The ARIES-CS – A Compact Stellarator Power Plant

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Abstract: Stellarators have many desirable features as fusion power plants. A detailed and integrated study of compact stellarator configurations as power plants, ARIES-CS, was initiated recently. Configurations with A $\leq$ 6 and excellent quasi-axisymmetry (QA) in both 2 and 3 field period versions were developed while reducing α-particle losses to < 10%. Stability to linear ideal MHD modes was attained but at the expense of reduced QA (and increased α-particle losses) and increased complexity of the plasma shape. Recent experimental results indicate, however, that linear MHD stability limits may not be applicable to stellarators. By utilizing a highly efficient shield-only region in strategic areas, the minimum stand-off was reduced by ~30%. This allows a comparable reduction in the machine size. The device configuration, assembly, and maintenance procedures appear to impose severe constraints: Three distinct approaches were developed, each applicable to a certain blanket concept and/or stellarator configuration. Modular coils are designed to examine the geometric complexity and to understand the constraints imposed by the maximum allowable field, desirable coil-plasma separation, coil-coil spacing, and other coil parameters. A cost-optimization system code has also been developed and will be utilized to assess the trade-off among physics and engineering constraints in a self-consistent manner in the final phase of the ARIES-CS study.

1. Introduction

Stellarators have many attractive features as a power plant because there is no large driven external current – they are inherently steady state (low recirculating power), are stable against external kink and axisymmetric modes, and are resilient to plasma disruptions. A detailed and integrated study of compact stellarator power plants, ARIES-CS, was initiated recently to explore attractive compact stellarator configurations and to define key R&D areas.

Earlier stellarator power plant studies led to devices with large sizes compared to recent tokamak power plant studies and a factor of 2 to 4 reduction in linear dimension is needed to bring the size and cost of these earlier stellarators in line with more recent tokamak studies. This has been a major thrust of the ARIES-CS study. The large size of stellarators is generally dictated by the constraints imposed by the minimum distance between the plasma and the coils. In a stellarator, the external coils generate a multipolar field and the high order harmonics of the magnetic field decay rapidly. Thus, the coil currents and irregular shape increase drastically as the distance between coils and plasma is increased. In fact, it is usually not possible to place the coils at an arbitrary distance from the plasma. On the other hand, in a burning plasma device and/or a power plant, the space between plasma and the coil is occupied by the first wall, breeding blanket, and shield which protect the superconducting coils – typically these components require a distance of 1.5-2 m between the plasma and the middle of coil winding pack. As such, for a given stellarator plasma/coil configuration, the average minor radius of the plasma (and the coil size) should be increased until sufficient space is available for the above components, *i.e.*, the radial build of the device dictates a minimum average plasma minor radius. As the earlier stellarator power plant studies all had a relatively large plasma aspect ratio, the radial built constraint had led to large devices. To reduce the device size, one could either 1) develop configurations with reduced plasma (or coil) aspect ratio while maintaining the "good" stellarator properties (Sec. 2), and/or 2) reduce the required minimum coil-plasma distance through engineering optimization (Sec. 3). Both avenues were explored in the ARIES-CS study.

This paper is a summary of ARIES-CS study to date. More detailed information can be found in Ref. 1-3. An overall summary of our research is given in Sec. 4.

2. Plasma Configurations

Stellarator configurations with an underlying quasi-symmetric magnetic field structure have attracted intense interests in recent years. In particular, quasi-axisymmetric (QA) configurations leads to particle orbits similar to those in a tokamak and have the potential to combine the desirable features of tokamaks (good confinement and moderate aspect ratio) with those of large-aspect ratio stellarators (steady-state operation, stability against external kinks and axisymmetric modes, and resilience to disruptions). The ability to operate at a relatively low plasma aspect ratio ($\sim$ 4-5) is of particular importance for burning plasma devices and power plant. A proof-of-principle device, NCSX, based on this configuration is under construction [4].

Physics optimization of the stellarator configuration involves a large number of variables. Typical criteria used in optimizing the configuration include the basic plasma properties (e.g., magnetic shear, magnetic well depth), as well as measures of MHD stability and transport. For our study, confinement of α particles is the most important criterion added to previous suite of criteria. An important factor in our optimization process is the recent experimental results from W7-AS and LHD stellarators. In both cases, the achieved experimental values were higher than those predicted from linear stability theory (e.g., 3.2% achieved β in W7AS versus 2% theoretical prediction). These results led us to explore new configurations with more emphasis on the quality of flux surfaces and transport, particularly the confinement of α particles, than strict adherence to linear MHD stability requirements. As will be discussed later, confinement of fast α particles is of great importance in a burning plasma and/or fusion power plant.

Three distinct classes of QA configuration have been considered for the ARIES-CS. First is NCSX-class configuration that maintains the basic characteristics of NCSX plasma and coils: It provides a good "balance" between quasi-axisymmetry and MHD-stability considerations: it has shown to have high β limits against linear MHD modes, and particular coils have been designed which recover all of the desirable plasma properties. For NCSX-class, we have developed new $A = 4.5$ configurations with coils sufficiently removed from the plasma. Three variants of this configuration have been developed to explore trade-off between confinement of fast α particles and linear MHD stability. Progress has been made to reduce loss of α particles to < 5%. It appears that the introducing a bias in the principle mirror term in the magnetic spectrum, plays an important but not yet fully understood role in helping reduce α particle losses.

The second class of configurations is MHH2 which aims that developing a very low aspect ratio geometry ($\sim$ 2.6) with relatively simpler coils to explore how compact a compact stellarator power plant can be. This configuration was reproduced in a 16-coil design with the total iota profile (sum of external and bootstrap) being very close to linear profile. The resulting configuration has excellent QA, low effective ripple (<0.8%) and α-particles losses are < 5%. The third class, SNS, is aimed at developing a configuration with excellent flux surface quantity and nearly flat rotational transform to explore how good and robust flux surfaces one can design. In both cases, strict adherence to linear MHD stability is deemphasized. Details of our physics optimization results are given in Ref. [1].

We have chosen a three-field period NCSX-Class plasma configuration as the base line for ARIES-CS detailed design. Key features of ARIES-CS design is given in Figs. 1 and 2 and Table 1.
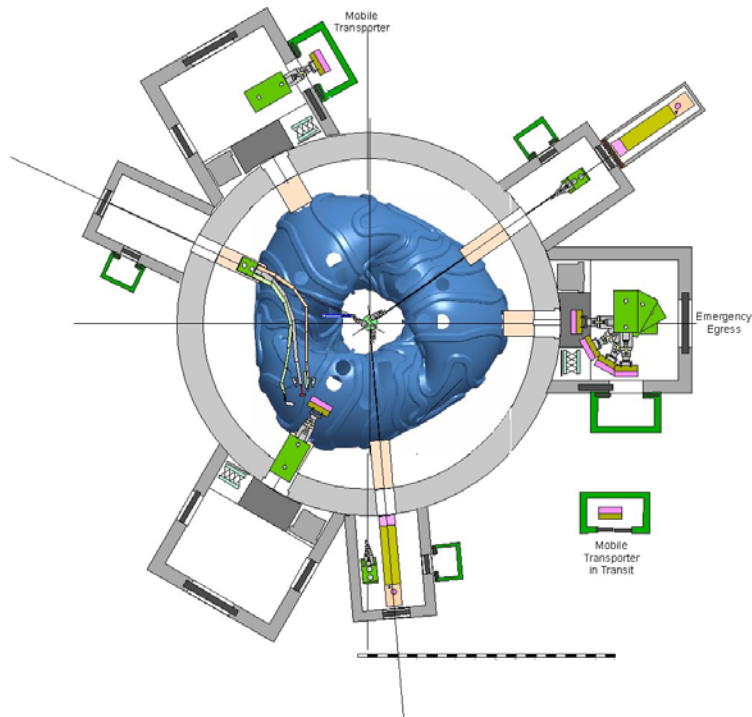

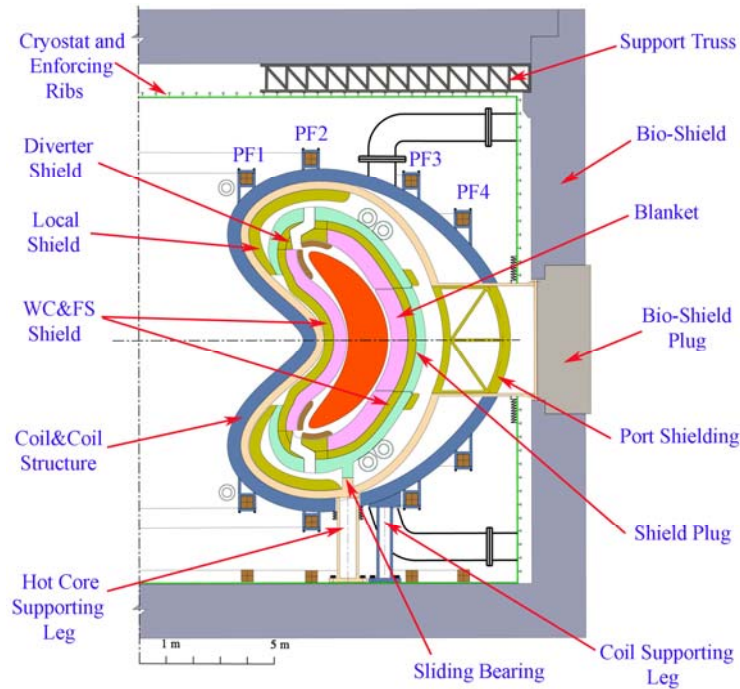*FIG. 1: A top of view of ARIES-CS including the maintenance equipment.*

FIG. 2: Cross Section of the ARIES-CS.

TABLE 1: ARIES-CS POWER PLANT PARAMETERS

Average major radius	7.75 m
Aspect ratio	4.5
Average field on axis	5.7 T
Maximum field at the coil	15 T
Number of coils	18
Average plasma β	5%
Average plasma density	$3.6 \times 10^{20}/m^3$
Average plasma temperature	5.7 keV
Central Plasma Temperature	8.6 keV
α−Particle Loss fraction	5%
Fusion Power	2.4 GW
Net electric output	1.0 GW
Average Neutron Wall Load	2.6 MW/m^2
Fractional Radiated Power	75%
Average Heat Flux on First Wall	0.58 MW/m^2

3. Engineering Options

3.1. Superconducting Magnets

The candidate superconductor materials for operation at high field (up to 16 T) are variants of Nb_3Sn which are glassy materials and their current carrying capability is drastically reduced with induced strain. For these types of superconductors, the superconductor material is wound around a large spool (with roughly the same radius of curvature as the coil itself) and is heat treated (700° C for a few hundred hours) to attain superconducting properties. The resultant cable is then gently unwound from the spool, insulated, and wound onto the coil. The reason for this two-step process is that organic insulators cannot survive the heat treatment process and need to be added afterwards. Because of the irregular shape of the coils, this method cannot be used. In order to keep the strain down, the superconductor material should be wound as a complete cable (including the insulator) on the coil and then heat treated. This method requires development of inorganic insulators that can withstand the heat treatment process. Recently, two groups (one in the US, the other one in Europe [5]) have developed glass tape that can withstand the process in small scale. Extrapolation of this insulator to large coils would allow stellarator magnets with B_{max} up to 16 T. The strain induced during the winding process is not an issue for ductile NbTi superconductors. However, the maximum field, B_{max}, is limited to ~7-8 T for 4 K operation (up to 9 T can be achieved by operation at 2 K, although there are issues with temperature margin).

In a compact stellarator, the absence of disruptions reduces the structural demand on the coil structure. The support structure must accommodate four kinds of static forces acting on the coils: radial forces in the coil plane, large centering forces pulling each coil towards the center of the torus, "out-of plane" forces acting between neighboring coils inside a field

period; and the weight of the cold coil system. Because of the complex shape of the windings, it appears that the best method to support the structural loads is a strong supporting toroidal tube (see Fig. 1) as opposed to providing support element between individual coils. As is shown in Figs. 1 and 2 for a three-field period configuration, the cryogenic modular coils are located outside the external vacuum vessel. These coils are wound into grooves at the inside of a strong supporting toroidal tube that supports the "out-of-plane" or inter-coil forces. This toroidal tube is separable at field periods as there is no net force (and no force transfer) between each field period. The radial ("in-plane") forces on the modular coil are supported locally by a strong-back layer. The centering forces can be supported by a bucking cylinder, operated at cryogenic temperature. The weight of the cold supporting tube has to be transferred to a foundation by ~3 long legs for each field-period. The legs have to be long enough to keep the heat ingress into the cold system within a tolerable limit. Deflection in the coils due to magnetic forces as well as changes in their shape and size when the coils are cooled to cryogenic temperature can be large. As the stellarator configuration critically depends on the location of the coils, these deflections should be accounted for in the design process such that coil would be at the desired position once the superconducting coils are cooled and energized. Field corrections coils may be necessary if the necessary tolerances cannot be achieved. Experience with W7-X would be critical in resolving this issue.

3.2. Configuration, Assembly, and Maintenance

Port-based maintenance involves the replacement of blanket modules using an articulated boom through a number of designated ports (see Figs. 1 & 2) -- one maintenance port per field period is used. In this arrangement the modular coils are kept at cryogenic temperatures during maintenance. A key feature of the layout is the desire to separate the hot core (including the shield and manifold, which are lifetime components) from the cooler vacuum vessel to minimize thermal stresses. In the absence of disruptions, the hot core does not need to be strongly attached to the vacuum vessel, being instead vertically self-supported. In this case, disassembling and re-welding of the vacuum vessel are not required for blanket maintenance. The maintenance ports are arranged between adjacent coils at locations with large port space and large plasma cross section. Transfer casks can be attached to the outside flange of the ports (See Fig. 1), and a system of double doors can be employed to avoid any spread of radioactivity (dust, tritium) into the containment building. The load capacity and required reach of the boom limit the weight (~5,000 kg) and size of the blanket modules. The drawbacks of this option are the large number of modules and large number of connects/disconnects and complex manifolds and joints. In addition, the coolant manifolds that are required to feed each blanket module increase the plasma-coil distance.

3.3. Blanket & Radial Build

Earlier stellarator power plant studies had assumed that the radial build of the fusion core is uniform around the plasma. This is not an optimum approach as the external coils are close to the plasma only in certain locations (~ 5% of the first-wall surface area) for the NCSX-like configurations. A novel approach was developed in ARIES-CS for the NCSC-like configurations in which a highly efficient tungsten-carbide-based (WC) shield module was used at the critical areas of minimum stand-off – each system has three radial builds: a shield-only region for locations where coils have to be close to the plasma, a nominal blanket and shield for most locations and a rather small transition region between the two. This approach reduces the minimum stand-off distance by ~30% but requires careful design of the

"nominal" blanket (increasing the tritium breeding ratio) to ensure overall tritium self-sufficiency.

The dual coolant concept is chosen as the reference blanket option. This concept utilizes He to cool the ferritic steel structure (including the first wall) and slowly flowing Pb-17Li as a self-cooled breeder in the inner channels [2]. Note that the ferritic steel in the blanket operates at temperature above its Curie temperature and, as such, is not expected to modify the magnetic field configuration. The configuration of this concept allows for the Pb-17Li coolant to exit the blanket at a higher temperature than the maximum allowable wall temperature, thereby increasing the power cycle efficiency. The Pb-17Li channels are lined with a SiC insulating layer (with no structural function as the thin Pb-17Li layer and the bulk Pb-17Li are pressure-balanced through a thin slot in the SiC), as illustrated in Fig. 14. This layer plays a key thermal insulation function to allow high temperature (~700°C) Pb-17Li in the channel while the steel temperature is maintained below its limit. This dual coolant concept was originally developed as part of the ARIES-ST study [6] and then at FZK in Germany [7] and it is considered as a test module candidate for ITER. A modular concept was adapted for the compact stellarator geometry with a particular focus on developing a more efficient coolant routing configuration and on optimizing the design performance of the blanket coupled to a Brayton power cycle [2].

3.4. Divertor

A suite of large analysis codes is utilized to carry out the divertor physics studies for the complex 3D magnetic geometry of a compact stellarator. Magnetic field lines were followed from the LCMS to the target plates and the first wall and the footprints of the field lines on the divertor plates as well as heat fluxes were calculated. A simple methodology is used to link the field line footprints to the heat load distribution on the target by associating each field line with a constant power value. Because of the 3D nature of the magnetic topology, determining the location and shaping of the divertor plates require a considerable number of iterations. The best location for the divertor plate is near the "tips" of the bean-shaped cross section as the flux expansion is maximum in those locations (See Fig. 2).

A range of different He-cooled divertor configurations have been considered in the past, including a tungsten plate design. More recently, a finger configuration utilizing tungsten caps has been evolved with the aim of minimizing the use of tungsten as structural material and of accommodating higher heat fluxes through the use of smaller units [8]. The ARIES-CS divertor builds on the W cap design but extend it to a mid-size unit. The proposed configuration is modular consisting of an assembly of W-alloy T-shaped tube, ~15 mm in diameter and ~100 mm long, with a He jet-cooling scheme and can accommodate up to 10 MW/m^2 of heat flux [2]. For a stellarator configuration, the absence of disruption relaxes the demand on the armor as no melting is anticipated and the required thickness of the W armor bonded to the T-tube would depend mostly on the sputtering rate. The thermo-mechanical analysis results showed that the total stress (<370 MPa) and operating temperatures (600-1300°C) were within the anticipated limits for W alloy [8].

3.5. α-Particle Loss

The impact of the loss of fast α particles on the power balance is usually small, but, the plasma facing components must accommodate the resulting heat load (which may have a high peaking factor) and provide the required lifetime. The high heat flux accommodation

could be provided by designing special modules based on the divertor geometry as fast-α dumps. The armor material lifetime, however, is a concern. For high-energy α particles, sputtering is not a major issue and armor lifetime would be governed by other mechanism, such as blistering, resulting from accumulation of He atoms in the armor. For a W armor, the implantation depth for He at ~1 MeV is ~1.5 μm. The implanted He would diffuse fairly rapidly through interstitials. However, the presence and/or formation of vacancies, defects or other trapping sites would trap the He, which can then lead to gas-atom precipitation, bubble formation and ultimately to exfoliation and loss of the material. It should be noted that the armor lifetime depends on the α-particle energy spectrum, armor temperature, and activation energy for defects and can vary by many orders of magnitude. In addition, annealing of defects at high temperature of the armor might help in releasing He through diffusion. Experiments and detailed modeling is needed to resolve this issue.

4. Summary and R&D Issues

Stellarators have many desirable features as fusion power plants: steady-state operation without externally-driven plasma current (low recirculating power) and stability against external kinks and large vertical displacement events without any feedback coils or conducting walls. A detailed and integrated study of compact stellarator configurations, ARIES-CS, was initiated recently to advance our understanding of attractive compact stellarator power plants and to define key R&D areas. The first phase of the study was devoted to initial exploration of physics and engineering options, requirements, and constraints. In the second phase, we explored the configuration space for stellarators, aiming at configurations that satisfy physics and engineering constraints. The results were reported in this paper.

In the physics area, we have explored several QA configurations. The physics basis of QA as a candidate for compact stellarator reactors has been assessed. New configurations have been developed, others refined and improved, all aimed at low plasma aspect ratios and, hence, compact size at a given fusion power. Configurations with excellent QA have been found with plasma aspect ratios ≤ 4.5. (Configurations with both 2 and 3 field periods are possible.) Progress has been made to reduce the loss of α particles. It appears that introducing a bias in the principle mirror term in the magnetic spectrum plays an important, but not yet fully understood, role in helping reduce α-particle losses. Numerical calculations using codes based on linear, ideal MHD theories show that stability to the kink, ballooning, and Mercier modes may be attained in most cases but at the expense of reduced QA (and increased α-particle losses) and increased complexity of the plasma shape. Recent experimental results indicate, however, that linear, ideal MHD stability theories may be too pessimistic and not applicable to stellarators.

Previous studies had assumed that the radial build of the fusion core is uniform around the plasma. This is not an optimum approach as the external coils are close to the plasma only in certain locations (~8% of first-wall surface area) for the NCSX-like configurations. A novel approach was developed in ARIES-CS for these configurations in which the blanket at the critical areas of minimum stand-off is replaced by a highly efficient WC-based shield – *i.e.,* each system has three radial builds: a shield-only region for locations where coils have to be close to the plasma, a nominal blanket and shield for most other locations, and a transition zone in between. In principle, by utilizing the shield-only region in strategic areas, we have been able to reduce the minimum stand-off, Δ_{min}, by -30% compared to a uniform radial build

that was assumed in previous studies. The reduced Δ_{min} together with the lower aspect ratio plasma lead to power plants that have similar size as advanced tokamak designs.

The device configuration, assembly, and maintenance procedures appear to impose severe constraints on the plasma configurations. Modular coils are designed to examine the geometric complexity and to understand the constraints imposed by the maximum allowable field, desirable coil-plasma separation, coil-coil spacing, and other coil parameters. In particular, because of the irregular geometry and small bend radius, development of inorganic insulators is necessary in order to access maximum field of >9T. A cost-optimization system code has also been developed and will be utilized to assess the trade-off among physics and engineering constraints during the integrated design phase of the ARIES-CS study.

Major R&D issues specific to stellarators that are identified to date include:
1) Development and experimental demonstration of compact stellarator configuration with reduced α-particle loss.
2) Understanding of β limits in stellarators.
3) Demonstration of profile control in compact stellarators with large bootstrap current to ensure the achievement and control of the desired iota profile.
4) Development and experimental demonstration of "pumped" divertor geometries in compact stellarators.
5) Development and experimental demonstration of plasma start-up scenarios
6) Development of high-field superconducting magnets with irregular shape (e.g., in-organic insulators and/or high-temperature superconductors)
7) Engineering accommodation of fast α-particle loss.

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