ARIES-AT: AN 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. Several avenues were pursued in order to arrive at plasmas with a higher β and better bootstrap alignment compared to ARIES-RS that led to plasmas with higher β_N and β. Advanced technologies that are examined in detail include: (1) Possible improvements to the overall system by using high-temperature superconductors, (2) Innovative SiC blankets that lead to a high thermal cycle efficiency of ~60%; and (3) Advanced manufacturing techniques which aim at producing near-finished products directly from raw material, resulting in low-cost, and reliable components. The 1000-MWe ARIES-AT design has a major radius of 5.4 m, minor radius of 1.3 M, a toroidal β of 9.2% (β_N = 6.0) and an on-axis field of 5.6 T. The plasma current is 13 MA and the current drive power is 24 MW. The ARIES-AT study shows that the combination of advanced tokamak modes and advanced technology leads to attractive fusion power plant with excellent safety and environmental characteristics and with a cost of electricity (5c/kWh), which is competitive with those projected for other sources of energy.

1. Introduction

The ARIES-RS study [1] showed that advanced tokamak modes, especially reversed-shear, are very attractive for fusion power. This mode of tokamak plasma operation has been under intense theoretical and experimental study in recent years. Since ARIES-RS study was completed, new insights on advanced tokamak modes have emerged. Theoretical studies indicate that even higher performance plasmas are possible. The ARIES-AT study was initiated to assess the potential of high-performance tokamak plasmas together with advanced technology (e.g., high-temperature superconductors, high-performance blankets, etc.) in the context of a fusion power plant.

The ARIES-AT study is a national U.S. effort. This paper summarizes the ARIES-AT design; further information can be found in the ARIES-AT project report [2]. Figure 1 shows the cross section of ARIES-AT. The major parameters of the 1000-MWe ARIES-AT power plant are given in Table 1. The ARIES-AT study shows that the combination of advanced tokamak modes and advanced technology leads to attractive fusion power plant with excellent safety and

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environmental characteristics and with a cost of electricity which is competitive with those projected for other sources of energy.

Table 1: Major Parameters of ARIES-AT

Plasma aspect ratio	4.0
Major toroidal radius (m)	5.2
Plasma minor radius (m)	1.3
Plasma elongation, κ_x	2.2
Plasma triangularity, δ_x	0.84
Toroidal β_N	5.4*
Toroidal β	9.2 %
Electron density (10^{20} /m^3)	2.3
Greenwald Density (10^{20} /m^3)	2.4
ITER-98H scaling multiplier	1.4
Plasma current (MA)	13
CD power to plasma (MW)	36
On-axis toroidal field (T)	6.0
Peak field at TF coil (T)	11.4
Thermal cycle efficiency	59%
Average neutron load (MW/m^2)	3.3
Fusion power (MW)	1,755
Recirculating power fraction	14%
Net plant efficiency	51%
Cost of electricity (c/kWh)	5

* Operates at 90% of theoretical limit.

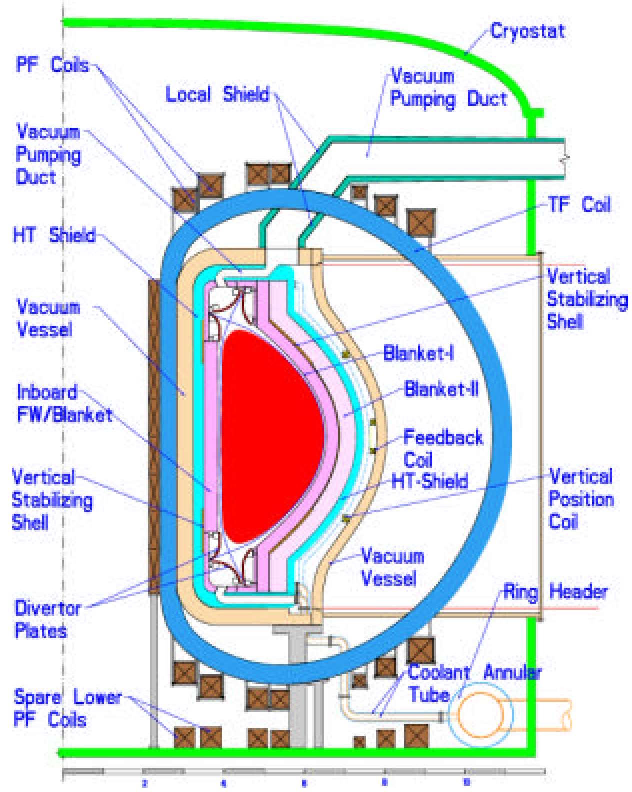

Fig. 1. Cross-section of ARIES-AT

2. Physics

Several avenues were pursued in order to arrive at plasmas with a higher β and better bootstrap alignment compared to ARIES-RS. Detailed stability analysis indicated that the presence of the X-point improve ballooning stability and lead to higher β_N and β (using $\geq$ 99% flux surface from free-boundary equilibria instead of 95% flux surfaces). In addition, the thinner blanket of the ARIES-AT allows the stabilization shells for both axisymmetric and resistive wall modes to be located closer to the plasma. This allows for a larger plasma elongation as well help increase kink stability limits. A large number of plasma equilibria were generated and their stability was studied. As operation with a radiative divertor is envisioned, our analysis then focused on those equilibria with a finite edge density and a Z_{eff} approaching 2, both of which reduced either the plasma β and/or led to a larger current-drive power. In addition to ballooning, kink stability of n = 1-6 modes were analyzed (n = 6 is found to be most unstable).

Three classes of equilibria were identified for further evaluation: A) β_N = 5.6, bootstrap fraction of 0.96, B) β_N = 6.0, bootstrap fraction of 0.94, C) β_N = 6.8, bootstrap fraction of 0.91.

All these equilibria have an elongation of κ_x = 2.2, triangularity of δ_x ~ 0.8, and plasma aspect ratio of 4. While all three have a high bootstrap current fraction of > 0.9, the bootstrap current alignment is very different in the three cases, with highest β case having the worst alignment. Analysis of the trade-off among the high plasma β with the current-drive power indicated that case B is the optimum equilibria and was adopted as the reference equilibria. (The design point operates at 90% of theoretical limit).

Analysis with the MARS code indicated that a kink stability shell consisting of 1 cm of tungsten behind the first blanket zone (τ = 10 ms), keeps the power requirements for n = 1 resistive wall mode feedback coils at a modest level (a few MW). Sixteen saddle coils are utilized for feedback and are located behind the shield and mounted on the vacuum vessel (see Fig. 1). The passive vertical stabilization shell is also located behind the first blanket zone and is made of 3 cm of tungsten. Both shells are hot and are passively cooled through conduction and radiation to the blanket zones. The vertical position feedback coils are also located behind the shield ans are attached the vacuum vessel (see Fig. 1). The feedback power is estimated by TSC to about 30 MVA (the real power is < 10% of the reactive power).

Current-drive analysis showed two RF systems are sufficient for steady-state operation: fast-wave (68 MHz, 3.8 MW) for on-axis current drive and lower hybrid (3.6 GHz, 31 MW) for driving edge current. In order to minimize intrusion in the 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 modules and cover less than 0.5% of the first wall area.

Analysis of neoclassical tearing modes for ARIES-AT profiles indicated that one mode is unstable. However, this mode is located near the plasma edge where extensive lower hybrid power is utilized for edge current drive and is expected to stabilize this mode. Transport analysis with GLF23 code has also been performed. The resulting density and temperature profiles are close to profiles assumed for ARIES-AT.

3. Fusion Core Engineering

Possible improvements to the overall system may be possible using high-temperature superconductors (HTSC) because of capability of higher field strengths, particularly if HTSC materials are operated at low temperatures. This capability of HTSC was not utilized in ARIES-AT (design optimized at 11.5 T because of the of high β). The HTSC offer operational and manufacturing advantages--higher operation temperature allows for a simpler cryogenic plant and the manufacturing process (deposition of thin superconducting film on a substrate) can simplify the manufacturing of large coils dramatically. Both cryogenic and high-temperature superconductors were considered for ARIES-AT. It is found that potential operational advantages of HTSC translate into considerable cost-savings.

Since SiC composites were first proposed in ARIES-I, substantial R&D has been performed and new design ideas developed. While, the ARIES-I outlet coolant temperature was high (900^{o}C), the best possible thermal cycle available at the time, Rankin cycle, let to a thermal conversion efficiency of 49%. Recent advances in low-cost recuperators make Brayton cycles very attractive even at relatively low coolant temperature (700^{o}C) as was shown in the ARIES-ST study. At higher temperatures possible with SiC-based blanket (~1,100 ^{o}C), the Brayton cycle efficiency approaches 59% [3] making these types of blanket very attractive for fusion application.

Separately, the ARIES-I design used solid tritium breeders, which led to a complicated nested shell design. Use of a liquid breeder such as LiPb simplifies the blanket considerably. Both directions were pursued in ARIES-AT. The ARIES-AT blanket is made of SiC composite cooled with PbLi (See Fig.2). The 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 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 ~1,100^{o}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 interface between SiC structure and LiPb to about 1000^{o}C. This blanket is well matched to an advanced Brayton power cycle, leading to an overall thermal efficiency of ~60%.

The very low afterheat in SiC composites results in exceptional safety and waste disposal characteristics that also help reduce the power plant cost. All of the fusion core components are low-level waste and qualify for shallow land burial under U.S. regulations (~90% of components qualify as class A waste).

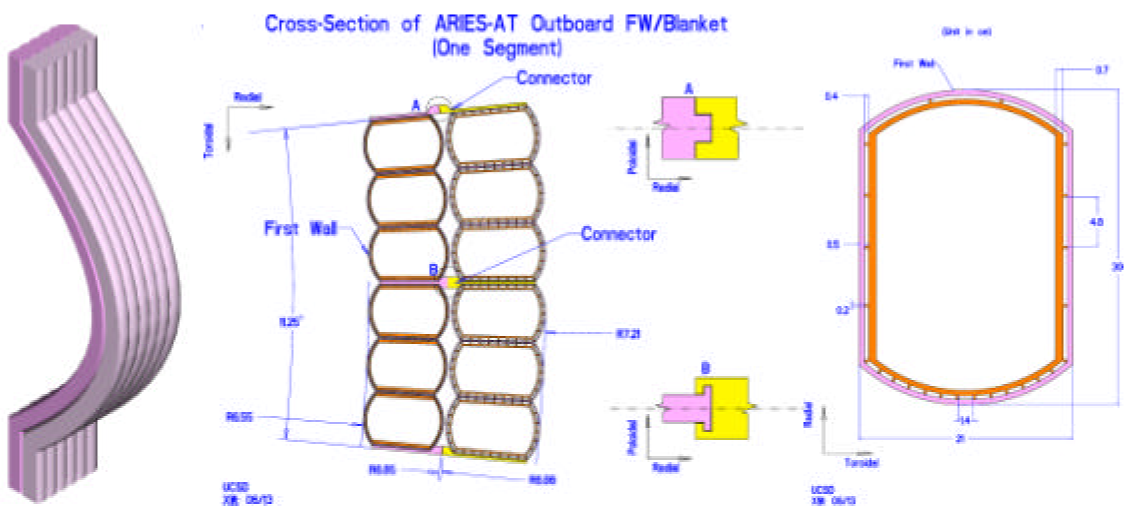

Fig 2. The ARIES-AT blanket module (1/2 of a blanket sector or 1/32 of the torus): A) Plan view, B) Cross section of the module, C) Cross section of a blanket "tube" showing the flow distribution insert.

References

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