ATTRACTIVE DESIGN APPROACHES FOR A COMPACT STELLARATOR POWER PLANT

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ABSTRACT

This paper summarizes the results from the engineering effort during Phase I of the ARIES-CS study, covering the different blanket configurations and maintenance schemes that were considered and assessed. The main design parameters are summarized and key issues are discussed including the impact of different physics configurations on the engineering choices. These results will be used as the basis to down-select to a couple of combinations of blanket configuration and maintenance scheme for more detailed studies.

I. INTRODUCTION

The ARIES-CS study has been launched with the goal of developing through physics and engineering optimization a more attractive power plant concept based on a compact stellarator (CS) configuration. On the physics side, the first phase of the study involves scoping out different physics configurations including two and three field period options. Key considerations impacting the design of the CS include the size of the reactor, access for maintenance and the minimum distance between plasma and coil that affects shielding and also breeding if sufficient blanket coverage is not provided.

As illustrated in Figure 1, the Phase I engineering effort, carried out in close interaction with the physics effort, has focused on scoping out maintenance schemes and blanket designs best suited to a CS configuration. This has been done by building on information and results from past studies. The results from this effort will enable a down-selection to a couple of most attractive combinations of blanket configuration and maintenance scheme for more detailed studies during Phase II, which will then culminate in the choice of a point design for a full system design study during the final phase of the study.

To provide a broad range of possibilities to accommodate the physics optimization of the number of coils and the machine size, three possible maintenance schemes were considered: (1) replacement of an integral unit based on a field-period including disassembly of the modular coil system; (2) replacement of blanket modules through a limited number of designated maintenance ports; and (3) replacement of blanket modules through maintenance ports arranged between each pair of adjacent modular coils. Several possible blanket/shield configurations compatible with the maintenance schemes and the CS geometry were considered, covering the following three classes: (1) self-cooled liquid metal blanket with SiC_f/SiC composite as structural material or with He-cooled ferritic steel (with or without thermal/electrical insulation); (2) He-cooled solid or liquid breeder blanket with ferritic steel; and (3) self-cooled flibe blanket with ferritic steel. The divertor heat load for a CS is still uncertain but it is assumed that a He-cooled divertor will be utilized. As guidelines for the first phase of the study, it was decided to develop each concept to an extent sufficient for a credible case to be made regarding performance, fabrication and maintenance.

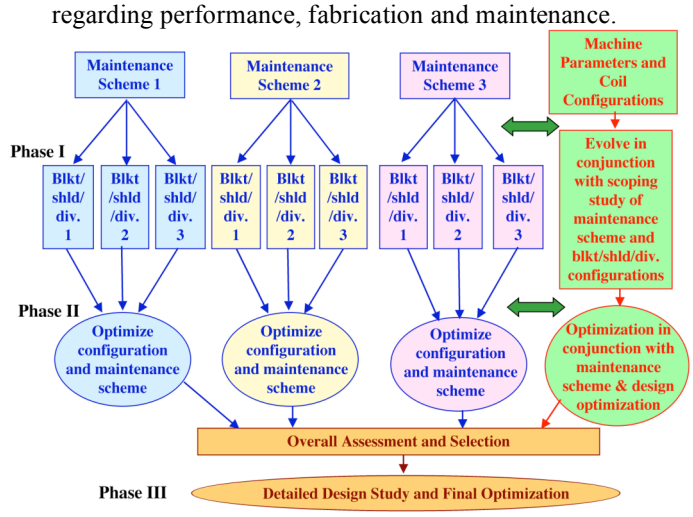

Fig. 1. Three-phase plan for ARIES-CS engineering activities.

This paper summarizes the engineering activities during Phase I of the ARIES-CS study. The example CS configurations used in the initial part of this study are first presented. Next, the maintenance schemes are briefly discussed and the blanket configurations then summarized, with the goal of down-selecting to a couple of combinations of blanket configuration and maintenance scheme for more detailed studies during Phase II of the study.

II. CS CONFIGURATIONS

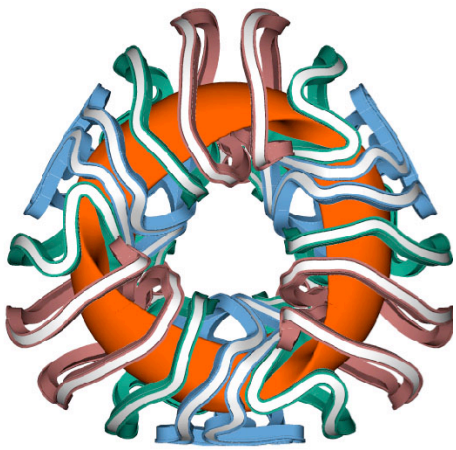

Fig. 2. Example CS configuration with three field periods and 18 coils (based on NCSX).

A major goal of the ARIES-CS study is to help evolve CS configurations that would result in attractive power plants. This requires close interaction among the physics, engineering and system efforts in order to understand the parameter space, the trade-offs and to help in optimizing the whole design. To initiate the engineering effort for the first phase of the study, sets of parameters for two example configurations with three field periods (NCSX-like[1]) and two field periods (MHH2[2]) were developed. The modular coils and the related reference plasma were scaled to a size expected to produce a fusion power consistent with a net power output of 1000 MWe. Clearly, these initial configurations are not yet optimized for a power plant and are only used to provide an initial basis for self-consistent evaluation and comparison for the engineering design activities with the understanding that they will evolve based on our physics and system optimization effort. These two example configurations are illustrated in Figures 2 and 3 and the assumed sets of parameters listed in Table I. For the two-field-period configuration shown in Figure 3, cases with 12 and 8 coils were also considered as the physics effort progressed. A key engineering parameter is the minimum coil-plasma distance which depending of the blanker design might require local regions of shield only and a corresponding tritium breeding loss.

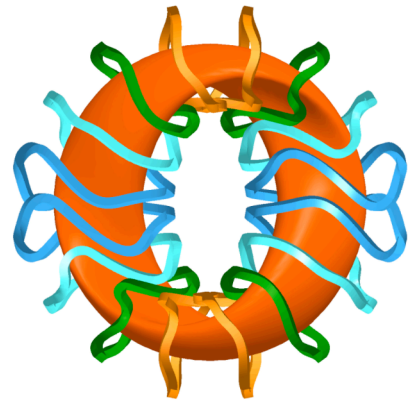

Fig. 3. Example CS configuration with two field periods and 16 coils (MHH2).

TABLE I. Parameters of Initial CS Configurations Assumed for the Engineering Studies

Parameters	3 field periods (NCSX)	2 field periods (MHH2)
Minimum coil-plasma distance (m)	1.2	1.4
Major radius (m)	8.3	7.5
Minor radius (m)	1.85	2.0
Aspect ratio	4.5	3.75
β (%)	4.1	4.0
Number of coils	18	16
Field on axis (T)	5.3	5.0
Max. field at coil (T)	14.4	14.4
Fusion power (GW)	2	2
Average wall load (MW/m^2)	2	2.7

III. MAINTENANCE SCHEMES

The three maintenance schemes considered are summarized below. More details can be found in Ref. [3].

III.A. Field-Period Based Maintenance

This proposed maintenance scheme is based on the same general principles proposed in the ARIES SPPS study[4] as well as in the ASRA6C study[5] performed jointly by IPP Garching, FZK Karlsruhe, and UW Madison. It involves the replacement of entire field period assemblies of blankets, shielding, and modular coils inside an external vacuum vessel (VV). Figure 4 illustrates the ARIES-CS approach for such a scheme for the assumed three-field-period configuration.

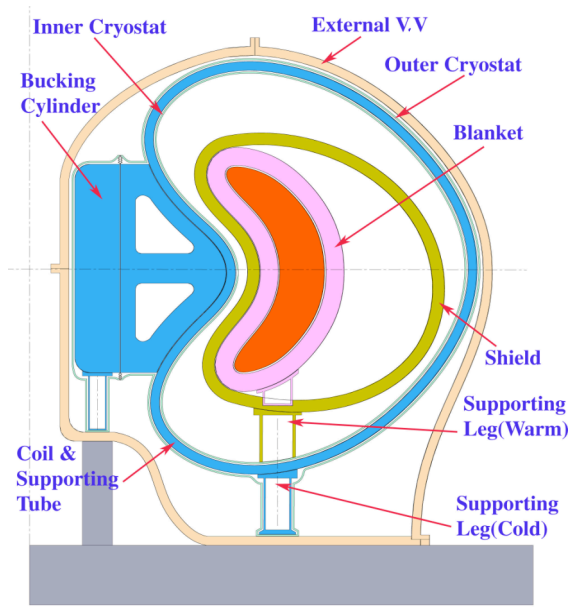

Fig. 4. Cross section of three-field-period configuration at 0° illustrating the layout for field-period based maintenance.

As shown in Fig. 4, the cryogenic modular coils are located inside an external vacuum vessel with a removable outer section. These coils are wound into grooves of a strong supporting toroidal tube, separable at field periods, as illustrated in Fig. 5. The radial forces on the modular coil support tubes are reacted by a strong bucking cylinder, operated at cryogenic temperature; toroidal forces occur within each field period and are reacted by the coil support but there are with no net forces (and no force transfer) between each field period. The blanket in a field period is subdivided into two units that allow toroidal removal and replacement. Temporary walls around the removed vacuum vessel door, field-period power core, and maintenance equipment are arranged to control the spread of radioactivity during blanket replacement. With this maintenance method, there is minimal limitation on the size and weight of the blanket units, the only limitation arising from the lifting and moving equipment. Overhead cranes will probably not be used due to the enclosure limitation and all lateral movements of these large components are assumed to occur on an air cushion.

Blanket replacement proceeds by first warming the coils to a safe temperature for movement. Then the outboard section of the external vacuum vessel is removed exposing one field-period power core assembly. The field-period assembly to be removed is structurally disconnected from the adjacent power core assemblies. This allows a radial movement of the power core assembly to an area for removal of the blanket modules, with a blanket unit removed toroidally in one direction and the other in the opposite toroidal direction. A key factor is the clearance between the stationary shield and the blanket unit when it is being removed toroidally. Increasing the reactor size would increase this clearance when applying this maintenance method to different blanket concepts. However, other measures can also be taken to avoid local interferences such as for example shaving off the shield and increasing the blanket thickness locally.

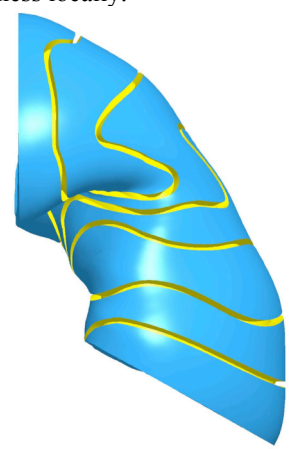

Fig. 5. Coil supporting structure for three-field-period configuration, illustrating the location of grooves for coil winding (it is preferable to wind the coils at the inner surface of the coil structure to better react the outward radial force on the coils).

III.B. Port-Based Maintenance Through a Small Number of Designated Ports

This maintenance scheme involves the replacement of blanket modules through a small number of ports, using an articulated boom for installing, inspecting, removing, and minor repairs of the modules inside the plasma chamber. Typically, there would be one or two maintenance ports per field. Since only the blanket modules are to be moved, this scheme calls for a different vacuum vessel and cryostat design, as illustrated in Figure 6. In this case, the vacuum vessel is internal to the coils and serves as an additional shield for the protection of the coils from neutron and gamma irradiation. No disassembling and re-welding of the vacuum vessel are required for blanket maintenance. 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. The load capacity of the boom is probably limited to about 5,000 kg, which limits the weight and size of the blanket modules and makes this scheme more suited for "lighter" blanket module concepts such as the flibe/ferritic steel and Pb-17Li with SiC_f/SiC concepts described in Sections IV.A and IV.B.

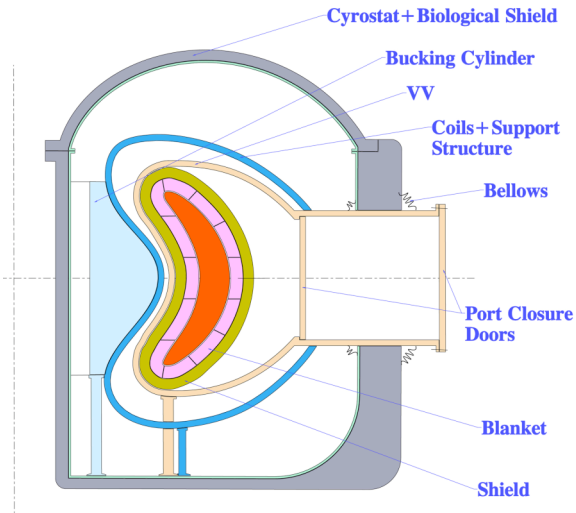

Fig. 6. Layout of 3-field period power core based on modular maintenance though a small number of designated ports

III.C. Port-Based Maintenance With Ports Between Each Pair of Adjacent Coils

This maintenance scheme can be viewed as an extension of the previous modular maintenance scheme but with the replacement of somewhat larger blanket modules through a larger number of ports arranged between each pair of adjacent coils. It also uses an internal vacuum vessel (as illustrated in Fig. 6), and has maintenance ports bridging the region between the vacuum vessel and the external cryostat. Both of these arrangements keep the modular coils at cryogenic temperatures during maintenance and helps contain radioactive contaminants. The significant improvement is that this approach requires shorter booms with much higher load capacities for installing, inspecting, removing and performing minor repairs on the module inside the plasma chamber. The disadvantage is that more ports are required and they are larger in size, which places more geometric constraints on the coil configuration. This maintenance method has been suggested in all IPP Garching Stellarator studies[6] but seems marginal for cases with small port sizes at certain locations, such as for the example three-field-period configuration, as illustrated in Table II.

IV. BLANKET CONCEPTS

As part of the first phase engineering effort, we have studied a number of blanket designs. The guidelines were to develop and analyze each concept to an extent sufficient for a credible case to be made regarding performance, fabrication and maintenance. This would then enable an assessment of all the concepts to help in down-selecting to a couple of concepts for more detailed studies during the second phase of the study. The following blankets concept were evaluated based on past magnetic fusion energy blanket studies (e.g. summarized in Ref. [7]) and potential adaptability to a CS configuration (listed in chronological study order):

1) Self-cooled flibe blanket with advanced ferritic steel (modular maintenance);
2) Self-cooled Pb-17Li blanket with SiC_f/SiC composite as structural material;
3) Dual-coolant blanket concept with He-cooled steel structure and self-cooled liquid metal (Li or Pb-17Li breeding zone) (field period maintenance); and
4) Helium cooled ceramic breeder blanket with ferritic steel structure (modular maintenance)

TABLE II. Port Space (m x m) Available for Example Three-Field-Period and Two-Field-Period Configurations

Configuration	3 field periods (NCSX-like) R=8.3 m	2 field periods (MHH2) R=7.5 m
Port #1	2.3 x 11.0	3.7 x 9.4
Port #2	1.5 x 10.2	3.8 x 8.3
Port #3	1.2 x 5.0	4.0 x 5.1
Port #4	2.0 x 3.0	3.6 x 4.3
Port #5	3.5 x 3.6	4.4 x 4.7
Port #6	2.2 x 10.5	3.7 x 7.4
Port #7		3.7 x 9.4
Port #8		4.4 x 10.2

To facilitate the study, some blanket concepts were evolved in combination with an example maintenance scheme as shown above with the understanding that these combinations are not necessarily exclusive. For example, a molten salt breeder leads to the thinnest blanket/shield zone and the lightest module for a given size, making it an attractive choice for a modular maintenance scheme. However, such a concept could also be used in combination with the field period maintenance scheme. The results of these scoping studies are summarized below for the different concepts; as needed for illustration purposes and in the interest of space, some concepts will be described in more details than others. Some of the major parameters for the different concepts are summarized in Table III. Details of the neutronics analysis including the radial builds and volume compositions are given in Ref. [8].

IV.A. Self-Cooled Flibe Blanket with Advanced Ferritic Steel

A flibe blanket has been considered in previous studies, such as for FFHR[9]. Flibe can be used as breeder and coolant, the self-cooled configuration enabling a simple geometry and high exit temperature, while the low electrical conductivity of flibe eliminates the need

TABLE III. Summary of Key Parameters for the Different Blanket Concepts Considered During Phase I of the ARIES-CS Study

Blanket Concepts Considered During Phase I of ARIES-CS	Self-Cooled Molten Salt	Self-Cooled Pb-17Li	Li Dual-Coolant Concept	Pb-17Li Dual-Coolant Concept	Ceramic Breeder
Breeder (form)	Flibe	Pb-17Li	Li	Pb-17Li	Li_4SiO_4 (pebble bed)
Multiplier (form)	Be(pebble bed)	None	None	None	Be(pebble bed)
Coolant	Flibe	Pb-17Li	He + self	He + self	He
Structure	ODS FS	SiC_f/SiC	RAFS & ODS FS (+SiC insert if required)	RAFS & ODS FS	RAFS & ODS FS
Structure T_{max} (°C)	700	1000	550 (RAFS) 700 (ODS FS)	550	550 (RAFS) 700 (ODS FS)
Breeder T_{max} (°C)	700	1100	800	700	950
Breeder T_{min} (°C)	550	650	500	460	
Multiplier T_{max} (°C)	750				750
Multiplier T_{min} (°C)					
Coolant T_{out} (°C)			He : 500	He : 480	610
Coolant T_{in} (°C)			He: 400	He : 300	400
Coolant P (MPa)	<0.5 (Flibe)	2 (Pb-17Li)	He : 8	He : 14	8
Blanket thickness (m)	0.33	0.5	0.67-0.75	0.52-0.6	0.65
Avg./peak neutron wall load for analysis (MW/m^2)	2/3	2/3	2/3	2/3	3/4.5
Upper limit on neutron wall load (MW/m^2)	3	4-5 (TBD)	4-5 (TBD)	4-5 (TBD)	~5
Surface heat flux (MW/m^2)	0.5	0.5	0.5	0.5	0.5
TBR	≥1.1	≥1.1	≥1.1	≥1.1	≥1.1
Cycle η (%)	~45	~58%	>45	~45	~42
Structural material lifetime and criteria	20 $MW\text{-}a/m^2$ 200 dpa swelling	18 $MW\text{-}a/m^2$ assuming 3% SiC burnup	21 $MW\text{-}a/m^2$ 200 dpa swelling	15 $MW\text{-}a/m^2$ 200 dpa swelling	20 $MW\text{-}a/m^2$ 200 dpa swelling

for insulating coatings to avoid MHD effects. This concept also requires thinner breeding zones for tritium self-sufficiency and shielding compared to other breeder materials, ~30 cm being sufficient to make the zone behind a life-time component. The low pressure of the system (<0.5 MPa) also minimizes the weight of structure making this concept particularly attractive for modular maintenance. A module of size 2m x 2m x 0.3 m weighs less than 1500 kg.

Issues that need to be addressed include the relatively poor heat transfer capability of flibe and its small operating temperature range (with a melting point of 459°C and a maximum compatibility temperature limit of 700°C at the ferritic steel/flibe interface). This means that flibe requires a structural material with a temperature range up to 650-700˚C (or more). In addition, flibe is rather aggressive to a number of candidate structural materials. This requires an excess of beryllium in contact with the flibe in order to stabilize the fluorine in the salt. This contact can be provided either inside the blanket or outside the irradiation environment.

The proposed modular design is illustrated in Figure 7. The coolant is fed to the module from a concentric coolant access tube with the "cold" inlet flow in the annulus and the "hot" outlet flow in the center tube. The entire structure is first cooled and then the flibe is flowed slowly in the large central ducts where it is heated up by the volumetric heat generation to a temperature equal or higher than the maximum interface temperature. With a flibe exit temperature of 700°C, it is believed that a cycle efficiency of > 45% is achievable when coupling a Brayton cycle to the blanket via a heat exchanger.

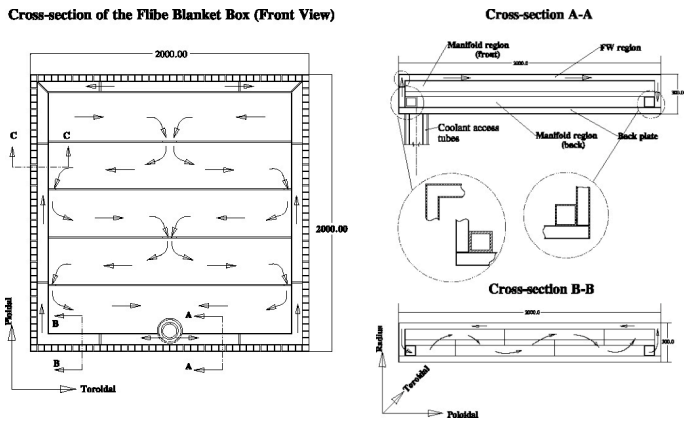

Fig. 7. Schematic cross-section of self-cooled flibe blanket concept.

The breeding capabilities of flibe are limited, making the presence of an additional neutron multiplier mandatory. The design includes a zone of a few cm's of Be pebble bed between perforated plates in the first wall region, serving both as neutron multiplier and for chemistry control of the flibe which flows at low velocity through the pebble bed. This gives rise to Be-specific issues that need to be addressed, including Be swelling under neutron irradiation (10-15 vol.% at end-of-life conditions), and the large tritium inventory in the Be (up to some kg's), which is a safety concern.

Such a self-cooled flibe blanket can only be utilized in combination with advanced ferritic steel that allows for high temperature operation (e.g. oxide dispersion strengthened (ODS) ferritic steel with nano-size oxide particles with operating temperature limit of up to ~800°C[10]). A dual-coolant version of the concept with helium cooling for the steel structure would allow for a more "conventional" reduced activation ferritic steel (RAFS with maximum temperature limit of ~550°C[10]), the use of lower melting point molten salts, and the possible replacement of beryllium multiplier by liquid lead.

IV.B. Self-cooled Pb-17Li blanket with SiC_f/SiC

This is a high-temperature concept utilizing low activation SiC_f/SiC as structural material and Pb-17Li as breeder and coolant, providing the possibility of high cycle efficiency when coupled to a Brayton cycle through a heat exchanger. However, there are key SiC_f/SiC issues that must be addressed, such as fabrication, thermal conductivity and maximum temperature limits.

The CS configuration, shown in Fig. 8, is based on the ARIES-AT blanket design[11]. It consists of two blanket regions, a replaceable one and a lifetime one (in the inboard, only a replaceable region is used); the shield at the back is also a lifetime component. A blanket unit consists of a number of submodules with an annular configuration; in the case of a modular design, a typical blanket module of size ~2 m x 2 m x 0.25m would include about 10 such submodules and would only weigh ~ 500-600 kg when empty. Each module is attached at the back with bolts and shear keys are used to take the load (except for modules at the top of the reactor chamber).

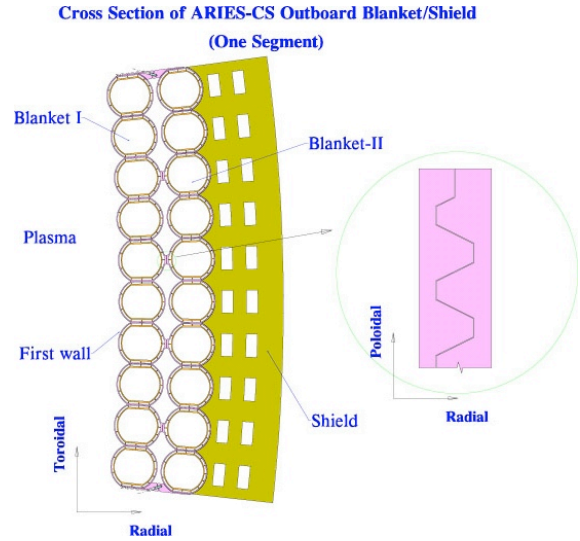

Fig. 8. Schematic cross-section of self-cooled Pb-17Li + SiC_f/SiC blanket concept.

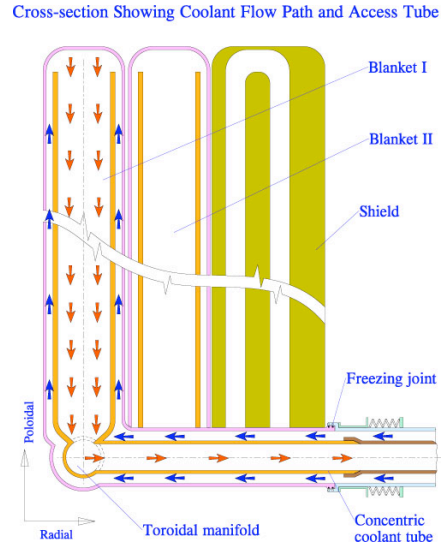

Fig. 9. Schematic of self-cooled Pb-17Li + SiC_f/SiC blanket submodule showing the coolant flow configuration.

As illustrated in Fig. 9, each submodule consists of a simple annular box through which the Pb-17Li flows in two poloidal passes. The first pass is a high-velocity flow through the annular channel region keeping the box walls cooled. The coolant then turns and flows very slowly (~0.05 m/s) as a second pass through the large

inner channel from which the Pb-17Li exits at high temperature. This flow scheme enables operating Pb-17Li at a high outlet temperature (up to 1100°C) for high cycle efficiency while maintaining the SiC_f/SiC composite and the SiC/Pb-17Li interface at a lower temperature (~1000°C) dictated by swelling and compatibility considerations.

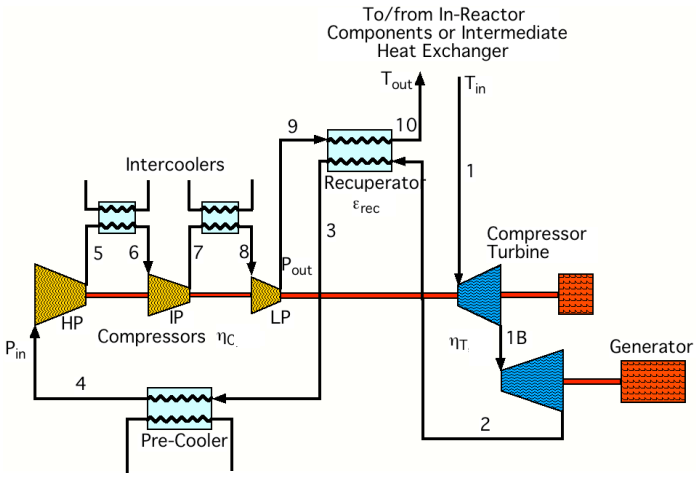

Fig. 10. Brayton cycle configuration assumed in the study[12].

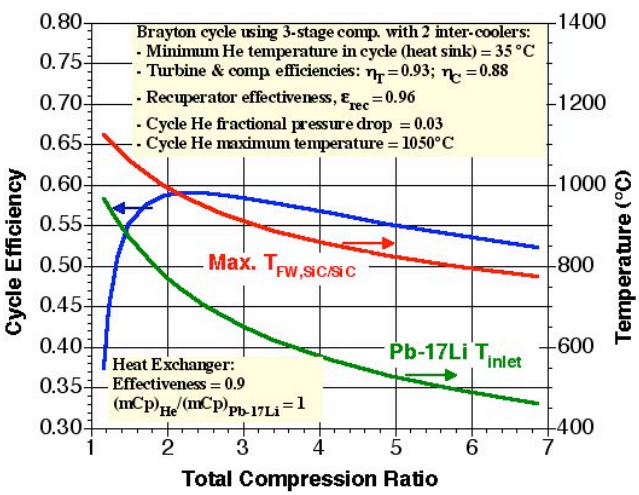

Fig. 11. Cycle efficiency as a function of compression ratio for example Brayton cycle.

A Brayton cycle (see Fig. 10) is chosen to maximize the potential gain from high temperature operation of the Pb-17Li that, after exiting the blanket, is routed through a heat exchanger with the cycle He as secondary fluid. The total compression of the cycle can be fine-tuned to optimize the cycle efficiency and set the minimum He cycle temperature (and in turn the Pb-17Li inlet temperature to the blanket), as shown in Figure 11 for a case with a maximum cycle He temperature of 1050°C (assuming a 50°C temperature difference in the heat exchanger). For example, for Pb-17Li inlet and outlet temperatures of 700°C and 1100°C, the maximum SiC_f/SiC temperature in the first wall is ~970°C (less than the 1000°C limit) and the cycle efficiency is ~58%. The corresponding temperature distribution in a module is shown in Figure 12. The flow enters the thin first wall channel at the (0,0) coordinate location, flows poloidally up about 2 m and then turns and flows slowly poloidally down through the large inner channel. For this case, the maximum SiC/Pb-17Li interface temperature is ~900°C which has not been clearly demonstrated as acceptable based on compatibility limits. This is a key R&D issue that needs to be further investigated.

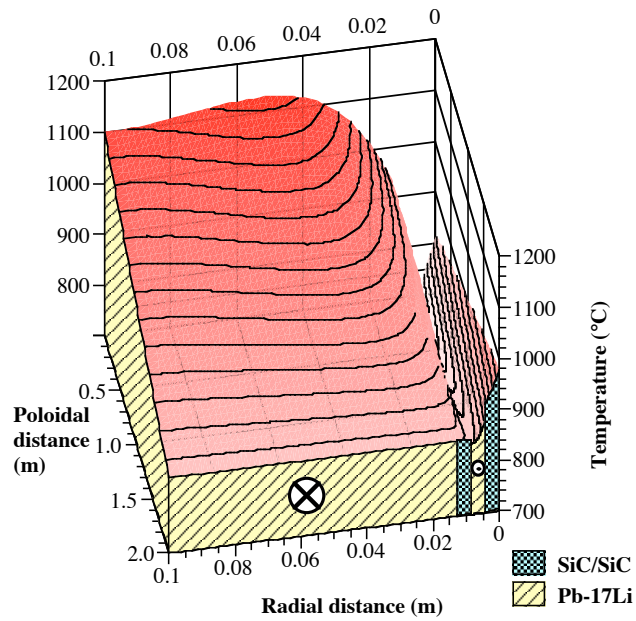

Fig. 12. Temperature distribution in the Pb-17Li blanket submodule.

IV.C. Dual-Coolant Blanket Concept with He-Cooled Steel Structure and Self-Cooled Liquid Metal

Liquid metal cooling of the first wall in self-cooled blankets would be feasible only with electrical insulating coatings to mitigate MHD effects. The feasibility and integrity of such coatings are very questionable. A dual coolant (DC) concept with He as first wall coolant and the liquid metal as blanket coolant would be preferable on this basis. Either of the two major liquid metal breeders (Li or Pb-17Li) can be used. Pb-17Li has slightly better breeding properties, and is less reactive with water or air, but its heat transport properties, specifically its thermal conductivity and heat capacity, are lower than those of lithium which has also a better compatibility with ferritic steel (allowable interface temperature up to ~600 °C compared to < 500°C for Pb-17Li[10]). Past designs such as the ARIES-ST[13] and FZK DC[14] blanket concepts have utilized Pb-17Li. These concepts included a He-cooled first wall

and box structure and a self-cooled Pb-17Li breeding zone. Flow channel inserts made of SiC_f/SiC composite are arranged in the large liquid metal ducts and serve as thermal insulator between the helium-cooled steel walls and the slow flowing Pb-17Li. A possible way to increase the maximum operating temperature of such designs is to plate the first wall with a thin layer (a few mm's in thickness) of ODS ferritic steel [14]. This restricts the use of ODS ferritic steel to zones where structural temperatures > 550°C are desired. At all other places the temperature is limited to values < 550°C for compatibility reasons.

In applying this configuration to the ARIES-CS study, it was decided to look at the possibility of using Li since the use of Pb-17Li in this context is well documented. Such a design is quite thick (and heavy) to provide the required breeding, (~67 cm in the case of Li and ~52 cm in the case of Pb-17Li[8]). In this sense it is probably better suited for a field period maintenance scheme, which has been assumed for the scoping analysis. However, it can also be used with a modular maintenance scheme, albeit with rather small modules.

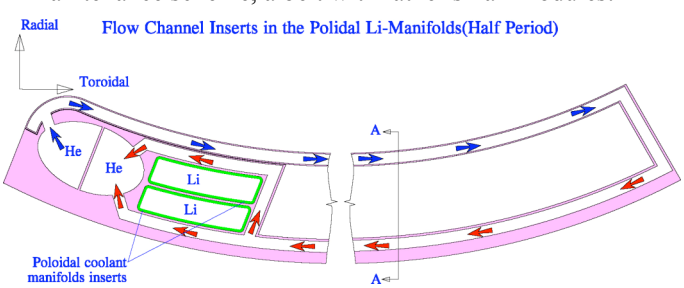

Fig. 13. Schematic of Li/He DC concept blanket unit

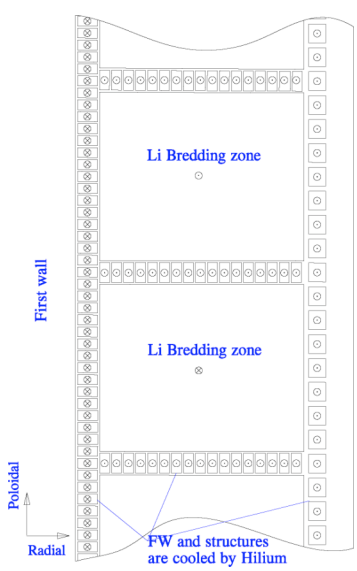

Fig. 14. Inner channel details for Li/He DC blanket concept.

The developed DC Li/He concept with ferritic steel as structural material does not need an additional neutron multiplier (in contrast to ceramic breeder and flibe blankets for example). Again, volumetric heating of the breeder/coolant provides the possibility to set the coolant outlet temperatures beyond the maximum structural temperature limits. The example concept is shown schematically in Figures 13 and 14. The first wall and the entire steel structure are cooled with He (with inlet pressure of 8 MPa and inlet/outlet temperatures of 400/500°C). Li is flown slowly (~0.12 m/s) in the toroidal direction (parallel to the major magnetic-field component) to minimize MHD pressure drop (~0.1 MPa). An electrical insulating coating between Li and ferritic steel is not required but a thermal insulating layer might be needed to maintain the Li/ferritic steel interface temperature within its limit (~600°C). The inlet and outlet temperatures of Li are 500°C and 800°C, respectively. The blanket is coupled to a Brayton cycle (similar to the example configuration shown in Fig. 10) through a heat exchanger with a cycle efficiency >45%.

IV.D. He-Cooled Ceramic Breeder Blanket with Ferritic Steel

A ceramic breeder blanket design has traditionally been coupled to a Rankine steam cycle (e.g. the EU FZK design[15]). However, safety concerns about possible steam/Be interaction in case of accident has led to high-pressure module design translating into more structure and less tritium breeding. To avoid this issue, it was decided to reconsider the possibility of coupling a Brayton cycle to such a blanket, by optimizing the cycle as well as by maximizing the coolant temperature through limited utilization of oxide-dispersion strengthened ferritic steel in high temperature regions. The blanket module is then designed to accommodate a relatively low pressure of ~1 MPa compared to a blanket He coolant pressure of ~8 MPa. A modular design (compatible with port-based maintenance) was assumed in line with past CB designs as it also provides the flexibility of setting the module sizes best suited to the particular reactor geometry. The design and analysis of this concept are described in detail in Ref. [16]. Here the major features and parameters are summarized.

The overall configuration consists of a number of CB and Be multiplier packed bed layers separated by cooling plates and arranged in parallel to the first wall, as illustrated in the cross-section view shown in Figure 15. Pebble beds with packing fractions of ~62% are assumed. Lithium ortho-silicate (Li_4SiO_4) is selected as ceramic breeder, with lithium titanate (Li_2TiO_3) as a possible alternative. The He coolant is first routed toroidally through the first wall cooling plate in alternating directions and then through a series of 3 toroidal passes in the blanket regions, each pass

consisting of parallel-flow routing through several cooling plates. The blanket box and inlet manifold are built of reduced activation ferritic steel with a maximum allowable temperature of 550°C. The first wall can be plated with a layer of ODS ferritic steel for higher temperature operation.

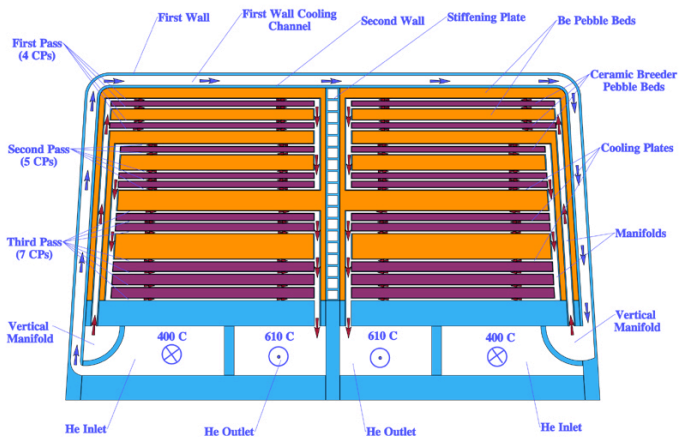

Fig. 15 Cross-section view of a ceramic breeder blanket module.

The number and thicknesses of the Be and CB regions (arranged in parallel to the first wall) were optimized for a tritium breeding ratio of 1.1, yielding a total blanket module thickness of 0.65 m. Constraints for the analysis included the maximum temperature limit of the ferritic steel as well as maximum CB and Be temperature limits of 950°C and 750°C, respectively[15,16]. Scoping studies of this blanket coupled to a Brayton cycle were performed. The results indicate the possibility of accommodating neutron wall load of up to 5-5.5 MW/m^2 and a surface heat flux of 0.5 MW/m^2 with corresponding cycle efficiencies of up to 42% (for a maximum ferritic steel temperature limit of 700°C) for a Brayton cycle with 3-stage compression and one-stage expansion (shown in Fig. 10). The peak efficiency is ~44%, corresponding to a wall load of ~3 MW/m^2. The maximum ferritic steel temperature limit in the first wall makes it very challenging to accommodate higher surface heat fluxes.

V. DISCUSSION AND CONCLUSIONS

The engineering effort during Phase I of the ARIES-CS study has provided a good understanding of a range of possible maintenance schemes and blanket concepts, highlighting their key features and parameters when applied to a compact stellarator (see Table III). Based on the information available, a purely quantitative comparison of these different options in order to select a couple of maintenance and blanket combinations for more detailed studies during Phase II would be a very ambitious task. Instead, it seemed more reasonable to proceed with the down-selection based on a balance of quantitative and qualitative measures while striving to maintain some diversity.

In the area of CS maintenance, it seems healthy to maintain two options. Field period replacement should be selected as one of the two candidates because it has a high potential for an attractive power plant. However, this approach implies huge components and a large extrapolation of the maintenance technology with precision placement. The other maintenance option should be the replacement of relatively small modules through a limited number of ports (perhaps 1 or 2 per field period) with the use of articulated booms. Maintenance through ports between each adjacent pair of coils is really a subset of this maintenance scheme but whose advantages seem minimal compared to the additional complexity imposed on the design of the machine.

In regard to the down-selection of blanket concepts, some key observations can be made:

a) The need for large heat transfer surfaces inside the blanket makes CB blankets always more complicated than self-cooled liquid breeder zones. This tends to be a burden on reliability and lifetime, and is associated with higher fabrication and material costs. In addition, a thicker, heavier breeding zone would be required with this breeder material, while, as illustrated in Table II for an assumed Brayton cycle, the cycle efficiency tends to be lower than that of a liquid breeder blanket. On this basis, CB blankets would be less attractive than liquid breeder blankets.

b) Molten salts in general have poor heat transfer performance and self-cooled concept would limit the surface heat flux and wall load that could be accommodated. In addition, a self-cooled flibe blanket with a melting point of 459°C is not feasible with regular ferritic steel and the performance of a self-cooled low melting point molten salt blanket would be rather low. On this basis, a dual coolant concept with He as first wall coolant would be preferable for molten salt breeders.

c) Helium cooling will be needed most probably for the divertor target plates in a compact stellarator (to be fully studied as part of Phase II of the ARIES-CS study). The additional use of this coolant for the first wall and module structure in dual coolant blankets facilitates the pre-heating of the blankets before the liquid breeder is filled in, serves as guard heating in case the liquid breeder cannot be circulated, and provides independent and redundant afterheat removal in case the liquid metal loop is

not operational. All this reinforces the selection of a dual-coolant liquid breeder concept with He as first wall coolant.

d) The choice of very advanced materials such as SiC_f/SiC composites or nano-size particles ODS-ferritic steels as structural material for blankets represents a high pay-off high-risk option, which deserves to be further studied as a second option during Phase II of ARIES-CS.

These considerations point to a down-selection to the following concepts for more detailed studies during Phase II: (1) DC blanket with a self-cooled liquid breeder zone and He-cooled RAFS structure; and (2) self-cooled Pb-17Li blanket with SiC_f/SiC composite as structural material. In principle, these concepts could all be developed in combination with either a field-period-based maintenance scheme or a port-based maintenance scheme, although for the self-cooled Pb-17Li + SiC_f/SiC option, fabrication constraints on the size of the blanket unit and the low density of the structural material makes it more amenable to a modular concept (port-based maintenance).

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