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US PLANS AND STRATEGY FOR ITER BLANKET TESTING

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Testing blanket concepts in the integrated fusion environment is one of the principal objectives of ITER. Blanket test modules will be inserted in ITER from Day 1 of its operation and will provide the first experimental data on the feasibility of the D-T cycle for fusion. With the US rejoining ITER, the US community has decided to have strong participation in the ITER Test Blanket Module (TBM) Program. A US strategy for ITER-TBM has evolved that emphasizes international collaboration. A study was initiated to select the two blanket options for the US ITER-TBM in light of new R&D results from the US and world programs over the past decade. The study is led by the Plasma Chamber community in partnership with the Materials, PFC, Safety, and physics communities. The study focuses on assessment of the critical feasibility issues for candidate blanket concepts and it is strongly coupled to R&D of modeling and experiments. Examples of issues are MHD insulators, SiC insert viability and compatibility with PbLi, tritium permeation, MHD effects on heat transfer, solid breeder "temperature window" and thermomechanics, and chemistry control of molten salts. A dual coolant liquid breeder and a helium-cooled solid breeder blanket concept have been selected for the US ITER-TBM.

I. INTRODUCTION

A critical element in the ITER mission since its inception has been testing integrated blanket modules in special ports[1,2]. Among the principal objectives of the ITER Test Blanket Module (ITER-TBM) Program are the development of the technology necessary to install breeding capabilities to supply ITER with the tritium necessary for operation in its extended phase, and the acquisition of experimental data vital to evaluating the feasibility, constraints, and potential of the D-T cycle for fusion systems (including limitations on options for improving plasma physics performance, e.g., conducting shells, passive coils, thick armors/first wall). Adequate tritium supply is a central issue for the extended operation of ITER and the development of fusion energy in general.

Among the specific objectives of ITER testing are:

- validation of structural integrity theoretical predictions under combined and relevant thermal, mechanical and electromagnetic loads;
- evaluation of uncertainties of tritium breeding predictions;
- validation of tritium recovery process efficiency and T-inventories in blanket materials;
- validation of thermal and fluid flow predictions for strongly heterogeneous breeding blanket concepts with surface and volumetric heat sources;
- information on first effects of radiation damage on component operation and first information on integrated component reliability;
- information leading to the demonstration of the integral performance of the blankets systems.

TBMs will be inserted in ITER from Day 1 of its operation. Very important information will be obtained from TBMs during the H-H phase; for example:

- demonstration of the structural integrity of the TBM structures and attachments during disruption and Vertical Displacement Events (VDE);
- assessment of the impact of Ferritic/Martensitic steel, used as a structure for most TBMs, on magnetic fields deformation in static conditions;
- testing of fluid flow, particularly MHD, in the complex 3-component magnetic field environment;
- establishment of TBM first wall requirements, e.g. need for beryllium coating;
- screening of flaws in TBMs so that they can be fixed prior to D-T operation.

It should be noted that ITER operating parameters such as wall load, neutron wall load, plasma burn and dwell times, and fluence have limitations in terms of providing all the required conditions to develop blankets for DEMO. However, ITER is currently the only planned fusion facility in the world fusion program over the next two decades that can provide integrated fusion environment for blanket and material testing. Therefore, maximum utilization of ITER for blanket and material testing is essential for the advancement of energy related fusion technology that in turn provides essential feedback to possible plasma physics operating regimes and boundary conditions.

With the US rejoining ITER, the US community has decided to have strong participation in ITER-TBM. The US has been a leader in the science and engineering of technology testing on ITER and other fusion devices and has many unique capabilities to contribute to the ITER-TBM program. A US strategy for ITER-TBM has evolved that emphasizes international collaboration. A study was initiated to select the two blanket options for the US ITER-TBM in light of new R&D results from the US and world programs over the past decade. The study is led by the Plasma Chamber community in partnership with the Materials, PFC, Safety, and physics communities. The study has focused on assessment of the critical feasibility issues for candidate blanket concepts and it is strongly coupled to R&D of modeling and experiments. The highlights of the study are presented in this paper. More details are reported in Refs. 3-8.

II. LIQUID BREEDER BLANKET OPTIONS

Liquid breeder blanket options were evaluated taking into account R&D results from the world program over the past decade. These options include:

- self-cooled lithium with vanadium structure (Li/V);
- helium-cooled lead-lithium (HCLL) with ferritic steel structure;
- dual coolant lead-lithium (DCLL) with He-cooled ferritic steel structure;
- self-cooled and dual coolant molten salts with ferritic steel structure.

Although the Li/V concept was the US favored concept for two decades, it was not selected for the US TBM because of seriously negative R&D results. In particular, the lack of progress on developing practical MHD insulator coatings with acceptable crack tolerance makes the Li/V not a suitable candidate for ITER TBM testing. There are also concerns about V alloy development cost and time schedule.

The helium-cooled lead-lithium alloy 83Pb-17Li (referred to in this paper as PbLi) option is being developed by the EU. It has some attractive features as a near term option but its thermal efficiency is limited and there are issues of tritium permeation and corrosion.

Many earlier studies concluded that reduced-activation ferritic/martensitic steel like EUROFER97 or F82H (referred to in this paper as FS) is the only attractive and practical structural material that is available in the ITER testing time frame. The US has decided to focus on testing Dual-Coolant liquid breeder blanket concepts (with ultimate potential for self-cooling) in order to obtain relatively high thermal efficiency while using FS as the structural material.

In the dual coolant concept, the helium cooled ferritic structure is always kept below 550°C, which is the FS operating limit based on yield strength considerations. The liquid breeder is “self cooled” and has an exit temperature substantially higher than that of the structural material. Two liquid breeders were evaluated: PbLi, and the low melting point molten salts $LiBeF_3$ and FLiNaBe. In the end, the DCLL concept was selected as the primary candidate for testing in ITER, with dual-coolant molten salt as the backup option.

II.A The Dual Coolant Lead-Lithium (DCLL) Concept

The DCLL concept investigated in this study is similar to that developed earlier by ARIES[9] and EU[10], which proposes the use of a flow channel insert (FCI) made of SiC_f/SiC composite in the breeder region flow channels in order to:

- thermally insulate the self-cooled breeder region from the helium cooled FS walls;
- electrically insulate the PbLi flow from current closure paths in the FS walls;
- provide a nearly stagnant PbLi layer near the FS wall that has increased corrosion temperature limits.

The FCI geometry is shown in Fig. 1, where a typical breeder unit cell is shown from the ARIES-ST design. The FCI has roughly a “C” shape, with a gap provided to allow pressure equalization from the interior region to the exterior of the FCI near the wall. Since electric circuits must be continuous, a single gap is still allowed from the perspective of electrical insulation. The flow is oriented poloidally, generally flowing one direction near the first wall, turning 180°, and returning in the rear of the blanket.

The FCI improves performance of the DCLL blanket over separately cooled PbLi blankets by allowing higher outlet temperature and energy conversion from the self-cooled LM breeder zone. This high temperature is possible because all structures are separately cooled by helium, eliminating any requirement for the PbLi region to be colder than the structure to promote heat transfer from the wall to the LM. With the thermal insulation provided by the FCI, the bulk PbLi temperature (in the interior of the FCI shown in Fig. 1) can in fact be significantly higher than the wall temperature, provided that the FS/PbLi interface temperature is kept below corrosion temperature limits for FS. Current designs have the bulk PbLi exit temperature at ~700°C, with peak PbLi temperatures near 800°C in the front channels near the first wall.

The FCI also serves to significantly reduce the degree of MHD pressure drop when compared to that of a purely self-cooled design. The fact that a high PbLi velocity is not needed for first wall or other structure cooling, combined with the high PbLi temperature change allowed due to the thermal insulation provided by the insert, results in a very low mass flow rate required to remove the nuclear heating in the self-cooled PbLi channels. This low velocity reduces the MHD pressure drop by roughly an order of magnitude when compared to bare-wall self-cooled designs (the degree of reduction depends somewhat on geometry and loading assumptions, but this is a rather conservative estimate). In addition, the electrical insulation provided by the FCI can reduce the pressure drop by another factor of 3 to 100 or more, depending on the electrical conductivity and thickness of the SiC FCI and steel wall. Together, these two effects are sufficient to keep wall stress below material limits of the FS, even in long, high field, inboard blanket channels where very high pressure stresses can be generated for LM blankets. The relative effect of velocity reduction and FCI insulation on maximum inboard cooling channel wall stress is shown in Fig. 2, assuming the dimensions in Fig. 1, a typical inboard neutron wall load of 2.2 MW/m^2, and an inboard flow length of 8 m. It is seen that moderate to low SiC conductivity is desirable to keep primary wall stresses low.

II.A.1 SiC_f/SiC FCI Issues and Feasibility

The feasibility of SiC_f/SiC flow channel inserts hinges on the compatibility of the SiC_f/SiC with the PbLi at high temperatures with characteristic impurities and with thermal stress loads. Issues related to the feasibility and testing of FCIs are presented in detail in Ref. 4. In summary, a SiC_f/SiC FCI must show the following behavior:

- The inserts have to be chemically compatible with PbLi at temperatures up to ~800°C in a flow system with strong temperature gradients and contact with FS at lower temperature.
- Liquid metal must not "soak" into cracks or pores of the composite in order to avoid increased electrical conductivity, high tritium retention, or explosively vaporized LM pockets. In general, nearly 100% dense "sealing layers" are required on all surfaces of the inserts.
- Secondary stresses caused by temperature gradients and primary stresses caused by developing flow conditions, field gradients or any off-normal electromagnetic event must not endanger the integrity of the irradiated SiC_f/SiC.

Static exposure experiments have not shown any dramatic attack of PbLi upon the SiC at these temperature levels, but definitive compatibility experiments, especially in systems with prototypic SiC_f/SiC with sealing layers in contact with PbLi in a flowing system with typical temperature gradients and FS will ultimately be needed.

The thermal stresses in the SiC insert must also be controlled. Preliminary calculations indicate that the stresses for a 5 mm thick insert with typical thermal load are all within the rated limits of the SiC composite.

There is room to optimize the electrical and thermal conductivity of the SiC_f/SiC for the desired application. Most SiC_f/SiC research for fusion and other applications is aimed at improving the thermal conductivity and strength of SiC by using high quality fiber and fiber coating "interphase" in complicated weave geometries. The requirements of low thermal and electrical conductivity, especially along the wall-normal direction, indicate that a simpler 2-D weave should be considered, and that the matrix infiltration technique should produce only closed porosity. Electrical and thermal conductivities near unity (in MKS units) are desired for best functioning of the FCIs in this application.

The FCI shapes must also be fabricable and affordable. Based on an assessment of current capabilities, the generic flow channel insert "C"-shape seems well within the scope of current manufacturing capability. More complex shapes for bends, manifolds, etc. may also be possible. It is anticipated that the various FCI shapes will be designed with overlapping "stovepiped" regions so that they can be slipped over one another in order to provide quasi-continuous insulation without the need for brazing.

II.A.2 Other DCLL Issues

The pressure drop and flow balance in complex geometry flow elements, developing flow regions, flawed FCIs, etc. where current closure along the flow direction (through the LM itself) is present will need to be the subject of future thermofluid MHD research. Some estimates have been done for pressure drop and velocity profiles for the European DCLL DEMO design[11] where it was shown that the pressure drop in the long straight channels was negligible compared to the 1-2 MPa estimated for some of the manifold regions. This is an area that must receive more scrutiny in the near future.

Flows for the DCLL with SiC FCIs can also exhibit side layer jet formation due to the incomplete electrical insulation expected for SiC with electrical conductivity on the order of 200 to 500 $\Omega^{-1}m^{-1}$. Figure 3 shows a typical velocity profile for a PbLi flow with a 5 mm thick SiC_f/SiC FCI with isotropic conductivity 500 $\Omega^{-1}m^{-1}$. Notice the strong velocity peaks near the walls, and even in the corner regions (usually considered to be stagnant) in between the FCI and the FS wall. It has been estimated that a reduced SiC conductivity of about 1 $\Omega^{-1}m^{-1}$ is necessary to avoid these jets. The impact that these velocity profiles have on thermal performance can be strong and has not yet been fully analyzed.

Compatibility between PbLi and FS has been extensively studied in the EC program. The corrosion rate as a function of temperature has been established, and the interface corrosion temperature limit was 475ºC for PbLi velocity ~0.1 m/s. To increase the exit PbLi temperature (for power conversion reasons), methods to increase the allowable interface temperature were evaluated. For instance, at low velocity, the allowable interface temperature can reach ~ 550ºC.[12] Since velocity has such strong effect on corrosion rate, the MHD effects may also be important, due to the change of the velocity profile, particularly in the boundary region. The experimental study of MHD effects to the corrosion rate is being evaluated.[13]

The PbLi has very low tritium solubility. The tritium partial pressure in the PbLi at the exit of the reactor is >0.01 Pa. The exact tritium partial pressure depends on the blanket design, and also the tritium recovery efficiency. At this high tritium partial pressure, it will permeate across the large surface area of the primary heat exchanger. Work has proceeded in Europe on the development of the tritium permeation barrier to reduce the tritium permeation rate. However, the results did not achieve the goal of a factor of 100 reduction factor[14]. One possible solution to control the tritium permeation, and loss to the environment is to select a gas cycle so that tritium can be recovered from the He gas, before it losses to the environment.

II.A.3 Potential for Testing in ITER

While the conditions of testing in ITER[15] will be somewhat limited, still ITER will provide a very challenging test environment for blanket modules that will have had no other significant prior integrated testing. Effective DCLL testing in ITER is possible and desirable, especially for investigating the performance of the SiC FCIs whose integrity and properties will depend on many integrated effects including: helium/hydrogen bubble formation, transmutation, corrosion, tritium partial pressure, SiC thermophysical properties, thermal gradients, and magnetic field. ITER provides an increasingly integrated environment in which failures in the FCIs need not result in catastrophic over-pressurization (see for instance Fig. 2 at 4-5 T ITER TBM fields), flow stoppage or overheating for properly designed tests. Such DCLL operation and FCI failures can be explored in ITER.

Table I gives a summary of a strategy for ITER testing progressing from basic structure and MHD performance to more integrated test modules as a function of the ITER operational phases during the first 10 years. The rather strong steady toroidal field (~5 T at the first wall) with prototypic gradients in a large magnetic volume provides in itself a unique environment for MHD testing and structural integrity in realistic fusion magnetic fields with typical plasma events including disruptions and other transients. Thermofluid tests and various neutronics tests are envisioned for the Low Duty DT phase, while performance tests with the buildup of some degree of radiation damage and property changes in materials will follow in the High Duty DT phase. This strategy is still evolving and the detailed analysis and scaling is currently the subject of research activities within the US Plasma Chamber program.

The unique issue for DCLL relative the European HCLL concept is the compatibility of the SiC insert with the PbLi under high temperature and temperature gradient conditions. Near-term focus in the US will be on improving our conception of DCLL reference design for DEMO application and ITER testing. The DCLL is a reasonable extrapolation to higher performance from the PbLi/FS/Helium blankets developed in the EU Program.

TABLE I. Outline of Testing Strategy in ITER Test Blanket Module Program for DCLL Concept Focusing on FCI Operation and Integrity

H and D	**Specific electromagnetic structure and MHD TBM:** ▪ Structural TBMs reaction to field environment and various transient plasma events –for instance "water hammer" effect during rapid plasma current quench ▪ Scaled pressure drop tests, flow balance test and critical velocity profiles affecting heat transfer in increasing field strength ▪ SiC FCI integrity under MHD loading
Low Duty DT	**Thermofluid TBMs:** ▪ Scaled stress response of thermally loaded SiC inserts and the effects of failures on pressure drop and thermal field ▪ Tritium production and permeation
High Duty DT	**Partially-integrated Thermofluid TBMs:** ▪ Corrosion and compatibility in fusion environment (nuclear/magnetic/etc.) ▪ Radiation damage ~1 dpa in the inserts ▪ Tritium control ▪ Demonstration of integrity and response of the TBM structure ▪ Demonstration of the integral performance of the DCLL concept

II.B Molten Salt Concepts

Since there are no recent conceptual designs for molten salt (MS) concepts, an effort was devoted over the past year to explore possible design options[3]. The objective of the

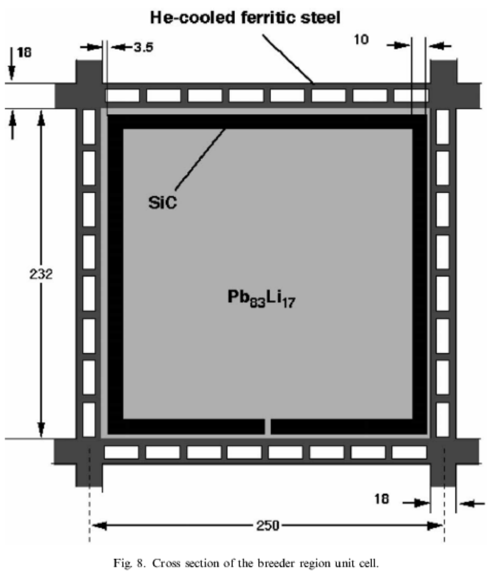

Fig. 1. Cross-section of breeder channel from ARIES-ST Dual-Coolant PbLi Blanket.[16]

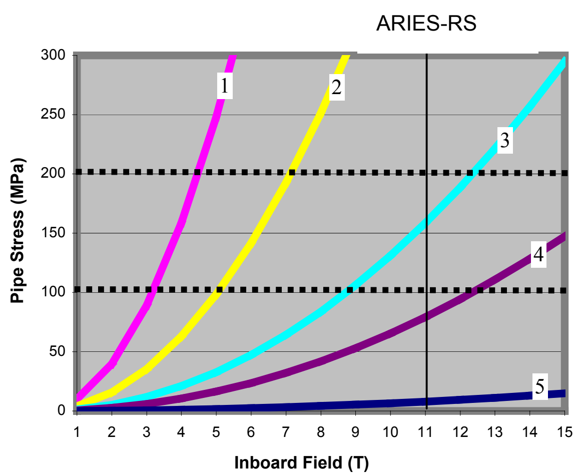

Fig. 2. Pipe stress estimates as a function of magnetic field for an inboard PbLi blanket channel 8 m in length with P_{NWL} 2.2 MW/m^2. (1) bare wall with FW coolant velocity 1 m/s, (2) bare wall breeder channel velocity 0.4 m/s, (3) DCLL breeder channel with high outlet temperature allowing low velocity 0.17 m/s but no electrical insulation, (4) Same as 3 with rather poor electrical insulation corresponding to σ_{SiC}=500 Ω^{-1}m^{-1}, and (5) same as 3 with moderate electrical insulation corresponding to σ_{SiC}=50 Ω^{-1}m^{-1}.

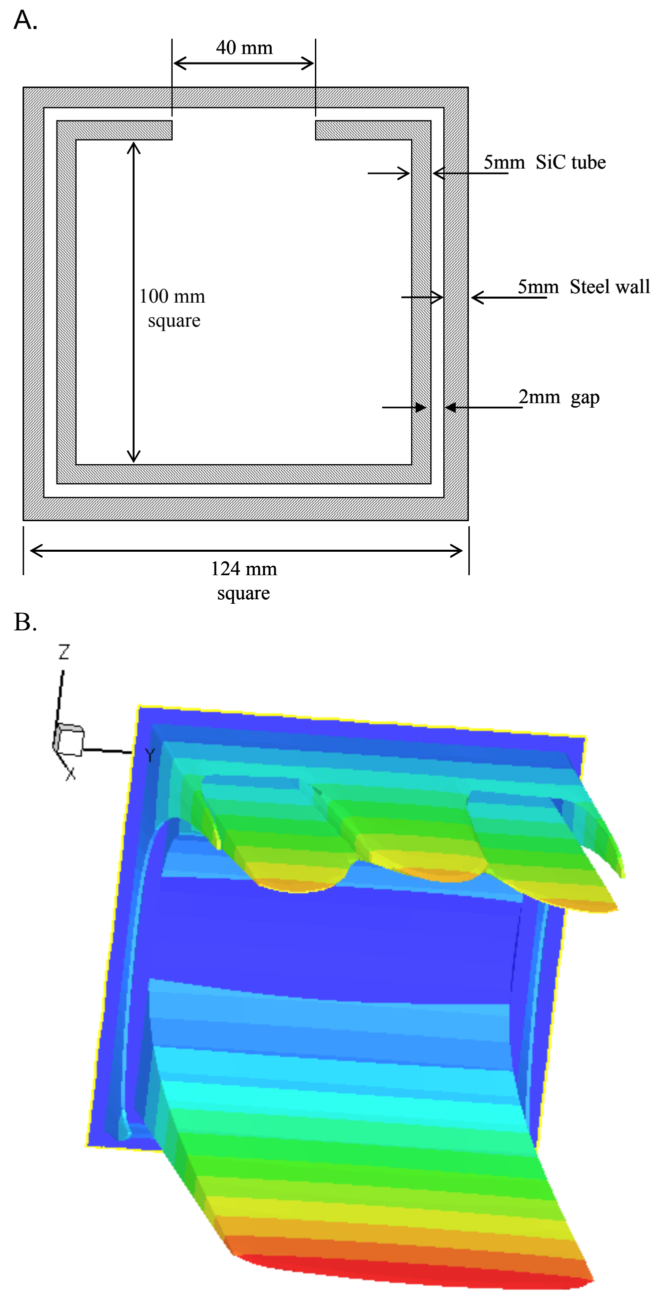

Fig. 3. Slice of breeder unit from 3D MHD calculation using HIMAG code. (A) Cross-section showing geometry, and (B) velocity profile in fully developed region showing side layer jets and near-stagnant flow in center of channel.[4]

conceptual design work was to enable better evaluation of the key technical issues and the suitability of molten salt concepts as candidates for ITER TBM. The molten salts evaluated included Li_2BeF_4 and the low melting point molten salts such as $LiBeF_3$ and FLiNaBe. Both self-cooled and dual coolant molten salt options were evaluated assuming FS structural material – typical geometries are shown in Figs. 4 and 5.

The first wall / blanket assessment was based on typical power reactor conditions with a maximum neutron wall loading of 5.4 MW/m^2 and a maximum surface heat flux of 1 MW/m^2 at the outboard mid-plane of the tokamak reactor. Molten salt blankets require an additional neutron multiplier like Be or Pb to provide adequate tritium breeding. In orderto meet the temperature limitation of FS, while obtaining high molten salt outlet temperature for high thermal efficiency, the dual coolant (DCMS) configuration was adopted. Helium is used to remove the first wall surface heat flux and to cool the entire steel structure. The self-cooled liquid MS breeder is circulated to external heat exchangers to extract the heat from the breeding zone, with an exit temperature substantially higher than that of the ferritic structure. We take advantage of the molten salt low electrical and thermal conductivity to minimize impacts from the MHD effects and the heat losses from the breeder to the actively cooled steel structure. The molten salt essentially acts as its own thermal insulator, supporting large temperature gradients near the helium cooled walls.

Some critical feasibility and design issues emerged during the study. It is difficult to avoid the formation of a thin solid Flibe layer on parts of the Helium cooled structure when the high melting point Flibe is utilized. It is likely that the lower melting point salt is required to open the temperature window and eliminate the possibility of a frozen layer. For the lower melting point MS, FLiNaBe in particular, thermophysical properties like will need to be further quantified. The MHD effect on turbulence and natural convection will play a critical role in determining the self-insulation properties of the Flibe, and temperature windows for DCMS concepts. Research related to this issue is being carried out at UCLA as part of the JUPITER-II collaborative research program with Japanese Universities.

Another key feasibility issue for all MS designs is the control of F and TF generated by nuclear reactions with the Be and Li in the MS. Some sort of REDOX chemistry control will needed to mitigate the compatibility issue between the generated F and TF and structural material. LiF and BeF_2 are very stable fluorides (see Table II) and by themselves do not lead to excessive dissolution of the constituents of FS. If an excess of Be is available in the coolant, then the reduction process can occur:

$$Be + 2TF \rightarrow BeF_2 + T_2 \qquad (1)$$

That will control the TF by reacting it into T_2 form. Based on the free energy of formations in Table 2, this reaction is very favorable from thermodynamics considerations. However, it is not certain if kinetically this reaction is fast enough. The Flibe REDOX experiment at INEEL is meant to confirm the kinetics of this reaction,[16] and initial data suggests that dissolved Be will be effective in controlling TF levels. This research is also a part of JUPITER-II.

TABLE II. Free Energies of Fluoride Formation[19]

Material	DG^f_{1000K} (kcal/g-atom of fluorides)
MoF_6	-50.2
WF_6	-56.8
NiF_2	-55.3
VF_5	-58.0
VF_4	-66.0
HF	-66.2
FeF_2	-66.5
NbF_5	-72.5
CrF_2	-75.2
TaF_5	-82.2
TiF_4	-85.4
LiF	-125.2
BeF_2	-106.9

The tritium produced will then exist in T_2 form. T_2 chemically acts like a metal, and metals do not readily dissolve in salt. The tritium solubility in the fluoride salt is thus very low and rather high tritium partial pressure will result, requiring a system to control the tritium permeation processes. Work on this issue for MS fusion systems has not yet begun in earnest.

An alternate REDOX control concept can be envisioned where MoF_6 is added to the MS so that the reactions:

$$MoF_6 + 3T_2 \rightarrow Mo + 6TF \qquad (2)$$

and:

$$MoF_6 + 3Fe \rightarrow Mo + 3FeF_2 \qquad (3)$$

take place. Here all tritium is oxidized into TF form and a stable corrosion barrier of Mo is formed on all FS walls. The tritium permeation is much smaller since TF has a much higher solubility than T_2 in Flibe, and also the chemical compound TF has lower permeability. Therefore, tritium control is less an issue. This process was demonstrated on both steel[17] and vanadium[18] containers. If the reaction 2 and 3 will complete to the right direction, Mo will precipitate out. Reaction 3 will stop after a Mo layer is established on the surface of the steel wall. An electro-chemical process:

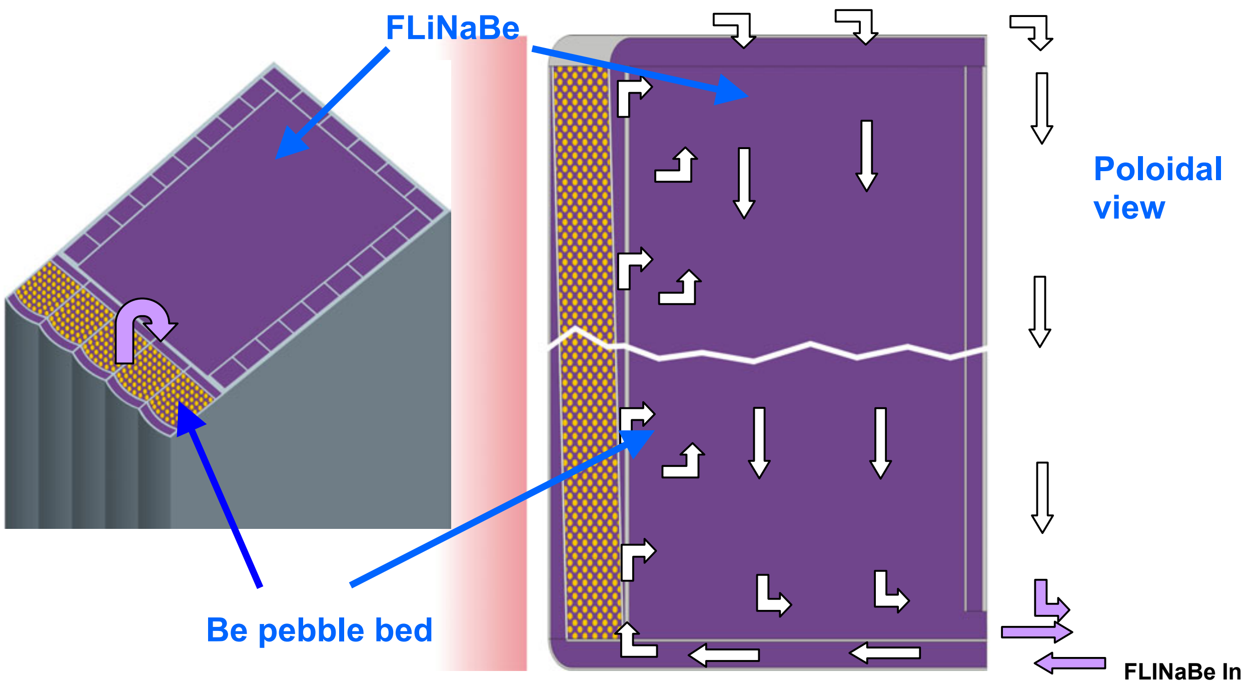

Fig. 4. Schematics of the self-cooled FLiNaBe design.

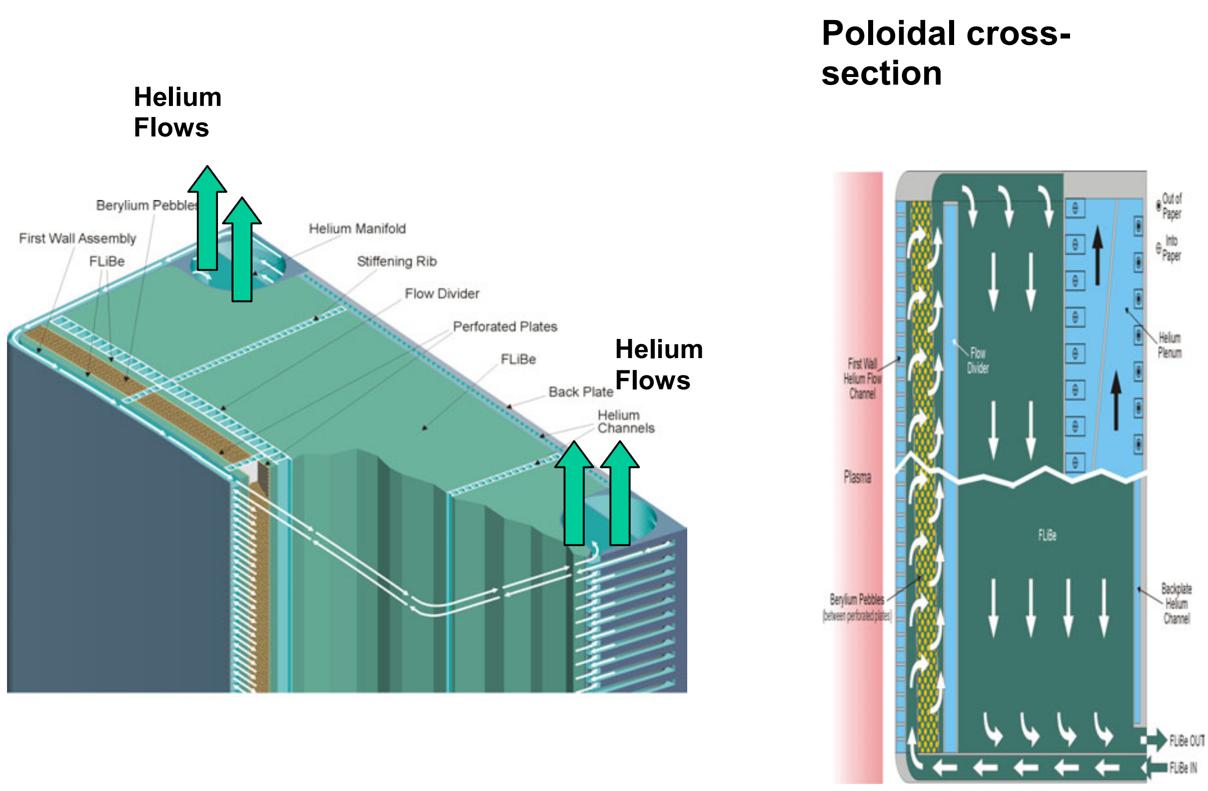

Fig. 5. Design configuration of the DC/He and FLiBe/Be neutron multiplier design.

$$2MoF_6 + 3T_2 \rightarrow 2MoF_3 + 6TF \qquad (4)$$

to control the reaction 2 is proposed. This process is similar to the REDOX process for the molten salt fission reactor, but has to be demonstrated for fusion blanket.

Equipment size is designed to the maximum power handling of surface heat flux of 0.5 MW/m^2 and a neutron wall loading of 0.78 MW/m^2 for a vertical half of an ITER test port. Based on the requirements of handling the first wall heat flux and maximizing the outlet temperature of the liquid breeder coolant up to 650°C, which is thermally insulated from the FS structure, we have to evaluate the ancillary equipment required for two coolant loops. The first one is the first wall and structure helium coolant loop, which are designed to carry 40% of the total FW/blanket energy at the maximum ITER operation parameters. Dedicated helium piping is designed between the TBM and the helium/water heat exchanger at the Torus Coolant Water System (TCWS) vault. The second one is the liquid breeder loop, which we designed to carry 100% of the blanket energy at the maximum ITER operating parameters. This second loop would also allow the testing of a single breeder/coolant self-cooled breeder concept if our blanket development evolves to such a direction in the future. For this breeder loop we designed a helium intermediate heat removal loop between the breeder and the TCWS water cooling system. Corresponding equipments for the intermediate helium-loop system were also designed.

A preliminary safety assessment of the system was also performed, which provided guidance to our ancillary equipment design in areas of minimizing the vulnerable breeder volume and the potential loss of tritium through permeation. For the liquid breeder loop design, the requirement of minimizing potential tritium loss from the breeder to the vicinity led us to the use of the helium intermediate heat transport loop as mentioned above. This intermediate loop also helps to minimize the required pressure drop when the high viscosity fluid $LiBeF_3$ is utilized by keeping the distance between the TBM and the liquid breeder/helium heat exchanger to a minimum. Concentric pipes are proposed to connect the liquid breeder between the TBM and the breeder/helium heat exchanger. For the FW-coolant loop, to minimize tritium permeation aluminum tubes are recommended for the He/water heat exchanger and permeation barrier like alumina coating or Al outside sleeve are recommended to be applied to the helium-coolant inlet and outlet piping. Results from the preliminary safety assessment for the ancillary equipment for the two FW/blanket concepts show that ITER safety criteria can be met, provided that we take care to control the amount of breeder used in the system and to reduce tritium permeation loss from the FW coolant loop and from the liquid breeder loop.

Details of key ancillary equipment, including heat exchangers, circulators, electrical heater, and helium storage and dump tanks for the two loops have been estimated, but must still be defined. All liquid breeder test equipment can be located within half of the test module transporter area as shown in Fig. 6. For the heat transport equipment in the TCWS vault, the required footprint is estimated to be a total of 20 m^2 and 5 m high for both helium to water loops, and a schematic of the necessary equipment in the TCWS is shown in Fig. 7. Piping size for the two loops connecting the TBM to the TCWS vault, which are separated by ~70 m in distance, has also been estimated, including the necessary inclusion of 10 to 15 cm of thermal insulation. Detailed designs of the PbLi/He and $LiBeF_3$/He heat exchangers are different because of the difference in thermophysical properties of the two breeders. Both heat exchanges will be able to handle the low coolant outlet conditions of 520°C and 440°C, respectively, and the high performance condition of 650°C. For the high performance case, the heat exchangers will have to be designed with the flexibility of reducing the heat removal surface by about 40%, and tube plugging is an example of achieving such an effect.

For this phase of the ancillary equipment assessment, the main focus was on providing input to ITER regarding the need for space and power requirements of key ancillary equipment. More detailed component design, fluid, tritium and safety handling design and analysis will still be needed. The scenario of testing FW/blanket concepts in different phases of ITER operation is currently being developed for the DCLL concept, and the corresponding sequence of installing necessary test components will have to be coordinated. But the requested envelope for the two testing loops proposed in this report[20] will be able to provide the flexibility of testing the selected DC liquid breeder FW/blanket concepts.

III. TESTING STRATEGY FOR SOLID BREEDER BLANKET OPTIONS

The US strategy for ITER TBM also includes participation in testing a helium-cooled solid breeder concept with FS structure and Be neutron multiplier. All ITER Parties have such a solid breeder concept as one of their options. In this case, the US will not provide an independent TBM, but rather will collaborate with the EU and Japan using their ancillary equipment. The US will contribute unit cells and sub-module test articles that focus on particular technical issues of unique US expertise and of interest to all parties. A unit cell will occupy a port area of about 19.5 x 21 cm and be housed behind another party's

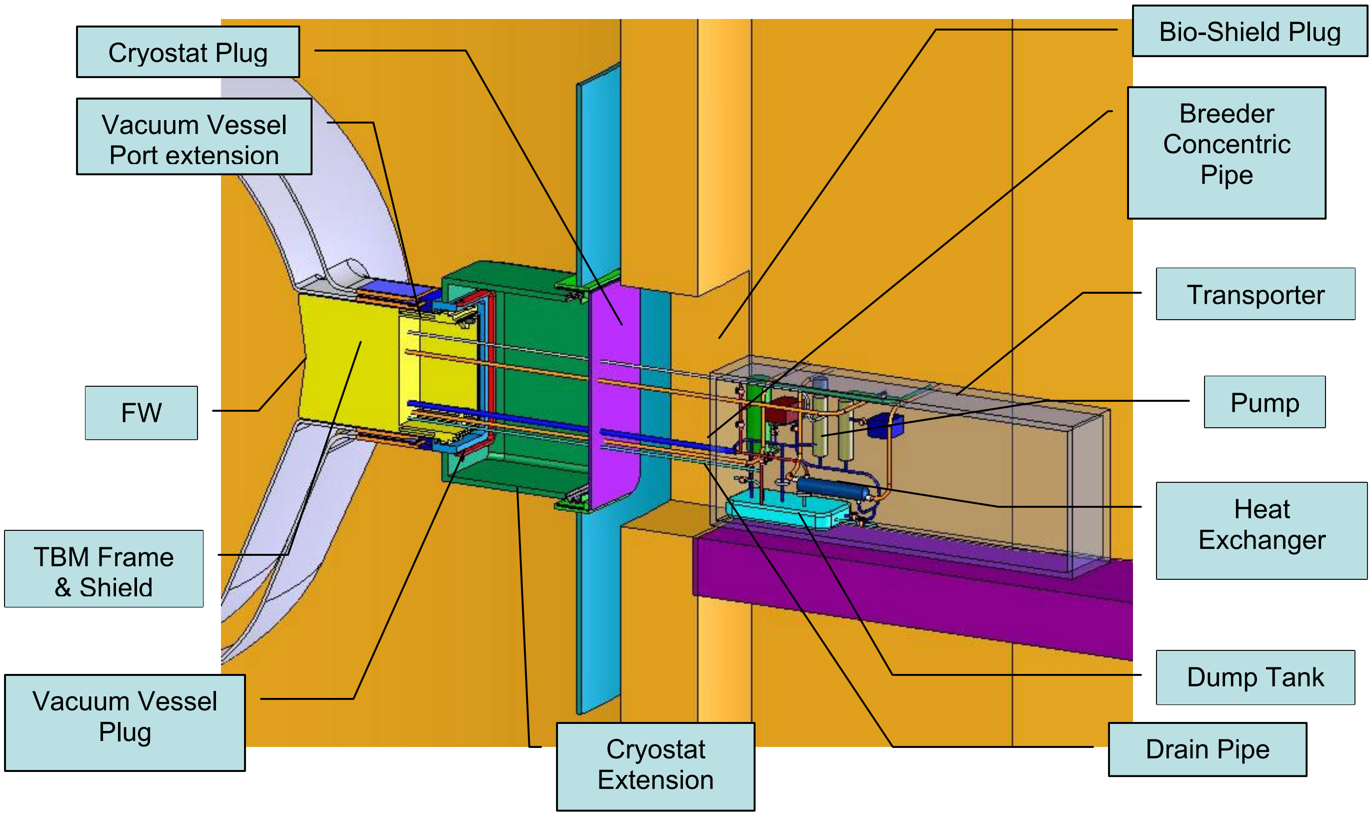

Fig. 6. ITER Test Port General Arrangement with the US DC Liquid Breeder Blanket Option and the Corresponding Equipment and Pipings Shown in the Transporter.

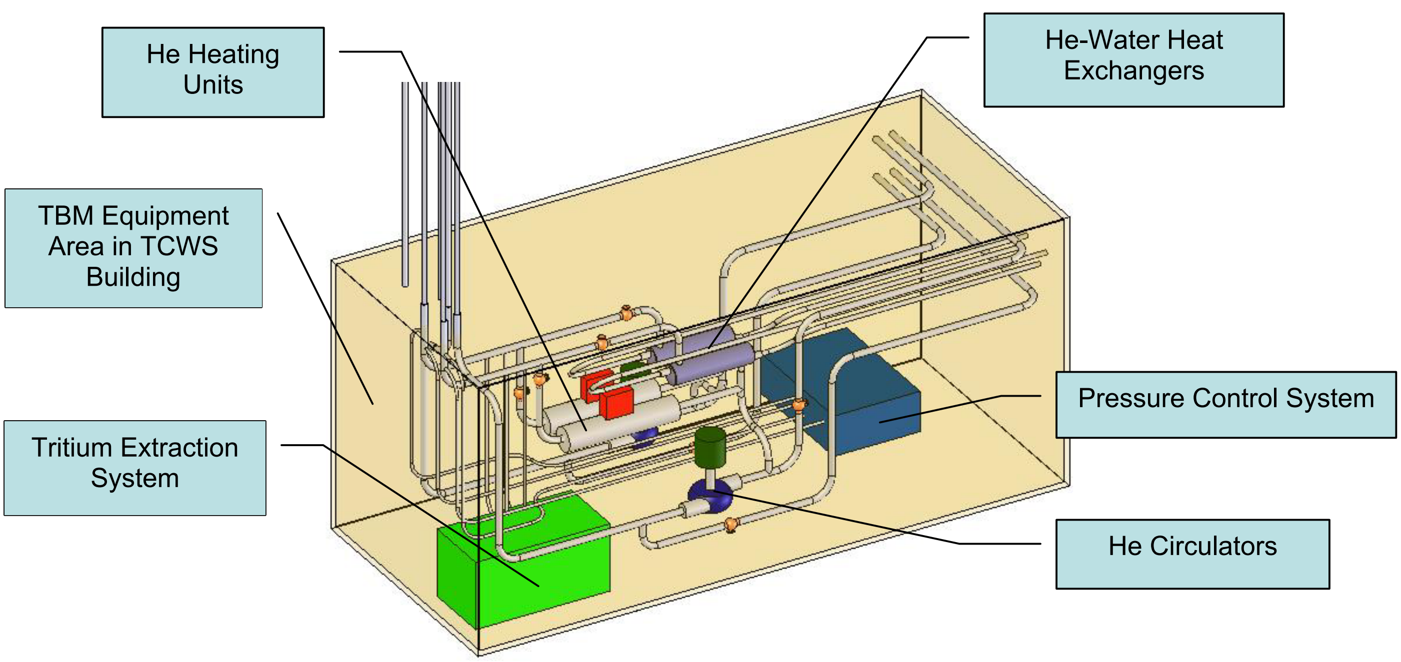

Fig. 7. US DC Liquid Breeder Blanket Primary and Secondary Coolant Loops in the TCWS.

structural box (Fig. 8), while a submodule will take up a testing space of a quarter port 73 x 91 cm and have its own structural box (Fig. 9).

III.A Test Sequence and Description

These test blanket units will be designed and inserted into the helium-cooled ceramic breeder test port according to the testing strategy synchronized with ITER operational characteristics with the following experimental objectives:

- *First wall performance and transient electro-magnetic tests during the H-phase. Testing objectives focus on the evaluation of predictive capability on the test blanket module's structural thermomechanical performance in response to an integrated fusion load of electromagnetic, thermal, and mechanical forces.*
- *Neutronics and tritium production rate prediction tests will be performed during the D-Phase and the early DT-phase***.** Testing objectives focus on the evaluation of tritium breeding performance and the validation of neutronic code prediction and nuclear data.
- *Tritium breeding, release and thermomechanics explorations tests during the DT-phase*. The objectives are to study configuration effects on tritium release and pebble bed thermomechanical performance. The data can be used to optimize configuration aspects of solid breeder blanket designs.
- *Initial study of irradiation effects on performance during the DT-phase*. Since several thermo-physical properties of breeding materials show the largest changes after initial exposure to irradiation, understanding their impacts on blanket performance will guide the design.

This testing strategy calls for three to four submodules/unit cells to be sequentially inserted into the designated test port from the commencement of ITER operation. While it is planned to share the same helium loop with the neighboring party or parties, any special requests to the coolant operating conditions (such as temperature) can be handled through a helium coolant conditioner located near the port area. This leads to only one coolant supply and one coolant return line running between the port area and the TCWS building per half-port. To maximize the use of ITER testing, the tritium concentration and gas composition from each breeder purge gas line will be analyzed at the test port area before merging it with other purge gas lines for tritium extraction process at the tritium building.

III.B Example Submodule Design Description

An example solid breeder submodule design is described here; other details related to each specific design can be found in Ref. 5. As shown in Fig. 9, two breeder design configurations are housed in one submodule to maximize the testing goals. In one configuration, both beryllium and breeder beds are placed perpendicular to the FW facing the plasma region (i.e. edge-on), while in the second configuration, a parallel

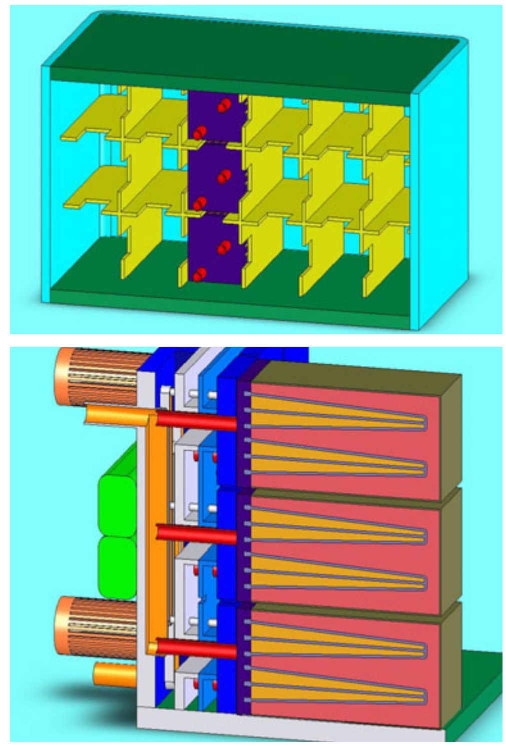

Fig. 8. Schematic view of proposed ITER test unit cells inserted inside EU structural box (top: back view; bottom: side view).

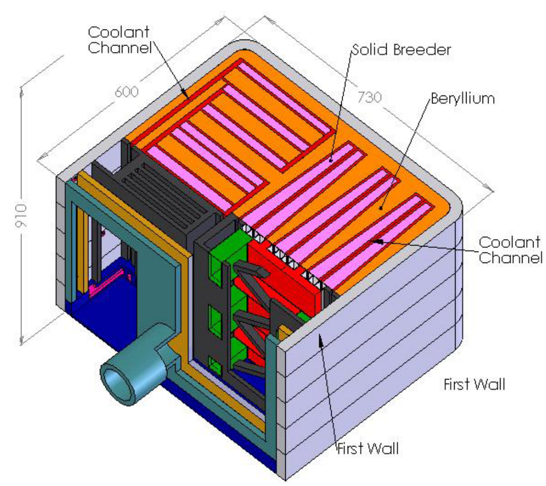

Fig. 9. Schematic view of proposed ITER solid breeder test blanket submodule.

geometry (i.e. layered) is considered. The layered option resembles the blanket concept considered in the US ARIES-CS and HAPL designs.[21]

The design of this submodule takes into account engineering scaling rules to preserve power reactor-like blanket operating conditions under ITER operational conditions including a surface heat load distribution ranging from 0.5 to 0.25 MW/m^2 and a neutron wall load of 0.78 MW/m^2. The 8 MPa helium coolant enters the submodule at a rate of 0.9 kg/s at a temperature of 300^oC (a typical value of helium coolant inlet temperature applicable to any helium cooled blanket designs with FS as a structural material) and is subsequently distributed into 10 first wall cooling paths for surface heat removal. Each first wall cooling path is made up of 5 coolant channels connected in series in order to reduce the coolant flow area and achieve a high helium velocity. A relatively high velocity is needed to ensure an adequately high heat transfer coefficient for removing the surface heat load of 0.5 MW/m^2. The overall first wall thickness is 28 mm, including a coolant channel of 16 x 13 mm^2 and a front wall thickness of 5 mm. The pitch between the coolant channels is 18.2 mm. These design parameters result in thermal-hydraulic and thermo-mechanical performance parameters as listed in Table III, indicating that the temperature and stress magnitudes of the first wall are within the maximum allowable limits of FS structural material. However, the flow rate of 0.9 kg/s gives a lower coolant outlet temperature as compared to typical values of 500^oC needed for achieving a high thermal efficiency in helium-cooled FS blanket designs, thus about 10% of the flow is by-passed away from the breeding zones after the first wall cooling. The remaining coolant in the submodule is divided into four paths for cooling upper and lower caps and two breeding configurations.

TABLE III. Key Operating Parameters for Example Solid Breeder Submodule

Parameter	Design value
Submodule size	0.73x0.91x0.6 m3
Surface heat flux	0.25- 0.5 MW/m2
Neutron wall load	0.78 MW/m2
Helium coolant pressure	8 MPa
Helium inlet/outlet temperature	300/500^oC
Mass flow rate to first wall	0.9 kg/s
Helium temperature rise from first wall	53^oC
Bypass mass flow rate	0.08 kg/s
Mass flow rate to breeding zone	0.82 kg/s
Helium temperature rise from breeding zone	146^oC
First wall maximum temperature	538 ^oC
First wall maximum stress	268 MPa

The helium mainly flows toroidally in the layered configuration layout and flows mostly radially in the edge-on configuration. These breeder unit arrangements create very different breeder temperature profiles. For example, the temperature gradient is mainly in the direction normal to the coolant plates (radial) in the breeder unit of the layered configuration, while there are two temperature gradients (radial and toroidal) found in the breeder unit of the edge-on configuration. The effect of a two dimensional temperature gradient on pebble bed thermomechanical interaction and dimensional stability, and their consequent impacts on thermal and tritium release performance is one of the key feasibility issues that only fusion testing can resolve.

Each breeder unit is scaled to operate with typical power-reactor ceramic breeder operating temperature windows by increasing the bed thicknesses. This compensates for the lower nuclear heating rates in ITER while increasing thermal resistance and thus the temperature gradient. The breeder unit width is about 1.8 cm near the first wall and about 3 to 4 cm at 35 cm away from the first wall. Example temperature distributions found in different breeder units are shown in Figs. 10 and 11. As shown, the calculated temperatures based on typical effective thermal conductivity of breeder (1 W/mK) and beryllium pebble beds (4 W/mk) combined with nuclear heating rates from 1-D nuclear analysis fall within typical operating temperature limits of 850^oC for Li_4SiO_4 and 600^oC for beryllium, although there appear some hot spots up to 670^oC in the beryllium region near the first wall of the edge-on configuration. This may be tolerable; while the temperature may be reduced during the operation due to creep compaction. It is worth noting that the helium velocity in the breeding zone is relatively slow due to a large coolant surface area available for heat transfer, an intrinsic characteristic of a low power density device such as a fusion reactor.

The effect of configuration on tritium breeding performance can also be studied from these submodule tests, although a 3-D neutronics analysis is needed to provide a neutronically-optimized design including beryllium fractional compositions for each configuration. The overall breeder temperature profile in each configuration has an impact on tritium release, which can be studied by analyzing the tritium concentration inside the helium purge gas. For this purpose, each breeder configuration is equipped with its own tritium purge gas line such that the tritium collected from breeder units can

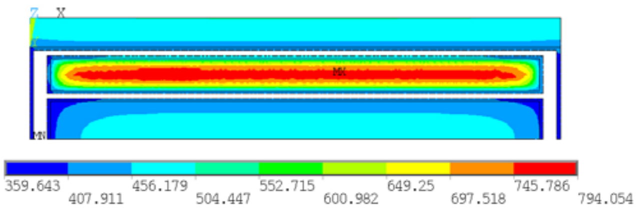

Fig. 10. Calculated breeding unit temperature profiles as designed in a layer configuration (maximum temperature in breeder zone: 794^{o}C; maximum temperature in beryllium: 500^{o}C).

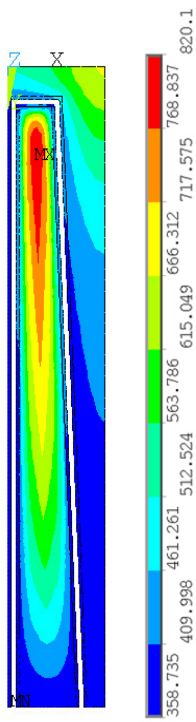

Fig. 11. Calculated breeding unit temperature profiles designed in the edge-on configuration (maximum temperature in breeder zone: 820^{o}C; maximum temperature in beryllium: 650^{o}C).

be traced. All data obtained from this submodule tests would help establish a best configuration for FS helium cooled solid breeder blanket designs for further analysis, in particular in the area of the irradiation effect on overall blanket performance.

IV. CONCLUSIONS ON US SELECTED TBM CONCEPTS

Conclusions on the specific technical issues were summarized earlier in the paper and are presented in more detail in Refs. 2-8. Among the broad conclusions are:

1. Selection between solid and liquid breeders can not be made prior to fusion testing in ITER;

2. All Liquid Breeder options have serious feasibility ("Go/No-Go") issues. Serious R&D and further technical evaluations are required to obtain fundamental data and resolve key issues in order to select with reasonable confidence the most promising concept.

The initial conclusion of the US community, based on the results of the technical assessment to date, is to select two blanket concepts for the US ITER-TBM with the following emphases:

- Select a helium-cooled solid breeder concept with FS structure and Be neutron multiplier, but without an independent TBM (i.e. support EU and Japan using their designs and their TBM structure and ancillary equipment). Contribute only unit cell and sub-module test articles that focus on particular technical issues of unique US expertise and of interest to all parties, for instance pebble bed thermomechanics.

- Focus on testing Dual-Coolant liquid breeder blanket concepts with potential for self-cooling. Develop and design TBM with flexibility to test one or both of these two options:
  - Dual Coolant PbLi concept: a helium-cooled ferritic structure with self-cooled LiPb breeder zone that uses SiC insert as MHD and thermal insulator (insulator requirements in dual-coolant concepts are less demanding than those for self-cooled concepts);
  - Dual Coolant MS concept: a helium-cooled ferritic structure with low melting-point molten salt. Because of the low electrical and thermal conductivity of molten salts, no insulators may be needed.

The key issues for molten salts (REDOX and MHD effects on heat transfer) are being investigated as part of the US-Japanese Universities Collaboration Program, called JUPITER-II. The larger part of US research for the liquid breeder TBM is currently focused on the R&D for the DCLL concept, particularly on the issues for the flow-channel insert.

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REFERENCES

[1] L. GIANCARLI, V. CHUYANOV, M. ABDOU, M. AKIBA, B.G. HONG, R. LÄSSER, C. PAN and Y. STREBKOV, "Objectives and Progress of the ITER Test Blanket Working Group Activities," Proceedings of the 20th IAEA Fusion Energy Conference, Vilamoura, Portugal, (Nov. 2004).

[2] V. CHUYANOV and the ITER INTERNATIONAL TEAM, "Interface of blanket testing and ITER design," these proceedings.

[3] C. P.C. WONG, S. MALANG, M. SAWAN, S. SMOLENTSEV, S. MAJUMDAR, B. MERRILL, D. K. SZE, N.B. MORLEY, S. SHARAFAT, M. DAGHER, P. PETERSON, H. ZHAO, S. ZINKLE and M. ABDOU, "Assessment of Liquid Breeder First Wall and Blanket Options for the Demo Design", these proceedings.

[4] N. B. MORLEY, S. MALANG and I. KIRILLOV, "Thermofluid Magnetohydrodynamic Issues for Liquid Breeders, these proceedings.

[5] A. YING, M. ABDOU, S. SHARAFAT, M. YOUSSEF, J. AN, P. RAINSBERRY and R. HUNT, "Engineering Scaling Requirements for Solid Breeder Blanket Testing", these proceedings.

[6] S. SMOLENTSEV, R. MIRAGHAIE and M. ABDOU, "MHD Effects on Heat Transfer in a Molten Salt Blanket," these proceedings.

[7] M. YOUSSEF and M. SAWAN, "Activation Analysis for Two Molten Salt Dual-Coolant Blanket Concepts for the US Demo Reactor", these proceedings.

[8] M. SAWAN, S. MALANG, C. WONG and M. YOUSSEF, "Neutronics Assessment of Molten Salt Breeding Blanket Design Options," These proceedings.

[9] M.S. TILLACK, S. MALANG, L. WAGANER, X.R. WANG, D.K. SZE, L. EL-GUEBALY, C.P.C. WONG, J.A. CROWELL, T.K. MAU, L. BROMBERG and the ARIES TEAM, "ARIES-ST breeding blanket design and analysis," *Fusion Engineering and Design,* **49-50**, 689 (2000).

[10] P. NORAJITRA, L. BÜHLER, A. BUENAVENTURA, E. DIEGELE, U. FISCHER, S. GORDEEV, E. HUTTER, R. KRUESSMANN, S. MALANG, D. MAISONNIER, A. ORDEN, G. REIMANN, J. REIMANN, G. VIEIDER, D. WARD, and F. WASASTJERNA, "Conceptual Design of the EU Dual-coolant Blanket (Model C)," Proceedings of the 20th IEEE Symposium on Fusion Energy, San Diego (2003).

[11] L. BUHLER and P. NORAJITRA, "Magnetohydrodynamic flow in the dual coolant blanket," Forschungszentrum Karlsruhe Report FZKA 6802 (2003).

[12] A. GESSI, A. AIELLO and G. BENAMATI, "Corrosion behaviour of Eurofer 97 in flowing Pb-17 Li," Presented at the International Workshop on Tritium Management and Corrosion Activities for Liquid Breeder Blankets, Brasimone, Italy (Sep. 27-29 2004).

[13] J. BUCENIEKS, E. PLATACIS, A. SHISHKO, and A. ZIK, "Investigating of MHD phenomena on corrosion of Eurofer steel in Pb-17Li flow under stationary conditions at the temperature of 550C." Presented at the International Workshop on Tritium Management and Corrosion Activities for Liquid Breeder Blankets, Brasimone, Italy (Sep. 27-29, 2004).

[14] J. KONYS, A. AIELLO, G. BENAMATI and L. GIANCARLI, "Status of Tritium Permeation Barrier Development in the EU," These proceedings.

[15] J-E. VETTER, V. CHUYANOV, H-W. BARTELS, M. ENOEDA, L. GIANCARLI, R. HAANGE, K. IOKI, I.R. KIRILLOV, Y. OHARA, S. PAÏDASSI, G. SHATALOV, Y. STREBKOV, S. TANAKA, L. BOCCACCINI, W. DÄNNER, T. HONDA, Y. KATAOKA, T. KURODA, S. MALANG, Y. POITEVIN, M. SEKI and S. J. BOOTH, "Report from the TBWG for the Period of Extension of ITER EDA," (2001).

[16] D. PETTI, Private Communications (2004).

[17] J.W. KOGER and E.L. LAWRENCE, United States Patent No. 3,783,014 (Jan. 1974).

[18] J. R. KEISER, J.H. DEVAN, and D.L. MANNING, "Corrosion Resistance of Type 316 Stainless Steel to Li2BeF4," Oak Ridge National Laboratory Report ORNL/TM-5782 (April, 1977).

[19] W.R. GRIMES and S. CANTOR, "Molten Salts as Blanket Fluids in Control Fusion Reactors," in the Chemistry of Fusion Technology, D. M. Gruen, Ed. Plenum Press, New York, (1971) 161.

[20] C.P.C WONG, M. ABDOU, D.P. CAROSELLA, M. DAGHER, M.P. LABAR, S. MALANG, B. MERRILL, N. MORLEY, M. SAWAN, R.S. WILLMS and D-K SZE, "Ancillary systems for dual coolant liquid breeder test blanket modules – Interim report to the ITER Test Blanket Working Group (TBWG)," General Atomics Report GA–A24754 (2004).

[21] A.R. RAFFRAY, S. MALANG, L. EL-GUEBALY, X. WANG and the ARIES TEAM, "Ceramic Breeder Blanket for ARIES-CS", these preceedings.