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The ARIES-AT advanced tokamak, Advanced technology fusion power plant

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Abstract

The ARIES-AT study was initiated to assess the potential of high-performance tokamak plasmas together with advanced technology in a fusion power plant and to identifying physics and technology areas with the highest leverage for achieving attractive and competitive fusion power in order to guide fusion R&D. The 1000-MWe ARIES-AT design has a major radius of 5.2 m, a minor radius of 1.3 m, a toroidal β of 9.2% ($\beta_N = 5.4$) and an on-axis field of 5.6 T. The plasma current is 13 MA and the current-drive power is 35 MW. The ARIES-AT design uses the same physics basis as ARIES-RS, a reversed-shear plasma. A distinct difference between ARIES-RS and ARIES-AT plasmas is the higher plasma elongation of ARIES-AT ($\kappa_x = 2.2$) which is the result of a "thinner" blanket leading to a large increase in plasma β to 9.2% (compared to 5% for ARIES-RS) with only a slightly higher β_N. ARIES-AT blanket is a simple, low-pressure design consisting of SiC composite boxes with a SiC insert for flow distribution that does not carry any structural load. The breeding coolant (Pb–17Li) enters the fusion core from the bottom, and cools the first wall while traveling in the poloidal direction to the top of the blanket module. The coolant then

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returns through the blanket channel at a low speed and is superheated to ∼1100 °C. As most of the fusion power is deposited directly into the breeding coolant, this method leads to a high coolant outlet temperature while keeping the temperature of the SiC structure as well as interface between SiC structure and Pb–17Li to about 1000 °C. This blanket is well matched to an advanced Brayton power cycle, leading to an overall thermal efficiency of ∼59%. The very low afterheat in SiC composites results in exceptional safety and waste disposal characteristics. All of the fusion core components qualify for shallow land burial under U.S. regulations (furthermore, ∼90% of components qualify as Class-A waste, the lowest level). The ARIES-AT study shows that the combination of advanced tokamak modes and advanced technology leads to an attractive fusion power plant with excellent safety and environmental characteristics and with a cost of electricity (4.7 ¢/kWh), which is competitive with those projected for other sources of energy.

Keywords: Fusion power plant; Tokamak; Reversed shear plasma

1. Introduction

The ARIES program research aims at establishing the economic, safety, and environmental potential of fusion power plants, and at identifying physics and technology areas with the highest leverage for achieving attractive and competitive fusion power in order to guide fusion R&D. The ARIES Team is a U.S. national effort with participation from national laboratories, universities and the industry, and with strong international collaborations. The Team performs detailed physics and engineering analyses using the most current and detailed models available, and then uses the results to perform optimization and trade studies via a cost-based systems code.

Progress in tokamak physics during the past decade has been remarkable. Our vision of tokamak power plants in the 1980s, large devices operating in the pulsed mode, has been replaced by high-performance, advanced tokamak modes today. During this period, the ARIES Team has studied a variety of tokamak power plants. Continuation of research has allowed us to apply lessons learned from each ARIES design to the next and explore the impact of different extrapolations in physics and technology on the overall attractiveness of the tokamak concept. The ARIES-AT is the latest tokamak power plant study undertaken by the ARIES program and represents the latest in the evolution of our thinking on the potential of the tokamak concept as an attractive fusion power plant. In Section 2, we will explore the design directions for ARIES-AT as derived from the lessons that we have learned from previous designs. Sections 3–5 present overviews of our detailed analysis and trade-off studies. Section 6 examines the extent to which ARIES-AT has met the top-level requirements for fusion power plants and describes the key R&D topics. A complete discussion of ARIES-AT systems can be found in the accompanying papers in this special issue [1–8]. Figs. 1 and 2 show the cross-section of ARIES-AT.

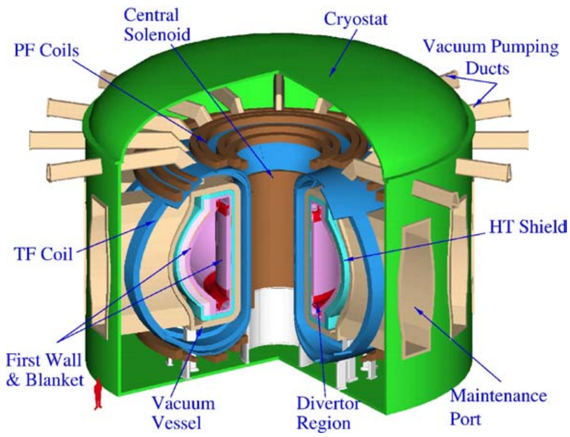

Fig. 1. The ARIES-AT fusion power core.

2. Design directions

The ARIES research aims at establishing the economic, safety, and environmental potential of fusion power plants. A set of top-level goals and requirements for fusion power has been developed in collaborations with representatives from US electric utilities and industries [9] and has been framed in a quantitative way [10,11] to help establish the minimum necessary features of a fusion power plant that would lead to

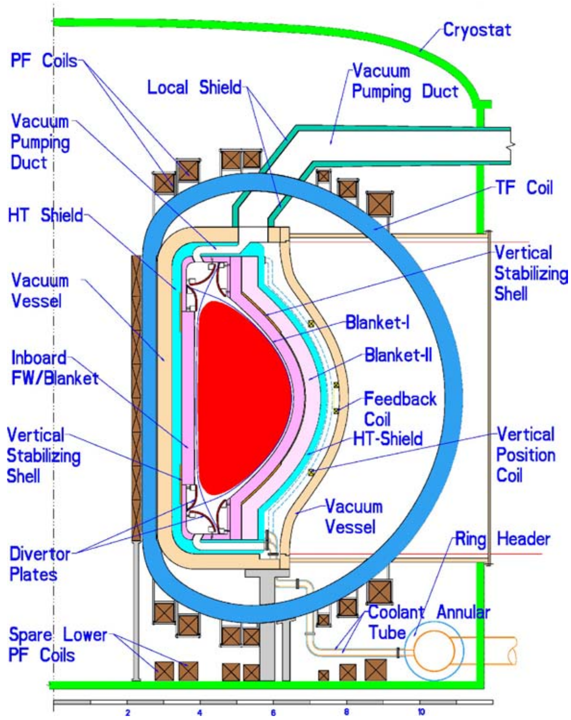

Fig. 2. Cross-section of ARIES-AT fusion power core.

its likely introduction into the US and world electricity market. These goals can be divided into three general categories: (1) power plant should have an economically competitive life-cycle cost of electricity, (2) it should gain public acceptance by having excellent safety and environmental characteristics, and (3) it should have operational reliability and high availability. Similar to all ARIES designs, ARIES-AT research was directed toward achieving these goals (as described below) while maintaining a balance between attractiveness (as measured by the customer requirements) and credibility (as measured by the extent of extrapolation from the present database).

2.1. Plasma physics

The plasma regime of operation is optimized mainly to reduce the plant cost (goal 1 above) through (a) reducing the recirculating power needed to maintain the plasma, (b) reducing the plant size by increasing plasma power density, and (c) reducing the cost of plasma support technologies such as magnets.

In the 1980s, tokamak fusion power plants were based on large devices operating in the pulsed-plasma mode. While the benefits of steady-state operation were clearly understood at the time, no credible steady-state tokamak plasma operation was identified. This is due to the fact that the non-inductive current drive, necessary for sustaining plasma current, is quite inefficient, leading to a very high recirculating power fraction (approaching unity). Operation with a high bootstrap current fraction as the approach to steady-state operation was proposed by the ARIES-I [12] and SSTR [13] studies simultaneously and independently in 1990. In order to reduce the current-drive power, the plasma current is reduced while the bootstrap fraction is maximized. In this manner, the amount of plasma current that should be driven by external means and the associated current-drive power are reduced and the power plant recirculating power fraction becomes acceptable (∼10–30%). In the first-stability regime (i.e., discharges with a monotonic q profile and stable to kink modes without a conducting wall) this can be accomplished by operating with a moderately high plasma aspect ratio ($A = 1/\varepsilon \sim 4.5$) to reduce the plasma current ($I \sim 10$ MA) at a relatively high poloidal beta ($\beta_p \sim 0.6$). This mode of operation, however, has a low value of plasma $\beta \sim 2\%$ because β_p (which determines the bootstrap fraction) is related to the achievable plasma β through

$$\varepsilon\beta_p \frac{\beta}{\varepsilon} = \left(\frac{\beta_N}{20}\right)^2 S \tag{1}$$

where $S = (1+\kappa^2)/2$ is the plasma shape factor, κ the plasma elongation, and $\beta_N = \beta$ a I/B is the normalized β (typically $\beta_N < 3.5$ in the first stability regime). In effect, there is a trade-off between the power density (lower β) and recirculating power fraction with the optimum system having a moderate power density and a moderate recirculating power fraction. Detailed MHD and current-drive analysis have shown that a bootstrap fraction of ∼60% can be achieved but with a slightly lower $\beta_N \sim 3$ (and a low plasma $\beta \sim 2\%$). Most of the driven current is located near the magnetic axis, requiring a current-drive power of about ∼100–200 MW delivered to the plasma. There is an ample experimental database for this regime, however operation in discharges with durations much longer than the current diffusion time as well as in a burning plasma are needed.

A comparison of major parameters of a 1000-MWe steady-state and pulsed-plasma tokamak power plants using the same physics and technology basis was generated as part of the Starlite study [10]. Our results indicate that both pulsed-plasma and steady-state first stability devices optimize in the same physics regime: low current, high A, and moderate bootstrap current fraction. Because of the long burn time in a pulsed-plasma power plant, the plasma is essentially in steady state. As such, the physics needs of pulsed-plasma and steady-state first-stability devices are identical (the only difference, the physics of non-inductive current drive, is well established). Generally, the low recirculating power fraction of pulsed-plasma operation is more than offset by the lower power density, the larger size, and the expensive PF system. As such, pulsed power plants tend to be more expensive than steady-state ones—a pulsed-plasma power plant is inferior to a steady-state one based on similar physics and technology extrapolations.

Starting from a steady-state, first-stability device, power plant economics can be improved by increasing the fusion power density and decreasing the recirculating power fraction. It should be noted that the economics improvement with increasing fusion power density or decreasing recirculating power fraction “saturates” after a certain limit. This is due to the fact that at low power density, any reduction in the plasma size (increase in power density and wall loading) proportionally decreases the volume of fusion core (blanket/shield/coils) that surrounds the plasma (and the system cost). However, when the plasma size is comparable and/or smaller than the blanket and shield thickness, the fusion core volume (and its cost) does not change appreciably with reduced plasma size and increased power density [14]. For 1000-MWe power plants, economic improvements with increased wall loading saturates at an average neutron wall load of $\sim$4 MW/m^2. As a reference, a first-stability steady-state device with a 16-T magnet technology achieves a neutron wall loading of $\sim$1.5 MW/m^2. Similarly, there is no economic benefit when the recirculating power fraction falls below a few percent.

High-field magnets can be used to increase the fusion power density of a first-stability device as is shown in Table 1. ARIES-I featured a 19-T cryogenic toroidal-field system. Because the fusion power density scales as $\beta^2 B^4$, the impact of increased toroidal-field strength is quite dramatic. The increase in fusion power density leads to a substantially smaller device and a slightly lower current-drive power, both factors combine to help improve power plant economics even though the magnet system is more expensive. Alternatively, one could seek plasma regimes with higher β_N and β that have a large bootstrap current fraction. In 1980s, second-stability operation with a high plasma β

Table 1
Major parameters of 1000-MWe advanced tokamak power plants

	First stability		Reverse-shear	
	ARIES-I	High-field: ARIES-I	ARIES-RS	ARIES-AT
Major radius (m)	8.0	6.75	5.5	5.2
Plasma aspect ratio	4.5	4.5	4.0	4.0
Plasma elongation, κ	1.8	1.8	1.9	2.1
β (%) (β_N)	2 (2.9)	2 (3.0)	5 (4.8)	9.2 (5.4)
Peak field at the coil (T)	16	19	16	11.5
On-axis field (T)	9.0	11	8	5.8
Average wall load (MW/m^2)	1.5	2.5	4	3.3
ITER89-P multiplier	1.7	1.9	2.3	2.0
Plasma current (MA)	12.6	10	11.3	13
Bootstrap current fraction	0.57	0.57	0.88	0.91
Current-driver power (MW)	237	202	80	35
Recirculating power fraction	0.29	0.28	0.17	0.14
Thermal efficiency	0.46	0.49	0.46	0.59
Cost of electricity (¢/kWh)	10	8.2	7.5	5

was the focus of the theoretical research. However, the ARIES-II/IV research [15] showed that high-β second-stability operation leads to bootstrap current overdrive and misalignment of the bootstrap current profile with the equilibrium current profile, resulting in a large current-drive power. Therefore, second-stability operation results in only marginal improvements in power plant attractiveness.

Reversed-shear [16] plasmas were proposed in mid 1990s. The benefits of this configuration are that it achieves both high β_N and β; obtains large bootstrap fractions with very good current profile alignment; and features an internal transport barrier necessary to sustain the peaked pressure profiles that is consistent with the high β and high bootstrap current. The negative central magnetic shear is responsible for stability to ballooning modes. A conducting wall, however, is necessary for stabilization of external kink modes—and the resistive-wall modes should be stabilized with plasma rotation and/or feedback coils.

Reversed-shear regime of operation was analyzed in the ARIES-RS study [11]. A large database of stable MHD equilibria was generated and their non-inductive current drive needs were calculated. This database was then used by the ARIES Systems Code to arrive at the optimum power plant configuration. Analyses showed that [11] to zeroth order, the cost of electricity is insensitive to the plasma aspect ratio in the range of 2.5–4.5 (the lower plasma β at the higher A is compensated by a higher toroidal-field strength on axis and a lower current-drive power). An aspect ratio of 4 was chosen for ARIES-RS based on engineering considerations.

Based on the results from our previous ARIES tokamak studies, the reversed-shear regime was chosen as the reference tokamak discharge for ARIES-AT. Since the ARIES-RS study, extensive theoretical and experimental studies of reversed-shear plasmas have been performed. As such, considerably more physics analysis was performed for ARIES-AT (see Section 3 and Refs. [1,2]) and, as a result, a more credible case exists for the ARIES-AT discharge. In addition, lessons learned from the ARIES-RS plasma optimization was applied to the ARIES-AT research. The major differences between ARIES-RS and ARIES-AT are:

(1) ARIES-RS has a bootstrap current fraction of ~88%. Unfortunately, a substantial amount of current-drive power was necessary to drive a relatively "small" plasma current mid-plasma and near the plasma edge. In ARIES-AT, effort was made to eliminate the need for non-inductive current drive in mid-plasma. As a result, while the bootstrap fraction was only increased slightly in ARIES-AT to 91%, the current-drive power was reduced by a factor of ~2.
(2) Typically the plasma profile optimization (e.g., MHD stability, bootstrap current, and current-drive analyses) is performed using fixed boundary equilibria. Plasma boundaries are forced to match the 95% flux surface boundary from the free-boundary equilibrium. For ARIES-AT, plasma boundaries used in the optimization are forced to coincide with the 99% flux surface from the free-boundary equilibrium. This has resulted in an increase in β_N as the plasma shaping is stronger as one approaches the separatrix. This approach also provides a strict self-consistency between the free-boundary equilibria generated by the poloidal-field coils and the fixed-boundary equilibria used for MHD stability and current-drive analyses.
(3) A distinct difference between ARIES-RS and ARIES-AT plasmas is the higher plasma elongation of ARIES-AT ($\kappa = 2.1$) which is the result of the ARIES-AT "thinner" blanket. In ARIES tokamak designs, passive vertical stabilization shells are located behind the blanket. As such, the blanket thickness helps set the plasma elongation. The higher plasma elongation of ARIES-AT and the corresponding larger plasma triangularity led to a large increase in plasma β to 9.2% (compared to 5% for ARIES-RS) with only a slightly higher $\beta_N = 5.4$ ($\beta_N = 4.8$ for ARIES-RS).
(4) More in-depth analysis has been performed for ARIES-AT (compared to ARIES-RS). For example, in the MHD area, we have considered the effects of edge density/temperature gradients and analyzed the stability requirements for the resistive wall modes (RWM), the neoclassical tearing modes (NTM), and the edge localized modes (ELM). Modern transport analysis tools, such as GLF23, were used to study consistency of plasma profiles. Radiative and conductive heat losses from the plasma and edge regions under a wide range of impurity doping were considered to arrive at a self-consistent solution for the divertor system.

The major plasma properties of ARIES-AT are also shown in Table 1. As in the ARES-RS study, we found that the cost of electricity is insensitive to the plasma aspect ratio in the range of 2.5–4.5. An aspect ratio of 4 was chosen for ARIES-AT based on engineering considerations (more space, more uniform load on components, etc.). An important observation from Table 1 is that ARIES-RS with a $\beta = 5\%$ and a 16-T Nb_3Sn superconductor achieved an average neutron wall loading of 4 MW/m^2. As discussed above, the power plant economic performance is insensitive to increased power density above this value. This can be seen in the ARIES-AT design point (Table 1) where the increased plasma β is used to reduce the toroidal field requirement instead of increasing the power density and reducing the system size.

2.2. Fusion technologies

Fusion power technologies (first wall, blanket, shield, and power conversion system) play an important role in the attractiveness of a power plant. Not only must these components perform the vital function of tritium breeding and power recovery, but the choice of materials and configurations associated with their designs also have a major impact on the safety and environmental characteristics of the power plant. In addition, the economics of the power plant is directly tied to the performance of fusion power technologies, i.e., thermal conversion efficiency. Three classes of low-activation structural materials are under consideration for fusion applications: ferritic/martensitic steels, vanadium alloys, and SiC composites. The ARIES program has studied a number of blanket designs based on these materials.

Use of SiC composites as advanced structural material for fusion power plants was first proposed in ARIES-I [12]. SiC composites offer a combination of high-performance at high temperature (necessary for high thermal efficiency) as well as a very low induced activation. In ARIES-I, solid breeder materials are located between SiC composite tube sheets in which flows the high-pressure He coolant. The ARIES-I design achieved a high coolant outlet temperature of 900 °C and excellent safety and environmental characteristics. The ARIES-I blanket, however, had two shortcomings. First, the large nuclear heat generation in the solid breeder, especially close to the first wall, resulted in small separations between the SiC composite tube sheets, making manufacturing of such a blanket difficult. Second, the coolant outlet temperature was too “low” for a reasonable thermal conversion efficiency from the Brayton cycle envisioned at the time. As such, a Rankine cycle was used in ARIES-I leading to a thermal conversion efficiency of ∼46%. The performance of the ARIES-I blanket was disappointing in this regard as the high coolant temperature did not lead to a high thermal conversion efficiency.

Self-cooled liquid breeder blankets are inherently simpler than blankets with solid breeders as the fusion neutron energy is deposited directly in the breeder/coolant. ARIES-RS featured a self-cooled design with liquid lithium as the coolant and breeder and vanadium alloy as the structural material. The high-temperature capability of vanadium alloys allowed a coolant outlet temperature of ∼700 °C and a thermal conversion of efficiency of 49% (better than ARIES-I due to a lower pumping power). Vanadium is a low-activation material and all components of ARIES-RS qualified for shallow land burial under NRC regulations. Lithium/vanadium blankets, however, have two main disadvantages: (1) the vanadium structure must be coated with an electrical insulator in order to have a reasonable MHD pressure drop and pumping power, and such an insulating coating has not yet been demonstrated; (2) care should be taken in design and operation in order to avoid safety concerns associated with lithium fires.

Ferritic/martensitic steels have the largest database of all fusion low-activation materials. It was perceived, however, that the relatively “low” maximum operating temperature (∼550 °C) of ferritic/martensitic steels would lead to a blanket with an unacceptably low thermal conversion efficiency. This perception was based on the assumption that the coolant outlet temperature from a blanket would always be lower than the maximum structure temperature. ARIES-ST [17] proposed an innovative dual-coolant ferritic steel blanket with an outlet coolant temperature of ∼700 °C (higher than the maximum steel structure temperature of 550 °C) and a thermal efficiency similar to those of Li/V blanket. The ARIES-ST first wall and blanket have a box-like geometry and are made of ferritic/martensitic steel. Pb–17Li is used as breeder and coolant, circulating slowly in the blanket: it enters the blanket at the bottom in the first row of blanket boxes, traverses to the top and return through the last three rows. Because the fusion neutron energy is directly deposited in the Pb–17Li breeder/coolant, it reaches an exit temperature of 700 °C. SiC inserts are used inside each box to help keep the heat conducted to the structural material from the coolant at a low level and maintain the ferritic/martensitic steel below its maximum operating temperature. The walls of the steel box, including the first wall, are cooled by high-pressure helium (flowing internally). Helium coolant exits the first wall/blanket at 500 °C and is then superheated by Pb–17Li coolant to ∼700 °C in a heat exchanger. Recent interest in gas-turbine power generators has resulted in large R&D efforts in high-efficiency gas cycles. In particular, the development of high-efficiency recuperators in a Brayton cycle provides the possibility of high efficiency even with a relatively low coolant temperature [18]. ARIES-ST utilizes such a Brayton cycle and achieves a thermal conversion efficiency of 46%.

Use of liquid breeder and the development of modern Brayton cycles directly address the shortcomings of the ARIES-I SiC blanket. As such, the ARIES-AT study revisited the use of SiC composite as blanket structural material (Section 4 and Refs. [3–5]) utilizing the same innovation as those in ARIES-ST for raising the coolant temperature. The ARIES-AT blanket is a simple, low-pressure design consisting of SiC composite boxes. Each box contains a SiC insert for flow distribution that does not carry any structural load. The coolant enters the fusion core from the bottom, and cools the first wall while traveling in the poloidal direction to the top of the blanket module. The coolant then returns through the blanket channel at very low speed and is superheated to ∼1100 °C. As most of the fusion power is deposited directly into the coolant, this method leads to a high coolant outlet temperature while keeping the SiC structure temperature as well as the interface between SiC structure and Pb–17Li to about 1000 °C. This blanket is well matched to an advanced Brayton power cycle, leading to an overall thermal efficiency of ∼59%.

High-temperature superconductors (HTS) offer several attractive features for fusion applications compared to cryogenic ones: (a) a higher magnetic field capability, (b) operation at liquid nitrogen temperature and simplifying cryo-plant, and (c) the possibility of using advanced manufacturing techniques to simplify the manufacture of superconductor coils and, thus, reduce the cost. As such, we considered HTS as the base line for ARIES-AT. Since, the ARIES-AT design point optimized at a maximum field of 11 T, we did not utilize the high-temperature capability of HTS. As such, Nb_3Sn superconductors can equally be used (in fact, with some cost penalty in increasing the size of the fusion core, NbTi superconductor will also be feasible). Section 4.4 and Ref. [6] summarize our findings on the impact of HTS on the attractiveness of fusion power plants.

3. Plasma physics

Plasma physics analyses of ARIES-AT were performed in an iterative manner with systems analysis and engineering design and with an increased level of detail and self-consistency. The ARIES-AT plasma optimization was started by developing a database of fixed-boundary equilibria for which the total parallel current and density and pressure profiles were prescribed. In each case, the bootstrap current profile was calculated and monitored for alignment with the prescribed current density profile. This approach allowed us to efficiently scan the MHD stability of these equilibria. Current-drive calculations were performed for selected optimized equilibria. The resultant plasma equilibria and current-drive database was then used by the systems code to arrive at the initial ARIES-AT design point. More detailed analysis was performed at this stage around this initial ARIES-AT design point. Self-consistent fixed-boundary equilibria were developed in which only the pressure profile and the parallel current density profiles from external current-drive sources (i.e., fast-wave and lower-hybrid current-drive systems) were prescribed. The bootstrap current-density profile was calculated at each stage of equilibrium and was added to the externally driven current to provide the total parallel current density. Through iteration with the current-drive calculations and system analysis, this approach was used to arrive at the final self-consistent ARIES-AT design point. In this analysis, plasma boundaries used in the fixed-boundary equilibrium calculations were taken from the free-boundary equilibrium at the 99% poloidal flux surface. This approach provides strict self-consistency between the free-boundary equilibria generated by the poloidal-field coils and the fixed-boundary equilibria used for MHD stability and current-drive analyses. In addition, this approach can have a strong impact on the calculated stable β values since the plasma shaping becomes stronger as one approaches the separatrix [2].

The robustness of the ARIES-AT equilibrium was studied by considering the impact of a variety of density/temperature gradients and current values at the plasma edge. Stability against the resistive wall modes (RWM), the neoclassical tearing modes (NTM), and the edge localized modes (ELM) were also analyzed. The transport properties were investigated using the GLF23 transport model to ensure self-consistency of assumed density/temperature profiles. This section summarizes the physics analysis in support of ARIES-AT. More details can be found in Refs. [1,2]. The ARIES-AT equilibria are shown in Fig. 3 and the major plasma parameters are summarized in Table 2.

3.1. MHD stability

Studies were performed to determine the impact of plasma elongation, triangularity, pressure and current profiles, and the kink-stabilization shell position on plasma performance. Fig. 4 shows the maximum nor-

Table 2
Major plasma parameters of ARIES-AT

Major radius (m)	5.2
Minor Radius (m)	1.3
Plasma current (MA)	12.8
On-axis field (T)	5.8
Plasma elongation, κ_x	2.20
Plasma triangularity, δ_x	0.90
Toroidal β[a] (%)	9.2
Normalized beta, β_N[a]	5.4
Poloidal beta, β_p	2.28
On-axis safety factor, q_0	3.50
Minimum safety factor, q_{min}	2.40
Plasma-edge safety factor, q_e	3.70
Plasma-edge safety factor, q^*	1.85
Internal inductance, $l_i(3)$	0.29
Location of kink-stabilization shell[b]	0.33
Bootstrap current fraction	0.91
Current-driver power (MW)	35
Electron and ion temperature (keV)	18
Temperature peaking factor ($T_0/\langle T\rangle$)	1.72
Electron density (m^{-3})	2.15
Total ion density (m^{-3})	1.71
Density peaking factor ($n_0/\langle n\rangle$)	1.34
ITER89-P multiplier	2.0

[a] ARIES-AT operates at 90% of maximum theoretical β limit.
[b] Distance from plasma edge normalized to minor radius.

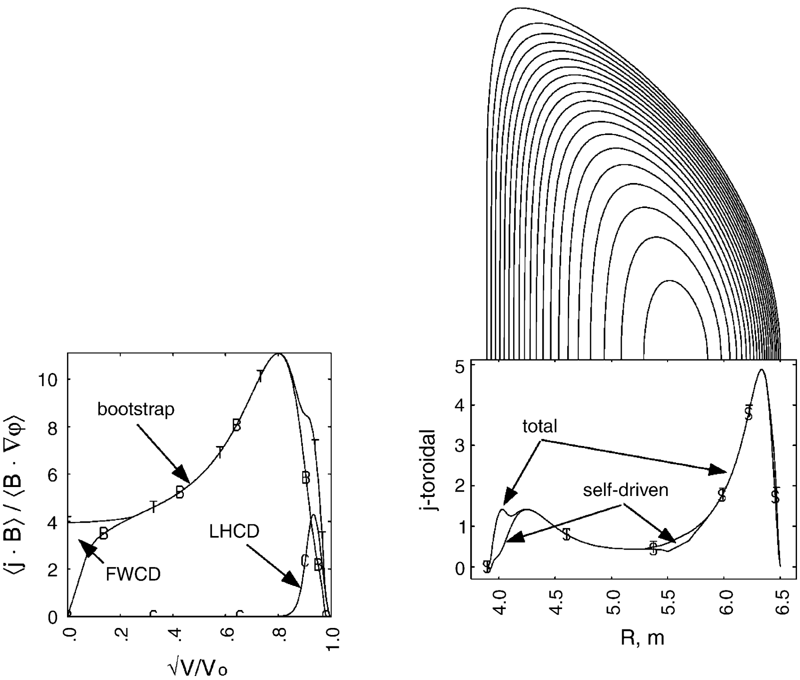

Fig. 3. The reference ARIES-AT plasma equilibrium: safety factor profile, parallel current density profiles, poloidal flux contours, and toroidal current density profiles.

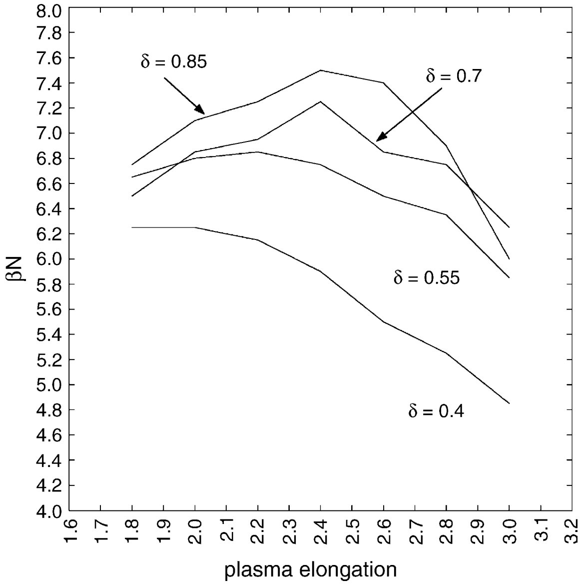

Fig. 4. Normalized beta values for ballooning modes as function of plasma elongation and triangularity. Note that there is an optimum triangularity for each value of plasma elongation.

malized beta (β_N) values for high-n ballooning modes as functions of plasma elongation and triangularity. In each case, the plasma pressure profile was optimized to provide a high stable β and a large well-aligned bootstrap current fraction simultaneously (larger stable β_N values are possible but at the expense of a much larger current-drive power). Fig. 4 shows that β_N increases with plasma elongation only if the triangularity is increased simultaneously—i.e., there is an optimum triangularity for each value of plasma elongation.

The low-n external kink stability limits of the above equilibria were analyzed and the marginal wall location was found for toroidal mode number up to 6 ($n = 6$–4 modes require a closer wall location). Our ballooning and kink stability analyses led to an elongation of $\kappa_x = 2.2$ and a triangularity of $\delta_x = 0.9$ for ARIES-AT under the requirement that the kink stabilization shell be located behind the blanket (see Fig. 2) at 0.33 times the minor radius. Note that this distance is measured between the shell position and the 99% flux surface (approximately the plasma separatrix).

Increasing the plasma elongation improves β, but makes the plasma more unstable to axisymmetric modes. The maximum elongation is limited by the passive-stabilization conducting structure and the feedback control system. It is desirable to utilize the kink-stabilization shell for passive stabilization of axisymmetric modes. The conducting structure is designed to provide a stability safety margin, $f_s = 1.2$ ($f_s = 1 + \tau_g/\tau_{L/R}$, where τ_g is the vertical instability growth time and $\tau_{L/R}$ is the longest up–down asymmetric time constant of the structure). The conducting structure is made of tungsten, 4-cm-thick, and operating at 1100 °C so direct cooling is not required. Feedback control simulations for the plasma vertical position were performed using the Tokamak Simulation Code [19]. A peak feedback power of 30 MVA is needed for 1 cm random vertical displacement (note that ∼85% of this power is reactive).

As the kink-stabilization shell is not an ideal wall, resistive wall modes (RWM) can grow although the growth time is reduced to values below the shell time constant. While physics of RWM is not fully understood, it is known that plasma rotation and/or feedback control coils can help stabilize these modes. Stabilization of RWM in the ARIES-AT plasma by plasma rotation was evaluated using the MARS stability code [20]. The critical rotation frequency for stabilization of $n = 1$ RWM modes is found to between 0.07 and 0.08 of the Alfven transit frequency and the stability window for the shell location is between 0.30 and 0.45 of the plasma radius. For $n = 2$ mode, the rotation required is substantially reduced. Our analysis indicates that neutral beam injection is not a practical mean for generating the necessary plasma rotation in ARIES-AT [1]. Another possible approach is utilization of “flux conserving intelligent coils” located at the resistive wall location (“intelligent shell feedback” scheme) [1]. Stabilization of RWM by control coils were examined by scaling the DIII-D C-coils [21]. Six saddle coils are employed (see Fig. 2) each carrying a maximum current of 50 kA-turns [1].

The stability of ARIES-AT with respect to neoclassical tearing modes has also been considered. In ARIES-AT, ELMs may seed perturbations to excite resonant $q = m/n = 5/2$ mode at $\rho = 0.92$ location. The large bootstrap current can produce a large island if excited and allowed to grow [1]. However, current profile control with radially localized current drive could suppress the island. It seems that a combination of current profile modification (to make the free energy available in the current profile more negative) and replacement of missing bootstrap (by the external current-drive system) is necessary and most effective. Design of the RF system for this purpose, however, has been left for the future.

The GLF23 drift-wave based model [22] was used to predict the transport properties of the ARIES-AT plasma and to analyze the consistency of the assumed plasma profiles with transport processes. The GLF23 model predict global energy confinement times which exceeds the ARIES-AT design requirement if the density profile is as peaked as assumed. The energy confinement improvement is primarily due to Shafranov shift stabilization of drift waves. Moreover, the computed pressure profile is close to the one found for optimum MHD stability. Results from the detailed GLF23 simulation of the ARIES-AT plasma are reported in Ref. [1].

3.2. Current drive

The ARIES-RS plasma required three current-drive systems: ICRF fast waves for central current drive; high-frequency fast wave (HFFW) for driving current in mid plasma; and lower hybrid for edge plasma current drive. The HFFW drive was necessary as the lower hybrid waves could not penetrate the inner parts of the plasma. As a result, a large current-drive power was necessary to drive a relatively "small" plasma current at mid-plasma. In ARIES-AT, effort was made to eliminate the need for non-inductive current drive in mid-plasma by optimizing plasma profiles. As a result, while the bootstrap fraction was only increased slightly in ARIES-AT to 0.91, the current-drive power was reduced by a factor of ~2. More importantly, a current-drive system with its associated launcher system was eliminated.

A large number of current-drive calculations were performed in support of plasma optimizations. The effect of impurities injected in the plasma to enhance plasma radiation (see Section 3.3) was taken into account. Calculations were performed using the ray-tracing code Curray [23]. Fig. 5 shows the reference plasma equilibrium current density profile with the various current-drive components. About 91% of plasma current is self-driven. Lower hybrid current drive is utilized to drive the bulk (1.1 MA) of the RF-driven current which is located in the outer part of the plasma (larger than 80% of minor radius). A small on-axis component (0.15 MA) is driven by ICRF fast waves.

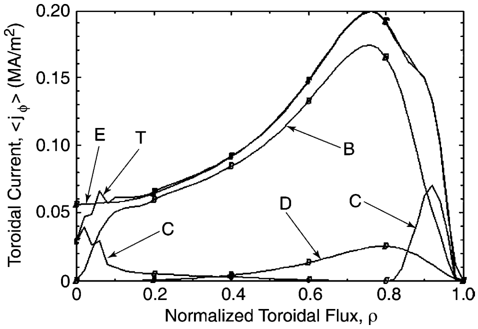

Fig. 5. Profiles of bootstrap (B), diamagnetic (D), and RF-driven (C) current densities. The sum of these components (T) matches well with the reference equilibrium current density profile (E).

The current-drive system details are given in Table 3. In order to minimize intrusion in the fusion power core, launchers with high power density capabilities were utilized: folded wave guides for fast wave and active/passive waveguide grill for lower hybrid. All RF launchers fits in one blanket module and cover less than 0.5% of the first wall area.

Using neutral beams (NBI) for driving current in the outer region of the plasma was considered as it was perceived that such a NBI system can also help generate the necessary plasma rotation. A 120-keV NBI system at a pivot angle of 70° can penetrate the outer region of the plasma and drive the necessary current. The needed NBI power is ~50 MW. Unfortunately, analysis indicated that this NBI system will not generate the necessary plasma rotation and this option was not pursued.

3.3. Divertor physics

Because of the high power density of ARIES-AT, the heat fluxes on the divertor structures can be high. Note

Table 3
The ARIES-AT current-drive system

System	Frequency	$N_{\|\|}$	Power (MW)	Driven current (MA)
ICRF fast wave	96 MHz	2	4.7	0.15
Lower hybrid	3.6 GHz	1.65	3.1	0.15
	3.6 GHz	2.0	4.4	0.2
	3.6 GHz	2.5	8.2	0.3
	3.6 GHz	3.5	8.9	0.2
	2.5 GHz	5.0	12.4	0.15

that in general, the heat flux capability of the first wall would be substantially smaller than that of the divertor because of the long cooling path of the inboard first wall. (In the ARIES-AT design, the heat flux capability of the first wall and divertor are, respectively, ~0.45 and ~5 MW/m^2.) As such, "radiative mantle" scenarios in which almost all of the plasma energy is radiated more or less uniformly on the first wall usually lead to an unacceptable solution. A combination of radiation in the core and edge/divertor plasmas is required.

To get an "upper" bound estimate of heat fluxes on the divertor, we considered a case with no impurity radiation (~30% of plasma energy is radiated through bremsstrahlung). Divertor plates were assumed to have an inclination of 10°. Assuming a ratio of 8:1 for the power flow into the outboard scare-off layer compared to the inboard and $\lambda_{q||}$ = 1.2 cm, the heat fluxes on the inner and outer divertor plates were estimated at 3.3 and 13.7 MW/m^2, respectively. Thus, some form of impurity injection is required to reduce the peak heat flux on the outboard divertor. (Note that $\lambda_{q||}$ is calculated based on formulation in Ref. [24] with ARIES-AT parameters which gives $\lambda_{q||}$ of roughly 1.2 cm for L-mode and 2.1 cm for H-mode.)

Because of the relatively low heat flux limit of the ARIES-AT first wall, the radiation fraction in the core should be below 36%. In this case, ~43% of power in the divertor should be radiated. The MIST [25] and STRAHL [26] codes are used to model impurity transport by injection of Ne, Ar, or Kr, and the resultant radiation in the ARIES-AT core and edge plasmas. Detailed analysis can be found in Ref. [1]. In all cases, the design requirements could be met with small increase in Z_{eff}. The results [1] indicate that higher Z rare gases are better as they lead to higher radiation at lower Z_{eff} and lower fuel dilution. The dynamics of feedback control of the radiated power using impurity injection, however, have not been investigated.

4. Fusion power core engineering

Silicon-carbide composite is selected as the structural material of the ARIES-AT first wall and blanket due to its high-temperature capability and its very low induced activation. This material helps the power plant to meet its economic goals (through a high thermal conversion efficiency) and its safety and environmental goals (because of the very low induced activation). Since SiC composites were first proposed in ARIES-I, substantial R&D have been performed and new design ideas have been developed. While the ARIES-I outlet coolant temperature was high (900 °C), the best possible thermal cycle available at the time (a Rankine cycle) led to a thermal conversion efficiency of 49%. Recent advances in low-cost recuperators make Brayton cycles very attractive even at a relatively low coolant temperature (700 °C) as was shown in the ARIES-ST study [17]. At the higher temperatures made possible with SiC-based blanket (~1100 °C), the Brayton cycle efficiency approaches 59% [18] making these types of blanket very attractive for fusion application. Separately, the ARIES-I design used solid breeders, which led to a complicated nested shell design. Use of a liquid breeder such as Pb–17Li simplifies the blanket considerably. Both directions were pursued in ARIES-AT.

Attempts were made to simplify manufacturing of the fusion core by utilizing a low-pressure design with simple geometry for blanket modules. In addition, the blanket and shield were designed in the form of integrated sectors to ensure rapid replacement of components (maximizing plant availability). Waste minimization and additional cost savings were also achieved by radially subdividing the blanket modules into two zones: a replaceable zone and a life of plant zone.

4.1. First wall, blanket, and shield

The ARIES-AT blanket concept is similar to that of the dual-coolant ARIES-ST. The first wall and blanket is a box-like structure. The breeding coolant enters the fusion core from the bottom and travels in the poloidal direction to the top of the blanket module. The coolant then returns through the blanket channel at very low speed and is superheated. In ARIES-ST, the ferritic/martensitic structure was separately cooled with He gas. The separate cooling circuit for the blanket box structure as well as the SiC insert allowed the breeding coolant in ARIES-ST to reach 700 °C, considerably higher than the maximum allowable temperature of ~550 °C assumed for the ferritic/martensitic steel structure.

While the same dual-coolant arrangement is possible for a blanket with SiC composite structures, a simpler design was chosen for ARIES-AT utilizing only Pb–17Li as breeding coolant because of: (1) the desire

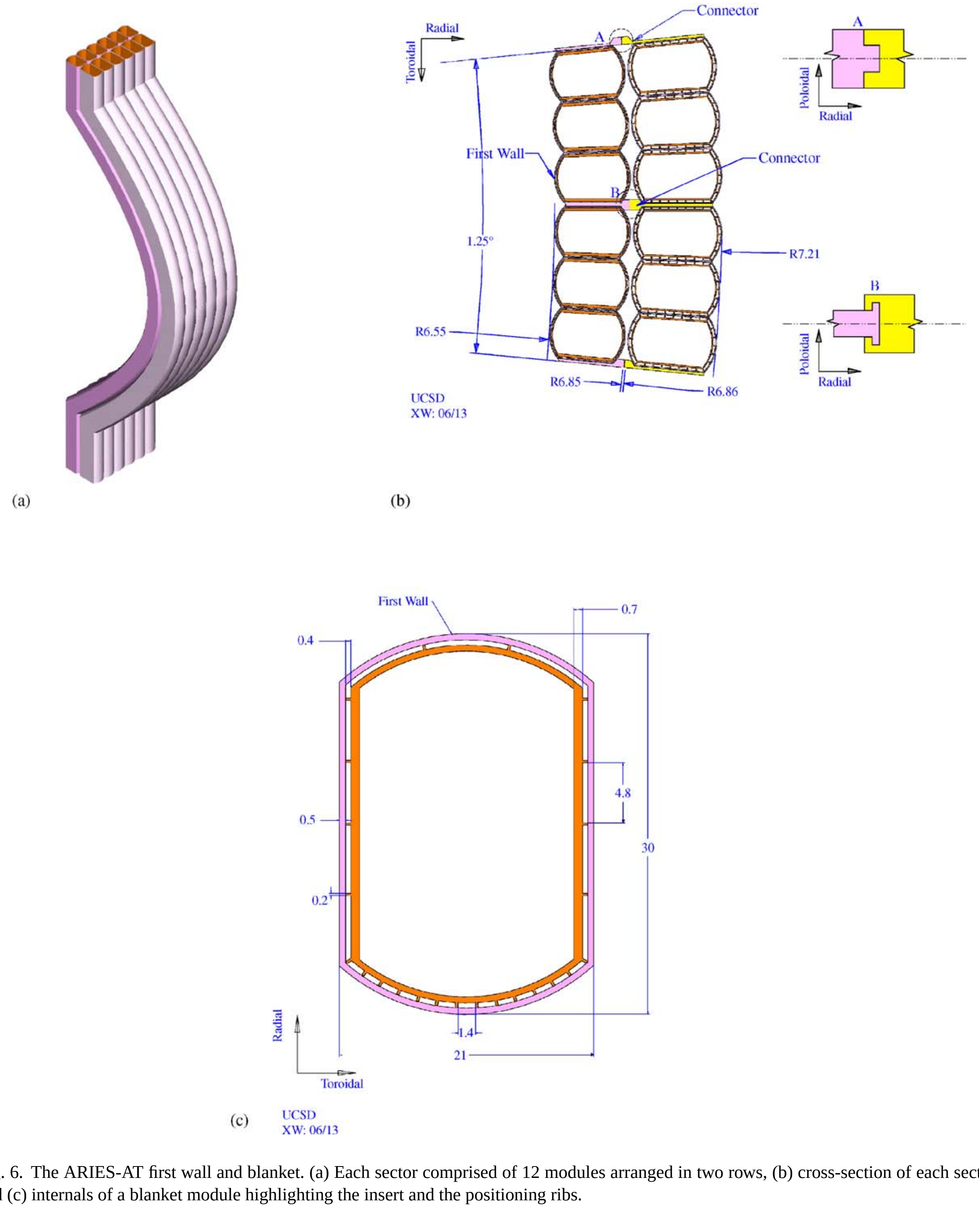

Fig. 6. The ARIES-AT first wall and blanket. (a) Each sector comprised of 12 modules arranged in two rows, (b) cross-section of each sector, and (c) internals of a blanket module highlighting the insert and the positioning ribs.

to have a simple low-pressure SiC composite structure with relative ease in manufacturing; (2) the desire to utilize only one coolant, and (3) the low nuclear heating of SiC composite as compared to a high nuclear heating in the Pb–17Li reducing the need for direct cooling of the box. However, a dual-coolant arrangement would allow for a much higher first wall heat load, if needed.

The ARIES-AT first wall and blanket is a simple, low-pressure design consisting of SiC composite boxes as shown in Fig. 6. Each module includes an insert with position ribs which is free flowing and does not carry structural loads. These ribs divide the annular region between the box and the insert into a number of channels through which the coolant first flows at high velocity to cool the walls of the box (from the bottom to top in the poloidal direction). The coolant turns at the top of the module and returns slowly through the large inner channel (insert). The breeding coolant exits the first wall and blanket box at the bottom at high temperature. This flow arrangement allows Pb–17Li coolant to exit at 1100 °C while maintaining the SiC composite structure and the SiC/Pb–17Li interface temperatures below 1000 °C. The wall of the box facing the plasma (first wall) consists of a 4-mm SiC structure on a which a 1-mm CVD SiC armor layer is deposited. Table 4 summarizes the power flow and major thermal parameters of the ARIES-AT blanket.

For stress analysis, a 1-MPa inlet pressure is assumed for the coolant to adequately account for both the pressure drop in the blanket (∼0.25 MPa) and the hydrostatic pressure of a ∼6 m Pb–17Li column (∼0.5 MPa). ANSYS analyses [3] indicate that the primary stress at the first wall is quite low (∼60 MPa) and the maximum thermal stress is 114 MPa resulting in a modest combined stress of 174 MPa, well within the 190-MPa limit adopted for SiC composites.

Table 4
Major parameters of ARIES-AT first wall and blanket

Power flows	
Fusion power (MW)	1719
Multiplied neutron power (MW)	1375
Total transport power (MW)	369
Core radiation (MW)	103
Average neutron wall load (MW/m^2)	3.2
Peak neutron wall load (MW/m^2)	4.8
Outboard blanket parameters (Region I)	
Number of sectors	16
Number of modules per sector	12
Power core coolant inlet temperature (°C)	654
Blanket outlet temperature (°C)	1100
Blanket pressure drop (MPa)	0.25
Maximum SiC temperature (°C)	996
Maximum SiC/Pb–17Li interface temperature (°C)	994
Average Pb–17Li velocity in the first wall (m/s)	4.2
Average Pb–17Li velocity in the inner channel (m/s)	0.11

The nuclear performance of the ARIES-AT fusion core was optimized in order to achieve a TBR of 1.1 (3D) while minimizing the radial build and waste stream. The blanket in the outboard is divided into two regions; a 30-cm-thick region which is replaceable and a 35-cm-thick region which is a life-of-plant component (see Fig. 2). On the inboard side, there is only one 30-cm-thick replaceable blanket region. The blanket is surrounded by hot shield zones (24-cm-thick in the inboard and 15-cm-thick in the outboard) which are also life-of-plant components. The vacuum vessel which is made of ferritic/martensitic steel and cooled by water also helps shield the magnets. This segmentation strategy has led to a reduction of ∼50% in rad-waste compared to ARIES-RS (see Section 6).

The ARIES-AT blanket with its high coolant outlet temperature is well matched to an advanced Brayton power cycle, leading to an overall thermal efficiency of ∼59%. The Brayton cycle considered includes three-stage compression with two intercoolers and a high efficiency recuperator. Recent advances in manufacturing of low-cost, high efficiency recuperators are the key in achieving the high cycle efficiency [18]. The power cycle requires using He as the secondary fluid. A heat exchanger between Pb–17Li and He coolants has not yet been developed but, in principle, is possible.

4.2. Divertor

The ARIES-AT divertor system is similar to that of ARIES-RS [27]. Each of the 16 ARIES-AT sectors includes a pair of top and bottom divertor modules. Each divertor module consists of three plates (inboard, outboard, and dome), a structural ring, and manifolds; SiC composite is used as structural material and Pb–17Li as coolant. Each divertor plate incorporates rectangular coolant channels with a toroidal pitch of 2.1 cm and a total radial dimension of 3.2 cm, including a 3.5-mm-thick W armor over a 0.5-mm-thick SiC composite wall on the plasma-facing side. The coolant flows in the poloidal direction in these coolant channels which serve as inlet and outlet headers for the 2-mm-thick channel on the plasma-facing side where the Pb–17Li flows toroidally over a short distance (∼2 cm) to cool the high heat flux region. This configuration is capable of handling a peak heat load of 5 MW/m^2 [3]. This divertor design was chosen because of the desire to integrate the divertor cooling circuit into the blanket circuit, to utilize only one coolant for both blanket and divertor, and to minimize the number of inlet and outlet coolant headers. Higher heat flux capabilities are possible by using high-pressure He as the coolant and utilizing W as the plate structural material (instead of SiC composite).

4.3. Configuration and maintenance

Achieving a high plant availability goal significantly influenced the overall configuration of ARIES-AT and the major design decisions. As for ARIES-RS, a sector removal scheme has been utilized for ARIES-AT. An important aspect of this approach is the integration of each sector (see Fig. 7). The first wall, blanket, parts of the shield, divertors, and stability shells form an integral unit. This integrated sector construction eliminates in-vessel maintenance operations and provides a very strong continuous structure which is able to support large loads (such as disruption forces). In addition, there is only one inlet-outlet header to each sector minimizing the number of connects/disconnects during maintenance. Maintenance is performed by extracting a complete sector and replacing it with a refurbished sector from the hot cell. Removal of the sector is performed through large ports (see Figs. 1 and 8) using a rail system together with transporter casks [7]. Port doors on the back of the shield and the cryostat prevent the spread of radioactive dust to the containment building.

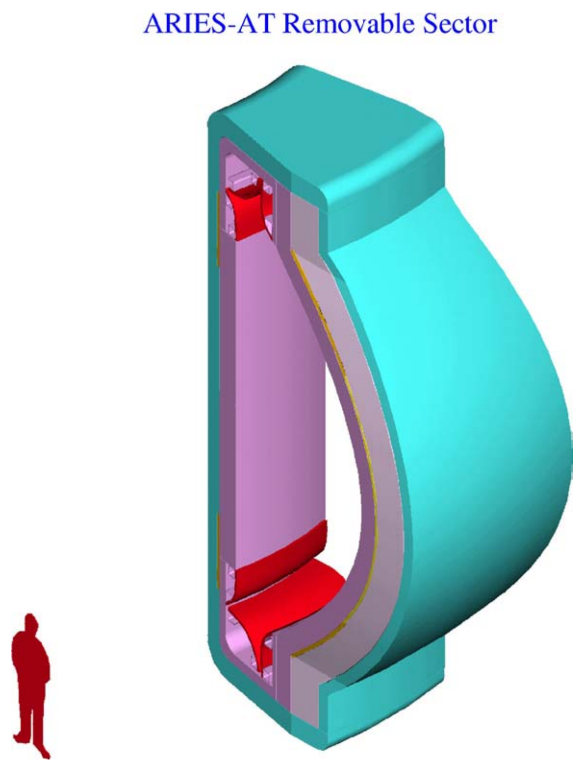

Fig. 7. An integrated ARIES-AT fusion core sector.

Extracting a complete sector requires positioning of the poloidal-field (PF) coils in regions above and below the maintenance doors. The outer leg of the toroidal-field (TF) coils must be sufficiently large to provide adequate clearance between the coils to extract a complete sector. This has the benefits of reducing the toroidal field ripple considerably. An important requirement for sector maintenance is that all replaceable components should have a nearly identical power core lifetime. Otherwise, the plant availability will suffer due to many plant shutdowns to replace smaller components.

4.4. Magnet systems

At temperatures greater than 20 K, the only HTS material that appears promising for fusion applications at present is YBCO tapes. Another HTS material, BSSCO, has great performance capability, particularly at high field but at cryogenic temperature. The YBCO thick film superconductor is highly anisotropic. For magnetic field parallel to the film, its current density capability is much larger than that of Nb_3Sn superconductors. Its current-carrying capability diminishes rapidly, however. as the field strength perpendicular to the tape is increased. For ARIES-AT, the perpendicular field strength has been kept below 5 T for this reason. The actual performance of the superconductor is a strong function of the film thickness; presently, thinner films have a higher current density. For the purpose of this study, it is assumed that it would be possible to make a relatively thick film (∼20–30 μm) of long length with a performance similar to that observed at present for thinner films.

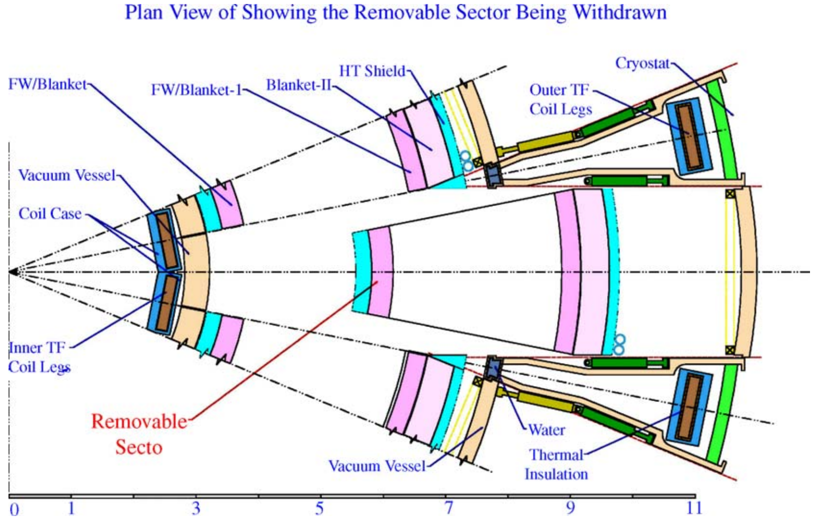

Fig. 8. Top view of ARIES-AT maintenance scheme. An integrated sector is removed horizontally.

A major benefit of HTS is the simplification of magnet design. High-temperature superconductors do not suffer from flux jumping when operated at temperatures above 10–20 K because of the large increase in the thermal capacity of the metals at these temperature compared to 4 K. As a result, only a small amount of stabilizer is needed which simplifies the magnet design and increases its average current density. In addition, it may be possible to manufacture the coil by directly depositing the superconductor on structural plates (with intermediate layers of insulator and stabilizer) by epitaxial techniques. This should lead to considerable savings in the magnet cost [6]. The HTS may be more damage-resistant compared to LTS (data to date indicate that irradiation damage limits are not lower than those for the LTS material; data at higher doses are not yet available). As a whole, HTS have the potential of substantially relaxing the design constraints due to lower irradiation damage, smaller stabilizer, and lower nuclear and AC heating at the cryogenic temperature. Magnet operation at 20 K will also substantially simplify the cryo-plants ("dry" magnets may be possible).

The ARIES-AT magnets are shown in Fig. 9. Because of the modest value of the ARIES-AT toroidal-field strength, the stresses are quite low [6]. A major constraint of YBCO superconductor is that the conductor strain should be less than 0.2%. As a result, the ARIES-AT magnet support structure is "thicker" and more massive than those of cryogenic superconductors. It should be noted that since the ARIES-AT design point optimized at a maximum field of 11 T, we did not utilize the high-temperature capability of HTS. As such, Nb_3Sn superconductors can equally be used (in fact, with some cost penalty in increasing the size of the fusion core, NbTi superconductor would also be feasible).

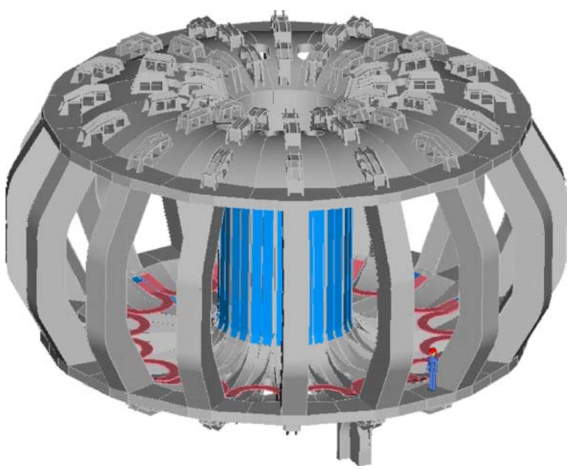

Fig. 9. The ARIES-AT toroidal-field coil assembly.

5. Safety and licensing

Safety and environment goals for ARIES designs include: (a) no need for evacuation plan and (b) no high-level rad-waste and minimization of low-level waste. In addition, we have tried to ensure that achieving these goals are transparent (e.g., by minimizing the source term) in order to simplify the licensing process.

Compared to previous ARIES designs, there is a greater depth in the safety analysis for ARIES-AT. For example, a large number of off-normal events were considered. In addition to loss-of-flow (LOFA) and loss-of-coolant (LOCA) events, we considered an ex-vessel loss of coolant, an in-vessel off-normal event that mobilize in-vessel inventories (e.g., tritium and tokamak dust) and bypass primary confinement such as a loss-of-vacuum and an in-vessel loss of coolant with bypass.

Fig. 10 shows the specific activity and specific decay heat in various outboard components of ARIES-AT as a function of time after shutdown. Note that the high initial decay heat of the first wall drops to levels below ferritic/martensitic steel components within 20 min to 1 h after shutdown. This is indicative of the very low-level of induced activity in SiC and the very small amount of total decay heat in SiC structure. As a result, analyses show [6] that following a LOCA or LOFA, the temperature of the first wall and blanket in ARIES-AT drops below the normal operation temperature and there is no temperature excursion. It is interesting to note that within 1 h after shutdown, the decay heat of SiC structure in the first wall and blanket falls below that of the Pb–17Li coolant. As such, component temperatures are estimated to be higher during a LOFA than a LOCA.

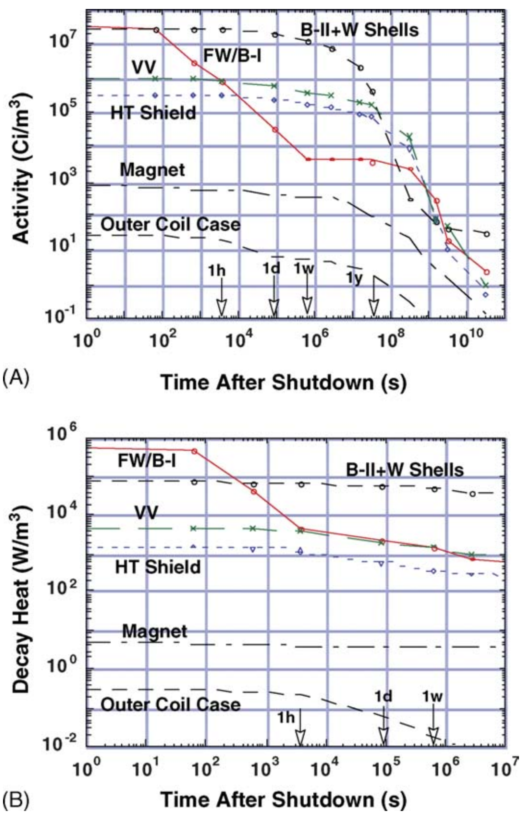

Fig. 10. (A and B) Specific activity and decay heat on the outboard components of ARIES-AT.

The major radiological inventories in ARIES-AT are tritium and activation products in fusion core components (plasma facing components, structural material, and coolant). Overall, no accident has been identified that can lead to mobilization of radioactive material in the ARIES-AT first wall and blanket. Therefore, inventories that may be mobilized during an accident are tritium, W dust, and ^{203}Hg and ^{210}Po isotopes in the Pb–17Li coolant. Detailed analyses [5] indicate that ARIES-AT can meet the no-evacuation limit even assuming a conservative on-line removal of Bi from the coolant. For example, the tritium inventory in ARIES-AT is estimated at 745 g, of which ∼680 g is estimated to be due to plasma interaction and/or co-deposited layer in the first wall. The worst case scenario considered (an in-vessel LOCA together with a by-pass event) would lead to the release of 7.6 g of tritium to the environment far below 150 g limit for no evacuation goal.

Activation analyses indicate that all ARIES-AT components meet the Class-C low-level waste requirements by a wide margin. In fact, ∼90% of ARIES-AT waste meets Class-A limits (lowest hazard classification under US regulations). Furthermore, the segmentation of ARIES-AT fusion core has led to a reduction of ∼50% in rad-waste compared to ARIES-RS. A total of ∼1270 m^3 of waste is generated after 40 full-power year (FPY) of operation (assuming that the coolant is reused in other power plants). Of this amount, ∼1000 m^3 is for lifetime components (end of service). The waste associated with the replaceable component is ∼30 m^3 every 4 FPY.

6. Design evaluation and R&D needs

Similar to all ARIES designs, the ARIES-AT research has aimed at arriving at a balance between attractiveness (as measured by customer needs) and credibility (as measured by the degree of extrapolation from present database). These two dimensions are explored in this section.

6.1. Design evaluation

The top-level goals and requirements for fusion power can be divided into three general categories: (1) cost, (2) safety and environmental features, and (3) reliability, maintainability, and availability.

Cost of electricity—The ARIES-AT cost of electricity of 4.7 ¢/kWh is competitive with current sources of energy and is smallest of modern fusion power plant studies. Table 5 provides a detailed breakdown of the ARIES-AT costs. Further reduction in the cost of electricity can be achieved by utilizing devices with higher net electric output (see Fig. 11).

Safety and environmental features—ARIES-AT meets the no-evacuation criteria with minimum need

Table 5
ARIES-AT power-plant economic parameters (1992 $)

Account no.	Account title	Million dollars
20.	Land and land rights	10.6
21.	Structures and site facilities	253.5
22.	Reactor plant equipment	761.0
22.1.1.	FW/blanket/reflector	64.3
22.1.2.	Shield	69.4
22.1.3.	Magnets	126.7
22.1.4.	Supplemental-heating/CD systems	37.1
22.1.5.	Primary structure and support	26.9
22.1.6.	Reactor vacuum systems (unless integral elsewhere)	98.8
22.1.7.	Power supply, switching and energy storage	50.7
22.1.8.	Impurity control	4.1
22.1.9.	Direct energy conversion system	0.0
22.1.10.	ECRH breakdown system	4.0
22.1.	Reactor equipment	482.0
22.2.	Main heat transfer and transport systems	126.0
23.	Turbine plant equipment	243.0
24.	Electric plant equipment	98.5
25.	Miscellaneous plant equipment	47.4
26.	Heat rejection system	23.3
27.	Special materials	83.8
90.	Direct cost (not including contingency)	1521.1
91.	Construction services and equipment	171.9
92.	Home office engineering and services	79.1
93.	Field office engineering and services	79.1
94.	Owner's cost	277.8
96.	Project contingency	311.8
97.	Interest during construction (IDC)	403.2
99.	Total cost	2844.0
Unit costs	Unit direct cost ($/kWe)	1521
	Unit overnight cost ($/kWe)	2441
	Unit total cost ($/kWe)	2844
Cost of electricity	Capital return (mill/kWh)	36.87
	O&M (2.10%) (mill/kWh)	6.87
	Component replacement (mill/kWh)	3.51
	Decommissioning (mill/kWh)	0.25
	Fuel (D) cost (mill/kWh)	0.03
	Total cost of electricity, COE (mill/kWh)	47.53

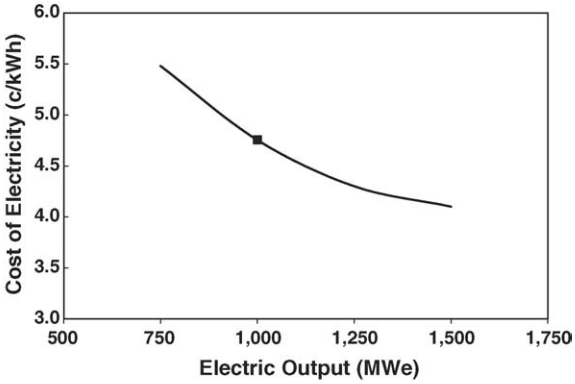

Fig. 11. Cost of electricity of ARIES-AT type power plant as a function of net electric output showing the nuclear economy of scale. The average neutron wall load is kept fixed at 3.3 MW/m^2 (ARIES-AT value). The ARIES-AT design point is shown by a square.

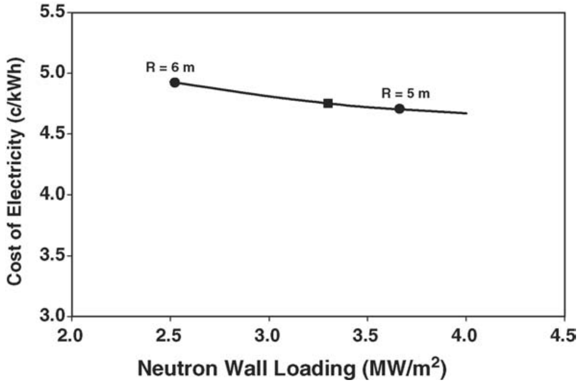

Fig. 12. Cost of electricity of 1000-MWe ARIES-AT type power plant as a function of neutron wall loading demonstrating insensitivity of COE to power plant size. The ARIES-AT design point is shown by a square.

for the confinement of radioactive material because of the low radioactive source term. In addition, ARIES-AT components meet the Class-C low-level waste requirements by a wide margin (∼90% of ARIES-AT waste meet Class-A limits). Furthermore, the segmentation of ARIES-AT fusion core has led to a reduction of ∼50% in rad-waste compared to ARIES-RS.

Reliability, maintainability, and availability—Detailed modeling of sector removal maintenance scenario has shown that schedule maintenance can be performed in 1 month. An availability of >90% was predicted for ARIES-AT [7,8]. (A plant factor of 85%, however, was used in ARIES-AT costing.) Special attention was given to manufacturability in designing ARIES-AT components.

Compared to ARIES-RS, these improvements in meeting customer requirements have been achieved mainly through using SiC composites as structural material and achieving a high thermal conversion efficiency. In addition, SiC composite has allowed the blanket to be thinner, moving the passive stabilization closer to the plasma resulting in a larger plasma elongation and plasma β.

As discussed in Section 2, the decrease in the cost of electricity with decreasing power plant size saturates above a certain value. This can be clearly seen in Fig. 12; the COE increases only by ∼0.15 ¢/kWh (∼3%) if the major radius is increased from 5.2 to 6 m (about 15%). As a result, the increased plasma β in ARIES-AT has not been utilized to decrease the machine size; rather it was used to reduce the toroidal-field strength. In addition, because of this insensitivity to size, ARIES-AT is more robust to variations in achievable physics or engineering parameters (e.g., a lower value of β and/or a higher divertor heat load can be easily accommodated by utilizing a larger size device with minor cost penalties).

6.2. R&D needs

A major aim of the ARIES-AT study was to identify physics and technology areas with the highest leverage for achieving attractive and competitive fusion power in order to guide fusion R&D. Refs. [1–8] describe the detailed analysis of ARIES-AT components and include a comprehensive list of R&D topics. Some of the more critical issues are listed below (the ordering is not intended to imply any priority). It should be noted that the ARIES-AT physics assumptions are similar to those of ARIES-RS.

6.2.1. Demonstration of a stable and controllable operating point

The benefits of the reversed-shear plasma operating mode are that it achieves both high β_N and high β, it obtains large bootstrap current fractions with very good current profile alignment, and it appears to provide the transport suppression necessary to sustain the pressure profile that is consistent with the higher β and high bootstrap current. Significant progress in understanding this regime of operation has been made in recent years and higher performance discharges has been obtained. Still, demonstration and control of these discharges in steady-state conditions and with a burning plasma remain a central issue. It should be demonstrated that plasma parameters (e.g., density, temperature, position, current, ...) and their profiles can be controlled precisely as large variation in fusion power would cause significant problems for the designs of fusion core components.

6.2.2. Divertor and edge physics

Divertors remain one of the most difficult physics and engineering challenges. Radiative modes which result in dispersing most of the plasma transport power on the first wall and divertor plates can help keep the heat load on the divertor plates to a manageable level. It appears that a combination of radiation in the core and in the divertor would be necessary. The higher density of an L-mode type edge can help to reduce the heat loads.

6.2.3. Disruption avoidance

One of the most serious problems with the tokamak is the possibility of a major plasma disruption requiring plant shutdown. Besides the potential for damage to the surrounding structures and the possibility to initiate events leading to chemical and radioactivity release, unplanned shutdowns are an unacceptable operating characteristic for a power plant. Aside from the impact on plant availability, it is likely that extended outages would follow an unplanned plasma shutdown to identify the source of the problem and ensure that it could not happen again.

As the SiC composite is a semiconductor, the disruption forces appear on the sturdy shield and vacuum vessel of ARIES-AT. However, thermal load on in-vessel components is still large and only a small number of disruptions (∼20) can be tolerated. Further studies are needed to determine operating points with very low probability of disruption (less than 1 disruption per year) and/or reliable active measures of disruption avoidance.

6.2.4. Development and qualification of SiC composites

Utilization of SiC composite is the key feature that has allowed ARIES-AT to achieve its relatively low cost of electricity and high level of safety and environmental attractiveness. A large amount of development is still needed before silicon-carbide composites can be qualified as structure material for fusion applications. It should be noted that major advances has been made in SiC composites during the last 10 years [28]. The latest results indicate that irradiated "third generation" composites show no degradation of properties up to 10 dpa. This improved performance is due to development of stoichiometric crystalline SiC fiber and advanced fiber/matrix interphases. Note that the irradiation lifetime of SiC composite is set probably by burn-up rate (a 3% burn-up rate is assumed for ARIES-AT, corresponding to ∼15 MW/m^2). For comparison, the irradiation lifetime of ferritic steel is estimated at 150–200 dpa.

6.2.5. Compatibility of Pb–17Li with SiC composites

The maximum interface temperature between Pb–17Li and SiC composites have been set at 1000 °C in ARIES-AT. Tests at ISPRA have shown that the two materials are compatible at 800 °C under stagnant conditions [29]. Test in flowing Pb–17Li is needed to better establish operating temperature limits.

6.2.6. High-temperature heat exchangers (Pb–17Li and He)

A high temperature heat exchanger between Pb–17Li coolant and He should be developed to utilize the Brayton cycle and achieve a high thermal cycle efficiency. This was considered a reasonable development step, although the technology needs to be demonstrated. Possible materials for such a heat exchanger are Nb or SiC composite (if Pb–17Li and SiC are shown to be compatible at 1100 °C).

6.2.7. High heat flux removal combined with high performance

The simultaneous requirements of high performance under extreme heat and particle loads, together with extended lifetime, maintainability, and safety pose a difficult problem for divertors and high heat flux components. The lack of predictive capabilities for the edge plasma exacerbates the problem. It appears that tungsten is the only acceptable material for divertor armor (and possibly divertor structure). Development of divertor systems with capability of removing high heat flux at steady state as high-grade heat is a major feasibility issue.

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