The ARIES-RS power core—recent development in Li/V designs[1]

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Abstract

The ARIES-RS fusion power plant design study is based on reversed-shear (RS) physics with a Li/V (lithium breeder and vanadium structure) blanket. The reversed-shear discharge has been documented in many large tokamak experiments. The plasma in the RS mode has a high beta, low current, and low current drive requirement. Therefore, it is an attractive physics regime for a fusion power plant. The blanket system based on Li/V has high temperature operating capability, good tritium breeding, excellent high heat flux removal capability, long structural life time, low activation, low after heat and good safety characteristics. For these reasons, the ARIES-RS reactor study selected Li/V as the reference blanket. The combination of attractive physics and attractive blanket engineering is expected to result in a superior power plant design. This paper summarizes the power core design of the ARIES-RS power plant study. © 1998 Published by Elsevier Science S.A. All rights reserved.

1. Introduction

The first wall/blanket/shield (FW/B/S) materials selection determines the performance of a fusion power plant. Proper selection of the materials have the following effects:

1. The structural material and coolant selection determines the blanket temperature, which in turn, decides the power conversion system and the efficiency. This requires high blanket temperature.

2. The structural material and the coolant determine the activation, the after heat and the waste disposal rating of the materials. Therefore, it determines the safety and environmental characteristics of the fusion power plant. This requires low activation materials.

3. The selection of the breeding material determines the tritium breeding, and tritium inventory in the blanket. This requires the selection of a breeding material which can give high tritium breeding.

4. The selection of the breeding and shielding materials determines the thickness required for the magnet protection. This requires the selection of effective shielding materials.

5. The complexity of the blanket/shield has a dominate effect on the reliability of the fusion power plant. This requires simple blanket geometry and low blanket pressure.

6. The first wall/blanket and the divertor designs determine the maximum acceptable neutron wall loading. This requires good heat transfer design.

The ARIES-RS design selected a self-cooled lithium design with a V-alloy as the structural material. It is the judgment of the ARIES team that this blanket has the best potential to fulfill most of those requirements with only a moderate extrapolation of today's technology. The V-alloy has low activation, low after heat, high temperature capability and can handle high heat flux. A self-cooled liquid lithium blanket is simple and, with the development of an insulating coating, has low operating pressure. Also, this blanket gives excellent neutronic performance. There are a number of engineering issues that have to be resolved. The most critical issues are:

1. V-alloy material development
2. V-alloy industry development
3. Insulating coating development
4. Power cycle development
5. Tritium recovery
6. Reliability and replacement assessment
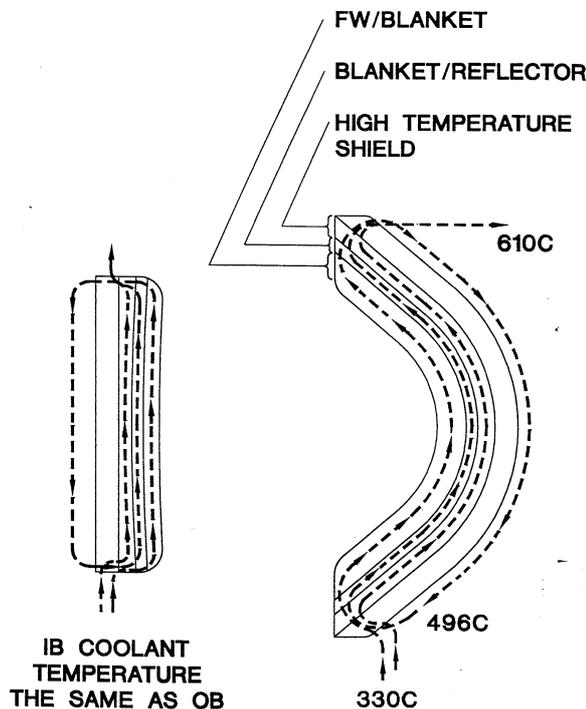
These issues will be discussed in the final section.

2. FW/B/S mechanical configuration

For a self-cooled liquid metal blanket design for a magnetically confined fusion power plan, the key consideration is to reduce the liquid metal MHD pressure drop. However, for the ARIES-RS design, it is assumed that an insulating coating can be developed to reduce the MHD pressure drop. With this assumption, the MHD pressure drop is no longer a major concern. The design of the FW/B/S can be optimized to improve heat transfer and simplified the configuration.

A very simple blanket configuration is based on once through, poloidal flow [1]. For the ARIES-RF design, major effort was made to radiate the alpha power to the first wall to reduce the heat flux to the divertor. For this reason, a first wall heat flux of 1 MW m^{-2} was assumed during the design study. The once through configuration is unable to handle this high heat flux, due to the high coolant temperature near the exit end of the first wall. The coolant configuration is shown on Fig. 1, with the coolant flows along the first wall

Fig. 1. ARIES-RS blanket flow direction.

first, and enters the blanket to remove the nuclear heating deposited in the blanket. This design modification from the once through flow significantly reduces both the coolant temperature and the coolant residence time for the first wall and improves first wall heat removal capability. The coolant for the reflector regime remains as once through. This different configuration for the blanket and reflector is selected to minimize the mechanical connection between the blanket regime and the reflector regime to simplify the maintenance procedures.

3. Primary loop and power conversion systems description

The selection of the heat transport system for the ARIES-RS design can be summarized as follows:

Primary coolant	Lithium
Primary structural material	V−4Cr−4Ti
Secondary coolant	Sodium
Secondary structural material	316 SS
IHX design	Double wall IHX
Power conversion	Steam
Tritium recovery method	Cold trap

The design of the primary loop and power conversion system is critical to the attractiveness of a power plant. This is the system to assure efficient power conversion, to isolate the radioactive products within the nuclear island, and to assure reliable operation of the power plant.

A key concern of designing the primary loop design for a power plant with a Li/V blanket is how to transfer the design from a Li/V system to a system compatible to the power conversion system. With a blanket temperature limit of 700°C, the power conversion system most likely will be an advance steam cycle. Therefore, it is best to have another material, most likely an iron-based alloy, to be used for the power conversion system. Also, the cost of the V-alloy is high. It is not certain whether V-alloy can be used for the entire system from an economic point of view.
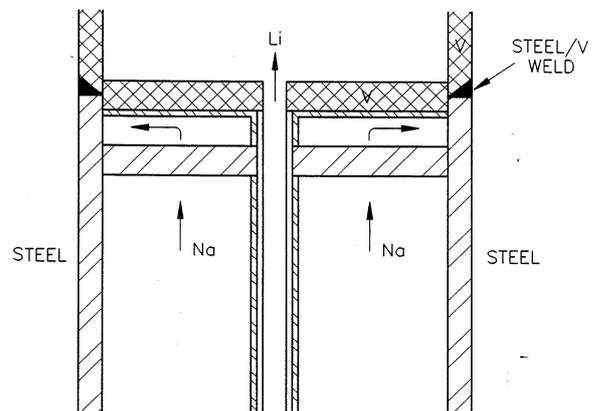

Fig. 2. The V-SS join at the IHX.

Therefore, the design of the primary loop has to make the transition from a V-alloy structure to an iron-based alloy structure.

The design proposed here is to use a double-walled IHX, both to improve the reliability of the IHX, and to provide a location to make the V-steel join. The primary coolant of the ARIES-RS is Li, while the most logical choice for the secondary coolant is Na. The problem is that, at the temperature we have, Li is not compatible with steel, while Na is not compatible with the V-alloy. Hence, the transition location has to be made at a site which does not face either Li or Na.

The design of the double-wall IHX is illustrated in Fig. 2. To join double-wall coolant tubes, two tube sheets are required, with the inner tube sheet joining the outer tubes and the outer tube sheet joining the inner tubes. This is a standard double-wall heat exchanger design developed by Westinghouse for breeder program applications [2]. The key feature for the ARIES-RS double-wall IHX design is to use the space on the HX container at the outer tube sheet to make the transition from V-alloy structure to the steel structure, as illustrated in this figure. The reason for doing this is that the steel/V weld does not face either Li or Na, to avoid the material compatibility problem. Further testing is certainly required to confirm this design, and to find a method to make the V/steel weld.

The power conversion system selected is an advanced steam cycle. This cycle was recommended by the EPRI for the next generation of power stations [3]. To maximize the power conversion efficiency, a double reheat system was selected. There are nine stages of feedwater heaters. The thermal converting efficiency for this steam cycle was reported to be 46.4%.

4. Tritium system description

Many processes have been proposed to recover tritium from liquid lithium [4]. The goal of the design process is to limit tritium concentration in the lithium to about 1 appm. This goal for a commercial power plant is to limit the tritium inventory in the lithium to less than 200 g. Due to the high solubility of tritium in the lithium and the required low concentration, the tritium recovery from lithium becomes a difficult technical issue. The only recovery process which has been demonstrated to be able to recover tritium to this level is the molten salt recovery process [5]. However, the salt used to recover tritium will be dissolved in the lithium, which may cause compatibility problems with the insulating coating required to reduce the liquid metal MHD pressure drop.

The tritium recovery process proposed here is based on cold trap process [6]. The cold trap process has been demonstrated to be able to recover tritium from lithium [7], sodium [8] and potassium [9] to their solubility limits. For the liquid lithium system, the hydrogen solubility at the cold trap temperature of 200°C is 440 appm, which is far above the design goal of 1 appm. For this reason, cold trap has not been considered as a candidate process for recovery tritium from lithium. The concept developed here is to add protium to the lithium so that the total hydrogen concentration in the lithium is higher than the 200 appm saturation value. At 200°C, $Li(H + T)$ will be supersaturated and precipitate out together. The co-precipitation of hydride and tritide has been demonstrated in the breeder program. The $Li(T + H)$ can be separated from lithium by a process called 'meshless cold trap' process which

was developed by the breeder program to separate NaH from Na by gravitational force [10]. The $Li(T + H)$ can than be heated up to 600°C for decomposition. The hydrogen stream will than be fed to the main isotope separation system (ISS) to separate tritium from protium. A calculation by the ITER-Naka estimated that the additional tritium inventory and the refrigeration power caused by the blanket tritium stream are acceptable (O. Kveton, personal communication).

5. MHD and heat transfer

The key design consideration for a self-cooled liquid metal blanket for a tokamak is the MHD pressure drop. With a moderate low plasma beta (that means high magnetic field) and a high neutron wall loading, there is no design window for a self-cooled liquid metal blanket for the in board blanket regime [11] Therefore, the development of an insulating coating is necessary for a self-cooled liquid metal blanket to be feasible. This insulating coating has to be reliable over a long period of time, under severe neutron irradiation, compatible with both the structural material and the coolant, and can survive repeated thermal cycling and maybe plasma disruption.

The development of the insulating coating is still at an early stage. The reference material in the US program is CaO [12]. With the assumption that a fully insulated coating can be developed, the blanket will be designed to be optimized to heat transfer. For the ARIES-RS design, a strong effort was made to radiate the alpha power to the first wall. Therefore, a first wall heat flux of 1 MW m^{-2} was assumed.

The blanket heat transfer has been calculated based on the configuration as shown on Fig. 1, and with different surface heat load. The calculation was done by a two-dimensional finite difference heat transfer code developed at ANL. The effect of MHD is represented by the coolant velocity profile, which is assumed to be a slug flow. The velocity is determined in an iterative mode to satisfy the following temperature: $T_{in} = 330°C$, $T_{out} = 610°C$, and $T_{structure} < 700°C$.

Fig. 3 summarizes the heat transfer calculation results for the case with 1 MW m^{-2} surface heat flux. The results indicate that a first wall heat flux of this magnitude can be handled by a self-cooled lithium blanket. This is an important conclusion because it demonstrates the capability of a self-cooled lithium blanket to be able to handle high wall loading.

6. Assessment and conclusions

The ARIES-RF is a design study to evaluate the performance of an advanced fusion power plant. A Li/V blanket was selected as the reference design for ARIES-RS. This selection was based on the performance and safety characteristics of the Li/V blanket.

Some key issues associated with the Li/V blanket have to be resolved during the next phase of fusion development. The important ones are:

1. Radiation damage effect: This is a key issue for fusion material development which is common to all structural material. Since there is no intensive 14 MeV neutron source, the effect of radiation damage to the structural material is unknown. All the radiation damage information

Poloidal Temperature Variation in Zone 1
Uniform Surface Heat Flux

BC 1 is the coolant in the first channel
BC 2 is the coolant in the second channel

Fig. 3. Heat transfer results for a surface heat load of 1 MW m^{-2}.

available is from fission spectrum, which may be very different from the fusion spectrum. In particular, the effects of He are difficult to assess. At this time, the first wall life time is set at 200 DPA. The real first wall life time may be very different from this. The 14 MeV radiation damage information will not be available until an intensive fusion neutron source is available.

2. Material fabrication: The fabrication of large V-alloy components has to be demonstrated. Based on a study for ITER, it is estimated that the development of V-alloy for ITER application will take 5–8 years and cost $130 million [13].

3. Primary loop design: For a fusion power plant with a V/Li blanket, the power conversion system will be either a high-performance steam cycle, or a closed-cycle He turbine. It is questionable if V-alloy will be compatible with either high-temperature water or high-temperature He. The structural material has, therefore, to be changed from V-alloy to some other alloy. The joining of two different structural materials will be difficult. For this design study, a concept is proposed to make this join at the double-walled IHX. Both the design of the double-walled IHX, and the performance of the join with repeated thermal cycling, have to be demonstrated.

4. V-alloy cost: To assure the low activation performance of the V-alloy, some of the impurities have to be reduced to low levels. The most important impurity is Nb, which has to be reduced to a level lower than 1 ppm. Methods are available to reduce the impurities to the level required. However, the increment of the cost is not certain.

5. Insulating coating development: The development of a reliable insulating coating is necessary for a self-cooled liquid metal blanket. Otherwise, the MHD pressure drop will be just too high. This coating has to be reliable over a long period of time, with repeated thermal cycling, under intense radiation damage, and facing high-temperature liquid lithium. Also, it has to be compatible to low activation requirements, material compatibility, as well as the tritium recovery system. The development of the insulating coating is still in an early stage and much further work needs to be done.
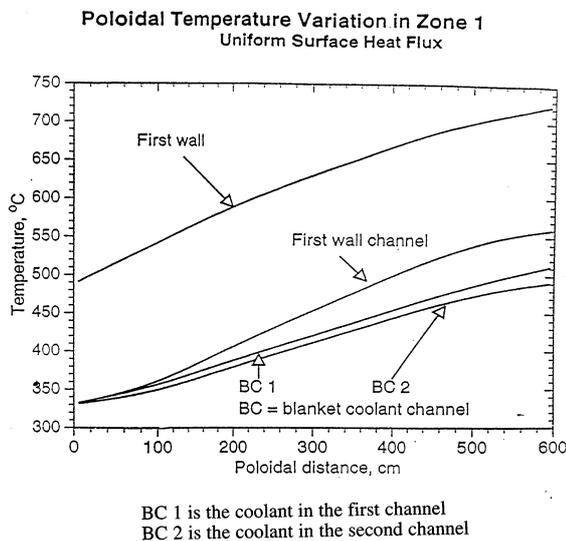
6. Tritium recovery: To recover tritium from lithium to a very low concentration (~ 1 appm) is very difficult. However, many processes have been proposed. The method proposed here is based on cold trap. There is wide experience of the cold trap process for both Na and NaK systems. Therefore, there is reason to believe that the cold trap process can be developed as designed.

7. Lithium reactivity: The chemical reactivity of the lithium is always a concern. However, many studies have concluded that a fusion power plant with a Li/V blanket can be safe. It is clear that the chemical reactivity of the lithium cannot be changed. Thus, it is up to the designers to make the design safe.

There are many other engineering issues to be resolved, such as the reliability of the blanket, the replacement of the blanket, the performance and the design of the power conversion system, etc. However, those issues are common to all the blankets.

References

[1] D.-K. Sze, M.S. Tillack, I.N. Sviatoslavsky, L.A. El-Guebaly, L.M. Waganer, Blanket selection for the Starlite project, Fusion Technol. 30 (3), Part 2B, 995.

[2] R.P. Waszink, J.Y. Hwag, L.E. Efferding, Thermal/hydraulics performance test results of sodium-heated steam generator model employing a duplex tune, Proc. AICHE–ASME Heat Transfer Conf., San Francisco, CA, August 11–13, 1975.

[3] G.M. Yasenchak, Engineering assessment of an advanced pilverized-coal power plant, EPRI Report CS-2555, Electric Power Research Institute, Palo Alto, CA, 1982.

[4] H. Moriyama, S. Tanaka, D.K. Sze, J. Riemann, A. Terlain, Tritium recovery from liquid metals, Fusion Eng. Des. 28 (1995) 226–239.

[5] V.A. Maroni, R.D. Watson, G.G. Stahl, Some preliminary considerations of a molten salt extraction process to remove tritium from liquid lithium fusion reactor blankets, Nucl. Technol. 25 (1975) 83.

[6] D.K. Sze, R.F. Mattas, J. Anderson, R. Haange, H. Yoshida, O. Kveton, Tritium recovery from lithium, based on a cold trap, Fusion Eng. Des. 28 (1995) 200–225.

[7] J.R. Watson, W.F. Calaway, R.M. Yonco, V.A. Maroni, Recent advance in lithium processing technology at Argonne National Laboratory, Proc. 2nd Int. Conf. Liquid Metal Technology in Energy Production, Richland, WA, April 20–24, 1980.

[8] C.C. McPheeters, D.J. Raue, Cold trap modeling and experiments on NaH precipitation, Proc. 2nd Int. Conf. Liquid Metal Technology on Energy Production, Richland, WA, April 20–24, 1980.

[9] J. Riemann, R. Kirchner, M. Pfeff, D. Rackel, Tritium removal from NaK-cold traps, first results on hydride precipitation kinetics, Fusion Technol. 21 (1992) 872.

[10] R.L. Eichelberger, Testing of a meshless cold trap from hydrogen removal from sodium, Proc. 2nd Int. Conf. Liquid Metal Technology in Energy Production, Richland, WA, April 20–24, 1980.

[11] D.K. Sze, R.F. Mattas, A.B. Hull, B. Picologlou, D.L. Smith, MHD considerations for a self-cooled lithium blanket, Fusion Technol. 21 (3), Part 2B, 2099–2106.

[12] J.H. Park, T.F. Kassner, R. Mattas, C. Reed, D.L. Smith, CaO insulator coating development, presented at ISFNT-4, Tokyo, Japan, April, 1997.

[13] H.K. Birnbaum, J.W. Davis, J.N. Doggett, W.D. Manly, Technical evaluation of the technology of vanadium alloys for use as blanket structural materials in fusion power systems, US DOE Report DOE/ER-0313/100, US Department of Energy, Washington, DC, 1993.