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Blanket system selection for the ARIES-ST☆

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Abstract

The ARIES-ST (Spherical Tokamak) is to investigate the attractiveness of a low-aspect-device as the confinement concept for a fusion power plant. The key driven force of the ST design is caused by the center column conductor. The design selected is a water-cooled Cu normal conductor. This selection has a major impact on the blanket design and selection, tritium breeding and over-all power balance. The blanket selected is a dual coolant concept, partially decided by the characteristics of the center conductor. The final blanket design is modified from the dual coolant concept, which developed under the EC DEMO program. The reason for this selection and the design issues are summarized in this paper. © 2000 Elsevier Science S.A. All rights reserved.

Keywords: Blanket system; Conductor; Dual coolant

1. Introduction

The ARIES-ST is investigating the attractiveness of using an ST configuration as plasma source for a commercial power plant. The ST configuration has a water-cooled, normal conducting center column. To protect the center column, a 20 cm shield is used. There is no breeding blanket on the in board (IB) of the power plant. Therefore, a ST power plant have to take the following points into consideration:

1. The center column will most likely be a water-cooled Cu magnet. Therefore, the design of the power plant has to be compatible to a system with a water-cooled major component.
2. A ST has high recirculation power, mainly due to the large ohmic power to the center column. Therefore, an ST power plant needs to have a high power converting efficiency to partially off-set the ohmic power for the center column.
3. There is no breeding blanket on the IB. Therefore, the breeding capability of the out board (OB) blanket has to be sufficient to compensate for the non-breeding IB.

Also, an ST has the potential to go to a high neutron wall loading due to the high beta. Therefore, it is important for the engineering design to identify a design with high neutron wall loading capability.

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The IB of the ARIES-ST has no blanket. However, a shield is added to protect the Cu center column from radiation damage. The cooling and heat removal from the center column is a difficult problem because of the space limitation, the length of the flow path and power density. Also, the refection of neutron from the IB to the OB

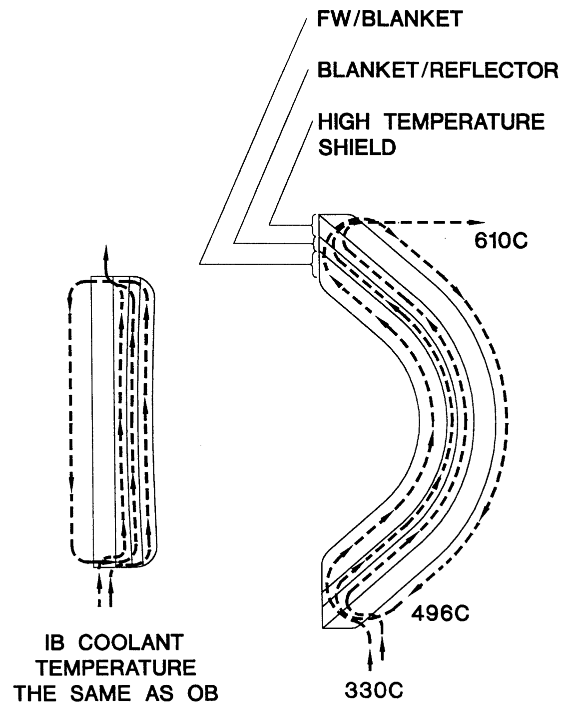

Fig. 1. Li/V blanket coolant flow path.

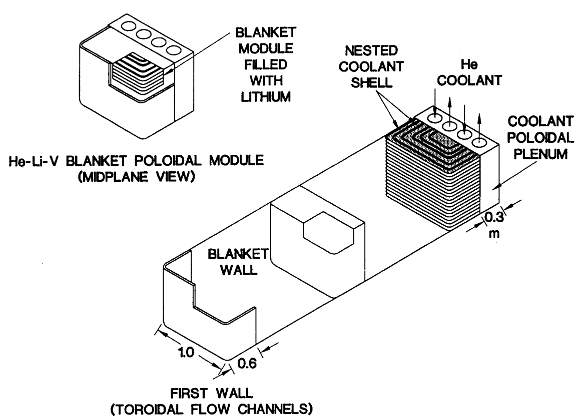

Fig. 2. He/Li/V blanket configuration.

breeding regime is required to assure tritium self-sufficiency. Therefore, the breeding requirement of the OB has a major impact on the selection of the material for the IB shield.

The ARIES-ST team investigated many blanket concepts which are potential candidates for the ARIES-ST. The evaluation of the concepts and the final selection of the reference blanket, is summarized in this paper.

2. Concepts description

2.1. A self-cooled lithium blanket with V-alloy as the structural material

The self-cooled lithium blanket has been investigated under ARIES-II and ARIES-RS. The concept is illustrated on Fig. 1. This is one of the most widely investigated power plant blanket within the US fusion program. Many of the key issues, such as the liquid metal magnetohydrodynamics (LMMHD), insulating coating development and V-alloy development, have active research program under US fusion program.

There are many advantages of this concept, such as,

- Simplicity in design.
- Low activation.
- High tenperature capability, which leads to high thermal efficiency.
- The best tritium breeding potential.
- Good first wall heat load capability.
- Low operating pressure.

A key concern of the Li/V blanket concept is caused by the chemical reactivity of lithium with both water and oxygen. For the ARIES-ST design, the presence of the water-cooled center column makes the using of this blanket concept unacceptable. However, if a different coolant for the center column can be identified, a Li/V blanket may be attractive.

2.2. A He-cooled, lithium breeding blanket with V as the structural material

This concept was proposed to the ARIES team as one of the candidate for the ARIES-ST. The

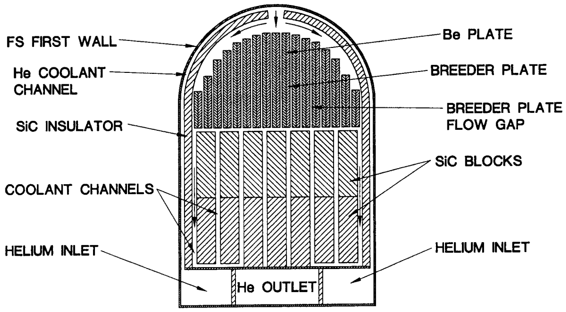

Fig. 3. He/FS/Solid breeder/SiC blanket configuration.

concept is illustrated on Fig. 2 [2]. He at very high pressure, 18 NWa, was proposed as the coolant. The key reasons to propose this concept are:

- Since lithium is not circulated for energy removal, but only for tritium recovery, the LMMHD pressure drop may not be that large. The development of an insulating coating may not be necessary. Using He-coolant at very high pressure improves the efficiency of the He cycle. It was estimated that, with a He exit temperature of 650°C, a cycle net efficiency of 46% is achievable, if a very high regenerator efficiency can be obtained.
- This concept reduces the lithium inventory comparing to a self-cooled lithium blanket. Therefore, safety concerns may be alleviated.

The conclusion from the ARIES team was that, although the total lithium inventory is less than that of a self-cooled lithium blanket, there is enough lithium to cause major safety concern.

Table 1
Wall loading limitation for a solid breeder blanket

Available temperature range for heat transfer (maximum allowable breeder temperature — coolant temperature	500°C
Breeder plate thickness	1 cm
DT over thye breeder	380°C
DT over the gap	51°C
DT over the side wall	31°C
DT to the coolant	38°C
Maximum allowable nuclear heating in the solid breeder	31 w cm^{-3}
Maximum allowable neutron wall loading	
With no BE	~3 MW m^{-2}
With BE	~1.5 MW m^{-2}

2.3. A He-cooled, Li_2O breeding, FS structure blanket with SiC acts as thermal insulator

Since water reaction with breeding material is a major concern, it is logical to investigate the possible usage of a solid breeder material for the blanket design. The Li_2O was selected because of its good breeding potential and low activation. Fig. 3 illustrated this concept.

The blanket module is U-shaped. The coolant will enter the blanket directing right toward the first wall along the side wall of the U. The key design consideration is to have He near the entrance temperature at the first wall to maximize first wall cooling. Also, the entire FS structure will be cooled by He near the He entrance temperature. The blanket will act as a super heater to the He coolant to heat it up to a high temperature for efficient thermal conversion. The FS structure is insulated by a SiC layer from the high temperature blanket to keep the FS temperature below the design temperature limit of 550°C.

The logics for this design are:

- The FS is used as the structural material for low cost.
- The coolant will cool the FS and first wall near the coolant entrance temperature for efficient cooling.
- The blanket will act as the super heater for the coolant to a high temperature for efficient power conversion.

This design has a very severe wall loading limitation based on heat transfer from the breeding material to the coolant, at the first zone behind the Be multiplier. Table 1 summarizes the heat transfer calculation from the breeding zone to the He-coolant. As can be seen, the maximum allowable heating rate within the breeding material is 33 w cm^{-2}. This heating rate corresponds only to 3 MWm^2 neutron wall loading, even in the case with no Be. The 3 MW m^{-2} neutron wall loading is the maximum. Therefore, the allowable average neutron wall loading is only 2 MW m^{-2}. If Be is added for neutron multiplication, the allowable neutron wall loading will be even lower. It is the judgment of the ARIES team that, with an allowable neutron wall loading of less than 2 MW

m^{-2}, the solid breeder concept will not be considered as the reference case for the ARIES-ST.

2.4. A dual coolants concept

This concept is very similar to the EC dual coolants concept for the DEMO blanket design [3]. The two coolants are He at 8 MPa and LiPb. The concept is illustrated on Fig. 4. The key difference between this design and the EC DEMO design is that a SiC layer is used here to decouple the He temperature from the LiPb temperature.

The He is used to cool the FS structure. The LiPb carried the nuclear heating from the blanket and is at a high temperature. The He will be used as the preheated, while the LiPb is used as the superheater, while transfer heat to a secondary He stream. A He turbine is selected as power conversion unit.

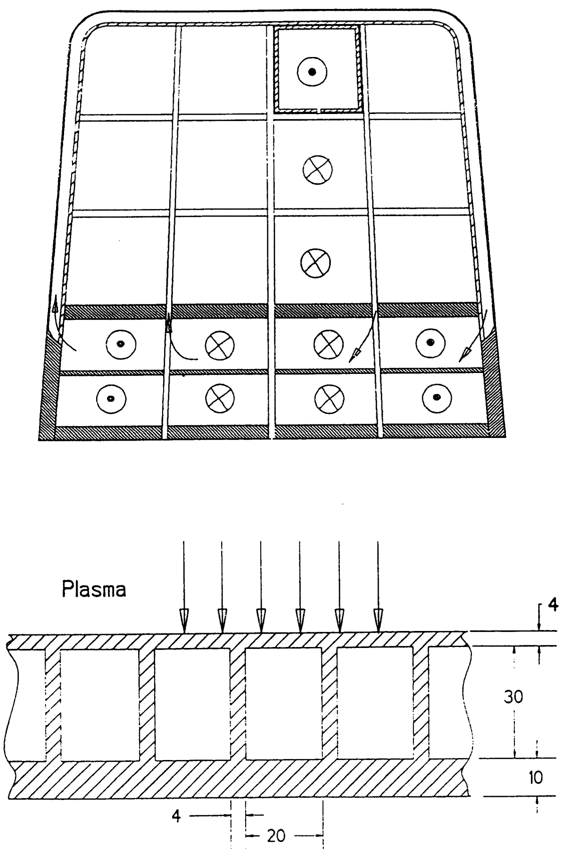

Fig. 4. Dual coolant first wall blanket configuration.

LiPb is an interesting breeding material. It is compatible to a water-cooled component. It has good tritium. breeding potential. Also, being a self-cooled concept, it has good nuclear heating removal capability. One of the problems with LiPb is the low thermal conductivity. Therefore, it has a limited surface heat removal capability. For this reason, another coolant, He, is used for first wall cooling. The He is also used to cooled the FS structural material to keep it below the design limit of 550°C. LiPb will be heated up to a high temperature to assure efficient thermal conversion. To keep hot LiPb away from the FS structure, a layer of SiC is used to keep hot LiPb away from the FS. The compatibility of LiPb with SiC is a concern. Some preliminary investigation suggested that static LiPb is compatible with SiC up to 850°C [4].

The tritium, partial pressure over the LiPb can be very high. To prevent tritium permeation concern, a close cycle He turbine is used for power conversion.

This concept has been selected as the reference design for the ARIES-ST. Some more important issues of design analysis are summarized here.

3. First wall cooling

For the dual coolant blanket concept, there is some concern about the maximum allowable first wall heat flux and neutron wall loading. The conclusion from ARIES-ST evaluation is that the allowable neutron wall loading is limited. The design window can be enlarged, however, by relaxation of some of the design parameters.

The reference ARIES-ST parameters are summarized on Table 2. Some of the parameters will be changed to assess the most effective improvement to increase the heat load capability. It is assumed that 50% of the alpha power will go to the first wall and there is no peaking factor.

The maximum allowable neutron wall loading for the dual coolant concept, based on the first wall heat removal capability, are summarized on Table 2. The allowable neutron wall loadings for the cases of relaxing the design limits are also presented. It appears that the most efficient way

Table 2
OB heat transfer parameters[a]

Coolant inlet temperature	350°C (reduced to 300°C)
Maximum allowable structural temperature	550°C (to be increased to 600°C
First wall thickness	4 mm (to be reduced to 3 mm)
Coolant heat transfer coefficient	1 w cm^{-2}-C
	0.64
He exit temperature	450–500°C

[a] q_x is defined as (Power deposited in the first wall He coolant)/(Total power deposited in He).

to improve the neutron wall loading capability is to increase the allowable first wall temperature. Even with a reasonable relaxation of all the parameters, the allowable neutron wall loading is about 8 MW m^{-2}, with a surface heat flux of 100 w cm^{-2}. It should be noted that this is a limit which is unlikely to be reached (Table 3).

4. Breeding

This section examines the breeding potential of the proposed blankets and documents the overall tritium breeding ratio (TBR) for the reference LiPb/FS blanket design. As a top-level requirement, all the ARIES designs should provide an overall TBR of 1.1. The 10% breeding margin is necessary mainly to supply the tritium needed to start other fusion plants and to account for uncertainties due to approximations and/or errors in the various elements of calculations such as nuclear data, calculational method and geometric representation [1].

Achieving tritium self-sufficiency in ST is a challenging task. It appears unlikely that a breeding blanket could be installed in the space-constrained inboard region. Furthermore, as the aspect ratio (A) of the ST design increases, it becomes more difficult to depend entirely on the outboard blanket to provide all the tritium needed for plasma operation. As an example of the effect of A on TBR, changing A within the range of interest for ST from 1.25 to 1.8, drops the outboard neutron coverage fraction from 90 to 80%. For higher A machines, the design should, therefore, respond and seek ways to augment the outboard breeding to compensate for the losses in the outboard neutron coverage fraction. This could represent a problem especially for blankets with marginal breeding.

Several blankets were proposed for the ARIES-ST design. The various blankets employ ferritic steel (FS) and vanadium alloy structures with solid and liquid breeders. The candidate breeders include LiPb, Li, Li_2O and Li_2TiO_3. A comparative l-D analysis was performed to assess the breeding potential of the proposed blankets. The main results are:

- blanket dimension and the Be content depend on the breeder type;
- solid breeders offer thinner blankets compared to liquid metals;
- Li/V blanket can easily meet the breeding requirement;
- breeding level of the LiPb/FS/SiC blanket is marginal;
- Li_2O could breed without Be if no structure is used;

Table 3
Allowable neutron wall loading, MW m^{-2}

He exit temperature	500°C	475°C	450°C
Case 1: reference case	3.3	3.8	4.3
Case 2 (t)fw = 3 mm	3.9	4.5	5.1
Case 3 (T)structure = 600°C			
Case 4 (T)coolant inlet = 300°C	3.8	4.3	4.9
Case 5[a]	7.0	7.7	8.4
The corresponding firstwall heat flues are	87.5 w cm^{-2}	96.3	1.5

[a] Case 5 assumes that the first wall thickness is 2 mm, the coolant inlet temperature is 300°Cand the maximum allowable structural temperature is 600°C.

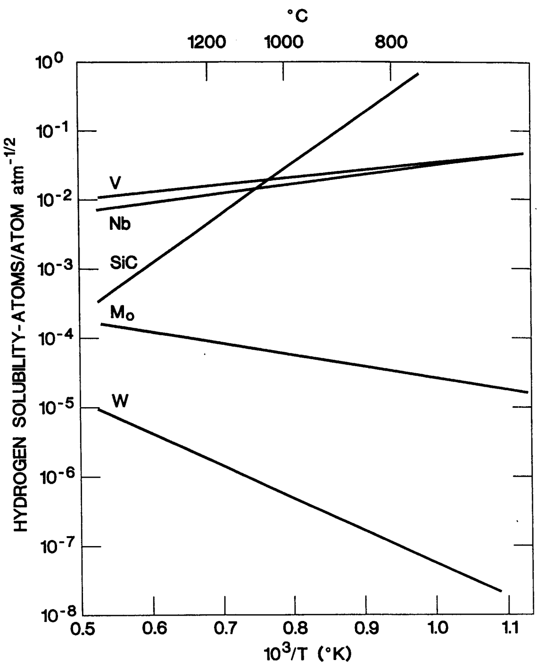

Fig. 5. Tritium solubility in SiC and other materials.

- titanate design needs a large amount of beryllium to enhance the breeding.

A three-dimensional Monte Carlo analysis was performed to determine the overall TBR for the selected blanket option [5]. The reference design employs FS as the primary structural material, Li17Pb83 as the tritium breeder, and helium as the main coolant. The water-cooled center post (CP) is protected by a 30 cm thick helium-cooled inboard shield [6].The 3-D model did not include the RF penetrations or gaps on the outboard side. The analysis indicated that the breeding level of the reference LiPb/FS blanket is slightly below the requirement. Options to enhance the breeding were then identified. The following agreed-upon changes will be implemented in the final design to achieve an overall TBR of 1.1:

- thicker outboard blanket; add a fourth LiPb cell;
- thinner outboard first wall; remove a few mm of FS;
- thinner SiC insulator in blanket; 0.3 cm thick SiC particularly in the first cell;
- eliminate the use of water in the inboard shield; water reduces TBR by $\sim 10\%$;
- design compact RF penetrations without Faraday shield to minimize neutron losses.

5. Tritium and power conversion

One of the key problem of using LiPb as the breeding material/coolant is the low tritium solubility, which causes very high tritium partial pressure. The high tritium partial pressure will cause problem on tritium inventory, if there is some material with high tritium solubility. Also, tritium permeation maybe an issue, especially if there is a high temperature heat exchanger.

The tritium solubility over LiPb has been measured. The Sievert's constant of tritium solubility in the LiPb is reported to be [7]:

$$Ks = 2.32 \times 10^{-8} \exp(-1350/PLT)$$

in which

Ks is in atomic fraction/$Pa_{1/2}$

T in K

R in j mole-K^{-1}

With this solubility, tritium partial pressure increase per coolant passage in the blanket is calculated to be 20 Pa. This is a very high tritium partial pressure, especially for a high temperature system.

SiC has a very high hydrogen solubility, as shown on Fig. 5. For the usual applications, the hydrogen diffusivity in the SiC is so low, that it takes a long time for the hydrogen to diffuse in. For this blanket, the SiC component will immersed in a bath of LiPb for a few years, and subjected to a high tritium partial pressure. The tritium inventory in the SiC is calculated to be 100 kg, it the blanket life-time is 2 years. This is potentially a severe issue of using SiC in a high tritium partial pressure environment. Other insulating materials are being investigated.

The high tritium partial pressure will also cause high tritium permeation across the heat exchanger. Therefore, a steam cycle is not a good candidate for this system. A power conversion system with a close-cycle gas turbine is being considered.

6. In board cooling

The IB shielding is designed to protect the center column magnet. The design of the heat removal system for the IB shield is challenging.

1. Due to the ST configuration, the space in the IB shield is very limited.
2. Due to the high neutron wall loading, the first wall surface heat flux is very high.
3. The total nuclear heating deposited in the shield is large.
4. The coolant flow path is very long.

Also, the selection of the coolant is restricted by other considerations:Liquid metal can not be considered due to the severe MED effects.

1. Water can not be considered due to the neutron absorption that reduces the neutron reflection to the OB blanket. The tritium breeding will not be sufficient.
2. Organic coolant can not be considered due to the radiolalysis. Therefore, the only possible candidate for the coolant is helium. The problem with He is the low thermal conductivity, which impacts heat transfer, and the low density, which requires large manifold and high pumping power.

The design of the coolant loop is shown in Fig. 6. There are two manifolds, for inlet and out coolants. The He coolant will exit from the inlet manifold, goes circtiferentially around the EB first wall five times to get enough velocity for heat transfer and returns to the exit manifold. The manifold is designed to maximize the channel dimension for easy heat removal and acceptable pressure drop.

The Ib-He coolant system parameters are summarized in Table 4.

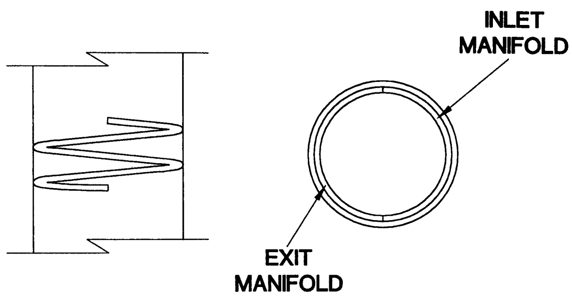

Fig. 6. In board shield coolant configuration.

Table 4
IB coolant parameters[a]

Total nuclear heating for the IB first wall	65 MW
Total nuclear heating for the IB shield	278 MW
Estimated total surface heating	40 MW
Total power to be handled	383 MW
Coolant temperature rise	200°C
Coolant velocity in the manifold	200 m s^{-1}
Manifold thickness	10.4 cm
First wall coolant velocity	100 m s^{-1}
First wall coolant channel diameter	1 cm
Total He pumping power	80 MW

[a] The configuration of the IB erquires high He pressure, high He velocity, results high pumping power. The space required for the manifold reduces thespace available for the Cu coil.

7. Summary

The selection of the blanket concepts, and the design of the reference system has been reported. The design of this blanket is challenging because of the more sever design requirements and the restriction from the design of the center column. The dual coolant concept is attractive because it can fulfill most of the requirements.

To assure the performance of this blanket to a high neutron wall loading with a high coolant exit temperature, an insulator has to be used. This insulator has to be compatible to LiPb at a high temperature, subject to high tritium partial pressure and substance high radiation damage. The material selected now is SiC. Whether SiC, or any other material, can provide long life time under these situation, is not certain. No other possible candidate insulating material has been identified.

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