INTEGRATION OF THE MODULAR DUAL COOLANT PB-17LI BLANKET CONCEPT IN THE ARIES-CS POWER PLANT

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A modular dual coolant lead-lithium (DCLL) blanket concept has been selected as the reference design for the ARIES-CS compact stellarator power plant study. The major characteristics of the blanket arrangement and power core integration associated with the modular maintenance scheme are summarized in this paper.

I. INTRODUCTION

A modular DCLL blanket is selected as the reference concept in the ARIES-CS power plant study. This concept is characterized by the following features:

1. Eutectic lead lithium alloy Pb-17Li serving as breeder and coolant (“self-cooled breeding zone”);
2. Helium-cooled FW and blanket structure made of the MF82H Reduced Activation Ferritic Steel (RAFS);
3. Flow channel inserts (FCI) made of SiC-composites arranged between the flowing liquid metal and the ferritic steel duct walls, serving as electrical and thermal insulator;
4. High outlet temperature of the liquid metal breeder/coolant (~700 °C), allowing the use of a Brayton cycle power conversion system.

In the past, DCLL blankets with self-cooled Pb-17Li breeding zones were proposed in the frame of the ARIES-ST (“Spherical Tokomak”) study, and were further developed in the EU Power Plant Conceptual Study (PPCS). The DCLL blanket became also the reference concept in the US-ITER Test Blanket Module (TBM) program.

Connected to the selection of the blanket concept are the issues of the integration of blanket modules, high temperature shield and coolant manifolds into the power core, and the methods for the replacement of blanket modules at the end of their life time or in case of failures. The main characteristics of the power core arrangement and integration of the DCLL blanket selected for the final phase of the ARIES-CS study are presented in this paper.

II. RADIAL BUILD OF THE POWER CORE WITH INTERNAL VACUUM VESSEL

The thickness of the different zones as required for tritium self sufficiency and adequate shielding of vacuum vessel (VV) and coils are determined by neutronics calculations. Special emphasis is placed on minimizing the amount of nuclear waste, and adequate shielding of the welds to be cut/re-welded for a blanket exchange. To achieve these goals, the following solutions are employed:

1. The thickness of the blanket module and back wall is adjusted to limit the neutron damage at the inner surface of the shield to < 200 dpa over the entire plant lifetime, i.e., 40 full power years based on the peak neutron wall loading of 5.3 MW/m², making all zones behind the blanket modules lifetime components.
2. Locating the manifolds and the coolant access pipes behind the shield in order to allow re-welding of this connections (helium generation during the entire lifetime of the plant < 1 appm).

It should be mentioned that the thickness of the breeding zone is not poloidally uniform. There are locations where the mid-coil to plasma distance is only about 1.3 m, allowing only a 25 cm thick breeding zone; therefore, a non-uniform blanket design is developed.

III. SUBDIVISION OF THE BREEDING ZONE INTO BLANKET MODULES

The blanket replacement method selected for the ARIES-CS study is similar to the one in PPCS, but opening of the FW panel is avoided by cutting the coolant access pipes from the outside of the module with orbital cutting/welding tools, starting with the module adjacent to the maintenance port. The disadvantage of disconnecting the coolant pipes from the outside is that it does not allow the replacement of every individual module independently of its location, but requires to start with the module neighbouring a port and to work up, down or over to the individual module needing replacement. However, even with the replacement method proposed in PPCS, it is not possible to replace any one of the modules separately
since the outboard modules are trapezoidal with width increasing with the radius.

In defining the maximum size of the blanket modules, a number of issues have to be considered. They include: (1) The load capacity of the articulated boom for the replacement of breeding modules is limited to about 5000 kg without installing a rail system and to about 10,000 kg if the boom is supported by a rail system. With a radial thickness of 0.63 m for a typical blanket module, and an average steel fraction of 20% leading to a weight of about 1200 kg/m², the maximum size of a module has to be limited to about 4 m² to stay inside the 5,000 kg weight limit for a module without coolant. (2) The maximum height of the maintenance ports in the VV is limited to < 4 m, and the maximum width to about 2 m due to coil interference; (3) Considerations of neutron streaming limits the size of the outer diameter of the concentric He coolant access pipe for a reasonable size of ~0.3 m ID, the corresponding pressure drop, pumping power and flow distribution limits the power per module to about 25 MW. For a maximum neutron wall load of 5 MW/m², a maximum surface heat flux of 0.8 MW/m², and a neutron energy multiplication of roughly 1.15, the maximum FW surface area of a module is limited to about 4 m². These considerations lead to a maximum module area of about 4 m², and a maximum height < 4 m.

IV. LAYOUT OF THE POWER CORE

Important questions for the lay-out of the power core include how and where to mechanically attach the blanket modules, and how to minimize thermal stresses caused by differential thermal expansion between the different components. The temperatures in the different components of a fusion power core based on DCLL blankets cover a large range:

- The VV is usually operated with an average temperature of ~200 °C;
- The helium inlet/outlet temperatures of the DCLL blanket are about ~350 °C/450 °C;
- The Pb-17Li inlet/outlet temperatures are about 450 °C/700 °C.

Stress calculations show that mechanical connected zones of ferritic steel structure must be operated either in a small temperature range (usually +/- 50 °C), or elastic connection elements (bellows, pipes with large bending radius, etc.) between the different structural elements are mandatory. In principle, bellows are interesting elements for the required elasticity, and they are extensively used in low temperature, low pressure systems (for example cooling pipes for the water cooled shielding blankets of ITER or the divertor plates there). Fig. 1 shows the sliding bearing and the layout of the hot power core with the VV and magnet (at a toroidal angle of 0°).

However, for the large diameter pipes (up to 0.3 m) necessary for helium or Pb-17Li cooled blanket modules, the high pressure (up to 10 MPa) and high temperature (up to 480 °C) of helium coolant pipes, or the extremely high temperature of the Pb-17Li pipes at the blanket exit (up to ~700 °C), the feasibility of bellows in contact with the coolants is questionable. To avoid the need for such elastic elements, we propose the following innovative features for the ARIES-CS power core:

A. Minimizing temperature variations of the blanket box by cooling the steel structure with helium and keeping the temperature rise of the helium below 80 °C.

B. Using concentric helium coolant channels in the manifolds and as connections between manifolds and blanket modules with the “hot” helium in the anulus (~ 441 °C) and the “cold” helium in the inner pipe (~ 369 °C).

C. Using concentric lead-lithium coolant channels in the manifolds and as connections between manifolds and blanket modules with the “hot” Pb-17Li (~711 °C) in the inner pipe and the “cold” Pb-17Li in the anulus (~ 452 °C).

D. Designing the concentric channels with sliding seals at the inner pipes. These sliding seals allow for differential expansion between inner tube and outer wall and, thus, minimize thermal stresses. Small leaks are not damaging since they open only very limited paths between inlet and outlet flow, considering the small pressure difference between inlet and outlet.

E. Connecting the high temperature shield and the poloidal coolant manifolds to form a strong “skeleton shell”, poloidally continuous and toroidally segmented. This skeleton shell is operated at nearly uniform temperature (~ 450 °C).
F. Each sector of this shell can freely expand relative to the VV since it is resting on sliding bearings at the bottom of the VV.

G. The blanket modules are mechanically attached to the skeleton shell and can float with it relative to the VV.

H. The coolant pipes connecting the skeleton shell with the external primary loop are designed as concentric pipes, operated also with a temperature close to 450 °C at the outer tube, and penetrating the VV at the geometric fixing point of each skeleton shell segment.

I. Arranging bellows between the VV and the coolant access pipes at the penetrations. These bellows to not see the coolant or a large pressure difference, because they have to seal only against the atmosphere in the VV and the cryostat.

IV.A. Coolant Access Pipes between Blanket Module and Poloidal Manifold

Each of the blanket modules is connected to the poloidal manifolds by two concentric coolant access pipes, one for helium and one for Pb-17Li. These pipes have to be cut/re-welded for every blanket exchange. Cutting re-welding the coolant access pipes from the outside of the module with orbital cutting/welding tools is recommended, starting with the module adjacent to the maintenance port. This requires first to remove all the components including the blanket, shield, and manifolds in port regions to allow tools to access to interior. As mentioned above, only the outer pipe has to be cut, because there are sliding seals at the inner pipe to allow for differential thermal expansion and the replacement without cutting/re-welding of the inner tube. For the Pb-17Li pipes there is the additional requirement to install flow channel inserts made of a SiC-composite in the bore of the inner tube as well as at the inner surface of the outer tube for thermal and electrical insulation.

An important criterion for the design of such coolant access pipes is the requirement that the helium concentration in the steel at the location of an assembly weld must be < 1 ppm. To ensure this, we locate the assembly weld of the blanket module at the interface between shield and poloidal coolant manifolds. The thickness of the shield is determined to meet the 1 appm He criteria at this interface. However, neutron streaming through the large helium duct increases the helium generation at the weld. Therefore, there are shielding rings installed inside the coolant access pipes for sufficient shielding of the weld. Cutting and re-welding of such complicated tubes is a very challenging task. We propose to use orbital welding heads for this purpose. To gain access to the outer tubes of the concentric coolant access pipes, small portions of the shield need to be removed. These removable shielding blocks can be made of WC, and can radiate their volumetric heat to the cooler surroundings. For this reason, no active helium cooling is needed for these blocks, facilitating in this way the blanket replacement to a large degree. Figs 2 and 3 show the access pipes connecting the blanket module to the manifold behind the shield.

![Fig. 2. Blanket coolant access pipes connecting to the manifold showing the removable shield blocks](image1)

![Fig. 3. Side view of blanket coolant access pipes connecting to the manifold](image2)
details about the steps for cutting/re-welding the coolant access pipes can be found in Ref. [6].

IV.B. Coolant Channels in the Poloidal Manifold

Each blanket module is connected to the manifold with two coolant access pipes, one for Pb-17Li, one for helium. Each of the poloidal manifold channels is connected to one blanket module. The cross sections of these manifolds are a rectangular outer channel and a circular tube inside as shown in Fig. 4 for the helium and the Pb-17Li channels.

![Diagram of poloidal manifold channels](image)

a. for helium

b. for Pb-17Li

Fig. 4. Poloidal manifold channels, (a) for helium, (b) for Pb-17Li

There are flow channel inserts shown in Fig. 5(b) for the liquid metal (LM) manifold, serving as electrical and thermal insulator. In case of helium the “cold” (~369 °C) inlet flow is in the inner tube and the “hot” (~441 °C) exit flow in the surrounding channel. The flow direction is reversed for the Pb-17 Li flow where the incoming liquid metal (LM) (~ 452 °C) flows in the rectangular outer channel, the “hot” (~711 °C) exit flow in the inner tube. The purpose of this arrangement is to keep the temperature in the manifold as uniform as possible in order to minimize thermal stresses.

IV.C. Combining Steel Shield and Poloidal Manifolds to Form a Strong Skeleton Shell

The high temperature steel shield is cooled by the helium stream leaving the breeding modules at a temperature of ~ 450 °C. Since the volumetric heat generation in this component is relatively small, the steel structure is maintained very close to this temperature and nearly uniform over the entire shield. Therefore, the high temperature steel shield and the poloidal manifolds described in the previous section can be combined to form one component (either by welding or by bolting), operated at nearly uniform temperature of ~ 450 °C. This component can be designed as shells continuous in the poloidal direction but segmented in the toroidal direction. We suggest sub-dividing the torus into 3 sectors for the three field-period compact stellarator. If it turns out that such large sectors are too difficult to fabricate and to handle, subdivision of the torus into 6 or 12 shells would be feasible too. These shells have sufficient strength to carry the weight of the breeding modules and to transfer the mechanical loads to the vacuum vessel. Since the temperature of these shells is (~250°C) higher than the temperature of the vacuum vessel, sliding bearings between the shell and the VV are required as shown in Fig. 5. The determination of the required contact area of these bearings requires a trade off between allowable contact pressure and resulting heat losses between the skeleton shell (~ 450 °C) and the VV (~ 200 °C). As a result of scoping estimates, we suggest to use for each of the three sectors (one per field period) three bearings with a contact area of 0.5-1 m² each.

The skeleton shells can freely expand in all directions, minimizing the thermal stresses. The concentric toroidal coolant access pipes attached to these shells are connected to the external loops for helium and Pb-17Li by one concentric coolant access pipe, penetrating the VV close to the location of the geometric fixing point of the shell. Small remaining differential expansions between these access pipes and the VV are allowed by bellows between these two components (see Fig. 5 (b), the A-A cross-section). These bellows are not in contact with the coolant and have to withstand only the pressure difference between the inner and the outer side of the vacuum vessel. This difference is close to zero during normal operation (plasma inside the VV, vacuum in the outer cryostat). Only in case of maintenance or under accidental condition, the pressure difference could increase up to a few bars.
V. CONCLUSIONS

Neutron damage in the FW requires the replacement of the breeding modules at a fluence of about 15.7 MWy/m², or after about 3 full power years for a FW neutron wall load of 5.3 MW/m². In order to avoid high penalty on the availability of the fusion power plant, a fast replacement of the breeding blankets at the end of their lifetime or in case of failures is an important requirement for the design of the fusion power core. For environmental and economical reasons it is also important to minimize the amount of materials to be disposed of when replacing the blanket modules. The design of the power core as suggested in this paper facilitates the achievement of both these goals. Minimum radioactive waste is achieved by designing all regions behind the breeding blankets as lifetime components. The breeding zone is subdivided into about 198 modules each with a FW surface area of ~ 4 m², to be replaced through a small number of horizontal maintenance ports, using articulated booms inserted into the plasma chamber. The blanket modules can be replaced without removing the shield and/or manifold regions using proven technologies for cutting and re-welding the coolant access pipes, minimizing the time required for blanket replacement.

The major features of the power core arrangement for the ARIES-CS power plant study are summarized, including the novel features associated with the concept of a “separate” hot power core design. All these measures result in the potential for reliable operation and reasonable down time for blanket replacement. Although this design is developed in the frame of a Compact Stellarator Power Plant Study, there are no inherent reasons why the design principle cannot be applied to a tokomak power plant.

ACKNOWLEDGMENTS

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

REFERENCES