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ARIES-ST plasma-facing component design and analysis

M.S. Tillack [a,*], X.R. Wang [a], J. Pulsifer [a], I. Sviatoslavsky [b], the ARIES Team

[a] *Fusion Energy Research Program, University of California, San Diego, La Jolla, CA 92093-0417, USA*
[b] *Fusion Technology Institute, University of Wisconsin-Madison, Madison, WI 53706, USA*

Abstract

ARIES-ST is a 1000 MW fusion power plant conceptual design based on a low aspect ratio 'spherical torus' (ST) plasma. The plasma-facing components include the inboard and outboard first wall, divertor plates, and plasma stability plates. Several unique aspects of ST plasma influence the engineering design. The limited inboard space precludes a full inboard divertor slot, such that substantial power and particle flows are expected to impact the inboard first wall. Strong requirements on plasma vertical stability require close-in conductors. The use of He-cooled tungsten for the stability plates allows them to act as plasma-interactive components operating at high temperature and moderately high (1–2 MW/m^2) heat flux. High plasma core radiation fraction and a natural tendency for low inboard transport losses help to alleviate these inboard problems, such that the peak power loads are not expected to exceed the capability of He-cooled tungsten and steel structures. The main features of the plasma-facing components are summarized here together with the analysis of their thermal hydraulic and thermomechanical performance.

Keywords: Plasma-facing components; Spherical plasma; Power

1. Introduction

The ARIES-ST plasma-facing components must survive the local environment, including high heat flux, particle flux, and electromagnetic forces. At the same time, they must meet the requirements of an attractive energy source, including low radioactivity, recovery of high-grade heat for energy conversion, maintainability and acceptable cost and reliability.

Tungsten was used in the divertor and vertical stability plates as both the plasma-facing and 'heat sink' material. Its advantages include low erosion rate, exceptional thermophysical and mechanical properties, acceptable safety and environmental characteristics, and an established and growing database. Its high electrical conductivity made it the logical choice for close-in (plasma-facing) vertical stability plates. One of the concerns with the use of pure tungsten is the difficulty in machining and welding. In addition, annealing (which might result from post-heat-treatment bonding) causes substantial loss in high-temperature yield strength. Therefore, simple design solu-

* Corresponding author. Tel.: +1-858-5347897; fax: +1-858-5347716.
E-mail address: tillack@fusion.ucsd.edu (M.S. Tillack).

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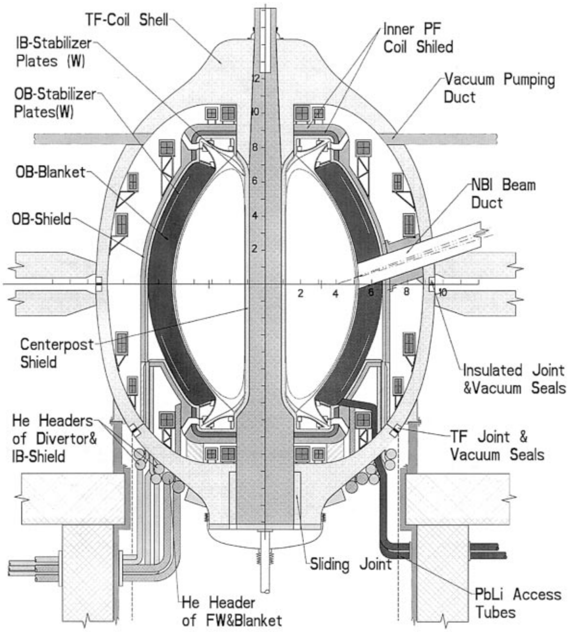

Fig. 1. Elevation view of the ARIES-ST power core.

tions were sought to minimize the need for metalworking. Rhenium addition is known to enhance the ductility and fabricability of tungsten. In this study, an attempt was made to use very simple structures and to maintain the stresses low enough such that pure tungsten can be used, thereby reducing the cost.

Helium is the preferred coolant for reasons of safety, material compatibility, and system integra-

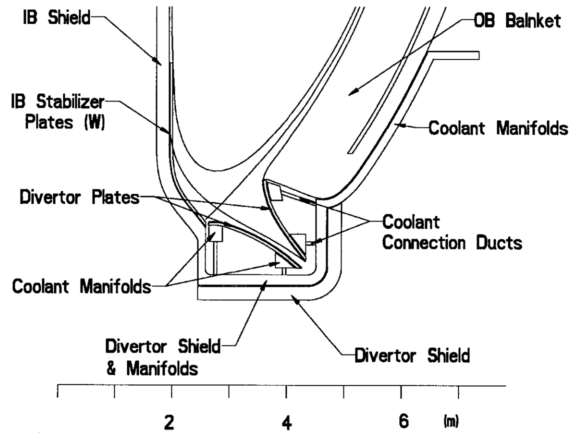

Fig. 3. Elevation view of the divertor region.

tion (since the blanket and other in-vessel systems use He). All the power deposited in the plasma-facing components contributes to electricity generation in the He Brayton cycle; therefore, matching of coolant temperatures and pressure to the power cycle was an important consideration.

The primary drawback of He is its limited heat transfer capability. Therefore, heat transfer enhancement techniques were explored to ensure acceptable temperatures in the structures, particularly the divertor. Some of the heat transfer geometries examined include slotted ducts with embedded W rods [1], porous metal beds [2] and foams [3], and normal flow options [4,5]. Significant design margins suggest that He cooling of W

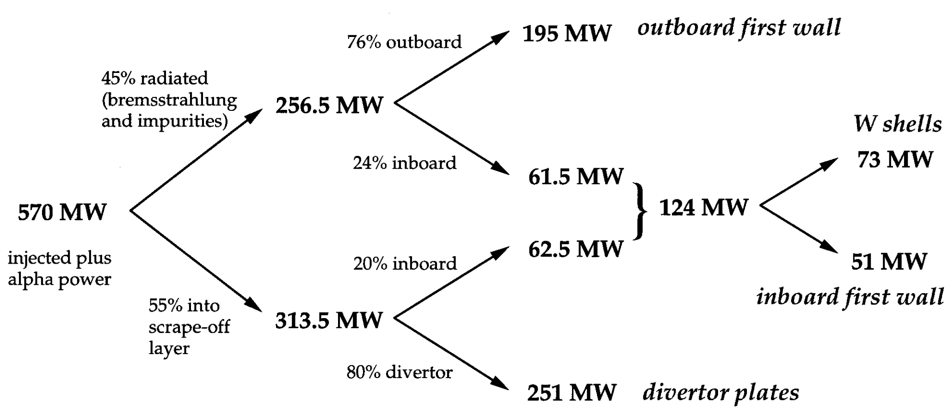

Fig. 2. Transport power flows.

is a feasible design option for ARIES-ST. Space limitations on the inboard side of the device necessitate once-through cooling of the inboard first wall and vertical stability plates. The thermal–hydraulic design is challenging here due to the long flow paths (~ 18 m), moderately high surface heat flux (0.76 MW/m^2) and strong nuclear heating.

2. Configuration and power flows

A cross section of ARIES-ST is shown in Fig. 1. The plasma configuration is similar to a 'normal' tokamak with a double null, except that the plasma is highly elongated ($\kappa = 3.75$ at the x-point) to increase the maximum stable plasma beta (~ 55%). The divertor and first wall are integrated parts of the power core. They are assembled ex-vessel and transported into the vacuum vessel with the remainder of the power core [6]. Single-piece construction is enabled in ARIES-ST by the elongated geometry of the plasma, the ability to provide electrical joints in the normal-conducting TF system, and the relatively small drained weight of a self-cooled blanket (~ 768 tons). It assumes that the lifetime and reliability of the plasma-facing components will be high enough to match the scheduled maintenance interval of up to 2 full power years. This goal is an important criterion for the plasma-facing component R&D program.

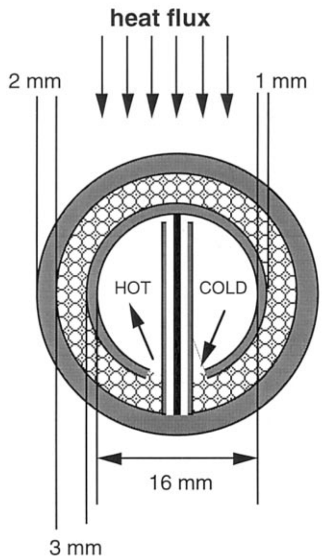

Fig. 4. Porous metal heat exchanger geometry.

Table 1
Divertor coolant tube parameters

Pipe length (m)	1.5
Pebble size (mm)	0.5
Power removed per pipe (kW)	84
Superficial velocity (m/s)	1.88
h_p (W/m^2 K)	3159
h_{eff} (W/m^2 K)	44 000
dp/dx (MPa/m)	1.5
Pumping power (W)	846

Fig. 5. Thermal plus pressure (12 MPa) stresses in the divertor outer pipe.

Due to expectations of low transport power on the inboard side [7], and also due to the lack of space to accommodate a full inboard divertor, inboard transport power flows are distributed along the entire first wall rather than being concentrated in a closed inboard divertor slot. Relatively large core radiation fraction (45%) is achieved using injected impurities, such that the total power into the outboard divertor slot remains modest. In addition, the location of the divertor far from the plasma center results in low neutron flux and relatively large available space. If a radiative divertor can be established with a deep slot (of the order of 1–2-m long), then the heat fluxes can be maintained quite low on the surrounding structures. Due to the large radiation fraction in the core, only 250 MW of transport power reach the divertors (125 MW each). The average surface heat flux distributed over 128 m^2 of area is only about 2 MW/m^2. The key concerns are the heat flux peaking factor and plasma erosion. Extensive edge physics analysis for an ST power plant is absent, approximate values of peaking factor and particle fluxes have been extrapolated from ARIES-RS [8].

The transport power flows (which result in surface heat fluxes) are summarized in Fig. 2. Core transport power is partitioned into radiation and particle flows. Since the outboard first wall is positioned well outside the scrape-off layer (SOL), the particle flows are assumed to deposit their energy on the divertor and on the inboard first wall and stability plates. Twenty percent of this power is assumed to be deposited on the inboard side of the machine; a conservatively high estimate is used, because handling power on the inboard first wall is more challenging than handling it in the divertor. This results in 125 MW on the inboard surfaces and 251 MW on the divertor plates.

All power core coolants enter and exit through the 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

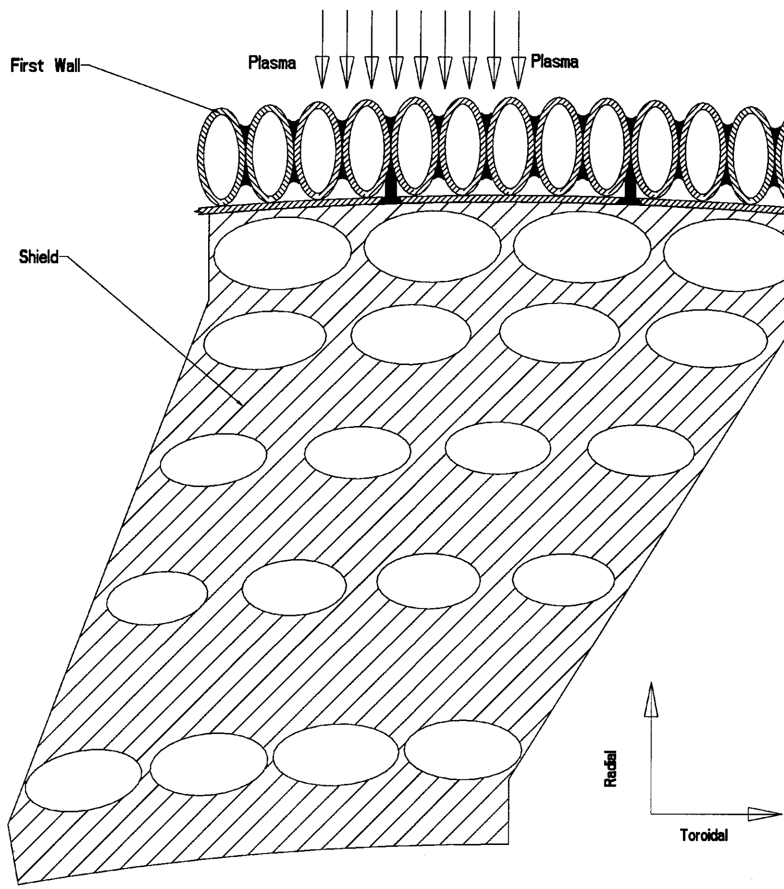

Fig. 6. A segment (1/39th) of inboard shield with superimposed first wall.

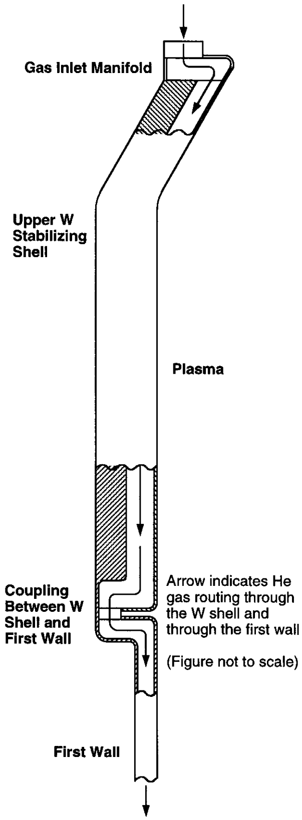

Fig. 7. Tungsten stabilizing shells and their interface with the first wall.

not only to route the coolant, but also to provide structural integration of the power core.

For compatibility with the Brayton power cycle, high coolant outlet temperature is extremely important. In ARIES-ST, PbLi in the deep blanket is used to supply the highest temperature heat to the power cycle. The inlet and outlet temperatures of helium in the first wall and divertors are chosen to match the 'low end' of the cycle, from 300 to 525°C. The relatively low bulk coolant temperature enables higher heat fluxes without sacrificing the net cycle efficiency, which is provided by the high-temperature operation of the deep blanket.

3. Divertor design and analysis

The divertor region is shown in Fig. 3. On the outboard side, a deep slot is provided to capture 80% of the transport power from the scrape-off layer. On the inboard side, the remaining transport power (63 MW) is distributed along the first wall and tungsten stability plates by tailoring the wall shape with respect to the outermost plasma flux surface.

Several geometries were examined for the high-heat-flux channel coolant configuration. This includes slotted ducts with embedded W rods, porous metal beds and foams, and normal flow options such as impingement jets. All these concepts achieve superior thermal–hydraulic performance with helium by using short coolant path length combined with extended surfaces in high-conductivity heat sinks. The ARIES-ST reference design concept uses a porous metal filler in a cylindrical pipe geometry. This concept was chosen in large part due to simplicity, fabricability and straightforward integration with the manifolding. Both sintered bed and foam variants were considered. For the purpose of numerical estimates, the packed bed concept was used because design correlations are more readily available.

Table 2
Thermal hydraulic parameters of the inboard plasma facing components

	Upper plate	First wall	Lower W plates
Inlet temperature (°C)	300	371.7	438.4
Exit temperature (°C)	371.7	438.4	510
Mass flow rate (kg/s)	193.19	193.19	193.19
Heat transfer coefficient (W/cm^2 K)	1.37	1.43	1.18
Channel diameter (cm)	2.84	1.81	3.14
Length of channel (m)	2.90	9.0	2.90
Pumping power (MW)	2.5	13.00	2.37

Fig. 4 shows a tubular design in which the central half-pipes serve as inlet and outlet manifolds 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.

The mass flow rate per channel needed to match the coolant temperature conditions of 300–525°C while removing an average heat flux of 2 MW/m^2 is 72 g/s, leading to a superficial velocity of 1.88 m/s. The effective heat transfer coefficient, h_{eff}, is given by [2]:

$$h_{eff} = \alpha h_p + \frac{1}{R_0 + [1/(\sqrt{h_p k_p S_p}\tanh(\sqrt{h_p S_p/k_p t}))]}$$

where S_p is the specific surface area, h_p the local particle-to-fluid heat transfer coefficient, R_0 the porous medium/wall interface resistance, k_p the porous medium thermal conductivity and t the porous medium thickness.

Using the parameters listed in Table 1 and assuming low contact resistance (achieved, for example, by sintering), the effective heat transfer coefficient is 44 000 W/m^2 K. This leads to a film drop of only 114°C at 5 MW/m^2 local heat flux. The conduction drop through the 2-mm wall is also $\sim$100°C. Large margins can be used to increase the wall thickness for protection against plasma erosion.

The corresponding pressure drop and pumping power are found using the Ergun formula. The regime is intermediate between laminar and turbulent flow. This design achieves a very modest ratio of pumping power to thermal power removed, only 1%.

Fig. 5 shows the stresses obtained in the outer pipe. In this design, a 180° U-bend is used to allow relatively unconstrained axial thermal expansion. As a result, peak stresses occur at the location of attachment, rather than at the location of peak temperature. These results were obtained by constraining the axial location of the attachment but allowing rotation. Joining to the toroidal manifolding is done using Ti–25Cr–3Be braze material. A design with increased flexibility at this joint is very desirable.

4. Inboard first wall and stability plates

The ferritic steel first wall (see Fig. 6) consists of elliptic coolant channels (1.9 × 3.0 cm) joined together and oriented with the small dimension facing the plasma, extending from the lower limits of the upper W plates to the upper limits of the lower W plates. The elliptic shape minimizes the hoop stresses in the surface facing the plasma. For the W stability plates, an equivalent conductor thickness of 5 cm is needed for plasma vertical stability. The coolant channels in this case are 2.84 cm in diameter spaced at 3.6 cm between centerlines in the cylindrical region of the plate, increasing to 5.7 cm at the top of the conical region. The W plates are joined to the first wall at the top and bottom, making it possible to cool them in series with the first wall. Manifolds are used to transfer the He across the interface between the first wall and stability plates (see Fig. 7).

The surface heat flux is spread evenly among the W plates and first wall; however, a safety factor of 2 is used on the W plates to allow peaking uncertainties. Particle fluxes to the inboard side of ST plasma are not well known. In the ARIES-ST design, no armor or coating was applied to the ferritic steel first wall. In principle, the inboard geometry could be tailored to preferentially direct particle fluxes to the tungsten stabilizing plates, which are capable of withstanding both power and particle fluxes far better than ferritic steel.

The results of thermal–hydraulic analysis are summarized in Table 2. The surface temperature of the W plates varies from 668 to 887°C, which is well within their operating limits. The temperature of the first wall peaks at 564°C just below the midplane. With a velocity of 150 m/s, the total pumping power is 17.87 MW, which is 8.5% of the thermal power recovered.

5. Summary

A high performance divertor was shown to be compatible with the ARIES-ST spherical torus power plant. The design uses materials that meet current safety and waste disposal requirements for a fusion power plant, while operating at a temperature that allows efficient power generation. The small major radius of the ST power plant is offset by a large vertical build, which provides a large area for a radiative outboard divertor. The modest heat fluxes combined with excellent thermophysical properties of tungsten provide a significant margin on peak temperatures and stresses. The key issue with the use of tungsten is to develop a fabricable and cost-effective alloy that maintains good properties under irradiation. Electromagnetic and ferromagnetic behavior of the divertor remains an important development issue.

Plasma power and particle flows in a ST reactor are uncertain, but preliminary indications are that the inboard power flows will be modest. This enables the elimination of the inboard slot, which is replaced with a tungsten stabilizing shell that is needed for plasma stabilization in any case. The final design adopted ferritic steel facing the plasma along the inboard first wall. The high heat flux and difficult cooling problem along the inboard first wall makes armor difficult to implement. Overall, the most critical issue for plasma-facing components is to obtain accurate characterization of the edge plasma conditions, including power and particle fluxes.

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