HIGH POWER DENSITY SELF-COOLED LITHIUM-VANADIUM BLANKET*

Y. Gohar, S. Majumdar, D. Smith

Fusion Power Program, Argonne National Laboratory,
9700 S. Cass, Argonne IL 60439, USA

Telephone: ++1 630 252 4816, FAX: ++1 630 252 5287, E-mail: gohar@anl.gov

RECEIVED
OCT 13 1999
OSTI

Abstract

A self-cooled lithium-vanadium blanket concept capable of operating with 2 MW/m² surface heat flux and 10 MW/m² neutron wall loading has been developed. The blanket has liquid lithium as the tritium breeder and the coolant to alleviate issues of coolant breeder compatibility and reactivity. Vanadium alloy (V-4Cr-4Ti) is used as the structural material because it can accommodate high heat loads. Also, it has good mechanical properties at high temperatures, high neutron fluence capability, low degradation under neutron irradiation, good compatibility with the blanket materials, low decay heat, low waste disposal rating, and adequate strength to accommodate the electromagnetic loads during plasma disruption events. Self-healing electrical insulator (CaO) is utilized to reduce the MHD pressure drop. A poloidal coolant flow with high velocity at the first wall is used to reduce the peak temperature of the vanadium structure and to accommodate high surface heat flux. The blanket has a simple blanket configuration and low coolant pressure to reduce the fabrication cost, to improve the blanket reliability, and to increase confidence in the blanket performance. Spectral shifter, moderator, and reflector are utilized to improve the blanket shielding capability and energy multiplication, and to reduce the radial blanket thickness. Natural lithium is used to avoid extra cost related to the lithium enrichment process.

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency thereof, nor any of their employees, make any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

DISCLAIMER

Portions of this document may be illegible in electronic image products. Images are produced from the best available original document.

1. Introduction

In previous self-cooled lithium blanket studies for magnetic fusion (1-3), the blanket concepts were developed to accommodate specific loading conditions and to satisfy predefined design guidelines. This study is intended to explore the capability of the self-cooled lithium-vanadium blanket concept for operation at high power density and high performance. Therefore, a new blanket concept has been developed, moreover the loading conditions, the energy multiplication, and the shielding capability are maximized. In addition, the radial blanket thickness is minimized to reduce the total cost of the reactor system. The blanket has liquid lithium as the tritium breeder and coolant to alleviate issues of coolant breeder compatibility and reactivity. Vanadium alloy (V-4Cr-4Ti) is used as the structural material because it can accommodate high heat loads. Also, it has several attractive features including good mechanical properties at high temperatures, high neutron fluence capability, low degradation under neutron irradiation, good compatibility with the blanket materials, low decay heat, low waste disposal rating, and adequate strength to accommodate the electromagnetic loads during plasma disruption events. Self-healing electrical insulator (CaO) is utilized to reduce the MHD pressure drop. Low pressure lithium is used to provide simple reliable design with enhanced performance. The blanket configuration is based on slotted poloidal flow concept (4). This blanket concept has high lithium coolant velocity at the first wall that reduces the peak temperature of the vanadium structure and accommodates high surface heat flux without excessive MHD pressure drop. Beryllium spectral shifter and titanium carbide moderator/reflector are utilized in the blanket to improve the blanket

performance and to reduce the total radial blanket thickness. Natural lithium is used to avoid extra cost related to the lithium enrichment process.

2. Strategy And Guidelines

The strategy of this study is to develop a high power density blanket concept, to study the blanket performance as a function of the loading conditions, and to enhance the performance of the self-cooled blanket concept. The top ranked blanket concept from the blanket comparison and selection study (1), the self-cooled liquid lithium with vanadium structure is employed as a starting point for the study. The V-4Cr-4Ti alloy is utilized for this study because of it appears to be near optimum in overall performance relative to the other vanadium alloys. Beryllium spectral shifter is incorporated in the blanket to reduce the required blanket thickness, to enhance the blanket energy multiplication factor, and to reduce the energy deposition in the shield. To achieve high surface heat flux capability for the first wall, a lithium coolant channel is located between the spectral shifter and the first wall, slotted channel design (4). This lithium coolant provides a direct heat transfer path for the surface heat flux that minimizes the maximum first wall temperature. However, the blanket neutronics requires the spectral shifter to be very close to the first wall. Neutronics, heat transfer, and hydraulics analyses were iterated to define the optimum design window for the first wall and the spectral shifter. Different reflector materials including carbon, copper, tungsten, Type 316 austenitic steel, manganese steel, vanadium alloy (V-4Cr-4Ti), calcium oxide, tungsten carbide, titanium carbide, zirconium carbide, and calcium carbide were considered in the analyses to define the best possible blanket performance. In addition, assessment was

carried out for these materials to define the best possible candidates for the moderator and reflector zones. Then, lead, and zirconium were analyzed with the titanium carbide, the top ranked moderator/reflector material to study the possibility of replacing the beryllium spectral shifter. The health hazard associated with the beryllium dust promoted this analysis.

The top ranked spectral shifter and moderator/reflector were utilized for the blanket design where neutronics, heat transfer, and hydraulics analyses were further iterated to satisfy the different design guidelines. The first wall/blanket/shield design and optimization system (BSDOS) was used to perform the iteration. BSDOS (5) represents the new generation of the blanket design tools featuring modular design and graphic interfaces. New capabilities have been added to the BSDOS to analyze the self-cooled blanket concepts. BSDOS provides fast and accurate tool to design, analyze, and optimize the performance of the first wall/blanket/shield concept based on multi-dimensional analyses.

Special attention has been given to the first wall design because of its impact on the allowable loading conditions, the attainable structural lifetime, the blanket structural design and the module toroidal span. The ITER Structure Design Criteria (ISDC) were applied to the first wall design. ISDC provide primary and secondary stress limits as functions of the reduced uniform elongation and ductility of the material with neutron fluence. The first wall lifetime was analyzed as function of the surface heat flux, the lithium coolant pressure, the toroidal span of the blanket module, the second wall

thickness, and the side wall thickness of the blanket module. The results from these analyses defined the first wall design, the main structure requirements of the blanket module, and the first wall lifetime.

Material issues have been carefully considered to insure satisfactory blanket performance. Baseline mechanical properties were compiled for the design process including the effect of neutron irradiation. Vanadium compatibility with the multiplier and reflector materials was assessed in the presence of the thermal bonding material. Current results from the electrical insulator R&D work for the lithium channels were reviewed to insure design consistency. Exploratory tests have demonstrated that a highly resistive calcium oxide coating can be formed on vanadium alloys. Also, self-healing characteristics have been demonstrated in preliminary experiments.

Several design guidelines were adopted for this study to insure a significant enhancement in the blanket performance. The reference loading conditions are 2 MW/m² surface heat flux and 10 MW/m² neutron wall loading. The energy multiplication and the shielding capability of the blanket are maximized. The local tritium-breeding ratio is maintained above 1.3 to insure tritium self-sufficiency with natural lithium. Simple blanket configuration is emphasized to reduce the fabrication cost, to improve the blanket reliability, and to increase confidence in the blanket performance. The blanket thickness and the energy deposition in the shield are minimized to reduce the capital cost of the reactor system. Low activation materials are selected for the different blanket components to enhance the environmental acceptance of the fusion system.

Electrical insulator is utilized for the lithium channels to maintain low-pressure system and long lifetime for the blanket structure. The temperature range of 400 to 800 °C is considered for the vanadium structure. The minimum temperature is intended to insure that the vanadium structure operates above the temperature at which significant irradiation embrittlement has been observed. The upper temperature is selected to provide adequate creep strength and to avoid high temperature helium embrittlement.

3. Spectral Shifter Materials

Neutron multiplier material is usually utilized for solid breeder blanket concepts to enhance its tritium breeding capability. Lithium has a good tritium breeding capability without neutron multiplier. In addition, the blanket has a low volume fraction of the vanadium alloy, which has insignificant impact on the neutron economy. These characteristics of the self-cooled lithium-vanadium are responsible for its poor neutron attenuation performance, which necessitates a large radial thickness to utilize the fusion neutrons for tritium production and to convert its kinetic energy to sensible heat. The use of a spectral shifter reduces the large mean free path of the DT fusion neutrons without significant energy loss through endothermic nuclear reactions and neutron loss through parasitic absorption. In this blanket concept, the spectral shifter is intended to reduce the required blanket thickness, to enhance the blanket energy multiplication factor, and to reduce the energy deposition in the shield. These changes significantly enhance the performance of the self-cooled lithium-vanadium blanket concept.

Several studies (6-8) were performed to define neutron multiplier, spectral shifter, and energy converter materials for fusion blankets. Beryllium, lead, and zirconium have been found to have the highest potential if the fissionable materials are excluded. High (n,2n) and low absorption cross sections are the main factors for achieving the required nuclear performance. Zirconium has the lowest (n,2n) cross section and the highest absorption cross section among the three materials. The high melting temperatures of beryllium and zirconium permit its operation without phase change. Beryllium has a health hazard associated with its dust that requires special handling procedure. In addition, the blanket design has to accommodate its swelling and the small amount of tritium produced during operation. Beryllium utilization in the blanket significantly enhances the energy multiplication factor and the tritium breeding ratio. This performance is unique for beryllium relative to the other two materials. Lead has low melting temperature that necessitates its use in the liquid phase to avoid volume changes during start up and shut down. Lead produces ^{210}Po and ^{205}Pb . ^{210}Po is an α -emitter, which is produced from neutron irradiation of bismuth. Bismuth is a transmutation product of lead by neutron capture and a natural impurity of lead. Such blanket requires on-line removal of polonium or its precursor bismuth to limit the ^{210}Po concentration in lead. ^{205}Pb is very long-lived isotope which decays by electron capture. Zirconium produces ^{93}Zr and ^{93}Nb isotopes, which have very long half-lives. In addition, zirconium generates more decay heat relative to beryllium and lead.

4. Moderator And Reflector Materials

Moderator and reflector materials are used to slow down the high-energy neutrons for absorption in the lithium and to decrease the loss of neutrons from the blanket by scattering back many of those that have escaped. The best moderator and reflector materials are consisting of elements of low mass number with low absorption cross section. In addition, this material has to contribute to the energy multiplication in the blanket, which favors materials with exothermic nuclear reactions. Water is excluded because of its interaction with the liquid lithium. Beryllium is also excluded because of the desire to reduce the total blanket material cost. Carbon, copper, tungsten, type 316 austenitic steel, manganese steel, vanadium, calcium oxide, tungsten carbide, titanium carbide, zirconium carbide, and calcium carbide are examined to select the most promising material(s).

Carbon has the lowest mass number possible to use and low absorption cross section, which improves the tritium breeding capability of the blanket. In addition, carbon has low activation characteristics. Carbide materials are considered for their carbon content and stability at high temperature. Copper is included because of its high thermal conductivity, which eases the blanket thermo-mechanical design. The good shielding capability of the tungsten promoted its consideration. The good fabrication characteristics, the low unit cost, and the current database of the austenitic stainless steel lead to its consideration. Calcium oxide was selected before as reflector material for the self-cooled lithium-vanadium blanket design (9) because it has low activation characteristics and low unit cost. Parametric analysis was performed to compare the

blanket performance with each material. Cost, activation, decay heat, shielding performance, and material issues were considered to select the best candidate(s).

5. Blanket Design

The developed blanket design is shown in Figure 1, which is capable of 2 MW/m^2 surface heat flux and 10 MW/m^2 neutron wall loading. The heat transfer results show that the 4-mm thick first wall has a maximum temperature of 754°C . In addition, this first wall satisfies all the structural design rules. The first wall structural analysis shows that the primary stress is controlled by the coolant pressure and the geometrical parameters of the blanket. The coolant pressure is only 0.5 MPa. The geometrical parameters include first wall span, first wall curvature, first wall thickness, side wall thickness, and second wall thickness. In contrast, the thermal stress is relatively insensitive to the same parameters. However, because of the combined effect of primary and secondary stresses on creep ratcheting, the blanket geometrical parameters have a significant influence on the creep ratcheting lifetime of the blanket. Thus, by suitably choosing the blanket geometrical parameters and the coolant temperature, the thermal loading capability of the blanket can be optimized with a reasonably long creep ratcheting and creep rupture lifetime. For example, Figure 2 show the effect of the coolant temperature on the creep ratcheting lifetime, which shows that the first wall lifetime in this design is 27.2 years or 272 MW.y/m^2 . Fatigue, creep-fatigue, and brittle fracture from pre-existing flaws are not performed because of the missing experimental data for the vanadium alloy under consideration. Swelling and fatigue will define the ultimate first wall lifetime.

The iteration of the neutronics, the heat transfer, and the thermal hydraulics analyses resulted in a total blanket radial thickness of 570 mm, which satisfies all the design guidelines. The radial thickness of the beryllium spectral shifter is 35 mm. The clad thickness of the spectral shifter is determined by the first wall structural requirements. An 8-mm thick clad is used on the front section facing the plasma and 2 mm on the backside. The maximum temperature of the spectral shifter is 800 °C that occurs at the coolant exit location. The spectral shifter has no mechanical or structural function in the blanket, therefore the high temperature degradation in its mechanical properties is acceptable. The titanium carbide moderator/reflector has a total radial thickness of 290 mm. This thickness is divided to four zones and each zone is cooled from both sides. These zones have unequal thicknesses to match the exponential distribution of the nuclear heating. The iteration was performed to achieve a maximum temperature less than 900 °C in each zone. The high melting point and the good thermal stability of the titanium carbide material allow its operation at high temperature. In addition, the reflector has no structural or mechanical function in the blanket. The blanket performance parameters are given in Table 1. The blanket energy multiplication factor is 1.33. The blanket deposits only 2.5% of the total nuclear heating in the shield. The tritium breeding ratio is 1.32 with natural lithium. The maximum vanadium temperature is 754 °C. Sodium is used as a thermal bound between beryllium or titanium carbide and the vanadium structure. In addition, the sodium bonding provides a mechanism to accommodate the differential thermal expansion and material swelling. This bonding technology was developed and used for both functions successfully in fuel rods of

fission reactors. In the spectral shifter zone, beryllium will getter any oxygen in sodium and prevent vanadium oxidation. A slight excess of titanium in the titanium carbide will prevent any carburization of the vanadium alloy.

In this design, the inlet lithium is used to cool the first wall with high velocity before removing the blanket nuclear heating. The high lithium velocity enhances the heat transfer coefficient and the use of the inlet lithium for the first wall reduces the maximum vanadium temperature to lowest possible value. In each blanket module, all the mass flow is introduced in the first channel (first wall channel) to cool the first wall first then the mass flow is distributed to all the other blanket channels at the blanket bottom. All the blanket cooling channels have the same radial thickness, which satisfy the neutronics performance and provide about the same mass flow rate in each blanket channel. The radial thickness of the first wall channel can be equal or exceed the radial thickness of the blanket channel. The poloidal length of the blanket module and the lithium velocity define the radial thickness of the first wall channel.

The first wall structural analysis results show that the allowable toroidal span for the current blanket configuration is 0.4 m. Assuming ARIES-RS reactor geometry (10), the poloidal length of the inboard blanket is 4.4 m. For the high loading conditions of table 1, the required coolant mass flow rate produces lithium velocities of 6.06 and 1.21 m/s in the first wall and the blanket channels, respectively, assuming 0.03 m radial thickness for each coolant channel. For the outboard blanket, the poloidal length is ~1.6 times the

inboard, which requires a 0.048-m radial thickness for the first wall coolant channel for the same maximum lithium velocity and the 200 °C increase in the lithium temperature.

In the thermal-hydraulics analysis, no improvements are considered due to the toroidal magnetic field although the design uses the slotted channel concept (4). For the slotted channel with turbulent flow, the magnetic field changes the flow pattern, which effects the heat transfer characteristics of the channel. First, the velocity and the velocity gradients near the channel wall parallel to the magnetic field are increased to generate an M-shaped velocity distribution, which has a positive effect on the heat transfer coefficient. Second, the velocity fluctuations (eddies) are damped, which has a negative effect on the heat transfer coefficient. These two computing effects define the net impact on the heat transfer coefficient. The limited experimental results (11) for rectangular channel flow in transverse magnetic field with heat transfer shows that MHD heat transfer is 1.2 – 1.3 times better than turbulent flow without magnetic field for Peclet number of 200 and Hartmann number of 800. Also, the experimental results show that the heat transfer coefficient increases as the Hartmann number changed from 0 (no magnetic field) to 800 or the Peclet number changed from 12.5 to 200. The MHD pressure drop is less than 0.064 MPa. This pressure drop does not cause any concern for the blanket design however, the total pressure drop is dominated by the pressure drop in the headers and the bottom section of the blanket. The blanket module is designed for 0.5 MPa lithium coolant pressure.

6. Conclusions

A self-cooled lithium-vanadium blanket has been developed with Beryllium spectral shifter and titanium carbide moderator/reflector. The blanket has very attractive features and enhanced performance. It is capable of operating with 2 MW/m² surface heat flux and 10 MW/m² neutron wall loading. The lithium outlet temperature is 600 °C to enhance the plant thermal efficiency. The blanket radial thickness is reduced to 0.57 m that has a significant influence on reducing the total reactor cost. The energy multiplication and the shielding capability are maximized while the energy deposition in the shield is minimized. The structural analysis of the first wall shows that its maximum thermal capability is determined not only by the thermal and mechanical properties of the first wall material but also by the geometrical parameters of the blanket.

*Work supported by the Office of Fusion Energy Sciences, U.S. Department of Energy, under Contract No. W-31-109-Eng-38.

References

- [1] D.L. Smith, et al., "Blanket Comparison and Selection Study - Final Report," Argonne National Laboratory Report, ANL/FPP-84-1 (1984).
- [2] Y. Gohar and S.T. Yang, "Energy Deposition and Shielding Requirements for All Concepts of The Blanket Comparison and Selection Study," Fusion Technology 8 (1985) 2010-2020.
- [3] Y. Gohar, "Trade-off Study of Liquid Metal Self-Cooled Blankets, Fusion Technology, 10 (1986) 1449-1445.
- [4] I. V. Lavrent'ev, "Liquid Metal Blanket of Fusion Tokamak Reactor," IV All Union Conference in the Engineering Problems of Fusion Reactors, Leningrad, (1988) 326-327.
- [5] Y. Gohar, C.Baker, H. Attaya, Y. Cha, S. Majumdar, and T. Scandora, "First Wall/Blanket/Shield Design and Optimization System," Fusion Engineering and Design 9 (1989) 375-379.
- [6] H.I. Avci, Y. Gohar, T. Y. SUNG, G. L. Kulcinski, and C. W. Maynard, "The Use and Performance of Graphite and Metal ISSECS in Tokamak Fusion Reactor, " Nuclear Engineering Design 45 (1978) 285-310.
- [7] Y. Gohar, "An Assessment of Neutron Multipliers for DT Solid Breeder Fusion Reactor," Transactions of American Nuclear Society 34 (1980) 52.
- [8] Y. Gohar and M.Abdou, "Neutronic Optimization of Solid Breeder Blankets for STARFIRE Design, " Proceedings 4th ANS Meeting on The Technology of Controlled Nuclear Fusion, King of Prussia, Pennsylvania October (1980) 628-641.

- [9] Y. Gohar, "Design Analysis and Optimization of Self-Cooled Liquid Metal Blankets," *Fusion Engineering and Design* 8-10 (1989) 71-77.
- [10] F. Najmabadi and the ARIES Team, "Overview of ARIES-RS Tokamak Fusion Power Plant, " *Fusion Engineering and Design* 39-40 (1998).365-370.
- [11] C. B. Reed, T. Q. Hua, I. R. Kirillov, S. I. Sidorenkov, A. M. Shapiro, and I. A. Evtushenko, "Liquid Metal MHD and Heat Transfer in A Tokamak Blanket Slotted Coolant Channel," *Proceedings of IEEE/NPSS 15th Symposium on Fusion Engineering*, Hyannis, MA (1993).263-272.

Table 1. Blanket Performance Parameters with 2 MW/m² surface heat flux and 10 MW/m² neutron wall loading

Parameter	Value
Blanket energy multiplication factor	1.33
Tritium breeding ratio	1.32
Shield energy fraction	0.025
Blanket radial thickness, m	0.57
Lithium enrichment	Natural
Lithium coolant inlet temperature, °C	400
Lithium coolant outlet temperature, °C	600
Max. vanadium temperature, °C	754
Max. beryllium temperature, °C	803
Max. titanium carbide temperature, °C	866
Lithium coolant velocity, m/s	
First wall coolant channel	6.06
Blanket channels	1.21
First wall lifetime for 1% strain, y	27.2

Figure 1. Plan view of the self-cooled lithium-vanadium blanket module design

Figure 2. Allowable surface heat flux versus first wall lifetime based on creep ratcheting rules for different bulk coolant temperatures and the coolant pressure.

Fig 1