Compact Stellarator Power Plants – Prospects, Technical Issue, and R&D Directions

F. Najmabadi, A. R. Raffray 1), L. El-Guebaly 2), L-P Ku 3), J. F. Lyon 4) and the ARIES-CS Team*

1) Center for Energy Research, University of California, San Diego, La Jolla, CA, 92093, USA
2) Fusion Technology Institute, University of Wisconsin, Madison, WI, USA
3) Princeton Plasma Physics Laboratory, Princeton University, Princeton, NJ 08520, USA.
4) Oak Ridge National Laboratory, Oak Ridge, TN 37831, USA.

fnajmabadi@ucsd.edu

Abstract. Results of a detailed and integrated study of compact stellarator configurations as fusion power plant, ARIES-CS, are reported in this paper. The first major goal of the ARIES-CS research was to investigate whether stellarator power plants can be made to be similar in size to advanced tokamak variants. We focused our analysis on quasi-axisymmetric (QAS) configurations as they are able to operate at a low plasma aspect ratio (~4-5). Our efforts to reduce α-particle loss rate- led to new criteria for optimizing QAS configuration. We also developed a non-uniform blanket and a WC-shield, optimized to provide shielding comparable to a regular breeding module but with a much reduced (~ 30%) radial thickness. The total tritium breeding including all modules is ~1.1. The second major goal od the study was to understand and quantify, as much as possible, the impact of complex shape and geometry of fusion core components. It became evident early on that the 3-D shape of the plasma and the coil (and the components between them) necessitates 3-D analyses of various components -- typical correlations and insight developed for axisymmetric fusion devices are not appropriate for stellarator geometry. As such, we directly used 3-D CAD models in many of our analyses. Moreover, we found that the results are quite sensitive to the details of 3-D shape of components and slight variations can result in substantial changes. Finally, we have found that engineering configuration as well as assembly/maintenance procedures are key elements in optimizing a compact stellarator – in some cases, these issues determine the choice of technologies. Examples include the selected port-based maintenance scheme which requires a compatible internal design of the fusion core and led to the choice of a ferritic-steel, dual-coolant blanket; and the irregular shape of the superconducting coil that necessitates development of inorganic insulators for high-field magnets. In this paper, trade-offs among physics and engineering constraints are highlighted; key design features and analyses are described; and the major R&D issues are discussed.

1. Introduction

Stellarators have many attractive features as a power plant because of the lack of a 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. On the other hand, the complex shape and geometry of various components introduce many engineering challenges. An integrated study of compact stellarator power plants, ARIES-CS, has been conducted to explore attractive compact stellarator configurations and to define key R&D areas. The detailed design of ARIES-CS can be found in Ref. [1-11]. This paper focuses on the directions we pursued to optimize the compact stellarator as a fusion power plant, summarizes highlights the key design aspects and constraints associated with a compact stellarator, and identifies the major issues to help guide future R&D. Major ARIES-CS Power Plant parameters are given in Table 1 and the ARIES-CS fusion core is shown in Fig. 1.

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* Institutions involved in the ARIES-CS study are University of California San Diego, Boeing, Georgia Institute of Technology, General Atomics, Idaho National Engineering Laboratory, Massachusetts Institute of Technology, New York University, Oak Ridge National Laboratory, Princeton Plasma Physics Laboratory, Rensselaer Polytechnic Institute, and University Wisconsin, Madison.

TABLE 1: ARIES-CS POWER PLANT PARAMETERS

Major radius, $\langle R_{axis} \rangle$ (m)	7.75
Minor Radius, <a> (m)	1.70
On-axis field, $\langle B_0 \rangle$ (T)	5.7
Peak field on the coil, B_{max} (T)	15.1
Plasma current (MA)	4
Plasma $\langle\beta\rangle$	6.4%
Plasma temperature, <T> (keV)	6.6
Electron density (m^{-3})	4×10^{20}
ISS95 confinement multiplier	2
α-particle loss fraction	5%
Fusion power (MW)	2,440
Thermal Power (MW)	2,920
Thermal Conversion efficiency	43%
Average neutron wall load (MW/m^2)	2.6
Peak neutron wall load (MW/m^2)	5.4

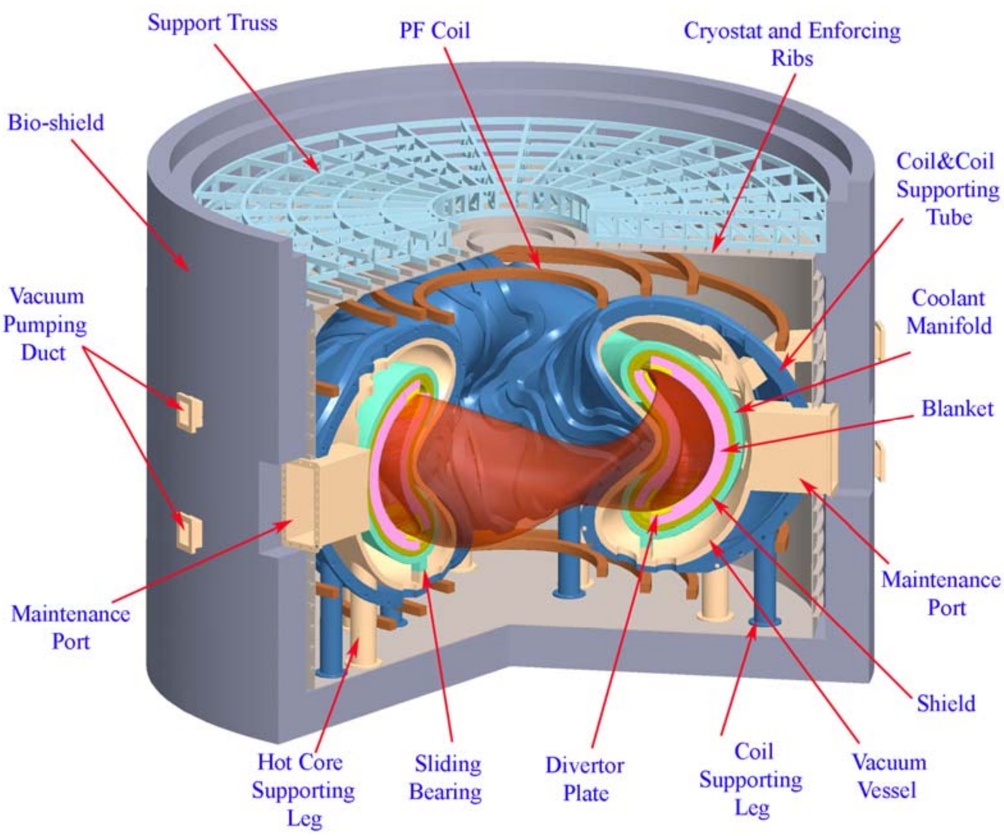

FIG. 1. The ARIES-CS fusion power core.

The large size and mass predicted by earlier stellarator power plants studies had led to cost projections much higher than those of advanced tokamak power plant. As such, the first major goal of the ARIES-CS research was to investigate if stellarator power plants can be made to be comparable in size to advanced tokamak variants while maintaining desirable stellarator properties. These issues are discussed in Sec. 2.

As stellarator fusion core components would have complex shapes and geometry, the second major goal of the ARIES-CS study was to understand and quantify, as much as possible, the impact of complex shape and geometry of fusion core components (discussed in Sec. 3.) Section 4 summarizes our major findings and highlights the key R&D directions.

2. Compact Stellarators

The first major goal of the study was to investigate if stellarator power plants can be made to be similar in size to advanced tokamak variants. It is well known that the cost of electricity (COE) decreases substantially as machine size is reduced and then levels off when the average plasma radius becomes roughly similar in size to the plasma-coil spacing. Most modern tokamak power plant studies envision machines with a major radius of 5-8 m.

Earlier stellarator power plant studies arrived at devices with an average major radius 2-3 times larger than advanced tokamaks. Because the external coils in a stellarator generate a multipolar field and the high order harmonics of the magnetic field decay rapidly with distance from the coils, the distance between the plasma (i.e., the last closed magnetic surface, LCMS) and the middle of the coil winding pack play a critical role in optimizing the stellarator configuration. For a given stellarator configuration, the machine average major radius is directly proportional to the plasma-coil spacing. However, this space is occupied by various components such as the blanket and shield – typically these components require a distance of 1.5-2 m between the LCMS and the middle of coil winding pack.

Based on the above observation, three avenues were pursued to reduce the size:

a) Develop stellarator configuration with lower aspect ratios,
b) For a given stellarator configuration (*i.e.,* plasma aspect ratio), increase the coil-plasma spacing by developing coil design with a lower coil aspect ratio and optimizing coil cross section.
c) Develop engineering options to reduce the required minimum distance between the plasma and the mid-point of the coil (Δ_{min})

2.1. Plasma-coil Configuration

We focused our analysis on quasi-axisymmetric (QAS) configurations as they are able to operate at a lower plasma aspect ratio (~4-5) compared to other stellarator configurations. The ARIES-CS reference plasma, N3ARE, is a NCSX-like configuration with 3 field periods, has an average major radius , $\langle R \rangle = 7.75$ m and a plasma aspect ratio, $\langle R \rangle / \langle a \rangle = 4.5$ [2]. The configuration has excellent quasi-axisymmetry, as measured by the effective helical ripple, e_{eff} ($e_{eff} < 0.6\%$ at LCMS and ~0.1 in the core region). The shaping of the plasma results in a vacuum rotational transform from ~ 0.4 to ~0.5. The plasma current (bootstrap-current) is ~4 MA which raises the rotational transform to ~0.7 near the plasma edge (See Fig. 2.)

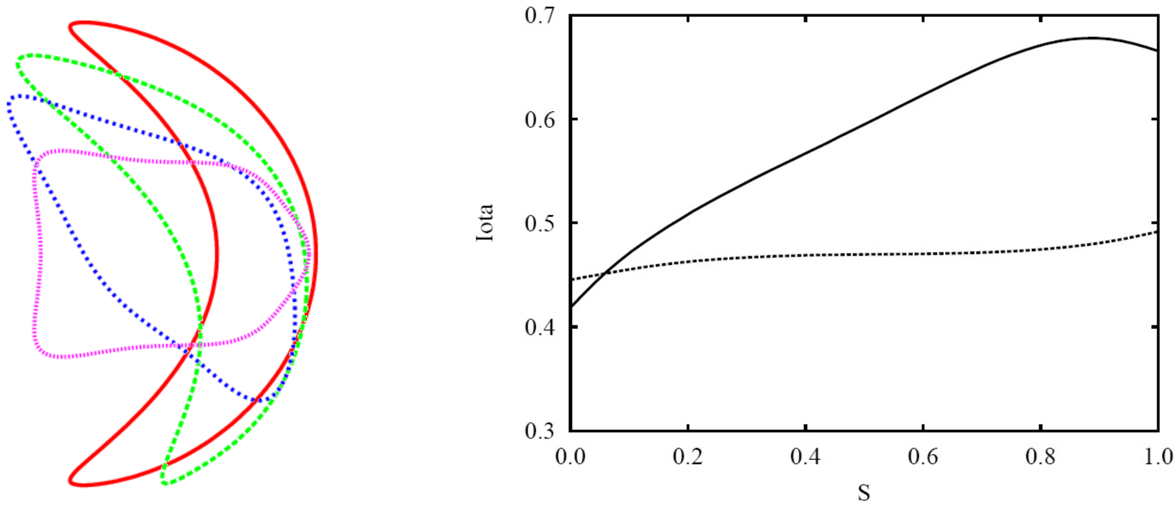

FIG. 2. A) Cross section of the LCMS at four toroidal sections and B) the rotational transform as a function of normalized toroidal flux s for the N3ARE configuration. The dashed line is the rotational transform from external coils (or transform at β = 0) and the solid line is the total rotational transform (iota) including the bootstrap current contribution at β = 5% .

The distinct feature of this configuration is that a bias is introduced in mirror and helical terms of the magnetic spectrum in order to alter the structure of the ripple and drastically improve α particle confinement [2]. Operation at low plasma temperature and high density also decreases α slowing down and further reduces α particle losses. Still, the α particle loss rate in ARIES-CS is high (~5%). While the heat flux from these lost α particles may be handled on the divertor target plate, erosion and, in particular, exfoliation due the accumulation of He atoms in the armor are major concerns. As such, developing configurations with lower α loss is a key research area.

We also optimized the coil design that can produce the designed target plasma shape in order to maximize the coil-plasma desistance for a given <R>. Several coil designs with different coil aspect ratio, A_c=<R>/Δ_{min} were produced. In addition, the coil cross section (shape and size) was varied [2]. The ARIES-CS base-line coil design has a coil aspect ratio of 5.8 with rectangular cross-section conductors (~0.19 m radially and ~0.74 m toroidally). The maximum field in the winding pack of the coils is ~15 T for an on-axis plasma field of 5.7 T.

2.2. Reducing Plasma-Coil Distance

Typically, the distance between the plasma and the front of the coils are set by the requirements of adequate breeding and shielding of the coils. An innovative approach was developed to downsize the blanket and utilize a highly efficient WC-based shield in the space-constrained regions where plasma is close to the coil [5]. The special modules in these regions are optimized to provide shielding comparable to a regular breeding module but with a much reduced module radial thickness. The final design of the blanket consists of a tapered breeding region with a blanket thickness ranging from 54 cm in regions where more space is available to 25 cm at the minimum plasma-to-mid-coil distances (for example see the “mid-plane” inboard region in Fig. 1). The corresponding plasma-to-mid-coil radial thickness for a regular breeding module is about 1.79 m, decreasing to 1.31 m at the closest plasma-coil position. The total tritium breeding including all modules is ~1.1, found from detailed 3-D neutronics analysis of the CAD model of ARIES-CS. Using a tapered blanket has had a major impact on the machine size: a device with a uniform blanket and shield at all location would have a major radius > 10 m compared to 7.75 m for ARIES-CS.

3. Engineering Complexity

The complex shapes and geometry of stellarator plasma and coils impose severe constraints on fusion core components such as non-uniform heat, particle, and neutron fluxes; accessibility for assembly and maintenance; and feasibility and cost of manufacturing.

The 3-D shape of the plasma and the coil (and the components between them) necessitates 3-D analysis of various components. Moreover, we found that the results are quite sensitive to the fine details of 3-D shape of components and slight variations can result in substantial changes. For example, nuclear assessment of the system (breeding, shielding, blanket thermal recovery) requires utilization of 3-D neutron transport codes such as MCNPX Monte Carlo code. We developed a new tool which uses CAD models for in-vessel components (i.e., first wall, blanket, etc.) to generate various zones for MCNPX [5]. This tool also uses plasmas equilibrium to generate 3-D source terms for both fusion neutrons and plasma radiation. This tool was utilized in designing the tempered blanket concept.

An important result of our nuclear analysis of ARIES-CS is the large peaking factor of ~2 in the neutron wall loading (peak value of 5.3 MW/m^2 and average value 2.6 MW/m^2) which is considerably larger than tokamaks (1.5 for ARIES-AT). This rather large peaking factor and the similarly large peaking factor for heat fluxes have a dramatic impact on the engineering design of the system.

We have found that the engineering configuration as well assembly and maintenance procedure are key elements in optimizing a compact stellarator – in some cases, these issues determine the choice of technologies that can be utilized. We considered several assembly and maintenance procedures early in the study. It appears that the only feasible option is a port-based maintenance scheme which involves the replacement of blanket and other in-vessel components through a number of designated ports. This is a complicated and time-consuming procedure as the size of the ports as well as weight limit of maintenance boom limits the size of a typical blanket module to ~3-4 m^2 on the side facing the plasma (about 200 such modules in ARIES-CS). Furthermore, module removal and replacement requires the cutting and re-welding of the coolant pipes which necessitates removal of a neighboring module first [4]. The need for a large number of modules with coolant pipes which can be cut and re-welded in a relatively short time drove us toward the choice of a ferritic steel-based based blanket as reference option instead of a higher-performance blanket utilizing SiC composite as structural material (e.g., SiC/Pb-17Li blanket of ARIES-AT [12]). The reference blanket concept is a dual-coolant concept with self-cooled Pb-17Li zones and He-cooled ferritic steel structure [4].

The technology and design of superconducting magnets are another example of the impact of configuration on the choice of available technologies. The candidate superconductor materials for operation at high field (up to 16 T) are variants of Nb_3Sn or MgB_2. Both are glassy materials and their current carrying capability is drastically reduced with induced strain. For these types of superconductors, the superconductor material is typically wound around a large spool (with roughly the same radius of curvature as the coil itself) and is heat treated (700^{o}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 a compact stellarator. 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. Glass-tape inorganic insulators that can withstand the heat treatment process have been developed recently in small scale [13]. Extrapolation of this insulator to large coils would allow stellarator magnets with B_{max} up to 16 T and is assumed for ARIES-CS. Otherwise, ductile NbTi superconductors should be used. For NbTi technology, the maximum field would be limited to ~7 T for 4 K operation. This reduction in maximum field strength will lead to substantial a increase in the machine size.

As most of the component cost in a fusion system is due to manufacturing process (and not the cost of the raw material), it is expected that there will be additional costs associated with the irregular and complex shape of stellarator components. These additional costs proved difficult to quantify because of lack industrial experience in manufacuritn large-scale irregular components and also because in some cases conventional manufacturing techniques would be quite expensive and challenging. However, advanced manufacturing techniques, under development, would allow for component size not to be a limiting factor and the process to be highly automated with minimal labor. In principle, these advanced manufacturing techniques are more suitable for monolithic structure such as superconducting coil structure as opposed to heterogeneous structures such as blanket modules. As an example, we explored such an advanced manufacturing technique for the ARIES-CS coil support structure [11] and in estimating the cost-of-electricity (COE).

4. Technical Issues and R&D Directions

Overall, ARIES-CS study demonstrates that compact stellarator power plants which are comparable in size to advanced tokamaks are possible. Because of the lack of industrial experience on feasibility and cost of manufacturing of irregularly shaped components, we were not able to assess the trade-off between increasing the size and mass to lower complexity of fusion core components. Lessons learned from the construction of NCSX and W7X would be valuable information for follow-up studies of compact stellarator power plants. Nevertheless, optimizing the plasma configuration in order to increase the relative distance between the plasma and the coils (*e.g.*, lower plasma aspect ratio, increased coil aspect ratio) is a desirable approach as it would lead either to components with smaller irregularity and complexity (for a fixed machine size) or it would lead to a smaller device (for same component complexity).

The device configuration, assembly and maintenance drive the design optimization and technology choices in many cases. Examples include port-based maintenance scheme which drives the internal design of fusion core and led to the choice of a ferritic-steel, dual-coolant blanket. Another example is the irregular shape of the superconducting coil that necessitates development of inorganic insulators for high-field magnets.

The configuration space for QAS stellarators is quite rich and many desirable configurations are possible. Of course, the QAS configuration with reduced α-particle loss should be developed and demonstrated experimentally. Profile control in compact stellarators is essential in order to ensure the achievement and control of the desired iota profile, including bootstrap current effects. While the alpha loss does not impact the plasma power balance appreciably, it leads to serious issues for in-vessel components: a) it can lead to localized heating and increase peak-to-average loads, and b) more importantly, energetic α particles can damage the material wall through blistering. Optimization of plasma configuration to reduce the "energetic" (i.e. > 10 keV) α loss fraction further is needed.

An important constraint in optimizing a stellarator configuration is the plasma β limit. The mechanisms limiting the plasma β in a stellarator are not well understood as stellarator experiments have operated at β values substantially larger than predicted by the linear MHD theory (*e.g.,* 3.2% achieved β in W 7-AS versus 2% for the theoretical prediction [14]). Understanding the mechanism that limits the plasma β is an important physics topic which should be addressed in the next generation of stellarators. Not only would relaxing linear MHD stability constraint allow optimization of other parameters (*e.g.,* α-particle losses), a larger β value can also be used to reduce the technological requirement of the superconducting magnets.

Similar to advanced tokamaks, a large portion of plasma power should be radiated in order to obtain a reasonable heat flux on the divertors. This mode of operation should be demonstrated in experiments. Field-line tracing analysis of the reference configuration resulted in peak load of 18.4 MW/m^2 (and average heat load of 0.77 MW/m^2) which exceeds the 10 MW/m^2 engineering limit. Because of lack resources and the 3-D nature of the problem, neither an aspect ratio scaling of the divertor heat load nor an optimized divertor design study has been carried out fully. Careful tailoring of divertor plates may lead to a reduction of peaking factor on the divertor plates and an acceptable solution but will probably require careful alignment of the divertor plates. We also focused on operation with a high density (4 x 10^{20} /m^3) because of the $\tilde{n}_e^{0.51}$ dependence in ISSS95 scaling and also the belief that high density operation would help reducing the heat flux on the divertor plates. Neoclassical theory predicts impurity accumulation in steady-state stellarator discharges with such high densities and this is typically observed in experiments. However, both LHD and W7-AS have operated in high-density regimes without impurity accumulation [15]. Development and experimental demonstration of "pumped" divertor geometries in compact stellarators with a highly radiative plasma including operation with a high plasma density without any impurity accumulation are critical R&D topics.

Because of lack of resources, we did not explore in detail the plasma start-up scenario for ARIES-CS. Plasma start-up scenarios and path to ignition with resonance-avoidance techniques for should be developed and experimentally demonstrated.

On the engineering side, key R&D items specific to stellarators include: A) Development and demonstration of methods to fabricate, assemble, and maintain large superconducting stellarators free of resonance-inducing field errors; B) Engineering accommodation of fast α-particle losses, and C) Development of high-field superconducting magnets with irregular shape (e.g., in-organic insulators and/or high-temperature superconductors).

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