THE ARIES-ST STUDY: ASSESSMENT OF THE SPHERICAL TOKAMAK CONCEPT AS FUSION POWER PLANTS*

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ABSTRACT

Recent experimental achievements and theoretical studies have generated substantial interest in the spherical tokamak concept. The ARIES-ST study was undertaken as a national U.S. effort to investigate the potential of the spherical tokamak concept as a fusion power plant and as a vehicle for fusion development. The 1000-MWe ARIES-ST power plant has an aspect ratio of 1.6, a major radius of 3.2 m, a plasma elongation (at 95% flux surface) of 3.4 and triangularity of 0.64. This configuration attains a β of 54% (which is 90% of the maximum theoretical β). While the plasma current is 31 MA, the almost perfect alignment of bootstrap and equilibrium current density profiles results in a current-drive power of only 31 MW. The on-axis toroidal field is 2.1 T and the peak field at the TF coil is 7.6 T, which leads to 288 MW of Joule losses in the normal-conducting TF system. The ARIES-ST study has highlighted many areas where tradeoffs among physics and engineering systems are critical in determining the optimum regime of operation for spherical tokamaks. Many critical issues also have been identified which must be resolved in R&D programs.

1. INTRODUCTION

Theoretical and experimental studies indicate that the MHD performance of a tokamak plasma is substantially improved with decreasing aspect ratio. While steady-state tokamaks with superconducting toroidal-field (TF) coils optimize at moderately high aspect ratio (A ~ 4), at low aspect ratio (*e.g.,* spherical tokamaks), the plasma β becomes large enough and the required toroidal-field becomes small enough that resistive TF coils with manageable Joule losses can be used. This eliminates the need for a thick, inboard shield for cryogenic TF coils so that fusion devices with smaller major radius may be possible.

The ARIES-ST study is a national U.S. effort to investigate the potential of the spherical tokamak concept. This paper summarizes the ARIES-ST design; further information can be found in the ARIES-ST project report. The cross section of ARIES-ST is shown in Fig. 1 and the major parameters of the 1000-MWe ARIES-ST power plant are given in Table 1.

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* Institutions involved in the ARIES-ST study, in addition to UC San Diego, are (1) Argonne National Laboratory, (2) Boeing, (3) General Atomics, (4) Princeton Plasma Physics Laboratory, (5) Rensselaer Polytechnic Institute, and (6) University of Wisconsin-Madison.

Table 1: Major Parameters of ARIES-ST

Plasma aspect ratio	1.60
Major toroidal radius (m)	3.20
Plasma minor radius (m)	2.00
Plasma elongation	3.40
Plasma triangularity	0.64
Toroidal β	54 %
Electron density (10^{20} /m^3)	2.74
ITER-93H scaling multiplier	1.83
Plasma current (MA)	30.8
CD power to plasma (MW)	31
On-axis toroidal field (T)	2.14
Peak field at TF coil (T)	7.6
Center-post ohmic losses (MW)	231
TF-coil ohmic losses (MW)	288
Average neutron load (MW/m^2)	4.1
Fusion power (MW)	2,860
Recirculating power fraction	0.32

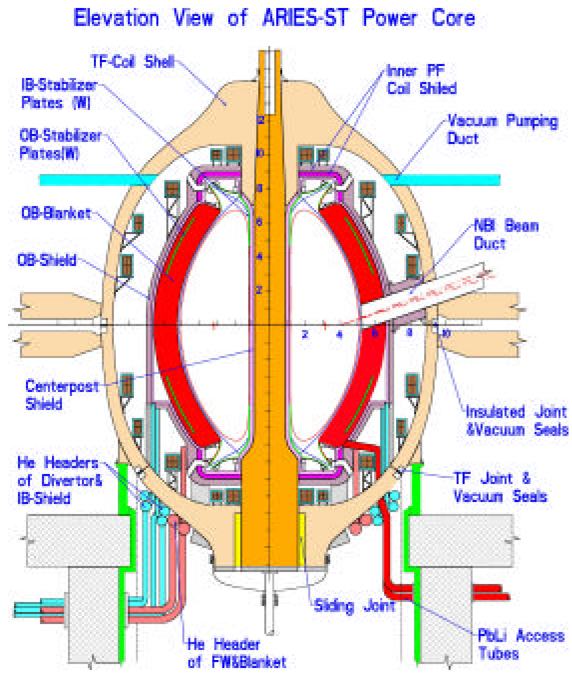

Fig. 1. Cross-section of ARIES-ST

2. PHYSICS

In order achieve a reasonable recirculating power fraction in a spherical tokamak power plant, it is essential to have a high toroidal β (to lower Joule losses in the TF coils) and almost perfect alignment of bootstrap and equilibrium current density profile (to minimize the current-drive power). An extensive study of the MHD stability limits of ST plasmas over a wide range of aspect ratio (A = 1.25, 1.4, 1.6, and 1.8), elongation, and triangularity were performed in order to map out the optimum operating regime for ARIES-ST. High β equilibria have been obtained which are stable to ballooning and kink modes (with a conducting wall located 0.15 a away from the plasma boundary). The β limits for these equilibria are: β= 74% and β_N = 8.8 (for A = 1.4 with κ =3.6 and δ = 0.64); β = 60% and β_N = 8.3 (for A = 1.6 with κ = 3.4 and δ = 0.64); and β = 46% and β_N = 7.8 (for A = 1.8 with κ =3.2 and δ = 0.64).

The above class of equilibria presents special challenges for several reasons. It is especially difficult to control the innermost boundary points of the equilibria because there are no PF coils in the inboard side. In addition, the strong shaping required for ideal MHD stability has to be obtained in the presence of strong natural shaping from the plasma interacting with its own field. The desire to obtain equilibria with high poloidal beta but with low internal inductance implies high order cancellations in the plasma equilibrium equations making the problem computationally difficult, such that special parameterization of profiles is needed. While the above equilibria have been shown to be stable to axisymmetric modes, studies of plasma position and shape control are on going to ensure practicality of these equilibria.

The ARIES-ST plasma equilibrium yields a pressure-driven current fraction of about 99%. In principle, the < 1% on-axis current can be partially “filled'” by the self-driven currents due to potato-like particle orbits near the magnetic axis. However, slight deviations from the “optimized” pressure (density or temperature) profiles drives the self-driven current away from the required equilibrium current-density profile. For ARIES-ST, it is assumed that capability to drive 5% of the equilibrium current profile externally is needed to ensure steady-state operation.

In the current-drive area, the unique magnetic topology of ST makes on-axis current drive with RF techniques difficult as most waves either do not penetrate to the center or suffer large damping. Low-frequency fast waves (LFFW) appear to be the only possible RF option. Unfortunately, LFFW requires a large antenna structure because of large parallel wavelength (~ 14 m). The data base for LFFW is small. For mid-plasma, high-frequency fast waves can drive the current efficiently. Current drive for profile control as well as during the start-up are additional issues.

Due to these difficulties, the reference current-drive system is based on positive-ion neutral beam injectors. With a beam energy of 120 keV, tangential injection at a major radius of 3.8 m drives a current profile which is peaked at $\psi = 0.8$. A total of 31 MW of NBI power is needed to drive 5% of the plasma current. The NBI system, together with additional 30 to 50 MW ECRH are also used during start-up and heating to ignition.

Because of the lack of space for a OH solenoid and the small amount of external current-drive power available, start-up is a challenge. For ARIES-ST, the plasma start-up process is envisioned to include two steps. Generation of a ~200 kA plasma either by ECH or using flux available from the divertor and outer poloidal-field coils is followed by a gradual ramp of plasma current (order of one to several hours) using bootstrap current over-drive.

3. MAGNET ENGINEERING

The design of the center-post is probably the most challenging engineering aspect of a spherical tokamak fusion system. Single-turn TF coils are preferred in order to reduce Joule heating through higher packing fraction and reduced shielding requirement (no insulation) even though such a TF system will require high-current, low-voltage supplies with massive busbars. The ARIES-ST TF is a single turn configuration. The TF return legs form a continuous shell with a constant area cross section, *i.e.*, the shell thickness decreases proportional to major radius away from the center-post. This shell configuration also allows the TF system to act as the primary vacuum boundary, so that the need for an additional vacuum vessel is eliminated.

The maintenance scheme envisioned for ARIES-ST centers around removal of the fusion power core and the center-post as a complete unit from the bottom (See Fig 1). Therefore, demountable joints are provided on the outboard side below the mid-plane. The fusion core, the center-post, and lower TF assembly are supported from below. The upper TF assembly is supported off the floor on the outboard of the demountable TF joint. Insulating joints are provided at outboard mid-plane for leads and bus connections; these connections are regularly spaced to ensure uniform current distribution in the TF shell and avoid error fields.

The center-post is flared to the degree possible in order to reduce Joule losses. The outer TF shell is extended on the top as a collar around the center-post with the center-post tapered for a tight fit to the outer shell. The electromagnetic loads on the collar tend to increase the contact pressure and lower the contact resistance at this point. At the bottom, the center-post is widened below the X point and a sliding joint provides the connection to the outer shell at this location. This arrangement allows removal and replacement of the center-post separately, if necessary.

The center-post is cooled with cold water with inlet and outlet coolant temperatures of 30 and 75^{o}C, respectively. The maximum conductor temperature is 125^{o}C. The coolant removes the center-post Joule losses (230 MW) and the nuclear heating (160 MW).

A 20-cm thick first-wall and shield is included in the inboard. This shield reduces the Joule losses in the center-post because of the reduced nuclear heating and the associated cooling requirement (leading to a higher conductor packing fraction). This shield also results in improved waste disposal, reduced irradiation damage (and increased resistivity due to transmutation), capturing the large nuclear heating as sensible heat, and additional life time.

4. FUSION CORE ENGINEERING

The high recirculating power fraction in a ST requires that a blanket design capable of high thermal efficiency be used. The high wall load and the water-cooled center-post narrow the options substantially. First wall and blanket design which include solid breeders require a major improvement in the thermal conductivity of solid breeders to handle high wall loads of a ST power plant. The option of self-cooled lithium with a vanadium structure technically could be utilized, but operation of such a blanket close to a water-cooled center-post is a safety concern. The reference ARIES-ST blanket design uses advanced ferritic steels as structural material with helium as coolant and LiPb as both a coolant and a tritium breeder. SiC composite inserts are used in order to achieve a high-coolant outlet temperature and reasonable power conversion efficiency.

Thermal hydraulic and stress analysis concluded that the first wall could tolerate up to 1 MW/m^2 of heat flux if the maximum temperature of ferritic steel could be extended to 600^{o}C. The 12-MPa helium coolant exits the blanket at 500^o (set by the maximum operating temperature of ferritic steel) but is superheated by the LiPb coolant, which has an exit temperature of 700^o. Through this technique, thermal cycle efficiency up to 42% has been obtained. Three-dimensional neutronics analysis of tritium breeding has been performed and a TBR of ~1.1 is obtained.

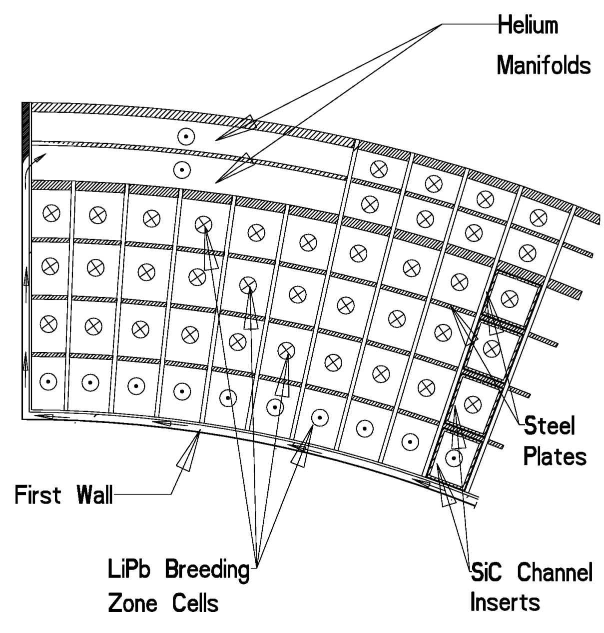

Fig 2. Cross-section of an ARIES-ST blanket module.