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BENEFITS OF RADIAL BUILD MINIMIZATION AND REQUIREMENTS IMPOSED ON ARIES COMPACT STELLARATOR DESIGN

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It is widely recognized among stellarator researchers that the minimum distance between the plasma boundary and the middle of the coil ($\square_{min}$) is of great importance for stellarators as it impacts the machine parameters considerably. Techniques for minimizing the radial build have made impressive progress during the first year of the ARIES-CS study. A novel approach has been developed for ARIES-CS where the blanket at the critical area surrounding $\square_{min}$ has been replaced by a highly efficient WC-based shield. As a result, an appreciable 20-90 cm savings in the radial build has been achieved, reducing the major radius by more than 20%, which is significant. The economic benefit of this approach is yet to be determined and the added engineering problems and complexity will be addressed during the remaining period of the study. This paper covers the details of the radial build optimization process that contributed to the compactness of ARIES-CS. Compared with previous designs, the major radius of ARIES-CS has more than halved, dropping from 24 m to less than 10 m, making a step forward toward the feasibility of a compact stellarator power plant.

I. INTRODUCTION

After two decades of stellarator power plant studies, it was evident that a new design that reflects the advancements in physics and improvements in technology was needed. To realize this vision, the multi-institutional ARIES team has recently launched a study[1] to provide perspective on the benefits of optimizing the physics and engineering characteristics of the so-called compact stellarator (CS) power plants. The primary goal of the study is to develop a more compact machine that retains the cost savings associated with the low recirculating power of stellarators, and benefits from the higher beta, smaller size, and higher power density, and hence lower cost of electricity, than was possible in earlier studies. Relative to tokamaks, stellarators promise disruption-free, steady-state operation with reduced recirculating power due to the absence of current-drive requirements. Such advantages could be offset, however, by the more complex coil system, nonuniform blanket/shield/divertor configuration, and challenging maintenance scheme.

The ARIES team has explored the potential of two configurations for the ARIES-CS power plant: two field periods (FP) with 16 coils and three field periods with 18 coils. Figures 1 and 2 illustrate both options showing the last plasma closed flux surface and the individual coils that create the compact configuration. The limited space assigned to the internals (blanket, shield, and vacuum vessel) calls for a well optimized, highly compact radial build, particularly at $\square_{min}$ as it controls the minimum major radius and the maximum field at the coil. During the first phase of the study, several blanket/shield systems have been considered employing advanced ferritic steel (FS) structure (such as IEA MF82H) and SiC/SiC composites. The list of candidates includes one solid breeder-based system (Li_4SiO_4/FS/Be/He) and four liquid breeder-based systems (self-cooled Flibe/FS/Be and LiPb/SiC, and dual-cooled LiPb/FS/He and Li/FS/He). The newly proposed dual-cooled Flibe/FS/Be/He system will be examined later. As predicted, each concept offers advantages and drawbacks and an integrated study with guidance from the economic analysis and maintenance scheme will later identify the preferred concept during the second phase of the study.

The nuclear assessment proceeded interactively with guidance from the thermo-mechanical analysis. We started the analysis by examining the breeding capacity of the candidate breeders, and then defined the blanket parameters (thickness, composition, and Li enrichment). Next, the shield was designed to protect the VV and both components were simultaneously sized to essentially protect the superconducting magnets. Finally, we specified the radial build and identified the key nuclear parameters: tritium breeding ratio (TBR), neutron energy multiplication (M_n), radiation damage to structural components and their service lifetimes.

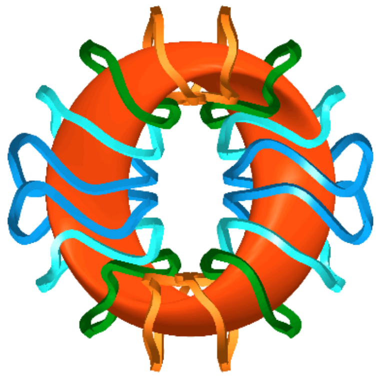

Fig. 1. Two field period design option for ARIES-CS.

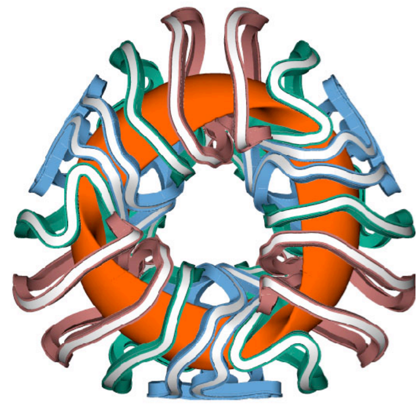

Fig. 2. Three field period design option for ARIES-CS.

II. HISTORIC BACKGROUND

Although the stellarator concept has been around for some time, very little in the way of conceptual design studies has been performed compared to tokamaks, of which many studies have taken place in the US and abroad. During the decade of the 1980s and continuing to the present, six stellarator power plants have been developed. These include UWTOR-M[2], ASRA-6C[3], SPPS[4], and ARIES-CS in the US, HSR[5] in Germany, and FFHR[6] in Japan. The studies varied in scope and depth and encompassed a broad range of configuration options. The 1982 UWTOR-M design[2] has 18 modular twisted coils with only two different coil geometries arranged in a toroidal configuration. The blanket employs FS as the main structure and LiPb for tritium breeding. Initiated in the mid-80s, the ASRA-6C study designed all 30 coils and internal components (FW, LiPb/FS blanket, and shield) to have identical elliptical bore. Next came the Stellarator Power Plant Study (SPPS) initiated in 1995 by the ARIES team to address key issues for stellarators based on the modular helias-like heliac approach. The baseline configuration has four field periods produced by 32 modular coils of four distinct types. Vanadium structure and lithium breeder are the reference materials for SPPS. On the international level, a helias stellarator reactor (HSR) study was initiated in Germany in the late 1990s based on the Wendelstein 7-X experiment that is currently under construction in Greifswald, Germany. The most recent HSR4/18 design[5] has four field periods with 40 coils and LiPb/FS blanket. Alternatively, the stellarator configuration can be produced using continuous helical coils. An example of this approach is the Forced Free Helical Reactor[6] (FFHR) presently under study in Japan. Vanadium structure, Flibe breeder, and beryllium multiplier are the materials of choice for FFHR. Note that all designs developed to date employed liquid breeders (Flibe, LiPb, or Li) for breeding and cooling to cope with the complex geometry of stellarators.

III. DESIGN PARAMETERS AND LIMITS

The limited space for the internals (blanket, shield, and vacuum vessel) calls for a well optimized, highly compact radial build and stresses on the crucial role of the neutronics activity. As such, certain features of these activities focused on a unique area to compact stellarators, mainly the minimum radial standoff between the plasma and magnet. A novel approach has been developed for ARIES-CS where the blanket at the critical area surrounding Δ_{min} has been replaced by a highly efficient WC-based shield. This approach places a premium on the blanket to supply all the tritium needed for plasma operation. For each blanket concept, the nuclear analysis determined Δ_{min} that meets the top-level requirements for the ARIES power plants. These requirements along with the adopted radiation limits and key design parameters are summarized in Table I. The reference design has a power level of 1000 MW_e. The blanket must breed sufficient tritium for plasma operation, recover ~90% of the neutron energy, and protect the shield for the entire plant life. The 10% energy leaking into the high-temperature (HT) shield must be recovered as high-grade heat to enhance the power balance. The blanket and shield help protect the vacuum vessel (VV) and all three components protect the superconducting magnets for life. An alternate design option has been proposed to facilitate the maintenance scheme where the VV is relocated outside the magnet. If so, a low-temperature (LT) shielding component should follow the HT shield to adequately protect the magnet. The nuclear heating deposited in the internal VV (or LT shield) is so low (< 1% of the total heating) to the extent that it could be dumped as low-grade heat. The implication for the design could be significant as coolants with high shielding performance (such as water) could be used for the internal VV and LT shield. The magnet design[7] calls for two consecutive winding packs running at 15 K and 4 K with MgB_2 and NbTi superconductors, respectively. At 15 K, the magnet cooling requirement is greatly relaxed and the cryogenic load is not a concern.

TABLE I. Design Parameters, Requirements, and Radiation Limits

Fusion power	~ 2000 MW
Major radius	8.25 m – 3 FP case
	7.5 m – 2 FP case
Minor radius	1.85 m – 3 FP case
	2 m – 2 FP case
Peak neutron wall loading	~ 3 MW/m^2
Average neutron wall loading	~ 2 MW/m^2
Overall TBR	1.1
Peak damage to structure	200 dpa - FS
	3% burn-up - SiC
He production at VV	1 appm
Magnet:	
Fast neutron fluence	10^{19} n/cm^2
(E_n > 0.1 MeV)	
Peak nuclear heating	5 mW/cm^3
dpa to Cu stabilizer	6x10^{-3} dpa
Dose to GFF polyimide	10^{11} rads
Machine lifetime	40 full power years
Availability	85%

IV. BLANKET DESCRIPTION

The team has examined five blanket systems. A brief description is given here and the accompanying paper[8] should be consulted for the details. The five blankets are:

- Self-cooled Flibe with beryllium multiplier and FS structure,
- Self-cooled LiPb with SiC/SiC composites,
- Dual-cooled LiPb or Li with He and FS structure, and
- He cooled solid breeder (Li_4SiO_4) with beryllium multiplier and FS structure.

For each system, we developed two radial builds: shield-only that fits in the local areas where the radial space between the plasma and magnet is highly constrained and blanket and shield for the nominal areas elsewhere. At this stage of the design, a few design parameters that are essential for the accuracy of the breeding and shielding parameters were missing. In order to move forward with the nuclear analysis, we had to make several educated assumptions. For instance, we assumed the peak to average neutron wall loading is ~1.5, the penetrations occupy 2% of the FW area, and the divertor plates/baffles cover 15% of the FW area. Moreover, we had to rely heavily on the simple 1-D poloidal cylindrical model using an average plasma minor radius of 1.85 m and make use of the DANTSYS code[9] to predict the overall TBR and M_n, combining 1-D estimates with blanket coverage fraction. As the ARIES-CS design progresses, a 3-D analysis is judged essential to confirm the key nuclear parameters and to generate the exact toroidal and poloidal neutron wall loading distribution[10].

IV.A. Self-Cooled Flibe/FS/Be System

The molten salt system always needs a beryllium multiplier to meet the breeding requirement. The Flibe cools the first wall (FW), turns around and flows through the beryllium and breeding zones. The system has a coolant outlet temperature of 700 ^{o}C and a thermal conversion efficiency (η_{th}) of 45%.

IV.B. Self-Cooled LiPb/SiC System

This design utilizes the advanced SiC/SiC composites as the main structural material. The LiPb flows through the SiC structure at a high speed, and then flows slowly within the breeding zone. The high operational temperature of the SiC makes it possible to achieve high LiPb outlet temperature (~1100 ^{o}C). The significance of the high operating temperature lies in the superior thermal conversion efficiency of the system (55-60%).

IV.C. Dual-Cooled LiPb/FS/He and Li/FS/He Systems

The dual coolant option holds the potential to operate the FS-based design at higher temperatures to ensure high η_{th} (~45%). Both LiPb/FS/He and Li/FS/He designs are very similar. The helium cools the FS structure while the LiPb (or Li) flows slowly in the breeding zone. Serving as a liner for the structure, 0.5 cm thick SiC insert must be used to control the MHD effect and maintain the FS temperature below 600 ^{o}C. The impact of the SiC insert on breeding has not yet been assessed.

IV.D. He-Cooled Li_4SiO_4/FS/Be System

Generally, solid breeder (SB) designs fail to achieve tritium self-sufficiency without neutron multiplier. The proposed design[11] 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 SB. This design can handle up to 4.5 MW/m^2 peak neutron wall loading and deliver electricity with η_{th} approaching 45%.

V. BREEDING PERFORMANCE

As will be discussed shortly, the nominal radial build varies widely with the proposed blanket concepts. We sized the blanket to essentially meet the breeding requirement and protect the shield for the 40 full power year (FPY) plant life. Addressing the breeding issue, it is informative to compare the breeding potential of all breeders. Figure 3 shows that the beryllium multiplier makes a difference as the Flibe and SB systems offer the highest breeding, resulting in thin blankets with excess breeding margin. Note that beryllium may raise safety and economic concerns that need further investigation.

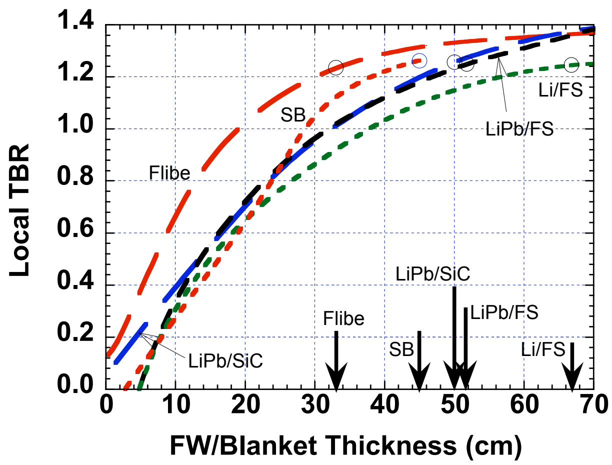

Fig. 3. Breeding potential of proposed blankets.

To achieve an overall TBR of 1.1 (local TBR ~1.25), we recommend 30% enriched Li for the Flibe system, 90% enriched Li for the LiPb system, natural Li for the Li system, and 20-90% enrichment for the multiple layers of the Li_4SiO_4 system. Excluding the back wall (that supports the blanket and serves as He manifolds), the FW and breeding zones of the Flibe/FS/Be, LiPb/SiC, LiPb/FS/He, Li/FS/He, and Li_4SiO_4/FS/Be/He systems should be 33, 50, 51, 67, and 45 cm thick, respectively. The arrows in Fig. 3 point to the recommended thickness of the blanket that satisfies the breeding requirement (overall TBR = 1.1). The analysis assumes that a few blanket modules can be installed behind the divertor plates (5 cm thick with 50/50 FS/He). The 1.1 overall TBR takes into account the 8% losses in the blanket coverage for the shield-only zones that will be discussed shortly. This statement is true for the 3 FP configuration only. A much larger fraction (~20% of FW area) could be dedicated to the shield-only zones of the 2-FP configuration causing a breeding problem. The remaining area occupied by the blanket may not supply enough tritium. A uniform blanket everywhere appears likely to be the most practical option for the 2 FP configuration.

VI. SHIELDING PERFORMANCE

The blanket along with the back wall provides an important shielding function as it protects the shield for the entire plant life (40 FPY). The life-limiting criteria for the FS and SiC structural components have traditionally been the displacement of atoms (200 dpa) and burn-up (3%) of the Si and C atoms, respectively. An additional shielding criterion relates to the reweldability of the VV. The blanket and shield must keep the neutron-induced helium at the VV below the reweldability limit (1 appm) at any time during plant operation. All three components (blanket, shield, and VV) help protect the superconducting magnets and externals. In our shielding analysis, we have considered a safety factor of three to account for the uncertainties in the computational tools and design elements. The selection criteria for the shielding materials included several design parameters that play an essential role in the acceptability of the materials. These are the compatibility with the structure and other components, radiation stability, safety characteristics, and operating temperature windows.

The blanket and shield have been sized to satisfy the design requirements and meet the radiation limits of Table I. There is only one blanket segment in all five designs except for the SiC system where two blanket segments are proposed, 25 cm thick each. The segmentation helps reduce the replacement cost and minimize the waste stream. The outer segment along with the shield is a lifetime component.

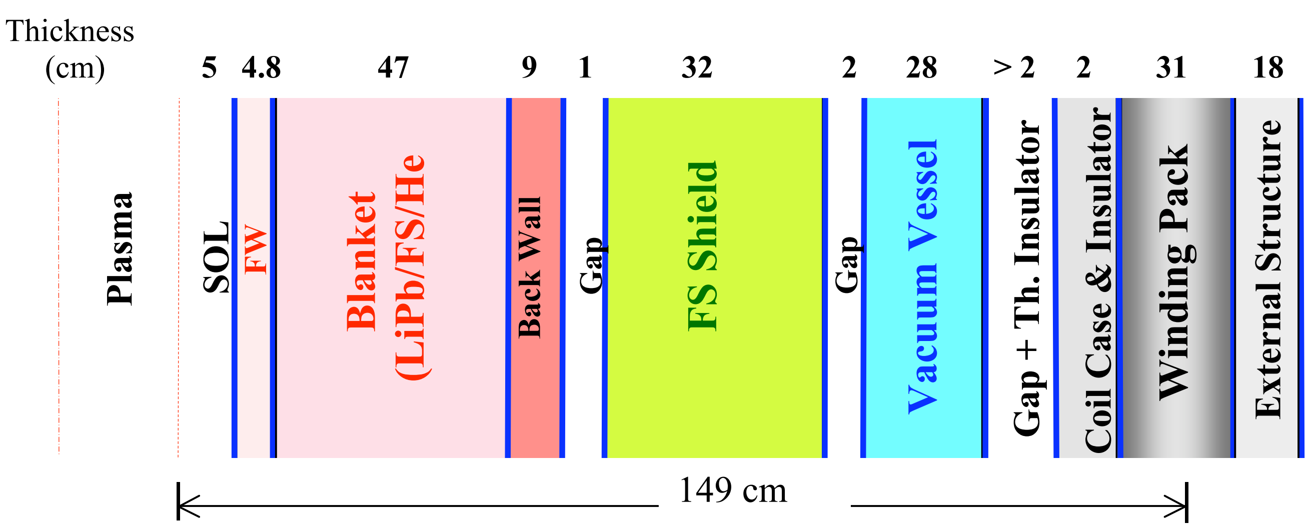

Fig. 4. Schematic of LiPb/FS/He radial build.

A representative radial build for the LiPb/FS/He concept is illustrated in Fig. 4. The reader is referred to the following URL for the composition of all components and the radial build definition for other blanket systems: http://fti.neep.wisc.edu/aries-cs/builds/build.html. Being the closest component to the magnet, the composition of the VV (or LT shield) influences the radiation damage at the magnet significantly. The double-walled VV was filled with shielding materials and optimized to achieve the necessary requirements for magnet protection. Several fillers have been identified for evaluation: water, borated water, FS, and B-FS (FS with 3 wt% B). No structural role has been envisioned for the fillers. Water was considered for its superior shielding characteristics relative to other coolants such as liquid breeders and He gas. In fact, liquid breeders were excluded as the blanket and, to a lesser extent, the shield provide all the tritium needed for plasma operation. Tradeoff analyses of the fillers were conducted for the Flibe, LiPb, and Li_4SiO_4 concepts. The strong dependence of the damage at the magnet on the choice of the VV fillers is displayed in Fig. 5 for the LiPb concept, showing the tradeoff between water and B-FS filler. The optimal 65% water content is driven by the more restrictive limit of the fast neutron fluence. Sandwiched between 3 cm thick face sheets, the central part of the VV consists of 5% FS ribs (dictated by the structural requirements), 65% water, and 30% borated FS filler, by volume. For the alternate design option where an external VV surrounds the magnets, the composition of the LT shielding component could resemble the internal VV described above, except for the Li system where water must be excluded for safety reasons. Only B-FS filler has been considered for the helium-cooled FS shield of the Li system. The use of an alternate hydrogen-based filler (such as zirconium hydride) could save tens of centimeters in the radial build, but may not be cost effective.

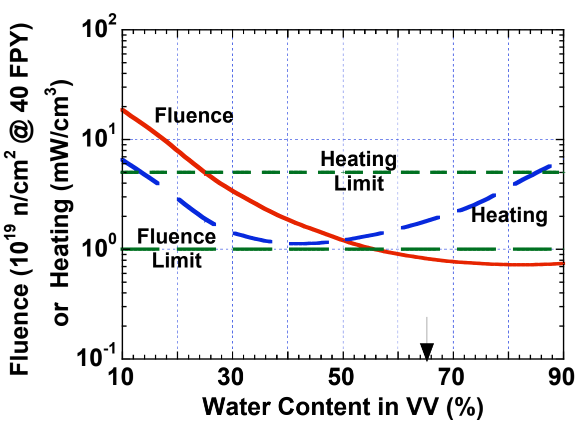

Fig. 5. Sensitivity of peak radiation effects at magnet to VV composition, substituting B-FS for water.

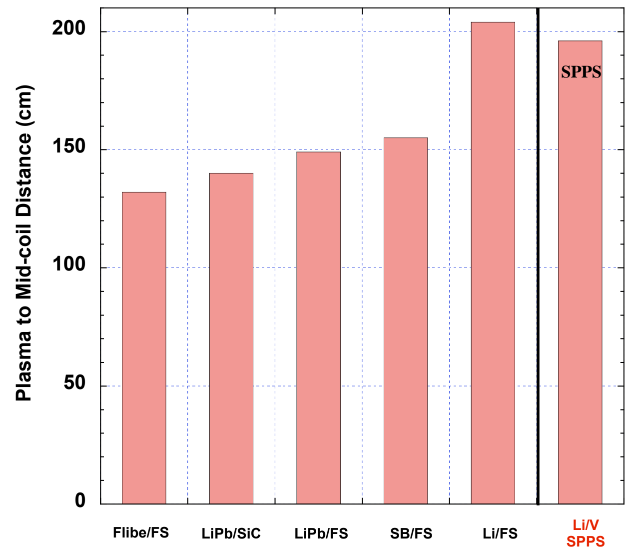

Fig. 6. Comparison of nominal distance between plasma boundary and mid-coil.

Figure 6 compares the nominal distance between the plasma boundary and the mid-coil (□) for the five systems. An appreciable reduction of 40-65 cm in the dimension of the internals relative to the 1996 Li/V SPPS design[4] has been observed for the Flibe and LiPb systems. The superior shielding performance of both breeders compared to Li and the ability to use water in the VV (or LT shield) along with the much thinner magnet contributed to the smaller build. Table II lists the radial dimension of the components of all designs. The alloying and impurity elements of all materials are available at: http://fti.neep.wisc.edu/aries-cs/builds/build.html. The helium coolant occupies 10-30 cm of the radial standoff. As noted, the Flibe system offers the thinnest radial build, followed by LiPb, SB, and Li.

TABLE II. Thickness of ARIES-CS Components

Thickness (cm)	Flibe/ FS/ Be	LiPb/ SiC	LiPb/ FS/ He	SB/ FS/ Be/ He	Li/ FS/ He
FW/B-I	33	25	52	45	67
Blanket-II	--	25	--	--	--
Back Wall	--	--	9	20	8
HT Shield	46	37	32	38	103
VV	25	25	28	26	--*
Total**	104	112	121	129	178

* External VV.

** Excluding assembly gaps.

To further minimize the overall size of the machine, we developed a high-performance, compact shield for the critical area surrounding Δ_{min}. The main idea is to use shielding materials at a few local spots and deploy the nominal blanket elsewhere. For the 3 FP configuration, Δ_{min} occurs twice per FP and the transition region between Δ_{min} and the nominal blanket covers ~8% of the FW area. Looking beyond conventional materials (such as steel, water, and borides), tungsten and its compounds possess superior shielding performance. Tungsten carbide, in particular, offers the most compact radial build when used in the blanket (replacing the breeder) and in the HT shield (replacing the B-FS). While more expensive than the breeder and steel, the material cost difference is not prohibitive for such limited space.

Figure 7 displays the sensitivity of the neutron fluence at the magnet to replacing the breeder of the blanket and B-FS filler of the shield with WC for the LiPb/FS/He system. The analysis assumes that all shields are helium-cooled and the VV dimension (28 cm) and its optimal composition remain fixed. Of interest is the FS case where FS filler replaces the blanket breeder. The WC shield is superior and helps reduce the overall thickness by 30 cm. Considering the positive impact on the overall machine and economics, it pays to incorporate the compact WC radial build at Δ_{min}. The radial arrangement of the shield-only components is shown in Fig. 8. Examining the other four blanket concepts, it is found that Δ_{min} varies within 10 cm (see Fig. 9) as the WC dominates the performance of the shield-only zones. This means switching from one blanket concept to the other would change the major radius of the machine within 1.2 m. A challenging task would be the integration of the local WC-shields with the surrounding blanket system. A blanket with variable thicknesses has been envisioned for the transition region as demonstrated in Fig. 10.

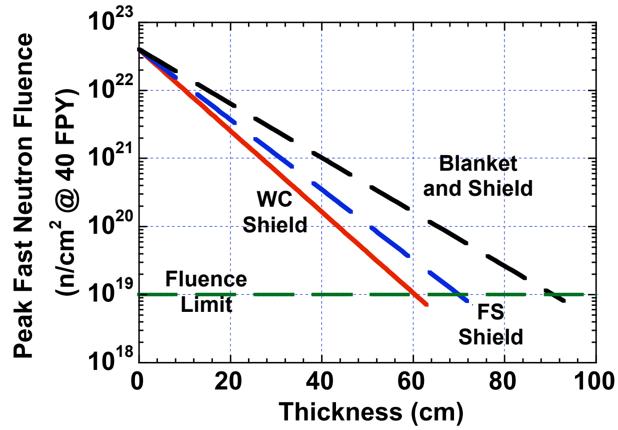

Fig. 7. Sensitivity of magnet damage to shield type.

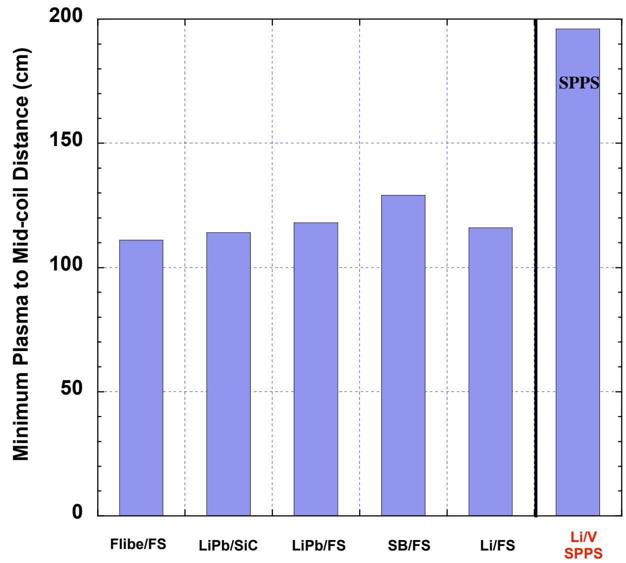

Fig. 9. Comparison between Δ_{min} of the five systems.

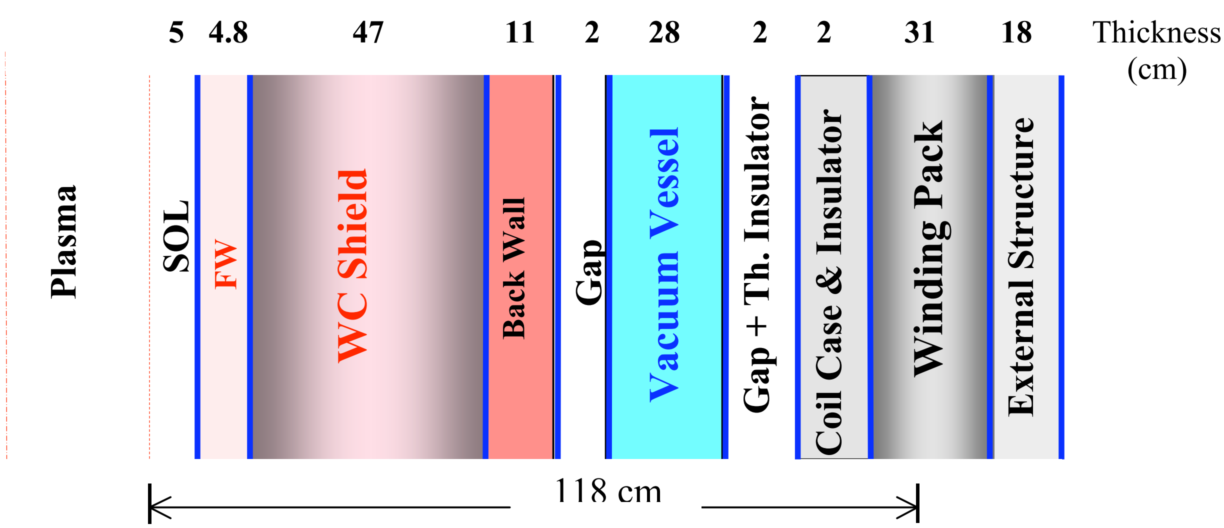

Fig. 8. Schematic of radial build of WC-shield for LiPb/FS/He concept.

TABLE III. Summary of ARIES-CS Parameters

Breeder Structure Multiplier He Coolant	Flibe FS Be	LiPb SiC ---	LiPb FS -- He	SB FS Be He	Li FS -- He
Δ_{min} (m)	1.11	1.14	1.18	1.29	1.16
Nominal Δ (m)	1.32	1.4	1.49	1.55	2.04
Overall TBR	1.1	1.1	1.1	1.1	1.1
M_n	1.2	1.1	1.15	1.3	1.13
FW lifetime (FPY)	6.5	6	5	4.4	7
η_{th}	45%	~55%	45%	45%	45%

Comparing the nominal and minimum radial builds of Figs. 4 and 8, approximately 20-90 cm saving can be achieved by replacing the blanket and FS-shield with WC-shield at the critical area near Δ_{min}. This translates into 20% or more reduction in the major radius, depending on the blanket system.

Table III summarizes the key parameters needed for the systems code to estimate the cost of electricity. Note that the solid breeder system offers the highest M_n with the largest Δ_{min} while the Flibe system offers the thinnest Δ_{min} with a moderate M_n. An integrated economic analysis can self-consistently assess the impact of Δ_{min}, M_n, and η_{th} on the overall cost of the machine.

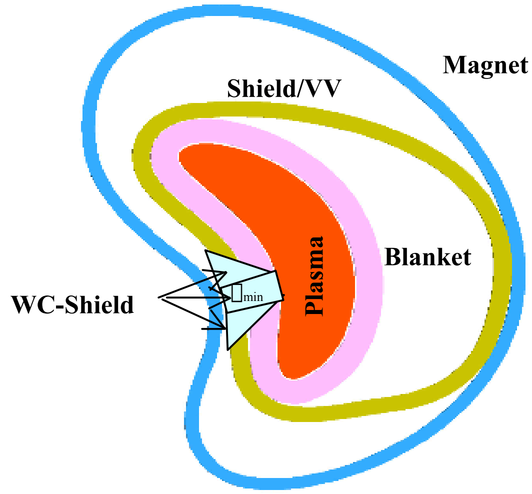

Fig. 10. Layout of shield-only zones and nominal blanket, shield, and VV inside the magnet.

VII. COMPARISON BETWEEN STELLARATOR DESIGNS

The value of the compact feature of ARIES-CS is fully recognized when comparing the major radii of all stellarator power plants developed to date (refer to Section II). The six designs are shown in Fig. 11. Note that over the past two decades, the major radius more than halved by the advanced physics and technology, dropping from 24 m in the early 1980s to 7-8 m for ARIES-CS. Thus, the compact stellarator designs are approaching the advanced tokamak size. Incorporating the latest advancements in physics and technology and means of radial dimension control, ARIES-CS achieved the compactness that other stellarator designs had not been able to accomplish before.

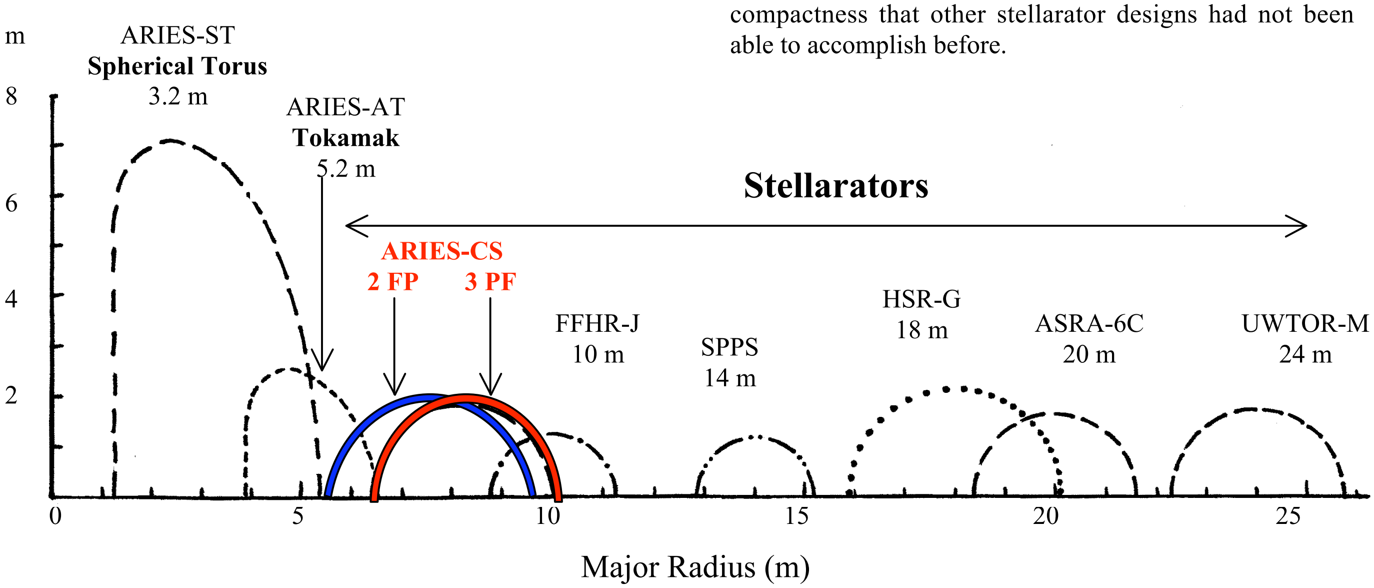

Fig. 11. Major and minor radii of stellarator power plant designs. Typical advanced tokamak and spherical torus are also included for comparison.

VIII. GENERAL REMARKS AND CONCLUSIONS

Developing an advanced stellarator that meets the demanding compactness goal required a collaborative effort between the physicists and engineers. Recently, the ongoing ARIES-CS study demonstrated the potential for compactness, with specific emphasis on plant maintainability[12]. The engineering activities focused on building a compact machine with a lifetime of 45-50 years, a thermal efficiency of 45-60%, and an availability of 80-90%.

All five candidate blanket concepts proposed for ARIES-CS meet the tritium breeding requirement if the blanket covers most of the FW area. A high performance WC-shield has been developed to fit locally in the highly constrained areas where the magnet moves closer to the plasma. The distance from the plasma to the mid-coil varies widely with blanket concepts, while the WC-shield region has much less variability. There will be a non-uniform thickness blanket surrounding the WC-shield zones. The Flibe option provides the thinnest blanket while retaining sufficient breeding margin. Water is a cost effective, efficient shielding material and is highly recommended for the VV and LT components. The engineering and safety limitations of all materials will be an important issue that needs further consideration during the continuing study. An integrated economic analysis will self-consistently assess the impact of the blanket and shield parameters on the overall cost of the machine.

The design benefited substantially from the well-optimized compact radial build that employs WC-shield. This new approach helps minimize the major radius and overall size of the machine. The economic benefit of this approach is yet to be determined. No attempt has been made during the first phase of the study to address the integration issues. Future work present some challenges: integration of the shield-only zones with the rest of the blanket, assessing the need for a separate decay-heat removal loop for the WC-shield, and handling of the relatively massive WC modules during maintenance.

The historic background provided perspective on the impressive benefits of the new shielding approach as well as the advancements in physics and technology. Having 7-8 m major radius approaching that of advanced tokamaks, ARIES-CS is considerably smaller than UWTOR-M[2], the first stellarator power plant developed in the early 1980s. Achieving the compactness goal will certainly improve the economic viability of stellarators. The positive trends in the ARIES-CS physics and engineering are positioning the compact stellarator for a bright future.

ACKNOWLEDGMENTS

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