



Major integration issues in evolving the configuration design space for the ARIES-CS compact stellarator power plant[☆]

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Abstract

The ARIES-CS study has been launched with the goal of developing through physics and engineering optimization an attractive power plant concept based on a compact stellarator configuration. The first phase of the study involved scoping out different physics configurations including two and three field period options. The engineering effort during that phase aimed at scoping out maintenance schemes and power core designs best suited to a compact stellarator configuration. This led to a down selection of the most attractive blanket configurations and maintenance schemes for more detailed studies during the second phase of the study. This paper summarizes early results from the second phase of the ARIES-CS study with a particular emphasis on the engineering effort.

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1. Introduction

In a stellarator, the confining field is produced by the external coils, which generate the poloidal field as well as the bootstrap current. The absence of a large externally-driven current make stellarators inher-

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ently steady state (with low recirculating power), stable against external kink and axis-symmetric modes, and resilient to plasma disruptions. These advantages need to be balanced against the engineering impact of the complicated coil geometry and the irregular cross-section of the plasma and in-reactor components. Earlier stellarator power plant studies led to devices with large sizes, with major radius up to ~ 20 m (e.g. Helias [1], FFHR [2] and SPPS [3]). Recent plasma/coil configurations with lower plasma aspect ratio have the potential of even smaller devices (compact stellarators). 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.

A detailed and integrated study of compact stellarator power plants, ARIES-CS, was launched to advance our understanding of attractive compact stellarator configurations and to define key R&D areas [4]. The study aims to benefit from the advancement in physics and engineering which would lead to a more compact machine retaining the attractive features of a stellarator while potentially benefiting from higher beta, smaller size, higher power density and lower alpha losses than was possible in previous studies. The first phase of the study was devoted to the initial exploration of physics and engineering options, requirements, and constraints. Several compact stellarator configurations such as quasi-axis-symmetric and quasi-helical were considered. In each case, trade-offs among plasma parameters (e.g. α -particle loss, β) were explored and possible coil topologies were studied 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. The engineering effort during that phase was aimed at scoping out maintenance schemes and power core designs best suited to a compact stellarator configuration. This led to a down selection of the most attractive blanket configurations and maintenance schemes for more detailed self-consistent analysis and optimization during the second phase of the study, which should help identify critical high-leverage areas for compact stellarator research. This is an integrated effort with strong interaction between the physics, engineering and

system activities to make sure that design constraints and requirements are taken into consideration when evolving the design space. One of the promising configurations chosen in phase II will be used for a detailed and self-consistent design study in phase III.

This paper summarizes early results from the second phase of the ARIES-CS study with a particular emphasis on the engineering effort. The major findings and results from phase I of the study are first summarized; the selected blanket design options and maintenance schemes for phase II are then described including a discussion of the key assembly and maintenance issues; next, the initial work on the divertor is highlighted; finally, the general observations from this stage of the study and the next step effort are summarized.

2. Phase-I effort

Several quasi-axis-symmetric configurations were explored during the first phase of the ARIES-CS study, including the example configurations with 3 field periods (NCSX-like [5], as illustrated in Fig. 1) and 2 field periods (MHH2 [6]). For the two-field-period configuration, cases with 12 and 8 coils were also considered. To initiate the engineering effort for the first phase of the study, sets of parameters for these two example configurations were developed and are summarized in Table 1. 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. These initial configurations were only used to provide an initial basis for self-consistent evaluation

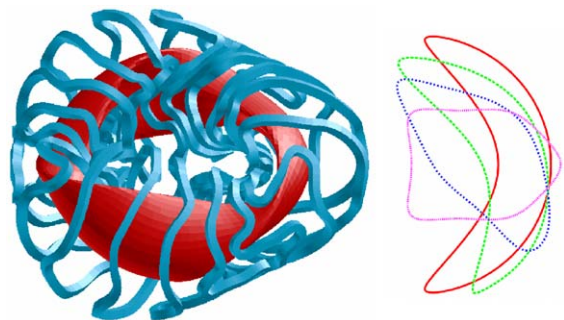


Fig. 1. Example CS configuration with 3-field periods and 18 coils (based on NCSX). Cross-sections of last closed magnetic surface at different toroidal angles are also shown.

Table 1
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
Maximum field at coil (T)	14.4	14.4
Fusion power (GW)	2	2
Average wall load (MW/m ²)	2	2.7

and comparison for the engineering design activities with the understanding that they will evolve based on our ongoing physics and system optimization effort [7,8]; recent results are very encouraging indicating the possibility of minimizing the alpha particle loss to <10% for these reactor sizes. The current modeling effort is directed at determining where these alpha particles deposit their energy.

A key engineering parameter is the minimum coil-plasma distance which, depending on the blanket design, might require local regions of shield only and a corresponding tritium breeding loss [9], and, ultimately could limit the minimum reactor size. Another constraint on the minimum reactor size is the neutron wall loading; typically, blanket designs can accommodate maximum wall load of about 4.5–5 MW/m². When including the peaking factor (~ 1.6 [10]) the corresponding average wall load is about 2.5–3 MW/m², which limits the minimum major radius of the machine for a given configuration.

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 [11]: (1) replacement of integral unit(s) based on a field-period including disassembly of the modular coil system; (2) replacement of blanket modules through maintenance ports arranged between each pair of adjacent modular coils; and (3) replacement of blanket modules through a limited number of designated maintenance ports. Several possible blanket/shield configurations compatible with the maintenance schemes and the CS geometry were

considered: (1) self-cooled liquid metal blanket with SiC_f/SiC composite as structural material; (2) dual coolant liquid metal (Pb–17Li or Li) blanket with He-cooled ferritic steel; (3) He-cooled ceramic breeder blanket with ferritic steel; and (4) self-cooled flibe blanket with ferritic steel. 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. To help in the blanket analysis and assessment during phase I, it was assumed that the divertor would be He-cooled; however, it was decided to carry out the detailed divertor design and analysis during phase II in anticipation of more information on machine geometry and possible divertor location and heat load. Details of the engineering effort and of the supporting neutronics analyses are provided in Refs. [9–14].

A purely quantitative comparison of the different blanket and maintenance concepts and combinations of them would have been a very ambitious task for down-selecting at the end of phase I. It was decided that the down-selection to a couple of combinations be done based on a balance of quantitative and qualitative measures in combination with engineering judgement while striving to maintain some diversity [12]. It was deemed healthy to maintain two maintenance schemes for the second phase to provide more flexibility in optimizing the physics and system configurations also. Thus, both field period replacement and port-maintenance through a selected number of ports were maintained as possible schemes. The following blanket options were selected for the more detailed studies of phase II based on a combination of providing a blanket with reasonable performance and moderate development risk as a first option and a higher-performance, higher-development risk blanket as an alternative:

1. Dual Coolant concept with self-cooled Pb–17Li zones and He-cooled ferritic steel structure (with the possibility of limited use of ODS ferritic steel in high temperature regions). A low conductivity SiC layer is used to thermally and electrically insulate the Pb–17Li zones.
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 make it more amenable to a modular concept (port-based maintenance).

The initial phase-II effort focused on evolving in more engineering detail the dual coolant modular concept and in better assessing the maintenance requirements (including assembly and removal) of this concept. In addition, a focused effort on the divertor was started to develop a potentially attractive concept in integration with the blanket (a parallel physics modeling effort is aimed at helping to determine the most appropriate divertor location and corresponding heat loads). These are summarized below.

3. Modular dual coolant concept

The dual coolant concept utilizes He to cool the ferritic steel structure (including the first wall) and slowly flowing Pb–17Li as self-cooled breeder in the inner channels, which can be operated at a higher temperature than the structural blanket walls to maximize the power cycle efficiency. Use of He coolant is compatible with the coolant strategy for the reactor since He

will be needed in any case for cooling the divertor in a compact stellarator. The use of this coolant for the first wall/structure in dual coolant blankets also facilitates the pre-heating of the blankets before the liquid breeder is filled in, serves as guard heating in case the liquid breeder can not be circulated, and provides independent and redundant decay-heat removal in case the liquid metal loop is not operational. In addition, cooling the first wall region of the blanket (where the heat load is highest) with helium (instead of a liquid metal) avoids the need for an electrically insulating coating in this high velocity region to prevent the large MHD pressure drop associated with liquid metal flow.

Such a dual coolant concept was originally developed as part of the ARIES-ST study [15] and then at FZK in Germany [16]. Recently, it has also been considered as a US test module candidate for ITER [17]. A modular concept was adapted for the ARIES-CS compact stellarator geometry with a particular focus on developing a more efficient coolant routing configuration and on optimizing the design performance of the blanket coupled to a Brayton power cycle [18]. The design concept is illustrated in Fig. 2, with both He and Pb–17Li fed to the back of the blanket module through concentric pipes. The helium coolant (assumed at 8 MPa) is routed to first cool the first wall in a single

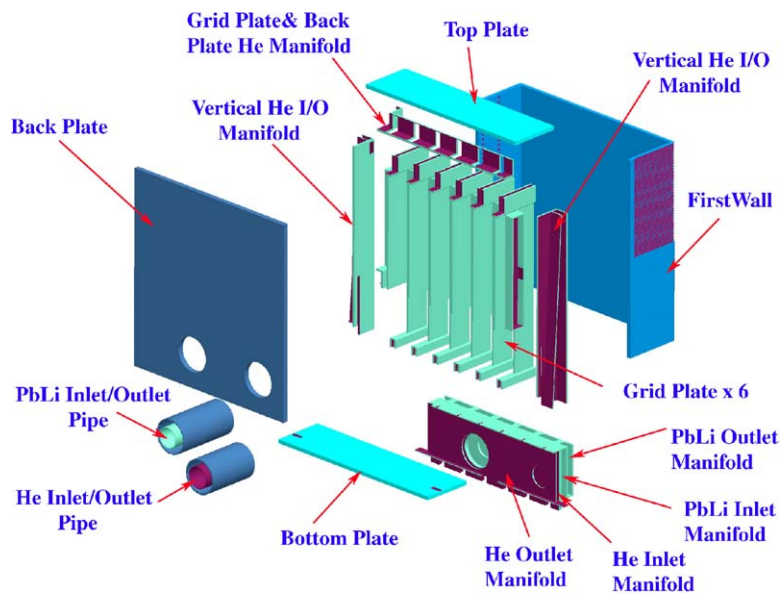


Fig. 2. Exploded view of dual coolant blanket module design for ARIES-CS.

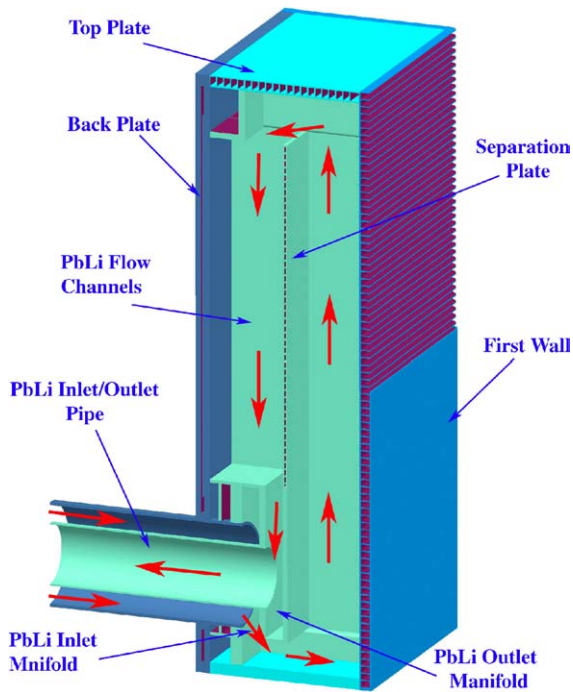


Fig. 3. Illustration of part of the Pb–17Li routing.

pass with an alternating toroidal flow configuration to create a more uniform temperature (and reduce thermal stresses); it is then routed in a combination of series and parallel flow to cool all other structural walls. The Pb–17Li flows slowly (~ 10 cm/s or less) in the large inner channels (of typical size ~ 0.25 m \times ~ 0.25 m) in a two-pass poloidal configuration, as illustrated in Fig. 3. The Pb–17Li channels are lined with a SiC insulating layer (with no structural function as the thin Pb–17Li layer and the bulk Pb–17Li are pressure-balanced through a thin slot in the SiC), as illustrated in Fig. 4. This layer plays a key thermal insulation function to allow high temperature (~ 700 °C) Pb–17Li in the channel while the Pb–17Li/FS interface temperature is maintained below its compatibility limit. The typical module size is about 2 m by 2 m with a radial thickness of 0.6 m required for sufficient tritium breeding (i.e. for a 3D tritium breeding ratio of 1.1 [9]). Note that in regions of minimal available space between coil and plasma, it is possible to use a much thinner shield-only design (e.g. 1.18 m instead of 1.49 m required for blanket + shield for the 3-field period example case [9]).

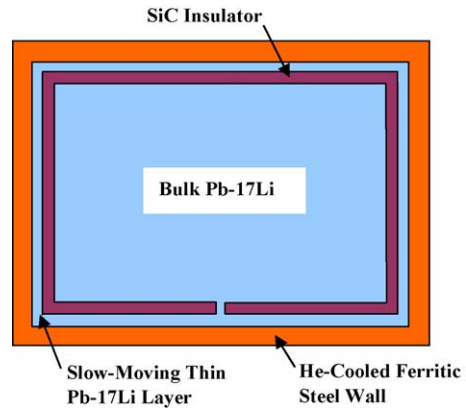


Fig. 4. Schematic of SiC layer in inner Pb–17Li channel.

The local loss of breeding can be adequately made up elsewhere.

Reduced activation ferritic steel is utilized as structural material, with a maximum temperature limit of ~ 550 °C based on thermal creep considerations [19]. A more constraining limit is the interface temperature between the thin Pb–17Li layer and the FS wall (see Fig. 4). A compatibility limit of 450 °C [19] has been assumed for quite some time for FS/Pb–17Li based on an old data set, which did not reproduce correctly the prototypical blanket conditions (e.g. MHD laminarization effect on the flow was not considered and a uniform 20 μ m/year corrosion limit was assumed based on plugging concern). Clearly, a focused experimental and modeling effort is required to derive more appropriate limits based on conditions correctly simulating the prototypical flow conditions as well as a better understanding and appreciation in developing a corrosion limit. For example, it might be acceptable to maintain the bulk of the interfaces below a certain limit but to allow for higher temperatures in very limited zones. It was decided in anticipation of such results not to overly penalize this blanket design and to assume that the interface temperature between FS and Pb–17Li could be up to ~ 500 °C in limited regions. Use of the SiC insulator in the return inner channel of the concentric coolant access pipes would also allow the use of FS for the coolant piping; however, the heat exchanger tubes would have to be made of higher temperature material (most probably a refractory alloy, such as niobium or tantalum). Use of niobium or tantalum alloys, which have a high tritium permeability, is also con-

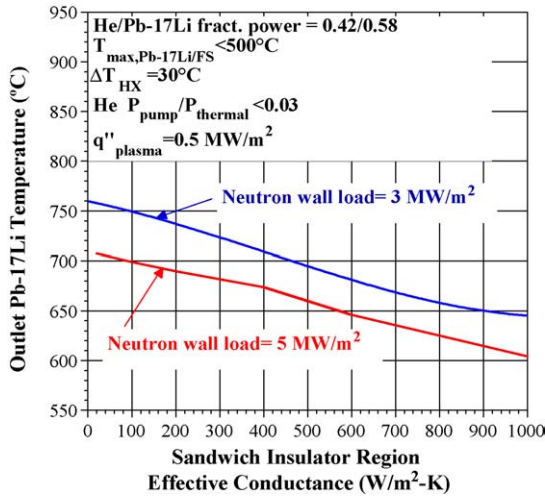


Fig. 5. Pb–17Li outlet temperature as a function of SiC insulation layer effective conductivity.

sidered for a permeator-based tritium recovery system from the hot Pb–17Li, which would then be routed to the heat exchanger with a minimal remaining tritium concentration (thus minimizing the tritium permeation there) [17].

Figs. 5 and 6 summarize the results for the performance of this blanket coupled to a Brayton cycle with 3 compression stages and a single expansion stage

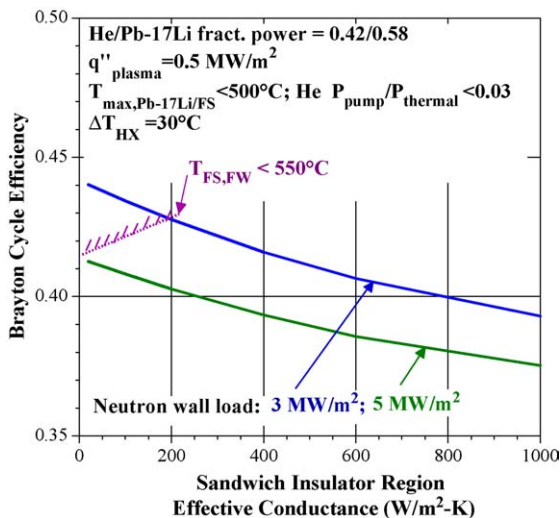


Fig. 6. Brayton cycle efficiency as a function of SiC insulation layer effective conductivity.

[20]. Fig. 5 shows the outlet Pb–17Li temperature and Fig. 6 the cycle efficiency as a function of the required conductance between the bulk Pb–17Li in the inner channel and the FS wall in order to accommodate the 500 °C compatibility limit. In addition, the He pumping power was limited to 3% of the total thermal power. In Fig. 6, the constraint of maintaining the radial average temperature of the first wall FS within 550 °C is also shown. For a 3 MW/m² wall load, the cycle efficiency is about 43% if a conductance of 200 W/m²-K is assumed (e.g. corresponding to a 5 mm thick layer of porous SiC with a conductivity of 1 W/m-K assuming a uniform Pb–17Li bulk temperature). For a maximum wall load of ~5 MW/m² (as roughly expected for ARIES-CS) and a conductance of 200 W/m²-K, the efficiency is slightly over 40%. In this case, the corresponding inlet/outlet temperatures of He and Pb–17Li are 355/485 °C and 475/690 °C, respectively. Although the MHD pressure drop of the Pb–17Li in bare inner channels could be acceptable (~0.1–0.5 MPa), the uncertainty linked with turning flows could increase the total pressure to an unacceptable value. Thus, the SiC layer also plays an important electrical insulation function resulting in a negligible pressure drop. Further studies are being conducted to assess the possibility (and R&D needs) of increasing the cycle efficiency through measures such as utilizing ODS ferritic steel in higher temperature regions (e.g. the first wall), assuming a higher Pb–17Li/FS compatibility limit in local areas, and reducing the SiC layer conductance.

4. Port maintenance scheme for modular concept

Port-based maintenance through a small number of designated ports involves the replacement of blanket using an articulated boom. Typically, there would be one or two maintenance ports per field period. For this scheme, 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 [9], as illustrated in Fig. 7 [11]. In this arrangement, the modular coils are kept at cryogenic temperatures during maintenance. A key feature of the layout is the desire to separate the vertically-supported hot core (including the shield and manifold, which are lifetime components) from the cooler vacuum vessel to minimize thermal

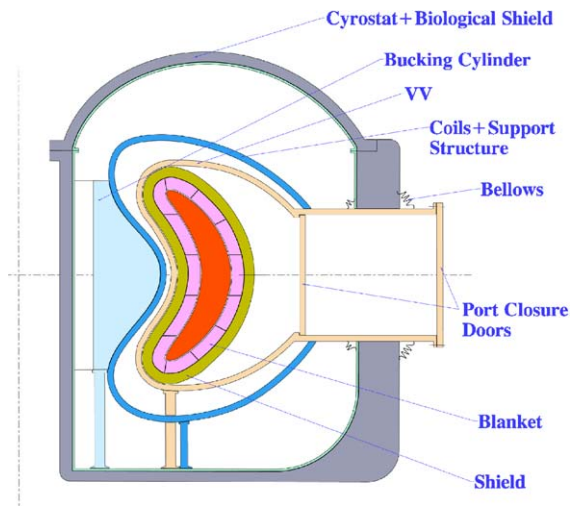


Fig. 7. Layout of 3-field period power core based on modular maintenance through a small number of designated ports.

stresses. In this case, 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 and required reach of the boom limit the weight (~ 5000 kg) and size of the blanket modules.

Cutting and re-welding the coolant pipes for blanket module removal and replacement can be done either by in-bore tools or from the outside of the tubes (using commercially available equipment such as shown in Ref. [21]). The latter is favored because of the rather complex geometry of the compact stellarator and coolant pipe routing, and the use of concentric pipes with sliding seals for the inner tube. In order to have access for performing these procedures a neighboring module must be removed first. Thus, the maintenance scheme requires in-series removal of specific modules in order to access a given module. Fig. 8 illustrates such a scheme where the shield pieces A, B and C must first be removed to access the coolant piping. The coolant piping cut is performed at the back of the shield where the He production is small enough to allow re-welding

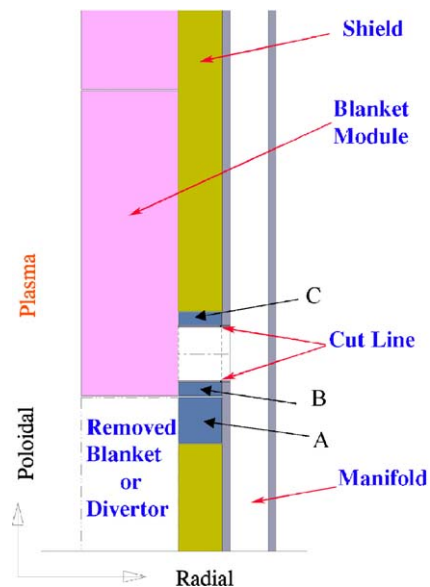


Fig. 8. Side view of module, shield and manifold illustrating coolant pipe cutting for blanket module removal.

(assumed as $\lesssim 1$ appm He). In the case of the He coolant piping, neutron streaming is a concern and is being further investigated since it may impact the possibility of re-welding.

As shown in Fig. 7, there is some concern about the available space at the bottom of the chamber for articulated boom access and operation to remove blanket modules at this location. This would be more problematic with more compact reactors. However, by removing a whole row of modules in order to access the lower module more space will be available in the chamber for this operation. As a last resort if space is really critical, there is also the possibility of using more than one port per field period, which would reduce the boom extension length required to reach the modules.

5. Divertor design

A concerted effort on the divertor has been launched as part of the phase-II effort. On the physics side, it involves adapting and using an array of codes to better assess the location of the divertor and to estimate the corresponding heat loads [22]. On the engineering side, the effort focused on evolving a design well suited to the

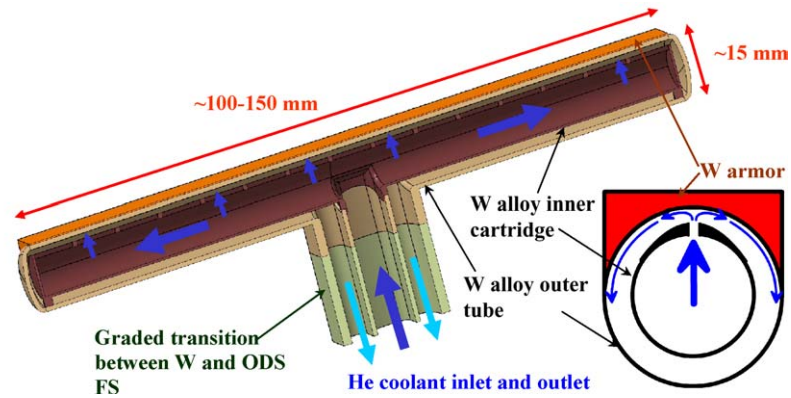


Fig. 9. Proposed T-tube divertor concept [25].

compact stellarator with the capability to accommodate a heat flux of 10 MW/m^2 (as a reasonable initial goal in anticipation of the physics modeling results) and which can be integrated with the in-reactor component design.

A range of different He-cooled divertor configurations have been considered in the past, including a tungsten plate design [23]. More recently, a finger configuration utilizing tungsten caps has been evolved with the aim of minimizing the use of tungsten as structural material and of accommodating higher heat fluxes through the use of smaller units [24]. It was decided to build on the tungsten cap design and to explore the possibility of a new mid-size configuration with good heat flux accommodation potential, reasonably simple (and credible) manufacturing and assembly procedures, and which could be well integrated in the CS reactor design. The proposed configuration consists of a “T-tube” illustrated in Fig. 9 [25].

The T-tube is about 15 mm in diameter and 100–150 mm long and is made up of a W alloy (e.g. plasma sprayed W-5% Re for better ductility, but the waste disposal impact would need to be assessed) inner cartridge and outer tube on top of which is attached a W armor layer (e.g. $\sim 5 \text{ mm}$ thick and castellated) subject to the plasma heat flux. The separately fabricated inner cartridge is inserted inside the outer tube and caps are welded at each end. Both W alloy pieces are connected to a base ODS FS unit through a graded transition (e.g. using diffusion bonded layers of graded W alloy/ODS FS composition) to minimize thermal stresses. The helium coolant is routed through the inner cartridge

first and then pushes through thin slots to cool the heat-loaded outer tube surface. A 2D-shaped impinging slot jet is created, leading to high heat transfer at reasonable pressure drop (a multiple jet impingement hole design [23] could provide even better heat transfer performance if required but at the cost of a slightly more complex cartridge fabrication). After impingement, the coolant flows as a highly turbulent wall jet along the large inside surface of the tube and then returns in the lower section of the annular gap between tube and cartridge. Recent results with He at inlet pressure/temperature of $10 \text{ MPa}/600^\circ\text{C}$ and a 0.5 mm jet-slot width indicate that the W alloy is maintained within 1300°C for a reasonable He jet pressure drop of the order of 0.1 MPa for a heat flux of 10 MW/m^2 . The maximum thermal stress is also reasonable, $\sim 370 \text{ MPa}$. A key assumption is that the W alloy will be able to operate in the temperature range of about $600\text{--}1300^\circ\text{C}$ based on ductility and recrystallization limits. The outlet He temperature is $\sim 700^\circ\text{C}$ in this case, which is similar to the blanket Pb–17Li outlet temperature and would fit within the overall heat exchanger and Brayton cycle scheme.

6. Conclusions

The results from the early phase-II effort are very encouraging. The modular design of the dual coolant concept has been evolved in more detail and key maintenance, assembly and integration issues

identified and studied. Future activities include developing in more integrated detail this blanket concept for field-period based maintenance, and revisiting the high performance SiC_f/SiC self-cooled Pb–17Li concept in sufficient detail for both maintenance schemes. These concepts will lead to the final selection of blanket concept and maintenance in coordination with physics and system optimization studies for the detailed phase-III design study.

Major progress has been made in developing an attractive He-cooled divertor design for the compact stellarator, which can accommodate a heat load of 10 MW/m² with the possibility of some thermal-hydraulic optimization margin. Future effort is directed toward better understanding the physics of the divertor to determine its location and heat load, and better integration of the divertor in the in-reactor design. Results of these activities will be reflected in the final divertor design for phase III.

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