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HELIUM–COOLING IN FUSION POWER PLANTS

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ABSTRACT

This paper reviews different helium-cooled first wall and blanket designs; and compares the selection of structural materials. We found that the solid breeder, SiC-composite material option generates the lowest amount of induced radioactivity and afterheat and has the highest temperature capability. When combined with the direct cycle gas turbine system, it has the potential to be the most economical fusion system and can compete with advanced fission reactors. When compared to martensitic steel and V-alloy, SiC-composite is the least developed of these three structural materials, a focused developmental effort will be needed. Fundamental research has begun in addressing the issues of optimized composite materials, irradiation effects, leak tightness and low activation braze materials. Development of helium-cooled high heat flux components and further development of the direct cycle gas turbine system will also be needed.

1. INTRODUCTION

Fusion power has the potential to be a safe and environmentally acceptable power source in the future. To realize this potential, fusion power plants must be economically competitive and environmentally attractive by controlling capital cost, achieving high reliability and minimizing the amount of generated radioactive waste and radioactive sources potentially available for release during normal and abnormal events. These attributes can be satisfied by the proper selection of suitable systems and materials for the tokamak first wall, blanket, and divertor components. For the purpose of comparison, a reasonable economic goal for fusion power plants is to compete with advanced fission reactors which may be economically available for hundreds of years. Projected future power costs associated with uranium fueled plants is about 45–60 mills/kWh in 1994$. To meet this economic goal fusion power plants must have high thermal conversion efficiency to compensate for the relatively high capital cost per thermal kilowatt. A way to reach high thermal efficiency is to operate the working fluid at high exit temperature. Compared to coolants like water and liquid lithium, helium has the potential to meet these economic and environmental attractiveness requirements.

2. HELIUM COOLANT

Helium coolant has been used for fission reactors and conceptual fusion power plant designs. The technology base of helium-cooling stems from the successful application to several fission reactors including Peach Bottom and Fort St. Vrain in the U.S. and HTR and THTR in Germany. For fusion application, advantages of helium-coolant for fusion application include its chemical inertness, its transparency to neutrons, and its stable heat transfer regime. Depending on the blanket coolant exit temperature, helium is applicable to either the Rankine or Brayton cycle power conversion system. The disadvantages of helium are its low density and correspondingly low volumetric specific heat. These characteristics lead to very low shielding effectiveness, modest heat transfer coefficient, and the requirement of relatively high pumping power. To alleviate these problems high coolant pressure designs in the range of 5–18 MPa have been studied [1,2,3].

3. MATERIAL SELECTION OF HELIUM-COOLED FUSION POWER PLANT DESIGNS

Helium-coolant has been used for different fusion power plant designs, cooling the first wall, blanket, shield and divertor components. Examples of helium-cooled designs are reported in the Blanket Comparison Study (BCSS) [4], Demonstration Power Reactor study in Europe [5,6], and ARIES designs [1,2] in the U.S. Helium cooling was also suggested for the ITER-EDA design [3]. A variety of structural materials were selected for these designs.

The BCSS designs employed martensitic steel as the structural material, and solid ceramic like Li_2O, Li_4SiO_4 or $LiAlO_2$, as the tritium breeder. Similarly the recent European designs used martensitic steel as the structural material, and solid breeder, $LiAlO_2$ or Li_2ZrO_3 [5] and liquid breeder, Pb-17Li [6] as tritium breeding materials. ARIES helium-cooled designs used SiC-composite as the structural material and solid ceramic, Li_2O or Li_2ZrO_3 as the breeder. The proposed ITER-EDA design [3] used martensitic steel as the structural material and stagnant lithium as the tritium breeder.

For the US Starlite Demo study, we are evaluating two helium cooled design options. The first one uses vanadium alloy (V-alloy) as the structural material and the second one uses SiC-composite as the structural material. These helium-cooled design options can have the coolant routing arranged in the poloidal or toroidal direction. For example, the toroidal flow configuration can be similar to the ARIES-I nested shell design which is shown in Fig. 1. The coolant can enter the blanket in the poloidal direction at the back plenum of the blanket and then distribute through the radially oriented pipes and flow toroidally cooling the first wall and blanket before returning to the back plenum of the blanket module. This configuration can be applied to metallic alloy and ceramic composite structural material design options. The innovation for the V-alloy helium cooled design is the use of a mixed tritium breeders of Li_2O and Li. This Li_2O/Li mixture can potentially control the problem of compatibility between V-alloy and the impurities of oxygen and hydrogen in the helium coolant. Instead of forming V-hydride and V-oxide which can weaken the structural material, the presence of lithium would lead to the formation of more stable lithium hydride and Li_2O. Feasibility of this suggestion will need to be demonstrated by experiment. Other advantages of this design option are the possibility of breeding adequate tritium without the use of Be neutron multiplier, and the elimination of the problem of contact resistance between solid breeder and structural wall. The helium coolant outlet temperature for this option is limited by the V-alloy which has a maximum temperature limit of 700°C, due to helium embrittlement. Therefore the thermal efficiency is limited to the steam Rankine cycle system which may have a gross thermal efficiency of 40% to 45%. The second option being evaluated by the US Starlite Demo study is the helium-cooled, solid breeder with SiC-composite structural material. By changing the structural material from V-alloy to a SiC-composite which has a higher temperature limit in the range of 1000°C to 1200°C, the coolant outlet temperature can be operated in the range of 850 to 1000°C, which allows the use of a closed cycle helium gas turbine Brayton cycle with a gross efficiency of 50%–55%. It should be noted that the mixed Li_2O/Li breeder approach will not be applicable for the SiC-composite material, due to the incompatibility of SiC and lithium at the temperature range of 900–1200°C [7].

4. MATERIAL PERFORMANCE COMPARISON AND ECONOMICS

The above mentioned materials: martensitic steel, V-alloy and SiC-composite materials are considered to be reduced or low activation materials. Activation levels for these materials as a function of time after reactor shut down is illustrated in Fig. 2 (HT-9 is representative of a reduced activation alloy of martensitic steel). It can be noted from Fig. 2, that during the period after shutdown to about 1 year after shutdown, SiC has the lowest level of induced radioactivity. In addition, SiC also has the lowest afterheat which implies a very low energy source term for radioactivity release. As a ceramic with a decomposition temperature of 2600°C, when compared with other materials, it will also have the highest capacity to absorb thermal energy under accidental conditions.

With the goal of determining the economic, safety and environment potential of tokamak fusion power plant, under the US-ARIES project [1,2,8] we have evaluated different blanket designs. The two higher performance ones are helium-cooled, SiC/SiC composite, solid breeder design, that we called ARIES-I [1,2] and the lithium-cooled, V-alloy, lithium breeder design that we called ARIES-II [8]. Using an advanced supercritical Rankine steam cycle design, at a net thermal efficiency of 39% for both designs, the corresponding cost of electricity for ARIES-I and ARIES-II are 75 and 60 mills/kWh, respectively. These are still higher than the projected advanced fission reactor cost of about 45–60 mill/kWh [9].

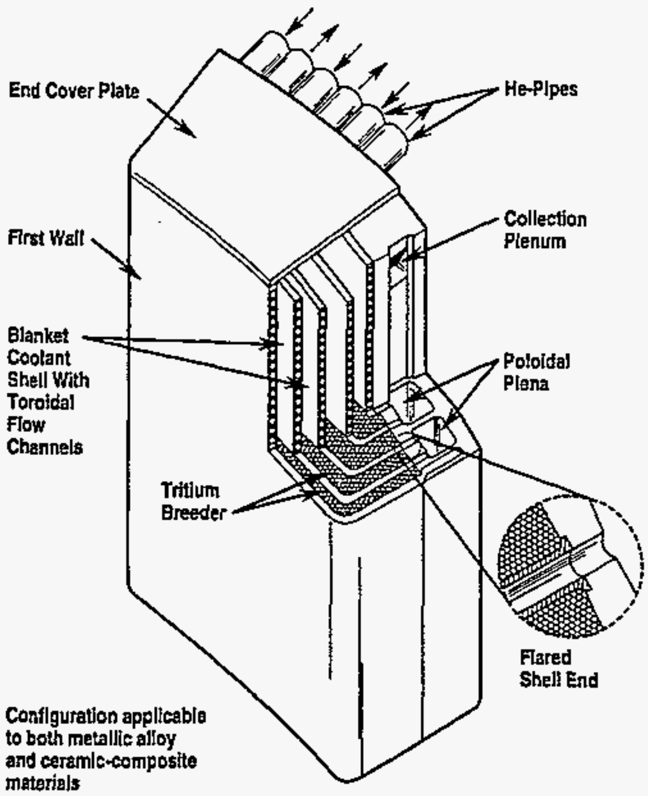

Fig. 1. ARIES–I nested shell blanket module.

For comparison, Table 1 shows the maximum temperature limit for HT–9, V-alloy and SiC-composite materials. These limits were derived from the projection of radiation damage of structure materials operated under the fusion neutron flux and fluence conditions. These radiation damages can be caused by displacement of atoms and/or by the generation of helium. The latter can lead to swelling and or embrittlement of the material followed by the loss of structural integrity. Table 1 shows that the combination of helium coolant and SiC-composite has the highest performance potential *i.e.* highest cooling outlet temperature for fusion power plant, mainly due to the higher temperature capability of SiC-composite.

Correspondingly, Fig. 3 shows the gross plant efficiency as a function of direct cycle gas turbine inlet temperature, which is the same as the blanket coolant outlet temperature. As the turbine inlet temperature increases, higher thermal efficiency can also be reached. Similarly, Fig. 4 shows the corresponding decrease of the cost of electricity with the increase of turbine inlet temperature. At a coolant temperature of 1100°C, the cost of electricity is about 55 mills/kWh, which is close to the project cost of advanced fission light water reactors. In summary, based on the desire of maximizing the turbine inlet temperature, only the helium-cooled, SiC-composite combination has the potential to compete economically with advanced fission power plant. As indicated in Fig. 2, the Helium-cooled, SiC-composite option also has the best potential to achieve the goal of inherent safety and the ease of waste disposal for fusion power plant.

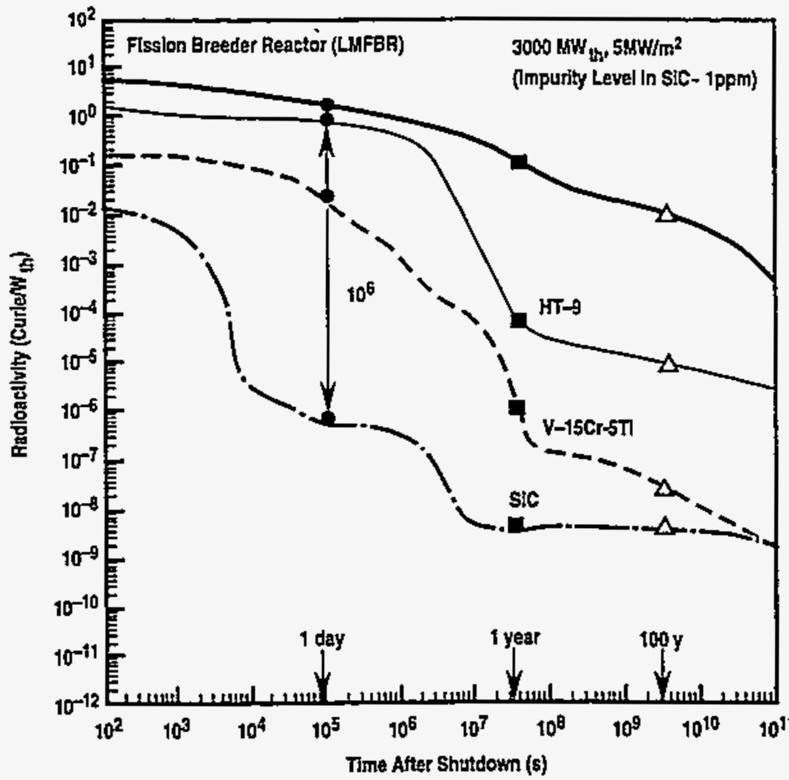

Fig. 2. Activation levels for typical tokamak power plant normalized to 3000 MW_{TH} and neutron wall loading of 5 MW/m^2. Low activation materials e.g., (SiC) reduce activation levels by 6 orders of magnitude (activation is dominated by impurities; high–purity SiC has ppb impurity levels). Other materials are V-alloy (V-15Cr-5 Ti), martensitic steel (HT–9). Comparing to Liquid Metal Fast Breeder fission reactor (LMFBR).

TABLE 1
Key Design Parameters for Different Structural Materials

	FERRITIC STEEL	V–ALLOY	SiC/SiC COMPOSITE
Material T_{max}, °C	550	700	≥1000
Coolant/T_{out}, °C	He/520 (Rankine Cycle)	Li/600 (Rankine Cycle)	He/650–750 Rankine cycle
Coolant/T_{out}, °C	LiPb/425 (Rankine Cycle)	He/550 (Rankine Cycle)	He/≥850 (Brayton cycle)
Industrial maturity	High	Low	Very low

5. DEVELOPMENT REQUIREMENTS

For the development of helium-cooled, SiC-composite structural fusion power plant design, there are several key technical issues that must be addressed. They are the development of composite material, high heat flux removal components, and direct cycle gas turbine.

5.1. SiC-COMPOSITE MATERIAL

SiC-composite is a low activation material with minimal afterheat and induced radioactivity. Compared to martensitic steel and V-alloy, SiC-composite is in the early stages of development as a structural material for fusion, as indicated in Table 1.

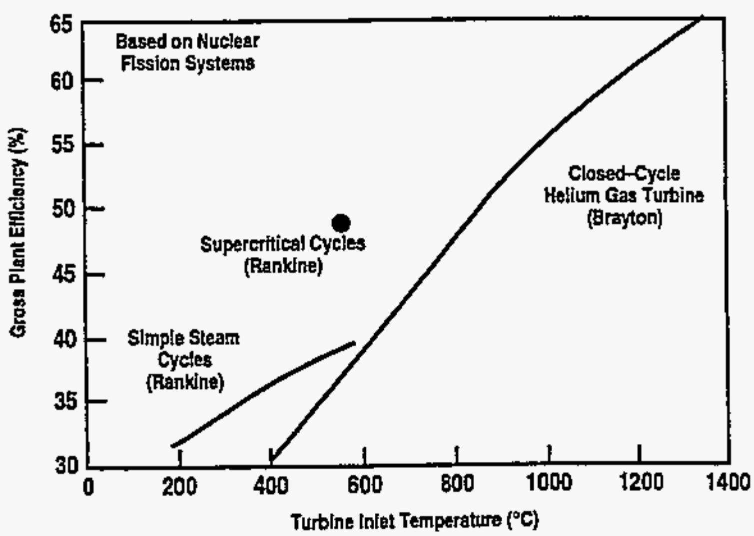

Fig. 3. Gross plant thermal efficiency as a function of turbine inlet temperature for Rankine and Brayton cycle power conversion systems.

Martensitic steel as a class is a mature structural material that has been used for many industrial components like steam generators. Relatively, the application of V-alloy is limited. There is essentially no industrial experience in the design and fabrication of significant structures from any V-alloy but fabrication tests have demonstrated that V-alloy exhibits fabrication properties similar to the stainless steels. If this fact is substantial by further test the capability of fabricating large structural components is available. For the SiC-composite material, it is being developed for different applications as rocket nozzles and advanced heat exchangers. However, the development of fusion relevant, high performance, helium-cooled SiC-composite components is in its infancy. Some of the critical issues of irradiation properties and lifetime, leak tightness, brazing and joining of composite parts, large components design and fabrication will need to be addressed. In the U.S., some of the SiC-composite material development efforts include the following: Relatively small samples (5×5 cm) were undergoing neutron irradiation tests in fission reactor at Pacific Northwest National Laboratory. Oak Ridge National Laboratory and General Atomics are developing optimized SiC-composite materials, investigating the selection of fiber, fiber coating, and matrix materials. At General Atomics, we have fabricated leak tight tubes which have been pressurized to 13 MPa at an elevated temperature of 1050°C. We are also developing low activation brazing materials for SiC-composite components. These are necessary but limited efforts. In order to match the U.S. development schedule of fusion power plants, *i.e.* demonstration power plant by the year 2025 and commercial power plant by the year 2040, a focused and well supported SiC-composite material development program will be needed.

5.2. HELIUM COOLING FOR HIGH HEAT FLUX COMPONENTS

A key concern in the use of helium as the coolant is its capability of removing high surface heat fluxes, this is especially true for the cooling of tokamak divertors at a maximum surface loading of about 5 MW/m^2. Experimental results of using helium at 4 MPa pressure to remove surface heat fluxes of 8 MW/m^2 [10] and 16 MW/m^2 [11] have been demonstrated. Copper alloy was used in these experiments. For the case of removing the surface heat flux of 8 MW/m^2, the test module consisted of a heated length of 80 mm and a width of 25 mm, fin height of 5 mm, fin pitch of 1 mm and a fin thickness of 0.4 mm. The corresponding pumping power fraction was only 0.8% of the removed thermal power. For the case of removing surface loading 16 MW/m^2, with a heat removal area of 1 cm^2, a porous medium configuration was used instead of the fin configuration. Similar demonstration components will need to be developed for SiC-composite material.

5.3. DIRECT CYCLE GAS TURBINE SYSTEM

Compared to other fusion related systems, the direct cycle gas turbine system is relatively well developed. The largest closed helium Brayton cycle turbine constructed to-date is the 50 MWe Oberhausen plant in Germany. Latest advances in helium Brayton cycle designs for individual components (turbo-machinery, heat exchangers, *etc.*) have been constructed in the size, temperature and pressure range required by fusion power reactors. [12] Comparatively, high temperature, high performance systems would likely to be available well ahead of fusion power plant development schedule.

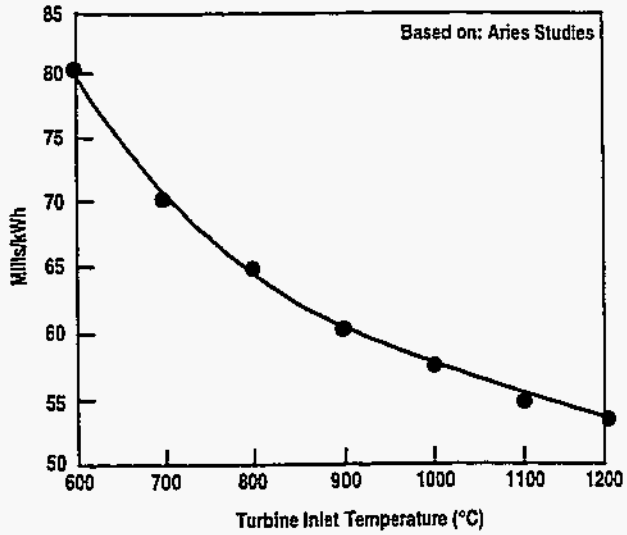

Fig. 4. Fusion economic potential depends on turbine inlet temperature, the reference point is based on AIRES–I [1] results.

6. CONCLUSIONS

Comparing different blanket designs of fusion power plant and the materials selected, we found that the helium-cooled, SiC-composite, solid breeder design, combined with the direct cycle gas turbine system has the best potential to compete economically with advanced fission reactors. This system also has the lowest induced radioactivity and afterheat, thus can lead to significant benefits in safety and licensing of fusion power plants. The key requirements for this power plant system are the development of the SiC-composite structural material, the development of SiC-composite high heat flux removal components and the further development of the direct cycle gas turbine system. Fundamental development on the optimization of irradiated SiC-composite materials, leak tightness and low activation brazes, and materials have been initiated, but a significant development program will be necessary to bring this system into reality.

ACKNOWLEDGMENT

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