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ABSTRACT

This paper reports the results of the second phase evaluation of the EVOLVE W-alloy first wall and blanket design cooled by vaporizing lithium. For the transpiration-cooled first wall and blanket concept, we identify the need to further quantify the data of lithium superheat from W-alloy heated surface and bulk lithium slabs. For the boiling lithium blanket, we identify the need to elucidate the impacts of magnetic field on various stable-boiling regimes. We also find that this FW/blanket concept should have no problem in achieving adequate nuclear performance. With the addition of passive cooling loops, the concept has a strong possibility of achieving the safety requirement of not needing a public evacuation plan under the loss of power accident conditions. Even though the irradiated W-alloy may be subject to embrittlement, due to the relatively low system pressure of the design, it is shown to be able to withstand a large number of cracks. The fundamental issues of W-alloy properties under high neutron fluence irradiation and the components fabrication technique remain. Preliminary investigations of W-alloy fabrication and heat flux removal through SBIR programs have begun and the initial results are encouraging. We recommend that the investigation of critical issues of the EVOLVE concept continue since this innovative design has a good possibility of showing a way to achieve high performance and passively safe designs that are necessary for the utilization of fusion power.

I. INTRODUCTION

To achieve high thermal performance at high power density, the Evaporation of Lithium and Vapor Extraction (EVOLVE) W-alloy first wall and blanket (FW/blanket) concept uses the vaporization of lithium to remove the FW/blanket thermal power. The lithium is maintained at the saturation pressure of 0.037 MPa and the high lithium outlet temperature of 1200°C leads to a helium closed cycle gas turbine efficiency of ~57%. The scoping design of this FW/blanket concept was completed in the first phase of the APEX work.[1] The basic design of the EVOLVE FW/blanket configuration is shown in Fig. 1. The first wall is composed of toroidally oriented tubes. Inside these tubes there are smaller feeding tubes connected to the first wall fluid channel formed by the capillary screen on the one side and the first wall on the other side. The blanket design schematic in Fig. 1 shows the lithium blanket trays. The lithium is fed by filling up the top tray until it is overfilled and drains to the lower trays. This process of overfilling the top tray and draining to the lower trays forms the supply approach of the coolant lithium. The vaporized lithium comes off the lithium surface and moves to the outlet vapor plenum located at the back of the blanket. For this phase of the EVOLVE concept evaluation, we focus on addressing critical issues. The areas of evaluation are summarized in this paper. More detailed descriptions of selected critical areas are presented in accompanying papers presented in

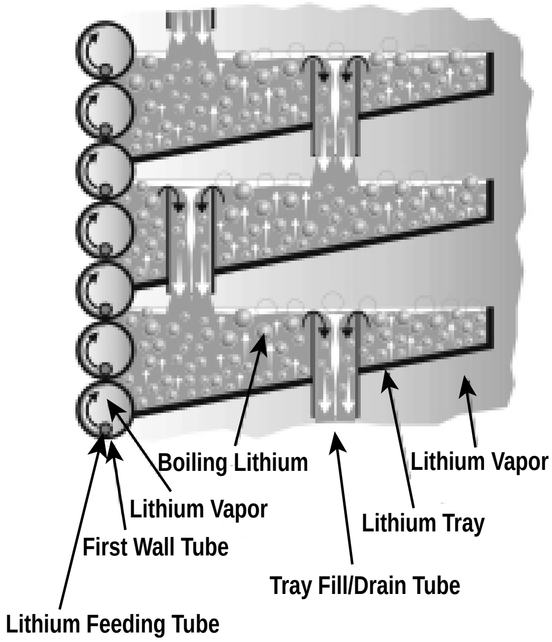

Fig. 1. Schematic of EVOLVE first wall and boiling blanket concept.

this conference. This paper begins with a review of the material assessment. Detailed analysis of the thermal-hydraulics of the transpiration FW design led to the concept of also cooling the blanket with the transpiration cooling approach, which is also presented in this paper along with the evaluation of the boiling blanket. Neutronics performance of these two blanket options is summarized. Due to the projected radiation and high temperature helium embrittlement and high afterheat properties of the W-alloy, the issues of coolant leakage to the plasma chamber and passive safety were addressed. Initial demonstration of W-alloy fabrication and the discussion of necessary experiments to address critical issues are also presented in this paper.

II. MATERIALS ASSESSMENT

The key structural materials issue for the EVOLVE concept is the compatibility of the material with the lithium that serves as the coolant and tritium breeder. Secondary issues include "low temperature radiation embrittlement" (typically important in all body center cubic (BCC) alloys for temperatures below ~0.3 melting temperature (TM), and doses above ~1 dpa), "high temperature helium embrittlement" (typically important in all materials for temperatures above 0.5 TM and >100 appm He concentrations), joining and fabrication issues (unirradiated components and repair welds of irradiated materials), creep strength and fracture toughness, and tritium solubility and permeability. Considering the high temperatures of the lithium vapor, the low mechanical stresses, and safety/afterheat issues, tungsten alloys appear to be the best choice for the structural material for temperatures above 1000°C.[2] Tungsten alloys have very good compatibility with lithium up to ~1400°C,[2] and offer a high thermal stress capability compared to most other materials. The main challenges associated with utilization of W alloys are fabrication difficulties as presented in Section VIII.

III. FIRST WALL AND BLANKET DESIGN

Figure 1 shows the initial configuration of the EVOLVE transpiration-cooled first wall and boiling lithium blanket concept. During this second phase of work we analyzed the thermal hydraulics of the first wall design and found that the transpiration cooling approach could also be applied to the blanket design. This approach has a more complicated configuration than the lithium blanket, but it has the advantage of being a very passive system. In contrast, the boiling blanket design has a much simpler configuration but requires the identification of stable operating regimes of boiling lithium in a magnetic field. Since both are innovative concepts that have no previous applications, we decided to evaluate both transpiration-cooled and boiling-cooled blanket options, and the results are summarized in the following sections. For the following the lithium vapor is nominally at 1200°C and saturated pressure of 0.037 MPa, with a first wall surface heat flux of 2 MW/m^2 and a neutron wall loading at 10 MW/m^2.

A. The Transpiration-Cooled First Wall and Blanket Concept with Poloidal First Wall Channels

In the transpiration-cooled concept, the heat from the solid first wall and the volumetric heat deposited in the blanket is removed by evaporation through capillary walls. The porous structure of these walls pumps the lithium by the capillary effect out of a sump to the evaporating surface formed by the capillaries with diameter d_C. Additionally, the fluid height in the tray is used to support this passive capillary pumping. Concepts with toroidally and poloidally oriented first wall channels are investigated. In this section we present the results of the poloidal design to illustrate the cooling concept. The routing of the liquid and vapor lithium of the poloidal flow first wall design is shown schematically in Fig. 2, and the orientation of the magnetic field is indicated. The coolant lithium is routed from the back of the blanket to the bottom of the vertical blanket slab, form by capillary sheet walls, to the first wall, and then through the first wall channel defined by the capillary sheet. The thin, 0.5 mm thick capillary sheet defines a first wall channel width (w) of 0.75 mm on the inner side of the first wall.

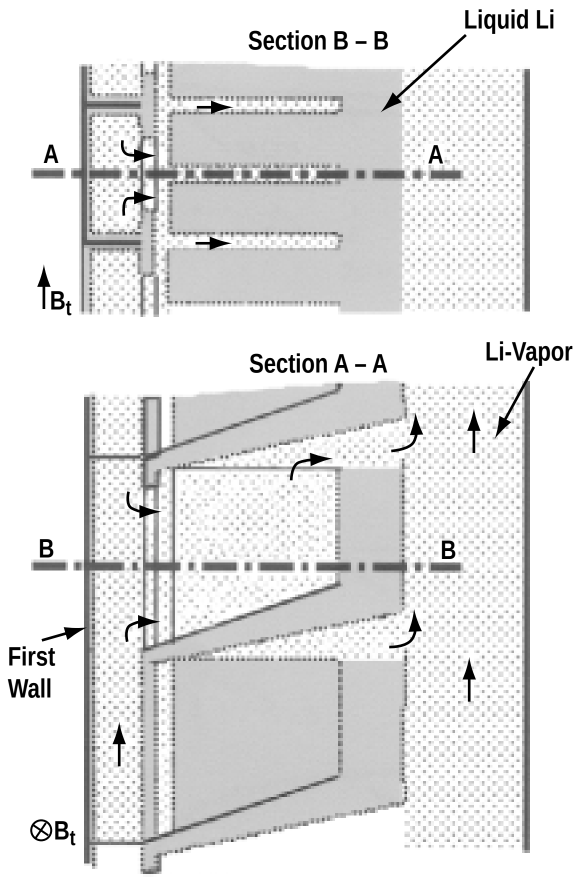

Fig. 2. The transpiration-cooled first wall and blanket schematic (section B-B is the top view and section A-A is the side view of the design).

In the blanket region 2 mm thick capillary sheets are used everywhere, insuring transpiration cooling of the Li-slab held within the sheets and within the back and bottom ducts feeding the Li to the first wall channels. The vaporized Li from the capillaries escapes between the lithium slabs. The transpiration-cooled concept works when the sum of the active and passive pumping head is higher than the total system pressure losses and when the temperature at the inner side of the first wall does not override the superheating limit of the coolant which would be about ΔT_{SH}=100 K for very smooth surfaces.[3] The pressure losses, especially in the liquid Li, are governed by magnetohydrodynamic (MHD) effects induced by the strong magnetic field and the pressure drop across the evaporation capillary surface. The first wall feeding gap width w is the critical dimension that determines both the first wall MHD pressure drop and the coolant temperature at the first wall. The selected width of 0.75 mm has been optimized for this design with the capillary diameter of 0.4 mm. The widths of the different liquid Li slabs, which are heated volumetrically and cooled at their surfaces by transpiration cooling, are defined by the maximal allowed superheating at their centerline. For this case, the superheat of 100 K was assumed before the formation of homogeneous nucleation. The calculations are conducted using the thermo-physical data of Li and tungsten from V.A. Kirillin[4] and N.B. Vargaftik[5] and applying correlations given in Ref. 3. Key blanket parameters for the design shown in Fig. 2 are given in Table I.

B. Boiling Blanket

In order to determine whether the EVOLVE fusion boiling blanket design is viable, thermal-hydraulic analyses were performed on the outboard liquid lithium blanket trays to determine the vapor fraction of the blanket. This will also help to identify the scenario of operating at different stable boiling regimes for the blanket concept. As shown in Fig. 1, trays of nuclear heated liquid lithium are stacked poloidally behind the first wall. The issue of volume fraction is important because it affects the tritium breeding and shield performance of this boiling lithium blanket. Three boiling/ evaporation scenarios are considered. Details of this work is presented in an accompanying paper in this conference.[6]

The first scenario explored assumes negligible magnetic field coupling to the liquid lithium pool. A standard drift-flux model[7] and considering the churn-turbulent boiling regime, was used to empirically fit data from liquid metal experiments utilizing nitrogen gas as the vapor, was applied to the liquid lithium trays. In the calculation, the tray was divided into five radial segments, and each segment was analyzed individually with the corresponding nuclear power density used as input. All the deposited nuclear power from the W-structure and the liquid lithium are utilized to vaporize the lithium. This scenario results in a maximum vapor fraction of up to 65% at the top of the pool.

Table I. Design parameters of the transpiration-cooled first wall and blanket design

FW surface heat flux	MW/m^2	2
Toroidal magnetic field strength	Tesla	6.0
Thickness of the W-alloy FW	mm	3.0
Thickness of the FW capillary screen	mm	0.5
Capillary open area	%	50
Capillary diameter	mm	0.47
Thickness of the FW/blanket capillary sheet	mm	2.0
FW radial feeding gap width	mm	0.75
True superheating at FW ΔT_{SH}	K	54
Blanket Li system pressure	MPa	0.037
Blanket Li slab thickness	cm	2.5–6.9
Δ P (Capillary + hydrostatic)	Pa	3707
ΔP-FW/blanket system	Pa	3674
Lithium Tmax	K	1514
W-alloy Tmax	K	1597

A second scenario[8] assumes that magnetic field effects are moderately coupled to the liquid lithium, and the lithium vapor escapes through vertical vapor channels. We define moderate magnetic interaction as one where the magnetic field influences the liquid lithium by strongly dampening its motion, but does not affect the nucleate boiling process. Mass, momentum and energy balances are performed to validate the potential heat removal scenario where these vertical vapor channels are held open by vapor momentum, frictional and magnetic field effects. The estimated vapor fractions for the vaporizing lithium pool based on this scenario are significantly reduced to the range of 6%–12%.

The third scenario involves the potential for large magnetic field coupling with the liquid lithium, whereby the nucleate boiling process is also affected. This can have impacts on the formation of the vapor channel as evaluated in the second scenario. The potential for these phenomena is being reviewed, and experiments to resolve the magnitude of the magnetic force on the liquid lithium pool are proposed and summarized in Section VII.

IV. NUCLEAR ANALYSIS

The neutronics performance of the EVOLVE concept was analyzed using two-dimensional calculations. Detailed results are given in a companion paper.[9] Iterations with the Li tray boiling analysis were carried out to determine a consistent set of void fraction and nuclear heating distributions. The boiling analysis, using the drift-flux model yields large void fractions of up to 65%. Conservative nuclear performance parameters were determined using these worst-case conditions. The overall tritium-breeding ratio (TBR) is 1.33 without breeding in the divertor region. The results imply that ~70% of the total thermal power is deposited as high grade heat in the front evaporation cooled zone and is carried by the Li vapor to the heat exchanger. Adequate shielding is provided for the TF-coils with all magnet radiation limits being satisfied with a large margin.

The impact of the higher vapor fraction in the Li trays was found to be relatively small. The TBR for the worst case conditions with highest vapor fraction is ~5% lower than that with 8% vapor fraction that might be achieved using a flow pattern with triggered vertical vapor channels, e.g. by engineered surface roughness.

The neutronics parameters for the preliminary design of the transpiration blanket option were compared to those obtained for the boiling blanket with the largest predicted vapor fractions. The results indicate that with these assumptions, the transpiration blanket has larger Li and structure content resulting in slightly higher TBR and a factor of 2–5 better shielding performance. The nuclear performance parameters for both designs are acceptable with large margins, implying that the nuclear performance is not going to be the deciding factor if one were to choose between the transpiration and boiling blanket concepts.

V. LITHIUM LEAKAGE ASSESSMENT

As mentioned in the materials assessment section, low temperature radiation embrittlement and high temperature helium embrittlement are issues related to the use of W-alloy. Since embrittled material is more likely to have enhanced crack growth, we evaluated the related issue of lithium leakage. Two problems may arise if fatigue cracks in the first wall of EVOLVE penetrate the full wall thickness. First, the leakage of lithium atoms into the plasma chamber may adversely impact plasma performance if the leakage rate exceeds 2×10^{20} atoms/m^2/s. Second, excessive leakage of the coolant may lead to localized heating of the first wall due to insufficient cooling. The nominal coolant mass flow required for cooling the first wall of EVOLVE in the region of the crack of 2.5 cm long is ~0.2 g/s. Calculations were performed to estimate the leakage rate of lithium through a fatigue crack in the first wall and to see if the two concerns are realistic.

Calculating the leak rate of high temperature lithium through a tight (low coolant pressure) and tortuous fatigue crack is not an easy task. Keeping in mind that the pressure of liquid lithium at the inlet to the fatigue crack is close to its saturation vapor pressure, as the coolant enters the crack, the viscous pressure drop will cause liquid lithium to flash to a vapor phase. The surface heat flux, as well as the surrounding liquid at high temperature, will supply the heat of vaporization. In fact, flashing will occur even without a surface heat flux. The local velocity of the vapor immediately after flashing will be very high, but it will be quickly damped by high viscous dissipation. Because of the large flow path length (3 mm) to width ratio (L/δ), equilibrium will be restored within some multiples of the crack width (10 µm) close to the entrance. If we assume that the pressure drop within this transition zone is small compared to the overall pressure drop, the mass flow rate will be determined by equilibrium of pressure and viscous forces. Such calculations showed that the mass flow rate of lithium vapor through a 25 mm long, 10 µm wide fatigue crack is 5×10^{-4} g/s at the assumed pressure of 0.15 MPa, which is much higher than the operating saturated pressure of the 0.037 MPa of the EVOLVE first wall. For a first wall area of 500 m^2, this leakage rate corresponds to 10^{17} atoms/m^2/s. Thus, a large number of fatigue cracks can be tolerated in the first wall of EVOLVE without compromising plasma performance or the cooling of the first wall.

VI. SAFETY

The EVOLVE lithium vapor cooled innovative FW/blanket concept allows low system pressure with the relatively slow circulation of liquid lithium. However, the afterheat of W-alloy is relatively high. It becomes necessary to evaluate the possibility of passive safety for this lithium coolant and high afterheat FW/blanket system. The reference point we use to assess fusion safety is the DOE Fusion Safety Standard.[10] This standard was developed to enumerate the safety requirements associated with D-T magnetic fusion facilities. One requirement called out in this standard is that: "the need for an off-site evacuation plan shall be avoided." This requirement translates into a dose limit of 10 mSv at the site boundary during a worst-case accident scenario. An accident that qualifies as a worst case scenario for the EVOLVE facility is a complete loss-of-power accident.[1] This scenario results in a complete loss of active cooling of the in-vessel components and an induced plasma disruption that could simultaneously cause window failure of a vacuum boundary port, allowing air from an adjoining room to enter the plasma chamber. The consequence would be the heating of in-vessel components by decay heat, release of plasma facing component (PFC) tritium inventories, and mobilization of PFC activation products by surface oxidation. A summary of the evaluation of the passive temperature control under the loss of power accident and the accidental releases are given in the following two sections.

A. Temperature Control from Afterheat

The accident scenario of loss of site power has been analyzed with the MELCOR code[11] modified to treat lithium as a coolant.[12] A one-dimensional model of the EVOLVE design was developed that accounts for thermal conduction and radiative heat transport from the inboard primary vacuum boundary (PVB) to the outboard PVB. The inboard thermal boundary condition is adiabatic, and the outboard thermal boundary condition is thermal convection and radiation to ambient temperatures. Lithium convection within the in-vessel components was included. The decay heat for the EVOLVE design at plasma termination is 55 MW. To aid in the transport of this heat in the radial direction, the passive natural convection system employed by the ARIES-RS design[13] was also used for EVOLVE. This system contains depleted lithium (Li-7) and consists of rectangular channels (40 mm × 80 mm) that are attached to the lithium tray back-plate and blanket stiffening ribs of the inboard and outboard primary blankets. These channels connect with similar channels attached to the wall of the outboard low temperature shield (LTS). During normal operation, MHD forces will minimize lithium circulation in this system, but under accident conditions these loops will self-initiate once the toroidal field coils have discharged.

Figure 3 shows temperatures predicted for the EVOLVE design during this accident scenario. From an initial operating temperature of 1400°C, the FW temperatures drop below 900°C in two hours, below 800°C in 13 hours, and below 700°C in 68 hours. The lithium velocity in the natural convection loops is initially 0.6 m/s, but gradually drops to 0.33 m/s by seven days. The LTS cooling system has been modeled as an active system capable of removing a maximum of 110 MW for this analysis. However, analyses are ongoing to develop a passive natural convection model for this cooling system.

B. Accidental Releases

To evaluate the release of in-vessel inventories during the loss of power and air ingress accident scenario, the free volume of the plasma chamber, a non-nuclear room, a bypass duct that connects the plasma chamber and the non-nuclear room, and the heating-ventilation-air-conditioning (HVAC) system duct that connects the non-nuclear room to the environment were modeled. The modeling assumptions for these items are based on default parameters from the ITER EDA safety study.[14] The HVAC system was assumed to stop at the beginning of the event due to the power outage.

The rate of mobilization of activation products by oxidation of the FW tungsten alloy was based on temperature dependent data from Smolik.[15] In addition,

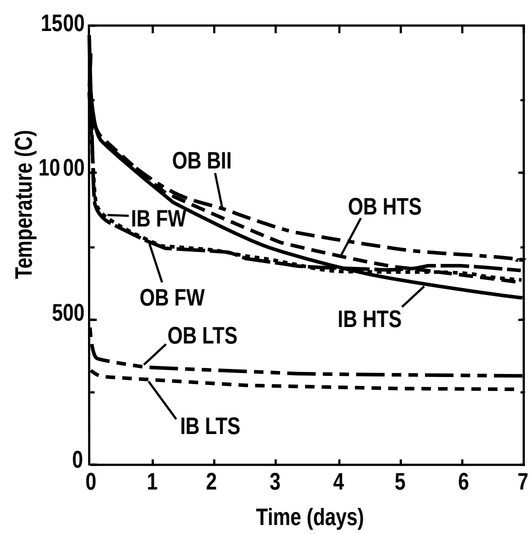

Fig. 3. Evolve component surface temperatures during a complete loss of power accident scenario.

the plasma disruption at the beginning of this event produces an additional 10 kg of dust. Both of these aerosol sources where given an initial mass mean diameter of 0.1 µm.[14,15] The initial specific activity of this aerosol mass was calculated from EVOLVE activation calculations[1] to be about 3.3×10^5 Ci/kg. In addition to this aerosol source, the release of the tritium from the PFCs must also be accounted for. Because implanted tritium inventories for EVOLVE are not available, the predicted ARIES-AT divertor inventory of 0.4 g/m^2 was adopted for this study.[16] When this inventory is adjusted for temperature differences between the ARIES divertor and the EVOLVE FW, the estimated EVOLVE FW value is 0.18 g/m^2. The resulting total FW inventory would be 76 g, all of which would be immediately released. Radiological dose calculations[17] for ground level releases of various radionuclides (1-km site boundary assuming average weather conditions) give a specific dose for the tungsten alloy aerosol of 130 mSv/kg, and for the tritium (HTO) of 67 mSv/kg. More than 90% of the tungsten alloy aerosol dose is due to Re184 and Re184 m.

Figure 4 contains the predicted tungsten alloy mass mobilized during the first week of this accident and the amount of this mass that is released to the environment by way of the HVAC duct if filtration is neglected. The total amount of tungsten alloy aerosol mobilized is 357 kg (95% of which is release during the first 2.3 days), and the amount vented to the environment without filtration is 7.8 kg. Given the additional dose of about 5 mSv associated with the HTO release, the non-nuclear bypass room would have to be isolated within 2.4 hours in order to meet the non-evacuation dose goal of 10 mSv if no filtration of the HVAC duct flow occurs. This time period could be extended to 1.6 days with filtration (99% filtration efficiency).[18] Both of these response times are reasonable. Therefore, EVOLVE can easily meet the non-evacuation goal for this accident.

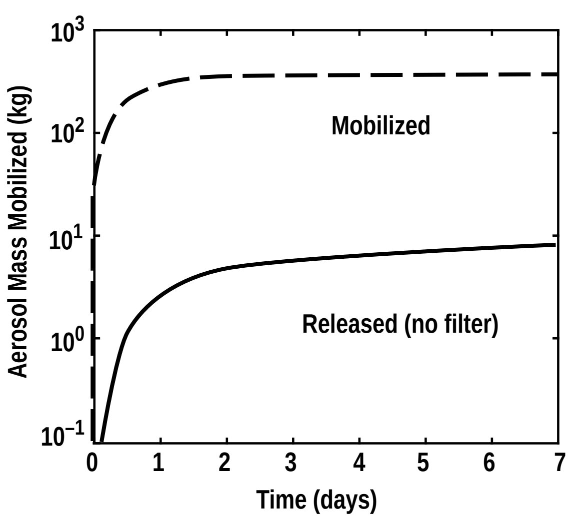

Fig. 4. Tungsten alloy mass mobilized and released during a complete loss of power accident scenario.

VII. FUNDAMENTAL HEAT TRANSFER EXPERIMENTS

The vaporizing lithium EVOLVE concept offers the possibility of passive heat removal at high temperature. The transpiration first wall approach is crucial to both the transpiration and boiling blanket concepts. To project the heat transfer performance of the transpiration first wall and blanket design, it is essential to quantify the degree of superheat from the first wall and from the bulk of the lithium slabs. Experiments are being planned to address this need. For the boiling blanket concept, based on previous work[19] and the current state of knowledge regarding pool boiling of liquid metals in the presence of a magnetic field, experiments are also needed to identify stable boiling regimes. Experiments are needed to determine the magnetic field effects on the nucleation and subsequent transport of bubbles from the W heated surface and the lithium in the boiling tray under the scenarios of startup to steady state operation. Real-time visualization of the developing flow patterns would be of great help in understanding the phenomena. Then a physical model explaining the effects of the magnetic field on the onset of boiling and the boiling flow regime can be developed. These experiments will allow us to determine the boiling rate and heat transfer for a given volumetric heat flux, which in turn will lead to the pool depth needed to balance heat generation and heat removal.

VIII. W-ALLOY FABRICATION, TESTING AND EXPERIMENTS

One of the main challenges associated with the EVOLVE concept in the utilization of W alloys is the need to develop suitable fabrication technologies. Satisfactory full-penetration welds have not been developed for W, despite intensive efforts over a >25 year time span (1960-1985). To initiate the evaluation we make use of the W-alloy fabrication technology being developed for the helium-cooled FW/blanket design.

Over the last four years, better fabrication technology for helium-cooled heat sinks, specifically the very large effective surface area for heat transfer in a small volume that porous media can provide, has led to dramatic increases in the performance of these heat sinks.[20-22] A description of the tungsten porous metal heat exchanger experiment is presented in a separate paper.[23] The thermacore porous tungsten module was fabricated in four separate assembly-brazing steps, all using BAu-4 (Nioro™) braze filler metal, either 0.010" or 0.020" diameter wire, in dry hydrogen atmosphere.

Both Thermacore and Ultramet, Inc. have been developing refractory helium-cooled modules through grants from DOE's Small Business Innovative Research (SBIR) Program. Ultramet designs and builds commercial products made of refractory metals for rocket nozzles and other applications. Ultramet has also used a process in which they build up refractory material with chemical vapor deposition to create a metallized foam that is integrally bonded to fully dense material to fabricate W tubes with integrally bonded porous W sections in the interior. These experiences will be useful to the development of W-alloy components in the future.

IX. CONCLUSIONS

This paper reports the results of the second phase evaluation of the EVOLVE W-alloy FW/blanket concept that is cooled by vaporized lithium. Results of the first phase evaluation are presented in the APEX interim report.[1] During the second phase, critical areas were evaluated and the results have provided us with further confidence on the possibility of utilizing this innovative concept to achieve a passively safe, under a loss-of-power accident, and high thermal performance FW/blanket design for fusion power reactors. Many fundamental issues remain and many of them can only be addressed by experiments. For the transpiration-cooled first wall and blanket concept, we have identified the need to further quantify the data of lithium superheat from W-alloy heated surface and bulk lithium slabs. For the boiling lithium blanket we have identified the need to elucidate the impacts of magnetic field on various stable-boiling regimes. We also found that this FW/blanket concept should have no problem in achieving adequate nuclear performance. With the addition of passive cooling loops, the concept has a strong possibility of achieving the safety requirement of not needing a public evacuation plan under a loss of power accident. Even though the irradiated W-alloy has projected problems in embrittlement due to the relatively low system pressure of the design, it is shown to be able to withstand a large number of cracks from lithium leakage with the initial crack length of 25 mm and a width of 10 μm. The fundamental issues of W-alloy properties under high neutron fluence irradiation and the technique of component fabrication remain. Preliminary investigations of W-alloy fabrication and heat flux removal through the SBIR program have begun, and initial results are encouraging. We recommend that the investigation of critical issues of the EVOLVE concept should continue since it has a good possibility of showing a way to achieve high performance and passively safe FW/blanket designs that are necessary for the utilization of fusion power.

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