Exploration of Compact Stellarators as Power Plants: Initial Results from ARIES-CS Study

Farrokh Najmabadi and the ARIES Team

Department of Electrical & Computer Engineering and Center for Energy Research, University of California San Diego, La Jolla, CA 92093-0438
fnajmabadi@.ucsd.edu

A detailed and integrated study of compact stellarators as power plants, ARIES-CS, was initiated recently to advance our understanding of attractive compact stellarator configurations and to define key R&D areas. We have completed phase 1 of ARIES-CS study—our results are described in this paper. We have identified several promising stellarator configurations. High α particle loss of these configurations is a critical issue. It appears that devices with an overall size similar to those envisioned for tokamak power plants are possible. A novel approach was developed in ARIES-CS in which the blanket at the critical areas of minimum stand-off is replaced by a highly efficient WC-based shield. In this manner, we have been able to reduce the minimum stand-off by ~20%-30% compared to a uniform radial build which was assumed in previous studies. Our examination of engineering options indicates that overall assembly and maintenance procedure plays a critical role in identifying acceptable engineering design and has a major impact on the optimization of a plasma/coil configuration.

I. INTRODUCTION

In a stellarator, the majority of the confining field is produced by the external coils (poloidal field is generated by the external coils as well as the bootstrap current). Because there is no large driven external current stellarators are inherently steady state (low recirculating power), and are stable against external kink and axisymmetric modes and resilient to plasma disruptions. These advantages should be balanced against complicated external windings and irregular cross section of the plasma and in-vessel components.

Earlier stellarator power plant studies led to devices with large sizes. The HSR (Helias) study is based on the W7-X plasma configuration. It has an average major radius, $\langle R \rangle$ = 22 m for a five-field-period configuration and $\langle R \rangle$ = 18 m for a recent four-field-period configuration[1]. The FFHR[2] is a 10-field period Heliotron/Torsatron (l=2 stellarator) and has $\langle R \rangle$= 10-20 m. The ARIES Stellarator Power Plant Study (SPPS), completed in 1996, was based on the four-field-period MHH (Modular Helias-like Heliac) configuration and led to a $\langle R \rangle$ = 14 m device and was the first step toward a smaller-size stellarator power plant[3]. More recent plasma/coil configurations with lower plasma aspect ratio (compact stellarators) have the potential of even smaller devices.

Because, the external coils generate a multipolar field, the distance between plasma and the coil is a critical parameter. As such, optimization of any stellarator configuration represents a large number of tradeoffs among physics parameters and engineering constraints. For example, fixed-boundary analysis of a stellarator configuration may lead to a high-performance plasma configuration which cannot be produced with any practical coils and/or cannot accommodate a power-producing blanket.

A detailed and integrated study of compact stellarator power plants, ARIES-CS, was initiated recently to advance our understanding of attractive compact stellarator configurations and to define key R&D areas. The stellarator configuration space is quite complex because of the large number of independent parameters (*e.g.,* β, α-particle loss, aspect ratio, number of periods, rotational transform, shear, *etc.*). Furthermore, engineering requirements and constraints such as coil topologies and maintenance approaches (which will have a major impact on in-vessel components, blanket, and power systems) may depend on details of a specific configuration. As such, the study ARIES-CS is divided into three phases. The first phase of the study was devoted to initial exploration of physics and engineering options, requirements, and constraints. Several compact stellarator configurations such as quasi-axisymmetric and quasi-helical were considered. In each case, trade-offs among plasma parameters (*e.g.*, α-particle loss versus β) were explored and possible coil topologies were studied. Initial estimates of device size, first-wall and blanket power loadings, divertor heat loads, *etc.* were made with a systems model. Promising configurations identified in phase 1 will be subjected to detailed self-consistent analysis and optimization in phase 2. Detailed self-consistent analysis of this phase will allow us to identify critical high-leverage areas for compact stellarator research. One of the promising configurations chosen in phase 2 would be used for a detailed and self-consistent point design study in phase 3.

We have completed phase of 1 of ARIES-CS study—our results are described in this paper and Refs. 4 through 11. We have identified several promising stellarator configurations (Sec. 2). It appears that devices with an overall size similar to those envisioned for tokamak power plants are possible. Our examination of engineering options (Sec. 3) indicates that overall maintenance approach plays a critical role in identifying acceptable engineering designs and has a major impact on plasma dimensions and performance. Overall summary and directions for phase 2 research are given in Sec. 4.

II. PLASMA CONFIGURATIONS

We have explored several quasi-axisymmetric (QAS) configurations during the first phase of ARIES-CS study.

Development of these and other recent stellarator configurations has been made possible by the efficient stellarator configuration optimization techniques pioneered by Nuhrenberg[12]. These techniques optimize the plasma properties (*e.g.,* rotational transform, MHD stability criteria, α-particle loss) by varying the shape of the last closed magnetic surface. The external coil set that would generate this configuration can then be found by matching the normal component of the magnetic field strength on the last closed magnetic surfaces with that generated by the external coils. (More modern codes use an integrated approach that optimizes the last closed flux surface and the coils simultaneously.) Because the external coils produce a multipolar field, the magnetic field intensity drops rapidly away from the coil. As such, the space between plasma and the coil (*e.g.,* scrape-off layer, fusion core) as well as constraint imposed by magnet technology (*e.g.,* minimum bend radius, support structure, and inter-coil spacing needed for assembly as well as maintenance of in-vessel components) play a critical role in configuration optimization.

The QAS configuration has attracted intense interest in recent years as the underlying quasi-axisymmetric magnetic field structure leads to particle orbits similar to those in a tokamak. As such, this configuration has the potential to combine the desirable features of tokamaks (good confinement, high β, 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). A relatively low aspect ratio (A=4.5) proof-of-principle device, NCSX, is under construction and is expected to operate by 2008 in the US[13].

Three distinct classes of QAS configuration have been considered for the ARIES-CS. First is the scale-up and upgrade[14-16] of the NCSX configuration. The NCSX-class configuration 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 configurations with A = 4.5 in which α-loss is reduced to ≤ 15% while the plasma remains stable against linear MHD modes for β ≥ 4%. This configuration is shown in Fig. 1. These configurations have good equilibrium properties for up to β = 8% and practical coils with a plasma-coil separation aspect ratio of ~ 6 is feasible. Configuration space was also extended to a broader rotational transform (iota) region. Initial systems analysis indicates that a 1-GWe power plant with an average major radius of ~ 8m is possible. Alpha-particles losses, however, are still large and design and operation of in-vessel components under such a high α flux remain a major issue. It may be possible to reduce α losses by relaxing the MHD linear stability constraints.

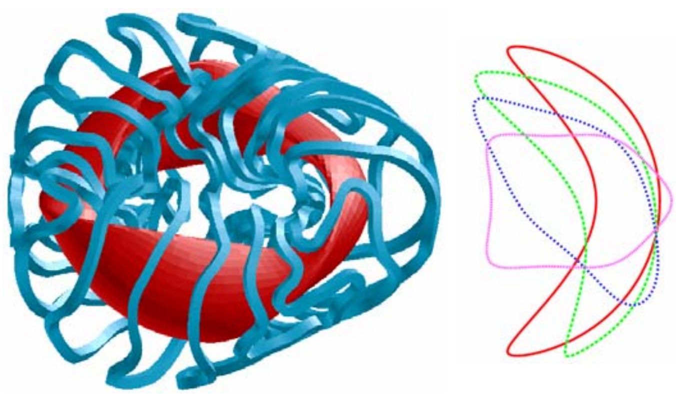


Fig. 1. A view of NCSX-like configuration developed for ARIES-CS study. Cross sections of the last closed magnetic surface at different toroidal angles are also shown.

We have also developed NCSX-class configurations with a plasma aspect ratio of 6, which has better quasi-axisymmetric properties compared with A = 4.5 case, in order to investigate the impact of the power plant performance on plasma aspect ratio in phase 2. NCSX-class configuration with 2 field periods has been developed although the plasma shape is "awkward" for these cases[16].

The second class of QAS configurations considered is aimed at improving equilibrium β limit and flux surface quality by careful tailoring of the external rotational transform[16]. Two lines of development has been pursued:

a) SNS-QA configuration in which externally generated iota is chosen to avoid low order resonances at finite β, and b) LPS-QA in which externally generated iota is chosen to minimize the impact of low order resonances but maintain high positive shear at full β. This class of configurations is in the early stage of development; although high quality flux surface, excellent quasi-axisymmetry and reasonable α-loss (~10%) have been demonstrated for SNS-QA.

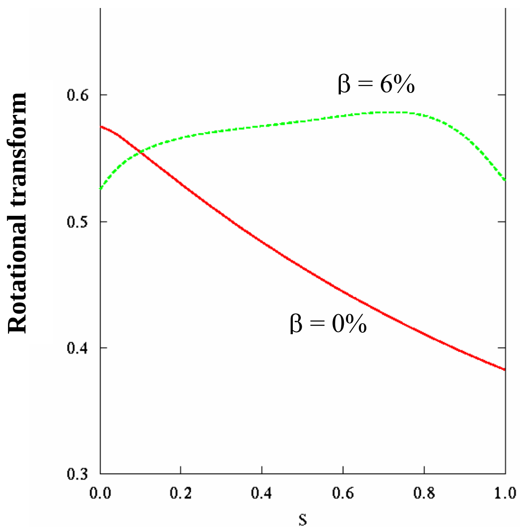


Fig. 2. Iota profile of SNS-QA configuration. Externally generated iota is chosen such that the iota profile at full β avoids low order resonances.

The third class of QAS configuration is the extension of the MHH configuration to 3 and 2 field periods[4] (hereafter referred to MHH2). The relatively simple shape of the plasma and external coils, especially in the 2-field-period case, makes this configuration especially attractive for a power plant (see Fig. 3). Configurations with various iota profile at a plasma aspect ratio of 2.5 has been found which have excellent quasi-axisymmetry, low effective ripple (<< 1%) and low α loss.

III. ENGINEERING OPTIONS

The choice of breeding blanket and shield plays an important role in optimizing the stellarator configuration for a power plant. First, the needed space between the plasma and the coil (scrape-off-layer, first wall, blanket, shield, *etc.*) is a critical parameter in determining the external coil design and overall device optimization – the device major radius directly scales with this minimum stand-off, Δ_{min}. This distance is set by the nuclear performance of the blanket/shield region, *i.e.,* tritium breeding and magnet protection. Second, the constraints on the external coils (*e.g.,* inter-coil spacing, support structure) play an important role in the device optimization. The above two sets of constraints are directly coupled to the proposed procedures for the machine assembly and the scheduled maintenance of the power core (regular replacement of first wall and blanket). Thirdly, the thermal performance of the blanket (*i.e.,* thermal efficiency) has a direct impact on the fusion power and machine size.

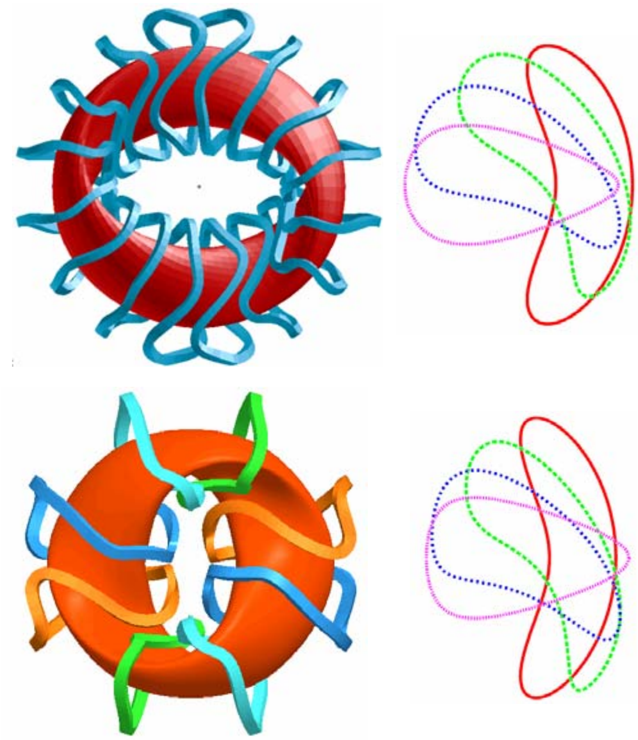


Fig. 3. MHH2 configurations developed for ARIES-CS study. The top configuration has A = 3.7 and 16 coils and the bottom configuration has A = 2.7 and 8 coils. Cross sections of the last closed magnetic surface at different toroidal angles are also shown.

We have considered five different blanket concepts for the first phase of ARIES-CS study:

1) Self-cooled flibe blanket with advanced ferritic steel structure. The molten salt system always needs a beryllium multiplier to meet the breeding requirement. The system has a coolant outlet temperature of 700 °C and a thermal conversion efficiency of 45%.

2) Self-cooled Pb-17Li blanket with SiC_f/SiC composite as structural material. This is an adaptation of the ARIES-AT[17] blanket to a compact stellarator configuration. The Pb-17Li flows through the first wall at a high speed, and then flows slowly within the blanket. This flow arrangement allows for Pb-17Li coolant exit

temperature of ~ 1100°C, which is higher than the maximum structure temperature, leading to a high thermal conversion efficiency of the system (55%-60%). This blanket has excellent safety and environmental characteristics.

3) Dual-coolant blanket concept with He-cooled ferritic steel structure and self-cooled Pb-17Li. This is an adaptation of the ARIES-ST blanket[18] to stellarator configuration. Silicon-carbide inserts (0.5-cm thick) are used to control the MHD effect and maintain the steel temperature below 600 °C while allowing the Pb-17Li exit temperature to be ~ 700 °C. This blanket has a thermal conversion efficiency of 45%.

4) Dual-coolant blanket concept with He-cooled steel structure and self-cooled Li. This concept is similar to no. 3 above except uses Li instead of Pb-17Li as coolant/breeder.

5) Helium cooled ceramic breeder blanket with ferritic steel structure and Be multiplier. The proposed design[8] features multiple Li_4SiO_4 and Be layers sandwiched between cooling channels to efficiently remove the nuclear heating and operate within the temperature windows for Be and the solid breeder. This design can handle up to 4.5 MW/m^2 peak neutron wall loading and has a thermal conversion efficiency of ~ 45%.

A detailed description of these blanket concepts as well as their thermal-hydraulic parameters are given in Ref. 6 and 8. The nuclear performance of these blankets is discussed in Ref. 7. Here, we focus on the minimum stand-off distance and the assembly/maintenance procedures for these concepts and their impact on optimizing the stellarator configuration.

III.A. Radial Build and Minimum Stand-off

Previous studies[1,3] 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 the first-wall surface area for the NCSX-like configurations). A novel approach was developed in ARIES-CS in which the blanket at the critical areas of minimum stand-off is replaced by a highly efficient tungsten-carbide-based (WC) shield – each system has two radial builds: a shield-only region for locations where coils have to be close to the plasma and a nominal blanket and shield for other locations (see Fig. 4). This dramatically reduces the minimum stand-off distance. This approach, however, requires careful design of the "nominal" blanket (increasing the tritium breeding ratio) to ensure overall tritium self-sufficiency.

Detailed neutronics analyses of ARIES-CS radial build is discussed in Ref. 9. These analyses were performed by a toroidal-cylindrical model which approximately captures the 3-D effects (a 3-D analysis is still needed to confirm these results). It is further assumed that the penetrations occupy 2% of the first-wall area, and the divertor plates/baffles cover 15% of the first-wall area (although blanket regions are located behind the divertor). The results are summarized in Table 1 for the five blanket concepts discussed above. In principle, by utilizing the shield-only region in strategic areas, we have been able to reduce the minimum stand-off by ~20%-30% compared to a uniform radial build which was assumed in previous studies. This would allow a comparable relative reduction in machine size. For comparison, the SPPS study[3] used a self-cooled Li blanket with Vanadium alloy as the structural material with a uniform radial thickness of 1.8 m while our new approach allows for a minimum stand-off distance of 1.1-1.3 m.

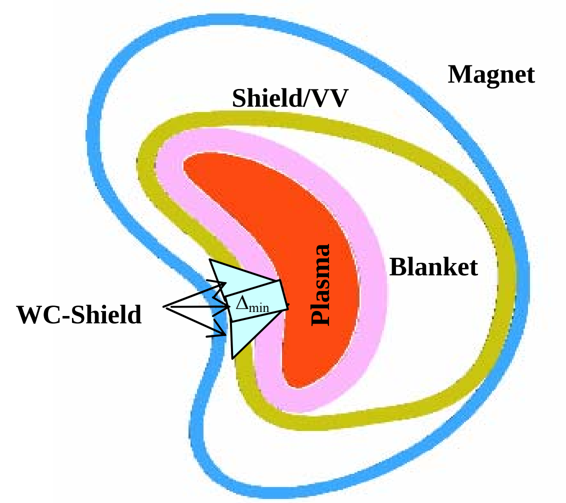


Fig. 4. Schematic of approach taken to reduce minimum stand-off between plasma and coil in ARIES-CS. A WC-shield is used in the critical regions where coils are in close proximity of the plasma. Note that these regions are also localized in the toroidal direction.

TABLE I. Radial thickness of shield only region (Δ_{min}) and of the nominal blanket for the five blanket concepts.

	Δ_{min} (m)	Nominal Δ (m)
1. FS/Flibe/Be	1.11	1.32
2. SiC/Pb-17Li	1.14	1.40
3. FS/He/Pb-17Li	1.18	1.49
4. FS/He/Li	1.16	2.04
5. Solid breeder	1.29	1.55

III.B. Assembly and Maintenance

Because of the complex shape and the precise location of stellarator external windings, access to the internals of the system (*e.g.,* first wall, divertor, blanket and shield) is very limited. The device configuration and assembly should also include the support structure for the external windings as well as support of the fusion core against gravity.

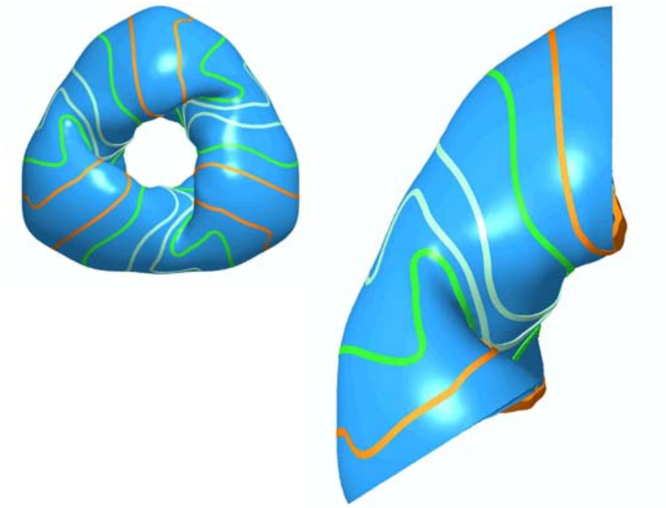
As such, assembly and maintenance of a stellarator fusion system are considerably more challenging compared with a tokamak. In tokamak, it is possible to extend the outer leg of the toroidal-field (TF) coil sufficiently away from the plasma such that the in-vessel components can be accessed and removed through the space between adjacent TF coils[17]. This cannot be done in a stellarator.

In order to assess the impact of configuration, assembly, and maintenance on the optimization of a stellarator configuration, we considered three distinct approaches: a) Field-period based assembly and maintenance; b) Modular assembly and maintenance through a small number of designated ports; and c) Modular assembly and maintenance through ports between each pair of adjacent coils. These concepts are discussed below. It appears that each favors a certain blanket concept and/or stellarator configuration.

III.B.1. Field-Period-Based Assembly and Maintenance

In this scheme, the external windings are wound in groves on shells that extend over an entire field period (See Fig. 5). Because the inter-coil forces cancel out over a field period, the shell can be made quite thin compared to discrete support elements. The hoop forces are also supported by the shell by winding the superconducting coils on the grove located inside the shell. A bucking cylinder, however, is required to support centering forces. The shell and the superconducting windings are enclosed in a cryostat. The entire system is enclosed in an external vacuum vessel as is shown in Fig. 6.

For maintenance purposes, the external part of the vacuum vessel is removed. (See the joints on the top and the right-bottom of the vacuum vessel in Fig. 6). The entire field-period unit is moved radially outward on special rails. The shield is a permanent, life-of plant component and is not removed during maintenance procedure. The inner components to be replaced (first wall, divertor, and blanket) are divided into two parts that are removed toroidally from each end of the field-period unit.



(a) (b)

Fig. 5. Support shells for the stellarator coils of a three-field period NCSX-class configuration: a) Assembled system, and b) Enlargement of one of the three shells. Note that the coils are wound into groves inside the shell, the outline of these groves are shown on the outside to highlight the location of windings.

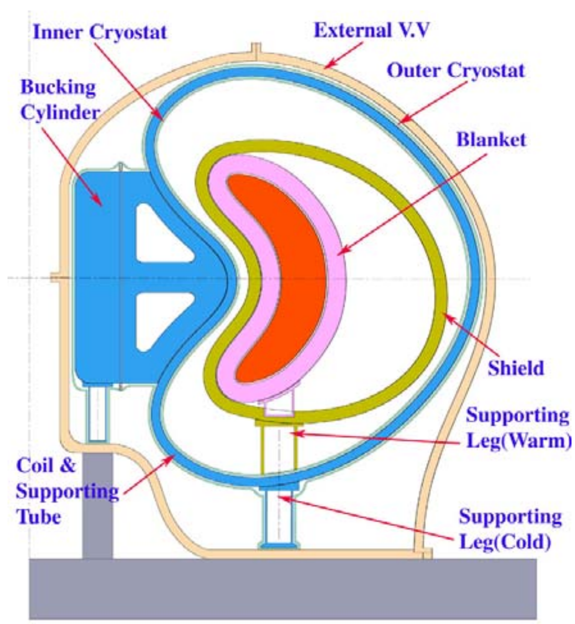


Fig. 6. Poloidal cross section of an NCSX-class configuration (at 0 degrees) based on a field-period assembly and maintenance scheme. The figure shows the bucking cylinder and the external vacuum vessel. The shield is a permanent (life-of-plant) component and is shaped such that only the first-wall, blanket, and divertor can be removed (see text).

The field-period based maintenance scheme enables the use of very large blanket units nearly without weight limitations. This scheme, however, requires large blanket modules in order to minimize the number of coolant connections to be cut/rewelded for blanket replacement, and to allow location of these connections close to the points of mechanical support in order to avoid problems with differential thermal expansion. The SiC/Pb-17Li and the dual coolant blanket concepts above (blanket concepts no. 2 through 4) appear to be well suited for this scheme.

The major issues with this scheme include motion of large components and the need to "warm up" the winding pack/shell that has to be moved. A key factor is the clearance between the stationary shield and the blanket unit when it is being removed toroidally. This needed clearance may introduce a "new" minimum stand-off distance that does not occur in the location of closest distance between the plasma and the coils.

III.B.2. Assembly and Maintenance through a Small number of Ports

In this scheme, the first-wall, divertor, and blanket are replaced through a small number of ports. Installation of a toroidal rail system inside the plasma chamber (similar to ITER) that supports the maintenance boom is not possible because of the 3-D character of the in-vessel components (the toroidal rail system would be similar to a roller-coaster). As such, an articulated boom should be utilized to install, inspect, and replace these components. This will dramatically limit the weight of any module.

Since only the blanket modules are to be moved, this scheme calls for a different vacuum vessel and cryostat design. In this case, the vacuum vessel is internal to the coils and serves as an additional shield for the protection of the coils. The maintenance ports are arranged between adjacent coils at a few locations with larger port space and larger plasma cross section. Transfer casks can be attached to the outside flange of the port, and a system of double doors can be employed to avoid any spread of radioactivity (dust, tritium) into the containment building.

This scheme does not require motion of large components or warm-up of the coils. By placing the vacuum vessel inside the coil winding, the need for cutting/re-welding of the vacuum vessel is also eliminated. On the other hand, the load capacity of the boom (limited to about ~5,000 kg) limits the weight and size of the blanket modules. As such, this scheme is more suited for blanket concepts such as ferritic steel/flibe or SiC/Pb-17Li (blanket concepts no. 1 and 2 above) that yield "lighter" blanket modules. In addition, this scheme require very high reliability for the permanent parts of the fusion core as replacement and repair of these component would require complete disassembly of the system. This approach may not be suitable for NCSX-like configuration as the space between coils is quite limited (see Fig. 6.)

III.B.3. Assembly and Maintenance through large ports between each pair of adjacent coils

This assembly and maintenance approach can be viewed as an extension of the previous modular maintenance approach but with replacement of much larger blanket modules through a larger number of ports arranged between each pair of adjacent coils, using a shorter boom. This approach was evaluated for the MHH2 configuration with 8 coils in which larger inter-coil separation exists – it is not suitable for NCSX-like configuration as the space between coils is quite limited (see Fig. 6.). The device is configured similar to the previous modular approach: an internal vacuum vessel and an external cryostat. The significant improvement is that this approach uses shorter booms with much higher load capacities. The disadvantage is that more ports are required and they are larger in size, which places more geometry constraints on the coil configuration. This maintenance method has been suggested in all IPP Garching Stellarator studies[1].

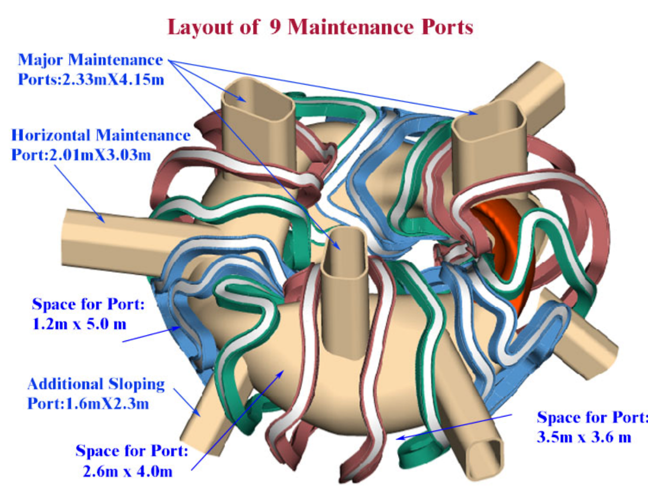


Fig. 6. Possible location of maintenance ports for an NCSX-like configuration. Assembly and maintenance through ports scheme is generally more suited to MHH2 – type configurations which have a larger inter-coil separation (see Fig. 3.)

IV. SUMMARY AND CONCLUSIONS

Stellarators have many desirable features as fusion power plants: steady-state operation without externally-driven current (low recirculating power) and lack of large plasma currents (stability against external kinks and large vertical displacement events). In addition, a compact stellarator with a relatively modest aspect ratio may lead to devices with reasonable size and cost. These desirable characteristics should be balanced against complicated external windings and in-vessel components with irregular cross-section and complex 3-D shape. In addition, because, most of the confining field is provided by external coils that generate a multipolar field, the distance between the plasma and the coil is a critical parameter. Therefore, optimization of any stellarator configuration involves a large number of tradeoffs among physics parameters and engineering constraints.

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 ARIES-CS study is divided into three phases. The first phase of the study was devoted to initial exploration of physics and engineering options, requirements, and constraints. Promising configurations identified in phase 1 will be subjected to detailed self-consistent analysis and optimization in phase 2 leading to a detailed point design in phase 3. This paper has summarized the results of phase 1 of the ARIES-CS study.

In the physics area, we have explored several quasi-axisymmetric (QA) configurations. The physics basis of QA as 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 $A \leq 6$. (Configurations with both 2 and 3 field periods are possible.) Progress has been made to reduce loss of α particles. Alpha losses of ~10% have been achieved. This is still higher than desirable, however. 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 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.

We have been developing the tools necessary to assess the location and heat and particle fluxes due to α particles and edge plasma. Detailed analysis of plasma facing components will be performed in phase 2 and is expected to introduce severe constraints on the acceptable level of α losses.

Our physics configuration optimization has been focused on the optimization of the final plasma configuration at full β. Start-up procedures require developing a series of configuration at different plasma β and require additional coils. This research has also been left for the second phase of the ARIES-CS study.

It appears that the minimum stand-off distance between plasma and coils is not as an important a parameter as envisioned previously. 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 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 two radial builds: a shield-only region for locations where coils have to be close to the plasma and a nominal blanket and shield for other locations. In principle, by utilizing the shield-only region in strategic areas, we have been able to reduce the minimum stand-off by ~20%-30% compared to a uniform radial build which was assumed in previous studies. This would allow a comparable relative reduction in machine size.

The device configuration, assembly, and maintenance procedures appear to impose severe constraints on the plasma configurations. We considered three distinct approaches: a) Field-period based assembly and maintenance, b) modular assembly and maintenance through a small number of designated ports, and c) modular assembly and maintenance through ports between each pair of adjacent coils. It appears that each favors 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. We have developed a cost data basis for components with irregular geometry. 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 second phase of ARIES-CS study.

ACKNOWLEDGMENTS

The work at UCSD was supported under U.S. Department of Energy Grant No. DE-FC03-95-ER54299. 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.

REFERENCES

[1] C, D, BEIDLER *et al.,* Recent Developments in Helias Reactor Studies," *Proc. 13th International Stellarator Workshop,* Canberra, Australia (Feb. 2003). Also, see C.D. BEIDLER, E. HARMEYER, et al., "The Helias reactor HSR4/18," *Nuclear Fusion*, **41**, 12 (2001).

[2] A. SAGARA, *et al.,* "Design and Development of the Flibe Blanket for Helical-Type Fusion Reactor FFHR," *Fusion Engineering & Design*, **49-50**, 551 (2000).

[3] R. MILLER, F. NAJMABADI, *et al.,* "The Stellarator Power Plant Study," University of California San Diego Report UCSD-ENG-004 (1996).

[4] P. GARABEDIAN, L. P. KU, and the ARIES Team, "Physics Basis for the ARIES-CS Compact Stellarator Reactor," *this issue*.

[5] J.F. LYON, L.P. KU, P. GARABEDIAN L. El-GUEBALY, L. BROMBERG, and the ARIES Team, "Optimization of Stellarator Reactor Parameters," *this issue*.

[6] A. R. RAFFRAY, L. EL-GUEBALY, S. MALANG, X. WANG, and the ARIES team "Attractive Design Approaches for a Compact Stellarator Power Plant," *this issue.*

[7] L. EL-GUEBALY, R. RAFFRAY, S. MALANG, J. F. LYON, L.P. KU, and the ARIES Team, "Benefits of Radial Build Minimization and Requirements imposed on ARIES Compact Stellarator Design," *this issue.*

[8] A. R. RAFFRAY, S. MALANG, L. EL-GUEBALY, X. WANG, and the ARIES Team, "Ceramic Breeder Blanket for ARIES-CS," *this issue.*

[9] L. EL-GUEBALY *et al.,* "Initial Activation Assessment for ARIES Compact Stellarator Power Plant," *this issue.*

[10] M. WANG *et al.,* "Three-dimensional Modeling of Complex Fusion Devices Using CAD-MCNP Interface," *this issue.*

[11] X. R. WANG, S. MALANG, A. R. RAFFRAY and the ARIES Team, "Maintenance Approaches for ARIES-CS Power Core," *this issue.*

[12] J. Nuhrenberg, W. Lotz, S. Gori, "Theory of Fusion Plasma," (Varenna 1994), Editrice Composoitiri, Bologna 3 (1994).

[13] G. H. NEILSON, M. ZARNSTORFF, L P. KU, *et al.,* "Physics Considerations in the Design of NCSX," *Proc. 19th International Atomic Energy Agency Fusion Energy Conference, IAEA-CN-94/IC-1,* Lyon, France, (Oct. 2002).

[14] L. P. KU *et al.,* "Development of Compact Quasi-Axisymmetric Stellarator Reactor Configurations: *Proc. 14th Int. Stellarator Workshop and IAEA Technical Meeting on Innovation Concepts and Theory of Stellarators*, Greifswald, Germany (Sept. 2003). Also, Princeton Plasma Physics Laboratory Report, PPPL-3874.

[15] L. P. KU *et al.,* "A Compact Quasi-Axisymmetric Stellarator Reactor," Proc. 20th IEEE/NPSS Symposium on Fusion Engineering, San Diego, California, (Oct. 2003). Also, Princeton Plasma Physics Laboratory Report, PPPL-3886.

[16] L. P. KU, "Progress in Configuration Development for Compact Stellarator Reactors," ARIES project meeting, Madison, WI (June 2004),

[17] F. NAJMABADI, S. C. JARDIN, M. TILLACK, AND L. M. WAGANER, "ARIES-AT: An Advanced Tokamak, Advanced Technology Fusion Power Plant" *Proc. of 2000 IAEA Fusion Energy Conference,* Sorrento, Italy, (October 2000).

[18] F. NAJMABADI and the ARIES Team, "Spherical Torus Concept as Power Plant – The ARIES-ST Study," *Fusion Engineering and Design*, **65**, 2003.