

## An overview of US ITER test blanket module program

A. Ying<sup>a,\*</sup>, M. Abdou<sup>a</sup>, C. Wong<sup>b</sup>, S. Malang<sup>c</sup>,  
N. Morley<sup>a</sup>, M. Sawan<sup>d</sup>, B. Merrill<sup>e</sup>, D.K. Sze<sup>f</sup>,  
R. Kurtz<sup>g</sup>, S. Willms<sup>h</sup>, M. Ulrickson<sup>i</sup>, S. Zinkle<sup>j</sup>

<sup>a</sup> University of California, Los Angeles, CA, United States

<sup>b</sup> General Atomics, Fusion Division, San Diego, CA, United States

<sup>c</sup> Fusion Nuclear Technology Consulting, Linkenheim, Germany

<sup>d</sup> University of Wisconsin, Madison, WI, United States

<sup>e</sup> Idaho National Laboratory, Idaho Falls, ID, United States

<sup>f</sup> University of California, San Diego, CA, United States

<sup>g</sup> Los Alamos National Laboratory, Los Alamos, NM, United States

<sup>h</sup> Pacific Northwest Laboratory, Richland, WA, United States

<sup>i</sup> Sandia National Laboratory, Albuquerque, NM, United States

<sup>j</sup> Oak Ridge National Laboratory, Oak Ridge, TN, United States

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### Abstract

A testing strategy and corresponding test plan have been presented for the two proposed US candidate breeder blankets: (1) a helium-cooled solid breeder concept with ferritic steel structure and Be neutron multiplier, but without a fully independent TBM and (2) a dual-coolant helium-cooled ferritic steel structure with self-cooled LiPb breeding zone that uses a flow channel insert as MHD and thermal insulator. Example test module designs and configuration choices for each line of ITER TBM are shown and discussed in the paper. In addition, near-term R&D items for decision-making on testing of both solid breeder and dual-coolant PbLi liquid breeder blanket concepts in ITER are identified.

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### 1. Introduction

The development of a blanket system represents a large-scale undertaking for fusion nuclear technology. Previous studies have shown that some of the key engineering feasibility issues of blanket concepts cannot be established prior to extensive testing in the

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\* Corresponding author. Tel.: +1 310 206 8815;  
fax: +1 310 825 2599.

E-mail address: [ying@fusion.ucla.edu](mailto:ying@fusion.ucla.edu) (A. Ying).

fusion environment [1]. Constructing a test blanket module (TBM) and subsequent testing in ITER marks the first step of fusion technology development in an integrated fusion environment. However, TBM testing in ITER is relatively expensive and any defect which might jeopardize the ITER operation should be avoided. This means that to the extent possible, all known key issues should be resolved and engineering models and codes should be developed to a high degree of confidence. Much of the testing in ITER will focus on integrated behavior of components; therefore, it is important to have an adequate understanding of individual phenomena for the interpretation of test data.

With the US rejoining ITER, the US community has participated in the discussions in the ITER test blanket working group (TBWG) and has proposed to develop, in collaboration with other parties, solid and liquid breeder blanket concepts to be tested in ITER [1]. Aside from the technical merits of the concepts, the criterion for concept selection concerns the resources available for the development and qualification of the TBM prior to installation in ITER. Nevertheless, the US has the goal of extending present designs with improved performance. Since a large R&D program is already available worldwide for the helium-cooled lead–lithium (HCLL) concept [2], with the possibility of using a thermal and electrical insert to achieve higher performance, we focus on the dual-coolant lead–lithium (DCLL) [3–5] with He-cooled ferritic steel structure for the liquid breeder option for ITER testing.

Presently, the US focus is on the following two candidate breeder blanket concepts:

1. a helium-cooled solid breeder concept with ferritic steel (FS) structure and Be neutron multiplier, but without an independent TBM program;
2. a dual-coolant helium-cooled FS structure with self-cooled LiPb breeding zone that uses flow channel inserts (FCIs) as MHD and thermal insulator. We are planning for an independent TBM that will occupy half an ITER test port with corresponding ancillary equipment.

Since much of the ITER TBM effort is a joint venture among participating parties, close coordination and collaboration among international partners are essen-

tial to ensure an efficient execution of TBM design, testing, and qualification.

## 2. Testing strategy, plan, and example TBM designs

### 2.1. Dual-coolant lead–lithium breeder blanket concept

The dual-coolant lead–lithium (DCLL) blanket concept proposed by the US [6] for ITER testing is similar to the HCLL, but utilizes a flowing self-cooled PbLi breeding zone separated from the FS structure by an FCI that serves as both a thermal and electrical insulator [7]. In the DCLL, the structure is cooled by helium at a moderate temperature ( $\sim 450^\circ\text{C}$ ), but the flowing PbLi is allowed to go to a higher temperature ( $\sim 700^\circ\text{C}$ ) with a large temperature drop across the FCI. This high PbLi outlet temperature allows access to the Brayton cycle for high efficiency power conversion, while the structural material is kept at moderate temperatures. Fig. 1 shows the basic geometry of the DCLL with poloidal flowing PbLi channels, and a generic breeding channel showing helium cooling channels in the structure and the FCI separating the hot PbLi from the FS wall.

The still evolving US strategy for ITER testing of the DCLL concept is to aim for flexibility. The test plan must remain flexible in order to respond to future technical issues, as well as the future budget schedule. The baseline assumption underlying the current planning and TBWG documentation is for a series of vertical half-port DCLL TBMs with dedicated ancillary equipment. The US strategy relies on close collaboration with the worldwide effort and interest in Pb–17Li systems. Currently, this effort is primarily concentrated in the EU for the helium-cooled lithium–lead (HCLL) breeder with China beginning to work on Pb–17Li as well. While no agreement has been formalized at this time, a shared and well-coordinated R&D and ITER testing program on the HCLL and DCLL concepts will be mutually beneficial and will result in the most productive and cost effective strategy for developing scientific understanding and technological systems needed for Pb–17Li breeding blankets.

The current US strategy for ITER testing is to progress from basic structural, hydraulic and MHD performance to more integrated test modules in concert

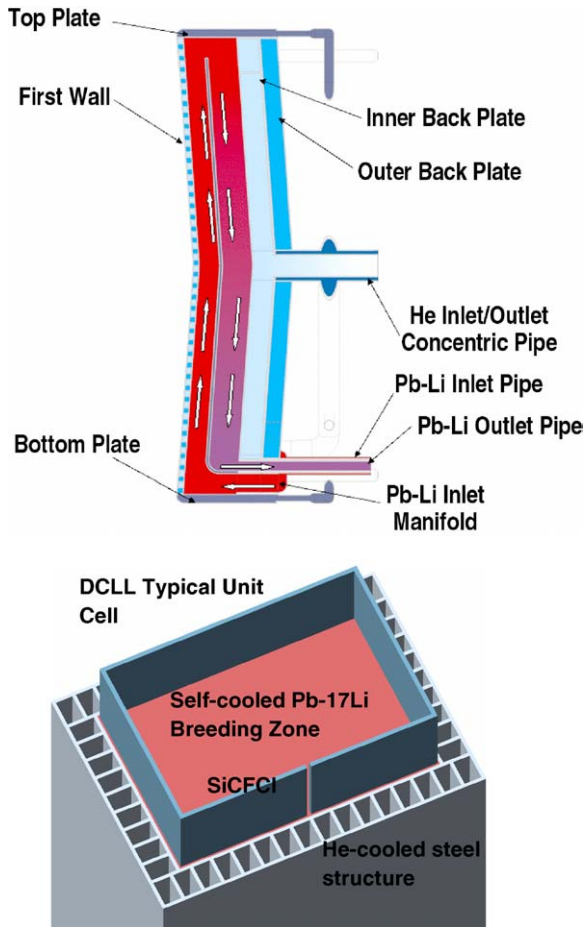


Fig. 1. (Above) DCLL blanket module and (below) typical breeder channel showing helium-cooled wall, SiC composite flow channel insert, and hot PbLi zone inside.

with the first 10 years of ITER operation. The first test module for DCLL is an electromagnetic/structural (EM/S) module designed to withstand EM forces and to measure response to such forces. The EM/S TBM should have similar electrical characteristics to the integrated TBMs as well, so that properly induced currents are simulated. A phased approach proceeding from an empty TBM to one filled with frozen metal (possibly a Pb-alloy other than Pb–Li), stagnant liquid metal and finally flowing liquid metal is suggested. During the H–H phase, only a nominal FW helium coolant flow rate will be required to remove the relatively low surface heat flux coming from the plasma for pulses of around 200 s. Over this period, we plan to pro-

vide additional ancillary systems equipment, including Pb–17Li circulation systems and diagnostic systems. Following the EM/S TBM, a Neutronics TBM will be required during the D–D and early D–T phases to deploy diagnostics specifically to characterize the nuclear spectrum and tritium production. Whether or not such measurements require an independent TBM, or could potentially be integrated in the EM/S and T/M (see below) TBMs are still being evaluated, but it is certain that such experiments will require the use of Pb–17Li itself and not a surrogate.

At the beginning of the low duty cycle D–T phase, a thermofluid/MHD (T/M) TBM is planned. The strategy for the T/M TBM is to allow testing of a variety of FCI geometries and integrated functions at different helium and Pb–17Li flow rates to achieve different outlet temperatures and temperature differentials. The plan during this period is for moderate temperature operation of the TBM with Pb–17Li temperature always below the temperature limits of the FS so that FCI effectiveness can be evaluated safely. Surrogate FCIs using FS or refractory alloy cladding of alumina insulators could potentially be used in testing if SiC composite FCIs are still under development at that time, although it is desirable to move to SiC as soon as possible during this phase to accurately characterize their behavior and effect of failures before more integrated testing. Various geometry and flow conditions will be explored during this period, where the goal is to understand and demonstrate the thermal and electrical insulation properties of the FCI, MHD pressure drop, flow distribution and natural convection effects, and how these features may change over time. During the high duty cycle D–T phase an integrated (I) TBM is planned where the long term operation of the system is explored, including some accumulation of radiation damage in the FCI and tritium and transmutation products in the Pb–17Li. SiC<sub>f</sub>/SiC composite FCIs should be used at that time, with operation of the Pb–17Li again at moderate temperature. When confidence is established, testing of the TBM itself at high Pb–17Li temperature is desired. This is required to demonstrate the high temperature capability and potential failure modes, but it is planned to include a TBM bypass circuit in the Pb–17Li supply system so that the cold-leg Pb–17Li is mixed with the hot Pb–17Li from the TBM before the Pb–17Li proceeds to the heat exchanger in the hot leg. In this way, the high temperature operation of the TBM itself can

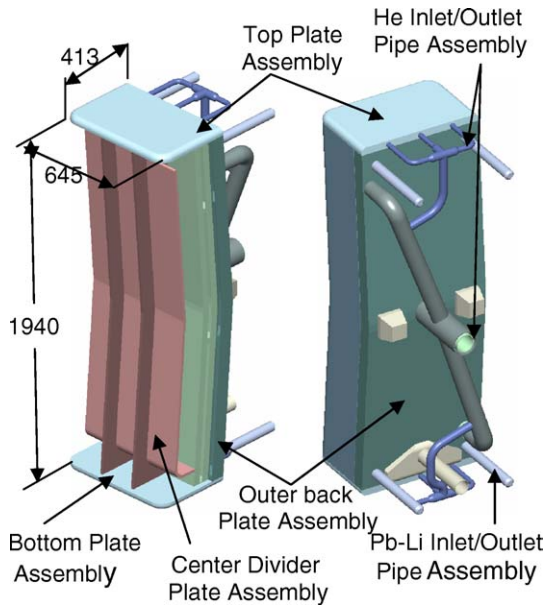


Fig. 2. Vertical half-port integrated DCLL TBM views.

be explored, while the added expense of the high temperature ancillary system can be deferred for testing in later phases of ITER operation beyond the first 10 years.

Fig. 2 shows a depiction of an Integrated TBM taking up a vertical half-port in one of the ITER test ports. Channel dimensions are shown in Fig. 3 (note that the FCIs are not shown). The I-TBM is pictured with a FW shape conforming to the contour of the test port support frame so that the TBM FW is recessed uniformly (the need for this shape is still being evaluated). The radial depth of the I-TBM is set so that the volume of PbLi in the TBM itself is less than the 230 l limit based on a hydrogen generation safety limit. The PbLi volume in the entire ancillary loop is close to 500 l. The division of the flow cross-section into three poloidal channels is a trade-off to keep the channels rather large (typical of power reactor dimensions) on one hand, but still with multiple channel manifolds and MHD phenomena affecting flow balancing, on the other hand.

## 2.2. Helium-cooled solid breeder blanket concept

In this particular concept, the unit cell approach incorporates consistent interface conditions that the host party (in this case EU [8]) requires, including

helium coolant operating pressure and coolant inlet temperature. In addition, the unit cell design is constrained by the physical boundary and dimensions imposed by the host party, with a typical space of about  $19.5 \text{ cm} \times 21.1 \text{ cm}$ . As shown in Fig. 4, testing of three unit cells simultaneously is proposed to provide multiple test data with statistical significance of the test results. The design of the breeder unit cells will coincide with ITER testing objectives. For example, the unit cell designed for neutronics and tritium production rate characterization tests during the early D–T phase will allow the breeder to operate at lower temperature regimes in order to immobilize the tritium inside the breeder regions during the testing. Subsequent removal of the breeder elements allows tritium concentration inside the breeder to be measured and compared with the neutronics code prediction. In this configuration, the breeder arrangement resembles a layered configuration, in which the breeder and beryllium multiplier are arranged parallel to the FW, with thicknesses varying in the radial direction. This is considered a better arrangement for the neutronics tests since relatively flat tritium production and heating rates are possible and thus a high spatial resolution for any specific measurement can be achieved. On the other hand, the thermomechanical test unit cell retains an edge-on configuration for the breeder/beryllium pebble bed arrangement, in which the breeder and multiplier beds are perpendicular to the FW facing the plasma. The differences between the proposed breeder unit cell design and that of the EU HCPB's breeder unit design [9] include: (1) the heat inside the proposed breeder unit is removed by conduction to the adjacent coolant panel perpendicular to the gravity direction and (2) it uses less beryllium by applying a taper design to the breeder element. Examples of the proposed US breeder unit designs are shown in Fig. 5.

The proposed sub-module will take up a testing space of a quarter port ( $73 \text{ cm} \times 91 \text{ cm}$ ) and have its own structural box. The design approach incorporates testing objectives of performance exploration and concept evaluation concurrently being addressed by the sub-module's built-in flexibility. This scheme leads to two breeder design configurations housed in one sub-module, as illustrated in Fig. 6. In one configuration, both beryllium and breeder beds are placed perpendicular to the FW facing the plasma region. In configuration two, a parallel configuration is considered. The latter

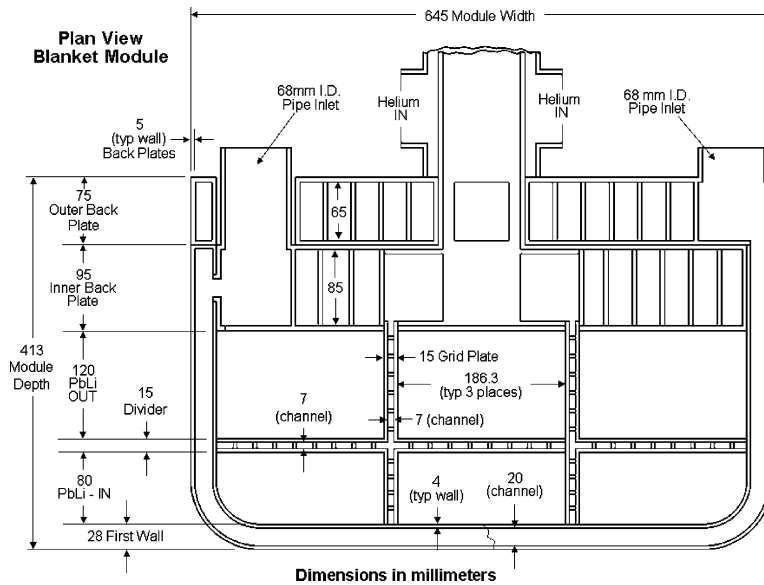


Fig. 3. Cross-section of DCLL I-TBM showing dimensions of poloidal channels.

option resembles the blanket concept considered in the US ARIES-CS design [10]. In addition to their impact on neutronics performance, the breeder pebble bed configurations display distinct thermomechanical performances due to dissimilar temperature profiles across the units. The effect of thermomechanical interactions

on the integrity of the breeder unit is a primary testing objective. Details on the performance analysis and the application of the engineering scaling to solid breeder test article design are discussed in references [11–13].

The decision to test the unit cell or sub-module test articles will be made in a few years, depending on

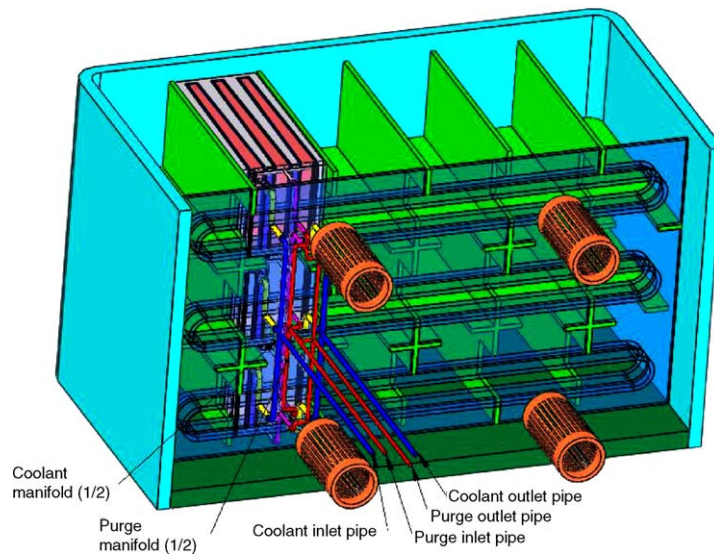


Fig. 4. Proposed solid breeder thermomechanical unit cell test blanket articles housed behind the EU structural box.

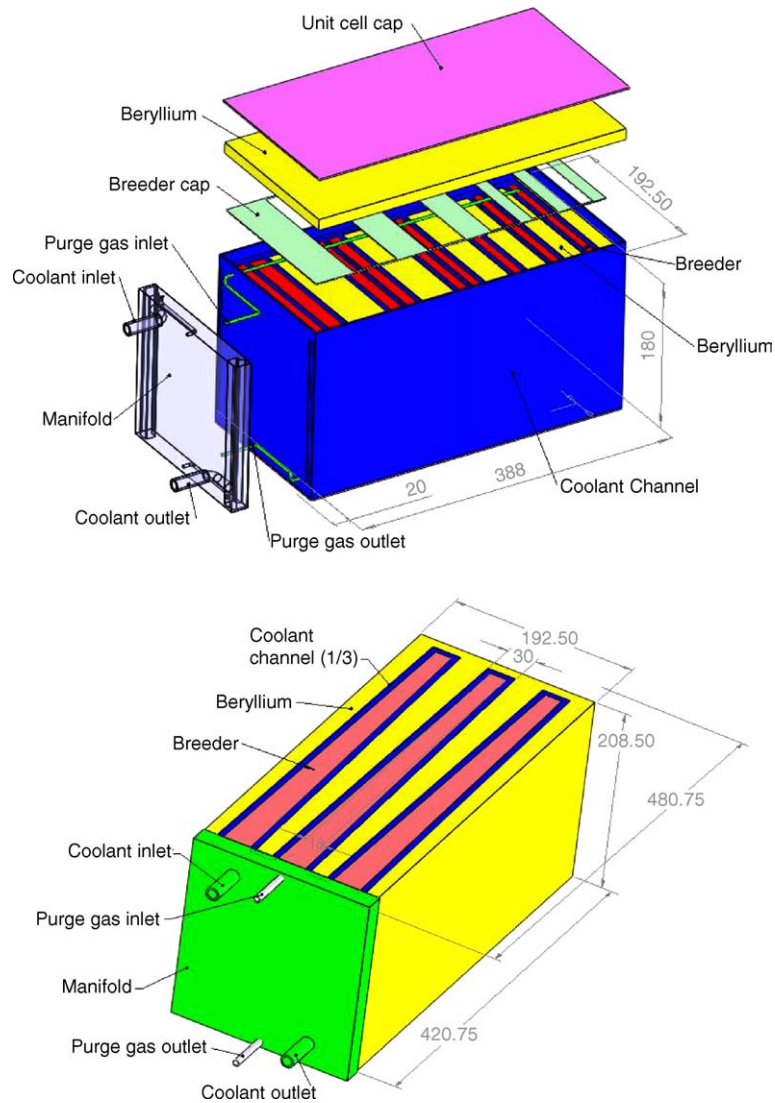


Fig. 5. Proposed unit cell test article designs for neutronics tests (top) and for thermomechanics and tritium release tests (bottom).

the international test program and the US budgetary situation. These test blanket units will be designed and inserted into the helium-cooled ceramic breeder test port (port A) in concert with the ITER operation. Three sequential phases can be envisioned: (1) FW structural thermomechanics and transient electromagnetic (EM/S) tests will be performed during the H–H and D–D phases, (2) neutronics and tritium production rate prediction (NT) tests will also be performed during the early D–T phase, and (3) tritium

breeding, release and thermomechanics explorations (TM) tests during the D–T phase with irradiation to higher neutron fluence. For the last phase, the integrated testing objectives are to study configuration effects on tritium release and pebble bed thermomechanical performance. In addition, since several thermo-physical properties of breeding materials show the largest changes after initial exposure to irradiation, initial study of irradiation effects on performance can be evaluated. Collected data can then be used to optimize

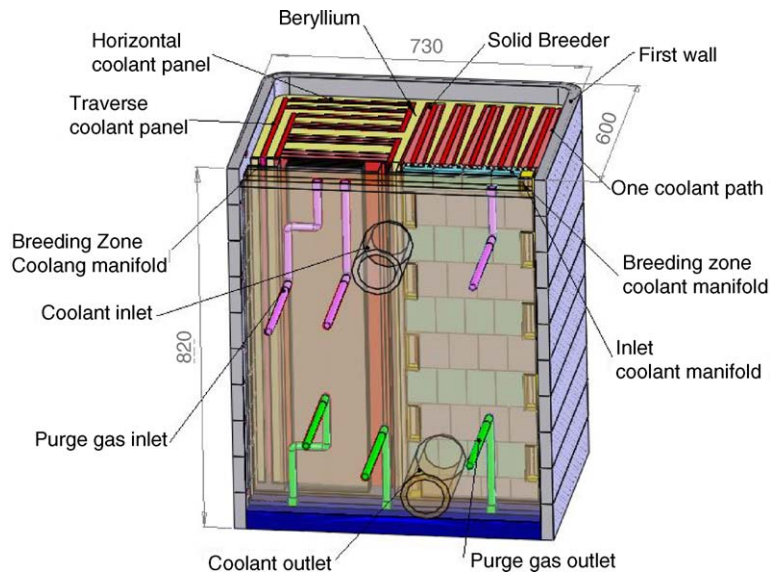


Fig. 6. A schematic view of the proposed sub-module for thermomechanics and tritium release tests.

the configuration aspect of the solid breeder blanket design.

### 3. R&D plan prior to ITER testing

The R&D prior to fusion testing in ITER is viewed as essential to the ITER TBM program from the following two perspectives: (1) the need for qualification to demonstrate safe performance and acceptable availability and (2) the need to acquire adequate knowledge to interpret data from ITER testing. It is necessary to eliminate any uncertainties existing in the proposed TBM.

Research into the DCLL concept in the US is proceeding along several avenues:

- Continued design and analysis of TBM including thermomechanical, thermalhydraulic, neutronics, EM, tritium systems, and safety analyses, including documentation for TBWG.
- MHD and heat transfer analysis of TBM flows in poloidal channels and manifold regions, and the function of FCI as thermal and electrical insulator. The need for a flow PbLi loop for testing MHD

effects and PbLi compatibility with FCIs is envisioned in the near future.

- SiC<sub>f</sub>/SiC composite properties and fabrication techniques including some compatibility experiments with PbLi in static tests and electrical conductivity measurements.
- Generic FS materials research, mostly on irradiation and compatibility effects.

A detailed planning and costing exercise to determine required R&D and resources needs to bring the DCLL concept to ITER testing is just now beginning in the US community. During this time as well, more contacts with the HCLL community in EU and the PbLi development in China are needed, hopefully leading to an agreement regarding joint development and resource sharing among the international community interested in PbLi breeder blankets.

The most important design uncertainties for solid breeder blanket concepts resulting from these issues relate to tritium breeding, tritium permeation and recovery, and breeder thermomechanical behavior. In particular, the integrity of the solid breeder/clad interface plays a key role impacting solid breeder thermal and tritium release performance. A better understanding of the occurrence of a gap at the interface,

the impact of this gap, and the potential and subsequent consequences of particle breakage remains as the near-term focus through continuous efforts on material properties characterizations and consecutive models derivations, as well as the benchmarking of experimental data resembling fusion relevant breeder unit operating conditions. The next stage of model development will focus on identification and quantification of potential failures/limiting factors related to pebble bed material system thermomechanics interactions under cyclic effects. The thermal-hydraulic and flow distribution behavior of helium coolant will also be addressed through small-scale experiments. Furthermore, it is highly desirable to develop and advance available tritium production, heating rate, and neutron spectrum measurement techniques for reliable and meaningful neutronics tests in ITER and this may require further instrumentation R&D effort. Finally, the R&D plan aims at completing the qualification and delivery of the first TBM (either unit cells or a submodule) to ITER 1 year before ITER installation. This allows 1 year for integration into Port A. It has been estimated that it may take about 36 months to complete fabrication and testing of a 0.5 m<sup>3</sup> module starting from the engineering design [14]. The R&D plan calls for beginning this process about 4 years before the target date of the ITER installation.

#### 4. Conclusions

In this paper, US ITER TBM program strategy, initial test plans and example TBM designs have been presented for the solid and liquid breeder concepts. For each line of blanket concepts, three to four test blanket units/modules will be tested from day one of operation to the end of the first 10-year phase of ITER. For both solid and liquid metal breeder designs, during the earlier stages of our testing, engineering scaling design approaches on particular technical issues will be applied. Some of the issues of interest to all parties are MHD effect and performance of the flow channel inserts and pebble bed thermomechanics. Progressively, we will then focus on a more integrated test module toward the later stage of the first 10 years of ITER operation. Prior to ITER testing, a larger part of the US research for the liquid breeder TBM will be focused on the R&D for the DCLL concept, particu-

larly on the issues for the flow channel insert design and performance.

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