

Spherical torus concept as power plants—the ARIES-ST study

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Abstract

Recent experimental achievements and theoretical studies have generated substantial interest in the spherical torus concept. The ARIES-ST study was undertaken as a national US effort to investigate the potential of the spherical tokamak concept as a fusion power plant. This 1000 MWe fusion power plant conceptual design 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 plasma β_T of 50% (which is 90% of theoretical limit). While the plasma current is 28 MA, the almost perfect alignment of bootstrap and equilibrium current density profiles results in a current-drive power of only 28 MW. The on-axis toroidal field of 2.1 T and the peak field at the TF coil of 7.4 T led to 329 MW of Joule losses in the normal-conducting TF system. The power core uses an advanced ‘dual-cooled’ breeding blanket with flowing PbLi breeder and He-cooled ferritic steel structures that can achieve a thermal conversion efficiency of $\sim 45\%$. The ARIES-ST study has highlighted many areas where trade-off among physics and engineering systems are critical in determining the optimum regime of operation for ST power plants.

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1. Introduction

Theoretical studies indicate that the MHD performance of a tokamak plasma is substantially improved with decreasing aspect ratio. Peng and

Strickler [1] showed that low-aspect ratio tokamaks have unique physics features such as large natural elongation, strong magnetic helical pitch, and the existence of a large near-omnigenous region in the plasma cross section which suggests

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improved confinement. They introduced the name ‘spherical torus’, or ST for this concept. Recent experimental results from the START device at Culham [2] has been quite encouraging. Although small, the START experiment has reached average β_T values of 40%, more than three times that what has been obtained in a tokamak of conventional aspect ratio (β_T , the toroidal β , is the ratio of plasma pressure to pressure of toroidal field at plasma center). These values were obtained with plasma current $I = 250$ kA, central electron temperature $T_{e0} = 200$ eV, central electron density $n_{e0} = 6 \times 10^{19} \text{ m}^{-3}$, aspect ratio $A = 1.31$, plasma elongation $\kappa = 1.78$, with a total discharge duration time of about 20 ms. START has achieved a normalized β_N of 5.3 ($\beta_T = \beta_N I / aB$ where a is the plasma minor radius). While these high β_T values have not yet been demonstrated with high bootstrap fraction, it is hoped that the longer pulse-length Mega-ampere experiments MAST [3] and NSTX [4] may be able to demonstrate this.

For conventional aspect ratio tokamak power plants ($A > 2.5$), the dissipated power in normal-conducting toroidal-field (TF) coils is prohibitively large, necessitating the use of superconducting TF coils. In order to protect the superconducting TF coil from neutron damage and nuclear heating, at least 1 m of shielding is necessary between the superconducting TF coil and the plasma on the inboard side. As a result, tokamaks with superconducting TF coils optimize necessarily at medium to large aspect ratios. At low aspect ratio (e.g. spherical torus), the plasma β_T 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 are possible.

Inherently, a ST power plant should operate in steady state because of lack space in the inboard for an ohmic solenoid. A ST device will have larger recirculating power than a conventional tokamak because of resistive losses in the TF coils and the need to drive a large plasma current. For a ST power plant to be viable, plasma equilibria with a high β_T (to minimize Joule losses in the TF coils) and high bootstrap current fraction ($\sim 99\%$)

should be found. More recent theoretical studies by Menard [5], Miller [6], and coworkers have shown that for certain specific classes of plasma equilibrium with close fitting conducting walls, there exist stable solutions to the plasma equilibrium equations with very large values of the plasma β_T and with essentially all of the plasma current self-provided by the bootstrap effect.

The major attribute of a ST as a power plant is high β_T . It had been argued [7] that the resulting increase in fusion power density ($\sim \beta^2 B^4$) more than compensates for inherently larger recirculating power fraction in a ST, leading to power plants that are more economical compared with advanced tokamak power plants with superconducting coils. Assuming advanced ST equilibria that require very little current-drive power, the optimization of ST naturally leads to a trade-off between the power density and the Joule losses in the TF coils. High plasma β_T in ST is due to the increase in β_N and, more importantly, increase in allowable plasma elongation, κ . As such, ST device are tall and elongated (see Fig. 1). The low aspect ratio also restricts the space available for the inboard legs of TF coils (TF centerpost), in fact, most of this space should be devoted to the TF centerpost. The TF centerpost is a central challenge for ST because of limited space and the high field and large forces on centerpost conductor (because of the $1/R$ dependence of toroidal field strength). Furthermore, in order to minimize TF Joule losses, previous studies have argued for the elimination of inboard breeding blanket, inboard shield, and even inboard scrape-off layer (by ‘leaning’ the ST plasma on the centerpost). These also make centerpost design more challenging.

As a result of favorable experimental results, theoretical studies, and power plant evaluation, a detailed investigation of the spherical torus concept as a fusion power plant, ARIES-ST study, was initiated. This paper provides an overview of ARIES-ST study. We will discuss optimization of ST power plants, trade-offs and high-leverage issues, key findings, and directions for ST research. Detailed results from ARIES-ST can be found in other papers in this issue [8–15].

Due to the need for high plasma β_T , considerable effort was focused on developing advanced

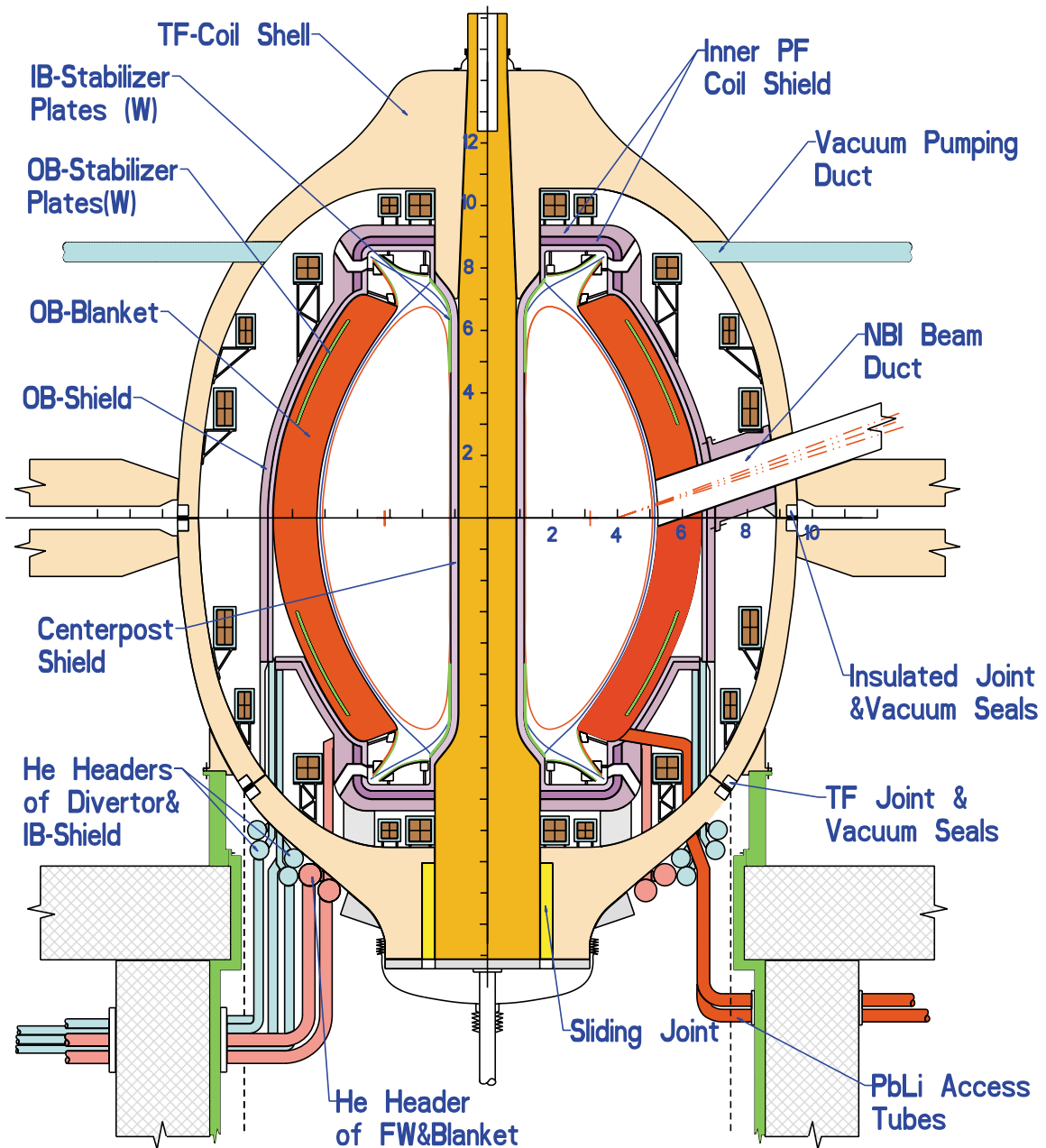


Fig. 1. Elevation view of the ARIES-ST power core.

ST equilibria with large bootstrap fraction. Section 2 summarizes this activity as well as other ST physics issues. Several approaches to TF centerpost design were examined. The most attractive

option appears to be a water-cooled centerpost made of copper alloys operating in ambient temperature (35 °C water inlet). We also found that TF Joule losses are actually reduced and

system performance is improved when a ~ 20 cm shield is located in front of the centerpost. Magnet engineering is presented in Section 3.

Fusion power core engineering is reviewed in Section 4. The high recirculating power fraction in a ST requires that a blanket design capable of high thermal efficiency be used. 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 major safety concern. The reference ARIES-ST blanket design uses advanced ferritic steels as structural material with helium as coolant and PbLi as both a coolant and a tritium breeder. SiC inserts are used in order to achieve a high-coolant outlet temperature and reasonable power conversion efficiency. This innovative design allows the coolant outlet temperature (700°C for PbLi) to be higher than the maximum allowable temperature of the structural material ($550\text{--}600^\circ\text{C}$ for advanced ferritic steel). In addition, recent advances in gas cycle thermal conversion systems allow a thermal conversion efficiency of 45% for ARIES-ST.

We will present the safety and licensing attributes of ARIES-ST in Section 5. Section 6 examines the extent to which ARIES-ST has met the top-level requirements for a desirable fusion power plant [16], identifies the major trade-offs in arriving at ARIES-ST design points, and describes the key R&D topics.

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 a plasma triangularity of 0.64. This configuration attains a β_T of 50% (which is 90% of the maximum theoretical β). While the plasma current is 28 MA, the almost perfect alignment of bootstrap and equilibrium current density profiles results in a current-drive power of only 28 MW. The on-axis toroidal field is 2.1 T and the peak field at the TF coil is 7.4 T, which leads to 329 MW of Joule losses in the normal-conducting TF system (222 MW in the centerpost). Figs. 1 and 2 show a cross section and perspective of ARIES-ST. Tables 1 and 2 summarize the key parameters and a set of cost parameters of ARIES-ST final design point.

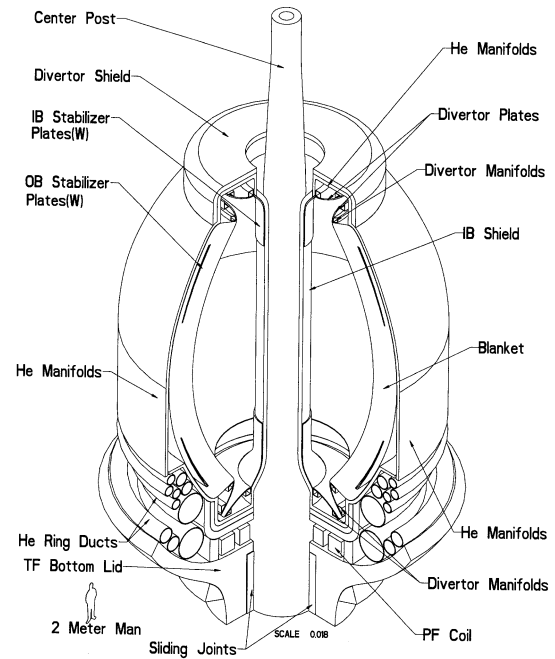


Fig. 2. Isometric view of ARIES-ST replaceable power core unit sitting on top of its movable/reuseable support platform.

2. Plasma physics

2.1. MHD stability

In order to achieve a reasonable recirculating power fraction in a ST power plant, it is essential to have a high β_T (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 with almost perfect bootstrap alignment were performed in order to map out the optimum operating regime for ARIES-ST. This study covered a wide range of aspect ratio ($A = 1.25, 1.4, 1.6$, and 1.8), elongation, triangularity, and squareness parameter [17]. For analyzing the stability of these equilibria, BALLOON [18,19] and BALOO [20] codes were used for high- n ballooning modes, and the PEST-II [21] and GATO [22] codes were used for computation of both marginal β values and the marginal wall position which stabilizes low- n kink modes. Kink stability analysis were per-

Table 1
Reference parameters of 1000 MWe ARJES-ST power plant

<i>Plasma parameters</i>	
Plasma aspect ratio	1.60
Major radius (m)	3.20
Minor plasma radius (m)	2.00
Plasma vertical elongation (κ_{95})	3.40
Plasma triangularity (δ_{95})	0.64
Plasma current (MA)	29
Bootstrap current fraction	0.96
Current drive power to plasma (MW)	28
Toroidal field on axis (T)	2.1
Peak field at TF coil (T)	7.4
Toroidal β	50% ^a
Normalized β_N	7.4 ^a
Poloidal β	1.70
On-axis safety factor	4.3
Plasma-edge safety factor	11
Electron temperature (keV)	16.5
Ion density (10^{20} m^{-3})	1.4
Electron density (10^{20} m^{-3})	1.6
Density profile peaking factor ($n_0/\langle n \rangle$)	1.23
Temperature profile peaking factor ($T_0/\langle T \rangle$)	1.14
ITER-93H scaling multiplier	1.47
<i>Plant parameters</i>	
Average neutron load (MW m^{-2})	4.1
Fusion power (MW)	2980
Total thermal power (MW)	3370
Center-post ohmic losses (MW)	222
TF-coil ohmic losses (MW)	329
Recirculating-power fraction	0.34
Net plant efficiency	0.30
Mass power density (kWe tonne^{-1})	55.0
Cost of electricity (mill per kWe h)	78.6

^a ARIES-ST operates at 90% of maximum theoretical β .

formed for $n = 1$ –6 kink modes as the intermediate- n kink modes often determine the minimum wall separation distance. All bootstrap current calculations use the collisionless model described in [23] and use an accurate approximate expression for the trapped particle fraction derived in [24]. These studies are summarized in [8] and described in detail in [5,6].

The generated database of the stable MHD equilibria was used by the ARIES systems code for trade-off studies. Several trends become clear at the outset: first, at each value of plasma aspect ratio, plasmas with highest value of β_T are the optimum configuration. Since $\beta_T \beta_p \propto \beta_N^2 (1 + \kappa^2)$, even modest improvements in either β_N or κ are important in power plant optimization. Second,

optimum aspect ratio for ST power plants is between $A = 1.4$ and 1.8 (with a shallow minimum between $A = 1.6$ and 1.8 [9]).

Use of a conducting wall and plasma rotation and/or active feedback to stabilize pressure driven kink modes is of crucial importance to the ST power plant concept, as it yields a significant increase in the stable β_N limit. Such wall stabilization is assumed in nearly all of the stability analysis that follows. As such, β_N depends on the location of the kink-stabilization wall. Similarly, highest attainable κ depends on the location of vertical stability shells. In ARIES-ST, tungsten shells are used for both functions. On the inboard, tungsten plates are attached to the first wall and also act as high heat flux components. The thickness of these plate is 6.5 cm including the coolant channels (the ‘effective’ conductor thickness is 5 cm). On the outboard side, tungsten plates are located inside the blanket (near the shield region), the location of these shells are mainly determined by the tritium breeding requirements. Our analyses indicate that elongations of $\kappa \geq 3$ are possible because of the natural elongation of ST plasmas as well relatively close location of the stabilization shells.

We also found that positive squareness ($\zeta = 0.1$) increases β_T by about 7–10%. Unfortunately, we found that generating a free-boundary shape with positive squareness ($\zeta = 0.1$) dramatically increases the poloidal field (PF) stored energy relative to shapes with zero or negative squareness. On the other hand, the impact on β_T only becomes significant when ζ is reduced below 0.0. For this reason, the equilibrium boundary of the final ARIES-ST configuration has a squareness near zero. The resulting final ARIES-ST power plant configuration has $A = 1.60$, $\kappa = 3.4$, $\delta = 0.64$. The ideal MHD stability limit for this design is $\beta_T = 56\%$ and $\beta_N = 8.2$. Consistent with the other ARIES studies, we choose the operating point to be 10% below the stability limit to give some margin to reduce the likelihood of plasma disruptions. Thus, the design point is at $\beta_T = 50\%$, $\beta_N = 7.4$, and $f_{BS} > 99\%$. The reference plasma configuration is given in Fig. 3 and Table 1.

The PF coil locations and currents to provide the required plasma equilibrium are determined

Table 2
ARIES-ST power-plant economic parameters (1992 \$)

Account number	Account title	Million dollars
20	Land and land rights	10.6
21	Structures and site facilities	370.8
22	Reactor plant equipment	1224.1
22.1.1	First wall, blanket, and reflector	50.2
22.1.2	Shield	113.1
22.1.3	Magnets	102.6
22.1.3.1	TF-coil 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
		Constant dollars

Table 2 (Continued)

Account number	Account title	Million dollars
[90]	Unit direct cost, UDC (\$ kWe ⁻¹)	2321.7
[94]	Unit base cost, UBC (\$ kWe ⁻¹)	3289.4
[99]	Unit total cost, UTC (\$ kWe ⁻¹)	3844.5
	Capital return (mill per kWe h)	65.04
[40–47, 51]	O&M (1.4%) (mill per kWe h)	9.28
[50]	Blanket replacement (mill per kWe h)	3.71
	Decommissioning (mill per kWe h)	0.50
[02]	Fuel (mill per kWe h) (LSA = 2)	0.05
	Cost of electricity, COE (mill per kWe h)	78.57

from free-boundary equilibrium calculations. The above class of ST equilibria presents special challenges for several reasons. It is especially

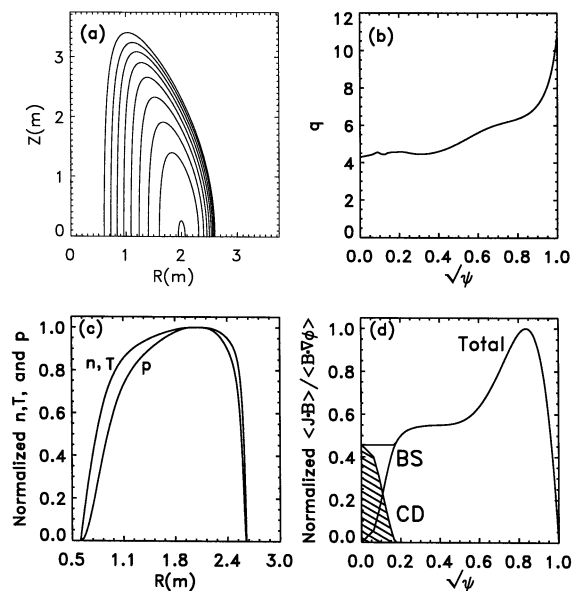


Fig. 3. The reference plasma equilibrium of ARIES-ST with $A = 1.60$, $\kappa = 3.4$, $\delta = 0.64$, $\beta = 56\%$ and $f_{BS} > 99\%$. (a) poloidal flux contours, (b) safety factor profile, (c) pressure, temperature, and density profiles, (d) total current, bootstrap current, and density current drive profiles.

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. We developed PF solutions with all of the PF coils located outside the TF coil shell (with the exception of the driver coils). This is desirable from a configuration and maintenance point of view. This solution, however, requires large currents in the PF coils, leading to large stored energy and cost. As such, all of the PF coils in the reference ARIES-ST design are located inside the TF coils (see Fig. 1).

The vertical stability analysis was also difficult due to the extreme plasma elongation combined with very high β and very low l_i . As such, the vertical position feedback control power could be estimated, but not directly simulated due to numerical difficulties for these plasmas. Resistive-plasma analysis by TEQ [25], using a plasma with approximate ARIES-ST plasma properties, indicated a plasma growth time of 70 ms. The required peak power required for vertical position control was estimated to be 105 MVA (for a 1 cm random vertical disturbance to the plasma). The feedback coils were located just behind the shield at about 45° from the outboard mid-plane. Some full nonlinear dynamic simulations with TSC [26] were also performed with approximate plasma profiles. These calculations showed that the plasma vertical instability was evolving on a resistive time scale, which implies that the structures are indeed influencing the instability. In addition, the dynamic simulation showed the highly non-rigid behavior of the instability at such high elongation.

2.2. Current drive

The ARIES-ST plasma equilibrium yields a pressure-driven current fraction of about 99%,

requiring $\leq 1\%$ on-axis current drive. Since the amount of driven current is small, driver efficiency is not a critical consideration. Rather, penetration to magnetic axis is the main issue. We have found that most RF current drive schemes cannot drive current on axis of ARIES-ST [8]. For example, lower hybrid and electron cyclotron (O-mode) waves cannot penetrate to plasma axis; and ICRF fast waves and high harmonic fast waves result in an off-axis current profile. The only possible RF scheme for on-axis current drive is low-frequency fast waves (8.5 MHz) resulting in ~ 0.03 A/W current drive efficiency. The concerns with this scheme include very limited experimental data base and the large size of antennas. High energy neutral beams (NBI) can also drive current on axis with a reasonable efficiency ~ 0.03 A/W. Beam energy, however, should be high enough (5 MeV) to peak the current on-axis. Issues with using NBI include extrapolation of NBI technology to 5 MeV. In addition, a broad pedestal of current is driven off-axis and requires re-optimization of equilibrium current density profiles. Recently, [27] proposed that the bootstrap current on the magnetic axis is nonzero as particle orbits close to the axis are not banana in shape, but actually are shaped like a potato, and the fraction of trapped electrons with potato orbits is not zero as ψ approaches zero. An estimate shows that the potato bootstrap current density on-axis can be significant compared with the banana bootstrap current density in the mid-plasma region. This theory, thus, provides the intriguing possibility of a 100% bootstrap driven tokamak.

Based on the above assessment and because the current-drive power needed to drive on-axis current is very small (~ 1 –3 MW), the ARIES-ST reference design relies on the self-driven current due to potato-like particle orbits near the magnetic axis to provide the on-axis seed current on ARIES-ST. We note that even though the ‘potato-orbit’ current is based on well-founded theory; it remains to be unambiguously reproduced in a fully kinetic particle simulation or measured in a tokamak experiment. Alternatively, high-energy NBI and/or low-frequency fast wave can be used for driving the on-axis current seed.

While ARIES-ST plasma equilibrium generates a pressure-driven current fraction of about 99%, a slight deviation from the ‘optimized’ pressure (density or temperature) profiles drives the self-driven current away from the required equilibrium current–density profile and, thus, some form of current-profile control is necessary. To be prudent, it is assumed in ARIES-ST that capability to drive 5% of the equilibrium current profile externally is needed to ensure steady-state operation. We have assumed that this current-profile control system should be capable of driving current with a profile similar to that of equilibrium current–density profile. For typical ARIES-ST equilibria that have been studied, this implies a driven current profile that is peaked at $\hat{\psi} = 0.8$, with $\hat{\psi}$ being the normalized poloidal flux. Most RF current drive techniques are capable of current profile control in ARIES-ST with reasonable current-drive efficiency. As profile control is mostly needed near the plasma periphery ($\hat{\psi} = 0.8$), low-energy neutral beam is an excellent candidate. The ARIES-ST reference design includes a 120 keV NBI system that requires ~ 30 MW of power. The added benefit of the low-energy NBI system is the induced plasma rotation necessary for stabilization of kink modes. This NBI system, together with an additional 30–50 MW of ECH power, is used during start-up and heating to ignition.

2.3. Divertor

There is little experimental data base on scrape-off-layer (SOL) physics in a ST. Even in modest aspect ratio tokamaks, the mechanism (or mechanisms) responsible for determining SOL properties in double-null divertors are at present not well understood. Therefore, there is a large uncertainty in calculated peak heat fluxes in divertor plates. We have used experimental data from DIII-D together with theoretical predictions to arrive at estimated divertor heat fluxes. The most important parameter is the mid-plane heat flux scrape-off length, λ_p . For DIII-D, λ_p at the outboard mid-plane is typically about 1 cm, while λ_p at the inboard mid-plane is much less than 1 cm and may, in fact, depend on toroidal field B_T . Based on available DIII-D data, we assumed that $\lambda_p \propto 1/B_T$

and a 5:1 ratio between outboard and inboard power flow [8]. For the most conservative case of no radiation in the SOL and divertor, the peak heat flux on the outboard and inboard divertor plates are calculated to be 16 and 31 MW m^{-2} , respectively (for plate normal to the SOL) [8]. Two methods were identified to reduce the heat flux to more manageable levels.

First, by inclining the divertor plates by 22° on the outboard and 11° on the inboard, the peak heat fluxes can be reduced to $\sim 6 \text{ MW m}^{-2}$, the design values for ARIES-ST divertors. This is indeed done for the outboard divertor.

It is very difficult to arrange for 11° inclination on the inboard divertor plate due to lack of space. Experiments on DIII-D have demonstrated that energy confinement in ELMing H-mode plasmas is unchanged when the plasma is lightly limited on the inboard wall and that this power is distributed nearly uniformly along the inboard protective armor. In principle, if the double-null separatrix is lightly limited on the centerpost armor, the entire inboard power flow can be directed to the centerpost and away from either inboard divertor. In ARIES-ST, the total power flow along the inboard SOL is estimated at ~ 44 MW. If this power is distributed uniformly on the centerpost first wall, we estimate that $Q_{\text{div,centerpost}} \approx 0.6 \text{ MW m}^{-2}$. The resulting heat flux on the centerpost first wall would be $\sim 1.1 \text{ MW m}^{-2}$ (its design value). Further reduction in the inboard SOL power can be achieved by shaping the inboard first wall such that most of the inboard SOL power is deposited on the tungsten stabilizing shells. If centerpost first wall can handle a higher heat flux (e.g. 2 MW m^{-2}), it would be even possible to limit the separatrix on the centerpost first wall and armor. In this case, the heat flux on the outboard divertor can also be greatly reduced.

It should be emphasized that the above heat flux estimates may serve as an upper bound for ARIES-ST, because the radiated power outside the separatrix has not been considered.

2.4. Plasma startup

Due to 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 PF coils, followed by a gradual ramp of plasma current (order of 1 to several hours) using bootstrap current over-drive [8].

3. Toroidal-field coil system

The inboard leg of the TF coils, the centerpost, is the most critical system in a ST. As mentioned before, the dominant optimization path in a ST is the trade-off between power density (lower aspect ratio, smaller devices) and recirculating power fraction (mainly Joule losses in the TF coils). Centerposts with small cross section are desirable as they allow for lower aspect ratio and smaller devices leading to a high power density. At the same time, smaller diameter centerpost increases the Joule losses in the centerpost and the power-plant recirculating power fraction. Mechanical design of the centerpost is also a challenge even though the on-axis field is low (~ 2 T in ARIES-ST). The forces on the TF coils are comparable to superconducting tokamaks because of the high field on the centerpost (~ 7.6 T in ARIES-ST) caused by $1/R$ dependence of the toroidal field and large plasma current. As such, detailed scoping studies of centerpost design were performed in the initial phases of the ARIES-ST research [13]. Results are summarized below.

3.1. Centerpost options

3.1.1. Conductor options

Superconducting centerposts are much bigger than normal conducting ones because of large space needed for shielding. In addition, assembly and maintenance of the device will be very difficult if the TF coils are not ‘demountable’. Use of liquid metal as conductor was considered in the context of an integrated blanket coil (IBC). In the IBC concept, liquid lithium serves as a breeding material, a conductor, and a coolant with the Joule losses directly deposited in the coolant and

partially recovered by the power conversion system. Calculations showed that such an IBC is unattractive because of excessive ‘net’ Joule losses. For the centerpost conductor, copper is the best because of its very high conductivity. In particular, copper alloys such as CuCrZr (precipitation hardened) or GlidCop AL15 (dispersion strengthened) provide sufficient strength to handle mechanical loads. For the outboard TF legs, both copper and aluminum are candidates since space is not an issue. Aluminum is particularly of interest because it leads to less massive legs.

3.1.2. Cooling options

Several cooling options were considered ranging from (a) cooling with liquid lithium or water at elevated temperature (to recover Joule losses) to (b) water at ambient temperatures to (c) cryogenic cooling with Liquid nitrogen (80 K) and gaseous helium (30 and 10 K) to take advantages of reduced copper resistivity at cryogenic temperatures. Analyses showed that operation at elevated temperature leads to 50–70% higher ‘net’ electric power consumption because the large increase in the copper resistivity with temperature more than offset any energy recovered. Operation at cryogenic temperatures did not lead to reduction in power consumption because of poor refrigeration efficiency. Cryogen-cooled options also appear to be much more complex. As a result, it appears that water cooling at ambient temperature (inlet ~ 30 °C) is the preferable option.

3.1.3. Single versus multi-turn

Single-turn TF coil is preferable in a ST because: (a) it results in a higher conductor packing fraction (no turn-to-turn insulation, flexible conductor and coolant geometry); (b) monolithic construction leads to a stronger mechanical design for the centerpost; (c) changes in electrical conductivity over time (due to transmutation of copper) are accommodated by natural current redistribution within the centerpost; (d) it leads to an acceptable assembly and maintenance scheme. The disadvantages of single turn coils include low operating voltage and high operating current that lead to large busbars and ‘unconventional’ power supplies. Overall, stringent requirements on Joule

losses and mechanical design strongly favor single-turn coils.

3.1.4. Centerpost shield

As the reduction of centerpost diameter is a critical parameter for a ST power plant, at first glance it appears that a ‘bare’ centerpost (i.e. without any inboard neutron shield) would be the best option. Copper conductor of the centerpost can act as a neutron shield and has the additional benefit of conducting current and reducing Joule losses. This observation, however, is incorrect because of two effects: (1) very large nuclear heating in the outer layer of a ‘bare’ centerpost (can be ten times the Joule heating or more), and (2) limited range of operation temperature for a water-cooled centerpost. Both effects lead to a large coolant (water) fraction in the outer layers of a ‘bare’ centerpost. As water is not as a good shielding material compared with metals, the neutrons penetrates further in the centerpost, requiring large coolant channel in the subsequent layers, etc. As such, the outer ~ 40 cm of centerpost has a very low packing fraction and does not contribute significantly in reducing Joule losses.

The same neutron shielding function can be performed by a thinner metallic shield that can operate at high temperature (and, therefore, with a much lower coolant fraction) and can be made of a better shielding material. As a result, the overall diameter of the centerpost will be reduced. In addition, as the nuclear heating is recovered as useful heat, the ‘net’ power consumption of the centerpost decreases. Such a centerpost shield has significant additional benefits such as: (1) it limits the increase in centerpost resistivity due to transmutation of copper; (2) it reduces the radiation damage to copper conductor; (3) it helps in achieving adequate tritium breeding ratio, and (4) it allows the centerpost to meet class C waste disposal and prolongs the centerpost life time. For these reasons, ARIES-ST includes a 20 cm inboard shield between the centerpost and the inboard first wall/armor (see Fig. 1).

3.2. Centerpost design

The ARIES-ST TF system is shown in Fig. 4. The centerpost is connected to an outer shell, made of aluminum that surrounds the first wall, blanket, shield, divertors, and PF coils. This outer shell is continuous and is effective in reacting both in-plane and out-of-plane electromagnetic loads. In addition, TF ripple is minimized with a continuous shell approach. The TF system also provides the primary vacuum boundary for the machine. The centerpost is designed to be physically separable from the power core assembly. The bottom portion of the centerpost is electrically connected to the outer shell by Feltmetal pads sliding joints (used on C-Mod and on MAST [3]). The centerpost and outer shell are keyed together in this location, permitting relative radial and vertical motions while keeping them registered toroidally. This sliding joint significantly reduces axial stresses in the centerpost and is critical in the mechanical design of centerpost. The centerpost has a conical shape above the upper divertor assembly where it is pulled against a mating surface in the outer shell for electrical continuity. This arrangement allows the centerpost to be removed either as part of the power core assembly or by itself without disturbing the power core assembly or (see Section 4.1).

Electrically, the TF system has only a single turn carrying 34 MA of current. This leads to a voltage drop across the TF coil system of 8.5 V and 290 MW of Joule losses. In order to minimize Joule losses in the leads, the TF power supplies have to be located as close to the TF coils as possible. A cylindrical biological shield is provided at a major radius of approximately 13 m. This concrete shield is > 2 m thick. The TF power supplies are located just outside this shield. The large conductor current (34 MA) in the single turn configuration, while formidable, does not appear intractable from a power supply standpoint [13]. Large busbars are also used in order to reduce Joule losses outside the TF coils. The stored magnetic energy in the TF coils is 6.2 GJ.

During initial operation (‘new’ centerpost), a total of ~ 405 MW of power is deposited in the centerpost, consisting of 242 MW of Joule heating

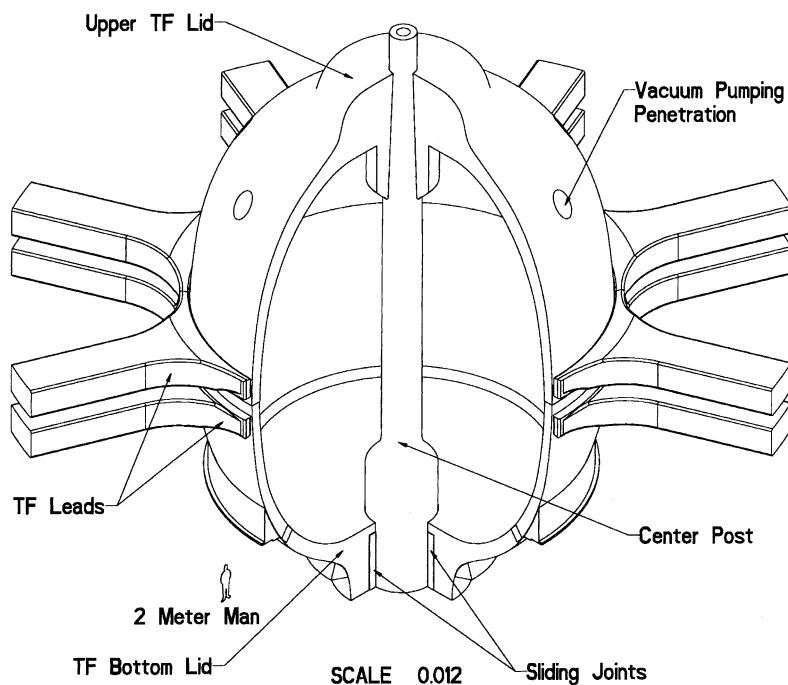


Fig. 4. Isometric view of ARIES-ST TF coil system.

and 164 MW of nuclear heating. The water coolant velocity is limited to 10 m s^{-1} to avoid excessive erosion. The required flow rate is 2155 kg s^{-1} with inlet and outlet temperatures of 30 and 75°C , respectively. The Joule heating in the centerpost increases by 33 MW after 3 FPY. With the same flow rate, the outlet temperature increases by 4°C to 79°C .

Optimal configuration of a ST power plant generally favors a low aspect ratio and a high elongation making the plasma to be very tall and very close to the centerpost. The 30 m long centerpost is comprised of a tapered section in the upper region to electrically mate with the outer shell, a straight cylindrical section that is in close proximity with the plasma, and a larger cylindrical section at the bottom to increase the conductive area to the maximum extent to lower ohmic losses. The 30 m, unitary centerpost is constructed of high-conductivity copper. It is cooled with low-temperature water in a single pass configuration. The centerpost is comprised of roughly 85% copper and 15% water. This part weighs approximately ~ 850 tonnes.

A continuous shell has been used as the outer legs of the TF coils for mechanical strength and reduction of the ripple. As shown in Figs. 1 and 4, there are four parts to the TF coil system. The outer TF shell has three distinct parts (see Fig. 1) The upper section extends from the centerpost to the outboard mid-plane where it is connected to one of the power supply busbar leads. The middle shell extends from mid-plane, where the other busbar connection is made, down to a maintenance break. The upper and middle sections are bolted together with an electrically insulating material in between. A bellows-type connection (with an insulating break) on the inside of the outer shell provides the vacuum barrier across the joint. The third shell is from the maintenance break down to the lower connection to the centerpost. This shell is removable during maintenance actions. A joint provides the electrical continuity between the middle and lower sections. The outer shells could be made from either copper or aluminum. Copper is more conductive than aluminum but it would be heavier for the same effective conductivity. The thickness of the alumi-

num shell is adjusted to yield the proper coil resistance and recirculating power losses. Again, the shells would be water-cooled with an overall 15% water fraction.

3.3. Centerpost fabrication

The conventional method of constructing the copper centerpost assembly with internal water passages would be to fabricate wedges with grooves for the coolant and then weld the entire assembly in a fixture to minimize distortion. This would be a difficult and costly technique involving extensive welding and inspection. The cost to fabricate this part is well over $\$80 \text{ kg}^{-1}$ (compared with $\sim \$3 \text{ kg}^{-1}$ material cost). Thus, the centerpost would likely cost $\$70$ – $\$100$ millions or more (note that the centerpost has to be replaced on a ~ 3 year basis).

The thicker cross-sections of the TF shells (up to 2.5–3.0 m thick) at the top and bottom complicates their fabrication approach. The most likely approach is to cast the TF shells in smaller pieces and then join sectors together by welding. It is difficult to reliably join the internal water passages. The unit cost to fabricate such an aluminum component might well be over $\$100 \text{ kg}^{-1}$ (compared with $\sim \$2 \text{ kg}^{-1}$ material cost). The total weight of the three aluminum shells is 2690 tonnes, and they would likely cost $\$270$ millions or more. If it were constructed of copper, the cost would be significantly higher. In summary, a conventional approach is a difficult and expensive way to fabricate the TF coils.

We have investigated two advanced fabrication techniques that can eventually lead to dramatic reduction in the cost of the TF coils for ARIES-ST [15].

3.3.1. Laser or plasma arc forming

The laser forming process starts with a thin panel or skin of the required thickness. Then a stream of powdered metal is directed toward the part and a laser melts the metal onto the substrate. As in the stereo-lithography process, the new material is applied only where required to form the 3-D part. The application rate of the material is limited only by the power of the laser and the

compatibility with the material. Titanium aircraft parts have been manufactured by laser forming process. Fatigue tests of these components show values comparable to high end of traditional cast titanium and the low end of wrought titanium [15]. It is anticipated that copper could be laser fabricated and exhibit properties slightly less than wrought material. Plasma arc sources (instead of lasers) may be better suited to copper at present. The manufactured parts typically have a desired form with high surface finish such that additional machining is minimized. While this technique is still in development stage (and, therefore, costly) and only small components have been produced, we have not identified any fundamental problem in scaling up this method to a large components such as the ARIES-ST centerpost. Using ten such laser forming heads, simultaneously directed to deposit layer by layer of material, we have estimated that the centerpost can be manufactured in about 6 months. Furthermore, because of the highly automated nature of the process, labor costs are minimized, and we estimate a unit cost of $\$8 \text{ kg}^{-1}$, about a factor of 10 reduction compared with a conventional approach [15].

3.3.2. Spray casting

The outer shell of the TF coils could also be constructed with laser forming. However, as the TF shell is several times more massive than the centerpost, the fabrication time would be similarly longer. Due to the relatively simple geometry of the outer TF shell, another fabrication technique, spray casting, has been proposed to speed up the process. Spray casting of a molten metal involves holding the metal just above the melting point temperature, atomizing it, and spraying it onto a preform structure [15]. This process is currently under evaluation and development for manufacturing large components. For ARIES-ST, we propose utilization of four spray heads that yield 7200 kg h^{-1} material casted (~ 26 times higher than ten-head laser forming process). As such, the TF shell can be manufactured also in about 6 month. Again, because of the highly automated nature of the process, labor costs are minimized; and we estimate a unit cost of $\$4$ – $\$5 \text{ kg}^{-1}$, about a

factor of 20 reduction compared with a conventional approach [15].

4. Fusion core engineering

The ARIES-ST power core is designed in response to unique features of an optimized ST power plant configuration. The highly elongated plasma together with an integrated outboard TF shell and vacuum vessel led to a vertical maintenance scheme and toroidally integrated blanket. An attractive single piece maintenance scheme is proposed for ARIES-ST and described in Section 4.1.

First wall, blanket, and in-vessel components choices are limited because: (a) high thermal conversion efficiency is needed to offset the effect of high re-circulating power in the normal-conducting TF system; (b) the use of water coolant in the power core (for the copper centerpost) strongly discourages the use of reactive materials such as lithium and beryllium; and (c) the absence of space on the inboard side for a breeding blanket places additional constraints on material selection and dimensions. The ARIES-ST blanket includes several innovative design features and leads to a new class of attractive blanket systems for fusion power plants:

- 1) Ferritic steels have the largest database of fusion low-activation material. It was perceived, however, that the relatively ‘low’ maximum operating temperature of ~ 550 – 600 °C of ferritic steel would lead to an unacceptably low thermal conversion efficiency. This perception was based on the assumption that the coolant outlet temperature will always be lower than the maximum structure temperature. ARIES-ST innovative blanket design leads to a PbLi coolant outlet temperature of ~ 700 °C, higher than the maximum structure temperature.
- 2) Recent interest in gas-turbine power generators has resulted in large R&D efforts in high-efficiency gas cycles. In particular, development of high-efficiency recuperators allows Brayton cycles with high efficiency even at a

modest temperature. ARIES-ST features such a Brayton cycle with $\sim 45\%$ thermal efficiency.

The ARIES-ST first wall and blanket design is described in Section 4.2 and the power cycle is presented in Section 4.3. Detailed thermal hydraulic and stress analysis of ARIES-ST power core can be found in [10]. Detailed neutronics analyses [11] have raised no serious difficulties about the tritium breeding (TBR of 1.1) or neutron shielding performance of the ARIES-ST power core.

4.1. Configuration and maintenance

Horizontal maintenance through large ports in the vacuum vessel was adopted in ARIES-RS to help meet availability goals by enabling rapid sector replacement [28,29]. However, the configuration and maintenance procedures for a ST power plant with a low aspect ratio is likely to be very different than those for a standard tokamak such as ARIES-RS because of the unique ST geometry. The fusion core components in a ST power plant are very tall and ‘thin’. The inboard geometry is substantially different than that of the outboard, and the inboard TF coil (centerpost) will most likely have to be replaced frequently. Probably the most important difference is the possibility for demountable normal-conducting coils. Both horizontal and vertical maintenance schemes were evaluated for ARIES-ST [30]. Vertical maintenance of fusion core as a single piece appears to be the best option for a ST power plant.

Due to the modest replacement cost ($\sim \$50$ millions), efforts to segment the blanket radially and/or re-use parts of the power core were not explored, although such techniques are possible, and could help reduce the waste stream considerably. Rather a single-piece maintenance approach is utilized. Such a single-piece maintenance can help improve plant availability considerably by shortening the power core replacement time.

In view of the weight of the replaceable power core unit, a system of hydraulic jacks (instead of overhead cranes) is employed for vertical replacement of the power core from the bottom. This approach reduces the required space above and at

the outer peripheries of the biological shield for component lay-down area, minimizing the size of the building. In addition, radioactive components and particles are confined to a smaller area.

Fig. 1 shows a cross section of the power core in its normal operating position within the vacuum vessel (which also serves as the TF return current leg). The replaceable power core unit is shown in Fig. 2, sitting on top of its movable (reusable) support platform which is the bottom portion of TF shell. The replaceable power core has three components: (1) the outboard breeding blanket and its first wall, (2) the inboard first wall and shield (including the inboard vertical stability shells), and (3) the divertors. The replaceable power core is assembled as an integrated unit outside the TF shell in ‘hot cell’ assembly and maintenance areas, and transported into place using a combination of rail and hydraulic lifting systems.

The power core components are then mounted on the TF shell, which incorporates the bottom divertor PF coils and the coolant headers (see Fig. 2). All power core coolants enter and exit through this lower support platform. This allows all maintenance connections to be made and broken outside the primary vacuum in a secondary containment area. Poloidal continuity is provided by the helium manifolds which surround the blanket and divertor, and are attached to the inboard first wall and shield. The manifolding, therefore, serves not only to route the two coolants but also to provide structural integration of the power core.

Fig. 5 shows the maintenance sequence for ARIES-ST. After separating the coolant headers, TF joints and vacuum seals that connects the bottom portion to the outboard TF leg (see Fig. 1) are disconnected. The replaceable TF assembly is then lowered and moved to a ‘hot cell’ area. A new power core assembly is lifted into place, the lower TF joints/vacuum seals are made, power coolant lines are connected, and power plant is brought on line. Provisions are also provided for removal of the centerpost by itself if repairs are necessary. Detailed CAD drawings of ARIES-ST machine assembly and maintenance sequence can be found at <http://aries.ucsd.edu/ARIES/>.

4.2. First wall, blanket, and shield

4.2.1. Outboard first wall and blanket

The ARIES-ST power core features an advanced dual-cooled ferritic steel blanket with He as coolant and flowing eutectic lead–lithium alloy Pb–17Li as breeder and coolant. This design borrows features of both the EU dual-coolant DEMO blanket [31] and the PbLi-cooled SiC composite blanket, ‘TAURO’ [32]. The ARIES-ST blanket includes several innovative design features that lead to a new class of attractive blanket systems for fusion power plants: blankets with coolant outlet temperature higher than the maximum structure temperature and using advanced Brayton cycles.

Cross sections of the outboard blanket and first wall is shown in Figs. 6 and 7. The first wall box is fabricated by diffusion welding and subsequent bending of the straight plates containing the milled coolant channels [31]. The resulting array of ‘I-beams’ creates stiffness against toroidal bending. This first wall forms, together with the helium manifolds at the back side of the segment, a box containing the flowing liquid metal breeder. A grid of steel plates inside this box creates large liquid metal ducts and reinforces the first wall box, providing stiffness against poloidal bending.

The helium coolant at 300 °C first cools the first wall and then the grid plates and exits at 525 °C. This coolant routing arrangement leads to lower temperatures in the first wall and higher coolant temperatures in regions with lower power density, minimizing temperature differences in the blanket structure. The helium coolant maintains the temperature of the steel structure below 600 °C.

The PbLi coolant/breeder enters the blanket at the bottom, flows upward in the front row of ducts, turns around at the top and flows downward in the parallel rows at the rear of the blanket. As most of fusion neutron power is deposited directly in the breeder, which is also the coolant, an exit temperature of about 700 °C is achievable, higher than the maximum structure temperature of 600 °C. To maintain this temperature difference, flow channel inserts made of silicon carbide are placed inside the liquid metal ducts (see Fig. 7).

ARIES-ST Machine Addemby Sequence

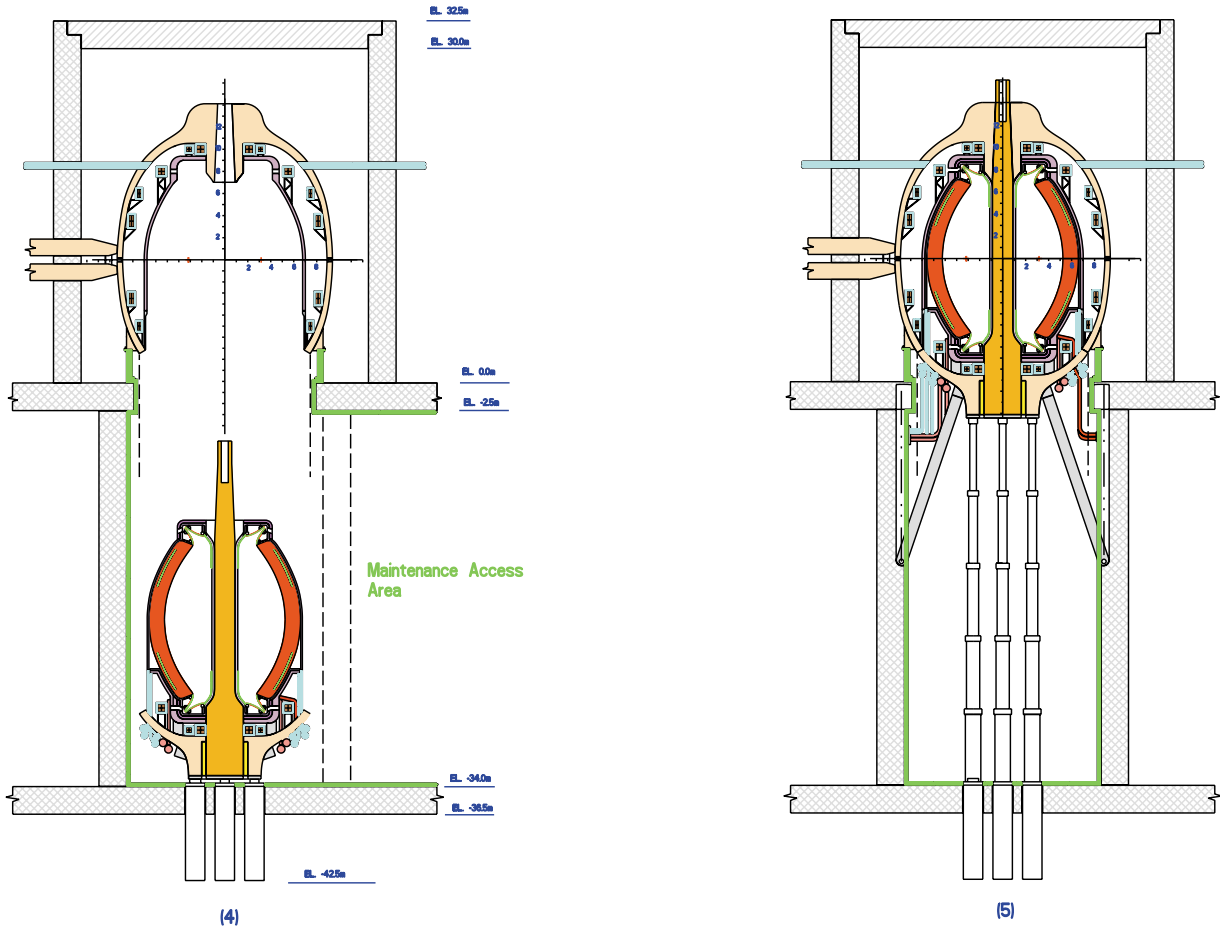


Fig. 5. Maintenance sequence of the ARIES-ST power core.

These inserts serve as electrical and thermal insulators between the flowing liquid metal and the steel structure. In order to equilibrate pressures and, hence, reduce stresses on the SiC, an opening is located along the back side and a small gap is provided between the SiC insert and the surrounding walls. A single opening to the electrically conducting structures is expected to have little or no effect on the MHD flow behavior and pressure drop.

Due to the high PbLi outlet temperature, special considerations are needed for the pipes leading to the power conversion system. A characteristic feature of PbLi is its low tritium solubility. The

resulting high tritium partial pressure can lead to intolerably high tritium permeation losses from the coolant access tubes and, therefore, generally requires additional permeation barriers. Another issue is the choice of materials. There is no available steel that will allow a temperature above 600 °C. To solve both problems, concentric access tubes are used with the liquid metal flowing in the inner tube and helium in the annular gap. In this way, helium with a temperature of 500 °C cools the liquid metal duct, especially if a thermal insulator is arranged inside the liquid metal tube. Fifteen millimeter thick SiC is proposed for this purpose. This design allows the use of steel for all

Cross Section of ARIES-ST Outboard Blanket

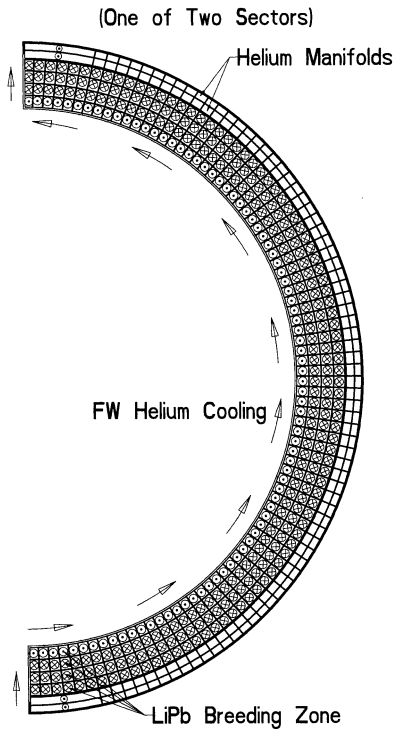


Fig. 6. Outboard blanket cross-section.

Partial Cross Section of Outboard Blanket

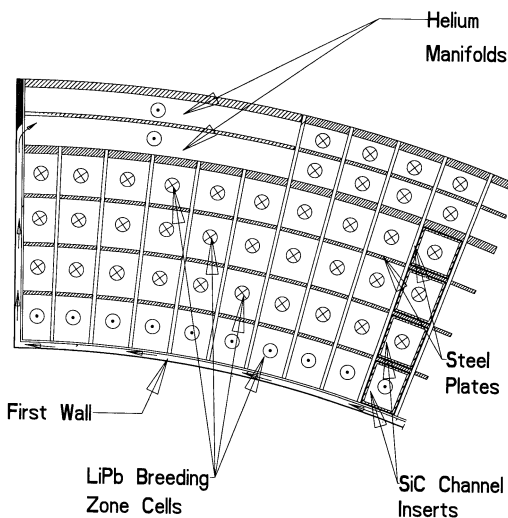


Fig. 7. Detail of blanket cross section.

access tubes. Tritium from the PbLi in this case will not enter the building atmosphere but rather the helium coolant, where it can be recovered easily.

One main argument against the use of PbLi as breeder/coolant has been the radiotoxicity of ^{210}Po . Investigations performed in Europe during the past 5 years, however, showed that this problem had been vastly overestimated [31]. More precise neutronics data as well as more adequate calculation methods indicate ^{210}Po generation is orders of magnitudes lower than previously estimated. Release experiments of ^{210}Po from PbLi proved that the release rates are not determined by the vapor pressure of Po but rather by the vapor pressure of a Po–Pb compound which is orders of magnitudes lower than that of Po itself. Both effects—the lower ^{210}Po generation and the lower release rates—led to the conclusion that ^{210}Po is no longer a significant safety issue. For a typical case, the contribution of ^{210}Po to the total dose to the public during a loss of coolant accident (LOCA) has been estimated to be less than 1%. The total release of radiotoxic materials was in any case so small that no evacuation of the public would be required.

4.2.2. Inboard first wall and shield

The centerpost shield surrounds the centerpost and protects it from neutron damage and excessive nuclear heating (see Section 3.1). It is 20 cm thick and contains five rows of elliptic coolant channels oriented in such a way as to mitigate neutron-streaming. The centerpost shield is cooled by helium and is made of ferritic steel.

The inboard first wall surrounds the inboard shield and is split into two halves, each subtending 180° of circumference. It consists of elliptic coolant channels joined together and oriented with the smaller radius facing the plasma in order to minimize the hoop stresses on the plasma-facing side. The first wall is attached to the inboard shield by joining the two semi-circular first wall assemblies around it (see Fig. 8).

Tungsten shells, cooled by helium, surround the centerpost at discrete vertical locations (see Figs. 1 and 2). They provide passive vertical stabilization of the plasma, and also act as high-heat flux

Cross Section of the Inboard First Wall and Shield

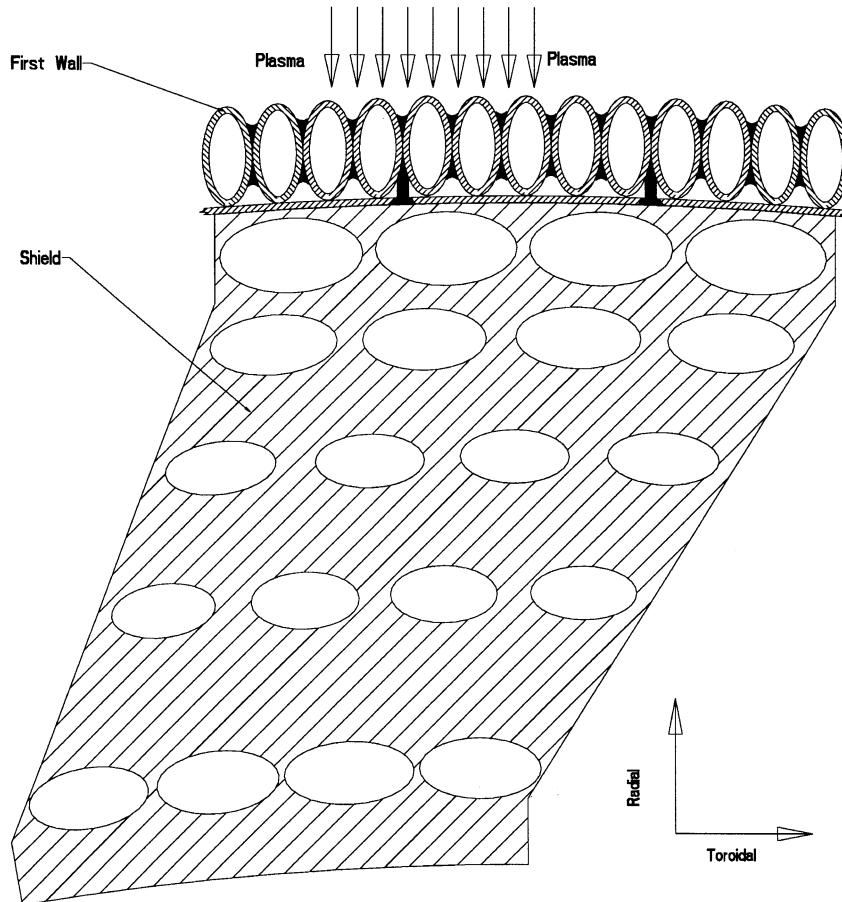


Fig. 8. Mid-plane cross section of the inboard shield.

components covering the transition region between the inboard first wall and the divertors. Tungsten is chosen as the structural material because of its high electrical and thermal conductivity, its high temperature capability, and its resistance to plasma erosion. Including the coolant channels, the total thickness of tungsten shell is 6.5 cm.

Particle fluxes to the inboard side of a ST plasma are not well known. In the ARIES-ST design, no armor or coating was applied to the ferritic steel first wall, but the design could be so modified if required. In principle, the inboard geometry could be tailored to preferentially direct particle fluxes to the tungsten stabilizing shells,

which are capable of withstanding both power and particle fluxes far better than ferritic steel. In the design of the W shells, a safety factor of 2 was used to allow for uncertainties in surface heat flux distribution along the inboard wall.

4.2.3. Divertor

Given the uncertainties in SOL physics in a ST, it is assumed that ARIES-ST can achieve an average divertor heat flux of 2 and a peak heat flux of 6 MW m^{-2} using a combination of methods (see Section 2.3). These values are used for engineering design considerations. In the reference design, tungsten was chosen for both the plasma-facing and 'heat sink' material. It

exhibits good thermophysical properties, low erosion capability, good high-temperature strength, and has an established and growing database. In this study, an attempt was made to use very simple structures and to maintain the stresses low enough such that pure tungsten could be used. This reduces the cost as it minimizes the need for metal-working. It also avoids addition of alloying elements such as rhenium (used for enhancing the ductility and fabricability of tungsten) that are activation concerns. Helium is the preferred coolant for reasons of safety, materials compatibility, and system integration (since the blanket and other in-vessel systems use He coolant). The primary drawback of He is its limited heat transfer capability.

Three design concepts were considered that can handle the particle and heat loads: a slotted duct configuration, a normal flow design, and design utilizing porous metallic filler material. The reference design concept chosen for ARIES-ST uses a porous metal filler in a cylindrical pipe geometry [10]. This concept was chosen in large part due to its simplicity, fabricability and straightforward integration with the manifolding. For the purpose of numerical estimates of porous media properties, the packed bed concept was used because correlations are more readily available.

Fig. 9 shows a tubular design in which the central half-pipes serve as inlet and outlet mani-

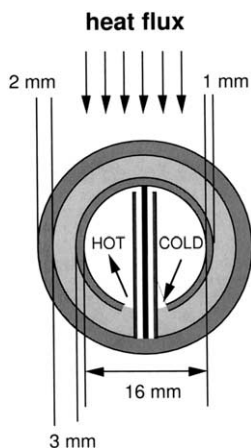


Fig. 9. Porous metal heat exchanger geometry.

folds and the outer annulus is the primary heat transfer region. Toroidal manifolds supply coolant to the pipes, which are oriented along a poloidal/radial direction. Coolant flows along the axis of the pipes until it is redirected around the circumference. The path length is very short and the effective duct cross sectional area is large, such that low coolant velocity is possible, leading to small pressure drop.

4.3. Power cycle

A distinguishing feature of the power conversion system for ARIES-ST is the use of two coolants: (1) a helium loop with an inlet temperature of 300 °C and an outlet temperature of 525 °C; (2) a liquid metal loop with an inlet temperature of 550 °C and an outlet temperature of 700 °C.

These heat sources can be used either for a Rankine cycle (steam turbine) or for a Brayton cycle (closed-cycle gas turbine). A gross thermal efficiency of at least 45% can be achieved with both power conversion systems based on the temperatures listed above. A closed cycle helium gas turbine process was chosen as the reference concept mainly because of concerns with liquid metal/water exchangers and tritium permeation losses [33].

Details of the ARIES-ST power conversion system are given in [10]. The power conversion efficiency for the gas cycle is estimated to be 45%. In order to achieve this high value, it is assumed that recuperators can be developed that achieve an efficiency of 96% while being able to operate reliably over many years of operation. Until the early 1980s, tubular recuperator designs were limited to effectiveness in the range of 82%. However, modern recuperators are available with effectiveness of 96% [34]. Some further development is required to achieve the exact set of conditions needed for ARIES-ST, but we believe the extrapolation to our reference value of 96% is reasonable.

Table 3 summarizes the power flows and peak temperatures in ARIES-ST fusion core.

Table 3
Parameters for the ARIES-ST power conversion system

Total fusion power (MW)	2859
Total thermal power (MW)	3107
First wall and divertor plate (MW)	674
Outboard blanket (MW)	1944
Inboard shield (MW)	288
Divertor shield (MW)	201
<i>Coolant load</i>	
For He	
First wall and divertor plate (MW)	674
Outboard blanket (MW)	553
Inboard and divertor shield (MW)	489
Total in He (MW)	1716
For Pb–Li (MW)	1391
He temperature (°C)	300–525
He pressure (Mpa)	12
Pb–Li temperature (°C)	550–700
He flow rate (kg s ^{−1})	1444
Pb–Li flow rate (kg s ^{−1})	47450
He temperature to turbine (°C)	300–680
Cycle efficiency (%)	45

5. Safety and licensing

Similar to previous ARIES designs, the safety design and analysis of ARIES-ST focused on achieving two key objectives: (1) generation of only low-level waste, e.g. CLASS C [35]; and (2) avoidance of the need for public evacuation following the most severe accident (i.e. the dose at site boundary per event must not exceed 1 mSv).

The rad-waste of different regions of ARIES-ST at end of service life were evaluated according to both NRC 10CFR61 [35] and Fetter [36] waste disposal concentration limits. The analysis indicates that all components would qualify for CLASS C low-level waste under both 10CFR61 and Fetter criteria. For example, Table 4 lists the waste disposal ratings (WDR) for major components of ARIES-ST fusion core. The waste disposal rating is defined as the sum over all isotopes of the ratio of the concentration of a particular isotope to the maximum allowed concentration of that isotope. If WDR is less than one, all isotope concentrations are below allowed limits and the component qualifies as low-level waste.

Regulatory limits require that the radiation dose at the 1 km site boundary be less than 1 mSv to

Table 4
WDR of ARIES-ST using 10CFR61 limits

Zone	Life (FPY)	WDR
Centerpost	2.86	0.88
Inboard shield	2.86	0.17
Inboard first wall	2.86	0.2
Outboard first wall	2.86	0.26
Outboard blanket	2.86	0.021
Outboard shield	2.86	3.7×10^{-3}
Divertor plate	2.86	0.18
Divertor shield	40	0.012

ensure that the impact of an accident to the surrounding population is sufficiently small such that no evacuation plan is necessary. Instead of a risk assessment methodology, we have considered the worst possible accident scenario, a LOCA in the fusion core. We have assumed that the fusion core maintains its geometry during the accident (e.g. there is no gross distortion of fusion core geometry). Following a LOCA, analysis indicates that the maximum temperature of centerpost is ~ 1000 °C (hottest location in fusion core) and the maximum temperature of the inboard and outboard first walls are ~ 890 °C. Under these conditions, the full mobilization of the radioactive products is impossible. The calculated temperature profiles and available oxidation-driven volatility experimental data were used to calculate doses at the site boundary under conservative release conditions. The vacuum vessel and the containment building were assumed to stay intact during the accident but can have large openings. As such, a ‘leak’ rate of 1% per day was assumed for both vacuum vessel and containment building. This is a conservative assumption as most of the released products condens on the cooler vacuum vessel and containment building surface area. Analysis indicates that the ARIES-ST design produces an effective whole body early dose of 1.8 mSv at the site boundary. Due to conservatism used in this analysis, it is expected that more thorough accident analysis for ARIES-ST would lead to lower doses and that no evacuation plan would be necessary.

6. Design evaluation and R&D needs

Overall the ARIES-ST study has shown that the ST concept leads to attractive fusion power plants. The cost of electricity from ARIES-ST is comparable to that of the advanced tokamak ARIES-RS (although, a more aggressive physics extrapolation has been used for ARIES-ST).

In the design of the ARIES-ST power core, extrapolation from existing physics data base and technologies was allowed in order to provide an attractive end-product, but minimized in order to obtain a credible design with an acceptable development program. Cases in which extrapolations beyond established technology were necessary are identified below as key R&D needs. The ordering is not intended to imply relative importance.

In the physics area, most of the ARIES-ST research has borrowed from tokamak data base extrapolated to low aspect ratios. Experiments with Proof-of-Principle and Performance Extension ST devices are needed to establish ST-specific physics data base. Some of the more critical physics R&D needs identified in this study are highlighted below.

- 1) Demonstration of a stable and controllable operating point. A ST power plant plasma should operate at high β with a large bootstrap current fraction ($\geq 95\%$) that is well-aligned with the equilibrium current–density profile. While ST concept will benefit from considerable research in advanced-tokamak modes worldwide, there are sufficiently subtle differences that these advanced modes should be demonstrated in a ST geometry.
- 2) Ignition physics. Physics of burning plasmas and α -particle dynamics remains unresolved issues for fusion research. Obtaining advanced tokamak modes in the presence of dominant α -particle heating is a critical issue that can only be addressed in a long-pulse, burning-plasma experiment.
- 3) Divertor and edge physics. Divertors remain one of the most difficult physics and engineering challenges. Even in modest aspect ratio tokamaks, the mechanism (or mechanisms) responsible for determining SOL properties in

double-null divertors are at present not well understood. ‘Leaning’ the ST plasma on the centerpost to reduce inboard divertor heat flux as well as establishment of radiative SOL and divertor plasma should be demonstrated in a high-power ST device.

- 4) Controlled start-up, shutdown, and partial power operation. Non-inductive start-up has been assumed for ARIES-ST with little help from the PF systems as there is no room for an OH solenoid in a ST power plant. While analysis indicates that non-inductive start-up, plasma shutdown, and partial power operation can be achieved without major difficulties, these techniques must be experimentally demonstrated.

As is often the case, uncertainties in materials performance, especially when subjected to the complete fusion environment, give rise to several key issues. This arises due to the use of novel or untested materials, or to the particular environmental conditions to which they are subjected.

- 1) Radiation damage effects and lifetime of advanced ferritic steel. Ferritic/Martensitic steel has been a top candidate alloy in fusion programs around the world for years. Its database is large and growing. Nevertheless, radiation damage and lifetime remain important issues. Low-activation ferritic steels are usually limited in operating temperature to less than 550 °C. In order to enable efficient power conversion, a higher temperature limit is needed. In combination with the novel use of heat from PbLi, a maximum steel temperature of 600 °C would allow a Brayton cycle with $> 45\%$ conversion efficiency.
- 2) Fabricability and radiation damage in tungsten. The excellent thermophysical and plasma-interactive properties of tungsten helps to provide a credible solution to divertor design. However, pure tungsten suffers from low fracture toughness and uncertain irradiation damage characteristics.
- 3) Electrical resistance of SiC. Measured resistivities for SiC/SiC composite often is not available. MHD analysis of the French TAURO

concept used a value of $0.2 \Omega \text{ cm}$, which had been measured for solid SiC at 1000°C [32]. Their finding was that the MHD pressure drop in a duct with such a resistivity is tolerable. For comparison, the resistivity of ferritic steel at 500°C is 95×10^{-6} and for PbLi $130 \times 10^{-6} \Omega \text{ cm}$. Pure alumina would have a resistivity of $10^6 \Omega \text{ cm}$ at 1000°C .

- 4) Ferromagnetism. Ferritic steel is ferromagnetic. This raises new classes of issues that have not been adequately addressed in R&D programs. The issues generally fall into three categories: (a) plasma interactions, (b) engineering interactions, and (c) diagnostic interactions. Plasma interactions include concerns over field ripple, impacts on start-up, and general position and shape control. Engineering interactions include forces during normal operation, disruption forces, and MHD effects on liquid metal flows. Diagnostic interactions include interference with magnetic field measurements.

Compatibility between materials at high temperature raises several concerns both inside the blanket and in the heat transport loop. The primary concern results from the use of PbLi, for which very limited data are available.

- 1) Compatibility of PbLi with ferritic steel. All of the steel structure in the blanket segment and the steel wall of the liquid metal coolant tubes are thermally insulated from the flowing PbLi by a SiC liner. However, there is a gap of stagnant PbLi between the steel wall and the flowing PbLi for pressure equalization. At this PbLi/steel interface the temperature will be slightly higher than the local helium temperature (525°C). For self-cooled PbLi blankets with a flow velocity of up to 2 m s^{-1} and ferritic steel as structural material, a maximum allowable interface temperature of 470°C has been used previously, based on a corrosion rate of less than $20 \mu\text{m}$ per year. It is assumed that a limit about 50°C higher can be allowed in the stagnant gap. Further testing is needed to demonstrate acceptable mass transport and corrosion levels.

- 2) Compatibility of PbLi with SiC. The maximum interface temperature between the two materials is below 700°C . Tests at ISPRA have shown that the two materials are compatible at 800°C under stagnant conditions [32]. Tests in flowing PbLi are under way to better establish operating temperature limits.
- 3) High temperature heat exchanger compatibility with PbLi. The main penalty for raising PbLi to 700°C is the need to develop a high-temperature PbLi heat exchanger. This was considered a reasonable trade-off, although the technology needs to be demonstrated. The top candidate heat exchanger material is SiC. Due to the absence of radiation damage concerns, relatively higher thermal conductivity composites should be possible.

Lastly, as ARIES-ST is the first large-scale ST study, ST concept can benefit from additional design studies. In particular, the centerpost is the most critical system in a ST. Further investigation of centerpost designs is warranted in order to better optimize this key component.

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