PROGRESS ON DCLL BLANKET CONCEPT

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Under the US Fusion Nuclear Science and Technology Development program, we have selected the Dual Coolant Lead Lithium concept (DCLL) as a reference blanket, which has the potential to be a high performance DEMO blanket design with a projected thermal efficiency of >40%. Reduced activation ferritic/martensitic (RAF/M) steel is used as the structural material. The self-cooled breeder PbLi is circulated for power conversion and for tritium breeding. A SiC-based flow channel insert (FCI) is used as a means for magnetohydrodynamic pressure drop reduction from the circulating liquid PbLi and as a thermal insulator to separate the high-temperature PbLi (~700°C) from the helium-cooled RAF/M steel structure. We are making progress on related R&D needs to address critical Fusion Nuclear Science and Facility (FNSF) and DEMO blanket development issues. While performing the function as the Interface Coordinator for the DCLL blanket concept, we were developing the mechanical design and performing neutronics, structural and thermal hydraulics analyses of the DCLL TBM module. We estimated the necessary ancillary equipment that will be needed at the ITER site, and a detailed safety impact report was prepared. This provided additional understanding of the DCLL blanket concept in preparation for the FNSF and DEMO. This paper is a summary report on the progress of the DCLL TBM design and R&D for the DCLL blanket concept.

I. INTRODUCTION

Both the US ARIES and test blanket module (TBM) studies have identified the Dual Coolant Lead Lithium (DCLL) as the primary high performance blanket concept for the US. This concept has been explored extensively in the US (Ref. 1,2), and the European Union (Ref. 3) uses PbLi liquid metal as breeder and coolant and helium as the primary structural coolant. Reduced activation ferritic martensitic (RAF/M) steel is the structural material. This design includes the SiC flow channel insert (FCI) as a required functional material to provide electrical and thermal insulation between the breeder and structural material. In support of the ITER Test Blanket Module (TBM) program, a preliminary DCLL TBM design was completed in 2005 (Ref. 4). From July 2009 to June 2012 the US was the Interface Coordinator developing the DCLL TBM design and corresponding ancillary equipment for ITER port 18. After a short description of the DCLL TBM design, including neutronics and safety assessments, this paper focuses on the research and development of the DCLL blanket concept in the areas of MHD, SiC material, FCI, structural material, and tritium production, as well as neutronics measurements.

II. DCLL TBM DESIGN

With the selection of RAF/M steel as the structural material, we are limited to a maximum steel structure temperature of <550°C and a PbLi and RAF/M interface corrosion temperature limit of ≤480°C (Refs. 4,5). At the same time, we need to remove the first wall (FW) heat flux, breed adequate tritium for the D-T fuel cycle and achieve high coolant outlet temperature for high power conversion efficiency when the design is used for DEMO. We conducted the development of the DCLL-TBM design for the D-T testing phase module [Fig. 1a] for ITER with the goal that the concept would provide adequate performance when it is extrapolated to DEMO (Ref. 4). The DCLL TBM vertical section view through the PbLi channels is shown in Fig. 1b. It has a two-pass poloidal PbLi flow configuration. The PbLi enters from the upper part of the module from the back, flows down the inside PbLi channels and turns at the bottom of the module before flowing up the outside channels to mitigate the risk of "hot spots" due to reverse fluid currents resulting from the buoyancy forces in the opposing flows. The PbLi poloidal channels are lined with SiC FCI to minimize MHD and thermal losses. The helium coolant enters from the bottom of the module and splits into two counter-flow circuits cooling the first wall in the toroidal direction while moving upward. After exiting the internal structure the helium is then routed out through the back of the TBM.

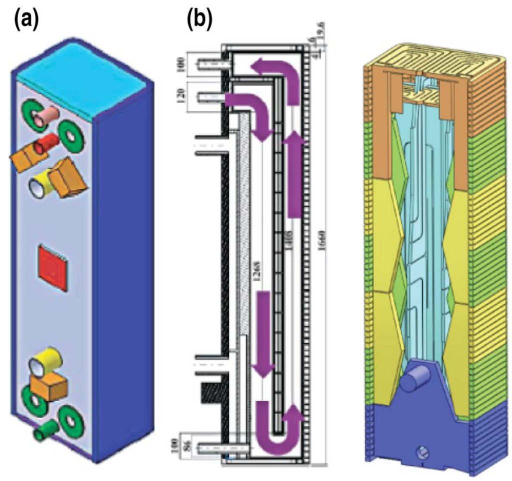

Fig. 1. (a) DCLL TBM module showing the coolant connections at the back and structural supports. (b) Poloidal PbLi flow, inlet and outlet channels located at the top, SiC FCIs are used in these channels (c) Helium flow distribution panels at the back to the first wall, inlet located at the bottom and outlet located at the top of the module.

The configuration has gone through a few design iterations, and we have improved the design by alleviating hot spots and increasing the evenness of the helium flow. Relocating the drain opening between the PbLi channels has alleviated a neutron and gamma heating hot spot. The internal structure within the PbLi channels has also been re-designed such that low flow or stagnant flow areas were eliminated. A sketch of the latest design is shown in Fig. 1c. In order to demonstrate the applicability of the first wall channel to the DEMO design, we continue to recommend that the first wall cooling channel surface on the plasma side be designed and fabricated with a one-sided 2-D roughened surface in order to enhance the local heat transfer coefficient and minimize the helium coolant pressure drop. Work remaining includes investigating the structural integrity of the entirety of the TBM under disruptions. Also, the structural supports where the TBM is mounted onto the port should be designed and analyzed.

III. NEUTRONICS

The primary function of the breeding blanket in a fusion power plant is to breed enough tritium to compensate for the tritium burned in the plasma. In addition, along with the shield, it should provide adequate protection for the lifetime of components such as the vacuum vessel (VV) and the TF coils. We performed a scoping analysis for the DCLL blanket when used in a DEMO plant with a major radius of 5.8 m, aspect ratio of 2.6, and a fusion power of 2116 MW. For this double null divertor design the overall tritium breeding ratio (TBR) is ~1.17. Three-dimensional neutronics calculations were performed also for the Advanced Tokamak version of the Fusion Nuclear Science Facility (FNSF-AT) utilizing the DCLL blanket.[6] The design has a major radius of 2.49 m and a minor radius of 0.71 m. It has a fusion power of 240 MW and allows for neutron wall loadings as high as 2 MW/m^2 and fluences of 3-6 $MW\text{-}yr/m^2$. The TBR was calculated to be 1.0 for the DCLL blanket excluding breeding in the double null divertor and the 16 mid-plane ports. Breeding in the 16 ports would add ~6% to the TBR.

The detailed design of the DCLL TBM for ITER revealed significant geometrical complexity that could influence the neutronics features of the DCLL blanket. Detailed 3-D results were generated for volumetric nuclear heating, tritium breeding, and material radiation damage using the exact CAD model.[7] Modeling the geometrical details resulted in significant local effects that should be taken into account. For example, detailed nuclear heating profiles should be taken into account in the PbLi MHD analysis and structural analysis of the blanket. We also found that tritium breeding and structural damage are affected by the geometrical complexity. It is essential to develop tools that would allow using a common CAD domain representation that facilitates the transfer of important quantities between the neutronics and other engineering analyses such as activation, computational fluid dynamics (CFD), and thermo-structural analysis to yield an accurate assessment of the DCLL blanket in a DEMO and future fusion reactors.

Under the severe high-energy neutron environment of D-T fusion systems, we found that the SiC FCI used in the DCLL blanket suffers significant transmutation. This results in changes in the properties that could affect its function as electric and thermal insulator. The type and amount of metallic transmutants produced in SiC composites can affect the electronic and thermal properties. Up to ~1.3% metallic transmutants are generated in the SiC FCI at the expected blanket lifetime of ~20 MWy/m^2 for DEMO.[8] Irradiation tests for structural materials are usually performed in fission reactors such as HFIR at ORNL. We found that irradiation in fission reactors to the same fast neutron fluence yield about an order of magnitude lower metallic transmutation products (dominated by P) than from a fusion spectrum where Mg is the dominant product. Accordingly, we will need to perform experiments using ion implantation and/or high-energy neutron sources in addition to modeling, to provide insights on the effects of radiation on SiC properties.

IV. SAFETY AND PrSR

The safety hazards associated with the US DCLL TBM have been examined in several design studies,[9,10] with the most detailed assessment presented in the US DCLL TBM Preliminary Safety Report (PrSR).[11] In these studies the safety and environmental impacts of this design were examined with the ultimate goal of ensuring that this blanket concept will not pose a risk to the public or plant workers, or create an operational or long-term radiological impact on the environment. These studies all identify that the largest radiological hazards are those associated with the dust generation by plasma erosion of the blanket module first wall, oxidation of module structures at high temperature in air or steam, inventories of tritium implanted in or permeating through the ferritic steel structures of the blanket module and blanket support systems, and the Po-210 and Hg-203 produced in the PbLi breeder/coolant. Other mechanisms for damaging the barriers confining these toxic radionuclides are the mechanical failure associated with the high-pressure helium coolant, and chemical releases associated with any hydrogen produced by possible PbLi steam or water reactions.

A failure mode and effects analysis (FMEA) was performed on the DCLL Test Blanket System (TBS), which includes the TBM.[11] The FMEA results that identified design basis events were analyzed with the MELCOR code to determine the safety consequences of coolant leaks from the DCLL TBS into the ITER vacuum vessel (VV), inter-space area, port cell, tokamak cooling water system vault, hot cell, and tritium building. Ultimate safety margin events were also analyzed. As for the overall impact on ITER safety of the DCLL TBS, based on the accidents analyzed, the impact is small. The increase in VV pressurization from helium and PbLi spilling into the VV, which spill initiates a plasma disruption that causes the ITER first wall (FW) to fail, is < 4% higher than a similar FW failure event anticipated for ITER, which is an ITER in-vessel coolant leak. The VV bypass event resulting from a helium spill into the inter-space area releases 40 times less of the ITER VV radioactive inventory to the environment than the ITER VV bypass event created by an ITER divertor cooling system ex-vessel large coolant pipe break. The inventory of tritium in the TBS is three orders of magnitude less than that in the ITER VV; and if all of the Po-210 and Hg-203 generated by the DCLL TBS over its lifetime were released to the environment, the dose to the public would be 600 times less than the ITER dose goal of 25 mSv for the most exposed individual at the ITER site boundary during conservative weather conditions.

The safety area that proved to be a larger challenge for the DCLL TBS is the occupational radiation exposure (ORE) hazards associated with TBS during maintenance activities or accident remediation.[11] For example, an ionizing radiation field produced by Pb-203 and ferritic steel (FS) corrosion in PbLi will be present even after the liquid metal has been drained from the TBS. The ~50 μm film that will be left behind emits a gamma field predicted to result in a contact dose rate between ~15 to ~80 μSv/h (the ITER administratively imposed maximum dose rate for hands on maintenance is 100 μSv/h). Based on the anticipated repair frequencies (from FMEA results), inspection, and TBM replacement activities, the estimated annual dose for DCLL TBS maintenance activities is 8.2 p-mSv/a, which is larger than the total goal for all six ITER TBS of 5 p-mSv/a. As a consequence, remote equipment and portable shields should be considered to further reduce the dose commitments associated with maintaining the DCLL TBS.

The possibility of inhalation doses during maintenance also exists from tritium permeating through components and volatile activation products (in particular Po-210 and Hg-203) being released from opening of the TBS. ITER will administratively control these doses by establishing radiation and ventilation zone requirements that are based on a derived air concentration (DAC). A DAC is the air concentration of an airborne radionuclide that equals the limiting or maximum inhalation dose of 20 mSv for one year (or a dose rate of 0.01 mSv/hr) as established by ITER procedures. A DAC is isotope dependent. For example, for tritium one DAC is a tritium air concentration of 3.4×10^5 Bq/m^3. For Po-210 and Hg-203 the concentrations are 4.1 Bq/m^3 and 9.1×10^3 Bq/m^3, respectively. For the ventilation rate of the ITER port cell, the predicted tritium permeation rate from the TBS during operation results in a DAC of ~135, where the limit for the port cell for all isotopes combined is 2.5 DACs. As a consequence, guard pipes or a TBS enclosure in which the atmosphere will be actively swept and cleaned of any tritium are under consideration. Of particular concern are Po-210 and Hg-203 exposure during maintenance activities that requires opening the TBS to the port cell atmosphere, because Po-210 is 10^5 times more hazardous than HTO, and Hg-203 is 10^2 times more hazardous than HTO. Fortunately, PbLi is a very low vapor pressure fluid, and the Po-210 and Hg-203 inventories are small and relatively immobile in solidified PbLi. However, caution should be used in opening any system that contains activated PbLi films or pools. Sweep gases, temporary glove boxes, and respirators will be procedurally employed for worker safety. These design improvements will be needed in the design of a DEMO PbLi heat transport system.

V. RESEARCH AND DEVELOPMENT

V.A. MHD Effects

At present, understanding the underlying physics of MHD flows in the DCLL blanket conditions and their impact on the blanket performance remains incomplete even at a qualitative level. Addressing MHD phenomena under such conditions is difficult due to their non-linearity, multi-scale nature, and complex geometry. Full numerical simulations for real flow configurations are often limited to relatively low values of operation parameters, *e.g.* magnetic field strength. The experimental limitations are caused by the requirements for a large magnet workspace, strong prototypical magnetic fields and prototypical neutron sources. Here, we summarize the most important MHD results from the DCLL TBM studies.[12]

Two types of induced electric current can circulate in the blanket. Cross-sectional currents are dominant in the poloidal ducts, where the flow over the major length is about fully developed with only small variations along the flow path. The MHD pressure drop associated with these currents can be reduced by orders of magnitude through a proper choice of SiC by lowering its electrical conductivity or by making the FCI thicker. Under the DEMO blanket conditions, near-ideal electrical insulation can be achieved with a 5 mm thick FCI if the electrical conductivity σ_{SiC} <1 S/m (Ref. 17).

In contrast to the polodoial flows, in such blanket elements as the inlet or outlet manifold or the coaxial pipe in the fringing magnetic field region, the flow is essentially three-dimensional. In fact, most of the MHD pressure drops in the DCLL module are due to the three-dimensional flows distributions.[13] The MHD drag cannot be reduced significantly using insulating flow inserts or other insulating techniques, and the FCI serves mostly as thermal insulator decoupling hot PbLi from the ferritic structure.

In the ITER TBM and especially under DEMO blanket conditions, the neutron flux is expected to drive buoyant flows whose intensity is comparable or even higher than that of the forced flow. In ITER and DEMO conditions (outboard), the Hartmann number (Hartmann number squared is the ratio of electromagnetic to viscous forces) is of the same order, while the Grashof number (the ratio of buoyancy to viscous forces) in DEMO is higher by about three orders of magnitude due to lager module dimensions and higher thermal loads. This suggests more intensive buoyant flows in a DEMO blanket compared with ITER TBM.[12] The complexity of the poloidal flows can also be affected by the FCI properties. More complex flows are expected to occur if the FCI is not perfectly insulating. In these conditions, the velocity profile is known to be "M-shaped," whose characteristic feature is two high-velocity jets at the sidewalls. In the DEMO blanket conditions, various MHD effects can appear at the same time resulting in very complex flow patterns. For example, the symmetry in the M-shaped velocity profile could be modified by the buoyancy-driven flows. The M-shaped velocity profiles, in turn, can be responsible for quasi-two-dimensional turbulence production. Considering the blanket conditions, such turbulence is of low damping due to its two-dimensionality[14] and is favorable for intensive fluid mixing due to the presence of large coherent structures, thus affecting heat transfer and tritium transport.

As an additional complexity, one should mention significant temperature-dependent variations of the physical properties in both liquid and solid structure, which can affect electric current distribution. Among other MHD effects, which might be important, are the multi-channel effect owing to leaking currents between neighboring poloidal ducts and those due to spatial variations of the magnetic field and due to other two magnetic field components, poloidal and radial.

V.B. SiC Ceramics and Composites

The current DCLL design uses a SiC-based FCI either in a form of continuous fiber-reinforced composite (SiC/SiC composite) or in an alternative form such as porous SiC. The fundamental requirement for the FCI material is the ability to maintain its MHD and thermal insulation functions and its structural integrity during a prolonged period of operation in a harsh environment that combines high temperature and large temperature gradient, flowing liquid metal, and neutron radiation.

SiC/SiC composites consisting of stoichiometric and polycrystalline beta-SiC constituents are in general highly resistant against neutron irradiation and are referred to as the nuclear-grade SiC/SiC composites.[15] The nuclear-grade SiC/SiC composites are anticipated to satisfy the requirements for the TBM FCI at low total neutron fluence of less than 3 dpa. The remaining critical issues that need to be addressed include the mechanical integrity during the evolving complex stress states arising from the anticipated temperature gradients and the transient swelling of SiC, the chemical compatibility within the FCI-steel-PbLi system in the presence of flowing condition and temperature distribution, and development of the joining technology that satisfies the basic requirements of the base FCI material.

Transient swelling in SiC occurs as a result of accumulation of radiation-produced defects on an atomic scale. Recent studies on the transient swelling and irradiation creep for SiC by Katoh, et al.,[16] showed the linear correlation between the swelling and creep strains in addition to the linear stress dependence of the creep strain. The Young's modulus-normalized swelling-creep coupling coefficient of ~40 (at ~390 ºC) to ~120 (at ~790 ºC) derived in this work indicates that only negligible creep-induced relaxation of the differential swelling-induced stress is anticipated. Regardless of the strength and elastic modulus of the FCI material, a through-thickness temperature drop of approximately 200 K will cause the cracking stress in a crystalline SiC-based FCI in case the irradiation creep does not relax the stress development.

The U.S. fusion materials program, in collaboration with the Japanese fusion programs,[17] is undertaking research and development of nuclear-grade SiC/SiC composites and their constituents for the DCLL TBM FCI and as the structural materials for advanced high temperature fusion blankets.

V.C. SiC Flow Channel Insert Fabrication

The development and demonstration of a manufacturable flow channel insert consisting of silicon carbide open-cell structural foam with solid silicon carbide face-sheets has been supported under the DOE Small Business Innovative Research (SBIR) funding, and Ultramet and Digital Materials Solutions (DMS) have made substantial progress. The goal of the project is to further optimize the FCI design under operational conditions in ITER and DEMO, materials, and scale-up, with particular attention to the development of an additional aerogel insulating material to fill the void space in the structural foam insulator to prevent PbLi ingress in the event of protective face-sheet damage. Testing has been performed in static PbLi and in a representative flowing PbLi environment in the MHD PbLi loop at UCLA. A foam segment at different stages of manufacturing is shown in Figure 2. SiC foam FCI specimens containing silica and carbon aerogels have been fabricated and are undergoing testing at UCLA in static PbLi at 700 °C and 1 MPa for at least 100 hours. After the testing of different filler materials, a down select will be made. The optimal FCI material will then be tested in the flowing PbLi experiment at UCLA at 350 °C (maximum possible) in a B-field up to 1.8 T and at velocities up to 10 cm/s. MHD pressure drop reduction and SiC-PbLi compatibility will also be tested.

Fig. 2. 30 cm long SiC foam FCI core at different stages of manufacturing.

V.D. Compatibility

A critical issue for implementing the DCLL concept, is the maximum temperature that can be obtained, especially in the eutectic PbLi. At present, the maximum PbLi and RAF/M interface temperature is limited to <480 °C because of the rapid dissolution of Fe-Cr alloys in flowing PbLi at higher temperatures.[5] Redeposition of dissolved material in the colder sections of the cold loop has the potential to cause flow restrictions if the rate of dissolution is too high. However, it has been demonstrated that Al-rich coatings and alloys can reduce the rate of dissolution in flowing PbLi and in Pb-Bi at temperatures up to 600 °C (Ref. 18). The addition of Al enables the formation of an Al-rich oxide at the alloy surface that inhibits dissolution. In PbLi, this oxide is $LiAlO_2$, as Al_2O_3 and is not stable.[19] To find the maximum temperature possible, capsule testing (i.e. isothermal PbLi) has been conducted at 600, 700 and 800 °C for up to 5,000 h. Figure 3 shows that a thin (~50μm) diffusion aluminide coating on Grade 92 (Fe-9wt.%Cr-2W) steel was effective for 5,000 h at 700 °C (Ref. 20). The next step in the evaluation process is to conduct coating evaluations in flowing PbLi. In preparation for that work, capsule tests have been performed on Kanthal alloy APMT (Fe-21Cr-5Al-3Mo-0.25Y), which is commercially available in tubing to build a thermal convection loop. The APMT mass loss at 800°C was minimal compared to conventional Fe-base alloys,[19] and there was no evidence of Al depletion beneath the surface oxide.[21] The advantage of an Al containing alloy is a larger Al reservoir compared to a coating. Higher operating temperatures could be achieved if oxide dispersion strengthened (ODS) ferritic steels were incorporated into the DCLL design. ODS alloys have significantly higher creep strength than conventional wrought ferritic-martensitic alloys. Some current efforts are investigating ODS FeCrAl alloys (rather than FeCr) for their improved corrosion resistance,[22] which have been available for at least 35 years.[23]

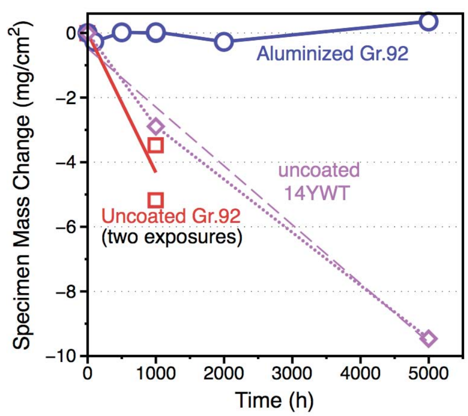

Fig. 3. Specimen mass change as a function of exposure time in PbLi at 700°C. Each data point represents a separate capsule experiment. The higher Cr content in ODS FeCr (14YWT, Fe-14Cr-3W) showed a similar mass loss as conventional wrought Grade 92 (Fe-9Cr-2W).

V.E. RAF/M

The U.S. Fusion Materials Science Program is engaging in a science-based effort to develop basic understanding of the damage mechanisms controlling performance-limiting phenomena of materials for fusion power systems. RAF/M steels are the reference structural material for the TBM and next-step plasma devices in all of the worldwide fusion materials research programs because of their attractive properties and technological maturity relative to low-activation alternatives, such as vanadium alloys and SiC composites.[24] The bulk of the work on these materials is concerned with characterizing the effects of neutron irradiation on their mechanical and physical properties.

Low-temperature neutron irradiation of materials results in considerably increased strength and decreased ductility. It is important to derive true stress-strain constitutive equations and plasticity rules for such materials so that appropriate fusion relevant failure evaluation models can be developed. Recently, post-necking true stress-strain constitutive equations for an RAF/M steel have been determined from engineering tensile stress-strain curves by an iterative fitting procedure that employs a finite element deformation model of the tensile specimen.[25] The most significant result from this work is that the true stress-strain curve of the irradiated steel is consistently higher than the unirradiated steel over a wide range of strain, which is a benefit, provided this increased strength does not lead to plastic flow instability.[25]

A major consequence of fusion neutron irradiation is transmutation production of He. Helium, when coupled with displacement damage, can significantly degrade mechanical properties at all temperature regimes. Measurements and modeling of He effects in RAF/M steels and ODS ferritic alloys have recently received considerable attention.[26] A unique experimental technique known as *in situ* He implantation has been used to concurrently investigate He and displacement damage at fusion relevant conditions. Recent experiments have examined the effects of about 1500 appm He on microstructural evolution in RAF/M steels and ODS ferritic alloys. 1500 appm He is about ¾ of the anticipated end-of-life concentration. The results show that high He promotes the formation of a large population of voids in RAF/M steels that would be largely absent at lower He levels.[26] Rate theory extrapolations of these data suggest that the observed voids are precursors to significant void swelling at higher displacement damage, indicating that RAF/M steels may not be suitable for high dpa service.[26] In contrast, ODS ferritic alloys appear to be much more tolerant to high He levels. In these materials He is benignly trapped in a very high density of nanometer size bubbles that principally form at the oxide particle/matrix interface, with no voids observed.[26] These results have stimulated work to elucidate the structure and composition of the oxide particles that give rise to improved He resistance along with increased high-temperature strength and radiation damage tolerance.[27] Critical research needed to develop RAF/M steels for FNSF and DEMO applications was described in a recent FESAC report.[28] Five Grand Challenges were identified that must be overcome to resolve nuclear degradation of materials and structures. In summary these challenges are: 1) understand and devise mitigation strategies for property changes occurring to materials exposed to high fluence, and high concentrations of transmutation produced gases from fusion neutrons, 2) develop science-based design criteria, 3) comprehend and control the processes that drive tritium permeation, trapping, and retention in neutron damaged materials, 4) understand the mechanisms controlling chemical compatibility of materials exposed to coolants and breeders, and 5) develop advanced fabrication and joining technologies.[28]

V.F. Tritium Extraction

Once bred in PbLi, tritium must be extracted. This will take place some distance from the fusion plasma where the overall environment, in terms of neutrons and magnetic field, is less harsh. But, still above molten PbLi temperatures, tritium will have to be extracted from PbLi. At these temperatures, issues such as material stability and tritium permeation will have to be addressed. Tritium's low solubility in PbLi will aid with extraction but may be problematic for containment.

The leading technologies for extracting tritium are gas stripping[29] and vacuum-driven permeation.[30,31] A gas stripper sparges a gas such as helium into the PbLi. Tritium moves into the He bubbles, and the mixture is collected. Subsequently, tritium is extracted from the He using established techniques. The vacuum permeator maintains vacuum on one side of thin tubes while PbLi flows on the other side. Tritium leaves the PbLi, permeates through the tube and is collected on the vacuum side. Experiments with the gas stripper indicate that this technique will need considerable development to meet requirements. No experiments have been performed on the hydrogen isotope-PbLi system.

The other key issue is tritium containment. One concern is the migration of tritium from the PbLi loop to the facility's room atmosphere. Possibilities for addressing this include the use of low permeation materials, permeation barriers, and secondary/tertiary confinement systems. Another concern is migration of tritium across heat transfer surfaces into the secondary cooling loop. Should this occur, tritium containment challenges would be extended to additional systems.

Since only limited work has been performed to address these issues, effort should begin with computational assessment of the system to identify potentially solutions. Then experimental measurement of fundamental parameters such as mass transfer coefficients will be needed and technologies based on the phenomena will be developed and demonstrated. Successful approaches should be combined into an overall tritium and heat extraction system, which must be tested. And initially these tests should include artificially introduced tritium (i.e. not bred).

The next stage of development will be more challenging and is under discussion. One approach is to perform experiments, which integrate tritium breeding and extraction in an integrated loop. The full experiment would include a neutron source to breed tritium and elucidate other phenomena. This tritium breeding and Extraction Facility would be constructed as small as possible so multiple configurations could be tested without excessive cost, but it would be large enough to show key system characteristics. If possible, such tests would use breeding areas in the 0.1 to 1.0 m^2 range. Tests of this type would collect the data necessary to ultimately design and deploy large blanket systems on full fusion machines such as those under consideration as stepping stones from FNSF to DEMO.

V.G. Neutronics Diagnostics

During our involvement with DCLL-TBM we have looked into the need for neutronics diagnostics, which will also be necessary for any DT device. The neutronics diagnostics can be classified into two major categories, namely, (a) dedicated diagnostics, and (b) supplementary diagnostics. The former is aimed at the verification of the adequacy of current neutron transport codes and nuclear data in predicting key parameters such as local tritium production rate (TPR), heating rate, gas production, and activation. The latter is aimed at providing supporting information to the dedicated diagnostics in examining the predictive capabilities of various computational methods, in addition to providing the source terms and their associated uncertainties (e.g. heat generation and tritium production rate) for other non-neutronics tests devoted to predictive behavior and engineering performance verification (e.g. tritium recovery tests, thermo-mechanics tests, afterheat removal tests, etc.). The measurements that will be performed in the former category include neutron yield and external DD (and DT) source characterization, local TPR, nuclear heating rates, neutron spectrum measurements, and several reaction rate measurements using various foils of specific materials (e.g. $^{27}Al(n,2n)$, $^{27}Al(n,\alpha)$, $^{197}Au(n,\gamma)$, $^{58}Ni(n,p)$, etc.). These multi-foil activation (MFA) measurements are performed to extract information on neutron spectrum.

Lithium-6 and Lithium-7 glass scintillation type detectors of 1-1.5 cm diameter, ~2 mm thickness (10-20 of such pellets) can be used to measure TPR at selected locations in the DCLL module (front, middle, back of TBM). This is an active on-line method. A mechanical or rabbit system to insert and retrieve the detectors, to and from selected locations in LiPb, could be envisioned utilizing the tubes that are inserted from the back of the TBM (2-3 tubes, ~2.5-in diameter each). Passive and in-situ Li_2CO_3 pellet detectors can also be used to measure TPR. They are placed at selected locations in the LiPb through utilizing one of the 2-3 tubes and they are kept in place for a period of time and then are to be retrieved after irradiation. In this case they will not be touching the LiPb. The other approach is to imbed the pellets inside the TBM and only retrieved them when the TBM is withdrawn or replaced. The amount of tritium generated can then be determined with the use of liquid scintillation counters.

Thermoluminescent dosimeters (TLDs), (~2 cm diameter x 3 cm length, 2-3 detectors) and ionization chambers/calorimeters (2-5 cm diameter x 2-5 cm length, ~10 detectors) can be used for heat generation rate measurement, but specially-designed, small-size detectors are necessary. Neutron spectrum measurements at selected locations can best be performed using the multi-foil activation method (MFA) that utilizes ^{197}Au, ^{58}Ni, ^{27}Al, and ^{93}Nb foils (~1 cm diameter, mcm thick in a string of about, 50-70 foils). The MFA method is an indirect technique and should first be confirmed in various available neutron fields.

ITER could become a test bed for the above diagnostics, but for FNSF and DEMO the environment of high heat flux, temperature and magnetic field coupled with high neutron fluence will be much more severe; therefore, significant R&D will be needed before deployment of these diagnostics to FNSF and DEMO.

VI. CONCLUSIONS

We worked on the design and development of the DCLL blanket concept for the ITER-TBM program, with support from necessary R&D activities. DCLL TBM mechanical design and analysis and 3-D neutronics analysis, safety impacts and PrSR were presented. Detailed MHD analysis included poloidal flow and 3-D effects at the manifolds and multi-channel effects. With fission neutron irradiation results, SiC/SiC is projected to be adequate as the FCI material for the DCLL TBM. Open-cell structural foam SiC FCI core has been fabricated and is being tested at UCLA. Static compatibility tests between PbLi and RAF/M have been conducted at different temperatures and it was found that Al containing Kanthal alloy APMT could be more compatible with PbLi. We continued to develop structural materials for fusion and found that RAF/M may not be suitable for high dpa service and the alternative ODS ferritic alloys appear to be more tolerant to high He levels. For the extraction of tritium from PbLi the options of gas stripping and vacuum-driven permeation are being considered. For neutronics diagnostics, we need to monitor heat generation and tritium production and provide quantification of neutron spectrum. Different passive and active diagnostics options can be considered. The general observation from this paper is that for the development of the DCLL blanket, from the focus of DCLL TBM, we have made progress in all areas of the necessary development and we have identified R&D

needs in all areas for a systematic development of the promising high performance DCLL blanket concept.

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REFERENCES

1. M.S. TILLACK, S. MALANG, High performance PbLi blanket, Proc. 17th IEEE/NPSS Symp. on Fusion Energy, San Diego, California, 1997 (Institute of Electrical and Electronics Engineers, Inc., Piscataway, New Jersey, 1998) pp. 1000-1004.
2. D.K. SZE, et al., Blanket system selection for ARIES ST, *Fusion Eng. Des*., **48**, 371-378 (2000).
3. P. NORAJITRA, et al., "The EU advanced dual coolant blanket concept," *Fusion Eng. Des*., **61-62**, 449-453 (2002).
4. C.P.C. WONG, et al., "An overview of dual coolant Pb–17Li breeder first wall and blanket concept development for the US ITER-TBM design," *Fusion Eng. Des*., **81**, (1-7) 461-467 (2006).
5. J. KONYS, et al., "Compatibility behavior of EUROFER steel in flowing Pb-17Li," *J. Nucl. Mater.,* 335(2) 249-253 (2009).
6. M.E. SAWAN, et al., "Neutronics Analysis in Support of the Fusion Development Facility Design Evolution," *Fusion Sci. Techol.*, **60,** pp. 671-675 (August 2011).
7. M.E. SAWAN, et al., "Three-Dimensional Nuclear Analysis for the US Dual Coolant Lead Lithium ITER Test Blanket Module," *Fusion Eng. Des*., **85**, pp. 1027-1032 (2010).
8. M.E. SAWAN, et al., "Transmutation of Silicon Carbide in Fusion Nuclear Environment," to appear in *J. Nucl. Mater*., (2012).
9. D. PETTI, et al., "ARIES-AT Safety Design and Analysis," *Fusion Eng. Des*., **80**, 111 (Jan. 2006).
10. B. J. MERRILL, et al., "Safety Assessment of the ARIES Compact Stellarator Design," *Fusion Sci. Techol.*, **54**, (Oct. 2008).
11. B. J. MERRILL et al., "US Dual Coolant Lead Lithium ITER Test Blanket Module Preliminary Safety Report," Idaho National Laboratory Report, INL/EXT-10-18169, Rev1 (June 2010).
12. S. SMOLENTSEV, et al., "Characterization of key magnetohydrodynamic phenomena in PbLi flows for the US DCLL blanket," *Fusion Eng. Des*., **83** 771-783 (2008).
13. S. SMOLENTSEV, et al., "Numerical analysis of MHD flow and heat transfer in a poloidal channel of the DCLL blanket with a SiC_f/SiC flow channel insert," *Fusion Eng. Des*., **81** 549-553 (2006).
14. S. SMOLENTSEV, R. MOREAU, "One-equation model for quasi-two-dimensional turbulent magnetohydrodynamic flows," *Phys. Fluids*, **19**, 078101 (2007).
15. Y. KATOH, et al, "Current Status and Critical Issues for Development of Sic Composites for Fusion Applications," *J. Nucl. Mater.*, **367,** 659-671 (2007).
16. Y. KATOH, et al, "Observation and Possible Mechanism of Irradiation Induced Creep in Cerami," *Acta Materialia*, (submitted).
17. T. MUROGA, et al, "Midterm Summary of Japan-Us Fusion Cooperation Program Titan," *Fusion Sci. Techol.*, **60,** 321-328 (2011).
18. J. KONYS, et al., *J. Nucl. Mater.*, **367-370**, 1144 (2007).
19. B. A. PINT, *J. Nucl. Mater.*, **417**, 1195 (2011).
20. B. A. PINT, K. A. UNOCIC, submitted to *J. Nucl. Mater*., (2011).
21. B. A. PINT, et al., *Mater. High Temp.*, **29**, 129 (2012).
22. J. H. LEE, et al., J. Nucl. Mater. **417**, 1225 (2011).
23. J. D. WHITTENBERGER, *Met. Trans*., **9A**, 101 (1978).
24. R.J. KURTZ, et al., "Recent Progress Toward Development of Reduced Activation Ferritic/Martensitic Steels for Fusion Structural Applications," *J. Nucl. Mater*., **386–388**, 411 (2009).
25. T. YAMAMOTO, et al., "Constitutive, Damage and Plasticity Laws for Candidate Fusion Alloys," in Fusion Materials Semiannual Progress Report for the Period Ending December 31, 2011; DOE/ER-313/51 78 (2012).
26. G.R. ODETTE, et al., "Helium Effects in Advanced Structural Alloys: Progress on Solving a Grand Challenge to Fusion Energy," in Fusion Materials Semiannual Progress Report for the Period Ending December 31, 2011; DOE/ER-313/51 90 (2012).
27. Y. WU, et al., "Transmission Electron Microscopy Characterization of the Nanofeatures in Nanostructured Ferritic Alloy MA957," *Acta Mater*., **60**, 3456 (2012).
28. S. ZINKLE et al., "Materials Science and Technology Research Opportunities Now and in the ITER Era: A Focused Vision on Compelling Fusion Nuclear Science Challenges," FESAC Report DOE/SC-0149 (2012).
29. I. RICAPITO, et al., (2011). "Tritium Extraction From Liquid Pb-16li: A Critical Review Of Candidate Technologies for ITER and Demo Applications." *Fusion Sci. Technol.*, **60**, 1159 (2011).
30. R. WILLMS, et.al., "Initial Testing Of A Low Pressure Permeator For Tritium Processing," *Fusion Eng. Des*., **49-50**, 963-970 (2000).
31. N. ALPY, et al, "Hydrogen extraction from Pb-17Li: results with 800 mm high packed column," *Fusion Eng. Des*., **49-50**, 775-780 (2000).