ITER
The Way to Fusion Energy

ITER
TBWG

Report from the re-established

Test Blanket Working Group (TBWG)

for the Period of the ITER Transitional Arrangements (ITA)

September 2005

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EXECUTIVE SUMMARY

The ITER Test Blanket Working Group (TBWG) has been re-established in September 2003 until the end of the ITER Transitional Activities (ITA) with a revised charter on the development work performed by the ITER Parties and the ITER Team leading to a coordinated Test Blanket Program in ITER for DEMO-relevant breeding blankets.
Among the technical objectives of ITER it is specifically stated that "ITER should test tritium breeding module concepts that would lead in a future reactor to tritium self-sufficiency and to the extraction of high grade heat and electricity production". All ITER Parties have defined DEMO breeding blanket designs and are performing (or planning to perform) a R&D program leading to the fabrication of blanket mock-ups, called Test Blanket Modules (TBMs), and to their installation and testing in ITER.

TBM Testing Conditions - ITER as an experimental machine will have a rather broad domain of operation around Q=10, with fusion powers between 300 and 600 MW, depending on achievable confinement, density and maximum pressure. The three equatorial ports # 16, 18, and 2 (1.75 m wide x 2.2 m high) have been allocated for TBM testing.
During the D-T phase, typical operating conditions for TBMs include a FW surface heat flux of 0.27 MW/m^2, a neutron wall load of 0.78 MW/m^2, a pulse length of 400 s with a duty cycle up to 22%. Because of possible plasma perturbations, it is however required to design the TBM assuming a localized surface heat flux of 0.5 MW/m^2 with an average of 0.3 MW/m^2. In the H-H phase, the typical TBM surface heat flux is about 0.11 MW/m^2 and the localized increase is up to 0.3 MW/m^2 (average of 0.15 MW/m^2).
The TBMs are inserted in a water-cooled steel frame, 20 cm-thick, which provides a standardized interface with the ITER basic structure, including thermal insulation of the basic machine. TBMs are mechanically connected to the back wall of the frame through flexible supports.

TBM Testing Objectives - Under these test conditions, essential information can be obtained, namely: i) validation of TBM structural integrity theoretical predictions under combined and relevant thermal, mechanical and electromagnetic loads, ii) validation of Tritium breeding predictions, iii) validation of Tritium recovery process efficiency and T-inventories in blanket materials, iv) validation of thermal predictions for strongly heterogeneous breeding blanket concepts with volumetric heat sources, and v) demonstration of the integral performance of the blankets systems.
On the other hand, because of the expected low neutron damage (3 dpa (eq. Fe) after 20 y of operation to be compared with the 70 dpa in a DEMO first wall), long-term radiation effects, failure modes and effects, and reliability have to be addressed in other facilities.

Testing Objectives Feasibility - Because of the significant difference between the testing parameters and DEMO operating conditions, the objectives can be achieved using specific "act-alike" TBMs and associated engineering scaling. It has to be stressed that for completion of the Testing Program it is necessary to have a relatively long sequence of back-to-back pulses, with longer pulse length (at least 1000 s) and with a well defined surface heat flux (in order to use a TBM design to the actual loads). Important information, such as structural integrity under disruption and VDE, acceptability of use of ferromagnetic materials, need of using of Be-protective layer, can be obtained during the H-H phase; this implies the installation of validated TBM since the beginning of ITER operation. Moreover, this information will be needed for the final safety dossier and will be required for licensing in the D-T phase.

TBM System Arrangement - Each TBM system include several associated equipment, such as primary coolant circuits and components, secondary coolant circuits and components, Tritium management components, liquid breeder loop for liquid breeder TBMs, instrumentation packaging and control system, safety-relevant detection systems and valves.
Its complexity indicates that integration in ITER machine and building is a quite difficult operation and that TBM tests will require considerable preparation.

To maximize measurement efficiency, Tritium-related components have to be located in the port cell. The extracted Tritium is directed toward the ITER Tritium system located in the Tritium building. Assuming two TBMs present in each port, the remaining space available in the port cell is mostly used for piping and is not sufficient for hosting the heat extraction components which will then be located in the TCWS vault, with the exception of the liquid metal heat exchanger which, because of their relatively limited dimension, may be possibly located in port cell.

DEMO Breeding Blankets and corresponding TBMs selected by Parties - All Parties have identified, at least, two kinds of DEMO-relevant blanket for testing them in ITER in support of their breeding blankets development programs.
In particular, all ITER Parties are interested in developing Helium-Cooled Ceramic Breeder (HCCB) blankets. The proposals are either to test independent TBMs based on different blanket concepts (China, EU, Japan and RF) or to collaborate with other parties on a common TBM concept (US and Korea). This kind of blankets requires Beryllium as neutron multiplier, and Ferritic/Martensitic Steel (FMS) structures. Depending on the Parties domestic experience, the ceramic breeder could be either Li_2TiO_3, or Li_4SiO_4, or Li_2O and is used in pebbles-bed form.
The other selected breeding blankets differ from Party to Party. Japan has selected a pressurized Water-Cooled Ceramic Breeder (WCCB) blanket, which is a water-cooled version of the corresponding HCCB blanket. All other Parties consider a liquid breeder option. In particular, EU has selected a Helium-Cooled Lithium-Lead (HCLL) blanket, using Lithium-Lead as breeder and neutron multiplier, and FMS structures. China is performing comparative studies between a HCLL blanket and the Dual Coolant version of it that uses He for cooling the FMS structures and LiPb as a self-cooled breeder zone. US has selected a Dual-Coolant Lithium-Lead (DCLL) blanket. Finally, RF has selected a self-cooled Lithium blanket using Be as multiplier and a Vanadium-alloy for structures, and South Korea a Helium-Cooled Lithium (HCLi) blanket using FMS structures.

TBM Testing Strategy - Significant steps need to be performed prior to the TBMs installation in ITER, such as materials fabrication routes and irradiation resistance, out-of-pile tests of mock-ups and associated systems up to the test of prototypical TBM systems, and RH equipment validation.
Concerning the tests in ITER, some Parties envisage focusing on confirmation tests. For this kind of tests, TBM size of half a port appears to be the best compromise between obtaining sufficient measurement sensitivity and maximizing the number of TBMs that can be tested simultaneously taking into account the space limitations and shield efficiency. Other Parties prefer to focus on parallel and sequential, functional tests, which require smaller size sub-modules, each of them designed using engineering scaling.
In all cases, taking into account the reduced FW loads compared to DEMO, the agreed TBM testing approach is to have, for each blanket concept, a series of different TBM designs, each one devoted to specific testing objectives, starting from the initial TBM in the H-H phase where no volumetric nuclear heat is present to the one installed at the end of the D-T phase where pulse-length longer than 1000 s and a large number of back-to-back pulses could be expected.

Coordinated Test Plan in ITER and Port Allocation - At present, the ITER Parties have proposed several independent DEMO-relevant TBMs that cannot all be tested simultaneously on day-one. Space limitation is not only due to the space available in the test ports but also to the limited space available in the port cells outside the bioshield behind each TBM, in the vertical shafts, and in the TCWS vault. The situation will become even more difficult during the D-T phase because additional TBMs are proposed.
However, although at different levels, all proposed TBMs need further specific R&D before proving their acceptability. It is likely that some proposals will be abandoned either for technical or financial reasons and, therefore, it is important to allow some flexibility on the final choice of TBM concepts to be installed in each port. For this reason, the TBWG has proposed that each Party prepares a Design Description Document (DDD) for each TBM system proposed for test in ITER, independently from port allocation or space availability in ports, in port cells and in TWCS vault.

In relation with the Parties proposals, five blanket design families have been defined: 1) He-cooled ceramic/Be blankets, 2) liquid eutectic Lithium-Lead blankets, 3) water-cooled ceramic/Be blankets, 4) liquid Lithium blankets, and 5) molten salt blankets.

For each blanket family, a Working Sub-Group (WSG) has been created in order to evaluate the status of the TBM design and R&D activities, to propose a development strategy agreed among interested Parties, and to define an appropriate testing program. The WSG on molten salt blanket has been put on hold because no significant activity could be performed pending technical results of on-going R&D on key issues. Each of the other 4 WSGs has prepared a test plan "as common and coherent as possible" and tried to identify interest and/or need of collaboration between Parties to possibly avoid duplication.

In order to allow such flexibility without disturbing the ITER construction phase, it is required to fix, for each port, the number and the dimension of the connection lines crossing the vertical shaft between port cells and TCWS vault and other ITER buildings. Moreover, the number of connecting lines has to be minimized because a limited space is available in the vertical shaft.

Looking at the Parties proposals, it appears that most TBMs need He-coolant lines. After optimization on the use of the available space, the TBWG has decided to have the following coolant lines: two He-lines for port #16, one water line and two He-lines for port #18 and three He-lines for port #2. Pipes characteristics have been standardized (same diameter, operating pressure, and material). Other service lines, such as those for heat rejection system and He purge gas, are also required.

The available connection lines at each port will become further constraints on future testing plans together with the space available in the port cell and in the TCWS vault. On the other hand, such constraints still allow sufficient flexibility on the choice of the TBM types to be tested in each port that will have to be performed few years before TBM commissioning.

The rules for the final TBM type selection will have to be defined in future when all required information about R&D results, manufacturing, performances, and funding availability will be known. The final selection of the TBM systems to be installed on day_one could be performed by an ad-hoc group, eventually charged with an appropriate mission. Financial commitments of parties to build, install and test the selected systems need to be developed and agreed in the framework of an international agreement.

In order to facilitate the final selection and to define a fully coordinated test program, a strong collaboration between Parties is required. This collaboration should be stronger than in the past and should aim not only to agree on space sharing or time sharing for testing, but also to promote technical convergence to similar TBM design, and even, whenever possible, to common TBMs.

Remote Handling, Maintenance and Safety Considerations - The general assumed rule is that the testing of TBMs must not spoil ITER operational safety. The present strategy is to consider that all accidental sequences that could involve TBMs remain within the envelope cases assessed for ITER.

Each TBM system has to be checked against specific accidental sequences, in particular in case of LOCA inside the VV (in-vessel Loss-Of-Coolant Accident), inside the TBM (in-box LOCA) or outside the vacuum vessel (ex-vessel LOCA). Failure of a TBM system should have no impact in the overall safety of ITER in term of VV and vault pressurization, tritium and hydrogen release and long-term temperature transient.

The performed safety analyses, summarized in the report and detailed in the DDDs, do not show any major difficulties. However, as the port cells form a containment volume, overpressure events, such as the rupture of a cooling pipe, have yet to be assessed.

TBM replacement occurs in the ITER hot cell, where the whole TBMs/shield plug system is remotely transported in a standard ITER transport cask. Because of the large number of TBM components present in the port cell, parking spaces need to be added in order to allow RH operation with the transport cask. Either the addition of an hot cell port or the modification of the TBM replacement procedure is required if the simultaneous replacement of the 3 test-port plugs has to be possible during a ITER planned shut down (1 month per year).

The ITER hot cell may be used to replace irradiated TBMs, but it is not designed to allow TBM repair and/or Post-Irradiation Examinations. Current ITER Hot Cell scheme considers the irradiated TBMs as a waste object. The presence of hot-cell facilities on the site selected for construction has also been taken into account and could help to solve this issue. Storage needs shall also be re-evaluated.

Main identified concerns on interfaces aspects - Several interface issues between TBMs systems and ITER machine and buildings have been identified and need to be further investigated. The main ones are the following:

- the space and remote handling capabilities in the unique ITER Hot Cell appears clearly insufficient. In fact, TBMs have to be replaced relatively frequently and quickly and PIE has to be performed on the dismantled TBMs;
- the proliferation of component casks and large permanent devices in the port cell is of concern for ITER. They will require temporary parking space during port plug removal and may affect the number of ports that can be maintained simultaneously;
- the space in the TCWS has become insufficient to install all TBM heat transfer equipments requested by Parties. There may be a similar problem in the T-Plant building. The use of common (shared) equipment could help to solve the issue;
- the impact of the confinement strategy for Port Cells recently adopted in ITER has to be checked;
- the port frames procurement is missing from the released ITER procurement packages. This point has to be clarified as soon as possible because it is an essential element for allowing the installation of TBMs at the beginning of ITER operation;
- the cost of the different activities which must be performed by ITER at all stages of testing starting from quality control of design and finishing by handling of irradiated modules are not currently included in the ITER budget. ITER costs associated with the TBM testing program must be recognized and addressed by Parties.

Conclusions and Future Work - The tests of DEMO-relevant TBMs in ITER will give essential information to such a development, although a part of the required R&D results will have to be obtained in other facilities, namely the high dose irradiation effects on blanket materials, materials interactions, and synergistic effects, as well as failure modes and reliability.

Besides the need of checking TBM compatibility with ITER operations, TBM testing in the initial H-H phase is essential both to demonstrate the safety-related performances of TBMs in view of licensing required to operate in D-T phase and to validate remote operations on the TBM systems.

The preliminary TBM integration work in ITER has shown that blanket testing in ITER will be very complex and very lengthy. It is therefore required to make extensive tests in dedicated out-of-pile facilities prior to the installation in ITER.

Present TBM designs are dictated by testing objectives and are performed to assess the TBM behavior under ITER operating conditions. However, to recover the required data, the development of appropriate instrumentation and data acquisition systems is necessary with high priority.

In order to ensure the feasibility of the installation of TBMs on day_one it is urgently required to perform a significant effort on TBM system integration in ITER machine and buildings. Some compromise with other ITER systems and some modifications of the present buildings design are likely to be required.

The work of the TBWG has shown that all TBMs still need significant R&D and that it is technically premature to select now the best performing candidates. It is therefore recommended to focus the blanket R&D programs in the near future on TBMs R&D and to focus the design effort on the TBM engineering designs and on their integration on the ITER machine and buildings.

Design effort to improve DEMO blanket designs will aid in developing better TBM designs that can maximize the effectiveness of tests in ITER. Testing results from ITER TBMs will, in turn, contribute substantially to improving DEMO blanket designs.

Breeding blanket development is one of the most challenging issues for the design and construction of DEMO. TBM tests in ITER will provide essential information toward resolving this challenge.

1 - INTRODUCTION

Breeding blankets represent one of the major technological breakthroughs required from passing from ITER to the next step, usually called DEMO, a demonstration reactor able to furnish electricity power to the grid. In fact, a DEMO breeding blanket and associated systems have to ensure Tritium breeding self-sufficiency, to show good power conversion efficiency and to withstand high neutron fluence.

For this reason, among the technical objectives of ITER it is specifically stated that "ITER should test tritium breeding module concepts that would lead in a future reactor to tritium self-sufficiency and to the extraction of high grade heat and electricity production" [**1-1**].

In order to comply with this mission, since the early stage of the ITER EDA, a working group, called ITER Test Blanket Working Group (TBWG), was officially established by the ITER Council and charged to define and coordinate an appropriate breeding blanket testing program in ITER. The TBWG is formed by representatives from ITER Team and from each of the ITER Parties participating to the corresponding ITER phase. The TBWG made a preliminary assessment of the testing capabilities of the present ITER machine in July 2001 at the end of the ITER EDA extension phase and defined a set of Tests Blanket Modules (TBMs) to be tested in the different operation phases of ITER in the framework of a coordinated testing program [**1-2, 1-3**].

The TBWG has restarted its activity in October 2003 with an enlarged official membership. The new group started its activities at the 11th meeting of the TBWG meeting series (TBWG-11) and includes four representatives of the ITER Team and three representatives of each of the six ITER parties (China, EU, Japan, Korea, Russian Federation, and USA).

The corresponding revised charter (see **Annex 1**) is aiming to:

- provide the Design Description Document (DDD) of the Test Blanket Modules (TBMs) systems proposed by parties including the description of the interfaces with the main ITER machine,
- promote co-operation among parties on the associated R&D programs,
- verify the integration of TBM testing in ITER site safety and environmental evaluations,
- develop and propose coordinated TBMs test programs taking into account ITER operation planning.
-

In accordance with this charter, a TBWG Workplan has been established since the first meeting (TBWG-11). Updating of the Workplan has been performed at each TBWG meeting. The latest and final Workplan is given in **Table 1-1**.

All ITER Parties have defined DEMO breeding blanket designs and are performing (or planning to perform) a R&D program leading to the fabrication of TBMs and to their installation and testing in ITER since the first day of ITER H-H operations.

The three ITER equatorial ports #16, 18, and 2 (1.75 m wide x 2.2 m high) have been allocated for TBM testing. ITER boundary conditions are given in chapter 2. ITER testing capabilities and TBM testing objectives are discussed in chapter 3.

Because of the limited available space for testing, not all the TBMs proposed by the Parties can be tested simultaneously and priorities have to be established. However, because of the many uncertainties pending on the various Parties TBM Programs, the TBWG has decided to leave some flexibility on the final TBMs port allocation provided that it does not endanger the possibility of testing.

In order to achieve this objective, the TBWG has required all details on Parties proposals, has established Working Sub-Groups to assess potential of common testing and of collaboration between Parties, has defined the minimum required interfaces with ITER machine and buildings for proceeding to construction, and, finally, has defined the constraints related to safety, remote maintenance and hot cell operation.

The main body of the report is therefore structured following these guidelines that will be further detailed in the corresponding chapters. In particular, chapter 4 describes in details all the Parties proposals, chapter 5 describes the potential Test Programs as defined by the Working Sub-Groups, chapter 6 presents the main constraints dictated by ITER and describes the interface features that have been fixed by the TBWG for each port. Finally chapters 7 and 8 address the issues of remote maintenance, hot cell, and safety.

The final part of the report includes some considerations about the TBM acceptance tests, the list of the information that TBMs designers have to furnish to the ITER site team for licensing, and finally the conclusions and some proposals of future work for TBWG.

References

[1-1] Summary of the ITER Final Design Report, July 2001, p.9.

[1-2] ITER Test Blanket Working Group, Report from the TBWG for the Period of Extension of the EDA, May 2001.

[1-3] V.A. Chuyanov and the ITER Test Blanket Working Group, ITER Test Blanket Working Group Activities: a summary, recommendations and conclusions, Fus. Eng. & Design 61-62 (2002) 273-281.

Table 1-1 : Main items of the ITER TBWG Workplan for the period of ITER ITA

Task	*Description of the activities / Milestones*	*Due date*
A	**Provide the Design Description Document (DDD) of the Test Blanket Modules (TBMs) Systems proposed by the parties**	
A.1	Definition of the breeding blanket domestic program in each party.	*Oct. 2003*
A.2	Creation of 5 Working Sub-Groups (WSGs), corresponding to the 5 most relevant blanket lines identifies y the Parties.	*Oct. 2003*
A.3	Parties proposal of testing goals and issues (H-H & D-T phases). Agreement on the definition of DEMO (from the blanket viewpoint)	*March 2004*
A.4	Identification of interfaces between TBM-systems and ITER machine & building	*Dec. 2004*
A.5	Parties proposals on TBMs design and R&D programs prior to the installation in ITER	*July 2004*
A.6	Ports allocation: agreement on the “flexibility” approach	*Dec. 2004*
A.7	Design documentation of the TBMs system (drawings, supporting analyses, tests and R&D results,..)	*June 2005*
A.8	Release of TBMs DDDs by Parties	*September 2005*
B	**Promote co-operation among ITER Partners on the TBMs development program**	
B.1	Presentation of the parties “Fusion Program”	*Oct. 2003*
B.2	Identification of common R&D lines (through WSGs)	*March/Dec. 2004*
B.3	Recommendations on exchange of information and tasks sharing for common research lines (eventually, IEA Agreements when applicable)	*July 2005*
C	**Verify integration of the TBMs in the safety & environmental evaluations made by the selected site**	
C.1	Identification of the information required by the site taking into account of the experience of the present site studies	*Oct. 2003*
C.2	Release information towards the Team Leader of the selected site on the TBMs testing program and request of an assessment of the potential impacts (after decision on Site)	*Summer 2005*
D	**Develop the co-ordinated blanket test program (by WSGs)**	
D.1	Identification of common testing objectives among partners and assessment of potential combined TBMs, space sharing and time sharing (WSGs)	*July 2004*
D.2	Development of a coherent testing schedules in each WSGs with reference to ITER operation schedule on the basis of the “flexibility” guideline	*December 2004*
D.3	Proposal of co-ordinated test program from each WSGs	*July 2005*
	Progress Report 1	*June 2004*
	Progress Report 2	*Feb. 2005*
	Report to Interim Project Director and PT Leaders	***September 2005***

2 - ITER BOUNDARY CONDITIONS & TESTING PARAMETERS

2.1 ITER Parameters

On the basis of experimental data available ~5 years ago ITER has been designed to achieve the following technical objectives:

- extended burn in inductively driven plasma with Q=10 (the possibility of controlled ignition should not be precluded) and with a duration sufficient to achieve stationary conditions on the time scales characteristic of plasma processes;
- aiming at demonstration of steady state operation using non-inductive current drive with Q~5;
- demonstration of the availability and integration of technologies essential for a fusion reactor (such as Super Conductivity and Remote Handling);
- test of components for future reactors (such as High Heat Flux components);
- test of tritium breeding blanket module concepts that would lead in a future reactor to tritium self-sufficiency the extraction of high-grade heat, and electricity production.

In accordance with these objectives it was expected that ITER as an experimental machine will have rather broad domain of operation around Q=10 with fusion powers between 300 and 600 MW (See **Figure 2.1-1**) depending on the ratio of the achievable confinement enhancement in H-mode to the expected one (H_H), the achievable density (n_e/n_{GW}) and the maximum pressure ($ß_N$).

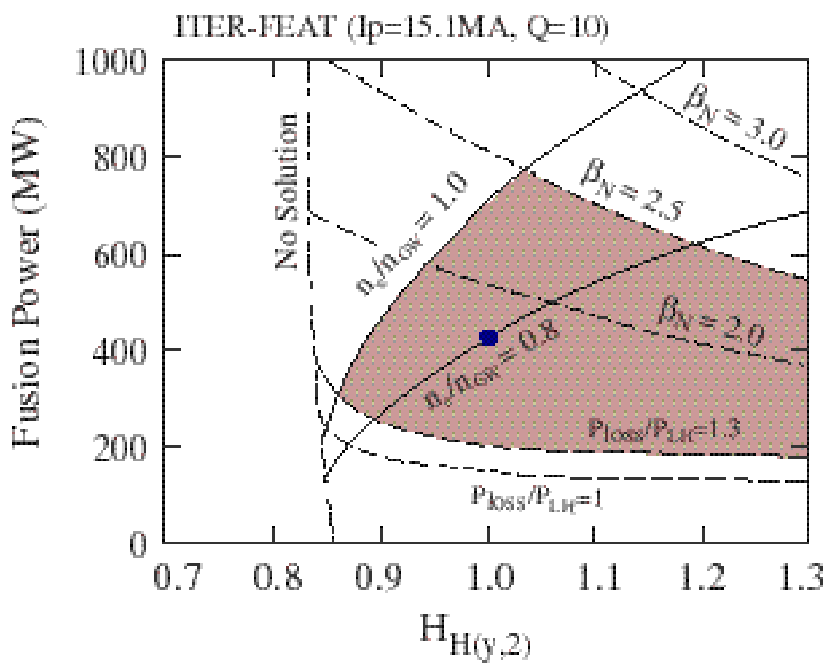


Figure 2.1-1: Operational space of ITER for Q=10

Three main regimes of operation were envisaged:

a. Inductive operation, when the plasma current is driven by the ITER central solenoid (CS) and other poloidal coils.

In this case the duration of plasma current is limited by total available magnetic flux and for a typical plasma current ~15 MA one can expect burn times ~400 s and minimum repetition time >1800 s.
ITER is optimized for this kind of operation. One can expect to reach the neutron loading on test modules ~ 0.76 MW/ m^2.

b. Non-inductive operation, when the plasma current is driven by injection of particle and/or HF/UHF energy beams in the plasma.

In this case duration of plasma current is limited by technical capabilities of external systems and for current ITER design one can expect to get pulses up to 3000 sec with a minimum repetition time > 12000 s. Physics of these regimes is not known so well as for the inductive scenario and a significant research and optimization will be needed before these regimes may be used for testing purposes. It is expected to get Q ~ 5, fusion power ~ 360 MW and neutral wall loading ~0.55 MW/ m^2.

c. Hybrid operation when the plasma current is driven by a combination of inductive (CS) and non-inductive means.

This scenario combines advantages and limitations of two previous ones. Physics is better known. Higher fusion power (~400 MW at Q=5.4) and higher wall loadings (0.62 MW/m^2) may be achieved, but the burning duration is limited ~1000 s. Minimum repetition time is 4000 s.
Reference plasma parameters of ITER are given in the **Table 2.3-1**.

During last several years there were no changes in the main parameters of ITER.
However significant physical researches have been done to justify selected parameters and clarify possible operational conditions and expected parameters. High plasma density and good confinement are achieved on JET at normalized parameters equal and even higher that was assumed for ITER (H~1 at n/n_{Gr} ~1, β_N>1.8, q_{95}~3). There is no degradation of confinement with increase of β_N (JET, D3D). Density profile in ITER will be not flat and as a result fusion power may be higher than expected. Significant progress has been achieved in understanding and experimental investigations of non-inductive and hybrid regimes of current drive. Hybrid regimes with plasma current I_{pl}=12 MA, Q>10 and duration of burn > 1000 s are expected now to be possible for ITER. These regimes if realized will be the most promising for blanket testing.

2.2 ITER Operation

ITER operational plan (**Figure 2.2-1**) has been discussed up to now only for the first 10 years. It includes 1 year of integration on sub-system level, 2.5 years of initial operation in hydrogen, a brief DD phase and a long tritium phase.

Tritium phase will start with initial operation with 400 s 500 MW inductive pulses which will be followed by “hybrid” operation with longer (at least 1000 s) pulses and after some additional studies by long non-inductive steady state pulses.

The program for the second 10 years will be decided later after review of achieved results. It will be focused on improvement of overall performance and reliability and testing of components with higher neutron fluence. It is difficult to believe that higher fusion power will be sustainable in pulses long enough for testing, but with fusion powers < 600MW longer pulses and higher duty factor will be probably achievable with a moderate investment.

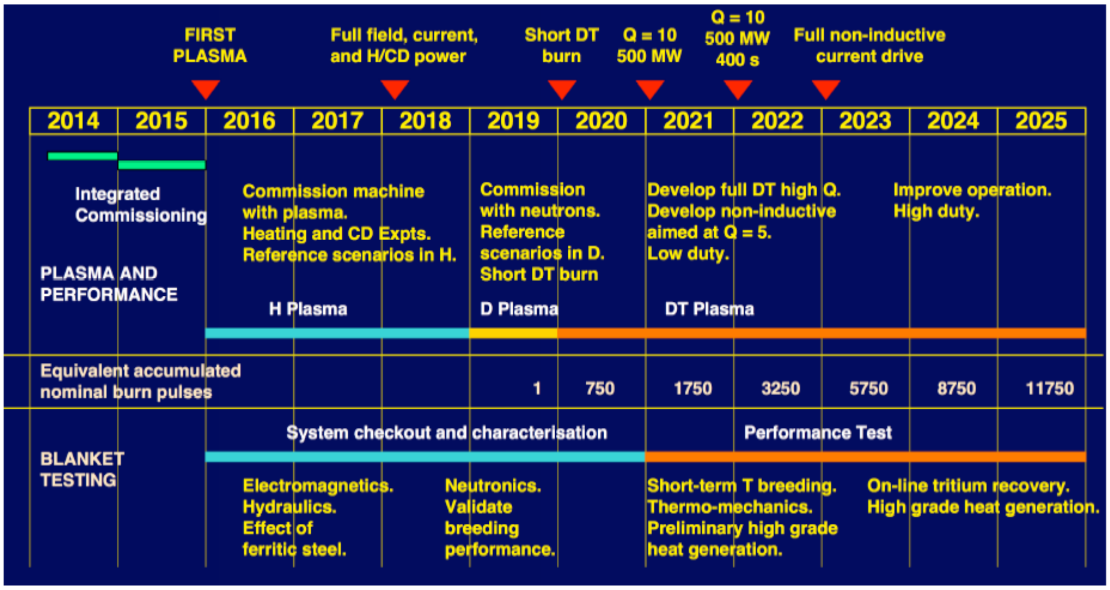


Figure 2.2-1: ITER operational plan
(assuming that ITER International Organization will be set up before the end of 2005 and the "License to Construct " will be granted in 2007)

ITER is designed for ~30000 pulses. Average neutron flux in the tritium phase is > 0.5 MW/m^2. Maximum neutron flux at the equatorial level is up to 0.8 Mw/m^2 at 500 MW. Average fluence after 20 years of operation may reach 0.3 MWa/m^2 (See **Figure 2.2-2**).

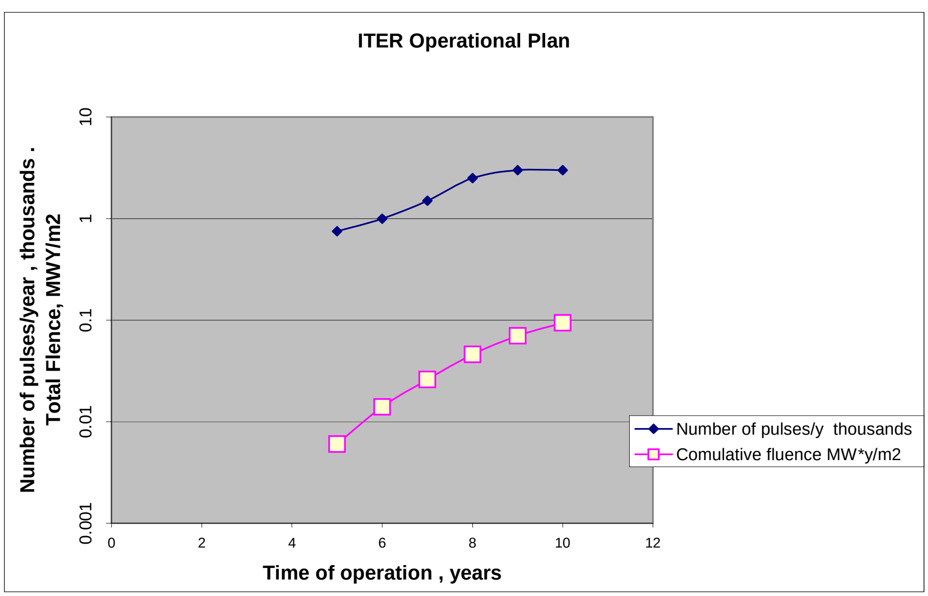


Figure 2.2-2: ITER Operational Plan

To be sure that test blanket modules are compatible with tokamak operation the test modules or their representative equivalents must be installed as early as possible before beginning of the DT operation.
There are several issues, which must be investigated at this stage:

- operation of test modules and supplementary equipment in strong magnetic field,
- forces , acting on test modules during disruptions,
- sputtering of the bare steel surface of the test module's first wall and necessity to use a Beryllium protective layer,
- interference of the test modules with plasma confinement,
- thermal loads on the test module's first wall.

Moreover, most TBMs will be made of a martensitic/ferritic steel. Their magnetization in the ITER field will generate "error fields" − small perturbations of the axial symmetry of the poloidal magnetic field. Even small error fields ($\sim 10^{-4}$ of toroidal field) can induce in the plasma locked (i.e. non-rotating) modes. Locked modes are not stabilized by plasma rotation. Magnetic islands grow, degrade fusion performance and lead to disruptions. The error field may influence confinement of fast particles and change heat load on the test modules themselves. There are also other sources of the error fields like TF or PF coil misalignment creating error fields of a similar amplitude but probably with different phases. The ITER magnet system is designed to compensate these error fields.
However, estimates show that the amount of ferritic steel in the current design is so high that the amplitude of the error fields created by test modules is close to limits for compensation. Taking in account uncertainties in prediction of the total error field and in tolerance of the ITER plasma to error fields ITER does not request to change the design of test modules today and to limit the amount of ferritic steel. However, if the experiments during the hydrogen phase will show that the level of the error fields is unacceptable, test modules designers must be ready to such a request.

2.3 Pulse characteristics, heat and neutron loads distribution

2.3.1 Pulse Characteristics

As described in the PID, variants of the nominal scenario are designed for plasma operation with extended-duration, and/or steady-state modes with a lower plasma current operation, with H, D, DT and He plasmas, potential operating regimes for different confinement modes, and different fuelling and particle control modes. Flexible plasma control should allow for "advanced" tokamak scenarios based on active control of plasma profiles by current drive or other non-inductive means.

Four reference scenarios are identified for design purposes and shown below. Three alternative scenarios are specified for assessment purposes where it shall be investigated if and how plasma operations will be possible within the envelope of the machine operational capability with the possibility of a reduction of other concurrent requirements (e.g. pulse length).

Design scenarios (more details are summarized in **Table 2.3-1**):

1. Inductive operation I: Fusion power = 500 MW, Q = 10, Ip = 15 MA operation with heating during current ramp-up, burn time = 400 s.
2. Inductive operation II: Fusion power = 400 MW, Q = 10, Ip = 15 MA operation without heating during current ramp-up, burn time = 400 s.
3. Hybrid operation: Fusion power = 400 MW, burn time = 1000 s.
4. Non-inductive operation I (weak negative shear operation): Fusion power = 356 MW, burn time = 3000 s.

The operation scenario for Inductive Operation I is summarized in **Table 2.3-2** and **Figure 2.3-1**. The minimum repetition time is 1800 s which gives the maximum duty factor 0.22 in the Induction Operation I. On the other hand, in Hybrid Operation or Non-inductive Operation I the maximum duty factor 0.25 is obtained, as shown in **Table 2.3-1**. These values of the duty factor are defined only for the period during repeated pulses without any pauses.

The present operation assumption (after initial stages of the ITER operation) is as follows:

- 10 cycles of operation per year,
- ~10 days of wall conditioning operation in one cycle,
- ~ 2 weeks of plasma operation in one cycle,
- 3000 equivalent number of nominal pulses (Inductive Operation I) per year,
- Average fluence on the FW is 0.024 MWa/m^2 per year.

The duty factor in this operation assumption is

0.04 average in year,
0.11 average in 2 weeks of plasma operation.

2.3.2 Heat Loads Design Conditions for TBMs

The surface heat load conditions in D-T Phase are summarized in **Table 2.3-3(a)**. The heat flux during burn time in normal plasma operation is 0.27 MW/m^2 for 3,000 cycles (equivalent nominal pulses) per year. The maximum heat load is 0.5 MW/m^2 for 100 cycles per year taking into account MARFE (transient) and other phenomena, such as re-ionization or toroidal field ripple effects (in steady state but localized). Since the area of 0.5 MW/m^2 is localized, the average heat load in TBM is not more than 0.3 MW/m^2 in the case of steady state condition, (the average can be for the TBM overall, or in the toroidal or poloidal direction). As a simplified approach, it is proposed that the test blanket module (TBM) withstands 0.5 MW/m^2 for 3,000 cycles per year to maintain an adequate design margin, where the design value for the FW in general is 0.5 MW/m^2 for 30,000 cycles for the whole ITER life. The definition of the disruption heat loads is also simplified to be 0.68 MJ/m duration 1 ms and 0.72 MJ/m^2 duration 40 ms, 300 cycles per year, as shown in **Table 2.3-3(a)**. In the H-H phase, the surface heat loads are somewhat lower than those in the D-T Phase. The heat flux during burn time in normal plasma operation is 0.11 MW/m^2 for 600 cycles per year (the total 1000 cycles for 2.5 years), as shown in **Table 2.3-3(b)**. The maximum heat load is 0.3 MW/m^2 for 100 cycles per year.

The neutron wall loading has been calculated based on 500 MW fusion power (Inductive scenario I). It has been calculated that the average neutron wall loading is 0.56 MW/m^2. The maximum neutron wall loading is located at the equatorial level in the outboard region. Therefore, the neutron wall loading on the TBMs is as high as 0.78 MW/m^2 (see **Figure 2.3-2**). It is defined as a design value that the average neutron fluence in the whole machine life is 0.3 MWa/m^2. This means that the total neutron fluence on the TBMs is 0.42 MWa/m^2. As shown in **Table 2.3-4** and **Figure 2.2-1** (operation plan for first 10 years), it will take more than 10 years to reach this fluence. On the other hand, there is a possibility to reach higher fluence when a long-pulse operation (Hybrid or non-inductive operation I) is achieved and higher duty factor is maintained.

Table 2.3-1: Main Parameters of Design Scenarios (PID chapter 3.2)

Parameter	1.Inductive operation I	2.Inductive operation II	3.Hybrid operation	4.Non-inductive operation I
R/a (m/m)	6.2 / 2.0	6.2 / 2.0	6.2 / 2.0	6.35 / 1.85
Volume (m^3)	831	831	831	730
Surface (m^2)	683	683	683	650
Sep. length (m)	18.2	18.2	18.2	16.9
Cross-section (m^2)	21.9	21.9	21.9	18.7
Toroidal field, B_T (T)	5.3	5.3	5.3	5.18
Plasma current, I_P (MA)	15.0	15.0	13.8	9.0
Elongation, κ_x/κ_{95}	1.85 / 1.7	1.85 / 1.7	1.85 / 1.7	2.0 / 1.85
Triangularity, δ_x/δ_{95}	0.48 / 0.33	0.48 / 0.33	0.48 / 0.33	0.5 / 0.4
Confinement time, τ_E (s)	3.4	3.7	2.7	3.1
$H_{H\text{-}IPB98\ (v.2)}$	1.0	1.0	1.0	1.57
Normalised beta, β_N	2.0	1.8	1.9	3.0
Electron density, $\langle n_e \rangle$ ($10^{19}m^{-3}$)	11.3	10.1	9.3	6.7
f_{He} [%]	4.4	4.3	3.5	4.1
Fusion power, P_{fus} (MW)	500	400	400	356
P_{add} (MW)	50	40	73	59
Energy multiplication, Q	10	10	5.4	6
Burn time (s)	400	400	1000[(1)]	3000[(1)]
Minimum repetition time (s)	1800	1800	4000	12000
Total heating power, P_{TOT} (MW)	151	121	154	130
Radiated power, P_{rad} (MW)	61	47	55	38
Alpha-particle power, P_α (MW)	100	80	80	71
Loss power, P_{loss} (MW) (conduction)	104	87	114	93
L-H transition power, $P_{L\text{-}H}$ (MW)	51	48	45	36
Plasma thermal energy, W_{th} (MJ)	353	320	310	287

[(1)] The Extended burn under the hybrid and non-inductive operations may be accomplished with additional investment for auxiliary systems.

Table 2.3-2: Design Scenario 1: Inductive Operation I

Phase	XPF[(1)]	SOH[(1)]	SOF/B[(1)]	EOB[(1)]	EOC[(1)]
t (s)	30	70	100	500	560
I_P (MA)	7.5	13	15	15	12
P_{add} (MW)	0	50	50	50	0

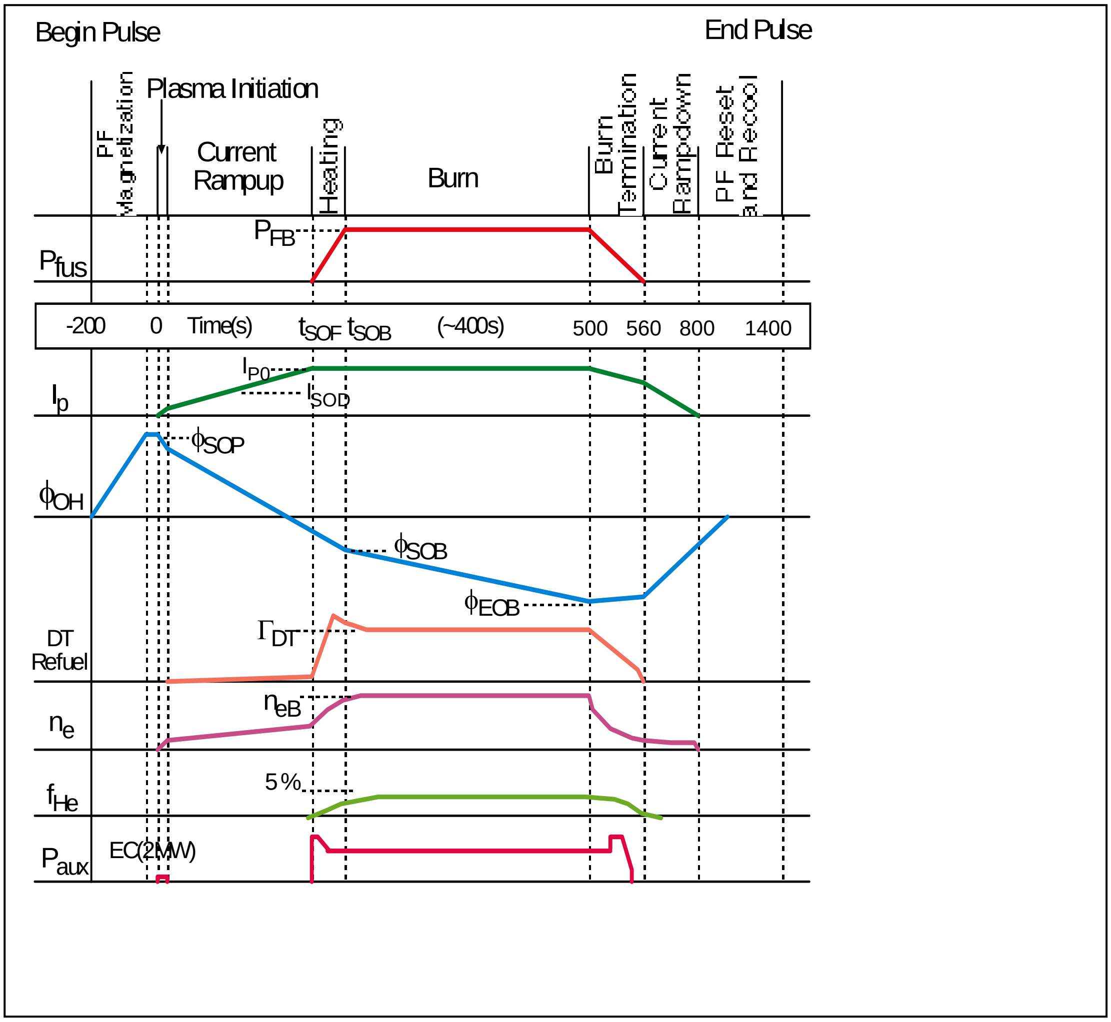


Figure 2.3-1: Burn cycle and plasma/PF parameter waveforms for Inductive Operation I

Table 2.3-3(a): Heat Loads Conditions During D-T Phase for Test Blanket Modules

Parameter	**Design values for test blanket module**	**Comments**	**Basis**
Inductive operation		**Fusion power; 500 MW, Burn time 400 sec**	
Heat flux during burn time in normal plasma operation	0.27(1)-MW/m^2 3,000(3) cycles (equivalent nominal pulses) per year	It is preferable that the test blanket module (TBM) withstands 0.5 (4) MW/m^2 3,000 cycles per year to maintain an adequate design margin, where the design value for the FW in general is 0.5(4) MW/m^2 30,000 (5) cycles for the whole ITER life.	(1): Radiation Loss:136MW / (680 m^2 x 1.06) x 1.41 (peaking factor: TBD) = 0.27 MW/m^2 DRG1Table1.15-1, Table 1.1-1, Table 1.21-4 (3): DRG1Table 1.29-1 (4): DRG1Table 1.15-2 (5): DRG1Table 1.1-1
Fusion Power Excursion	0.30(6)-MW/m^2 Duration 10(7) sec 1,000 cycles	Fusion power excursion: +20%(7) will increase the radiation loss by 10-15 %.	(6): Total Radiation Loss: (100 MW x 1.2 + 73 MW) x 1.05 x 0.75 = 152MW instead of 136MW. This gives 0.30 MW/m^2 (7): DRG1Table 1.15-1
Surface heat flux due to MARFE or other phenomena	0.5(8) MW/m^2 (*) Steady-state (in a localized region(*)) 100(4) cycles per year	MARFE (Duration; 10(9)sec) in the outboard region has a small probability. When MARFE is detected, the plasma will be shut-down. The heat load due to other phenomena can be steady-state, but the high heat load is localized..	(8): Estimation based on A. Kukushkin's private communication (9): DRG1Table 1.15-2 (*) In the case of steady -state condition, the average heat load in TBM is not more than 0.3 MW/m^2 (the average can be for the TBM overall, or in the toroidal or poloidal direction).
Disruption heat load	TBM is recessed: 0.55(10)MJ/m^2 Duration 1(11) ms 300 cycles per year	Peak energy deposition is defined to be 0.36(11) MJ/m^2 in the present DRG1. However, according to recent data (JET,ASDEX-U), the maximum heat load on the FW can be higher. DRG1 will be updated..	(10): 350MJ(11) x 0.8(12) / (680 m2 x 1.06) x 1.4 (peaking factor) =0.55 MJ/m^2 (11): DRG1Table 1.16-2 (12): ~80 % of the thermal energy can go to the FW at maximum. (Experimental data in JET and ASDEX-U)
Disruption heat load during current quench	0.72MJ/m^2 Duration 40 ms 300 cycles per year	All of the internal magnetic energy is assumed to be radiated with peaking factor 1.4.	370MJ / (680 m2 x 1.06) x 1.4 (peaking factor) =0.72 MJ/m^2
Heat load due to ELM and "blob"	Negligible (TBD)	The heat load due to ELM and "blob" will be negligible, considering the distance from the separatrix and recess of the TBM FW locations.	The heat load conditions due to these effects might be revisited based on additional experimental data in the future.
Neutron wall load on TBM FW	0.78(5) MW/m^2	Average neutron wall loading is 0.56(5) MW/m^2, and the local neutron wall loading in the outboard equatorial port region is 0.78 MW/m^2	

Pulse length	Typical case; 400(16)sec (burn time) 1800(16)sec (repetition time)		
Duty factor	Peak burn duty factor: 0.25(5)		
Non-inductive operation		**Fusion power; 356(16) MW, Burn time 3,000(16) sec**	
Heat flux during burn time in normal plasma operation	0.20(13) MW/m^2 Duration 3,000 sec TBD cycles (equivalent nominal pulses) per year	From thermal fatigue point of view, this condition will be less severe than Reference Case.	(13): Radiation Loss:(71+59)MW x 1.05 x 0.75 = 102.4 MW instead of 136MW. This gives 0.20 MW/m^2 DRG1Table1.3-1
Fusion Power Excursion	0.23(14)MW/m^2 Duration 10(7) sec TBD( ) cycles (?)	Fusion power excursion: +20%(7) will increase the radiation loss by 10-15 %.	(14): Radiation Loss:(71 x1.2+59)MW x 1.05 x 0.75 = 113.6 MW instead of 136MW. This gives 0.23 MW/m^2
Surface heat flux due to MARFE or other phenomena	0.5(8) MW/m^2 (*) Steady-state (in a localized region(*)) 100(4) cycles per year	MARFE (Duration; 10(9)sec) in the outboard region has a small probability. When MARFE is detected, the plasma will be shut-down. The heat load due to other phenomena can be steady-state, but the high heat load is localized..	(*) In the case of steady -state condition, the average heat load in TBM is not more than 0.3 MW/m^2 (the average can be for the TBM overall, or in the toroidal or poloidal direction).
Disruption heat load	TBM is recessed: 0.45(15)MJ/m^2 Duration 1(11) ms		(15): 287MJ(16) instead of 350 MJ gives 0.45 MJ/m^2 (16): DRG1Table 1.3-1
Disruption heat load during current quench	0.26MJ/m^2 Duration 24 ms	All of the internal magnetic energy is assumed to be radiated with peaking factor 1.4.	133MJ / (680 m2 x 1.06) x 1.4 (peaking factor) =0.26 MJ/m^2
Heat load due to ELM and "blob"	Negligible (TBD)	Same as above	
Neutron wall load on TBM FW	0.56 MW/m^2	The wall loading is proportional to the fusion power. Average neutron wall loading is 0.40 MW/m^2, and the local neutron wall loading in the outboard equatorial port region is 0.56 MW/m^2	
Pulse length	Typical case; 3000(16)sec (burn time) 12000(16)sec (repetition time)		
Duty factor	Peak burn duty factor: TBD		

Possibility of high power operation		**This is not a design requirement, only for assessment. Fusion power; 700 MW, Burn time = 100 [20]~200 sec**	
Heat flux during burn time in normal plasma operation	0.38[17] MW/m^2 TBD cycles (equivalent nominal pulses) per year		[17]: Radiation Loss:(140+110[5])MW x 1.05 x 0.75 = 170 MW . This could give 0.38 MW/m^2
Fusion Power Excursion	0.43[18]MW/m^2 Duration 10[7] sec TBD cycles	Fusion power excursion: +20%[7] will increase the radiation loss by 10-15 %.	[18]: Total Radiation Loss: (140 MW x 1.2 + 110 MW) x 1.05 x 0.75 = 219MW. This gives 0.43 MW/m^2
Surface heat flux due to MARFE or other phenomena	0.5[8] MW/m^2 [*] Steady-state (in a localized region[*]) 100[4] cycles per year	MARFE (Duration; 10[9]sec) in the outboard region has a small probability. When MARFE is detected, the plasma will be shut-down. The heat load due to other phenomena can be steady-state, but the high heat load is localized..	[*] In the case of steady -state condition, the average heat load in TBM is not more than 0.3 MW/m^2 (the average can be for the TBM overall, or in the toroidal or poloidal direction).
Disruption heat load	TBM is recessed: 0.68[19]MJ/m^2 Duration 1[11] ms		[19]: 434MJ[20] instead of 350 MJ gives 0.68 MJ/m^2 [20]: DRG1Table 1.3-6
Disruption heat load during current quench	0.72MJ/m^2 Duration 40 ms	All of the internal magnetic energy is assumed to be radiated with peaking factor 1.4.	370MJ / (680 m2 x 1.06) x 1.4 (peaking factor) =0.72 MJ/m^2
Heat load due to ELM and "blob"	Negligible (TBD)	Same as above	
Neutron wall load on TBM FW	1.09 MW/m^2	The wall loading is proportional to the fusion power. Average neutron wall loading is 0.79 MW/m^2, and the local neutron wall loading in the outboard equatorial port region is 1.09 MW/m^2	
Pulse length	Typical case; 100[20]sec (burn time) TBD[20]sec (repetition time)		
Duty factor	Peak burn duty factor: TBD		
Other Conditions			
Neutron fluence at the TBM FW	Minimum:0.42MW a/m^2 ForAssessment:0.7 0 MWa/m^2	Average neutron wall loading at the FW is 0.3[5] MWa/m^2 (minimum), 0.5[5] MWa/m^2(for assessment).	

First wall armor material	Coated beryllium (TBD)	Sputtering erosion: Be ~ 50 µm per year (3,000 equivalent nominal pulses)	
Simplified burn time heat load	**0.5 MW/m^2 3,000 cycles per year**	**(Simplified disruption heat loads: 0.68 MJ/m Duration 1ms, 0.72MJ/m^2 Duration 40 ms, 300 cycles per year**)	**<Proposed envelope conditions>** It is preferable to use a simplified load condition envelop the conditions described above.

Table 2.3-3(b): Heat Load Conditions During H-H Phase for Test Blanket Modules

Parameter	Design values for test blanket module	Comments	Basis
Inductive operation		**Fusion power; 0 MW, Flat-top time 100-200sec**	
Heat flux during burn time in normal plasma operation	0.11$^{(1)}$-MW/m^2 up to 600 cycles per year (Total 1000 cycles for 2.5 years)		$^{(1)}$: Total Radiation Loss: 73 MW x 1.05 x 0.75 = 57.5 MW Peak heat flux: 57.5 MW / (680 m^2 x 1.06) x 1.41 (peaking factor: TBD) = 0.11 MW/m^2 $^{(3)}$: DRG1Table
Surface heat flux due to MARFE or other phenomena	0.3 MW/m$^{2\,(*)}$ Steady-state (in a localized region$^{(*)}$) 100 cycles per year	MARFE (Duration; 10$^{(9)}$sec) in the outboard region has a small probability. When MARFE is detected, the plasma will be shut-down. The heat load due to other phenomena can be steady-state, but the high heat load is localized..	$^{(9)}$: DRG1Table 1.15-2 $^{(*)}$ In the case of steady -state condition, the average heat load in TBM is not more than 0.15 MW/m^2 (the average can be for the TBM overall, or in the toroidal or poloidal direction).
Disruption heat load	TBM is recessed: 0.42$^{(10)}$MJ/m^2 Duration 1$^{(11)}$ ms 180 cycles per year		$^{(10)}$: 270MJ$^{(12)}$ x 0.8$^{(13)}$ / (680 m2 x 1.06) x 1.4 (peaking factor) =0.42 MJ/m^2 $^{(11)}$: DRG1Table 1.16-2 $^{(12)}$: DRG1Table 1.1-1 $^{(13)}$: ~80 % of the thermal energy can go to the FW at maximum. (Experimental data in JET and ASDEX-U)
Disruption heat load during current quench	0.72MJ/m^2 Duration 40 ms 180 cycles per year	All of the internal magnetic energy is assumed to be radiated with peaking factor 1.4.	370MJ$^{(12)}$/ (680 m2 x 1.06) x 1.4 (peaking factor) =0.72 MJ/m^2
Heat load due to ELM and "blob"	Negligible (TBD)	The heat load due to ELM and "blob" will be negligible, considering the distance from the separatrix and recess of the TBM FW locations.	The heat load conditions due to these effects might be revisited based on additional experimental data in the future.
Neutron wall load on TBM FW	0 MW/m^2		

Pulse length	Typical case; 100-200 sec (flat-top time) 1800 sec (repetition time)		
Duty factor	Peak duty factor: Less than 0.25		

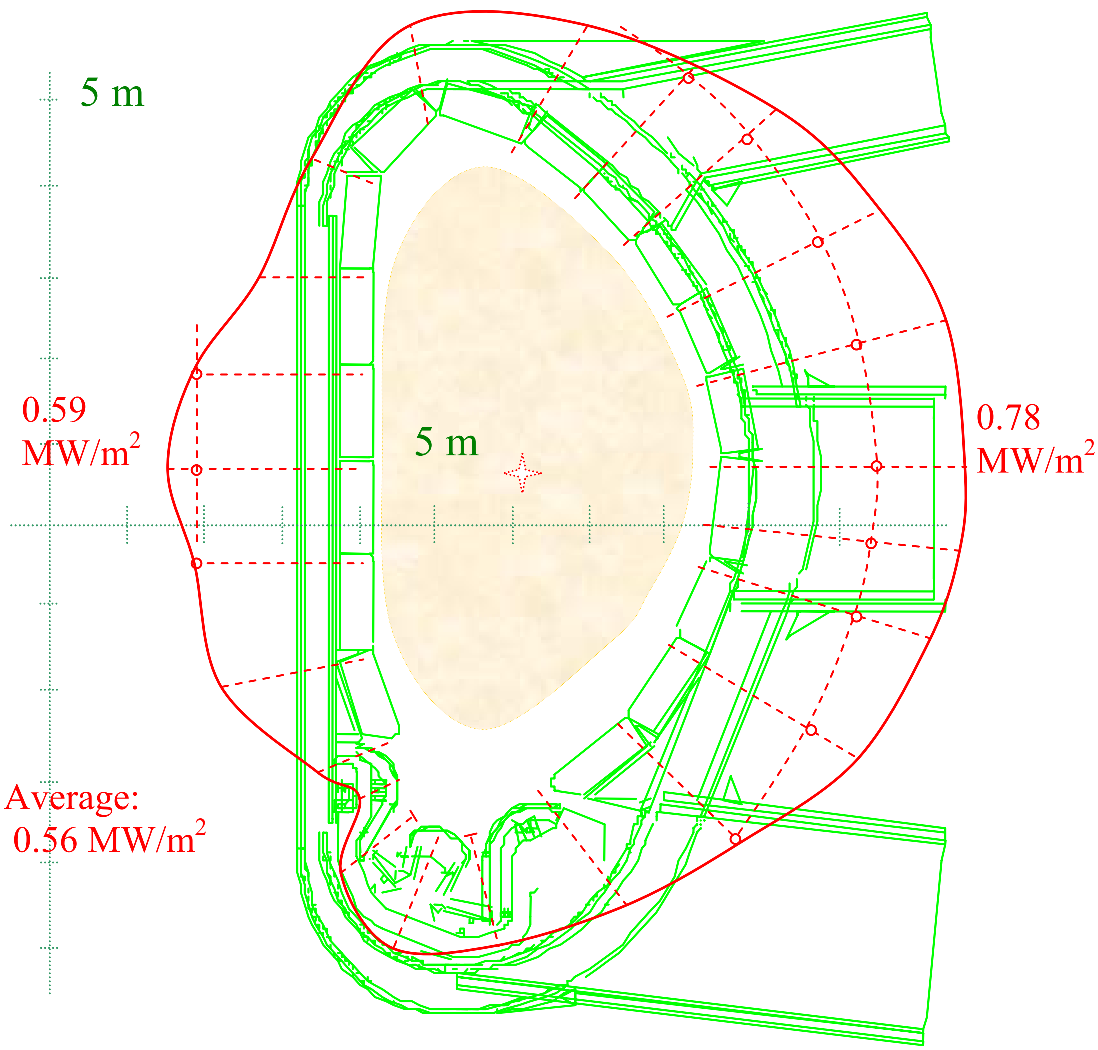


Figure 2.3-2: ITER Poloidal Neutron Wall Loading Distribution. (Fusion Power 500 MW)

Table 2.3-4: Neutron fluence during the first ten years of ITER operation (MWa/m^2)

	1~3	4	5	6	7	8	9	10	total
Equivalent number of nominal pulses	0	1	750	1000	1500	2500	3000	3000	11751
Average neutron fluence at FW	0.0	0.0	0.006	0.008	0.012	0.020	0.024	0.024	0.09

2.4 Test Port Location

ITER has 17 ports at the equatorial level. Fourteen of these ports are standard radial ports and three of them are tangential ports for neutral beam injection. Three of the standard ports are dedicated for test blanket modules (see **Table 2.4-1**). They are Port #16, #18 and #2 as shown in **Figure 2.4-1** and **-2**. In 2001 ITER design, Port #18, #1 and #2 were allocated for test blanket modules, and neighboring 3 ports were chosen considering the configuration of cooling pipes and service lines. However, according to optimization of the diagnostics port layout, Port #16 has been newly allocated for test blanket modules instead of Port #1.

The current port layout (Port #16, #18 and #2) gives an advantage that these ports are not located at the field joint of the vacuum vessel (VV). One of the issues of the ports for TBMs is a possible lower shielding efficiency for the VV. In Port #16, #18 and #2, the reweldability of the VV field joint is not required to be evaluated. However, shielding issues should be assessed carefully in any case, including the nuclear heating in the VV and the magnet. Port A, Port B and Port C are allocated to Port #16, #18 and #2, respectively, as shown in **Figure 2.4-2**.

Each standard equatorial port has an opening 1.748 m wide and 2.2 m high which accommodate the frame 1.708 m wide and 2.160 m high. The face of each TBM is to be recessed by ~50 mm from the first wall of blanket modules to protect modules from interaction with plasma.

Table 2.4-1: Equatorial port allocation for each scenario (PID Chapter 3.5)

Port	Startup	Scenario 1	Scenario 2	Scenario 3	Scenario 4
#1	Diagnostics	Diagnostics	Diagnostics	Diagnostics	Diagnostics
#2	(Test Blanket)	(Test Blanket)	(Test Blanket)	(Test Blanket)	(Test Blanket)
#3 (RH port)	Diagnostics	Diagnostics	Diagnostics	Diagnostics	Diagnostics
#4 (small rad.)	D-NB	D-NB	D-NB	D-NB	D-NB
#4 (tangential)	H-NB	H-NB	H-NB	H-NB	H-NB
#5 (tangential)	H-NB	H-NB	H-NB	H-NB	H-NB
#6 (tangential)		Diagnostics	H-NB	H-NB	H-NB
#7	Closed	Closed	Closed	Closed	Closed
#8 (RH port)	Limiter, Diagnostics	Limiter, Diagnostics	Limiter, Diagnostics	Limiter, Diagnostics	Limiter, Diagnostics
#9	Diagnostics	Diagnostics	Diagnostics	Diagnostics	Diagnostics
#10	Diagnostics	Diagnostics	Diagnostics	Diagnostics	Diagnostics
#11	Diagnostics	IC	Diagnostics	Diagnostics	Diagnostics
#12 (RH port)	Diagnostics	Diagnostics	Diagnostics	Diagnostics	Diagnostics
#13	IC	IC	IC	IC	IC
#14	EC	EC	EC	EC	LH
#15		LH	LH	IC	LH
#16	(Test Blanket)	(Test Blanket)	(Test Blanket)	(Test Blanket)	(Test Blanket)
#17 (RH port)	Limiter, Diagnostics	Limiter, Diagnostics	Limiter, Diagnostics	Limiter, Diagnostics	Limiter, Diagnostics
#18	(Test Blanket)	(Test Blanket)	(Test Blanket)	(Test Blanket)	(Test Blanket)

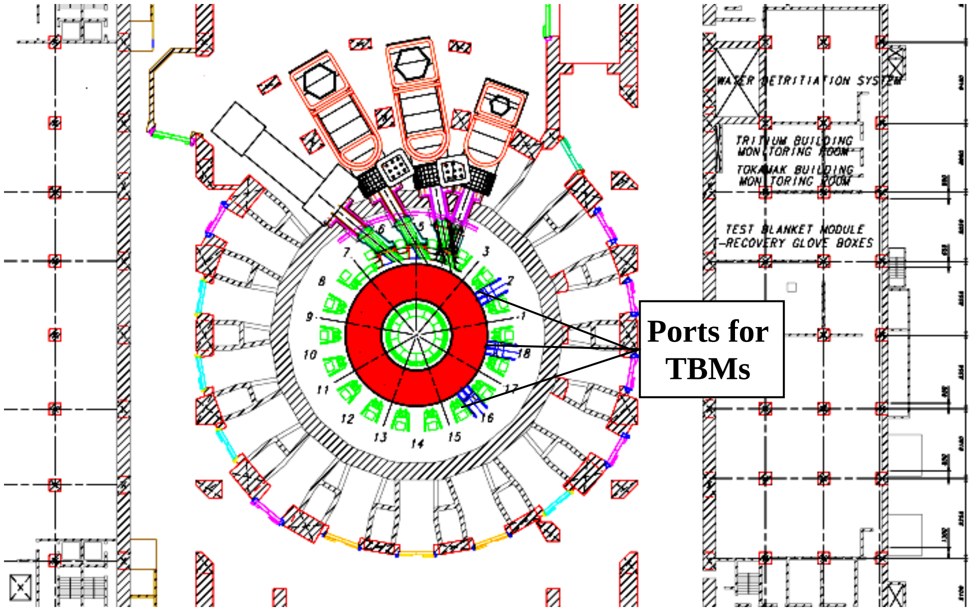


Figure 2.4-1: ITER machine layout and layout of the Ports for TBMs

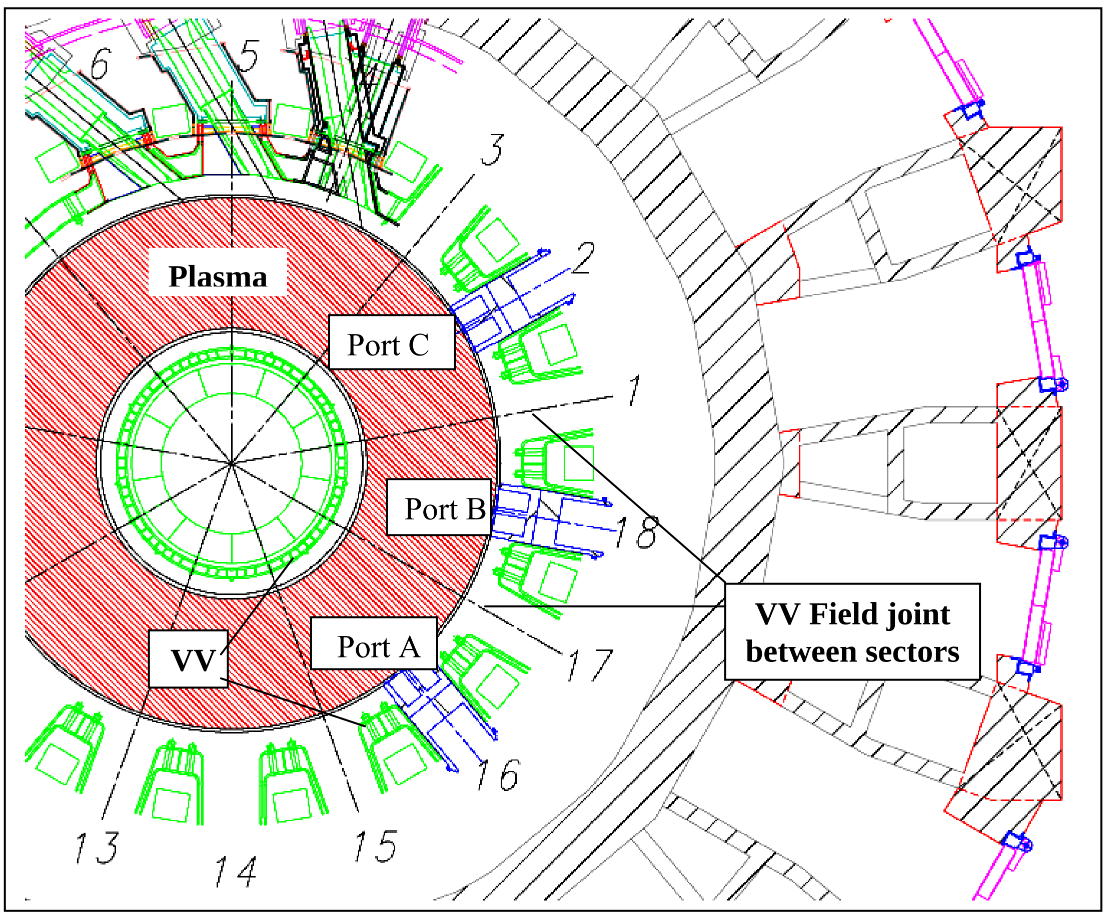


Figure 2.4-2: Plan View of the VV and Locations of TBMs

2.5 Requirements to TBMs

2.5.1 General Requirements

The testing program may be successful only if ITER itself will operate successfully. As a result, the testing of breeding blanket modules must neither interfere with ITER operation, nor decrease ITER reliability or compromise safety of operation or contradict ITER operational plans.
In cases the presence of the TBMs may by itself interfere with a proper plasma confinement (surface sputtering or error field generation), experiments during hydrogen stage must clarify mitigation measures.

2.5.2 Operational and safety requirements

Test modules must not interfere with ITER operational plans or compromise reliability. The TBMs may be changed once per year in accordance with the ITER operational schedule. Some modules may need additional replacements, but would need to be synchronized with requirements of other modules and with machine operation. Target replacement time must be less than one month for the three test ports together.
Integrity of TBMs must not influence the machine availability.
The TBM shall not compromise the safety objectives, principles, requirements and guidelines of ITER. This guidelines and requirements applied to the TBMs are presented in the Safety section of this document (Chapter 8).

2.5.3 Acceptance and Licensing requirements

For licensing, documentation similar to the ITER generic site safety report (GSSR) is required. The GSSR structure is also shortly listed in the Safety section of this document.
To guarantee safety and reliability of operations, the same quality assurance and quality control programs will be applied to TBMs as for ITER itself at all stages of design, production and testing. Today, general requirements have been established for TBMs (operational conditions, compatibility with ITER design and services, compatibility with ITER operation and safety). During the design process frequent and detailed reviews of the TBM design and material specifications, validations of the structural integrity of the designed TBMs, and establishment of requirements to quality control during design and production, will be carried out. Full non-destructive examination and proof tests (pressure tests; leak tightness, etc.) must be performed during factory production. The requirement for quality control during design and production of TBMs and especially also for the out-of-vessel part of the TBM loop, will be as for *SIC (Safety Important Class) components*. Acceptance tests will be done at the ITER site after transportation, and additional testing will be done after installation.

2.5.4 Cooling /baking requirements

The test modules shall be designed to use the ITER heat rejection system (pressurized water) inside the TCWS-vault which has the following parameters: pressure 0.1 MPa, temperature at the inlet of a heat exchanger 35 °C, temperature difference between inlet and outlet < 40°C.

The first wall shall be baked as other blanket modules at the nominal temperature of 240°C.

2.5.5 Mechanical Requirements

The Test Blanket modules inside a port must be contained in a "frame", which provides a standardized interface with the ITER basic structure and provides thermal isolation from the basic machine.
A water-cooled shield must be located behind the TBMs and the frame to assure neutron protection for the vacuum vessel and magnets and to reduce the neutron load at the VV boundary to allow hand-on access outside the VV boundary**.**
The mechanical interface of the frame with the ITER machine will be provided by a flange structure with bolt connections and lip-seal welding joint. The plumbing, which extends through the VV closure plate up to the cryostat boundary, is considered also as a part of the test blanket subsystem (TBS). All the VV plug penetrations will have rigid connection or double bellows. Bellows will be allowed when they are inevitable, but they are used only for the vacuum boundary and not for coolant boundaries. This arrangement enables the whole TBS (TBMs, frame, shield, VV plug and plumbing) to be a self-contained unit that may be installed and removed as a single piece without remote handling operations inside the VV port extension. This assembly must be completely assembled and tested prior to installation. The TBM structural connections shall use remote handling compatible connectors.

2.5.6 Remote Handling Requirements

The test blanket module shall be designed for full remote replacement (see chapter 7 for details).
The weight of a TBM (without weight of coolant and frame) has been considered to be up to 2 t. However, this weight limit in the hot cell will be redefined later after detail design if required. Test Blanket Modules will not be repaired but just refurbished or replaced in the ITER hot cell. The hot cell may be used to replace irradiated test modules but it is not designed for post-irradiation studies of the test modules.
The weight of the integrated structure consisting of TBM/Frame/Shield plug to be carried in a transfer cask and installed on the machine must be limited to 40 t. Welded joints within the plasma chamber and the vacuum vessel extensions shall be done, repaired, and leak tested remotely. Remote coolant draining shall be possible. Assembly and maintenance tools shall be provided for the structural attachment of the test blanket article: Be capable of joining, cutting, and leak testing the 0.5 to 2 cm diameter pipes required for the tritium purge gas lines and instrumentation. Gripping points must be provided on all replaceable components or assemblies, capable of supporting their full weight over the full range of motion required for installation and removal.

2.5.7 Assembly Requirements

To help protecting the first wall of the TBM, it can be recessed below the adjacent shielding blanket first wall and, thus, will not have an explicit requirement for alignment to the magnetic surface.

2.5.8 Electrical Requirements

The test module shall be electrically connected to the frame with an electrical connector, and through the coolant and purge gas pipes.

2.5.9 Layout Requirements

Structural and leak tightness welds shall be removed as far away as possible from high neutron flux locations. Special attention shall be given to gaps. Radiation streaming shall be minimized by design.

2.5.10 Instrumentation & Control Requirements

The TBM must provide independent instrumentation with data connection through a local controller to the CODAC system. Sensors should monitor the system temperatures, flow rates, pressure, and stresses/deflections to insure that they are within prescribed values. The following minimum set of parameters is mandatory.

TBM Instrumentation required for operation:
a. Inlet and outlet TBM coolant temperature,
b. TBM coolant flow rate,
c. Temperatures inside the test module,
d. Inlet and outlet purge gas pressure and temperature,
e. Inlet and outlet purge gas flow rate and tritium concentration.

TBM instrumentation to signal acceptability to operate or to shut down:
a. Strain gauges,
b. Temperatures at critical points inside the test module,
c. Pressure and moisture sensors in each purge gas line.

2.6 Frame design

2.6.1 Purpose

TBMs are accommodated in special port plugs, which are dedicated to TBMs. TBMs are supported onto a structure called “frame” (**Figure 2.6-1**). The frame provides a standardized interface with the VV/port structure and provides thermal isolation from the basic machine.

The frame consists of first wall, box and backside shield (**Figure 2.6-2**). All these components are cooled by water to remove heat loads and provide enough nuclear shielding for the vacuum vessel/ports and magnets. The shielding allows hand-on access behind the backside shield in the port cell.

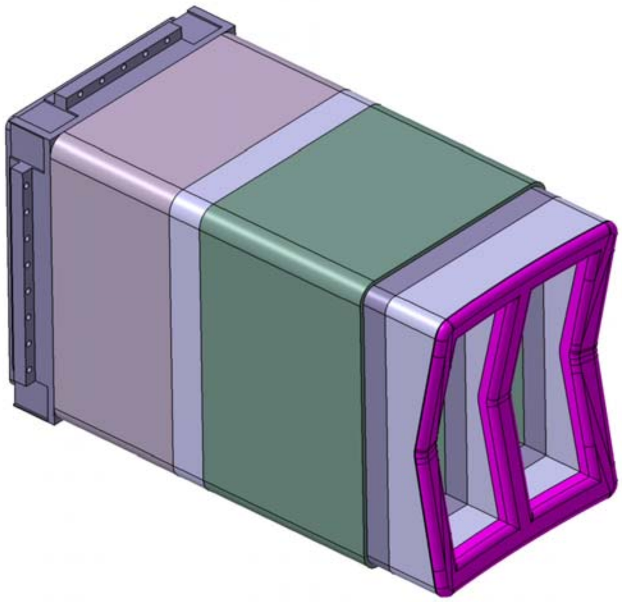


Figure 2.6-1: Overall view of the Test Blanket Frame (Frame for vertical TBMs)

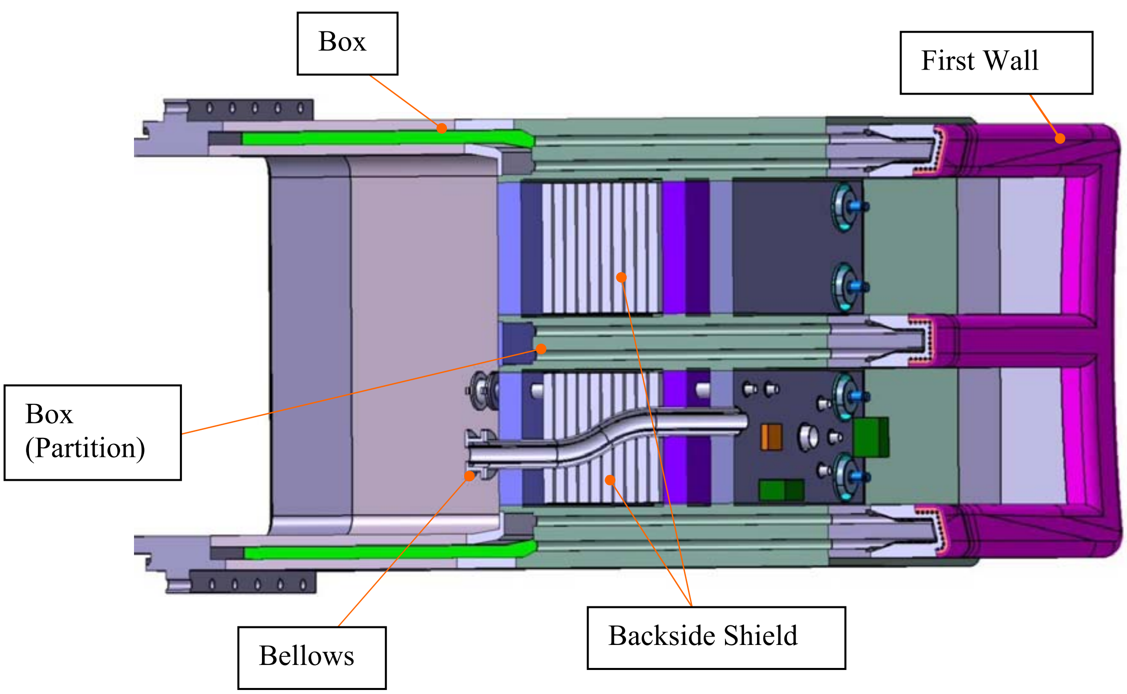


Figure 2.6-2: Horizontal cross section of the Test Blanket Frame (Frame for vertical TBMs)

2.6.2 Frame structure

The first wall (FW) covers the plasma facing surface of the box. The FW consists of Be armor, copper heat sink and stainless steel body as shield blanket (**Figure 2.6-3**). The box is the main body of the frame. The box is separated by an intermediate plate called "partition". Each TBM is accommodated inside of it separately. The TBM is supported from the box by flexible supports and keys (**Figure 2.6-4**). This support concept is similar to the support of the shield blanket, which has been demonstrated through the R&D during EDA. An electrical connection is equipped at the centre area of the TBM to reduce electromagnetic force on the pipes.

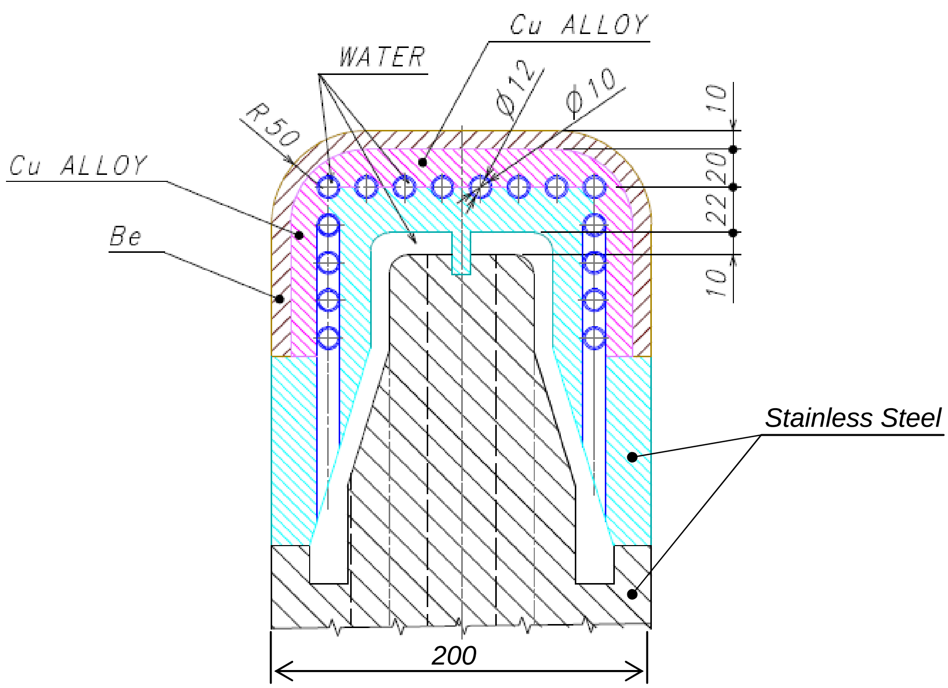


Figure 2.6-3: First Wall of the frame
(Horizontal cross section)

There are two types of box, so-called "horizontal type" and "vertical type", which are used for "horizontal module" and "vertical module" TBMs, respectively. This report mainly describes about the vertical type of frame but the same design concept can also be adapted to the horizontal type. The box with the FW guards the TBMs from plasma. It also contributes to shielding. The thickness of the box is 200 mm to provide adequate shielding to prevent excessive nuclear heat in the vacuum vessel and to limit the amount of TBM structural material mostly made of ferromagnetic steel that aggravates the toroidal ripple.

Considering flexibility of pipe layout for the TBM, alternative design option of the frame, which allows removing backside shield, has been developed, instead of the reference design, which has permanent backside shield. This option is called "frame with removable backside shield" (**Figure 2.6-5**). In this design, the backside shield can be removed from the box together with support structures (keys, flexible supports and electrical connection) and pipes. This design makes different pipe configurations possible according to a design change of the TBM at the future stage without replacing the whole box structure, which is massive and equipped with FW made by HIP. The backside shield is cantilevered and supported by the box with bolts.

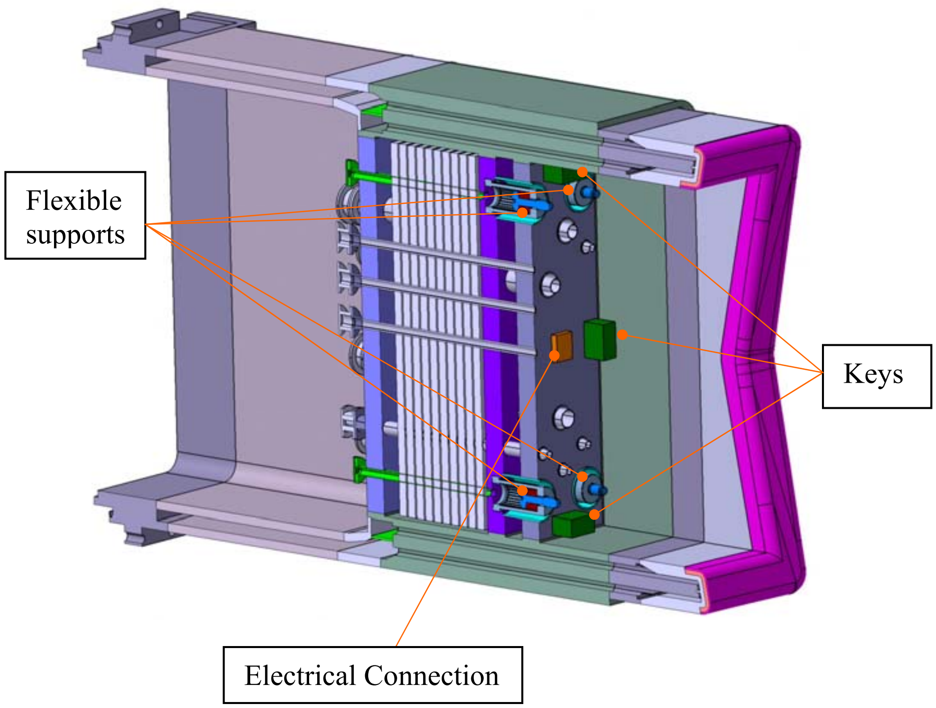


Figure 2.6-4: Support structures/interfaces for TBM

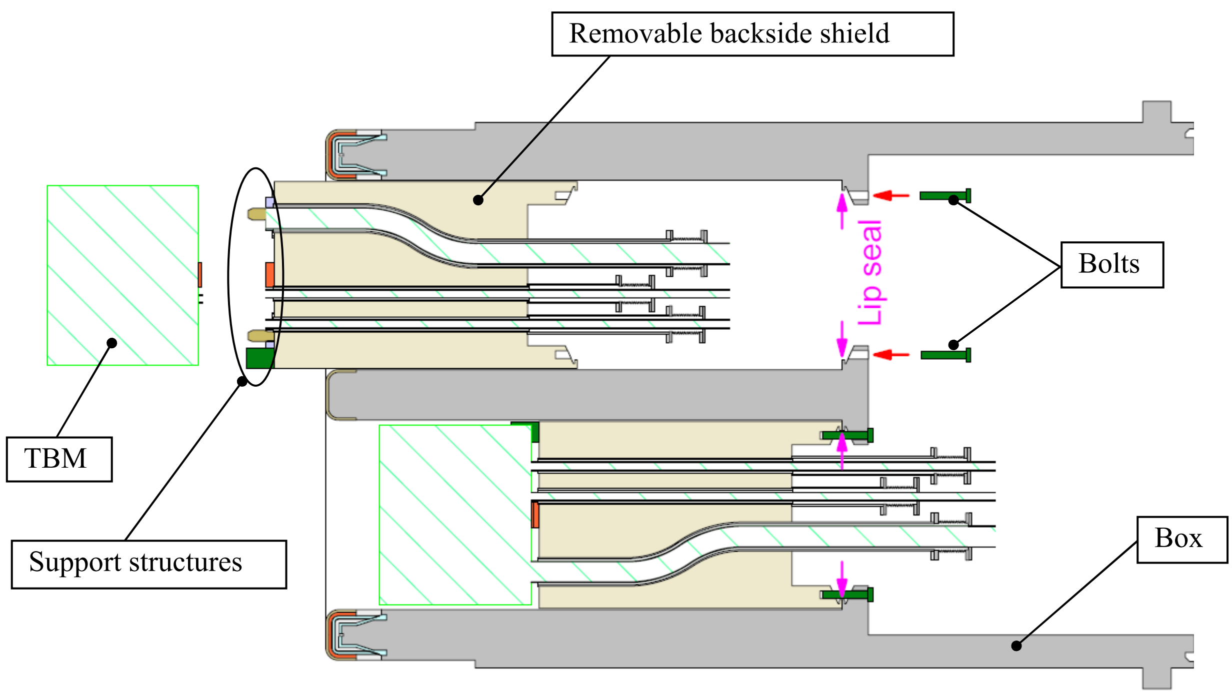


Figure 2.6-5: Frame with removable backside shield

Table 2.6-1 compares advantages and disadvantages of the frame with the permanent backside shield and the frame with the removable backside shield. The frame with the removable backside shield is useful, but the available backside space is getting smaller and there are some drawbacks to be assessed.

Table 2.6-1: Comparison between the frame with the permanent backside shield and the frame with the removable backside shield

	Permanent backside shield (Reference option)	Removable backside shield (Alternate option)
TBM replacement in the Hot cell	Replace TBM	(1) Replace (TBM + backside shield) together (2) Replace TBM Two steps are required for the TBM replacement. Heavy component is to be replaced in the hot cell. Hot cell storage needs to be enlarged.
Pipe cutting/welding	Bore tool method (access from the backside)	The method can be chosen among the options. - Bore tool method (access from the backside) - Orbital tool method (side access) or combination [depending on the replacement scenario]
Available backside area for TBM attachments (pipes etc.)		- Available backside space is smaller (due to flange and lip-seal structures).
Flexibility in different configurations of TBM attachments (pipes etc.)		Different layouts of TBM pipes and attachments can be accommodated by using different backside shields.
Cost		- Slightly higher cost due to additional structures - More cost with different backside shields

2.6.3 Pipes layout

A typical configuration of penetrations (pipes, instrumentations, etc) has been studied taking into account the available space for pipes of TBMs. A half-port TBM, which occupies half of the frame, has been considered. The typical pipe configuration has been studied based on the dimensions and number of penetrations for the HCPB TBM of the EU. Although the HCPB TBM of the EU is a horizontal module, the frame design for vertical TBMs is used for this study because it is expected that the space availability of the penetrations is more severe. The typical pipe configuration is shown in **Table 2.6-2**. It includes 3 large pipes with 85.4 mm I.D /101.6 mm O.D., 1 pipe with 57.3 mm I.D /76.3 mm O.D. and 7 small pipes with 30.0 mm I.D /35.0 mm O.D. One of the possible layouts for the vertical type of the frame is shown in **Figure 2.6-6**. Pipe layouts of the frames with permanent and removable backside shield are shown in the figure. This typical configuration can be achieved without any difficulties in the permanent backside shield. However, in the removable backside shield, it is very marginal whether we can achieve this configuration.

Table 2.6-2: Typical configuration (based on HCPB, half-port TBM)

	Pipe diameter (ID/OD)	**Temperature/Pressure**	**Number of pipes (penetrations)**
He cooling (inlet)	85.4 / 101.6 mm	300 °C / 8MPa	1
He cooling (outlet)	85.4 / 101.6 mm	500 °C / 8MPa	1
He cooling (bypass)	85.4 / 101.6 mm	400 °C / 8MPa	1
Purge gas	30/35 mm	RT / 0.1 MPa	2
BU	30/35 mm	400 °C (inlet), 500 °C (outlet) / 8MPa	4
Instrumentation	57.3 / 76.3 mm	TBD	1
Others	30/35 mm	TBD	1

Table 2.6-3(a-k) shows proposed pipe layouts by each PT together with comments by IT about compatibility with available space. Most of proposed penetrations of half-port modules seem to be accommodated in the backside shields. However, more details need to be studied. For quarter-port modules which are smaller than half-port module, smaller diameters of the penetrations and a limited total number (up to ~6 penetrations) will be required. It might be difficult to allocate completely independent lines to each quarter-port module or sub-module. It is also requested to use a common “base-plate” to support quarter-port modules. This base-plate will be supported on the backside shield in the same way as for a half-port module.

Engineering and material issues associated with 700/650 °C of outlet temperature proposed by CHN and US for the Dual-Coolant Lithium-Lead TBMs are to be studied.

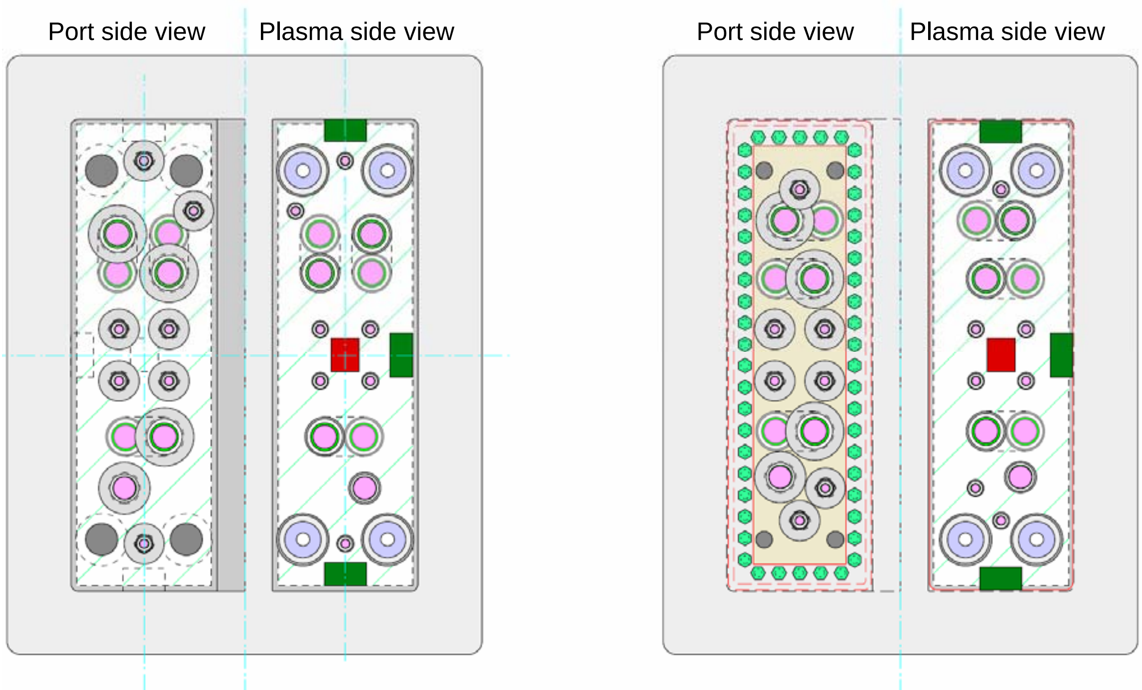


Figure 2.6-6: Pipe layout for the typical pipe configuration shown in Table 2.6-3

Table 2.6-3(a): Proposed pipe configuration for the EU He-cooled solid breeder (HCPB) TBM

Port: A Party: EU Layout: Horizontal half-port module Type: He-cooled pebble bed			
	Pipe diameter (ID/OD)	**Temperature/Pressure**	**Number of pipes (penetrations)**
He cooling (inlet)	80/91.2 mm	300 °C / 8MPa	1
He cooling (outlet)	60/71.2 mm	500 °C / 8MPa	1
He cooling (bypass)	60/71.2 mm	370 °C / 8MPa	1
Purge gas	30/38 mm	500 °C / 0.2 MPa	2
BU	30/38 mm	300 °C / 8MPa 500 °C / 8MPa	4
Instrumentation	30/38 mm	TBD	1

Table 2.6-3(b): Proposed pipe configuration for the JA He-cooled solid breeder TBM

Port: A Party: JA Layout: Horizontal half-port module Type: He-cooled solid breeder			
	Pipe diameter (ID/OD)	**Temperature/Pressure**	**Number of pipes (penetrations)**
He cooling (inlet)	85.4 / 101.6 mm	300 °C / 8 MPa	3 (or 1, TBD)
He cooling (outlet)	85.4 / 101.6 mm	500 °C / 8 MPa	3 (or 1, TBD)
He cooling (bypass)	49.5 / 60.5 mm	472 °C / 8 MPa	3 (or 1, TBD)
Purge gas (inlet)	18.4 / 25.4 mm	RT / 0.1 MPa	3 (or 1, TBD)
Purge gas (outlet)	18.4 / 25.4 mm	450 °C / 0.1 MPa	3 (or 1, TBD)
Instrumentation (cable conduit)	50.0 / 60.5 mm		6 (or 2, TBD)
Instrumentation (micro- fission chamber)	7.8 / 13.8 mm		6 (or 2, TBD)

Comment from IT: 3 times more penetrations seem not feasible

Table 2.6-3(c): Proposed pipe configuration for the JA Water-cooled solid breeder TBM

Port: B Party: JA Layout: Vertical half-port module Type: Water-cooled solid breeder			
	Pipe diameter (ID/OD)	**Temperature/Pressure**	**Number of pipes (penetrations)**
Water cooling (inlet)	76.2 / 101.6 mm	280 °C / 15.5 MPa	1
Water cooling (outlet)	76.2 / 101.6 mm	325 °C / 15.5 MPa	1
Purge gas (inlet)	18.4 / 25.4 mm	RT / 0.1 MPa	1
Purge gas (outlet)	18.4 / 25.4 mm	450 °C / 0.1 MPa	1
Instrumentation (cable conduit)	57.3 / 76.3 mm		2
Instrumentation (micro-fission chamber)	7.8 / 13.8 mm		2

Table 2.6-3(d): Proposed pipe configuration for the EU He cooled liquid lithium lead (HCLL) TBM

Port: B Party: EU Layout: Vertical half-port module Type: He cooled liquid lithium lead			
	Pipe diameter, mm (ID/OD)	**Temperature/Pressure**	**Number of pipes (penetrations)**
He cooling (inlet)	60/70 mm	300 °C / 8 MPa	1
He cooling (outlet)	70/80 mm	500 °C / 8 MPa	1
He cooling (bypass)	60/70 mm	400 °C / 8MPa	1
Lithium lead (inlet)	87/99 mm	300 °C / up to 1 MPa	1
Lithium lead (outlet)	87/99 mm	450 °C / up to 1 MPa	1
Instrumentation	30/35 mm	TBD	1
Other (mini-tubes…)	30/35 mm	TBD	1

Table 2.6-3(e): Proposed pipe configuration for the CHN Dual-Functional Lithium Lead (DFLL) TBM

<table>
<tr><td colspan="4">Port: B Party: CHN Layout: Vertical half-port module
Type: Dual-Cooling Lithium Lead</td></tr>
<tr><td></td><td>Pipe diameter (ID/OD)</td><td>Temperature/Pressure</td><td>Number of pipes (penetrations)</td></tr>
<tr><td rowspan="2">He cooling (concentric)</td><td>outer pipe
110/120 mm</td><td>300 °C / 8MPa</td><td rowspan="2">1</td></tr>
<tr><td>inner pipe
80/90 mm</td><td>410 °C / 8MPa</td></tr>
<tr><td rowspan="2">LiPb loop pipe (concentric)</td><td>outer pipe
110/120 mm</td><td>480 °C / 1MPa</td><td rowspan="2">1</td></tr>
<tr><td>inner pipe
80/90 mm</td><td>~700 °C / 1MPa</td></tr>
<tr><td>Draining breeder in emergency</td><td>90/100 mm</td><td>Max.~700 °C / 1MPa</td><td>1</td></tr>
<tr><td>Instrumentation</td><td>70 / 80 mm</td><td>TBD</td><td>1</td></tr>
<tr><td>Attachment</td><td>60 / 70 mm</td><td>TBD</td><td>4</td></tr>
<tr><td>Others</td><td>30 / 35 mm</td><td>TBD</td><td>1</td></tr>
</table>

Comment from IT: Layout of the penetrations is tight. The concentric pipe for LiPb coolant can be smaller.

Table 2.6-3(f): Proposed pipe configuration for the US Dual-Coolant Lithium-Lead (DCLL) TBM

<table>
<tr><td colspan="4">Port: TBD Party: US Layout: Vertical half-port module
Type: DCLL</td></tr>
<tr><td></td><td>Pipe diameter (ID/OD)</td><td>Temperature/Pressure</td><td>Number of pipes (penetrations)</td></tr>
<tr><td>Concentric PbLi1</td><td>71/77 mm,
46/52 mm</td><td>400 °C / 650 °C / 2MPa</td><td>1 concentric pipes</td></tr>
<tr><td>He cooling (inlet)2</td><td>76/86 mm</td><td>380 °C / 8MPa</td><td>1</td></tr>
<tr><td>He cooling (outlet)2</td><td>76/86 mm</td><td>460 °C / 8MPa</td><td>1</td></tr>
<tr><td>PbLi drain pipe</td><td>77 mm (OD)</td><td>650 °C / 2 MPa</td><td>1</td></tr>
<tr><td>Pressure relief line</td><td>50 mm (OD)</td><td>TBD</td><td>1</td></tr>
<tr><td>Power/control cable connection</td><td>75 mm</td><td>TBD</td><td>1</td></tr>
<tr><td>Instrumentation connection</td><td>50 mm</td><td>TBD</td><td>1</td></tr>
</table>

Table 2.6-3(g): Proposed pipe configuration for the RF Li self-cooled/V TBM

Port: C Party: RF Layout: Vertical half-port module Type: Li self-cooled/V			
	Pipe diameter (ID/OD)	**Temperature/Pressure**	**Number of pipes (penetrations)**
Li cooling (inlet)	30/33 mm	250-450 °C / 0.5 MPa	2
Li cooling (outlet)	30/33 mm	350-550 °C / 0.3 MPa	2
H2O (inlet)	45/50 mm	100 °C / 1 MPa	1
H2O (outlet)	45/50 mm	150 °C / 1 MPa	1
Ar (gas drain during Li filling up)	30/35 mm	200 °C / 0.1 MPa	1
Instrumentation	~80 mm (TBD)		2 (TBD)

Table 2.6-3(h): Proposed pipe configuration for the RF He-cooled solid breeder (HCSB) TBM

Port: C Party: RF Layout: Vertical quarter-port module Type: He-cooled solid breeder			
	Pipe diameter (ID/OD)	**Temperature/Pressure**	**Number of pipes (penetrations)**
He cooling inlet (sub-module casing)	74/80 mm	300 °C / 8 MPa	1
He cooling outlet (sub-module casing)	80/86 mm	355 °C / 8 MPa	1
He cooling inlet (breeding zone)	55/60 mm	300 °C / 8 MPa	1
He cooling outlet (breeding zone)	62/68 mm	500 °C / 8 MPa	1
Purge gas	30/35 mm	RT / 0.1 MPa	2
Instrumentation	30/35 mm	TBD	2
Tritium monitoring channel (CO2)	20/22 mm	50-150 °C / 0.5 MPa	2

Comment from IT: Too many pipes/penetrations for each quarter-port module.

Table 2.6-3(i): Proposed pipe configuration for the CHN He-cooled solid breeder (HCSB) TBM

Port: C Party: CHN Layout: Vertical quarter-port module Type: He-cooled solid breeder			
	Pipe diameter (ID/OD)	**Temperature/Pressure**	**Number of pipes (penetrations)**
He cooling (inlet)	100/108 mm	300 °C / 8MPa	1
He cooling (outlet)	100/108 mm	500 °C / 8MPa	1
Purge gas	30/35 mm	RT	2
BC He cooling	20/24 mm	400 °C (inlet),500°C (outlet) /8MPa	2
BC Purge gas	10/14 mm	RT / 0.12MPa	2
Instrumentation	30/35	TBD	1
Others	30/35mm	TBD	1

Comment from IT: Too many pipes/penetrations for each quarter-port module.

Table 2.6-3(j): Proposed pipe configuration for the KOR He-cooled solid breeder TBM

Port: A Party: KOR Layout: Horizontal half-port module Type: He-cooled solid breeder			
	Pipe diameter (ID/OD)	**Temperature/Pressure**	**Number of pipes (penetrations)**
He cooling (inlet)	80/85 mm	300 °C / 8 MPa	1
He cooling (outlet)	60/65 mm	500 °C / 8 MPa	1
Purge gas	30/35 mm	RT / 0.1 MPa	2
Instrumentation	30/35 mm	TBD	1
others	30/35 mm	TBD	1

Table 2.6-3(k): Proposed pipe configuration for the KOR He-cooled Molten lithium breeder TBM

Port: C Party: KOR Layout: Vertical half-port module Type: He-cooled molten lithium breeder			
	Pipe diameter (ID/OD)	**Temperature/Pressure**	**Number of pipes (penetrations)**
He cooling (inlet)	80/85 mm	350 °C / 8 MPa	1 (concentric)
He cooling (outlet)	60/65 mm	410 °C / 8 MPa	
Li flow (in, out)	80/85, 60/65 mm	TBD	1 (concentric)
Instrumentation	30/35 mm	TBD	1
others	30/35 mm	TBD	1

2.6.4 Assembly

TBMs are assembled with the frames outside of the ITER tokamak and installed to the equatorial ports of the vacuum vessel on the site. The frames are cantilevered and supported by the vacuum vessel port flanges, which is the general method to support in-port components, as other port plugs. Interface structure with the port flange is equipped at the end of the box (**Figure 2.6-7**). The TBMs can be disassembled from the frame and be replaced in the hot cell with remote handling. To take care of possible assembly displacements, an appropriate gap will be provided between the TBM and the box. More detail study about assembly is needed. As pointed in **Table 2.6-1**, the assembly scheme of the frame with removable backside shield is also to be studied more.

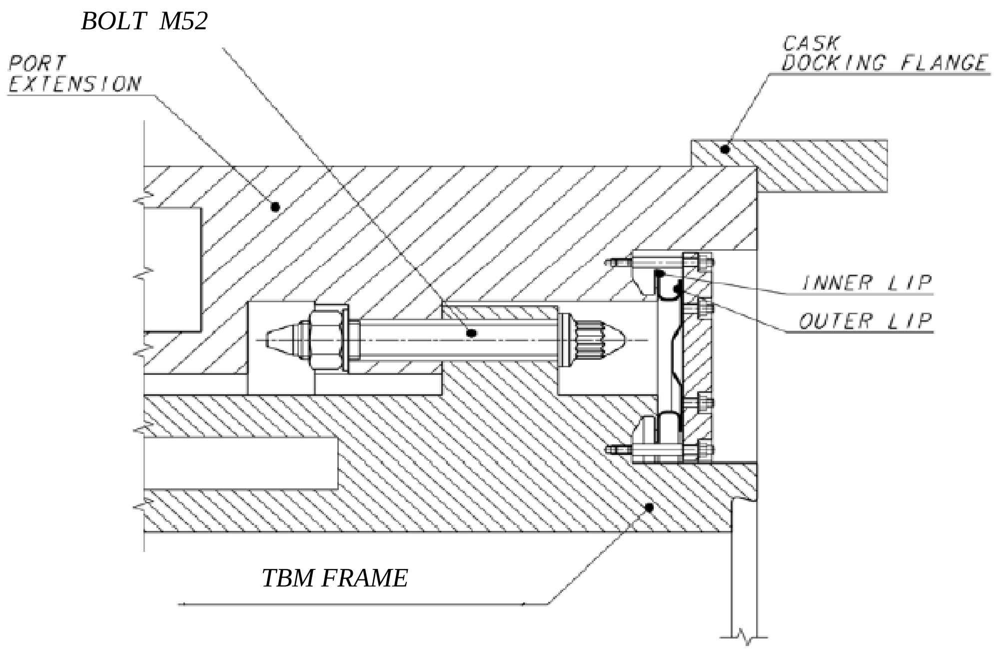


Figure 2.6-7: Frame – Port Flange Joint with lip seal

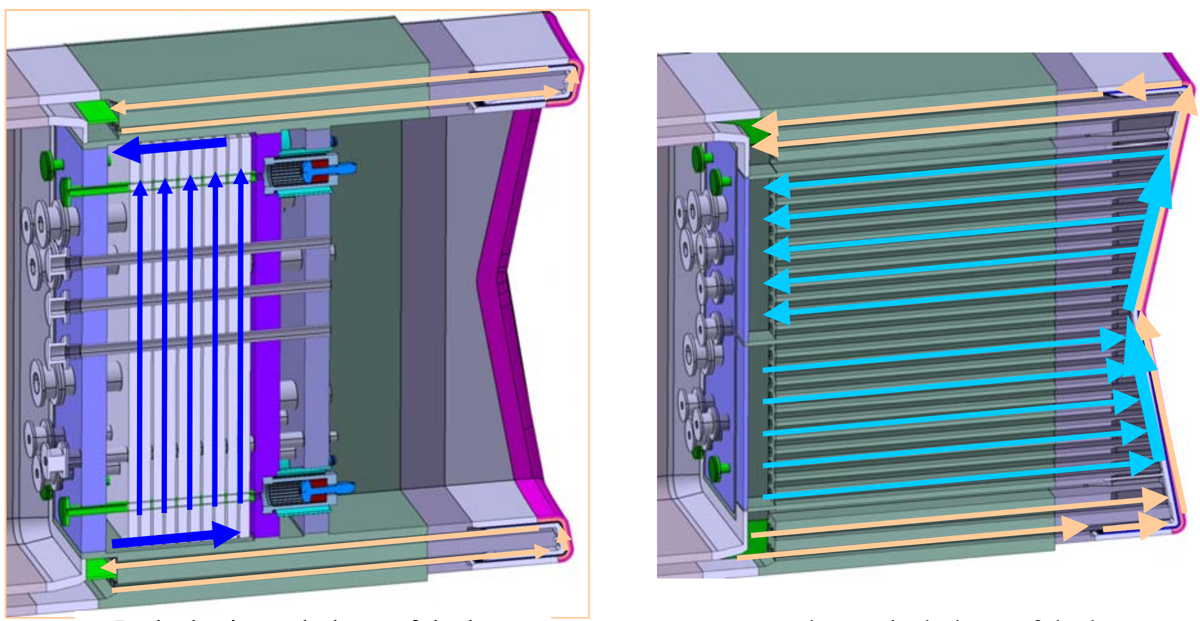
2.6.5 Vacuum boundary

Vacuum boundary is made with lip seal between the equatorial port and the frame as the other port plugs (**Figure 2.6-7**). There are several penetrations for pipes, which come from the TBMs. The pipes, whose temperatures are much higher than the frame, cannot be connected to the frame directly. This means that gaps exist between the pipes and the frame. Bellows are used to seal the gaps. For removable backside shield type frame, additional vacuum boundary exists between the box and removable backside shield. This boundary is also made with lip seal (**Figure 2.6-5**). The detail design of the lip seal for the removable backside shield is to be carried out taking into account assembly study.

2.6.6 Cooling structure

Cooling water in the frame is supplied from the blanket cooling system. The inlet temperature to the frame is 100 °C during the normal operation and the nominal temperature is 240 °C (maximum temperature 250 °C) during the baking operation. Coolant line for a frame has one inlet pipe and one outlet pipe. But the inlet line branches to three regions of the frame. The cooling channels to cool the FW, the box and the backside shield region are independent from each other to provide sufficient flow rates for the each region (**Figure 2.6-8**).

The coolants for the FW region feed into the bottom horizontal plate of the frame in radial direction horizontally. After the coolants reach the front (plasma side) end of the frame, they flow up in the FW region. Then, they flow out through the top horizontal plate of the frame.
The coolants for vertical plates of the box region feed into the lower half of the frame in radial direction horizontally. After the coolants reach the front end of the box, they flow up in the front header region, which is just behind the FW of the box and then they flow out through the upper half of the frame.



In the horizontal plates of the box and in the backside shield

In the vertical plates of the box

First wall cooling water

Box cooling water

Backside shield cooling water

Figure 2.6-8: Cooling water concept for the frame

2.6.7 Frame provision

There is possibility to prepare two frames for each port considering the time limitation for the replacement of TBMs in the hot cell. **Table 2.6-4** compares advantages and disadvantages of “1 Frame / port” and “2 Frames / port”. Generally speaking, “2 Frames / port” will be useful from operational flexibility point of view, but the storage space for frames in the hot cell should be checked.

Table 2.6-4: Advantages and disadvantages of 1 Frame / port and 2 Frames / port.

	1 Frame / port	**2 Frames / port**
TBM replacement in the Hot cell	- On-line replacement of TBM(s) will be required. - Replacement time should be estimated and its acceptability should be checked.	- Replacement of TBM(s) can be done by the next maintenance period. - The required maintenance time can be shorter.
Storage space for Frames in the Hot cell	No additional storage spaces for Frames are required in the Hot cell.	Additional storage spaces for 3 Frames are required in the Hot cell.
Fabrication cost of Frames	No additional fabrication costs of Frames are required.	Additional fabrication costs of 3 Frames are required.
Operational flexibility of TBMs and attachments		- The operational flexibility is higher because an additional frame can be used when a frame has a problem. - Slightly different frames can be considered to accommodate different attachments/pipes.

2.7` Port cells

2.7.1 Overall dimensions

The area outside the bioshield in front of each port is enveloped by a compartment (port cell) that provides space available for a movable container (cask) with interfacing equipment. The container would have to be temporarily removed in case the TBM had to be replaced. The vertical shafts connect these ports to the the Vault area.

The port cells are about 8 m long from the bioshield to the port cell door and have a minimum width of 4 m, measured at the inner corner of the vertical shafts. Additional space of trapezoidal shape is available between the bioshield and the vertical shafts at the right side of port cells 2 and 16 (looking towards the machine center) and at both side of port cell 18 (see **Figures 2.7-1 to 2.7-3**).
The nominal height of the port cell is 4.88 m.

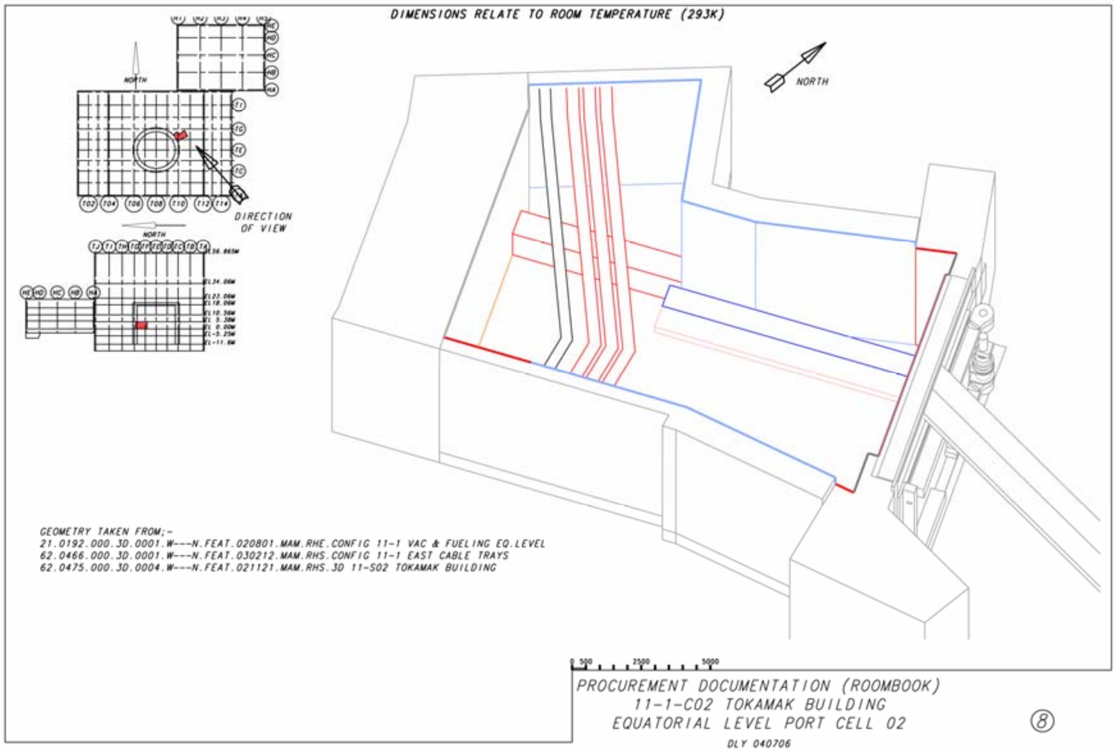


Figure 2.7-1: Isometric view of the Equatorial port cell #2

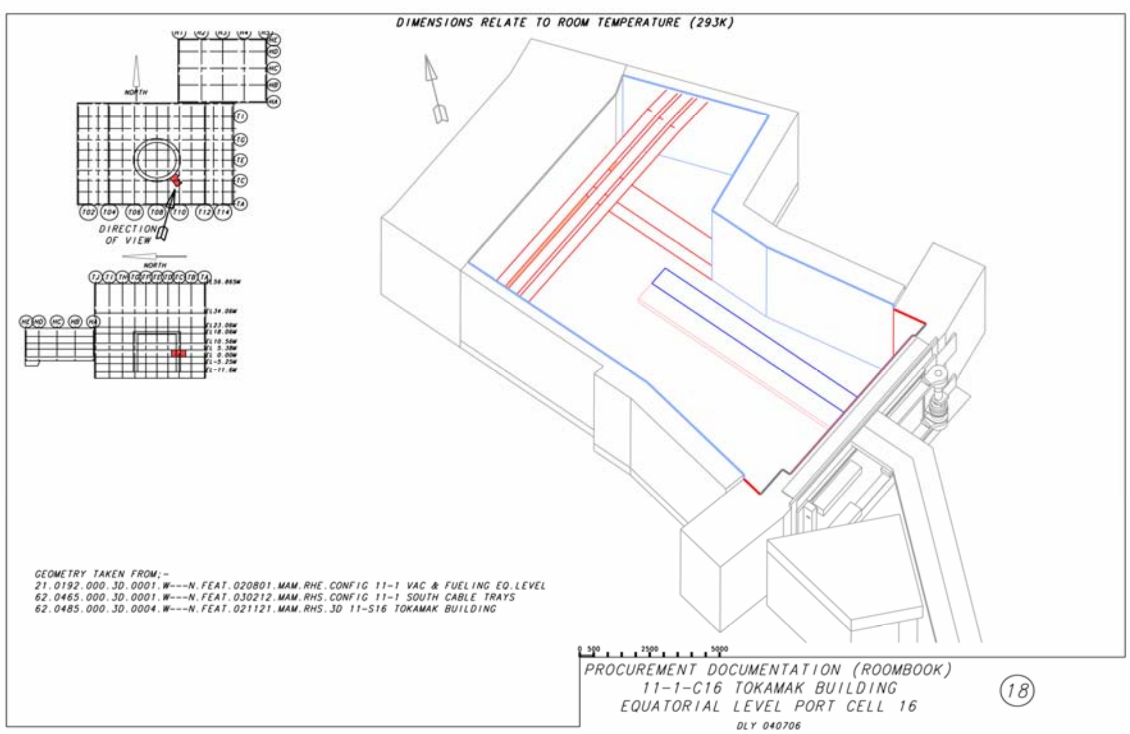


Figure 2.7-2: Isometric view of the Equatorial port cell #16

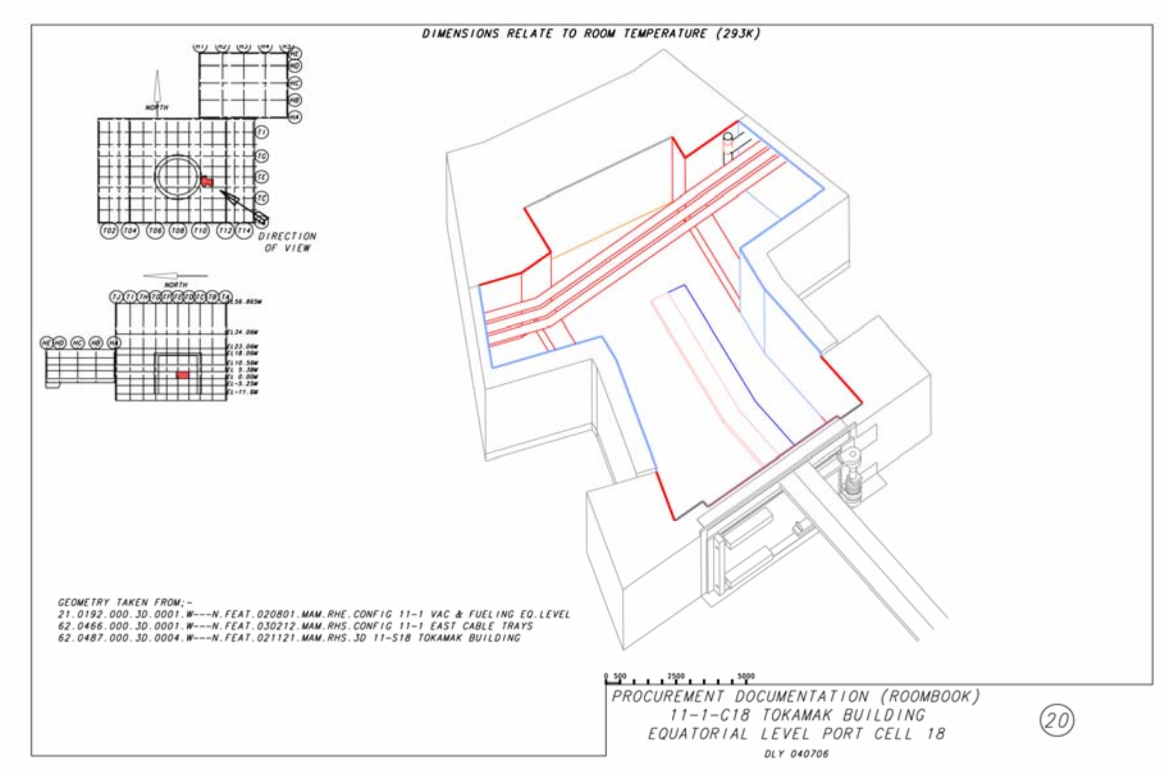


Figure 2.7-3: Isometric view of the Equatorial port cell #18

2.7.2 Classification and functional description

For radiation hazards the port cells are classified in the ITER Plant Design Specification (PDS) as Zone B (Supervised area), which means limited access for Non-Radiation Workers and Unlimited access for Radiation Workers during the ITER scheduled maintenance periods. The total dose rate calculated 2 weeks after shut-down in limited to 10 µSv/h. The access to the port cells is governed by the ITER site access control system and is normally possible only during maintenance periods.

The port cells are part of the second confinement barrier of ITER. This is provided by the walls of the port cells, by floor and ceiling and the shielding doors at the rear end of the port cell. The port cell is designed to resist up to a pressure of 0.16 MPa, i.e. 0.06 MPa overpressure. In case of break of high temperature water pipe, the steam is released in the vault area through relief panels that open to the pipe shaft.
The Port cell Atmosphere Detritiation System controls that the pressure in the port cell is kept at –100 Pa relatively to the pressure in the Gallery to prevent uncontrolled flow of contaminants from the port cell to the galleries.
The port cells have no HVAC, but use one or two cooler units (composed of chilled water cooled heat exchanger with fan) to extract heat and to limit the maximum air temperature below 35˚C. To limit the capacity of this cooler, the hot faces of port plugs must be thermally insulated to be equivalent to 100mm thick layer of rock wool and the total heat release from the components and pipes inside the port cell to be ≤5 kW. This is different from the previous port plug designs, as it has been decided that the cryostat closure plates are eliminated and hence the port plug face is now exposed to the atmosphere inside the port cell.

To limit the capacity of the vent detritiation system and the production of (tritiated) water by this system, air leakage into the port cell has to be minimized. Pneumatic valves, in particular control valves, must have the released air collected into a common header for release into the external atmosphere.

It should be noted that these areas are subjected to a relatively strong magnetic field caused by the superconducting magnets. The peak values are estimated to be 2000 Gauss near the bioshield, and therefore, the use of electric, electronic and magnetic equipments is strictly limited inside the port cells. In particular it will not normally be possible to deploy power distribution panels, control and instrumentation cubicles and similar inside the port cells. Solenoid valves must be shielded down to around 50 Gauss, which requires exorbitant amount and volume of shielding materials.

The fire load in each port cell should be minimized as much as practical. In this context the use of organic coolant in secondary heat transfer loops will need to be scrutinized in detail.

In each port cell there are routed general utilities including:

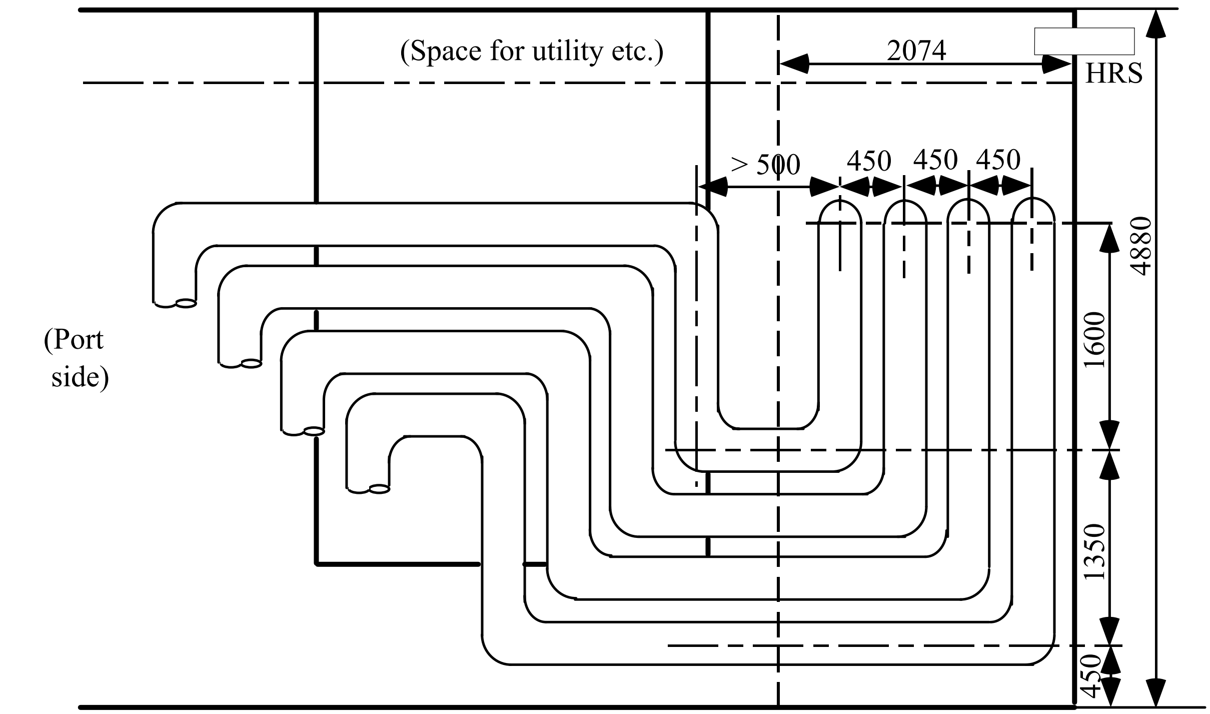
- Low voltage electric power (400V if required) 230V,
- Instrument air (probably 6bar), routed outside the cells,
- “House vacuum” (for rough vacuum),
- Nitrogen or special gases,
- Breathing air (routed outside the cells),
- Collector for released instrument air.

The systems to be placed near the ports are primarily the tritium interface and the components of the primary heat transfer loops. In case the space inside the container is insufficient for all these components, the heat transfer components may be placed in the Tokamak Cooling Water System (TCWS) vault. In this case, the cooling pipes of TBMs would be routed from the port cell through the vertical shafts to the heat transfer components in the TWCS vault.

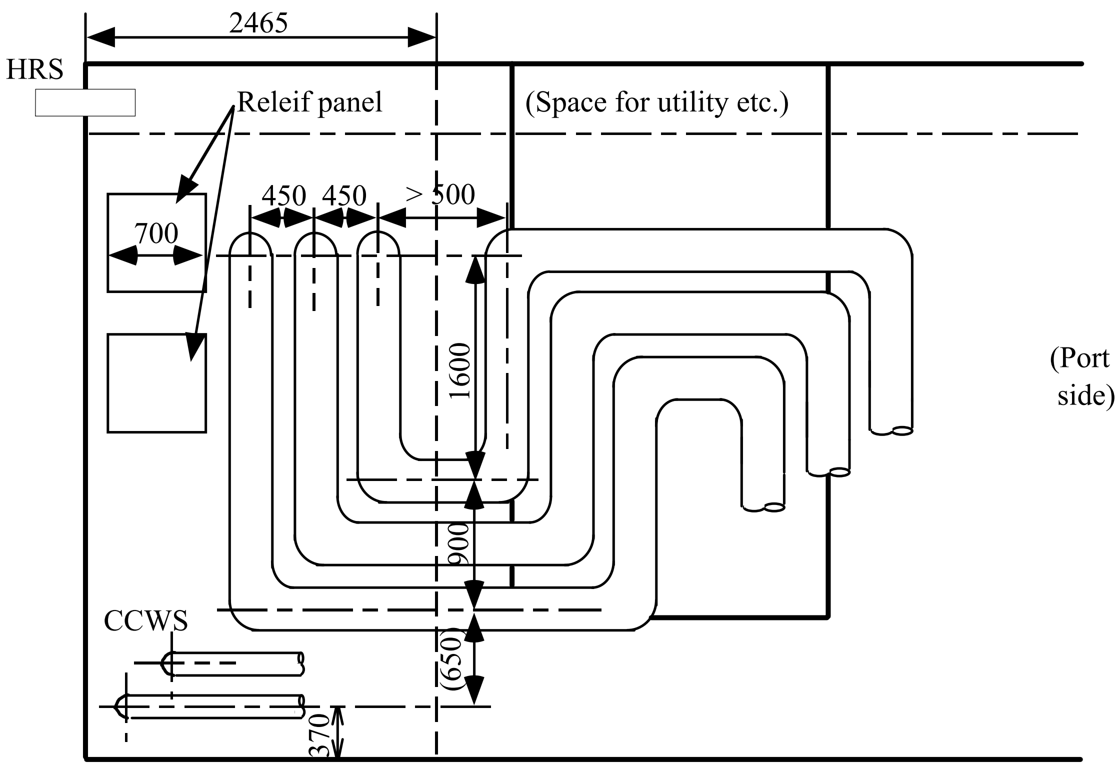
In 2001 ITER FEAT design, two TBM PHTSs for RF TBMs were contained in a cask (container) located in a port cell. Now the maximum external size of the container is 2.62 m (W) x 6.5 m (L) x 3.68 m (H). The length is limited by the bioshield requirements and the latest building design.

The cooling pipes layout inside the port cell and the vertical shaft has been preliminarily investigated [**2-1**], and the result is illustrated in **Figure 2.7-4**. Based on this, the maximum number of cooling pipes that are routed to the TCWS vault is limited to seven. Note that the external diameter of the cooling pipes in Figure 2.7-4 is assumed to be 300 mm including thermal insulation (100 mm thickness) around the pipe. Human access to the port under the cask presence etc. remains an issue.

The allowable opening size of penetrations through the bioshield was evaluated [**2-2**] and has been given to each Party as a design requirement. The investigation results show that the total opening size on the bioshield in a port cell has to be limited below 0.6 m radius equivalent circle to achieve attenuation of the radiation level by a factor of 10.
The size and amount of cables (power and C&I) that are connected to the TBM assemblies must be further specified in detail.



(a) Right side wall



(b) Left side wall

Figure 2.7-4: Preliminary layout of TBM pipes inside port cell

2.7.3 Interfaces with other system in the port cells

In the Port cells #2, 16 and 18 a number of other systems exist, whose presence need to be considered for the layout of the TBM piping and auxiliary system. The complete list of systems is being developed. **Table 2.7-1** identifies some of these systems and provides a reference to the existing CATIA models.

Table 2.7-1: List of other systems in TBM port cells

System	ITER Drawings	Comments
FUELLING LINE UPPER PORT GIS PIS VALVE BOX	18.0054.0.P-G.040511.CGS.SMA	to be re-routed
VENT RING MANIFOLD	21.0237.0.P-G.041006.MAM.MAM	These occupies the front part of the port cell just below the ceiling
TRACER GAS SUPPLY MANIFOLD	21.0238.0.P-G.041006.MAM.MAM	space allocation only
INSTRUMENT AIR (INCL) VALVE ACTUATION MANIFOLD	21.0239.0.P-G.041006.MAM.MAM	space allocation only
INSTRUMENT AIR COLLECTOR MANIFOLDS	21.0240.0.P-G.041006.MAM.MAM	space allocation only
SPECIAL GAS MANIFOLD	21.0241.0.P-G.041006.MAM.MAM	space allocation only
WELDING GAS MANIFOLD	21.0242.0.P-G.041006.MAM.MAM	space allocation only
BREATHING AIR MANIFOLD (IN GALLERIES)	21.0243.0.P-G.041006.MAM.MAM	space allocation only
WATER FEED MANIFOLD	21.0244.0.P-G.041006.MAM.MAM	space allocation only
CHILLED WATER RETURN MANIFOLD	21.0245.0.P-G.041006.MAM.MAM	space allocation only
RELIEF VALVE HEADER MANIFOLD	21.0246.0.P-G.041006.MAM.MAM	space allocation only
COMPRESSED HE (GM PUMPS) MANIFOLD	21.0247.0.P-G.041006.MAM.MAM	space allocation only
COMPRESSED HE (GM PUMPS) MANIFOLD	21.0248.0.P-G.041006.MAM.MAM	space allocation only
PFW/BLK PHTS	26.0090.200.R-N.010326.YYA.YKI	In the connecting duct
Service Vacuum system RING MANIFOLDS	31.0182.03.W-G.040325.RMS.MWS	
CELL COOLER		To be designed
ADS/VDS SYSTEM		Not routed yet
CASK CONNECTION SYSTEM		To be designed

Detailed layout of many of the systems has still to be developed. A preliminary share of the volume in the port cell should be based on the following assumptions:

- The space below the ceiling for a height of 500 mm is reserved for ring manifolds (vacuum fuelling, venting, special gas) and service lines (electric power, instrumentation).

- A path way (about 3.8 m in height and 2.8 in width) in the center of the port cell is reserved for the movement; parking of the port plug maintenance cask should not be occupied by permanent component of the Test Blanket system.
- The TBM pipe works can either be routed at the side of the cask using the shaded areas indicated in the **Figure 2.7-5** below, or can be made of dismountable parts that must be removed before the cask is brought in.
- The pipe works in the port cell must penetrate the side walls of the vertical shaft. This penetrations needs to be leak tight, because the vertical shafts are part of the TCWS that can experience a pressure of up to 0.2 MPa in accidental case. The routing of the pipes (both inside the port cells and in the vertical shaft should consider the presence of 2 relief square panels in the left side walls of the port cell. These relief panels and structural frame occupies an area of about 1 m (from the port cell door) and 2- 3 m from the ceiling.

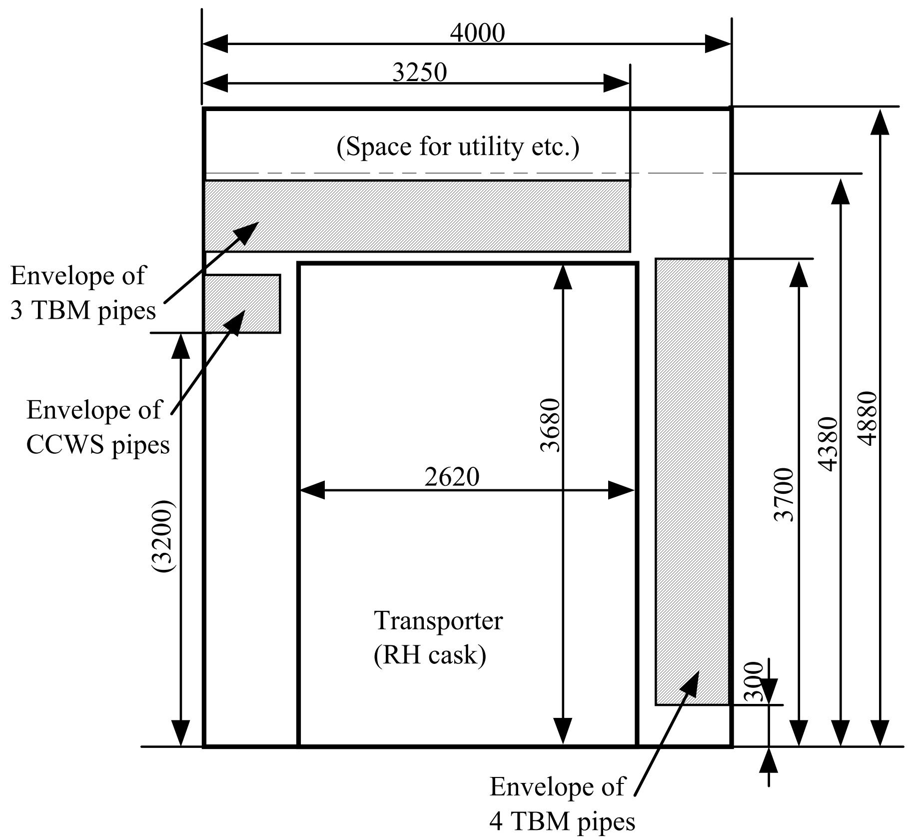


Figure 2.7-5: Space allocation in the test cells (alternative)

2.8 TCWS Vault

In the 2001 ITER FEAT design, a space of 16.6 m (L) x 7.3 m (W) x 6 m (H) was assigned in the south east corner of the TCWS vault for the primary heat transfer systems (PHTSs) of the TBMs, as shown in **Figure 2.8-1.** Four PHTSs for EU and JA TBMs and a pressure relief tank for the RF TBM PTHS were integrated in the space at that stage. Though now the same space is assigned for the TBMs, it is noted that an emergency escape to the tritium building is located near the TBM area and therefore at least 0.5 m width along the east wall has to be assigned for the escape corridor.

The ITER heat rejection coolant (raw water, nominal supply temperature 35˚C, maximum return temperature 75˚C at ~ 0.1 MPa) interfaces with these TBM PHTSs in the assigned space. In addition, the cooling pipes are to be routed from the port cell to the TCWS vault through the limited space of the upper pipe chase and the vertical shafts. On the other hand, the total space proposed for TBMs by Parties is about twice the maximum space currently available. Integration and common use of the cooling loop, especially for the He cooling loops, among Parties is highly recommended.

The nominal air temperature is kept to a limit of 35˚C by internal coolers. Air leakage into the vault may be chronic through valve stems and through pump shaft seals etc, but is removed by a dryer system. The relative humidity is therefore expected to be varying between extremely dry to 100% relative humidity.

During maintenance periods the vault will be connected to a HVAC system such that the air is likely conditioned to be around 25˚C and 40% relative humidity. The range of variation around these nominal points has not been estimated.

In the corner of the TCWS vault the magnetic field strength is sufficiently low so that conventional equipment may be used.

The TCWS vault is kept under a depression of 1 mbar with respect to the galleries and the extracted air is detritiated prior to release into the environment. As for the port cells, released instrument air must be collected into a common header for release into the environment.

Because of the very limited access and the environmental conditions, as well as for space reasons, the cubicles for control and instrumentation must not be placed inside the TCWS vault. Based on the design principle of the large primary heat transfer systems located inside the vault, the safety related cubicles and power distribution panels should be located in special rooms inside the tritium plant building, whereas the non-safety related cubicles and power distribution panels may be located on top of the roof of the vault near a load centre and/or the four corner of the gallery area in the tokamak building. The control cubicles may also be located at the four corner of the gallery area in the tokamak building.

In the TCWS vault there are routed general utilities including:

- Instrument air (probably ~ 0.8 MPa),
- Breathing air,
- Collector for released instrument air,
- Heat rejection water, nominal supply temperature 35˚C, maximum return temperature 75˚C, at ~ 0.1 MPa.,
- Component cooling water (dematerialized water, normal inlet temperature 40˚C at a few bar.

Installed and consumed electrical power must be defined for the TBM components. Cable penetrations must be of the pressure tight type, as the vault could experience an overpressure of nearly 0.1MPa during an accident.

Access into the vault with equipment is gained via the main lift and a pathway is available from the lift to the area where the components are located. The pathway is 1.5m (W) x 3.7m (H).

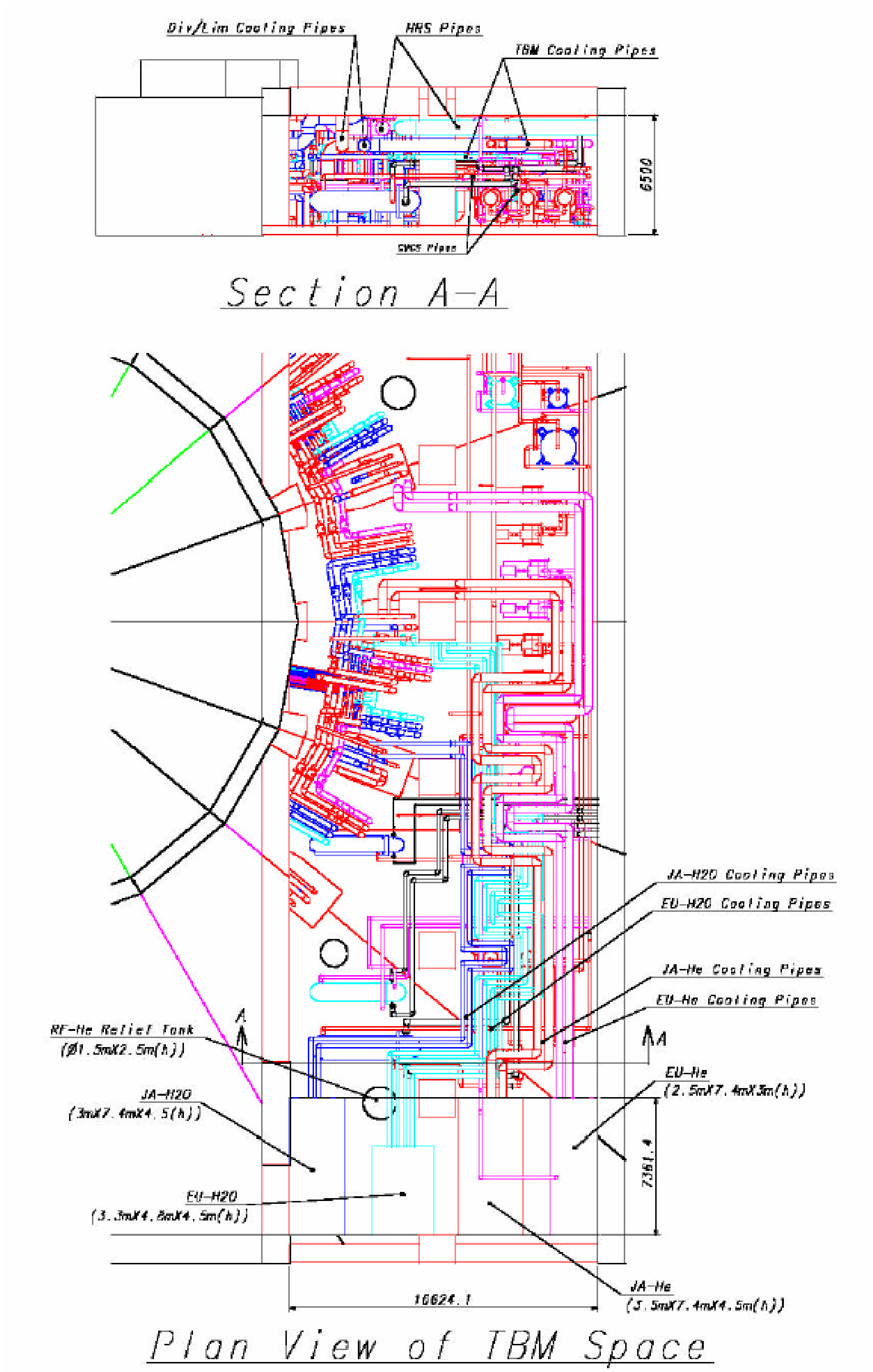


Figure 2.8-1: Layout of TBM cooling system inside TCWS vault (from 2001 ITER FEAT DDD)

2.9 Remote service and hot cells

The ITER Remote Handling (RH) system and equipment will be available and will be operated for the replacement of the Test Blanket (TB) plugs in the same way as for the other ITER port plugs. This means that, prior to the TB plug removal, all equipment located in front of the TB port must be removed and the port cell cleared of any TB related equipment, services, supply lines and cables to allow the deployment of the RH transfer cask.

The ITER Hot Cell (HC) is currently available to receive, temporarily store and process prior to final disposal (or return to the Vacuum Vessel – VV) 3 TB plugs. Depending on the complexity of the operation, TB plug repair/refurbishment/testing inside the HC may also be possible (see chapter 7).

2.10 References

[2-1] Y. Kataoka, N 26 MD 49 W 0.2, September 2004.

[2-2] H. Iida, "Allowable size of openings in the bioshield from radiation shielding point of view" (NAG-254), January 2005.

3 – ITER TESTING CAPABILITIES AND TESTING OBJECTIVES

ITER may be the only opportunity for testing DEMO-relevant TBMs in a real fusion environment before the construction of a DEMO reactor ensuring: i) in the initial H-H phase, relevant magnetic fields, surface heat fluxes, and disruption-induced loads, and, ii) in the following D-T phase, an additional relevant neutron flux, volumetric heat, and Tritium production with corresponding T-management capabilities.

Therefore, ITER should take this opportunity to test "tritium breeding blanket concepts that would lead in a future fusion reactor to tritium self-sufficiency, the extraction of high-grade heat, and electricity generation".

The most important restriction for blanket testing is that in ITER the magnitude of neutron flux, neutron fluence and volumetric power density is considerably lower than what is expected in a DEMO reactor. As a consequence, in most cases, for each selected blanket concept, several TBMs have to be developed making use of "engineering scaling" for testing specific DEMO "act-alike" TBMs for addressing the different aspects of the TBM performances.

The TBWG agreed that essential results can be obtained from TBMs testing both during the D-T phase and the H-H phase, provided appropriate instrumentation will be available. Details are given below.

3.1 Main test objectives for the D-T phases

Taking into account these limitations, the major overall testing objectives are the following:

- validation of structural integrity theoretical predictions under combined and relevant thermal, mechanical and electromagnetic loads, with consequent validation of the used fabrication techniques and processes;
- validation of Tritium breeding predictions and their acceptability for the DEMO self-sufficiency proof;
- validation of mechanical, thermal and thermo-mechanical predictions for strongly heterogeneous breeding blanket concepts with volumetric heat sources, including the maximum breeder material temperature and structural stability of the breeder elements, with the determination of safety factors to be included in DEMO designs development;
- validation of Tritium recovery process efficiency and T-inventories in blanket materials, determination of the tritium permeability through breeding elements structure and of the efficiency of anti-permeation coatings;
- observation of possible low-dose irradiation effects on the performance of the blanket modules;
- confirmation of remote maintenance approach and equipment;
- demonstration of the integral performance of the blankets systems.

It must be pointed out that, because of the expected relatively long time constants, tests concerning Tritium-related performances generally need longer pulses (>1000 s) than the ITER reference one and an uninterrupted series of back-to-back cycles; therefore, they will essentially be performed only at a later stage. On the other hand, most of TBM subcomponents thermal time constants are lower than 400 s and then relevant temperatures can be achieved during a reference pulse length.

Maximum expected neutron damage is about 3 dpa (in ferritic steel) even after 20 years of operation, while dpa levels required for DEMO are greater than 70 dpa. Therefore, ITER cannot give answers to long-term irradiation effects on blanket performances, failures, functional and structural materials and interfaces, synergistic effects. These should be addressed in other facilities, e.g. fast-neutron spectrum fission reactors, future neutron sources such as IFMIF [**3-1**], future component test facilities [**3-2**].

3.2 Main test objectives for the H-H phase

Important data can be obtained during the H-H phase, which implies the installation of validated TBMs since the beginning of ITER operation. In fact, information on TBMs systems compatibility with ITER constraints and TBMs systems operation has to be obtained prior TBM activation that is expected to become larger than the hands-on limits after only a few short D-T pulses.

The main results expected in this phase are the following:

- demonstration of the structural integrity of the TBM structures and attachment during disruption and Vertical Displacement Events (VDE);
- assessment of the impact on Ferritic/Martensitic steel, used as a structure for most TBMs, on magnetic fields deformation in static conditions;
- demonstration of the performance of TBMs remote handling operation and transport to hot cell, including required operation in port cells;
- demonstration of the overall TBMs system functionalities, taking into account connections to the port cell, to the TCWS vault and to the Tritium building;
- verification of the need of Be-coating on the FW (if necessary) by testing uncoated TBMs.

Moreover, essential TBMs validation data can be obtained in this phase to be used in support of the safety dossier that will be required for licensing.

For TBMs using liquid metals, additional data can be obtained for the validation of the MHD pressure drops estimations and for the T-control and management simulated with addition of H/D in the flowing liquid.

3.3 Required Instrumentation

In order to achieve the expected testing objectives, an important issue is the availability of appropriate instrumentation and measurement techniques.

Most of the instrumentation available on the shelves are not compatible with the severe TBM environment which, since day-one, is characterized by high magnetic fields, high temperatures and pressures, materials compatibility issues and, also, by limited accessibility of the TBMs both for repair and for data acquisition. In the D-T phase, additional high neutron and gamma fields will be present and direct access to TBMs will be no more possible.

A large R&D program on instrumentation may therefore be required. Moreover, a detailed assessment of the feasibility of the instrumentation integration in the TBMs designs and their impact of the TBMs performances has to be performed.

3.4 General comments on tests feasibility

The above general objectives apply to all blankets to be tested in ITER. They provide the framework for the specific objectives to be developed for the different blanket concepts taking into account the particular generic design features and issues, the capabilities and limitations of ITER as test environment, and the alternatives to carry out particular tests in out-of-pile facilities or fission reactors at lower cost or under more relevant conditions.

Going from one objective to the next one may require the exchange of some TBM's. It may be also necessary to use more than one TBM's for the achievement of an objective. Therefore, regular replacement periods for all modules have to be scheduled. In addition, it may be necessary to replace after short operation periods a part of the instrumentation especially in the neutronics test modules. These replacements have to be synchronized among all parties to minimize the required down time.

In chapter 2 it is indicated that, because of possible magnetic field perturbations due to the TBM ferro-magnetic materials leading to a loss of fast particles and then to localized hot spots, the expected surface heat flux on the TBM FW in the D-T phase could be up to 0.5 MW/m^2 on 10% of the FW area, with the consequence that TBMs has to be designed to this value despite the typical average heat flux will be about 0.27 MW/m^2. This specification has a strong impact on TBMs designs and it appears that it can be accepted by making specific design modification (e.g., FW coolant bypass).

However, because of the unpredictable character of these operating conditions, the interpretation of some of the testing results (such as FW thermal stresses) may not be possible under such conditions. It is therefore required that, at least in the later D-T phase, well-defined and predictable operating conditions be available for a sufficiently large number of pulses, in order to be able to reach the required TBM operating parameters for performing DEMO-relevant tests.

The monitoring of the test modules, in particular of parameters relevant for the specific objectives, requires appropriate instrumentation. This is of particular importance with respect to the validation and calibration of the design tools, one of the main objectives of blanket testing in ITER. Besides the data obtained by the on-line instrumentation, post-test examinations in the Hot Cells will deliver significant information on the operational behavior and for code validation.

3.5 References

[3-1] H. Matsui, et al., Roles of IFMIF in the Fusion Technology – Status and Perspectives, Proceeding of ISFNT-7, Tokyo, May 22-27, 2005.

[3-2] Y.-K.M. Peng, et al., Fusion Engineering and Plasma Science Conditions of Spherical Torus Component Test Facility, Fusion Science and Technology, Vol. 47, Nb. 3, April 2005, pp. 370-383.

4 – BLANKET DEVELOPMENT AND CORRESPONDING TEST BLANKET MODULES PROPOSED BY PARTIES

4.1 Introduction and General Approach

This chapter presents the different breeding blanket development programs of each ITER Party and the corresponding TBMs test proposals. Following the agreed guidelines given by the TBWG, the Parties have written their own proposals without taking into account space availability in ITER which will be dealt with at a later stage.

The agreed guidelines are the following:

- In order to give relevant information for DEMO breeding blanket development, ITER TBMs and associated systems are based on the same blanket design, the same functionalities and the same technologies as the corresponding DEMO blankets. Moreover, the design of "act-alike" or "look-alike" mock-ups needs a direct reference to DEMO blankets design and performances. Reference DEMO blankets are therefore presented by all Parties.

- Each Party has written a Design Description Document (DDD) for each proposed TBM system to be tested in ITER independently from port allocation or space availability in ITER test ports. *All DDDs are available on the TBWG ftp server.* The objective is to give a clear status of the present TBM development. The synthesis of the DDDs contents is given in the following chapters. The final selection for test port allocation will depend on future progress of the Parties TBM development program in term of obtained R&D results, financial commitment and on possible evolution of the Parties testing strategy that could lead to collaborative actions and/or common test program between two or more Parties. This choice leaves the maximum of flexibility of the final TBM test program.

Therefore, the following sub-chapters describes the original Parties proposals that will be used as a basis for the presentation of the overall testing program (§ 5) and of the Port Cell arrangement (§ 6).

It is stressed that the design work on TBMs by different Parties presented in the following sections has been performed in parallel to the discussion on the detailed specifications whose conclusions are given in Chapter 2 (e.g. loads, boundary conditions, geometry, interfaces and frame dimensions). Therefore, the assumptions made for the TBMs designs are not always fully consistent with such specifications. The same consideration applies to the specific technical requirements derived from quality assurance and acceptance tests as described in Chapter 9.

4.2 People's Republic of China Proposals

4.2.1 DEMO Studies and Testing Strategy in ITER

4.2.1.1 Solid Breeder Blanket Options

(1) HC-SB DEMO Blanket Studies

(2) HC-SB TBM Testing Strategy in ITER

4.2.1.2 Liquid Breeder Blanket Options

(1) SLL/DLL DEMO Blanket Studies

(2) SLL/DLL TBM Testing Strategy in ITER

4.2.2 TBMs design and analyses

4.2.2.1 HC-SB TBM design and analyses

(1) TBM configuration and design parameters

(2) Neutronics analyses

(3) Thermo-hydraulic analysis

(4) Safety analyses

4.2.2.2 DFLL-TBM design and analysis

(1) System overall description

(2) Material

(3) Neutronics analysis

(4) Thermalhydraulics and MHD analyses

(5) Thermo-mechanics analysis

(6) Safty analysis

Activation analysis

Tritium inventory and permeation analysis

Severe accident analysis

4.2.3 TBMs ancillary systems

4.2.3.1 HC-SB TBM ancillary systems

(1) Helium coolant system (HCS)

(2) Coolant purification system (CPS)

(3) Tritium extraction system (TES)

(4) Space requirement and allocations

4.2.3.2 DFLL TBM ancillary systems

(1) Lithium lead system

(2) Helium coolant system

(3) Helium coolant purification systems

(4) Tritium extraction system

(5) Space requirement and allocations

4.2.4 Supporting Activities to other Parties TBMs

4.2.4.1 HC-SB TBM Supporting Activities to other Parties TBMs

4.2.4.2 DFLL TBM Supporting Activities to other Parties TBMs

4.2.5 Supporting R&D and Validation Program prior to the installation in ITER

4.2.5.1 HC-SB TBM Supporting R&D and Validation Program

4.2.5.2 DFLL TBM Supporting R&D and Validation Program

4.2 People’s Republic of China Proposals

4.2.1 DEMO Studies and Testing Strategy in ITER

4.2.1.1 Solid Breeder Blanket Options

(1) HC-SB DEMO Blanket Studies

He–Cooled Solid Breeder (HCSB) blanket with Ferritic/Martensitic (FM) represents the main stream in DEMO relevant blankets and has foundation of the world R&D database. The solid breeder blanket is considered as one of the main options for the Chinese DEMO blankets, which are based on the definition and development strategy of China DEMO fusion reactor [4.2.1.1-1&2] According to the ITER engineering design activity (EDA), the philosophy and design features of HC-SB DEMO blanket are supposed: 1) validating the reliability of calculation codes including neutronics, thermal-hydraulics and electro-magnetic, etc.; 2) efficiently removing high-grade heat; 3) tritium technologies and tritium self-sufficiency; 4)vertical scheme of blanket assembling/disassembling.

The major design parameters of HC-SB DEMO blanket are summarised in Table 4.2.1.1-1. HC-SB blanket studies aim at establishing physics and engineering bases and constraints for the development of fusion power reactors and are the important aspects of long-term national fusion program to evaluate the technology, economics, safety, environmental impact for the potential magnetic fusion applications.

The HC-SB DEMO blanket is designed as modular structure. There are 14 sectors along the poloidal direction, 4×18 sectors on the toroidal direction. The segments mounting/ dismounting operations are carried out through the vacuum vessel vertical ports. The total thickness of inboard blanket is 630 mm and the radial thickness of outboard blanket is 800 mm.

The first wall is a complicated shape plate with toroidal coolant channels and beryllium protective coating. It is supported by stiffness ribs to provide required strength in abnormal operation conditions. The internal space between the first wall module and back plate is used for breeding zone arrangement.

The inboard/outboard breeding zone contains 14 rows of circular coolant channels, 5 rows of Li_4SiO_4, 7 rows of Be pebble and 1 row of Be protection. The single pebble-bed (38% porosity) of lithium orthosilicate is used as breeder material. The free space between first wall, back plate and external surfaces of coolant circular channels is used for multiplier location. The beryllium binary pebble-bed (20% porosity) is used as multiplier material. The total coolant pressure drop in the circulation circuit is 0.15 MPa, what is an acceptable value. Optimization by changing thickness and enrichments was performed within limitations. MCNP code was used in calculations of 3-D neutronics.

All design features were taken into consideration. The optimisation resulted in tritium breeding ratio of 1.1 at the beginning of reactor operation. Lithium burn up can decrease this value by 3-6 % after 3 FPY.

(2) HC-SB TBM Testing strategy in ITER [4.2.1.1-3]

The important goal of ITER is to test component and demonstrate tritium breeding and transfer of heat to electricity in DEMO. DEMO reactor will conduct the direction of R&D for ITER. China plans to develop relevant technical of helium-cooled solid blanket and obtain enough key testing data from ITER, which can support development of next step DEMO.

China will develop the EM, NT, TT and IN TBMs to test or demonstrate the electric-magnetic effect of blanket, nuclear issues, tritium extraction, heat removal and integration performance of DEMO blanket. Table 4.2.1.1-2 gives HC-SB DEMO lifetime of testing requirement for replacement, fluence of neutron irradiation and number of pulses, which are used to help make testing plans on ITER. The tests performed in phase of H-H, D-D, low-duty D-T and High -duty D-T operations have been listed in Table 4.2.1.1-3.

After completing all testing and experiments ITER, China will develop own DEMO and PROTO fusion power plant. Development of low active and high performance materials will be independently preceded. The advanced materials, such as CLAM being developed, will be applied at DEMO.

Based on China fusion energy strategy and its definition of DEMO reactors, China has planned to independently develop the HC-SB TBMs and test them during ITER operation period. China hopes to equally share space with other parties and expects to deliver a HC-SB TBM on day one. According to the operation plan of ITER facility, four operation phases are expected, H-phase, D-phase, low duty DT-phase and high-duty D-T phase. China plans to test its EM-TBM during the H-phase on the first day operation. EM-TBM test is devoted to the investigation of electromagnetic effects. Following the EM-TBM test, other TBMs will be tested during D-phase and two D-T phases. Different blankets have different test objectives.

a) Electro-magnetic (EM) test during the H-phase. Testing objectives: measurement of magnetic fields, eddy currents, forces and moments acting on blanket during fast transients to ensure the integrity of the TBM.
b) Neutronics and tritium production test (NT) during the D-phase. Testing objectives: evaluation of tritium breeding performance, validation of neutronics codes and evaluation of the nuclear data.
c) Thermal and mechanical test (TT) during low-duty D-T phase. Testing objectives: investigation of thermal and mechanical behaviours. TBM's design will simulate as close as possible the condition in the DEMO reactor.
d) Integrated tests (IN) during high duty D-T phase. Testing objectives: evaluation of the integrated behavior of TBM and required external systems for cooling and tritium extraction under a large variety of loads and operating conditions.

A draft possible test scenario is proposed in Table 4.2.1.1-3 based on ITER-FEAT different operation phases.

Reference:

[4.2.1.1-1] K.M. Feng, et al., Design Description Document for the Chinese Helium Cooled Solid Breeder (CH HC-SB) TBMs, (Draft Design Report), Dec. 30, 2004.

[4.2.1.1-2] K.M.Feng, C.H.Pan, G.S.Zhang, X.Y.Wang, Z.Chen, T.Yuan, et. al., Preliminary Design for a China ITER Test Blanket Module, Presented at the 7th International Symposium of Fusion Nuclear Technology (ISFNT-7), Tokyo,, 22-27, June, 2005.

[4.2.1.1-3] K.M.Feng, et. al, Chinese Helium Cooled Solid Breeder (CH HC-SB) TBMs Design Description report, Presented at the 12th - 15th ITER TBWG-meetings (2004-2005).

Table 4.2.1.1-1 Design Parameters of HC-SB DEMO Blanket

Fusion Power, P_f Plasma Radius, R/r Neutron Wall Loading, NWL Max. Surface Heat Load, SHL Tritium Breeding Ratio, TBR	MW m MW/m^2 MW/m^2	2000 7.0/2.1 2.64 0.7 1.05 -1.1
Coolant Temperature Range, T Pressure, P	 °C MPa	He 300-500 8
Tritium Breeder [% ^{6}Li enrichment] Packing Factor, PF Tempreature Range, T	 °C	Li_4SiO_4 [80%] pebbles 63% 400-950
Neutron Multiplier Tempreature Range,T	 °C	Be pebbles at 80% PF 400-620
Structure Material Temperature Range, T	 °C	CLAFM 300-550

Table 4.2.1.1-2 HC-SB DEMO lifetime of testing requirement

Reactor elements lifetime, years - Replaceable components (blanket, divertor etc) - Permanent components	 ~ 8-10 ~ 15-20
Lifetime FW fluence, MWa/m^2 - Replaceable components - Permanent components	 ~10 ~50-
Lifetime number of pulses - Replaceable components - Permanent components	 ~150-1500 ~700-7000

Table 4.2.1.1-3. Possible testing scenario for HC-SB TBM

Operation phases	Test description	Test requirement (estimated pluses)	TBMs
H-H phase	Safety, surface heat flux, E-M force	~ 600	EM
D-D phase	D performance, surface heat flux	~3000	EM
D-T low duty	Tritium inventory, Tritium permeation, code validation, stress distribution	~ 3000	NT TT
	PIE	HC	
D-T high duty	Reliability test, Tritium extraction, integrated test,	~ 5000	TT, IN

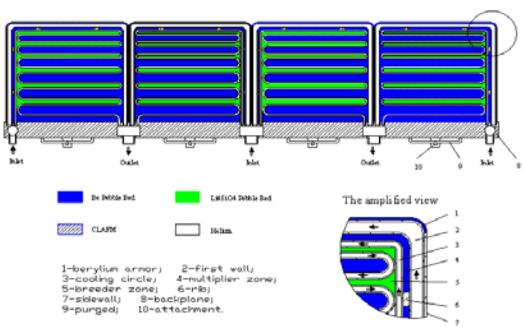


Fig. 4.2.1-1 HC-SB DEMO blanket module

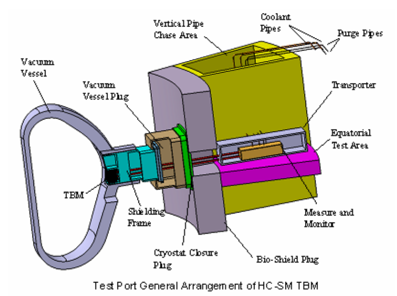


Fig.4.2.5-1 Test port general arrangement data libraries for HC-SB

4.2.1.2 Liquid Breeder Blanket Options

(1) **SLL/DLL DEMO Blanket Studies**

The liquid lithium lead breeder blanket is considered as one of the main options for the Chinese DEMO blankets due to their potential attractiveness of economy and safety and relatively mature technology base. Two specific liquid metal blanket concepts are proposed, namely the Dual-cooled Lithium Lead (DLL) breeder blanket and the Quasi-Static Lithium Lead (SLL) breeder blanket for the fusion DEMO reactor. Details on the SLL/DLL blanket and relevant reactor design has been seen in [4.2.1.2-1&2]. The DLL blanket is a dual-cooled lithium lead (LiPb) breeder system with helium gas to cool the first wall and main structure and LiPb eutectic to be self-cooled. The Reduced Activation Ferritic/Martensitic (RAFM) steel, e.g. the CLAM (China Low Activation Martensitic) steel [4.2.1.2-3], is considered as the structural material.

As the reference DLL module design shown in Fig.4.2.1.2-1, LiPb in the front row of flow channels adjacent to the FW goes down from the top of the module, divides into two branches at the bottom and goes up in the rear two rows of the channels. The FCIs (Flow Channel Inserts) e.g. SiC_f/SiC, are designed as thermal and electric insulators to reduce the MHD pressure drop of LiPb flow. Coating may be considered as the tritium permeation barrier if necessary. The opposite U-shaped helium flow scheme is adopted in the FW, and the U-shaped and W-shaped flow schemes in toroidal and radial steel grids, respectively. The structural box is reinforced with several radial and toroidal stiffeners to withstand the disruption-induced forces and the He/ LiPb pressure under normal and abnormal conditions.

The average/maximum neutron wall loadings are 2.72/3.54MW/m^2, and the average/maximum surface heat flux on FW 0.54/0.70 MW/m^2. The inlet/outlet temperatures of helium gas coolant at 8MPa are 300°C/ 450°C, considering the temperature limits of RAFM steel under irradiation. Due to the utilization of FCIs, the outlet temperature of LiPb as high as 700°C can be achieved with designing the inlet temperature of 480°C while considering the pinch point. Main reference design parameters of the DLL/SLL-DEMO blanket modules are listed in Table 4.2.1.2-1.

The SLL blanket is another option of the Chinese DEMO blankets if the critical issues of the DLL blanket could not be solved and validated by testing in ITER. To avoid or mitigate those critical problems resulting from magneto-hydrodynamic (MHD) effects and FCI technology, the SLL blanket is designed to use quasi-static LiPb flow instead of fast moving LiPb with the similar structure as of the DLL module. In this case, coating is probably needed to protect the steel structure against corrosion of LiPb and to reduce tritium permeation and MHD effects.

DFLL (Dual-Functional Lithium Lead)-TBM system [4.2.1.2-4&5] is designated to check and validate the technologies of both the SLL and the DLL concepts which are proposed to be tested with as similar as possible basic structure and auxiliary systems except for including FCIs and more quickly flowing LiPb in DLL-TBM. These two options will be assessed and tested earlier under out-of-tokamak conditions and in the EAST superconducting tokamak before a final decision can be made for testing in ITER.

(2) SLL/DLL TBM Testing Strategy in ITER

To balance the reduction of potential risk and the pursuit of potential attractiveness of the blanket development, the two-stage testing strategy of DFLL-TBM is proposed, including the SLL-TBM testing on the earlier stage and DLL-TBM testing on the later stage.

Prior to ITER, two types of size-reduced (1/3) SLL-TBM and DLL-TBM may be tested in EAST, the superconducting tokamak device, in China, which is expected to operate in 2005/2006 on the basis of the fact that the designed electromagnetic parameters and average heat flux on the FW of EAST are comparable to those of ITER as shown in Table4.2.1.2-2. EAST can server as a valuable pre-testing platform for TBMs. Some of the testing tasks for H-H and D-D phases of ITER can be pre-conducted in EAST prior to ITER. The data and experiences deriving from EAST-TBM testing can be applied to optimise and improve the design of DFLL-TBM system and the testing plan for ITER.

The testing program of DFLL-TBM covers three phases as follows: First, test of small scale semi-mock-up under out-of-pile condition, mainly for validation of the TBM mechanic performances, the fabrication route and techniques, and for assessment of reliability and safety with regard to ITER standards; Second, test of middle-scale (1/3) TBM in EAST to validate the design tools and codes for electro-magnetic, thermo-mechanic and neutronics, and to check the availability of instruments and assessment of MHD effects, as well as to demonstrate the TBM system design before DFLL-TBM is installed in ITER; Finally, test of full-scale consecutive TBMs corresponding to different operation phases of ITER during the first 10 years. Therefore, the ‘act alike’ DFLL-TBMs are designed in turn as the EM-TBM for testing of electromagnetic effects, as the NT-TBM for performance of neutronics, as TT-TBM for thermo-mechanics and tritium behavior, and as IN-TBM for integrated performance testing, respectively. This program will allow to consecutively validate SLL/DLL concepts, technologies and design tools with reliable and safe operation, and finally to demonstrate relevant technologies for the fusion DEMO reactors. It should be noted that this test program is developed assuming successful testing in earlier phases. The detailed testing schedule is listed in Table 4.2.1.2-3.

Because the available space in port cells does not allow accommodating several ancillary system units, the space-sharing testing strategy among parties seems to be difficult although it would be possible if size-reduced TBMs and ancillary systems could be designed with common interfaces. The time-sharing strategy may be arranged, e.g. the DFLL-TBM probably possibly shares the half of port with another party, which needs to be investigated further.

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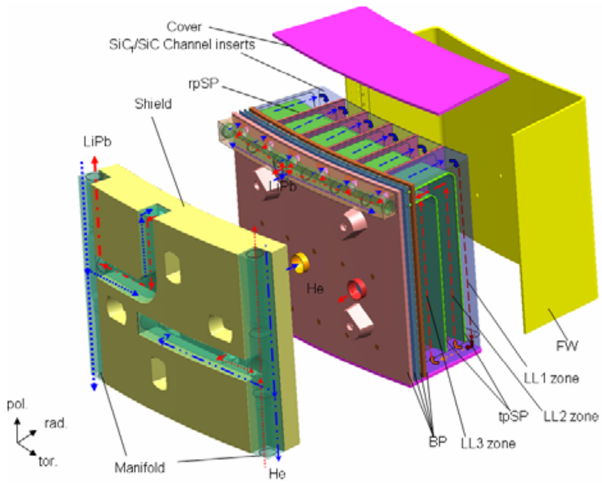


Fig.4.2.1.2-1 The exploded view of DLL DEMO blanket module

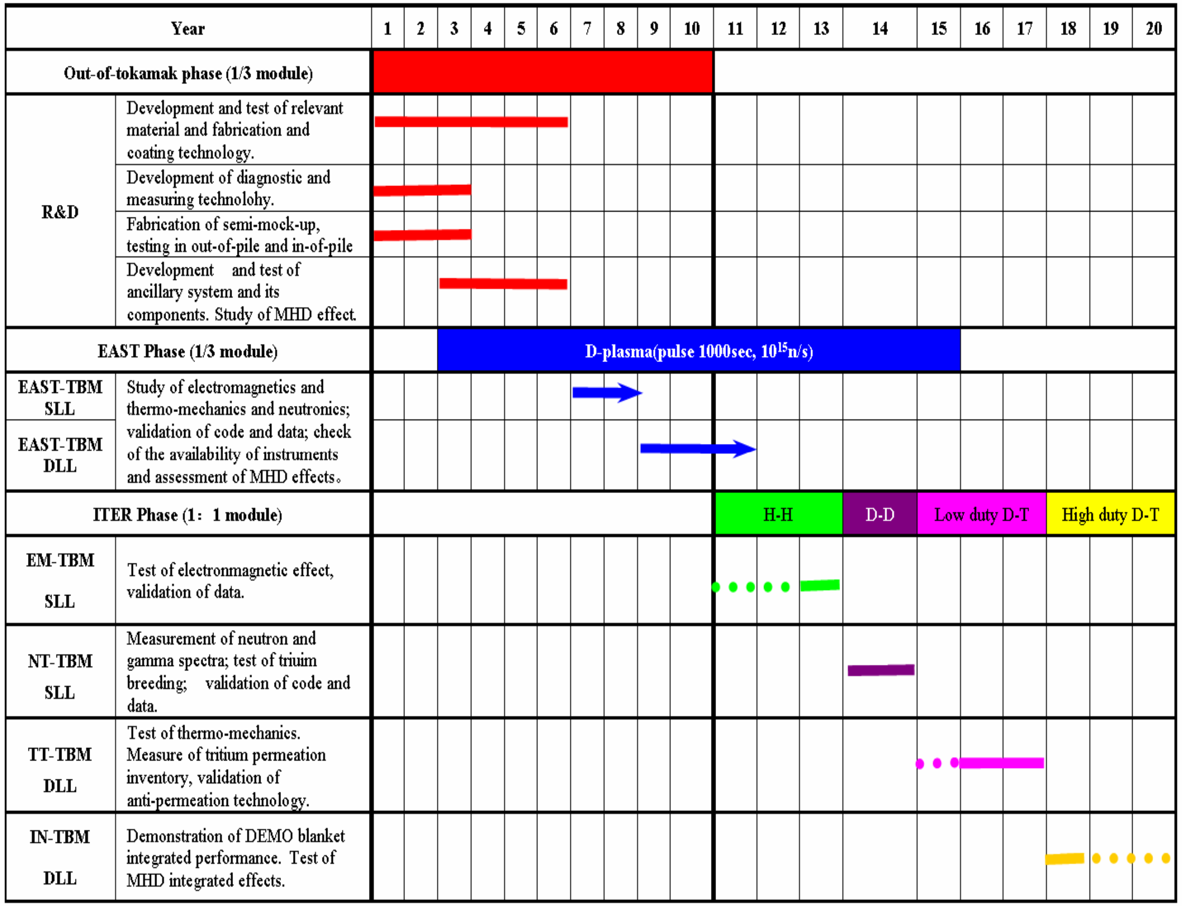
Table 4.2.1.2-1 The reference design parameters of DLL/SLL-DEMO blankets

Blanket		DLL	SLL
Structural material		RAFM (CLAM)	RAFM (CLAM)
Functional material(s)		FCI：e.g .SiC_f/SiC, Coating: e.g.Al_2O_3	Coating: e.g. Al_2O_3
Plasma core	Fusion power /MW	2500	2500
	Major radius /m	6	6
	Minor radius /m	2	2
Heat source	Max. heat flux /$MW\cdot m^{-2}$	0.7	0.7
	Max. neutron wall load /$MW\cdot m^{-2}$	3.54	3.54
	Nuclear heat deposition (FW/breeder zone) /MW	5.7/15.3	~5.7/15
Coolant He	In/Out temp. /^oC	300/450	300/450
	FW/SP velocity /$m\cdot s^{-1}$	115/40	~120/80
	Pressure /MPa	8	8
Breeder material LiPb	In/Out temp. /^oC	480/700	/450
	Ave. velocity of LiPb in breeder zone /$mm\cdot s^{-1}$	99/78/12	~1
TBR		1.2	

Table 4.2.1.2-2 Main parameters of EAST in comparison with ITER

Device	**EAST**	**ITER**		
Phase	DD	HH	DD	DT
R (m)	1.95			6.2
A (m)	0.46			2
Bt (T)	3.5-4.0			5.3
Neutron rate (n/s)	10^{15}~10^{17}			1.77×10^{20}
Avg.HF (MW/m^2)	0.1~0.2	0.11		0.27
Port Size	0.97m × 0.53m	2.2 m×1.7m		
Pulse (sec)	~1000	100-200		400

Table 4.2.1.2-3 Testing schedule for



4.2.2 TBMs design and analyses

4.2.2.1 HC-SB TBM design and analyses

(1) Configuration and design parameters

As one of options, the HC-SB TBM design adopt SB/He/FM concept. HC-SB TBM design will aim at the IN-TBM modules and its relevant ancillary system. After completing IN-TBM, others TBMs (e.g. EM-, NT-, TT-TBM) for different operation condition and operation parameters of the ITER plan will be designed in detail. Up to now, a draft Design Description Document (DDD) for HC-SB IN-TBM based on the definition and development strategy of China DEMO fusion reactor has been completed [4.2.2.1-1]. The detailed schedule of relevant R&D, testing program will be established.

The structure outline of the HC-SB TBM has been carried out and shown in Fig.4.2.2.1-1. The design parameters for the HC-SB TBM are listed in Table 4.2.2.1-1[4.2.2.1-2].

The structure of TBM is consist of the following main components: first wall, caps, grids, manifolds, attachments, cooling pipes, purge gas pipes and sub-modules, which are presented in Fig.4.2.2.1-2. A dual-layer structure with the thickness of 30 mm is used in first wall design. An U-shaped helium cooling channel in series connection are used in the cooling circuit design. The grids and caps with their own helium cooling channels are considered. The grids are welded on first wall, which will enhance the safety and reliability of structure. The integral HC-SB TBM consists of 9 sub-modules (cell). Each cell as a sub-module has independent cooling circuit and purge gas circuit. As shown in Fig.4.2.2.2-3, the cooling and tritium extraction path with parallel connection for each sub-module has been designed.

The configuration and dimension of the integral HC-SB module design are based on the consideration of 1/4 port space position on ITER Port C. It is easily extended to a half-port-space module. The lithium orthosilicate, Li_4SiO_4 is selected as tritium breeder. The ferritic/martensitic steels, such as CLAM, EUROFER etc., may be used as the reference structure material. The helium gas is used as the coolant and the tritium purge gas. To assure an adequate tritium-breeding ratio (TBR), beryllium pebbles are adopted as neutron multiplier; enriched lithium-6 of 80% is used as the tritium breeder.

The tritium generation rate is 0.033g/d under the ITER standard operation condition. In order to improve the distribution of power density in the blanket module, the arrangement of the neutron multiplier Be in the breeding zone will be further optimised in the DDD document. Be pebbles of diameter 0.5 -1 mm are chosen for Be zone.

(3) **Neutronics analyses** [4.2.2.1-3]

The neutronics calculations are performed using the neutronics transport code: 1-D ONEDANT, 2-D TWODANT and 3-D MCNP. The 3-D results from MCNP are definitely selected as input data for other systems design. The 1-D and 2-D calculations are mainly used in optimization calculation for geometry and materials.

The data library is based on FENDL2.0. The results of 1-D ONEDANT neutronics transport calculation show the local tritium breeding ration (TBR) of 1.15. Fig.4.2.2.1-5 shows results of the global TBM tritium breeding ratio (TBR) in different breeding zones by means of using the one-dimensional model. Detailed three-Dimensional neutronics analysis results will be given in the final DDD report.

Fig.4.2.2.1-4 shows the distribution of power density with materials zones. It is the important input parameters for thermal-hydraulic analysis. In calculation, the neutron loading of 0.78 MW/m^2 is adopted. It is shown from figure that a peak power density of 8.9 MW/m^3 in the module occurs at the end of first breeding zone of Li_4SiO_4. Fig.4.2.2.1-5 shows production rate of tritium in breeding zones with distance to FW. The tritium generation rate is 0.033g/d under 22% duty factor for ITER, which will be offered to design of tritium extraction system (TES) and coolant purification system (CPS).

(3)Thermo-hydraulic analysis [4.2.2.1-4]

The temperature distributions of the FW (3-D) and sub-module (2-D) obtained by ANSYS code and FLUENT code are shown in Fig.4.2.2.1-6. The result shows that the temperatures of all zones drop in the permissible temperature limitation. The thermal-hydraulic and thermo-mechanical calculations have been preformed by ANSYS code. The results show that the peak temperature in the test module is 737 °C with the surface heat loading of 0.5 MW/m^2. Total heat power of 0.76 MW will deposit in the module. The inlet and outlet temperature of the helium cooling are 300°C and 500°C. Fig. 4.2.2.1-7 illustrates the analysis result of FW and sub-module cooling plate. The peak temperature is 530°C in FW and 522°C in sub-module cooling plate, and peak stress is 244MPa in FW and 219 MPa in sub-module cooling plate. This result satisfies the requirements of structure strength regulations.

（4）Safety analyses.

- **Radioactive Inventory** [4.2.2.1-5]

Neutron fluxes and energy spectra are provided by one-Dimensional neutron transport and burn-up code BISON1.5 calculation for each specified material zone. Total radioactivity and afterheat are calculated by using the activation code, FDKR and its decay chain library DCDLIB [4.2.2.1-6]. Activation calculation was performed assuming a continuous irradiation over 0.53 years at full fusion power (500 MW) with a neutron wall loading of 0.78MW/m^2, this results in a total first wall fluence of 0.41MWa/m^2, which is a conservative estimates for the safety analyses. The HC-SB TBM employs the low activation (LA) steel Eurofer as structure material, Li_4SiO_4 pebbles as breeder material and Beryllium pebbles as neutron multiplier. The activation behaviour of these materials is affected to a large extent by impurities and other minor elements. At shutdown, the total radioactivity inventory is as low as about 0.54MCi. They are mainly due to the Eurofer structure present in the shield, back plate, the back breeder channel, and the FW. The level is about 0.09MCi after 1 year and is about 0.008 MCi after 10 years. This is an extremely low level and therefore it imposes no concerns with regard to disposing the activated materials of the TBM. Fig.4.2.2.1-8 and Fig.4.2.2.1-9 show the afterheat and the activity as a function of the shutdown time, respectively.

- **Decay Heat Generation**

The total decay heat at shutdown is about 0.005MW. The decay heat levels after 1 hour, 1 day, 1 year, 10 years are 3.3×10^{-3}MW, 1.19×10^{-4}MW, 4.3×10^{-5}MW, 3.27×10^{-6}MW, respectively. The total decay heat is dominated by the contribution from the structure for all time.

- **Radwaste Assessment**

The waste disposal rating (WDR) depends on the level of the long-term activation. For the Eurofer structure, ^{59}Ni ($T_{1/2}$=75ky), ^{93}Zr ($T_{1/2}$=1.5My), ^{94}Nb ($T_{1/2}$=20ky) are the main contributors. According to the US 10CFR61[4.2.2.1-7] regulation, if the waste contains a mixture of nuclides, then the waste disposal must meet the requirement of WDR<1. The main results of WDR for ^{59}Ni, ^{93}Zr, ^{94}Nb ,are 1.48×10^{-2}, 4.72×10^{-9}, 1.034×10^{-3}, respectively. The results show that the WDR values are very low (<<1), according to 10CFR61 Class C limits of waste disposal rating, these radwastes which come from HC-SB TBM can be qualified for Shallow Land Burial (SLB).

- **Tritium inventory**

Tritium will be bred in this HC-SB TBM in the solid breeding material and in the beryllium FW cladding. The tritium production rate for the solid breeder is estimated to be 0.033g/d under the ITER operation condition.

The accident and safety analysis, such as the E-M stress, LOCA and LOFA, etc., are almost completed. The results will be given in final DDD document.

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Table 4.2.2.1-1. Design parameters for the HC-SB TBM

Configuration	BOT (Breeder Out of Tube)	Modules: 3×3 Sub-modules
First wall area Neutron wall loading Surface heat flux	0.664m(W)×0.890m(H)	$0.591m^2$. $0.78MW/m^2$ $0.50MW/m^2$
Total heat deposition	NT-TBM, PI-TBM	0.76 MW
Globe TBR	Lithium orthosilicate, Li_4SiO_4	1.15(1-D), 80% Li-6
Tritium production rate	ITER operation condition	0.033g/d
Sub-module dimension	(P)× (T) × (R)	0.260m×0.190m×0.420m
Ceramic breeder (Li_4SiO_4)	Two size Thickness Max. Temperature	Diameter: 0.5~1mm, pebble bed 90mm (four zones) 737℃
Neutron multiplier (Beryllium)	Two size Thickness Max. Temperature	0.5~1mm, Pebble bed 200mm(five zones)+2mm(armor) 543℃ (Armor) 617℃ (Be Pebble bed)
Structure Material	Ferritic steel Max. Temperature	Eurofer 530℃
Coolant helium (He) Pipes size	Pressure Pressure drop Temperature range (inlet/outlet) Mass flow Diameter (OD/ID)	8 MPa 0.294 MPa 300/500℃ 0.73kg/s 85/80 mm
He purge flow (He)	Pressure Pressure drop	0.12MPa 0.02MPa

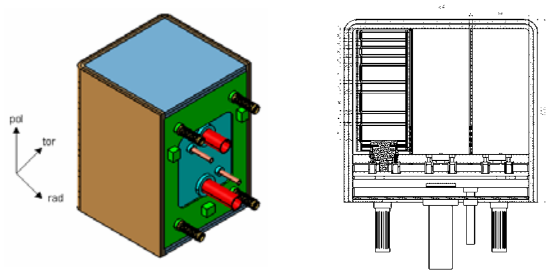


Fig.4.2.2.1-1 Schematic view of CH HC-SB TBM

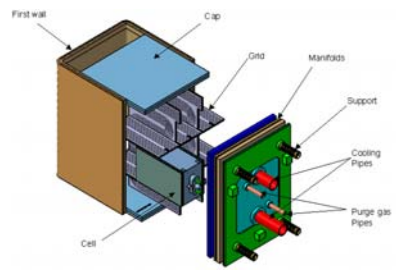


Fig.4.2.2.1-2 Exploded 3-D view of CH HC-SB

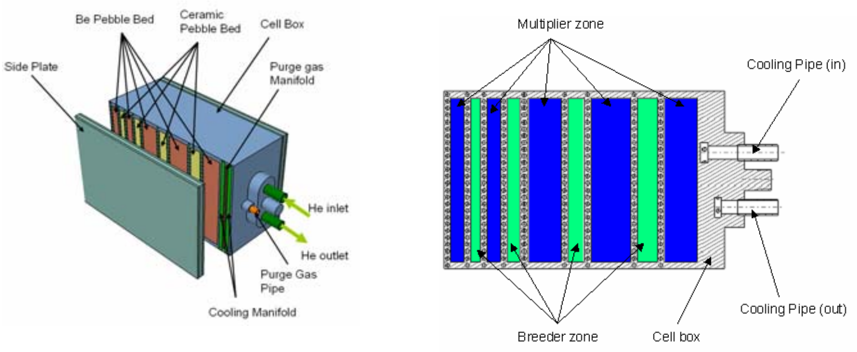


Fig.4.2.2.1-3 Schematic view of HC-SB sub-module

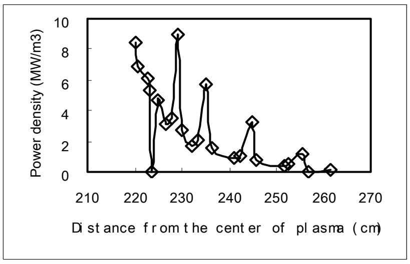


Fig.4.2.2.1-4 Power density with distance from the center of plasma in HC-SB module

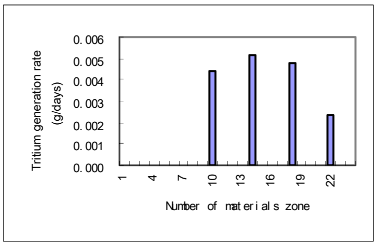
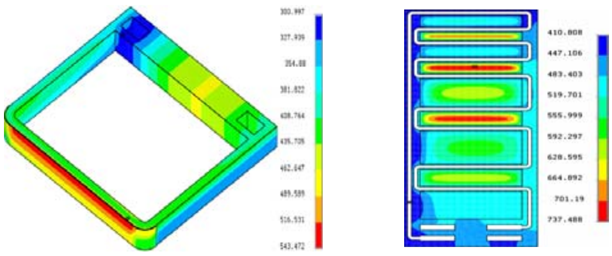
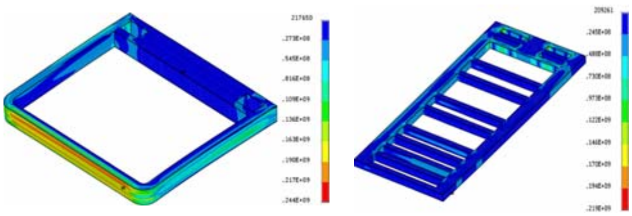


Fig.4.2.2.1-5 Generation rate of tritium vs. materials zone in HC-SB module



FW Sub-module

Fig. 4.2.2.1-6 Temperature distribution of the HC-SB test module



FW Sub-module (Single plate)

Fig. 4.2.2.1-7 Stress distribution of the HC-SB test module

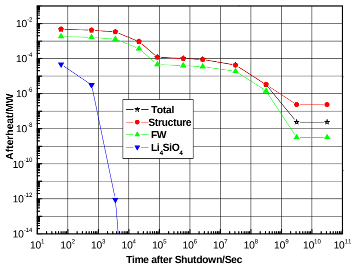


Fig. 4.2.2.1-8 Afterheat as a function of the time in HC-SB module

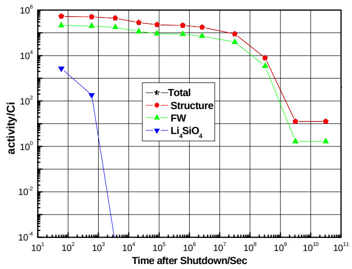


Fig. 4.2.2.1-9 Activity as a function of the time in HC-SB module

4.2.2.2 DFLL-TBM design and analysis

（1） System overall description

The DLL-TBM, as one of the consecutive DFLL-TBMs, consists of a 626 mm x 1832 mm x 476 mm steel box, reinforced by two radial-poloidal (rpSP) and six '˥' shape toroidal-poloidal (tpSP) stiffening plates, containing the self-cooled LiPb breeder / multiplier as schematically shown in Fig.4.2.2.2-1. The module box is formed by a U-shaped FW with radial/ toroidal/ radial He cooling channels. It is closed in the rear by a multi-shell Back Plate acting also as helium gas collector and distribution system.

The He coolant flowing scheme is shown in Fig. 4.2.2.2-2. Helium gas of 8 MPa, 300°C is delivered to the TBM through concentric pipes and exits at the outlet temperature of 410°C. The helium concentric pipe is separated into two branch pipes, which run to a heat exchanger located in the TCWS vault. The helium gas is mainly used to transfer the surface heat on FW and the volumetric nuclear heat in the TBM structure to the He/water heat exchanger.

The LiPb flowing scheme in DLL-TBM is shown in Fig. 4.2.2.2-1. The LiPb (1MPa) flowing in the poloidial direction with the inlet/outlet temperatures of 480°C /700°C is designed to carry the volumetric nuclear heat in the breeding zones to the LiPb/He heat exchanger which is installed in the transfer cask.

The main characteristics and reference design parameters of DFLL-TBM (a maximum surface heat flux up to 0.5 MW/m^2 for 10% of the FW area) are summarized in Table4.2.2.2-1. Design and analysis considering an average surface heat flux of 0.3 MW/m^2 is presented in the DDD report.

(2) Materials

In the design of DFLL-TBM, the RAFM steel (e.g. CLAM) is selected as the structural material assuming 300℃/550℃ as the lower/upper operation temperature limits. Considering the compatibility temperature limit of the RAFM steel with liquid LiPb, the operation temperature should be designed below 480℃ if no corrosion barrier is installed between them. Based on the progress in several RAFM steels being developed in the world, a preliminarily optimized design of chemical compositions of CLAM is proposed[Ref.4.2.2.2-1], The content of tungsten is set to 1.5wt%, which is higher than that of Eurofer97 (1.0%) and lower than that of F82H (2.0%). It is a compromise between the strength and the possibility of deposition of Laves phase in heat affected zone (HAZ) appearing during welding process. The 9.0wt% of chromium is selected to obtain the lower DBTT both before and after neutron irradiation. The testing with standard Charpy V impact samples show that the DBTT of CLAM (HEAT 040801) is about -100℃. Ultimate tensile strengths of CLAM at room temperature and 600℃ are 660MPa and 330MPa, respectively. These are similar to those of Eurofer97. Further optimization of compositions and heat treatments for CLAM is underway.

To obtain a good insulation performance, it commonly recognized that the product of the electrical resistivity and thickness of coating should exceed a value of ~100$\Omega \cdot cm^2$. To ensure complete coverage, a coating thickness of 5~10μm is considered desirable. Therefore, an electrical resistivity of 10^5~$10^6$$\Omega \cdot cm$ would be adequate. High tritium permeation reduction factor (TPRF>100～1000) and good compatibility with LiPb under operation temperature for coating are also required. Al_2O_3 is one of the candidate materials of coating based on current experimental results. Further investigation on their properties before and under neutron irradiation is greatly needed.

The SiC_f/SiC composite is a candidate material for FCIs. The most important requirements for SiC_f/SiC composite are low electrical conductivity, e.g. lower than 20~500 $(\Omega \cdot cm)^{-1}$, low thermal conductivity, e.g. lower than 2~5W/m·k, and good compatibility with LiPb at high temperature.

For pipes of the TBM auxiliary systems, 316 stainless steel and ferritic steels may be candidate materials. However, great effort should be paid to the material development of the high temperature LiPb/He heat exchanger.

(3) Neutronics analysis

The nuclear design calculations for DLL-TBM and SLL-TBM in ITER have been performed with the Monte Carlo code MCNP/4C and the nuclear cross-section data library FENDL-1 on the basis of the updated 3-D 20 degree torus sector model of ITER. The 3D views of the MCNP models of ITER and DLL-TBM converted from the existing MCNP input file and the CAD model by the code MCAM are shown in Fig.4.2.2.2-3.

MCAM is the Chinese home-developed CAD/MCNP interface code between commercial CAD softwares and MCNP, which can support various neutral CAD file formats, such as STEP/IGES, and MCNP input syntax. It can be used as a converter of large complex 3-D CAD models into MCNP models and vice versa as well as an analysis tool of MCNP models by the way of visualization to contribute the QA (Quality Assurance) of neutronic analysis.

MCNP-calculations were performed to obtain the neutron flux, tritium production and nuclear heating in DLL-TBM and SLL-TBM based on a normalized fusion power of 500MW. The tritium production by natural LiPb is about 1/3 of that by 90% Li-6 enrichment. The calculated local TBR (tritium breeding ratio) is 0.44 in DLL-TBM and 0.46 in SLL-TBM for 90% Li-6 enrichment in LiPb. The corresponding daily tritium production at the assumed duty factor of 22% (burn time 400s and repetition time 1800s) is 16.6mg/day in DLL-TBM and 17.3mg/day in SLL-TBM. The total volumetric nuclear heating amounts to some 0.57MW in DLL-TBM and 0.55MW in SLL-TBM.

(4) Thermal hydraulics and MHD analyses

For the helium flow circuit, the calculations and analysis of pressure drop were performed. Total pressure drop in the helium circuit is estimated as 0.79 MPa by analytical solutions, which amounts to about 10% of the circuit inlet pressure of 8 MPa. Accordingly, the pumping power for the TBM helium gas system is about 213KW, which assumes that the pumping efficiency is 80%.

The MHD effects are critical issues in DLL-TBM design and often pose a constraint to achieve higher performance due to additional pressure drop, flow redistribution and influence on heat transfer. FCIs are used as electrical and thermal insulators to reduce pressure drop and to allow LiPb outlet temperature as high as 700℃. To address the MHD pressure drop and flow features of LiPb coolant, numerical computations were performed considering two models of LiPb flow inside TBM and in manifold shown in Fig. 4.2.2.2-4.

The user-defined routine based on the commercial Computational Fluid Dynamic (CFD) software FLUENT is used to numerically simulate the LiPb flow in DLL-TBM. The code solves the governing equations in the fluid domain including interaction between flow field and electromagnetic field. The κ-εtwo-equation model of turbulence was applied in the analysis of MHD turbulent flows. The sink term standing for the Joule dissipation and the destruction term were added to the equations of turbulence kinetic energy (k) and its dissipation rate (ε), respectively.

The total MHD pressure drop of LiPb flow is about 0.10MPa from simulation results, including the MHD pressure loss (0.002MPa) in the straight poloidal channels and the 3D MHD pressure drop (0.098MPa) associated with the sharp contraction/expansion and manifold. Accordingly, the required pumping power of LiPb for the TBM is about 80W by using the equation, $P = \frac{1}{\eta}\Delta p \frac{\dot{m}}{\rho}$, which assumes that the pumping efficiency is 80%.

As shown in Fig4.2.2.2-5, the results of simulating LiPb flow in the inlet of manifold show magnetic field strongly influences the velocity profiles and flow redistribution. The velocities in three outlets of the manifold are non-uniform. The LiPb velocity in middle the outlet of manifold is less than that in two others.

(5) Mechanics Analysis

Temperature and stress analyses of DFLL-TBM were carried out using the finite element code ANSYS with the 3-dimensional poloidal section model with 4-channel helium coolant as shown in Fig.4.2.2.2-6. For DLL-TBM, this section is located near the bottom of the TBM, where the highest FW temperature is expected with LiPb breeder temperature rising along the flow direction. For SLL-TBM, this section is located at the top of the TBM, where the highest helium gas temperature in the r-p SPs reduces the heat transfer ability from LiPb to the structure.

For DLL-TBM, the maximum temperature of 536 °C and the maximum Von Mises stress of 400MPa are lower than the engineering limit for the structural material.

For SLL-TBM, the temperatures are everywhere lower than the upper temperature limit of RAFM steel. The maximum Von Mises stress is 552 MPa (> allowable 3Sm of 480MPa for

RAFM at 370 °C), which can be reduced by including rounding radius and more radial and toroidal SPs in LiPb flow channels in the stress analysis model. Further detailed analysis is necessary considering specific design models.

The resistance of the box in case of an accidental pressurization , which has been assumed that the rupture either of FW or SPs would imply the pressurization of the entire box to the He pressure of 8MPa, has been furthermore tested. Analyses showed that, according to the ITER Structural Design Criteria code [4.2.2.2-2], the box will be able to withstand this type of load.

Calculations of the eddy currents induced by disruptions were not included in the above analyses. Transient electromagnetic analyses of the blanket structure for plasma disruption events should be carried out to investigate the dynamic structural characteristics and the feasibility of the blanket structure for DFLL-TBM.

(6) Safety Analyses

- **Activation analysis**

Activation calculations were performed to assess the level of afterheat, dose rate and activity for DLL-TBM (IN-TBM) as a function of time after shutdown by using the code system VisualBUS [4.2.2.2-3] and the European Activation File EAF-99. The 8500 full power (high duty D-T operational phases) ITER pulses scenarios are accounted for. Neutron flux spectra were calculated by the Monte Carlo transport code MCNP/4C and the cross section library FENDL1.0. The total activity and afterheat are 6.8×10^{16}Bq and 0.018MW at shutdown, respectively. Assuming an extraction tritium efficiency of 95%, the activity level generated by the residual tritium in the TBM is 5.3×10^{13}Bq at shutdown. Referring to the Safety and Environmental Assessment of Fusion Power (SEAFP) strategy for the management of activated materials, all the activated materials of DLL-TBM may be recycled with a cooling time not exceeding 50 years and can be recycled by hands-on operation after 100 years cooling except for the Al_2O_3 coating.

- **Tritium inventory and permeation analysis**

The main purpose for tritium inventory and permeation analysis are: (1) to estimate the tritium inventory and permeation rates for the DFLL-TBM cooling systems; (2) to analyze the factors which will affect tritium permeation, such as the percentage of LiPb flow in TES (F), Tritium Permeation Reduction Factor (TPRF) and so on. A schematic of tritium permeation model is shown in Fig. 4.2.2.2-7.

The following assumptions are made in the tritium inventory calculations: (1) About 10% of LiPb flow in the loop is circulating into TES and 0.1% of He flow in the He loops into CPS; (2) The tritium extraction efficiency is assumed to be 95%; (3) The TPRF is 10 in TBM and 100 in auxiliary system.

The results show that ~1.038 g-T is produced annually (3000pulse/yr) in DLL-TBM. About 80% tritium can be recovered from LiPb, 10% from primary He coolant and 5% from secondary He coolant respectively. 3% tritium permeates through the TBM walls into the ITER VV, 1.0% permeates through LiPb piping, only 0.1% tritium permeates through He piping, the remaining 0.9% tritium resides in the TBM systems. The tritium permeation rate into the ITER heat removal system through the TBM He gas/water heat exchanger is ~0.04mg-T/a as HTO, which is much lower than the total TBM annual release target of 1mg-T/a as HTO. After 3000 pulses, the total inventory of tritium calculated in TBM systems is 51mg, with 1.5mg in LiPb, 49.5mg in structure of TBM and auxiliary system.

Fig. 4.2.2.2-8 shows the effect of F and TPRF in TBM on the tritium permeation into primary He coolant by keeping the TPRF in auxiliary system be 100. It is clear that the most important factor influencing the tritium permeation is F. When TPRF is 50, the tritium permeation will change slowly with the increasing of TPRF in TBM. Thus, it is reasonable to choose F as 10% and TPRF as 50 in the DFLL-TBM tritium system design.

- **Severe Accident Analysis**

The decay heat removal capacity must be demonstrated for DFLL-TBM in order to meet ITER safety requirement. A 1-D heat transfer model with finite element code ANSYS has been developed to analyze the temperature response of DLL-TBM after shutdown of ITER. It represents a radial unit cell column cut of the TBM from the FW all the way through the breeding zone, manifold region, back plate, shield and VV. In this analysis, a complete loss of all active cooling systems for the TBM and ITER, and only radiation heat transfer is assumed between the back of the TBM and the shield and between the shield and the ITER VV. The ITER VV cooling system is assumed to operate in natural circulation mode, maintaining the VV temperature at 135^{o}C. The emissivity of the steel is taken as 0.3. The initial TBM temperature is assumed to be 500^{o}C. The calculated results indicate passive decay heat removal is assured, resulting in long term TBM temperatures less than 240^{o}C after 10 days. The decay heat removal capacity of DLL-TBM is less than that of SLL-TBM due to the weak thermal conductivity of the SiC FCI in DLL-TBM. The decay heat removal capacity of DFLL-TBM can meet the ITER safety requirement.

Possible hydrogen sources were examined in this analysis. It is concluded that the maximum quantities of hydrogen produced under accident conditions is less than the ITER limit of 2.5 kg. The results also demonstrate that the pressurization of the VV, TBM test cell and TCWS vault induced by the helium coolant system in case of accident does not pose a serious threat to these confinement structures.

References

[4.2.2.2.-1] Qunuying HUANG et. al., "Overview on the Development of Low Activation Martensitic Steels for Fusion Reactor", Chinese J. of Nuclear Science & Engineering Vol.24 No.1 (2004) p.56-64.

[4.2.2.2-2] ITER structural design criteria for in-vessel components(ISDC); SECTION B: In-Vessel COMPONENT; ITER IDoMS S74MA 197-12-12 R 0.2; ITER document IRB. M51; 1998.

[4.2.2.2-3] Gao Chun-jing，et al.，Integral Data Test of HENDL1.0/MG and VisualBUS with Neutronics Shielding Experiments (I) [J], Plasma Physics and Technology, 2004,6(5): 2507~2513

Table 4.2.2.2-1 **Main characteristics and design parameters of the DFLL-TBM**

	DLL-TBM	SLL-TBM
Heat Flux	Ave.0.3MW/m^2, Max. 0.5 MW/m^2	
Neutron Wall Load	0.78 MW/m^2	
Structural material	China Low Activation Martensitic steel (RAFM Steel)	
TBM dimensions	Pol. 1832 mm × Tor. 626 mm × Rad 476 mm (without external headers) Gap TBM/Frame = 20 mm	
Total deposited power	0.92MW	0.89MW
Coolant	He: Tin/out = 340/420 °C; Pin = 8 MPa; Q_{tot}=1.69kg/s	He: Tin/out = 340/420 °C; Pin = 8 MPa; Q_{tot}=2.15kg/s
First Wall	U-shape; Toroidal He cooling; 4 paths; Thickness: 30 mm (5/15/10) Cooling channels: (15 x 20)mm^2, pitch 25 mm	
	$T_{in/out\ He}$ = 340/404 °C; V_{He} = 53 m/s	$T_{in/out\ He}$ = 340/400 °C; V_{He} = 67 m/s
Stiffening plates	Thickness: 10 mm (3/4/3); Cooling channels: (4 x 8)mm^2, pitch 11 mm	
'˥' type tpSP; rpSP	V_{He} =49m/s; $T_{in/out\ He}$ = 404/417 °C	V_{He} =62m/s; $T_{in/out\ He}$ = 400/420 °C
Covers	Thickness: 32 mm; 8 parallel cooling channels; (8 x 16) mm^2, pitch 17.5mm	
	V_{He} =49m/s; $T_{in/out\ He}$ = 404/414 °C	V_{He} =62m/s; $T_{in/out\ He}$ = 400/416 °C
He collector	3-stage collector; radial direction size: 20/20/10/20/10/20/20 mm	
Breeder/multiplier LiPb	3 rows poloidal flowing channel	
	V_{LiPb}=20/11/5 mm/s ; $T_{in/out}$ = 480/700 °C	~1mm/s; $T_{in/out}$ = ~/450 °C

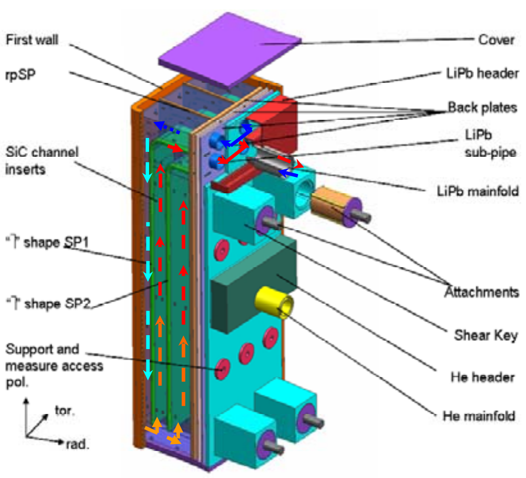


Fig4.2.2.2-1 3D Structure View of DFLL-TBM

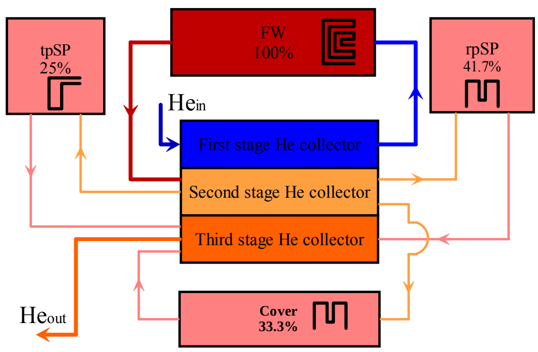


Fig4.2.2.2-2 He flowing scheme in DFLL-TBM

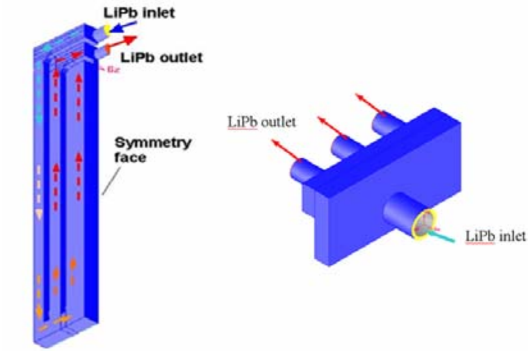


Fig4.2.2.2-4 Simulation model of LiPb MHD flow inside DLL-TBM and in manifold for DFLL-TBM

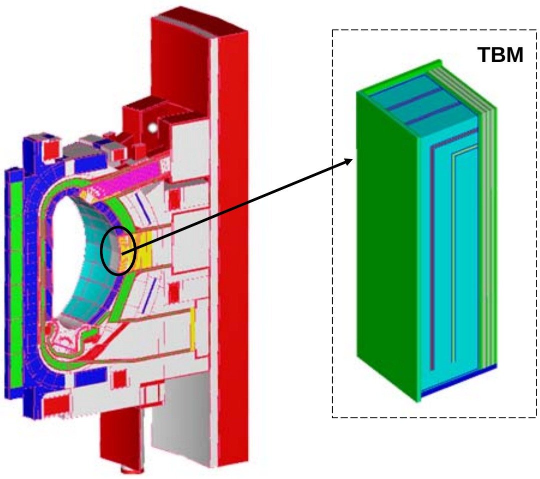


Fig.4.2.2.2-3 3D views by MCAM of ITER and DLL-TBM

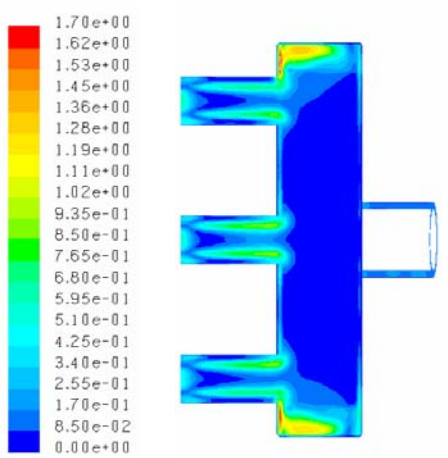


Fig4.2.2.2-5 Velocity distribution of LiPb flow in the inlet manifold of DFLL-TBM

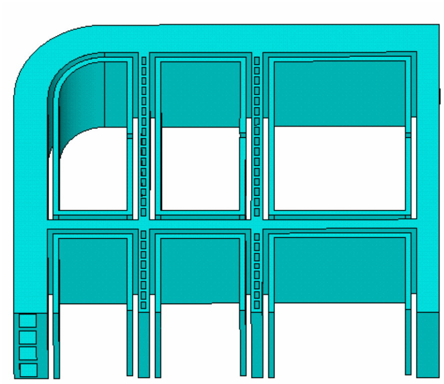


Fig.4.2.2.2-6 3-D analysis model for DFLL-TBM

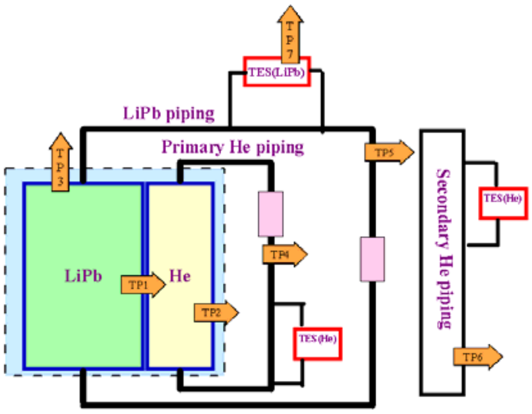


Fig. 4.2.2.2-7 Tritium permeation model of DFLL-TBM

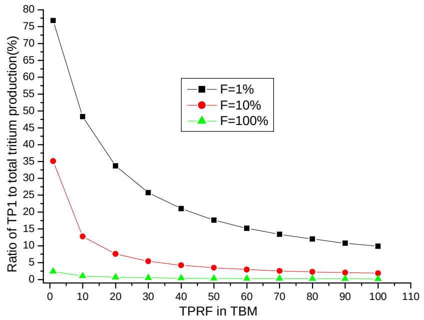


Fig. 4.2.2.2-8 Tritium permeation into primary He coolant of DFLL-TBM

4.2.3 TBMs ancillary systems

4.2.3.1 HC-SB TBM ancillary systems

HC-SM TBM requires several ancillary systems for supporting its experiment. The description of these ancillaries are listed in the follows.

(1) Helium cooling system (HCS)

The cooling system includes the primary helium circulation loop with all components and the secondary heat removal loop. The secondary loop employs water with the temperature of 35°C as coolant which is supplied by the ITER tokamak cooling water system (TCWS). The cold water at 35°C is heated in the main heat exchanger by absorbing the heat transferred from the hot helium of the primary loop. The hot water at the outlet of the main heat exchanger possesses the temperature of below 75°C and then flows back to the TCWS. The pressure of the secondary water loop is lower than 1MPa. The cooling system is installed in the TCWS vault, approximately 70m away from the TBM. The pipelines have to run about 18m horizontally, 14m vertically within the shaft and again 60m horizontally plus 10m between components. To mitigate thermal stresses due to high temperature operation conditions, it results in a total length for the hot leg and cold leg of about 100 m and 95 m, respectively.

It has been acknowledged that there will be no crane available for vertical component handling. A minimum net foot print size of 16 m^2 and approximately 5m height will be needed for housing the CH HC-SB TBM helium cooling system. The TBM cooling subsystem consists of a single main loop. Since the decay heat removal was demonstrated to be achievable by passive heat dissipation without relying on a residual heat removal loop. Besides the main components, a number of tanks, a shelf for pressure control equipment and a chamber for local electrical equipment have to be accommodated.

The thermal power from the CH HC-SB TBM is removed to the ITER secondary cooling water which is designed to work under the conditions of about 1MPa pressure, 35°C /75°C inlet/outlet temperature referenced to the main heat exchanger, and about 4.95 kg/s in mass flow rate. The helium cooling subsystem is designed for the extreme operating conditions with 0.78MW/m^2 neutron load and 0.5MW/m^2 surface heat flux. The Table 4.2.3.1-1 contains a set of values belonging to the specified load conditions of the TBM taking into account some load uncertainties. The basic design requirement is that the cooling subsystem must possess the capability of adjusting the flow rate so that the TBM internal temperatures can be bounded within the safety limits specified to the TBM material. A flow chart of the HCS is shown in Fig.4.2.3.1-1.

The most demanding conditions for component layout pertain to the TBM in terms of temperature (up to 500°C) limit, and result from the pressure drop induced by mass flow rate passing through the TBM. It should be noted that the conditioning of the TBM for Tritium release at 500°C during shutdown phase requires additional heating, which is possible to operate at reduced pressure.

(2) Coolant purification system (CPS)

For the cooling system of the test blanket modules a purification system should be provided to purify a fraction of the helium coolant stream, i.e. to extract hydrogen isotopes as well as solid, or gaseous impurities. The CPS is connected to the HCS loop at both sides of the circulator of HCS in parallel via small pipes delivering a small by-pass flow. A fraction of gas ingresses the CPS through partial tube valve, then back to the HCS loop after tritium and other impurities are removed using the catalysis oxidation and molecular sieve adsorption methods. The main design parameters of the system are listed in Table 4.2.3.1-2. A flow chart of the CPS is shown in Fig.4.2.3.1-2.

(3) Tritium extraction system (TES)

Tritium extraction system (TES) will be placed in T-Plant building of the ITER device Main design parameters for the TES subsystem are shown in Table 4.2.3.1-3.

The main function of the TES is to extract tritium produced in TBM by purge gas, to control the gas composition of the low pressure purge flow, to provide tritium-breeding data of TBM by measuring and monitoring operation parameters of the system (temperature, pressure, gas components, gas flow rates, tritium concentration), and to test the reliability of TES.

A flow diagram and layout of the tritium extraction system is shown in Fig.4.2.3.1-3. The instruments for process control like sensors for temperature, pressure, flow rate, etc. are not included in this figure. The key components of the system include cold trap, cryogenic molecular sieve beds, Pd/Ag diffuser, gas circulation pump, hydrogen isotope separation subsystem, hydrogen storage beds, heat exchangers etc. The TES design has following features: firstly, a hot Mg bed is used to decomposed HTO released during regeneration of the MSB beds for tritium recovery in HT; secondly, a small size ISS system is designed to separate product HT of the TES for H_2 recycle, thus greatly reduce the amount of the discharged waste H_2.

(5) Space requirement and locations

The space requirement of HC-SB TBM all ancillary system have been designed. The detailed dimension can find in the final DDD report and [4.2.3.1-1].

- **Helium cooling system (HCS):**

Helium cooling system (HCS) is located in the ITER TCWS vault. The control panel for operation/monitoring of the helium cooling system will be installed in the main control room requiring a space of approximately 12m^2. Small electrical control equipment, such as signal transducers, are planned to be placed inside the transfer cask in which some equipments of the tritium monitoring and extraction subsystem should also be housed [4.2.3.1-2].

- **Coolant purification system (CPS)**

The system is housed in a separate glove box. A space of 1.5m×1.2m×2.2m (L×W×H) is needed for its assembly, maintenance and operation of the CPS. Additional area of 3m^2 is needed for the remote control station and data acquisition instruments. Electrical power, helium, noble gas for glove box, gas for pneumatic valves and wasted gas process system, etc. are of necessity for running and maintenance of the CPS. This system will be located in the TCWS vault in the proximity of the HCS system [4.2.3.1-3].

- **Tritium extractionsystem (TES)**

Tritium Extraction System is installed in the tritium plant building supplied by ITER. It will be housed in a glove box with the dimensions of 5.5m×1.2m×5.5m (L×W×H). Additional space of about 35 m^2 is needed for a control station, electrical cabinets, elevating equipment, and for the working area and the maintenance. It should be pointed out, that a forklift and an elevator platform for maintenance of the TES at the upper position in the high glove box are necessary. The dimensions of pipes and penetrations through the walls of the TES glove box have been estimated in the [4.2.3.1-3]

Reference:

[4.2.3.1-1] K.M. Feng, et al., Design Description Document for the Chinese Helium Cooled Solid Breeder (CH HC-SB) TBMs, (Draft Design Report), Dec. 30, 2004.

[4.2.3.1-2] Z.W.Zhou, Y.W.Yang, R.Q.Jing, HC-SB TBM Heilum Cooling System (HCS) Design Description Report, Dec.15, 2004.

[4.2.3.1-3] D.L.Luo, C.A.Cheng, et al., HC-SB TBM Coolant Purification Systems and Tritium Extraction System Design Description Document, Dec. 15, 2004.

Table 4.2.3.1-1 Operating conditions of the HCS system for HC-SB

Parameters	Unit	Specified Values
Total heat to be removed	MW	0.79
Primary coolant		Helium
Temperature at module in/out	°C	300/500
Pressure	MPa	8
Number of circuits		1
Mass flow rate	kg/s	0.73
Secondary coolant		Water
Temperature at heat exchanger in/out	°C	35/75
Pressure	MPa	<1
Number of circuits		1
Mass flow rate	kg/s	4.95

Table 4.2.3.1-2 Design Parameters for the Coolant Purification System of HC-SB

Parameters	Predefined value
Pressure of helium coolant	8 MPa
Maximum flux of CPS	450 mg/s
Partial pressure at inlet	
P (H_2)	<100 Pa
P (HT)	<0.3 Pa
P (Q_2O)	<35 Pa
Tritium extraction efficiency	<95 %
Temperature at inlet/outlet of CPS	300^{O}C/500^{O}C

Table 4.2.3.1-3 Main design parameters for the TES of HC_SB

He Mass Flow	0.85g/s
Swamping Ratio	He: H2=1000:1
Pressure of purge gas at TBM inlet at TBM outlet Pressure drop in TBM	 0.12Mpa 0.10Mpa 0.02Mpa
Temperature of purge gas at TBM outlet	300 ^{O}C
Tritium Generation Rate	0.033g/d
Tritium Extraction Efficiency	≥95%

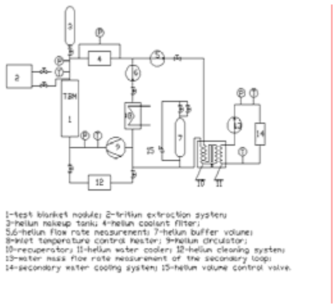
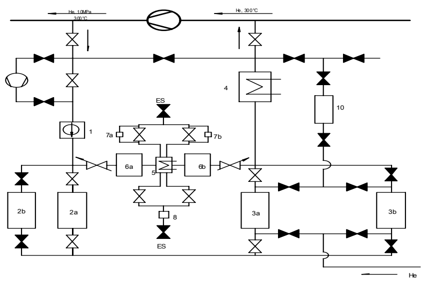
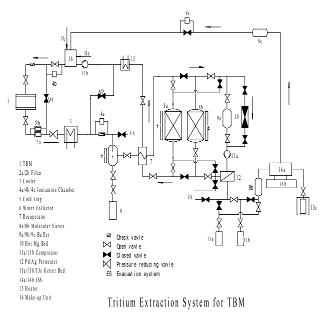


Fig.4.2.3.1-1 Flow diagram of the HCS Sub-system

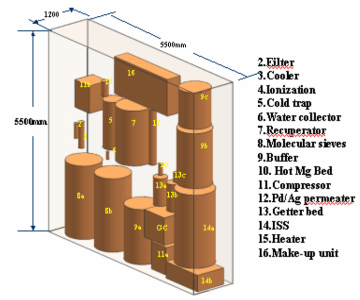


1 - gas flow controller; 2a/2b - impurities getter bed; 3a/3b - tritium getter bed; 4 - heater; 5 - cooler; 6a/6b - buffer bank; 7a/7b - ionization chamber; 8 - gas chromatograph 9—circulator; 10—ZrCo bed

Fig.4.2.3.1 -2 Flow chart of the Coolant Purification System of HC-SB



(1) Flow chart of TES system



(2) Layout of TES system

Fig.4.2.3.1-3 The Flow and layout of Tritium Extraction System (TES) for HC-SB

4.2.3.2 DFLL TBM ancillary systems

(1) Lithium lead system

LiPb auxiliary system is an important external loop equipment of TBM for liquid LiPb blanket. According to the testing strategy of DFLL-TBM which is designed to test the technologies of the SLL and the DLL blankets, a conceptual design of the LiPb auxiliary system is performed to support the dual functional blanket design. The auxiliary system can be used to achieve the recovery of tritium, heat rejection, monitoring of impurity in LiPb, LiPb purification and so on.

The flow diagram of the LiPb circuit is shown in Fig. 4.2.3.2-1. Different from the SLL operating mode, the flow rate of liquid LiPb in the DLL mode is much higher. The LiPb auxiliary system not only has the functions of tritium extraction and impurity elimination, but also can transfer thermal power from LiPb to helium gas in the secondary circuit.

The LiPb auxiliary system for DFLL-TBM has the unique features such as dual functions combining SLL and DLL testing, in-situ impurity monitoring to eliminate impurity in time; more efficiently removing impurity in LiPb with a magnetic trap.

In addition, the high LiPb exit temperature (~700°C) in the DLL mode indicates the following problems: selection of the structural material for the LiPb exit pipes and the heat exchanger, and the tritium permeation. A solution to these issues is proposed by designing the primary loop pipes as concentric tubes with the hot outlet fluid flowing in the centre tube and the cold inlet fluid in the annulus. In this case, both tubes are cooled by the inlet flow together with a thermal insulator (e.g. SiC_f/SiC) installed inside the inner tube. Consequently, the temperature of both steel tubes can be maintained below 500°C. That will considerably reduce tritium permeation losses into the building atmosphere.

To avoid using advanced materials to handle high outlet temperature of the LiPb loop, a bypass loop system may be considered. Hot LiPb (700°C) returning from TBM is mixed with the bypassed cold LiPb (480°C) at the bypass section, which results in only a warm stream going to the tritium extraction and heat exchanger systems. In this way, the high temperature features of the TBM, especially the function of the FCIs as a thermal insulator at high temperature, can be tested in ITER without requiring high temperature materials in the tritium extraction and heat exchanger systems.

(2) Helium coolant system

The helium cooling system supports not only DLL-TBM testing, but also SLL-TBM testing. This system is located in the TCWS vault, with four pipes connecting to the TCWS which has been designed to supply cold water with a temperature of 35°C and accept hot water with a temperature of 75°C. This cooling system consists of two helium loops, one of which, the primary helium loop, cools the first wall/structure and removes 0.94MW thermal power during the DLL-TBM test. The secondary helium loop exchanges 0.21 MW thermal power with LiPb. During the SLL-TBM test, the primary loop will remove the whole nuclear heat (~1.12MW). The two helium cooling loops will share some equipments such as pressure control and helium purification sub-systems. The flow diagram of the helium cooling systems is schematically shown in Fig. 4.2.3.2-2.

(3) Helium coolant purification system

The Coolant Purification System (CPS) is a subsystem for the Helium Cooling System (HCS). It is designed to extract tritium which permeates into coolant from FW and the breeder zones. Besides tritium extraction, this system is also proposed to remove any additional gaseous, solid impurities. The flow diagram of TES is shown in Fig4.2.3.2-3. The main design parameters of the system are listed in Table 4.2.3.2-1.

Tritium in the helium coolant is removed by catalytic oxidation and molecular sieves absorption method. HTO and T_2O adsorbed in molecular sieves bed are treated by the water gas shift reaction and Pd-Ag membrane separation methods. The tritium is recovered by ZrCo alloy. ZrVFe alloy is used to remove other impurities like O_2, N_2, and CO_2 etc. The key components of the system include catalytic oxidizer, molecular sieve bed, water gas shift reactors, circulator, tritium reservoir, ionization chamber, gas chromatograph, etc.[4.2.3.2-1]

(4) Tritium Extraction System

The main function of the Tritium Extraction System (TES) is designed to extract the tritium produced in liquid LiPb by purge gas, and to control the tritium content of the purge gas. The bubble column is a gas-liquid counter-current contactor. The tritium is removed from the LiPb as it moves from the liquid LiPb into the gaseous helium. The helium/tritium mixture leaves the separator and is sent to the tritium plant. The flow diagram of TES divided into two parts, which are installed in the transporter and the tritium building respectively, is shown in Fig4.2.3.2-4 [4.2.3.2-2].

(5) Space Requirement and Allocations

- **Piping Layout**

The DFLL-TBM system needs half space of the transporter. The equipments of the LiPb loop (e.g. LiPb/He heat exchanger, dump tank, expansion tank, cold trap unit, etc.) and of the tritium extraction sub-loop (e.g. purging bubble tank, heater, blower, etc.) are installed in the half space of the transporter. Fig 4.2.3.2-5 shows arrangement of the testing port.

Four pipes connecting TBM to the transporter for cooling and diagnostics are designed to penetrate the two barriers i.e. the bio-shield and the seal. The 1st pipe (concentric, reference outer pipe o.d. 120 mm) carries LiPb with the hot liquid running in the inner tube. The 2nd pipe (concentric, reference outer pipe o.d. 120 mm) of which carries helium coolant for cooling the FW/structure and the breeder. The 3rd pipe (reference o.d. 100 mm) is used to drain the breeder in case of emergency. The 4th pipe (reference o.d. 80 mm) is used for diagnostics.

Four helium pipes (reference o.d. 100 mm) of approximately 90~100 m run from DFLL-TBM to He Cooing System located in the TCWS. Two purge gas pipes (reference o.d. 15mm) running between the transporter and the glove box in the tritium building are designed for TES, and four bypass pipes (reference o.d. 15 mm) in helium loops are designed to the glove box in the tritium building for CPS.

The layout of above pipes including dimensions and numbers, etc. needs to be further optimized.

- **TCWS building layout**

A space of 16.6m× 7.3m × 5m(H) is assigned in the TCWS vault for all TBM cooling systems. A minimum space of 6m (L) ×3.6m(W) × 5m(H) is needed for the equipments of the DFLL structure-He cooling loop and the Li-He heat exchange loop, the helium cooling loops layout in the TCWS vault is shown in Fig.4.2.3.2-6. The control panel for operation/monitoring of each loop requires a space of approximately 4 m^2.

- **Tritium building layout**

According to the ITER design, eight glove boxes for tritium extraction system will be installed in the tritium building, and each one is 3m (L) x 1m (W) x 2.7m (H). According to current primordial designs of TES and CPS for DFLL-TBM, Two glove boxes are needed for accommodating the components of TES and CPS, respectively. The number of required glove boxes would be reduced if the TES and the CPS are further optimized and combined.

References

[4.2.3.2-1] Wang He-yi, et al., DFLL-TBM Tritium Extraction System Design Description Document, 2005.05

[4.2.3.2-2] Luo Del, Cheng ChangAn, Huang ZhiYong, et al., DFLL-TBM Coolant Purification Systems Tritium Extraction System Design Description Document, 2005.0

Table 4.2.3.2-1. Main Design Parameters for the Coolant Purification System of DFLL-TBM

Parameters	Predefined value
Pressure of helium coolant at inlet	8MPa
Operation pressure	2MPa
Maximum flux of CPS	2 g/s
Partial pressure at inlet P(T_2)	<0.15 Pa
Tritium extraction efficiency	≥ 95%
Temperature of coolant at inlet/outlet of CPS	573K/573K

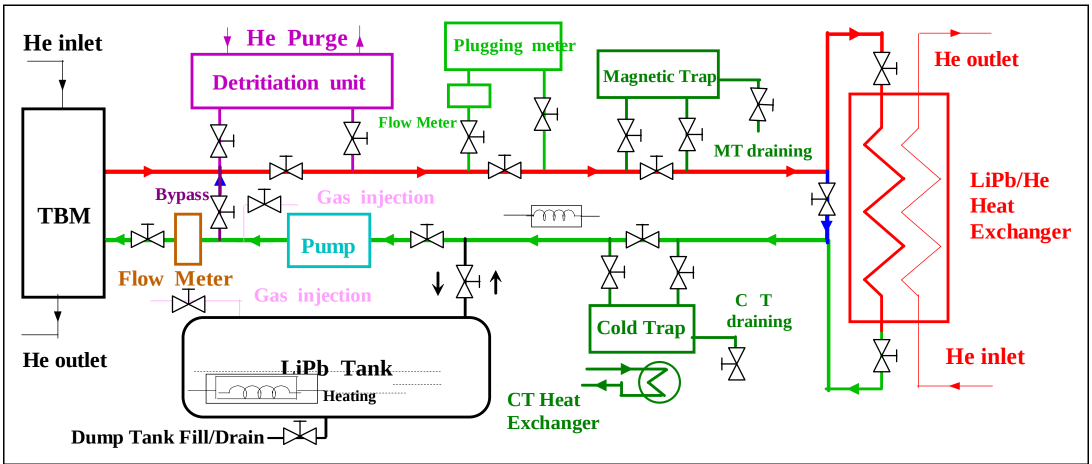


Fig. 4.2.3.2-1 Flow diagram of the LiPb auxiliary system for DFLL-TBM

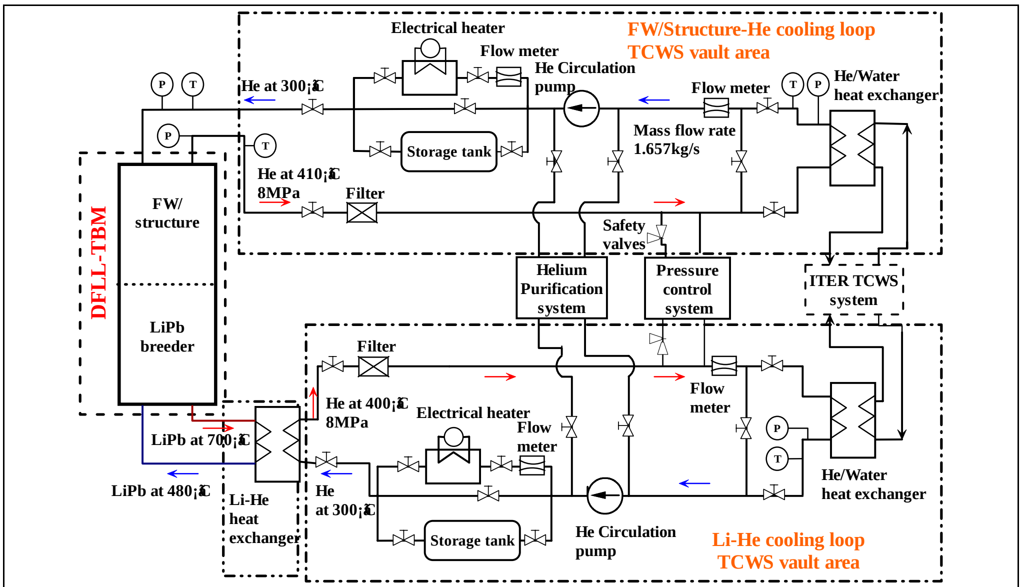


Fig. 4.2.3.2-2 The flow diagram of the helium cooling systems for DFLL-TBM

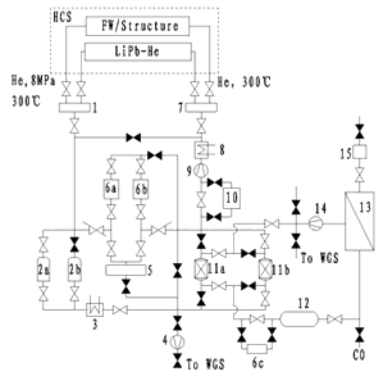


Fig.4.2.3.2-3 Flow diagram of CPS for DFLL-TBM

1.Diffluence Gas flow controller 2.Catalytic oxidizer 3.Cooler 4.Vacuum pump 5.Gas chromatograph 6.Ionization chamber 7.Refluence Gas flow controller 8.gas heater 9.blower 10.ZrVFe bed 11.Molecular sieve bed 12. Relief tank 13. Water gas shift reactor 14.circulator 15.ZrCo bed

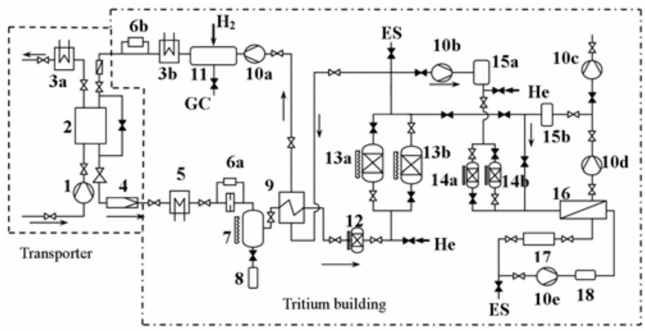


Fig. 4.2.3.2-4 Flow diagram of TES for DFLL-TBM

1.Blower 2. Purging bubble tank 3.Heater 4. Filter 5.Cooler 6.Tritium Monitor 7.Cold trap 8.WaterCollector 9.ecuperator 10.Compressor 11. Make-up unit 12.Adsorption implement 13.Fisrt step of GC column 14Second step of GC column 15.Buffer 16. Pd/Ag diffuser 17.Getter bed 18.He tank ES Evacuation Check valve Open valve Closed valve Pressure reducing valve

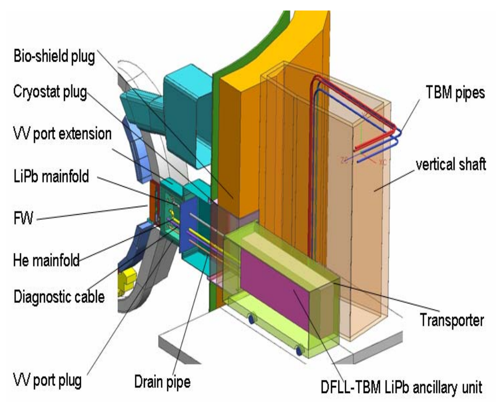


Fig. 4.2.3.2-5 Test port general arrangement for DFLL-TBM

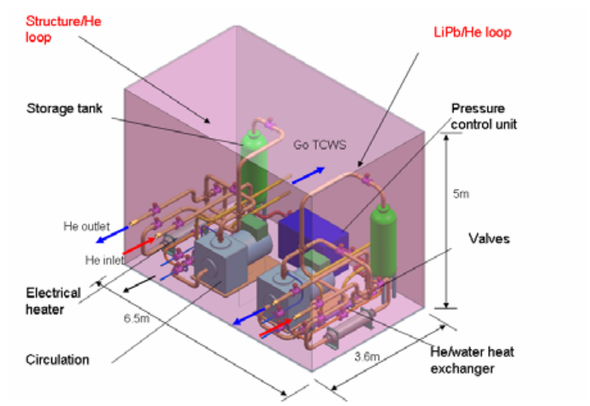


Fig. 4.2.3.2-6. Layout of the helium cooling loops in the TCWS vault for DFLL-TBM

4.2.4 Supporting Activities to other Parties TBMs

4.2.4.1 HC-SB TBMs Supporting Activities to other Parties TBMs

(1). China has studied tritium-processing technology supported by national fusion program for many years. Knowledge accumulated in this field is useful for the HC-SB TBM tritium work. China looks for cooperation with other parties on solid breeder, tritium extraction and hydrogen isotopes separation technologies.

(2). China has built a high temperature gas-cooled reactor (HTGR). The technologies and experiences gained in HTGR project are useful for the design and test of HC-SB TBM. China also hopes to strengthen cooperation with other parties.

(3). A High Temperature He Experiment Loop (HTHEL) with 700 ^{O}C and 8 MPa, which is useful for HC-SB TBM design and R&D activities, is proposed to be built in China. China also seeks cooperation with other parties on this issue.

4.2.4.2 DLL-TBMs Supporting Activities to other Parties TBMs

China intends to share the necessary R&D activities related to liquid lithium-lead blankets, including mock-up fabrication and validation program prior to the ITER operation and testing program in ITER, with other parties of mutual interest. The critical issues related to liquid lithium-lead blanket includes at least technologies of liquid LiPb experimental loop, FCI, insulation coating and the RAFM steel (e.g. CLAM steel) etc. The EAST tokamak can be open internationally for testing of TBMs (at least EM-TBM and possibly NT-TBM) from other parties.

4.2.5. Supporting R&D and Validation Program prior to the installation in ITER

4.2.5.1 HC-SB TBM Supporting R&D and Validation Program

(1) TBMs Basic objectives for R&D [4.2.5.1-1&2]

The following two goals necessary to design the HC-SB Demo-blanket system should be achieved: 1) to verify blanket design under the integrated load conditions; 2) to verify integrated performances of the blanket system and its reliability. Specific issues to be verified and/or evaluated for the HC-SB TBM are:

a) Tritium production consistent with neutron spectrum;

b) High grade heat generation relevant to the electricity generation;

c) Thermo-mechanical behavior of the module with pebble beds under surface heat load and neutron irradiation;

d) Tritium release characteristics from the pebble bed;

e) Tritium permeation to the coolant.

Therefore, different kinds of components, sub-modules of HC-SB TBM should be tested in the out and in -of-pile.

Out-of-pile Test：

- Small-medium scale mock-up tests
- Pebble bed thermo-mechanical test
- First wall heat removal test and thermal cycle test
- Prototype TBM mock-up test (scale 1:4 or larger)
- Thermal-hydraulic test
- Heat removal test and thermal cycle test
- TBM check-out with the auxiliary systems prior to ITER installation

In-pile Test：

- Tritium release behavior and required sweep gas conditions
- Pebble bed thermo-mechanical behavior under neutron irradiation

Material Test：

- Reduced-activation ferritic steel (CLAFM)
- Tritium breeder (Li_2TiO_3, Li_2O, Li_4SiO_4)
- Neutron multiplier (Be pebble bed)
- Tritium permeation barrier

(2) HC-SB TBMs R&D and components manufacture schedule

A detailed test plan for further operation of every operation phase has not developed yet because it will depend on the final test arrangement, the ports allocation and the time schedule from the ITER TBWG. A draft schedule on R&D and components manufacture is listed in

References

[4.2.5.1-1] C.H. Pan, Definition and development strategy of DEMO fusion reactor in China, Presented at TBWG-12 meeting, ITER Naka Joint working site, Mar. 11-14, 2004.

[4.2.5.1-2] K.M. Feng, et al., Design Description Document for the Chinese Helium Cooled Solid Breeder (CH HC-SB) TBMs, (Draft Design Report), Dec. 30, 2004.

Table 4.2.5.1-1HC-SB TBM Design and R&D Schedule

Phases	now	-9	-8	-7	-6	-5	-4	-3	-2	-1	**Day one**
Design & Analysis											
Conceptual Design (DDD)											
Detailed Design											
Engineering											
R&D Phase											
Breeder and Be development											
Structure material development (FM)											
Components fabrication & technology											
Components testing											
Tritium technology											
Mock-up testing phase											
Mock-up testing											
Prototype TBM testing											

4.2.5.2 DFLL TBM Supporting R&D and Validation Program

The common R&D for the DLL and the SLL concepts will be performed in the following areas: development of RAFM steel (e.g. CLAM steel) and its fabrication technology, testing of full mechanical performances, development of coating and FCI technology including testing of combinability of coating material and CLAM steel, tritium extraction and control technology from LiPb and Helium gas circuits, diagnostic instruments and ancillary systems.

A small-scale LiPb loop with thermal convection has been developed to investigate the LiPb compatibility with structure materials. A forced convection LiPb loop will be built to test MHD effects and related flowing performances under out-of-tokomak conditions. A middle scale (1/3) LiPb loop and relevant components need to be developed for (1/3)-sized TBM testing in EAST before a full scale testing system is developed for full-sized TBM testing in ITER.

For the DLL concept, necessary R&D for FCI materials as thermal and electric insulators is needed, including test the fabrication techniques, thermo-physical properties and compatibility of FCI material with LiPb under high temperature conditions.

4.3 European Union Proposals

4.3.1 - DEMO studies and Testing Strategy in ITER

The European Union supports two designs of DEMO Blanket, the "Helium Cooled Pebble Bed", HCPB, and the "Helium Cooled Lithium Lead", HCLL. Both blankets have been studied in the recent EU Power Plant Conceptual Study [4.3-1], are candidate for the Demo reactor and have been selected for the test in ITER.

4.3.1.1 – <u>The Helium-Cooled Pebble-Bed (HCPB) blanket</u>

4.3.1.1.1 - HCPB DEMO design

The Design of a HCPB blanket has been improved in the last years [4.3-2, 4.3-3] in its structural and thermo-hydraulic configuration: a robust box reinforced by a grid and able to withstand the full pressure of the coolant Helium (8 MPa) in case of in-box LOCA, a radial cooling for the box components (except the FW that is cooled toroidally), grid and Breeder Units (BUs) with the high pressure manifolds integrated in the back plate structure, BUs arranged in a modular array in the space defined by the stiffening grid and a design of the BUs based on Cooling Plates (CP) for extracting the heat from the breeder/multiplier.

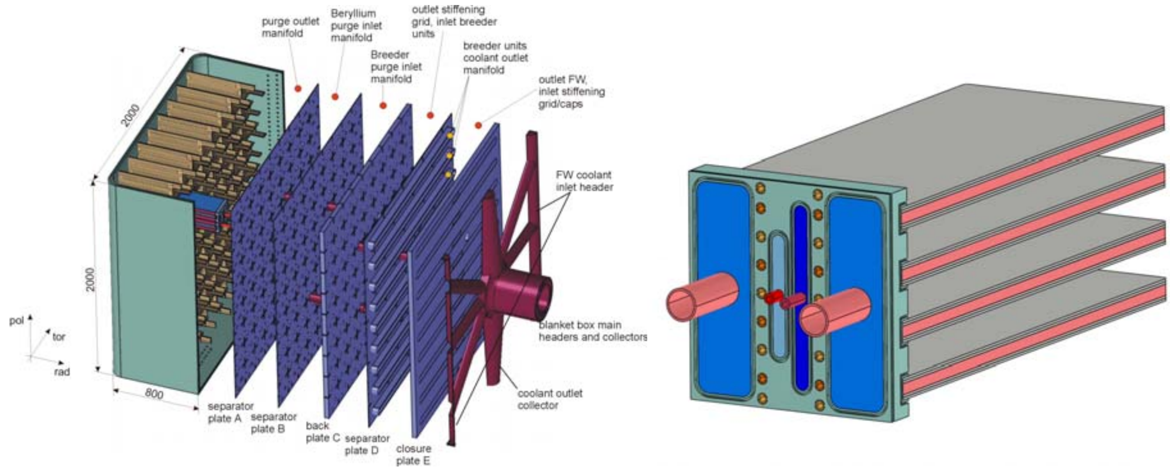


Fig. 4.3-1: Exploded 3D view of a HCPB DEMO module (left) and detail of the BU (right).

The new version proposed for the HCPB blanket is shown in Fig.4-3-1. The outer shell of the blanket box is made up from a steel plate with internal cooling channels bent into U-shape, the two remaining sides being closed by cooled cap plates. Inlets and outlets of all channels are located at the radial back of the box. Welded into the box is a stiffening grid of radial-toroidal and radial-poloidal plates; each grid plate is cooled by He flowing in internal channels that are fed in the rear part. This grid results in cells open in the rear radial direction with toroidal-poloidal dimensions of about 20 cm x 20 cm that accommodate the breeder units (BU). The spacing of the grid is determined by the mechanical strength of the box's walls according to the 8 MPa fault condition. The joints of each group of four stiffening plates form a cross that extends into the radial back and is needed for a strong connection of the grid to the module back plate.
The breeder unit is a base plate that holds four breeder canisters, each providing space for a shallow ceramic breeder pebble beds. The CPs contain a dense array of meandering internal rectangular cooling channels, fed from headers in the base plate. The space left by the canisters is filled with Beryllium pebble beds.
The blanket back region is built up from several large steel plates. The thicker outer (D) and the inner plates (B) are connected by radial ribs to provide the mechanical strength needed for contain the high He pressure; the intermediate plates have only the function to divide the space realising the separate manifolds necessary for the radial cooling scheme. The space between the breeder units and the high

pressure manifolds is used by the purge system with a thin plate (A) creating two headers for the inlet and outlet of the purge flow.
Supplying all structures with sufficient cooling is one of the challenging tasks of designing the blanket. In the present HCPB design, Helium coolant passes the major blanket parts in series: the first pass is through the U-shape First Wall/side walls, the second pass is the stiffening grid plates (75%) and the caps (25%) in parallel and the third pass is the breeder units.

4.3.1.1.2 - HCPB DEMO Performances

Neutronic calculations based on PPCS model B [4.3-4] have shown the achievement of the target TBR with a breeder zone thickness of about 46 cm. Thermo-hydraulic and thermo-mechanic assessment, can demonstrate that temperature and stress distribution are inside the design limits. Mechanical analyses showed that one of the main design requirements; i.e. withstand of the box in case of an in-box-LOCA, can be demonstrated. After the accident, the box should be replaced. The main performances of the HCPB blanket are summarized in Table 4.3-1.

Table 4.3-1: Main HCLL DEMO blanket design characteristics and performances

Heat Flux on FW	0.5 MW/m^2
Neutron Wall Loading	2.4 MW/m^2
He inlet/outlet temperature	300/500 °C
He coolant operating pressure	8 MPa
Estimated pressure drops in the blanket module	0.403 MPa
Max. temperature in: • FW (steel) • CP (steel) • Ceramic Breeder • Beryllium	 552°C 547°C 920°C 650°C
Tritium Breeding Ratio (breeder radial thickness)	1.14 (46 cm)

Tritium permeation in the Helium coolant (with potential release to environment) can be controlled without additional permeation barriers with a proper selection of the working point for the Tritium Extraction and Purification Systems. This result was found in previous analyses and recently confirmed for the new version of the DEMO blanket [4.3-5]. A new assessment is planned in the next years to account recent results on helium-water heat exchangers and material properties.

4.3.1.1.3 - HCPB testing strategy and test programme in ITER

The generation of data in ITER supporting the design of the DEMO blanket is a very important mission in the EU road map to the commercial fusion reactor [4.3-6], but it is also an ambitious task. Due to the irradiation conditions present in ITER that are for some parameters very different from the DEMO ones, the direct testing of the DEMO blanket components in ITER is not possible; large differences in the neutron wall load (only 30% of the DEMO values) and in the surface heating (about 50%) lead to different conditions for the blanket testing in terms of power density, temperature field and stresses. Furthermore, the total neutron fluence will be not more than 0.3 MWa/m^2, only a small fraction of the 7 MWa/m^2 that are indicated as target for a DEMO blanket module; the observation of the irradiation damages in the TBM material will be limited almost to an early phase of the damage processes (only few dpa in steel can be generated by these fluences). In addition the TBM is an ITER component; it must be integrated in the reactor and therefore it must be compatible with the systems and operational procedures of ITER, in particular the more frequent temperature cycling due to pulsed operation. TBMs must also not compromise safety and a reliability of the experimental machine. From these considerations it is clear that the design of the TBM will be a compromise between different requirements.

A testing strategy has been developed in the past years to obtain relevant DEMO data from the test in ITER; a proposal of a test plan for the first 10 years of ITER operation including the objectives and the rationale of these tests was elaborated as included in the final report of the TBWG-10 [4.3-7]. This kind of strategy has been confirmed for the present TBM design; this lead to the proposal of different versions of the TBM to perform specific experiments in the different fields of neutronics, thermomechanics and electromagnetic. Furthermore, these tests have been adapted to the operational plan of ITER that foresees different plasma phases from the first H-H pulses to operational scenarios with long pulses (up to about 3000 seconds) and back-to-back pulses for several days.

Four modules have been considered for the HCPB: 1) the Electro Magnetic module (EM) used in the beginning of ITER operation during the H-H phase to investigate the response of the structure to electromagnetic transients and to test operational function (cooling systems, remote handling, etc.); 2) the Neutron and Tritium module (NT) for investigation of neutronic responses and Tritium generation/extraction during the D-D and in the early D-T phase of ITER; 3) the Thermo-Mechanic module (TM) for investigation of the pebble bed behaviour at relevant temperatures for the DEMO during the low duty D-T plasma phase and 4) the Plant Integration module (PI) to demonstrate operational behaviour of the blanket component in the heat extraction and tritium management.

At present the design of the different variants of these modules is ongoing, with the priority on the EM-TBM that should be delivered to the ITER site in time for the commissioning procedures; it is planned that the TBM will be installed in the machine ready for the beginning of ITER operation, thus minimizing the number of test needed to check only the integration of the various systems. It is assumed that all 4 modules will share the basic architecture; this is foreseen in particular for the design of the structural part of these modules (first wall, caps, grid and back plate/attachment system) whose design will be qualified during the out-of-pile testing programme and step-by-step during the different phases of ITER operation. This strategy will assure a relatively stable interface between TBM and ITER during all the operation time with benefits for the availability and safety of the machine.

The main differences in the design of each TBM will concern the integration of the specific instrumentation and the design of internals; in particular the design of some BUs could be modified for testing optimised design variants or to achieve the required testing conditions; for instance the thickness of the beds will be increased if the requirements of the test call for relevant temperatures in the beds. The modular design of the box allows, in principle, to allocate for testing up to 18 different BUs in each TBM. Of course the testing possibilities are limited by the necessity to have a common helium feed system for all the BU units; this will not allow a completely independent regulation of the inlet Helium temperatures during the test. For some TBMs (e.g. the TM-TBM) the possibility is studied to add at least two independent small inlet/outlet Helium pipes; this will allow dedicated experiments in which the individual control of the Helium temperature in the BU is necessary.

4.3.1.2 – The Helium-Cooled Lithium-Lead (HCLL) blanket

In 2002 EU has endorsed the decision to concentrate the work on blanket modules for testing in ITER on a single coolant, Helium. Up to that time, two different coolants were envisaged for the EU Breeding Blankets: i) pressurized water for the Water Cooled Lithium Lead (WCLL) concept and ii) pressurized He for the HCPB concept (Helium-Cooled pebble-Bed). In order to maintain open the possibility of adopting both types of breeder – lithium-lead and ceramic – the choice of high pressure He coolant was made, water appearing indeed incompatible with the solid breeder solution (accidental Be/water interaction), and He presenting the advantage of higher potential thermodynamic efficiency. This option has therefore been precisely investigated with regard to design and material (9Cr ferritic-martensitic steel) limits under high thermal and neutron loading in future fusion reactors.
In this frame, a Helium Cooled Lithium Lead (HCLL) breeding blanket concept for DEMO has been developed and optimized (with regard to tritium breeding, heat removal and shielding capability), sharing with the Helium Cooled Pebble Bed (HCPB) concept the maximum basic technology (structures fabrication techniques, segment box, He technologies, etc). Previous developed concepts have permitted to identify the basic orientations for a common HCLL/HCPB modular design structure. In particular, the radial orientation of the He cooling in the Breeder Zone has been chosen, which allows the positioning of the collector region in the rear of the module. Furthermore, the U shaped FW/SW has been assumed for the module box. On the basis of these orientations an HCLL concept has been developed, which seems to be the best compromise with regard to its multiple functional requirements. Neutronic, thermal, thermal-hydraulic and mechanical calculations have been carried out for the dimensioning of the module, a feasibility fabrication study has been performed, and a manufacturing sequence has been envisaged. Design and analyses on the DEMO HCLL are reported in [4.3-8, 4.3-9].

HCLL DEMO design
The standard blanket module consists of a 2 m × 2 m × 1 m directly cooled steel box, reinforced by a grid structure, containing the flowing PbLi breeder/multiplier. Fig. 4.3-2 shows an exploded 3D view of a DEMO module. The module box is formed by a U-shaped First Wall (FW), cooled by He (pressure of 8 MPa) circulating in toroidal channels, and closed by two cooled caps (top and bottom). It is closed in the rear by a multi-shell Back Plate acting also as He collector and distribution system. The box is stiffened internally by poloidal-radial and toroidal-radial stiffening plates (SP), 8 mm thick and forming square radial cells of ~(210 x 210) mm^2. SP are actively cooled by He flowing in 4 parallel channels forming three U-turns in a way that He enters and exits from the rear manifold system.
The box is filled with liquid PbLi at 90% ^{6}Li enrichment for tritium breeding and neutron multiplication. It is slowly flowing for extracting the tritium outside the reactor (~10 recycles/day). The liquid metal enters from the rear collector, then it vertically circulates in some distributing boxes, each one supplying one column. The distribution boxes are constituted by vertical parallelepipedal chambers located behind the BU, between the BU He collectors and separated to form the inlet and outlet legs by an oblique internal wall. Through appropriate openings of the BU back plate, the PbLi enters in a BU and exits from the one below, feeding in this way all the BU in parallel (in couple). In each cell delimited by the internal stiffening structure is inserted a Breeder Cooling Unit (BU) (Fig. 4.3-2), consisting of 5 parallel Cooling Plates (CP). Each CP is 6.5 mm thick and is cooled internally by He flowing in 8 parallel square-shape meandering channels, collected in the rear Back Manifold (BM). The BM system is a four-stages collector that distributes/collects He and PbLi from/to the different components (FW, caps, SP and BU).

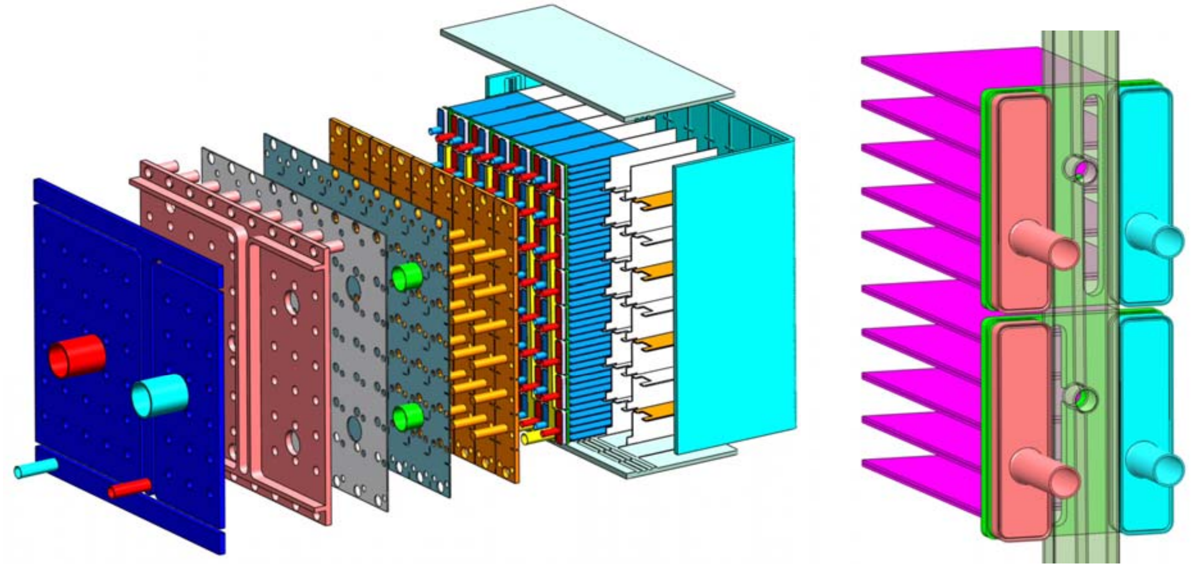


Fig. 4.3-2: exploded 3D view of a HCLL DEMO module and view of two breeder units

<u>HCLL DEMO Performances</u>

Neutronic calculations have been performed showing the tritium breeding capability of the modular concept. In particular, a TBR = 1.22 has been obtained for a radial thickness of the BZ of 75 cm and a ^{6}Li enrichment of 90%. A thickness of 55 cm would be sufficient to achieve the target TBR of 1.15, thus the reduction of the Li^{6} enrichment could be envisaged to reduce the costs [4.3-10]. An appropriate shield (with Tungsten Carbide) can ensure the required shielding performance of the blanket [4.3-11].

Thermal hydraulic and thermal-mechanical analyses have shown that the concept is able to recover the deposited heat with a quite good thermo-dynamic efficiency. Actually, although the maximum temperature value reached in the FW steel is high (~560°C), the module is able to resist to the mechanical and thermo-mechanical loads in normal operating conditions. A main concern could be the interface liquid metal /steel temperature, which reaches a local maximum of 544°C, higher than the limit fixed for the Water Cooled Lithium Lead blanket concept. However, an experimental campaign is ongoing to assess the behavior of this interface up to 550°C and develop corrosion barriers if needed.

Mechanical analyses showed that, in case of a LOCA accident, the module could withstand the entire coolant pressure. Furthermore, the deformations are compatible with the gaps foreseen between two adjacent modules, so guaranteeing the module remote removal and replacement with a new module. However, the return in service of the module after the accident could not be guaranteed.

The pressure drops in the module have been estimated at 0.34 MPa, leading to a pumping power of around 6 % of the gross electrical power. This value is acceptable if compared with the 10 % assumed as limit, however it does not take into account the pressure drops in the manifolding system, neither the one in the He ancillary circuit. The main performances of the HCLL blanket are summarized in Table 4.3-2.

Table 4.3-2: Main HCLL DEMO blanket design characteristics and performances

Heat Flux on FW	0.5 MW/m^2
Neutron Wall Loading	2.4 MW/m^2
He inlet/outlet temperature	300/500 °C
He coolant operating pressure	8 MPa
He flow velocity in FW/SP/CP	85/22/35 m/s
Max. temperature in: • FW (steel) • CP (steel) • Breeder/multiplier	 563°C 537°C 544°C at PbLi/steel interface
Tritium Breeding Ratio (breeder radial thickness)	1.15 (55 cm)

Tritium control analysis has not yet been performed for DEMO, therefore it is not yet decided if tritium permeation barriers will be necessary. The answer has to come from a tritium balance evaluation performed on the whole reactor blanket taking into account all ancillary circuits performances. This evaluation is on going within the EU programme, as well as the possibility of having permeation barriers on the He side both from the barrier efficiency and feasibility point of view.

HCLL testing strategy and test programme in ITER

In order to test in a real fusion environment DEMO-relevant blanket concepts before the construction of a DEMO reactor, a HCLL Test Blanket System (TBS) has been developed to be introduced and tested in ITER [4.3-9, 4.3-12]. Starting from the general testing objectives dedicated to TBM in ITER, some specific objectives have been identified for the HCLL concept:

- Validation of structural integrity of TBM under integrated acting of thermal, mechanical, and electromagnetic loads;
- Study of Pb-17Li flow and assessment of the MHD effects;
- Validation of thermal calculations results, and neutronics codes and models, including nuclear libraries used in ITER and DEMO analyses especially for the prediction of tritium generation rates, nuclear heat deposition, neutron multiplication and shielding efficiency;
- Study of tritium recovery process efficiency (from both Pb-17Li and He), temperature dependence of residual tritium inventory in the blanket, and T-permeation towards the main He-coolant stream;
- Validation of irradiation effects studied in fission reactor spectrum with the aim to check out the impact of the neutron spectrum at least for low fluence irradiation of EUROFER structures, welds and joints;
- Demonstration of HCLL DEMO blanket ability to generate high temperature heat suitable for electricity generation

In addition, as the Test Blanket Module (TBM), part of the TBS, is a FW component comparable with the ITER Shielding Blanket, it must fulfil the basic functions common to all in vessel components.

The TBS is described in sections 4.3.2.2 (TBM) and 4.3.3.2 (ancillary components).

For the first 10 years of operation, the ITER time-schedule envisages four plasma phases, H-H, D-D, low and high duty cycle D-T with related operating conditions. Because of the differences between the ITER loading conditions and those expected in DEMO, the option of testing several 'act alike' TBMs is selected, which will allow to adapt each module and relevant instrumentation to the ITER parameters in each phase. Moreover, this process will allow to gradually validate the HCLL blanket concept, technologies and design tools while having the minimum impact on ITER safety and availability. The HCLL blanket testing strategy in ITER is therefore organized around the following lines:

- HCLL TBMs shall be inserted since day-1 of ITER;
- HCLL TBMs shall include relevant DEMO technologies, provided they have an impact on the concerned test.
- A progressive TBM qualification and testing program adapted to the different ITER operation phases;
- Four HCLL TBM types are envisaged today to cover the first 10 years.

Taking into account the objectives and the testing strategy, a test programme has been defined. The test program based on the use of four different TBMs has been defined. It is summarized in the Table 4.3-3.

The foreseen 4 types of HCLL TBM corresponding to different ITER phases are the following:

1) EM-TBM: Electromagnetic TBM (will be installed in ITER from the first day of operation and tested before and during the H-H phase);
2) NT-TBM: Neutronic TBM (plasma D-D and first period of the D-T low cycle phases);
3) TT-TBM: Thermo-mechanic & Tritium Control TBM (last period of the D-T low cycle and first period of the D-T high duty cycle phases);
4) IN-TBM: Integral TBM (last period of the high duty cycle D-T phase).

The minimum duration needed to achieve the different test objectives has not yet been assessed in detail, although if some indications could be derived from previous studies.

Table 4.3-3: HCLL TBM test program proposal in ITER

#	Test Description	Test Requirement	TBM
1	installation of TBM, leak tests, remote handling tests	Prior to day 1 Start operation: day 1	EM
2	resistance against EM forces		EM
3	functionality, safety, thermal kinetics of ancillary circuits	vacuum in plasma chamber	EM
4	heat extraction from FW, verification of H-H surface heat flux data	during plasma pulses	EM
5	MHD pressure drop as a function of PbLi flow-rate	during plasma pulses	EM
6	H/D permeation from PbLi into coolant, evaluation of D inventory	vacuum in plasma chamber	EM
7	calibration of T source (n, flux) test of extractor w/ D	D-T plasma (no long pulses required)	NT, TT
8	deuterium permeation into FW coolant, verification of D-D surface heat flux data	D-D plasma	NT
9	T inventory (experimental conditions TBD)	D-T plasma	NT
10	temperature fields for code validation	D-T plasma	TT
11	stress distribution for code validation	D-T plasma (pulse length > 100 s)	TT
12	T permeation into coolant: • No PbLi circulation, no extraction	D-T plasma with pulse length as long as possible and frequent repetition time	TT
13	• with PbLi circulation, no extraction	D-T plasma with pulse length as long as possible and frequent repetition time	TT
14	• with PbLi circulation, with extraction	D-T plasma with pulse length as long as possible and frequent repetition time	TT
15	Reproducibility of preceding tests, reliability	D-T plasma with pulse length as long as possible and frequent repetition time	In

With the first two types of modules, useful information can be obtained about the impact of the TBM on the plasma stability, as well as on the TBM structural integrity and system functionality. The provisional capability of the calculation tools (neutronic, EM, MHD) can be furthermore validated and sources can be calibrated for the following phases.

The two others types of modules will allow to complete the code validation (thermo-mechanic), the tritium control, up to the integral qualification of the HCLL blanket and of PbLi and He coolant circuit components under DEMO relevant conditions. He parameters can be varied to achieve the DEMO relevancy under different loading conditions, compatibly with the response time of the system regulators. Therefore, the meaningfulness of most of the tests in the D-T phase, in particular for the TT-TBM and for the In-TBM, will depend on the capability to predict the actual surface heat load with sufficient advance and to keep it constant for a sufficient long time.

4.3.2 – TBMs design and analyses

4.3.2.1 – The Helium-Cooled Pebble-Bed (HCPB) TBMs design and analyses

TBMs requirements, engineering description, performance analyses including safety and required R&D are described in details in the HCPB TBM DDD [4.3-13].

4.3.2.1.1 – Design Description and operational work-point

The description of the design of the HCPB TBMs has been already presented in several posters and oral presentations in the main fusion technology conferences in 2004 and 2005 [4.3-14, 4.3-15, 4.3-16]; we refer to these publications for further detail in the TBM design description . Fig. 4.3-3 shows a CAD drawing of the HCPB TBM according to the status of the work as reached in May 2005.

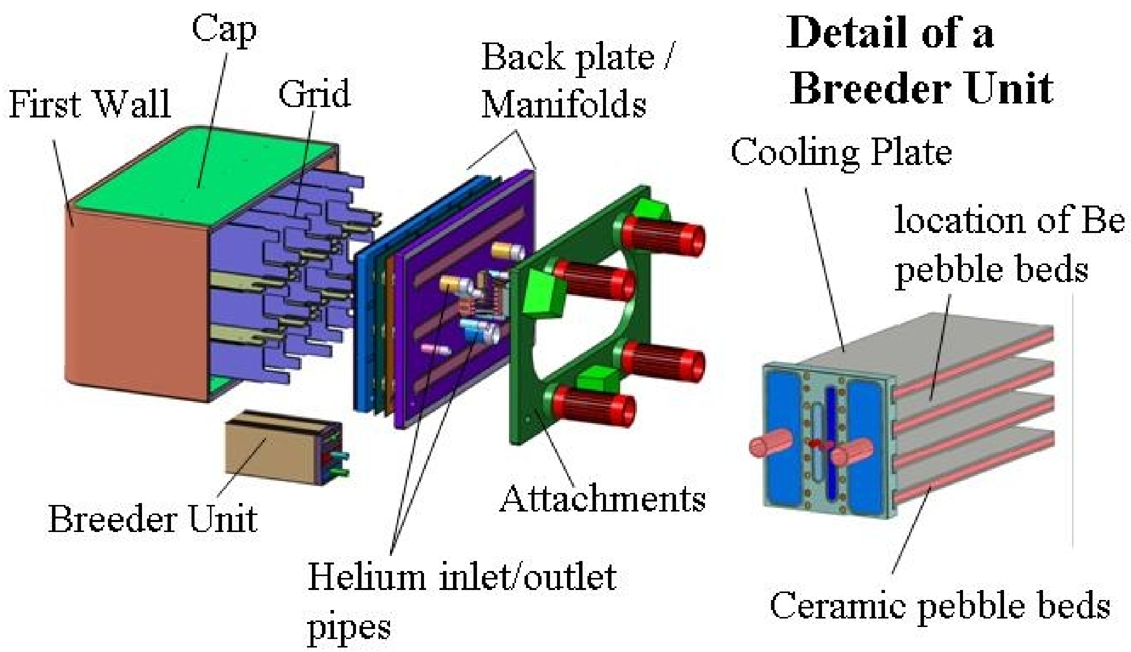


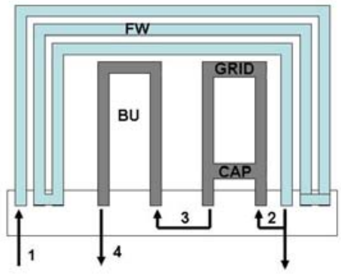
Fig: 4.3-3: CAD drawing of the EU HCPB TBM in an exploded view

As mentioned in Section 4.3.1.2, the strategy followed by the European Programme for the testing of DEMO concept in ITER calls for several (4 in case of the HCPB) test objects that should be inserted in succession in ITER. The dimensions of these objects have been selected to be about a half of the port, allowing a test volume of about 0.8 m^3 with 1 m^2 of FW exposed to the burning plasma. With these dimensions the test object in ITER will allow to reproduce a relevant portion of the DEMO blanket, keeping the typical dimensions of the breeder units (about 20 cm x 20 cm), but reducing the number of them; hence, the TBM represents about 1/4 of a typical module foreseen for DEMO. The HCPB TBM has been designed for a horizontal configuration, characterised by 18 cells (6 in toroidal and 3 in poloidal direction). The HCPB TBM is assumed to be installed in a frame-Port Plug in the equatorial port #16 of ITER; the total space in the Port Plug is shared with another TBM in an up-down configuration.

The design of the HCPB TBMs is based on the common architecture that was developed for the HCLL and HCPB proposal for a DEMO blanket in 2003 (see section 4.3.1.2). The major differences are in the manifold region that has to cope with the specific interface to ITER (mechanical attachments, piping connections, etc.) and in the thermo-hydraulic lay-out that has been partially modified to allow relevant test conditions. An example of this is the lay-out of the FW; the design reproduces the U-shaped plate with internal cooling channels like in DEMO. However, to perform tests using the same mechanical features of the DEMO component i.e. the possibility to withstand a 8 MPa over-pressurisation of the box, the dimensions of the FW (thickness and channels dimensions) must be very close to that used in DEMO. Due to the particular conditions in ITER (shorter module of 1.2 m instead of 2 m, and lower neutron wall load) this causes too low velocity of He in the channels and a reduced heat transfer capability at the plasma side. Hence, a double sweep is used to increase the channel length by a factor of 3 allowing to increase the velocity of Helium in the channels without changing the DEMO-relevant geometry of the channel itself. This solution causes as a drawback an increase of pressure drops in the fist wall up to 0.3 MPa.

Another important difference in the lay-out is the introduction of a by-pass line for partial decoupling the cooling of the first wall from the cooling of the breeder zone. As it has been previously mentioned, the nominal surface heat flux for the TBM in ITER is relatively low (270 kW/m^2 in comparison to 500 kW/m^2 foreseen in DEMO), but the simultaneous possibility of unpredictable hot spots of 500 kW/m^2 affecting 10-15% of the FW surface, obliges to dimension the cooling capacity of the whole FW according to cooling needs of the local spot, but extrapolated to the whole plasma facing surface area. A study of the transient behaviour of the first wall surface temperature due to a surface heat flux increase from 270 to 500 kW/m^2 showed a local temperature rise rate of about 8 K/s with a time constant of about 10 s. As there is no way to cope with such an off-set condition regulating the inlet coolant mass flow or temperature, the operational cooling conditions of the first wall must be designed for such extreme loading conditions. As the breeding zone (BU and Grid) is connected in series after the FW cooling, the chosen FW mass flow causes lower outlet temperature of the outlet He, missing the testing goal to reproduce the high performance of the DEMO blanket. To cope with these conditions and assure a major flexibility in TBM operation, a by-pass after the FW and before the cooling in the breeding region has been added. This by-pass flow can be tuned by an external valve regulating the mass flow in the BUs (a time constant of about 50 s is available for the regulation of the temperatures in the BU).

Table 4.3-4: Working point parameters for nominal conditions (270 kW/m^2 surface heating and 0.78 MW/m^2 neutron wall load) during a long pulse (>1000s)

	HCPB
Thermo-hydraulic scheme	
FW surface	0.94 m^2
He pressure	8 MPa
He: Temperatures and mass flow	1) 300°C – 1.3 kg/s 2) 370°C – 0.65 kg/s 3) 420°C – 0.65 kg/s 4) 500°C – 0.65 kg/s
Pressure drops	<0.3 MPa
Tritium production	1.1 µg/s
Material max temperatures:	Steel: ~550°C Beryllium: ~650°C Ceramic Breeder: ~920°C

4.3.2.1.2 – Performance analyses

The proposed design has been evaluated under neutronic, thermo-hydraulic, structural and tritium control point of view to assure that the main requirements for the structural integrity, safe operation and testing in ITER are fulfilled. Neutronic analyses have been performed with MCNP codes to determine the tritium production and the total volumetric power generated in a HCPB TBM; it will be in the order of 560 kW, with only small deviation from the different geometry analysed. The tritium production will be in the range of 100 mg/day at full power conditions.

For the analysis of the temperatures and stresses a combination of models for thermo-hydraulic and structural calculations has been used. RELAP/ATHENA codes have been used to simulate the whole He loops (including a relatively detailed simulation of the TBM flow schema) for transient evaluation. 3D Fluid dynamic calculations (with STAR-CD) have been used for analysing detail of the flow in the Manifolds and in the FW channels, resulting in a evaluation of velocity distribution, pressure drops and heat transfer coefficients. 2D and 3D (up to ¼ of the TBM) ANSYS models with different detail have been developed for calculating temperature distribution and stresses; the resulting stresses have been assessed according to the material design data for EUROFER recently presented by EU [4.3-17]. Table 4.3-4 summarises the main parameters of the present TBM lay-out.

4.3.2.1.3 – <u>HCPB TBMs integration in ITER</u>

The TBM is located inside a Port Plug (PP) that offers a standardised interface to the ITER VV and allows the replacement of the TBMs. The frontal part of the PP (the so called "frame") provides a containment structure that thermally insulates and shields each TBM from the other one and from local ITER structure. The whole structure is cantilevered with a flange to the VV Port Extension. A thick neutron shield (about 1m thickness) inside the PP protects the magnets, the structures and the buildings from the neutron flux. The shield constitutes also the first interface for TBM integration providing a support for the mechanical attachments and penetrations for the TBM pipes. The mechanical attachment is designed to cope with the large electromagnetic forces that are created during plasma disruptions (a radial torque of 0.72 MNm has been calculated and used as design reference for the HCPB design). The attachment consists of 3 shear keys capable of withstanding the EM torque and acting as the fixed point (in the middle of the TBM back plate), and 4 “flexibles” that locate the TBM with bolts in radial direction. The presence of large Helium pipes caused some difficulties due to space requirements for the integration. As Helium is transparent to neutrons and does not contribute to the shielding, only pipes of internal diameter (ID) smaller than 30 mm can be routed straight through the Port Plug shield; tubes of larger diameter must be bent within the shielding to reduce neutron streaming.

The vacuum boundary is realised at the rear part of the shield; here the vacuum at the penetrations coming from the TBM should be ensured in relation to 0.1 MPa in the inter-space cell. To attach the TBM to the shield with screw straight holes are needed in the shield for access; to ensure the shielding function these holes should be closed by shield plugs. Flexible support (bellows) in the Helium pipes to the PP must be designed in order to compensate the differential thermal expansion (e.g. up to 500°C for the hot leg in comparison to ~150°C of the PP structure). In line with the ITER requirement each bellow must be secondarily contained with monitoring of the inter-space. The resulting interface is quite complex; this interface is one of the most critical factors and may limit the number of independent TBM systems that can be tested at the same time in ITER.

Following the present philosophy adopted by ITER for the PP handling, the replacement of the TBMs is done in the Hot Cell after the Port Plug is extracted from its position in the VV, transported in the cask to the hot cell and docked to the PP refurbishment workstation. The replacement system studied in [4.3-18] is based on the use of in-bore tools and mechanical screw actuators that operate from the clean side of the docking station reaching the Interface 1 (see Fig. 4.3-4) inside the PP shielding. For the Helium pipes, two types of in-bore tools are required, a cutting/welding/testing device for straight pipes of 30 mm ID, and an analogous device for bent tubes of 80 mm ID and bend radius not lower than 400 mm. The PbLi re-circulation pipes for the HCLL should require tools of the first type (30 mm ID). In addition, the electrical grounding and the instrumentation cables must be connected/disconnected as well; for these systems the design is ongoing.

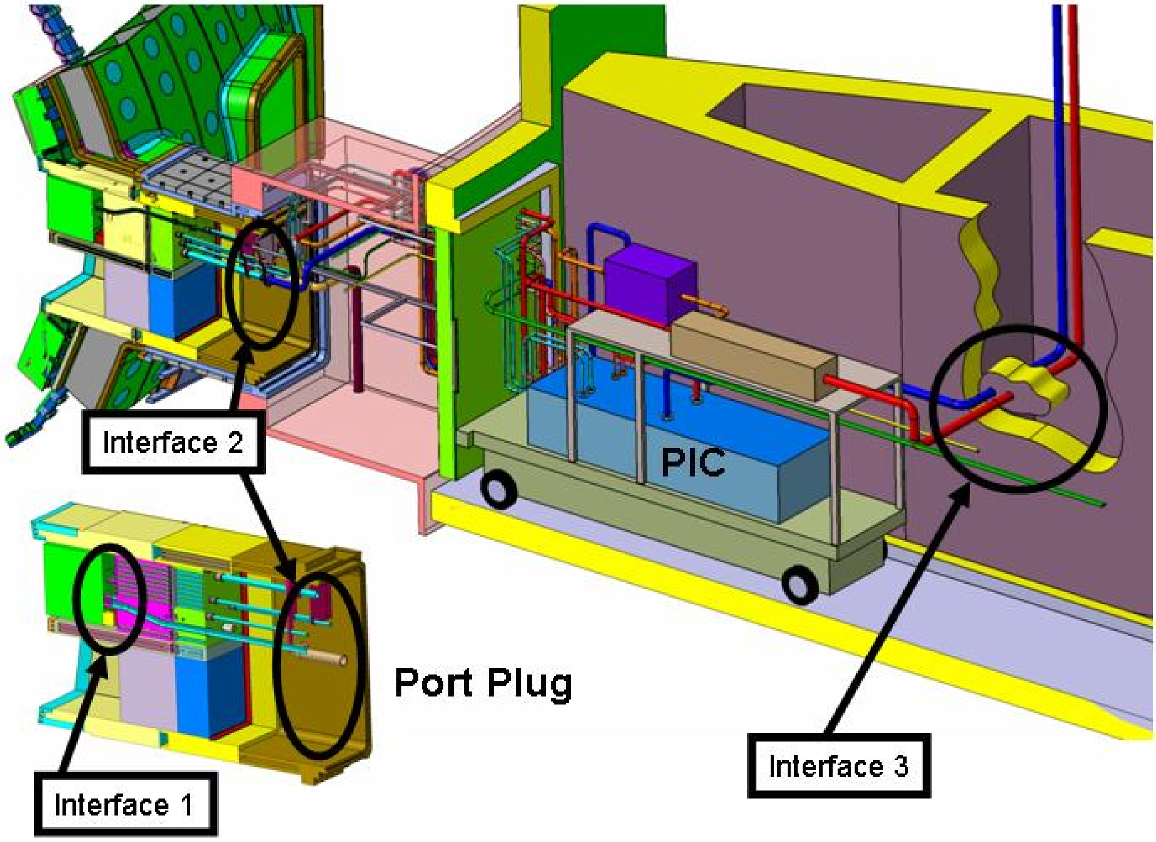


Fig. 4.3-4 Example of integration of the HCPB TBM in ITER

Several systems should be accommodated in the space in front of the VV Port (the inter-space in the VV Extension inside the Bio-shield, and in the Port Cell); for both HCPB and HCLL concepts the piping of the main coolant system, some component of the by-pass (valve and thermal sleeves), Helium or PbLi pipes for the Tritium extraction, measurement systems, etc. A further issue is the lay-out of the main Helium pipes to the respective HCS allocated in the Torus Coolant Water System (TCWS) vault. An example for the HCPB integration is presented in [4.3-18]; again the major requirements for the lay-out are the restricted space and the necessity to realise a quick exchange of the components. The use of an integrated device (the so called Piping Integration Cask, PIC) allows the pipes to be cut and re-welded only at the boundary interfaces, removing all the intermediate components as a block. The lay-out of Helium pipes of 80-100 mm ID from the interface 2 to 3 (see Fig.4.3-4) is dictated mainly by the necessity to compensate the thermal expansion of the pipes (maximum design values are 500°C and 8 MPa); large radius bends have been designed to accommodate this expansion. A conclusion of this study was that the space is restricted, but the accommodation of a second analogous system to support the TBM in the lower port is still possible if a common PIC is used for the integration of both systems.

4.3.2.1.4 – Safety

Detailed safety analyses for the HCPB TBM have been already performed for the previous versions of the TBM in 1998 and 2001 (references). Three different scenarios were analised as envelope of the worst case.:

1) In-vessel LOCA. The accidental sequence starts with the rupture of 4 FW channels, followed by plasma shut-down caused by the entrance of Helium in the Plasma chamber.

2) In-box LOCA. Rupture of an internal Helium channel with over-pressurisation of the box. According to the previous design, the intervention of a rupture disc protected the box from a possible explosion.

3) Ex-vessel LOCA. In this scenario a break of a main He pipe in the ITER building is not detected and the TBM is irradiated without cooling until the evaporation of the Be layer in the first wall shuts down the plasma.
The three scenarios were analyzed and the consequence (in term on VV pressurization, T release, H production, short and long term cooling, etc.) was found compatible to the ITER safety requirements.
For the new design, these analyses should be repeated (they are ongoing and will be included in an advanced version of the DDD). Also if no significant changes from the conclusion of the previous analyses are expected, at least the scenario 2 necessitates to be re-evaluated. In fact the introduction of a box design able to withstand 8 MPa and the consequent cancellation of the use of rupture discs for box protection will modify strongly the accidental sequence of this case.

4.3.2.2 – The Helium-Cooled Lithium-Lead (HCLL) TBMs design and analyses

TBMs requirements, engineering description, performance analyses including safety and required R&D are described in details in the HCLL TBM DDD [4.3-19].

Preliminary assessments on EM-TBM definition

From the preliminary global list of test objectives to be achieved during ITER program (see §4.3.1.3), some fields of test activities have been identified as relevant to be started during the HH phase, since day-1 of ITER, on the bases of the following criteria:

- to gain earlier confidence in the TBM before DT phase (e.g. structural robustness, compatibility of the module with ITER operations),
- to develop experimental skills on sensitive activities in real ITER environment (e.g. T management using safe D equivalent),
- opportunity to use the specific ITER environmental conditions not easily available out of this facility (e.g. high magnetic fields, stress induced by plasma disruptions),
- interest to host extensive or intrusive instrumentation before encountering more stressing conditions due to neutron load (validation of MHD codes for PbLi circulation with numerous flow-meters).
-

The global design of the EM-TBM is quite similar to the one of the In-TBM, described in details in the next section. One of the main differences with the In-TBM is the lack of tritium production and neutron heating inside the module. As the thermal analyses have shown that the heat flux on the FW is the dimensioning factor, the EM-TBM FW, thus, will not differ from the In-TBM one.

After the development of a preliminary mounting sequence [4.3-20], the design of this first TBM to be installed in ITER has been modified to take into account results of an industrial expertise performed to identify fabrication and mounting sequence issues [4.3-21]. Some design guidelines have been adopted in consequences, such as: avoid welding triple points on the junction First Wall / Stiffening grid, no sharp points on welding trajectories, specific grooves to avoid possible interference between welding beams and welded parts, optimisation of the welding depths for thick plates, etc. In parallel, development of the sub components fabrication method is the subject of an intensive R&D programme [4.3-22]. The global design approach is detailed in [4.3-23], and the corresponding development and qualification programme is synthesized in [4.3-24].

Assuming a dimensioning heat flux of 0.3 MW/m^2, 344 KW are deposited on the TBM. The He (0.6 kg/s) enters in the FW at 300°C and exits at 410°C, then it passes in the Breeder Zone (BZ), where it allows keeping the lithium lead over 300°C, which is the temperature of the liquid metal at the exit of the PbLi ancillary circuit. No precise calculations have been carried out on the thermal power lost from the TBM by radiation, however no supplementary heating should be needed to avoid the PbLi freezing.

Analytical and FEM computations have been performed to assess the possibility to use externally D-saturated PbLi in order to evaluate T-permeation and T-control in the absence of T bulk production by neutron. The expected very low concentration of D in He coolant has led to design a dedicated purge gas circuit in the EM-TBM.
Innovative sensors offering high potential in term of installation facility and insensitivity to nuclear fusion environment could be tested (e.g. optic fibre engraved with Bragg gratings for temperature measurement or even for other physical data).

<u>The HCLL In-TBM design</u>

The In-TBM (see Fig. 4.3-5) looks alike a generic HCLL breeder blanket module for DEMO, the dimensions of which have been adapted to fit it in a ITER test port. It features a steel box of 575 (rad) × 626 (tor) × 1838 (pol) mm^3 cooled by horizontal multi-passes rectangular cross section channels and closed by top and bottom cooled covers and, in the rear, by 4 steel plates acting also as distributing/collecting chambers for the He coolant.

The box is stiffened by poloidal radial and toroidal radial cooled plates (vertical and horizontal stiffening plates, SPs) in order to withstand the internal pressurization at 8 MPa which could take place in case of accident (loss of coolant inside the TBM). The grid also stiffens the box against the torques acting on it during disruptions.

The grid forms 24 radial cells (3 columns × 8 rows) of ~380 (rad) × 180 (tor) ×214 (pol) mm^3 in which circulates the multiplier/breeder PbLi, so allowing external tritium extraction. In each cell is inserted a breeder cooling unit (BU), ensuring the heat recovering from the breeding zone. Each BU consists of five radial toroidal plates (Cooling Plates, CPs) cooled by internal double U rectangular channels and welded to the BU back plate. The lower row BUs contains four CPs to allow a safe PbLi draining/replenishing (see Fig. 4.3-5). Two BU collectors located behind the BU back plate distribute/collect the He circulating in the CPs.

The thickness of the FW, CPs and SPs and their relevant channels pitch and cross section are summarized in Table 4.3-5.
The back manifold is reinforced by stiffening steel rods for pressure withstanding. The rods have a tubular cross-section so allowing to use them either as an access to the module body for the instrumentation connections, or as thread for the bolts of the attachment system. This tubular rod design presents also the great advantage that structural function (relying on threads and conical surfaces) and tightening function (relying on welding) are decoupled.

One He circuit is envisaged to cool both the FW and the breeder zone (BZ). The He enters through the inlet pipe into the first chamber (between the 1st and 2nd BP) from which it is distributed to the FW tubes; it is collected then into the second chamber (between the 2nd and the 3rd BP) from which it feeds in parallel the SPs, the covers and the CPs. It is then collected in the 3rd chamber (between the 3rd and the 4th BP) from which it leaves the module. A by pass is envisaged at the exit of the FW in order to increase the He outlet temperature at the exit of the CPs.

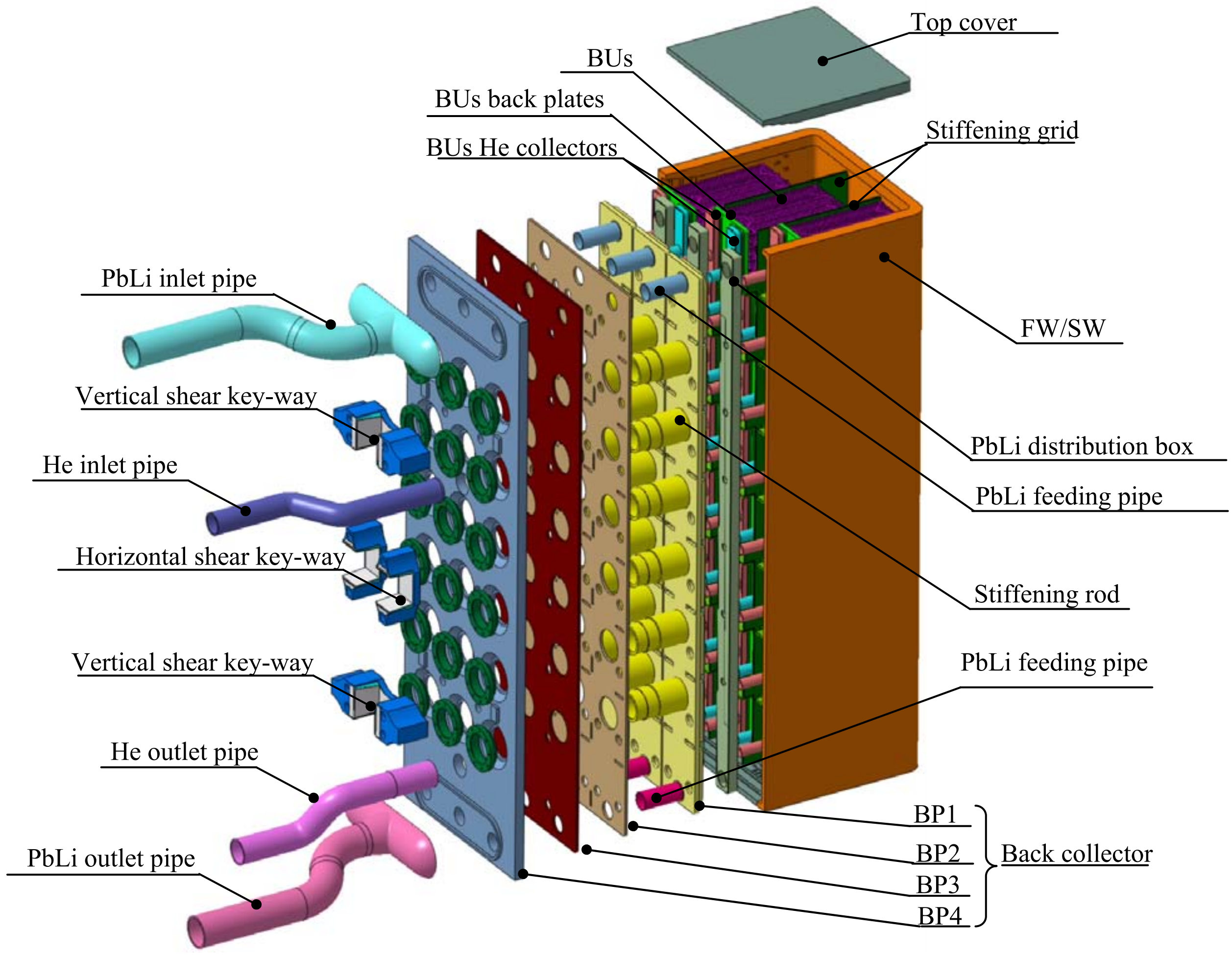


Fig. 4.3-5: Exploded view of the In-TBM

Table 4.3-5: Cooling channels dimensions in the various TBM components

	FW	hSPs	vSPs	CPs
Thickness (mm)	25	8	8	6.5
Ch. Pitch (mm)	22.2	10	12.2	5.5
Ch. Sect. (mm^2)	15(pol) × 14(rad)	3 (pol) × 8 (tor)	10 (pol) ×3 (tor)	4 (tor) × 4.5 (pol)

The liquid metal circulates in the same way as in the DEMO module (see section 4.3.1.1). Assuming 10 recirculations/day, the PbLi mass flow rate will amount to 0.33 kg/s. With this figure, the liquid metal velocity will be ~6 mm/s in the feeding pipes and in the distribution box, ~0.08 mm/s in the BU current section and will increase to ~2.8 mm/s in the front opening between the SP and the FW. The PbLi draining is realized from the TBM bottom, to allow the draining by gravity so improving safety. When draining PbLi from the module, it is necessary to insure that, in case of residual PbLi, its solidification does not lead to wall failures. This has lead to lower as much as possible the exit pipes used for the draining, and to eliminate the lowest cooling plate of the bottom BUs, in order to avoid its potential interaction with residual PbLi.

The present TBM design assumes a frame thickness of 140 mm for the external walls and 100 mm for the central wall. Thus, the total volume of Pb-Li within the TBM is today 0.292 m^3. If needed, especially for safety concerns, this quantity could be easily reduced under 0.28 m^3, either by reducing of about 8 % the radial thickness of the breeder zone or by fitting the TBM dimensions in a 200 mm thick frame.

HCLL TBMs integration into the frame

The module attachment system to the frame is based on the principle of ITER modules attachments, and combines (Fig. 4.3-6):

- A flexible fixation (flexible cartridges) on 4 points to recover the radial mechanical loads while authorizing the thermal expansion in the poloidal and toroidal directions;
- A gliding shear keys system along a cross-shaped key way on the external back plate, to lock up the module displacements in poloidal and toroidal directions during the disruption loads, the thermal expansion in these directions being free; it also contributes to bear the weight of the module.
-

The whole system is positioned on the external plate of the module's back collector (Fig. 4.3-6). The bolts of the system of flexible cartridges are screwed in four of the stiffening rods of the back collector (the cartridges being screwed in the frame). The key ways are laid out on the back plate according to a cross centred on the module, the shear keys being fixed on the frame. Due to lack of data on Electromagnetic Forces (EF) acting on the TBM during a disruption, the attachments have been dimensioned to resist a maximum radial torque of 1.5 MNm.

The He and PbLi pipes feeding the TBM are bent in the crossing of the frame in order to limit the neutron streaming. The radii of curvature of the pipes are imposed by geometrical constraints (length of the crossing and diameter of the tubes) and by the fact that the tubes must present significant straight portions to authorize their cutting and their welding during the replacement of TBM.

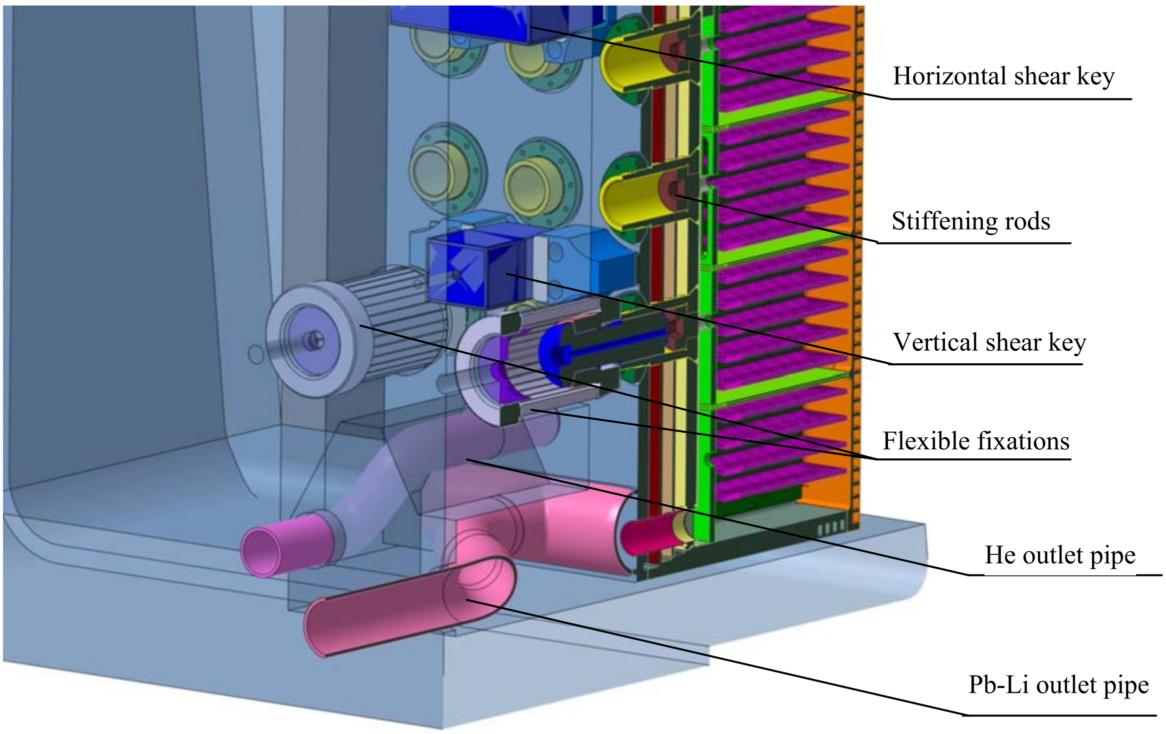


Fig. 4.3-6: Detail of the TBM (section) into the frame

HCLL In-TBM Thermal-hydraulic, thermal and mechanical analyses

Analyses performed for the HCLL TBM are detailed in [4.3-25].The first In-TBM was designed to resist to a surface Heat Flux (HF) = 0.25 MW/m^2 and to a Neutron Wall Loading (NWL) = 0.78 MW/m^2. Recently, ITER Team has requested TBMs to be designed to withstand a surface heat flux of 0.5 MWm^{-2}, even if most of the time the real heat flux will be lower. 1.2 MW are deposited on the TBM of which about 0.7 MW on the FW and 0.5 MW on the breeder zone (CPs, SPs and liquid metal). Due to higher ratio between the HF and the NWL (0.5/0.78 against 0.5/2.4 in DEMO) neither the cooling schema nor the He parameters adopted in DEMO are fully suitable for the TBM.

Steady state analyses, thermal and thermal-hydraulic, have been performed to adapt the outline design to these new "dimensioning loading conditions". The He flow schema and the flow parameters (mass flow, first wall channels cross-section) have been optimised in order to reach a minimal pumping power for the whole He circuit, while keeping a maximal first wall temperature to 550°C. The resulting flow schema is shown in Fig. 4.3.2-7, in which He parameters and pressure drops in the various circuits are also reported. A by pass is envisaged at the FW exit in order to reduce the total pressure drop and to be able to increase He temperature in the breeder units. The mass flow in the SPs and in the CPs are optimised in order to obtain in the BZ steel temperatures of the same order of magnitude as those obtained in DEMO so guarantying a good relevance in terms of T permeation. This mass flow distribution should be achieved through appropriate diaphragms, the feasibility of which, nevertheless, has not yet assessed.

The total pressure drop (TBM plus external circuit) has been found to be 0.290 MPa, leading to a needed pumping power of 83 KW. The results for the corresponding thermal calculations are summarized in Table 4.3-7. In the table, the maximum/average/minimum temperatures obtained in the various TBM regions without varying the He mass flow, for three different heat flux values on the FW are compared with those obtained in the generic DEMO module. It can be seen that no DEMO relevancy (in terms of thermal behaviour) could be reached if no adaptation of the mass flow to the heat flux is possible. Analyses showed that He coolant parameters could be adapted to cope with different heat loads. However, automatic feedback control will require a relatively long response time compared with the FW time constant. Therefore, the meaningfulness of most of the tests in the D-T phase, in particular for the TT-TBM and for the In-TBM, will depend on the capability to predict the actual surface heat load with sufficient advance and to keep it constant for a sufficient long time.

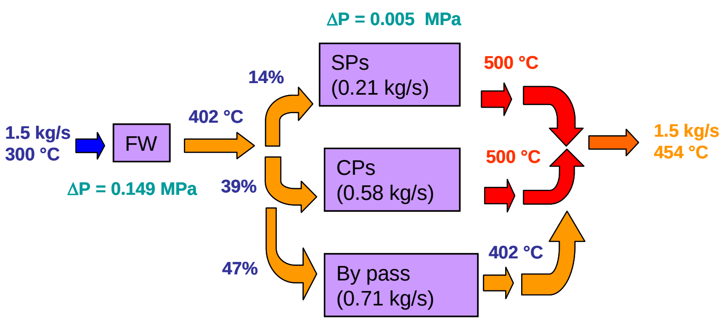


Fig. 4.3-7: Schema of the He flow with flow parameters and pressure drops

Table 4.3.2-6: Maximum/average/minimum temperatures (°C) in the TBM and DEMO regions

	FW	vSPs	hSPs	CPs	PbLi	Interface
DEMO	420/475/563	443/467/515	453/472/537	415/475/544	418/490/660	418/ - /544
TBM 0.5 MW/m^2	337/409/550	441/459/486	436/459/477	414/463/520	418/468/543	418/ - /520
TBM 0,27MW/m^2	334/393/481	407/424/452	402/424/442	380/427/485	383/434/511	383/ - /485
TBM 0.1MW/m^2	329/381/430	382/399/423	377/399/418	355/403/460	358/409/388	353/ - /460

In order to avoid the difficult problem of pressure drops equilibrium, a first alternative scheme consisting in cooling in serial FW then SP then CP is under evaluation. This would imply some important design modifications (such as addition of a fifth back plate, increase of the SP thickness). A second alternative scheme without by pass has been studied. Although the needed pumping power would be the same as in the previously described case, this scheme would imply higher pressure drops, so it is considered only as back up solution.
Transient analyses have been carried out considering an ITER pulse with a duty cycle of 400 s / 1800 s and showed that in terms of thermo-mechanical behaviour, stationary conditions would be reached in the TBM front regions, where maximum temperatures and stresses are located, after some tens of seconds (60 in the FW).

Mechanical analyses have been performed to verify the resistance of both the box and the CPs to He internal pressure. Primary stresses are everywhere under the allowable limits following the IISDC code. Analyses demonstrated that the design modifications as required by the fabrication (thinning of the 1st back plate in correspondence of the welding with the SPs) do not jeopardize the resistance of the rear manifold to the He internal pressure.
The resistance of the box in case of an accidental pressurization (it has been assumed that the rupture either of a CP or a SP would imply the pressurization of the entire box to the He pressure, 8MPa) has been furthermore tested. Analyses showed that, according to the IISDC criteria, the box will be able to withstand this type of load.

HCLL TBM safety approach

A first safety analysis of the TBM based on HCLL concept has been performed for further HCLL design and licensing. Starting from the review of both HCLL concept and ITER safety guidelines, several safety aspects have been proposed:

- a dedicated safety approach to initiate the safety analysis of the ITER/HCLL coupling;
- an approach of the specific risk, safety functions and related design requirements;
- a preliminary list of postulated events in the Design Basis domain;
- the possible change of the ultimate safety margins and of the radiological release risk.
-

According to this analysis detailed in [4.3-26], the integration of HCLL system into ITER does not raise strong drawbacks in terms of ITER plant reliability and safety. Nevertheless, some studies are needed to confirm the feasibility, in particular about: the reliability of helium cooling means; the safety provisions against tritium leaks and management; the detailed assessment of the consequences of the listed events.

4.3.3. - TBMs systems

4.3.3.1 – HCPB TBMs systems

The TBM requires several ancillary systems for supporting its operation modes (heat removal, tritium extraction, etc.) and to reach the objectives of the testing programme (e.g. measurement systems for tritium and neutronic). The main systems foreseen for the HCPB TBM are listed in the follow:

a) Helium Coolant System (HCS). The HCS provides the coolant helium necessary for TBM heat removal at the conditions of pressure, temperature and mass flow required by different operating conditions. Table 4.3-7 summarises these requirements for TBM type and related operation states.

Table 4.3-7: Coolant requirements for the HCS during the different operational states

<table>
<tr><th rowspan="2">Name of state</th><th rowspan="2">System pressure
(MPa)</th><th rowspan="2">He mass flow
(kg/s)</th><th rowspan="2">$t_{TBM\ inlet}$
(°C)</th><th rowspan="2">$t_{TBM\ outlet}$
(°C)</th><th colspan="4">TBM type</th></tr>
<tr><th>EM</th><th>NT</th><th>TM</th><th>PI</th></tr>
<tr><td>Maintenance</td><td>1</td><td>0</td><td>20</td><td>20</td><td>X</td><td>X</td><td>X</td><td>X</td></tr>
<tr><td>Cold standby</td><td>1</td><td>0.13
(0.18)</td><td>20</td><td>20</td><td>X</td><td>X</td><td>X</td><td>X</td></tr>
<tr><td>Component testing</td><td>1-8</td><td>0.13
(0.18)</td><td>-</td><td>-</td><td>-</td><td>-</td><td>-</td><td>-</td></tr>
<tr><td>Baking</td><td>1</td><td>0.13
(0.18)</td><td>250</td><td>250</td><td>X*</td><td>X</td><td>X</td><td>X</td></tr>
<tr><td rowspan="2">Hot standby</td><td rowspan="4">8</td><td rowspan="4">1.3 (1.8)</td><td>100</td><td>100</td><td>X</td><td>X</td><td></td><td></td></tr>
<tr><td>300</td><td>300</td><td>X</td><td></td><td>X</td><td>X</td></tr>
<tr><td rowspan="2">Reactor operation</td><td>100</td><td>Up to 300</td><td>X</td><td>X</td><td></td><td></td></tr>
<tr><td>300</td><td>Up to 500</td><td>X</td><td></td><td>X</td><td>X</td></tr>
<tr><td>Tritium outgassing</td><td>1</td><td>0.13
(0.18)</td><td>500</td><td>500</td><td>X*</td><td>X</td><td></td><td></td></tr>
<tr><td colspan="9">Values in brackets for mass flow are required for TBM-extended
* For pre-testing of equipments that will be used in later phases</td></tr>
</table>

The cooling subsystem is housed in the TCWS vault at the CVCS level, approximately 90-100 m (length of the pipes) away from the TBM. It consists of a single main loop arranged in a 8-shape-configuration (see Fig. 4.3-8); a recuperator splits the loop in two temperature regions. This configuration is being considered at the present because it allows the gas compressor to work at low temperature (max ~100°C). The Heat Exchanger is also located in the low temperature region reducing the temperature gradient with the secondary side that is cooled by the water of the Heat Rejection System of ITER (35-75°C). Fig.4.3-9 presents an isometric view of the HCS without thermal insulations. The system should be completed with a pressure regulation systems (not indicated in the figure) and with the connection to the Coolant Purification System.

b) Coolant Purification System (CPS). A purification system is provided for the HCS extracting continuously ~ 0.3% of the main flow by means of a by-pass. It is designed to purify the helium coolant stream keeping the partial pressure of T species (mainly HT and traces of HTO) lower than the design value of 0.3 Pa. In addition it controls the gaseous impurities like N_2 and provides addition of H_2 and H_2O in order to keep the gas composition required for the testing. The requirements of this system are summarised in Table 4.3-8. In the proposed purification process Q_2 (Q=H,D,T) in the feed stream is oxidised to Q_2O in a reactor filled by the oxygen donor Cu_2O-CuO. After cooling to 173 K, Q_2O removal is accomplished by a cold trap. The task of the impurity removal impurity (as well as not oxidized Q_2) is accomplished in a third step by a molecular sieve beds. Purified He is then routed back to the HCS. This system is located in the TCWS vault in the proximity of the HCS

Table 4.3-8: Specifications for the CPS and TES

	Specifications for the coolant in the HCS	Specifications for the purge flow at TBM inlet
Mass flow	By pass to the CPS for a max mass flow of 5 g/s	Adjustable in the range 0 – 1 g/s
Nom. Pressure	8 MPa	0.11 MPa
Gas composition:	$p(H_2)$ adjustable in the range 0-1000 Pa $p(H_2O)$ adjustable in the range 0-100 Pa p(HT, HTO) < 0.3 Pa Impurities: $p(N_2, O_2, CQ_4$, etc) < 8 Pa	$p(H_2)$ adjustable in the range 0-110 Pa p(HT,HTO) < 0.1 Pa Impurities $p(N_2, O_2, CQ_4, H_20$, etc) < 0.1 Pa

c) Tritium Extraction System (TES). The extraction of T from the ceramic and Be pebble beds (about 1.1 µg/s T production during the burning) is achieved by help of a low pressure (~0.1 MPa) Helium stream containing up to 0.1% H2; the addition of H is foreseen to facilitate the tritium extraction from the ceramic by isotopic exchange. To test the purging conditions for the HCPB concept, a Tritium extraction system is provided capable of removing the T and impurities, and of controlling the gas composition at the TBM inlet as summarised in Table 4.3-8.

For the removal of T from the He stream, the envisaged process considers:

- Q_2O trapping, that is carried out by a cold trap operated at around –100°C.
- Q_2 trapping from the purge gas, that is accomplished by a TSA (Temperature Swing Adsorption) system operated at the liquid nitrogen temperature in the adsorption phase where tritiated and not tritiated impurities are also trapped on molecular sieve. In the industrial TSA practice for H_2 purification, the regeneration phase is normally carried out at 20 °C, under a pure He stream. The TES will be allocated in the T building in standard glove boxes.

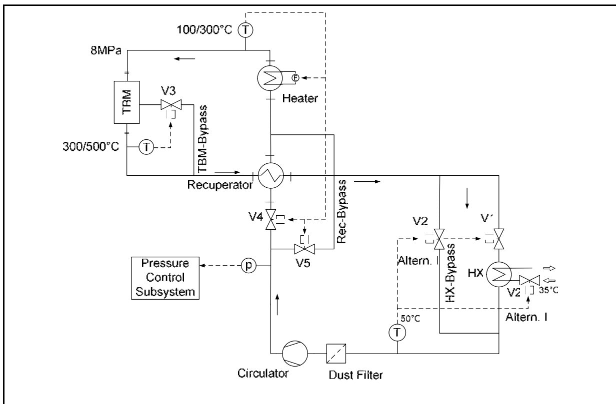


Fig. 4.3-8: Flow diagram for the HCS

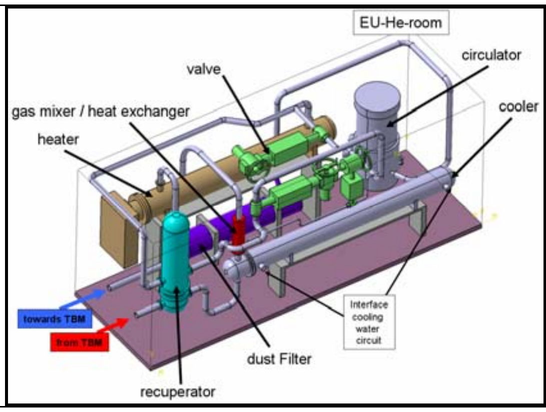


Fig.4.3-9: HCS isometric view

d) Measurement Systems. Additional ancillary equipment is required to perform measurements during the operation of TBM. In particular for the NT-TBM, two systems are foreseen for tritium and neutron flux measurement. The Tritium Measurement System (TMS) will substitute the TES for tritium extraction and accurate measurement of tritium production. The Neutron Measurement System

(NMS) will provide a mechanical inserting/extracting system with three channels connecting the TBM to the position of the NMS outside the Bio-shield. These channels will be used in several experiments for locating inside the TBM fission chambers, activations foils and materials samples for neutron flux measurements and material irradiation investigations. Both systems will be allocated in the Port Cell correspondent to the TBM location in reactor.

e) Helium Conditioning Units. Especially during the TM-TBM campaign a direct control of the helium temperature in some BUs is desirable for thermal-mechanical experiment of pebble bed assemblies. Additional equipment is requested in the Port Cell to control directly the inlet temperature and mass flow of few single units.

4.3.3.2 – HCLL TBMs systems

The TBS is composed of the following elements and their components:

- **The Test Blanket Module (TBM).** The test blanket module encompasses the function of the first wall, breeding blanket, shield and structure. Its principal functions in ITER are, besides serving as test module with the objectives defined in section 4.3.1.2, to remove surface heat flux and energy from plasma during normal and off-normal operational conditions, and to contribute to the shielding to the vacuum vessel and super-conducting coils. It is will be inserted in an Equatorial Port of ITER inside the vacuum vessel in front of the plasma (see section 4.3.2.2).
- **The TBM Port Plug (PP).** The PP contains the TBMs and provides thermal and neutronic insulation from the ITER basic machine. It allows the TBM replacement through the port itself. The front part (called “frame”) has also FW functions. The mechanical interface with the ITER machine will be provided by the PP rear part (“flange”); this flange is supported by the VV port extension. The PP provides also neutron shielding; a thick water cooled “shield” is placed behind the TBM. The pipes coming from the TBMs and crossing the PP up to the boundary with the interspace are considered part of the PP.
- **Interspace/Port Cell Piping (IP).** The Piping and loop components belonging to the main helium coolant and that connect the port plug to the interface to the Torus Coolant Water System (TCWS) (shaft wall).
- **The Helium Coolant System (HCS).** This system shall provide the He coolant at the characteristic of pressure, temperature and mass flow required by the TBM for the testing and for the extraction of the heat produced. The components (compressor, heat exchanger, etc.) and the piping allocated in the TCWS vault are part of this system (see section 4.3.3.1).
- **The Coolant Purification System (CPS).** Function of the CPS is to remove the Tritium that can permeate in the coolant and other impurities, and control the gas composition in the HCS (partial pressure of H2 and H2O). This system is connected to the HCS and its allocation is anticipated in the TCWS.
- **The PbLi ancillary system**.
- **Other systems:** measurement or Helium conditioning Systems, instrumentation control and management. These systems will be allocated in front of the VV Port and almost integrated in the IP.

The TBS presents boundaries to the ITER machine with: Vacuum Vessel, Shielding Blanket, Cryostat, Bio-shield, Remote Handling Equipment, Primary Heat Transport System, Vacuum Pumping System, Tritium Plant Building, Waste Treatment and Storage, General Testing Equipment and Hot Cells.

HCLL TBM Helium Coolant System (HCS)
The design of this system is shared with the HCPB TBM system (See section 4.3.3.1)

HCLL TBM PbLi ancillary system

The PbLi auxiliary system should ensure feeding and circulation of PbLi liquid metal in the breeding blanket and removal of tritium produced by a nuclear reaction in TBM. The container with the PbLi auxiliary system (dimensions H x L x W: 2.315 m x 2.19 m x 1.6 m) will be placed as close as possible to the TBM to prevent tritium permeation from the connection piping. A scheme of the circuit is shown on Fig. 4.3-10.

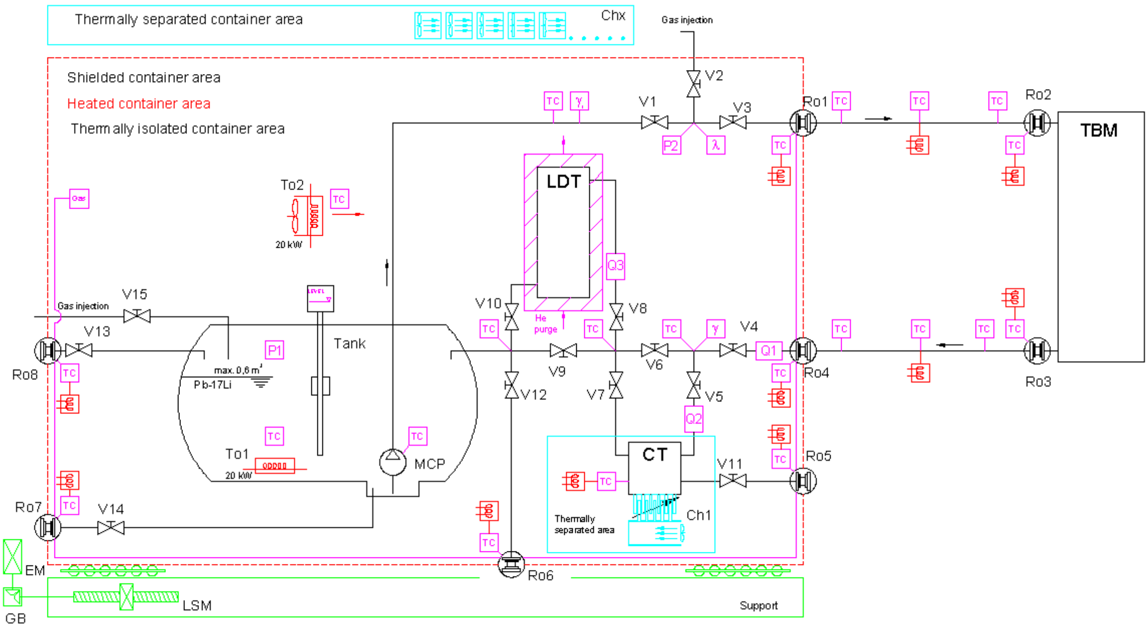


Fig. 4.3-10 Flow diagram of the PbLi ancillary circuit

From the functional point of view, the PbLi ancillary system is divided into the following parts: main circuit, detritiation unit and cold trap, dosing and sampling systems, heating and cooling systems, and shielding and insulation.

The PbLi circuit is a closed loop with a forced circulation of PbLi. From the tank that, at the same time, is a PbLi storage tank, liquid metal is pumped into the TBM where tritium is produced. The flow velocity in the PbLi system will be controlled in the range of 0.1 to 1 kg/s. PbLi outlet temperature from the TBM is 550°C. Tritium is removed from PbLi in a detritiation unit. Corrosion products and impurities are removed in a cold trap. [4.3-27] details the design of the key system components as well as their structure material and determines and describes the PbLi auxiliary system operating modes as filling, start-up, operation at nominal parameters, shut-down, emergency operation and sampling. Also, the limits and terms of the PbLi auxiliary system safe operation are defined. Requirements for the PbLi auxiliary system installation, testing and maintenance have been specified.

4.3.4. – Supporting R&D and validation Program prior to the installation in ITER

4.3.4.1- Supporting R&D in the EU

4.3.4.1.1 - Development and qualification of fabrication technologies

The fabrication of HCPB EUROFER plates with internal square-shape channels (First Wall, Cooling Plates) was preliminary demonstrated using uniaxial diffusion bonding of two grooved plates [4.3-28]. Alternative manufacturing processes for both concepts HCPB and HCLL are under examination and

optimization [4.3-29] within the EU, based on diffusion bonding achieved through Hot Isostatic Pressure (HIP).
In parallel to the manufacturing of plates, the assembly techniques have taken a growing part of the task effort, the HCPB and HCLL designs relying indeed on a large number and different type of welds for plate assembly.
A preliminary assembly assessment by Industry confirmed the principle feasibility of the TBM manufacturing if the proposed test programs and developments will be performed successfully. An exhaustive list of possible welding techniques for each joining has been produced, proposing conventional technologies (e.g. TIG) where applicable or welding processes requiring more development (e.g. laser welding) for some cases. A dedicated development program for plate assembly and welding development and characterization is on-going with the objective to define reference welding techniques and to produce an assembly feasibility mock-up by 2006.

Testing of Cooling Plate mock-ups will be also carried-out for validation of the fabrication technology and in particular the mechanical resistance of the diffusion bonding joint under endurance loading conditions (DEMO relevant) and cyclic loading conditions (ITER relevant).

4.3.4.1.2 – Tritium control and extraction technologies

In the HCLL concept, as a consequence of the low tritium Sieverts' constant in Pb-17Li, the high mean temperature of the He cooling system and the high surface available for tritium permeation, a high tritium permeation rate from the liquid metal into the primary cooling system (HCS, Helium Cooling System) is expected, at least in absence of highly efficient tritium permeation barriers. This affects the whole blanket tritium management consisting of i) tritium extraction from the liquid breeder (TES, Tritium Extraction System), ii) tritium removal (TRS, Tritium Removal System) from He stripping gas used in TES and iii) tritium removal from He primary coolant (CPS, Cooling Purification System).

Studies have shown [4.3-30] that the process of tritium extraction from the liquid metal (TES) and the subsequent tritium removal from He stripping gas (TRS) will not present, at least in principle, significant problems in terms of technical-economical feasibility.

However, some concerns are caused by the cooling purification system (CPS), which has to remove the significant amount of tritium permeated into the HCS loop from the blanket modules. The reference process for CPS is based on three steps: i) catalytic oxidation of Q_2 to Q_2O, ii) Q_2O removal by cold trap or molecular sieve beds, iii) removal of the remaining impurities in the coolant, mainly O_2, N_2, hydrocarbons and not oxidized Q_2 (microporous adsorbent beds operating at cryogenic temperature in the adsorption phase). In order to stay within reasonable CPS technological requirements (CPS feed flow-rate < 10^5 Nm^3/h), a compromise must be found between the important technical requirements: the maximum allowable specific tritium activity in the HCS, the required tritium permeation barrier efficiency (PRF, Permeation Reduction Factor) and the liquid metal mass flow-rate in the blanket modules. An attractive and technologically feasible combination of such parameters has been defined [4.3-30].

In the HCPB concept, the extraction of tritium from the ceramics/Be beds is performed by an independent low pressure (~1 bar) Helium flow. The mass flow is chosen as compromise between the two opposite requirements of minimising the permeation of T in the main He flow (which calls for a higher mass flow) and of reducing pressure drops and T dilution in the He stream. To recover the T that permeates into the high pressure He coolant, a purification system takes a bypass flow of about 0.1% of the main stream. Typical design values for this concept are a purge mass flow of about 0.4 kg/s (for a T production of 385 g/day) and a maximum allowable T partial pressure in the main coolant flow lower than 0.8 Pa.
The recovery technology taken as reference for the HCPB concept to separate T from the purge He is based on an oxidation step, followed by cold traps (-100°C) for removing the HTO and molecular

sieves (liquid N temperature) for the extraction of the residual gaseous H isotopes and other gaseous impurities [4.3-27].

4.3.4.1.3 – Solid breeder/multiplier development and irradiation

In the EU, production processes for the Li_4SiO_4 and Li_2TiO_3 are available. The fabrication processes have reached semi-industrial level with a potential production capability of 150 kg/years, also if up to now only limited batches of a ~10 kg per year were produced. The characterisation of these materials, especially the determination of the thermo-mechanical behaviour (of importance for the pebble beds) and under irradiation are still continuing in EU. One of the two materials could be considered as a back-up depending on future R&D (poor unexpected properties on one material, e.g. after long irradiation and heavy burn-up).

Another key issue of the HCPB blankets is the long term behaviour of the Be pebble under irradiation; especially their in-pile tritium release behaviour. Conclusive data are still missing (irradiation at high temperatures and up to 6000 appm in fission reactor will be available only in 2009 with the HIDOBE irradiation); in the mean time a systematic programme has been started to investigate and analyse this neutron multiplier material under DEMO relevant conditions, and fabrication of new grades of Be pebble are attempted to improve the T release and the mechanical characteristics.

The solid breeder/multiplier irradiation program concerns numerous fields of the HCPB blanket development like irradiation of Pebble Bed Assembly, tritium release from ceramic breeder and inventory, high dose Beryllium irradiation. The on-going and foreseen irradiation programs are summarized in the table below.

Table 4.3-9: Status of HCPB in-pile experimental program

Irradiation campaign	Description	Target burnup or damage (expected achievement date)
EXOTIC-8	Tritium release and post irradiation inventory in ceramic breeder	MTi 17% Li burn-up and OSi 11% Li burn-up, corresponding to > 20,000 h in DEMO. (irradiation achieved, PIE to be achieved 2005)
EXOTIC-9/1	Tritium release and post irradiation inventory in newly developed ceramic breeder (MTi at natural and 30% ^{6}Li enrichment, OSi at 20% ^{6}Li enrichment)	10 cycles (2006)
HICU	Irradiation tests on ceramic breeder at high neutron fluence (irradiation started in 2005)	20 dpa in OSi (2008)
PBA	In-pile testing of Pebbles Bed Assembly	2 dpa in Eurofer, Li burnup 2-3% (irradiation achieved end of 2004, PIE to be achieved in 2006)
HIDOBE	High dose irradiation program for Beryllium (irradiation started in 2005)	3000 & 6000 appm He, i.e. 18 & 36 dpa in Be (2009)

4.3.4.1.4 – HCLL Magneto-Hydrodynamics

The work on MHD experiments and modelling of the reference TBM design has been started (FZK). The aim of the MHD R&D studies is to analyze MHD issues in the HCLL TBM poloidal manifold and multi-channel effects in the BUs. An experimental mock-up consisting of a test section of 4 BUs will be installed in 2005 in the FZK/MEKKA facility and experiments will be performed with detailed measurements of liquid metal flow rates, velocities and pressures. In addition, some experimental work is on-going in Latvia for assessing the magnetic field effects on the corrosion behaviour of EUROFER in PbLi.

4.3.4.1.5 – Structural material R&D program

Currently, the main part of the qualification of EUROFER has been completed in the un-irradiated condition in the form of plates, bars and tubes as well as for various joints and welds. EU devoted also considerable efforts to a series of irradiation campaigns and the characterization of post-irradiation mechanical and micro-structural properties of EUROFER. In particular, irradiation temperatures lower than 400°C have been considered as a key issue (neutron hardening and embrittlement). The program in the EU is progressing with irradiations of EUROFER to a wide range of radiation damage: from 0.3 up to 70-80 dpa to fully cover all needs from the support to the radiation damage modeling program, up to engineering data base for ITER TBM and finally DEMO application.

Post-irradiation examinations (PIE) of the low to medium dose irradiations at 300°C up to 3 dpa at BR2 (Mol) and up to 12 dpa at HFR (Petten) show coherent results: an increase of the yield stress of 9CrWVTa heats and a shift of the DBTT as a function of the n-irradiation dose. Further studies to even higher dpa rate will be performed and modeling is being performed to get a better understanding of the underlying physical effects.

Irradiation experiments with the focus on specific manufacturing processes or joining techniques (TIG, laser, EB-welds and Diffusion-welded joints) are underway and will be complemented if needed for particular TBM design issues.

The next step in the development is the definition of EUROFER-2, which aims at further improvement of mechanical properties after irradiation, in particular the reduction of the DBTT. A decision on that can be taken by the end of 2005 at earliest when significant PIE results from irradiations up to 40 dpa will be available. Otherwise, the next milestone is end 2007 when the PIE results of the 80 dpa irradiation campaign at Dimitrovgrad, Russia, will be at hand. The third step (EUROFER-3), finally, aims towards a real low activation material with re-cycling times within 100 years. This goal seems to be technically feasible but has to be proven in industrial heats where the impurity control is extremely difficult to manage unless special clean production lines are available.

4.3.4.2- TBM validation program prior to the installation in ITER

In addition to sub-components testing, functional tests of TBM mock-ups integrating all components and reproducing complete He cycles and hydrogen (tritium simulating) extraction are mandatory; the main issues to be addressed are: i) validation of the TBM design performances (heat removal, H extraction), ii) validation of the fabrication route, and iii) reliability and safety with regard to ITER standards (structural integrity).

For the first step of this qualification, the size of reduced-scale HCLL and HCPB integrated mock-ups were chosen to be compatible with existing EU He facilities, while keeping relevant level of information. The HeFus3 Helium facility in ENEA/Brasimone offers currently operating conditions compatible with testing of about ¼ scale TBM mock-ups. In addition He purge loop will be available and a Pb-17Li loop at HCLL relevant conditions is under construction and shall be available end of 2005.

¼ scale TBM mock-ups shall be available around 2007-08 and a wide scope of tests is foreseen, like:

- Testing of heat removal from FW and Breeder (neutronic heat deposition simulated by electrical heating on FW and in breeder);
- Testing of hydrogen (tritium relevant) control and extraction (H injected at PbLi/He purge inlet);
- Testing and validation of ancillary circuit components (TES, TRS, CPS);
- Testing of instrumentation and validation of test procedures for ITER;
- Validation of the TBM system design before manufacturing 1/1 TBM prototypes.

After this validation step, some issues will need to be further tested at full TBM scale:

- Design qualification/optimization of the Back Manifold (BM) and total He flow: The BM features today a relatively complex design that will be preliminary optimized with the help of numerical simulation and tests on Plexiglas mock-ups with air. However, a full-scale qualification will be required because i) the He flow presents highly non-symmetrical paths in the back manifold, ii) a precisely tuned flow balance is crucial for the integrity of the TBM (components temperatures are often close to design limits), iii) the sensitivity of the thermal-hydraulics behaviour under different flow regime has to be known prior to TBM insertion in ITER, and iv) the total operating He flow cannot be oversized for covering design uncertainties, without impacting loop size, integration of He components and cost.
- Thermo-mechanical testing of the box structure: The thermo-mechanical validation of the box structure under ITER conditions requires an extensive use of the FE modeling and design codes. However, the final validation of the TBM structural behavior shall be carried-out on a full-scale prototype because i) the qualification shall be performed on a mock-up assembled with the reference welding/joining techniques (full-scale), ii) the verification of the impact of overall deformation of the box structure on local area is highly recommended before TBM installation in ITER, iii) in a ¼ TBM size mock-up, a central breeder cells cannot be represented; only a near full–scale testing can allow testing of this configuration, and iv) at the end, the sensitivity of a full-scale He flow on the thermo-mechanical behaviour of the box shall also be evaluated clearly (in ITER flow regime will be imposed by operating conditions; no sensitivity study will be possible).

- He TBM loop components qualification: The He loop components for ITER TBM shall be developed and qualified according to foreseen operating conditions (300-500°C, 8 MPa, pressure drop ~5 bars) and ITER standards (size, reliability, leak rate, etc.). For the most critical ones, their qualification at full-scale prior to ITER operation shall be ensured.

The question of He facility suitable to offer full-scale He testing (Q~1.6-1.7 kg/s) is currently under examination within the EU. Such tests are not envisaged before 2009-2010.

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4.4 Japan Proposal To TBWG
4.4.1 DEMO Study and Testing Strategy in ITER
(1) Roadmap and Strategy of DEMO Blanket Development in Japan

The Fusion Council of Japan has established the long-term R&D program of the blanket development in 1999. The development of blankets is performed according to this program in Japan. In the program, JAERI has been nominated as a leading institute of the development of solid breeder blankets, in collaboration with universities, as the primary candidate blanket for the fusion power demonstration plant[4.4.1-1], while, universities and NIFS are assigned mainly to develop advanced blankets for future blanket options of the fusion power demonstration plant and commercial power plants.

Figure 4.4.1-1 shows the roadmap of blanket development in Japan. ITER blanket module testing is regarded as the most important milestone, by which integrity of candidate blanket concepts and structures are qualified. In parallel with the blanket development, material development is being performed on major candidate materials. The blanket of the fusion power demonstration plant will be decided by the qualification data of the test blanket module (TBM) by ITER blanket module testing and irradiation data by International Fusion Material Irradiation Facility (IFMIF). The fusion material development is being performed on reduced activation ferritic/martensitic steel (RAFM)[4.4.1-2], as the primary candidate structural material, and silicon carbide composite (SiC_f/SiC)[4.4.1-3], vanadium alloy[4.4.1-4], ODS ferritic steel as the advanced structural materials, and also on functional materials such as tritium breeder, neutron multiplier, insulation materials and so on. The blanket development has been, and is being performed under interactive relationship with material development.

As the primary candidate blanket for the fusion power demonstration plant, solid breeder test blankets made of RAFM cooled by helium and water are being developed by JAERI with cooperation of universities. In all essential issues of blanket development, elemental technology development has been almost completed and is now stepping further to the engineering R&D phase, in which scalable mockups of solid breeder test blanket modules will be fabricated and tested to clarify the total structure integrity for final specification decision of test blanket modules.

As the advanced blanket options, solid breeder blanket module with SiC_f/SiC structure cooled by high temperature helium gas, liquid LiPb breeder with SiC inserts cooled by helium and its dual-coolant option, liquid Li self-cooled blanket made by V alloy and molten salt self-cooled blanket made by RAFM are under development by universities and NIFS with cooperation. Key issues have been addressed and critical technologies are being developed.

The development of blankets in Japan has shown sound progress on both of solid and liquid breeder blankets under coordinated domestic development programs, for both of primary and advanced options. Japan is investigating the possibility of testing all types of blankets under Test Blanket Working Group (TBWG) framework with both of JAERI and universities/NIFS involvements.

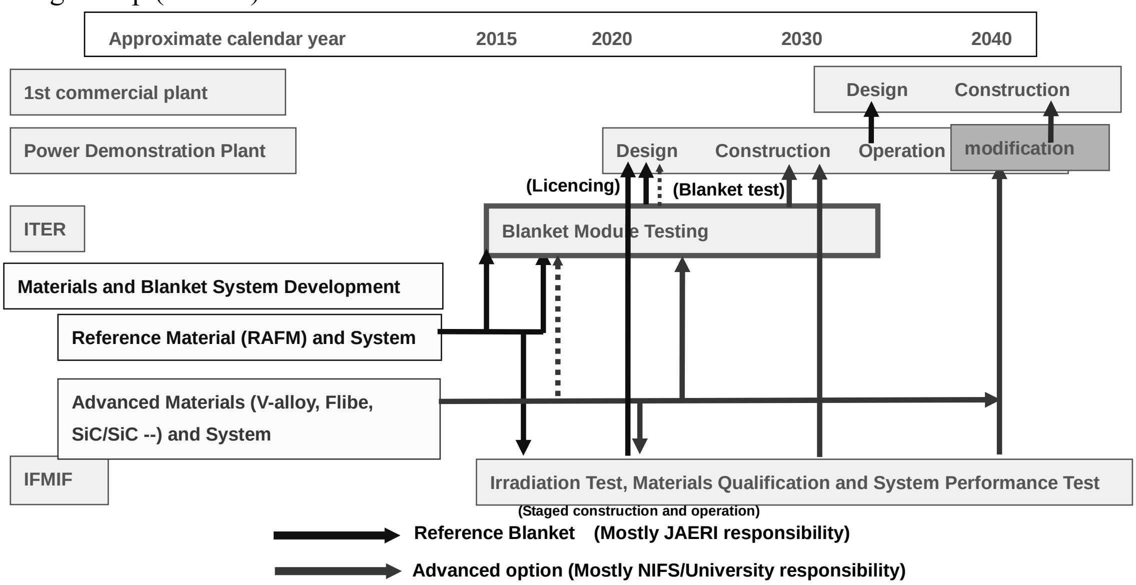

Fig. 4.4.1-1. Road map for materials and blanket development in Japan.

(2) DEMO Blanket Design as the Development Target of TBMs

One of conceptual designs of the DEMO reactors has been performed by JAERI, aiming at the achievement of similar plasma performance to SSTR[4.4.1-5], such as fusion power, Q value, and neutron wall load with more economical merits[4.4.1-6]. In line with the reactor design, DEMO blanket design has been conducted[4.4.1-1], as the advanced solid breeder blanket for DEMO, cooled by the supercritical water. The basic structure of this blanket design is regarded as common structure for basic coolant option of high pressure water (280 – 320°C, 15 MPa) or backup coolant option of high pressure Helium (300 – 500°C, 8 MPa). Major design parameters of the DEMO blanket are summarized in Table 4.4.1-1. Load conditions and applied materials are similar to those of SSTR.

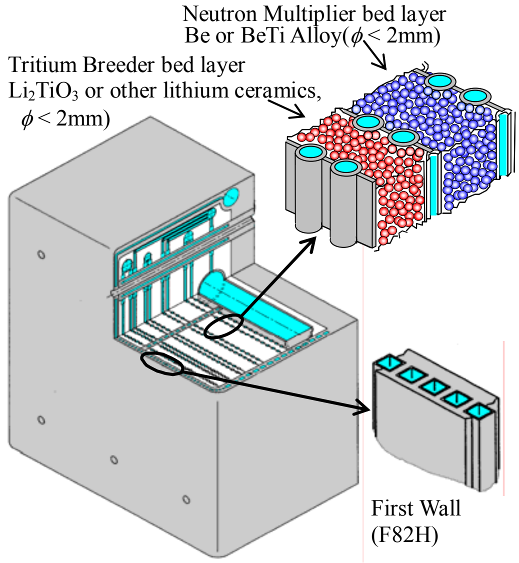

Fig. 4.4.1-2. Schematic structure of one of cadidate DEMO blanket.

Conceptual layout of the blanket module is shown in Fig. 4.4.1-2. Dimension of the blanket module is about 2 m high, 2 m wide, and 0.6 m thick. Reduced activation ferritic steel, F82H, which is currently under development by JAERI, was selected as the structural material. Ceramic breeder and beryllium neutron multiplier are packed in a form of small pebbles in a layered structure as shown in the figure. Li_2TiO_3 was selected as the primary candidate tritium breeder material in this design work and other lithium ceramics are regarded as the backup candidate breeder material. With respect to the neutron multiplier material, Beryllium was selected as the primary candidate and inter-metallic compounds, such as $Be_{12}Ti$, are regarded as the advanced neutron multiplier materials.

Table 4.4.1-1. Typical Design Parameters of DEMO Blanket.

Item	Value
Surface heat flux	0.5 (peak 1) MW/m^2
Neutron wall load	3.5 (peak 5) MW/m^2
Neutron Fluence	>10 MWa/m^2
Coolant Material	Supercritical water
Coolant Pressure	25 MPa
Inlet / Exit Temperature	280 /510 °C
Tritium Breeding Ratio	>1.05
Structural Material	RAFS* (F82H)
Tritium Breeder	Li_2TiO_3 or other Li ceramics
Neutron Multiplier	Be or BeTi alloy

* Reduced activation ferritic steel

Table 4.4.1-2. Estimated TBR for DEMO Blanket.

Materials	Li_2O / Be		Li_2TiO_3 / Be		Li_2TiO_3 / $Be_{12}Ti$				
^{6}Li Enrichment	30%	90%	30%	90%	90%	30%	90%	30%	90%
Packing Structure	Breeder / Multiplier Separate					Breeder + Multiplier Mix			
Temperature Limits	Breeder 900°C				900°C	600°C		900°C	
	Multiplier 600°C				900°C				
Local TBR	1.53	1.56	1.41	1.52	1.37	1.24	1.35	1.35	1.43
Coverage Requirement*	69%	67%	74%	69%	77%	85%	78%	78%	73%

* Required coverage fraction of the plasma facing surface of the breeding region of the blanket in the total area of the plasma facing surface, to achieve net TBR, 1.05.

Net TBR is required to be more than 1.05 for stable operation and accumulation of startup fuel for next fusion plant[4.4.1-7]. Table 4.4.1-2 summarizes estimated values of TBR with major candidate options of materials, ^{6}Li enrichment and structure. Li_2TiO_3 and $Be_{12}Ti$ are expected to have better compatibility with water in high temperature than Li_2O and Be. Even in case where Li_2TiO_3 or $Be_{12}Ti$ are applied, net TBR satisfied more than 1.05[4.4.1-1]. As the design study of water cooled solid breder blankety for DEMO, thermo-mechanical analyses of blanket module structure, tritium inventory analysis, evaluation of tritium permeation[4.4.1-8], configuration and thermal Efficiency evaluation of power plant have been conducted.

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4.4.2 TBMs Design

(1) Design of TBMs

The primary candidate blanket is supposed to be solid breeder blankets for DEMO which will be planed next to ITER. Therefore, Japan is proposing to deliver Japanese solid breeder TBMs from the first day of ITER operation and to perform module testing. Therefore, design work has been performed on solid breeder TBMs intensively. As described in subchapter 5.4, Japan is planning to have 4 TBMs, for system check and environment measurement (TBM W-1, H-1), neutronics test (TBM W-2, H-2), tritium production test (TBM W-3, H-3) and thermo-mechanical test (TBM W-4, H-4). TBM W-1 and H1 have the same structure as TBM W-4 and H4. Dimensional variation will be made to TBM W-2, H-2, W-3 and H3, based on the design of TBM W-4 and TBM H-4. Therefore, detailed design of TBMs is mainly performed for TBM W-4 and TBM H-4, TBMs for thermo-mechanical performance test. Design of solid breeder TBMs have been performed under the design conditions stated in Chapter 2, to represent major structure features of the design of DEMO blanket described in the previous sub-chapter[4.4.2-1].

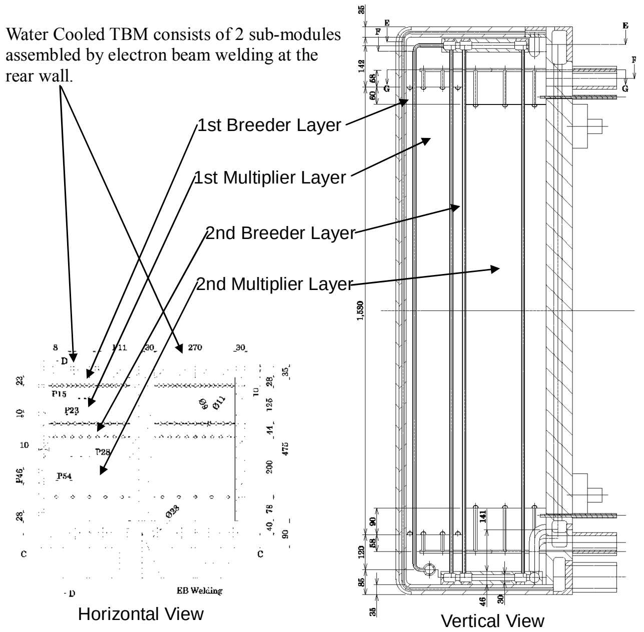

Fig. 4.4.2-1 Structure of Typical Cross Sections of Water Cooled TBM[4.4.2-2]

The design of Japanese solid breeder TBMs has the following features.

(a) First wall and side walls are fabricated in near-net-shape by Hot Isostatic Pressing (HIP) using RAFM, for realizing built-in cooling channel structure.

(b) Vertical slots was adopted to split the blanket module into smaller sub-modules, in less than 50 cm intervals, for the purpose of reduction of electro- magnetic force in vertical disruption event (VDE) and increasing the endurance to internal over-pressure in the case of coolant ingress in the module[8]. Sub-modules are integrated at rear wall by welding.

(c) For Helium Cooled TBM, By-pass flow, which is merged to the outlet flow after temperature homogenization by heat exchanger, is planned to adjust the TBM operation temperature to DEMO relevant condition. In addition, coolant manifold for each sub-module is independent to each other. Correspondingly, the heat exchanger for by-pass flow of each sub-module can be independent. This design allows the flexibility of testing different concepts of sub-modules.

(d) Breeder and multiplier are packed in layered pebble beds whose partition walls are integrated with cooling pipes. The internal structure is designed according to the same concept as the breeding blanket for fusion power demonstration plant[1].

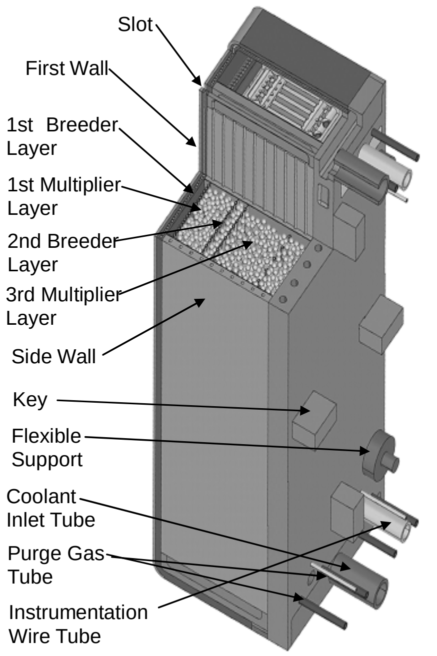

Fig. 4.4.2-2 Three dimensional view of the structure of Water Cooled TBM[4.4.2-1]

Table 4.4.2-1 summarizes the major specification of Water Cooled and He Cooled Solid Breeder TBMs proposed by Japan. Detailed structure design has been performed on TBMs for thermo-mechanical tests[4.4.2-2].

Figure 4.4.2-1 shows the typical drawing of vertical and horizontal cross sections of Water Cooled TBM[4.4.2-2]. In the water-cooled TBM, two sub-modules have same box structures and internal structures. The first wall made of F82H has built-in rectangular cooling paths. As for internal structure, it has multi-layer pebble beds structure same as the DEMO blanket. Breeder and neutron multiplier formed by small pebbles are packed separately in inner box structure made of F82H thin plates, which is separated into four layers by cooling panels. The cooling panel consists of F82H tubes, which are the inner diameter of 9mm and the thickness of 1.5mm, and thin plates connecting adjacent tubes. The inner box structure is welded to the first wall and the back plate. The thickness of each layer and pitches between tubes at each cooling panel were optimized to experience similar level of temperatures and possibly stresses as those in the DEMO blanket according to the transient performance analyses of temperature evolution and tritium generation / release performance. Figure 4.4.2-2 shows the schematic three dimensional drawing of

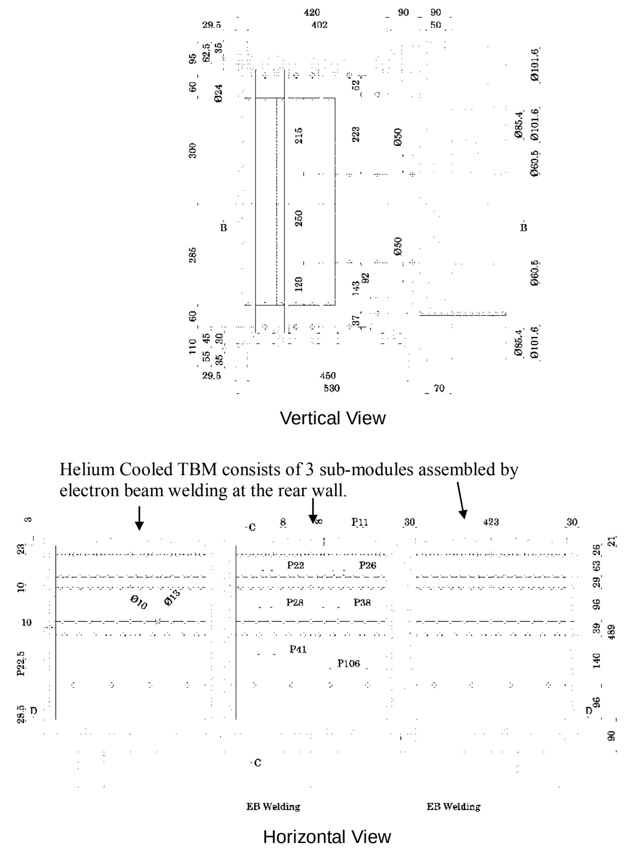

Fig. 4.4.2-3 Structure of Typical Cross Sections of Helium Cooled TBM[4.4.2-2]

Table 4.4.2-1 Major Specification of Water Cooled and Helium Cooled Solid Breeder TBMs[4.4.2-2]

Items		Unit	Water Cooled		Helium Cooled
			Reference	Advanced	
Structural Material			F82H		F82H
Coolant			Pressurized Water	Supercritical Water	Helium Gas
Multiplier	Primary		Be		Be
	Temperature Limit	°C	< 600		< 600
	Advanced		$Be_{12}Ti$ or BeTi alloy		$Be_{12}Ti$ or BeTi alloy
Breeder	Primary		Li_2TiO_3		Li_2TiO_3
	Temperature Limit	°C	< 900		< 900
	Backup		Other Li Ceramics		Other Li Ceramics
Area of First Wall		m^2	0.68 × 1.94		1.49 × 0.91
TBM Thickness		m	0.6		0.6
Surface Heat Flux		MW/m^2	0.3		0.3
Nuetron Wall Load		MW/m^2	0.78		0.78
Total Heat Deposit		MW	1.55		1.61
Total Tritium Production		g/FPD	0.156		0.180
Coolant Pressure		MPa	15.5	25.0	8.0
Coolant Inlet Temperature		°C	280.0	360.0	300.0
Coolant Outlet Temperature		°C	325.0	380.0	500.0
Coolant Flow Rate		kg/s	6.15	6.53	1.80
Coolant Bypass Flow Rate		kg/s	—	—	0.51

Water Cooled Solid Breeder TBM composed by established drawings[4.4.2-1].

Figure 4.4.2-3 shows the typical drawing of vertical and horizontal cross sections of He Cooled TBM[4.4.2-2]. As for the helium-cooled TBM, one or two sub-modules may have different structures to the Japanese sub-module(s), which are designed and developed by other parties. The sub-module proposed by Japan has almost same box structure and multi-layer internal structure as that of the water-cooled TBM. It is noted that there is a space of 96mm thickness in front of the back plate where neutron multiplier pebbles are not packed to lay coolant connecting pipes, helium purge gas connecting pipes and cable conduits for instrumentations. The thickness of each layer and pitches between tubes at each cooling panel were also optimized same as the water-cooled TBM. Table 4.4.2-2 shows major materials and weight which compose ceramic breeder test blankets of Japan. Total weight is about 2.5 tons in both TBMs. As the important information on interface condition with the Shield Plug of the Common Frame, specifications of penetration pipes through the Shield Plug is listed in Table 4.4.2-3 and 4.4.2-4 fro Water Cooled and He Cooled Ceramic Breeder TBM. Both interface conditions meet the limitation of the penetration pipe numbers and dimensions from the design of the Shield Plug. In the case of He Cooled Ceramic Breeder TBM, there is a possibility to install more numbers of penetration pipes for independent sub-module cooling. However, in this case, the pipe diameters are limited to smaller size than 35 mm^{OD} to meet the Shield Plug requirement.

Table 4.4.2-2 Major materials and weight of Japan Ceramic Breeder TBMs

Material	Water Cooled Ceramic Breeder TBM	He Cooled Ceramic Breeder TBM
F82H	2041 kg	1961 kg
Be Armor	23 kg	7 kg
Li_2TiO_3 pebble	91kg	104 kg
Be pebble	430 kg	305 kg
Coolant	38 kg	0.5 kg
Total	2624 kg	2378 kg

Table 4.4.2-3 Interface pipe specifications of Water Cooled Ceramic Breeder TBM to Shield Plug.

	Coolant		Purge gas		Cable Conduit	
	Inlet	Outlet	Inlet	Outlet	Thermocouple Strain gauge	Micro-fission chamber
Fluid	Water	Water	Helium	Helium	-	-
Press.	15.5MPa	15.5MPa	0.1MPa	0.1MPa	-	-
Temp.	280^{o}C	325^{o}C		450^{o}C	-	-
Pipe size	101.6mm^{OD} 76.2mm^{ID}	101.6mm^{OD} 76.2mm^{ID}	25.4mm^{OD} 18.4mm^{ID}	25.4mm^{OD} 18.4mm^{ID}	76.3mm^{OD} 57.3mm^{ID}	13.8mm^{OD} 7.8mm^{ID}
Number	1	1	1	1	4	4
Dest.	TBM cooling system		TBM tritium recovery		Control panel	

Table 4.4.2-4 Interface pipe specifications of He Cooled Ceramic Breeder TBM to Shield Plug.

	Coolant		Purge gas		Cable Conduit	
	Inlet	Outlet	Inlet	Outlet	Thermocouple Strain gauge	Micro-fission chamber
Fluid	Helium	Helium	Helium	Helium	-	-
Press.	8.0MPa	8.0MPa	0.1MPa	0.1MPa	-	-
Temp.	300^{o}C	472^{o}C		450^{o}C	-	-
Pipe size	101.6mm^{OD} 85.4mm^{ID}	101.6mm^{OD} 85.4mm^{ID}	25.4mm^{OD} 18.4mm^{ID}	25.4mm^{OD} 18.4mm^{ID}	60.5mm^{OD} 50.0mm^{ID}	13.8mm^{OD} 7.8mm^{ID}
Number	1 (3*)	1 (3*)	3	3	6	6
Dest.	TBM cooling system		TBM tritium recovery system		Control panel	

* Depending on the number of independent sub-modules, there is a possibility of more pipe numbers only with smaller diameter (about 35mm^{OD}).

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4.4.3 TBM Systems

(1) Cooling System for Water Cooled TBM

Design conditions of the cooling system for the JA water-cooled TBM are summarized in Table 4.4.2-1. The first wall area facing the plasma is 0.68m × 1.94mH with 20 mm-wide gap between the TBM and the common frame. Thermal power (removal heat) of the TBM is 1.55 MW with maximum 0.5 MW/m^2 (0.3 MW/m^2 average) of surface heat flux and nuclear heating due to neutron wall loading of 0.78 MW/m^2. Primary coolant conditions are 280 °C and 325 °C at TBM inlet and outlet, respectively, and pressure of 15.5 MPa. The flow rate of the primary coolant water is 6.15 kg/s. About 5 % of the primary coolant flow is bypassed and circulated through a purification system (CVCS: chemical and volume control system). The thermal power of the TBM is transferred to the ITER secondary coolant water of 35/75 °C at a heat exchanger inlet/outlet, respectively, and 0.5 MPa. Demonstration of electricity generation utilizing the power from the water-cooled TBM is planned. When the electricity generation is demonstrated, the electricity generation system is connected, via the steam generator, as an intermediate loop to the ITER secondary cooling system. For the common frame, water coolant of the ITER first wall/blanket will be used.

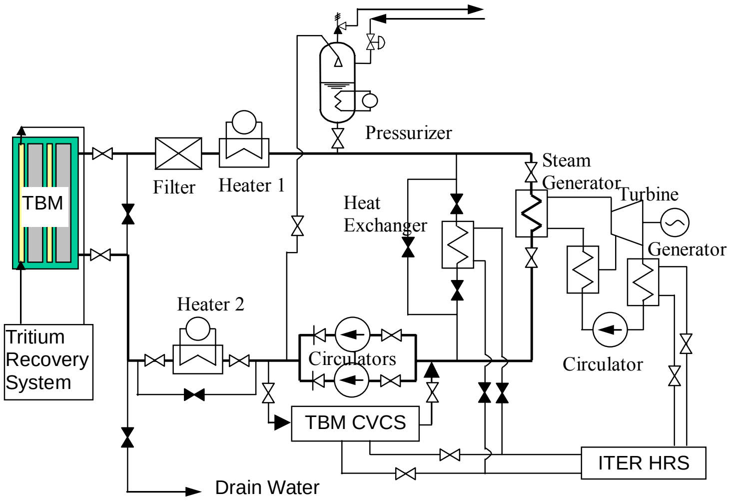


Fig. 4.4.3-1 Flow Diagram of Cooling System of Water Cooled TBM

The flow diagram of the cooling system for the water-cooled TBM is shown in Fig. 4.4.3-1. Major components in the main loop are a steam generator, a main heat exchanger, a circulation pump, a pressurizer, a heater 1 and a heater 2. The high-pressure primary coolant flows inside tube and low-pressure secondary coolant outside tube. For circulation pump system, two circulation pumps are planned for redundancy in case of a pump trip accident was reduced to one pump because of the limited layout space. The pressurizer is designed to accommodate the volumetric change of water coolant due to its temperature rise from room temperature (20 °C) to 300 °C. The heater 1of the same power as the TBM, 1.17 MW, is equipped between the TBM outlet and the steam generator inlet and used during the demonstration of electricity generation. Namely, the heater 1 will provide the power to compensate the power reduction of the TBM during the dwell time, thus to keep the water temperature at steam generator inlet constant and to avoid excessive load change for the turbine system. The heater 2 of 450 kW is equipped to warm-up the system by temperature rising rate at about 50 °C/h and also to adjust the TBM inlet temperature during operation. Though it is not shown in the diagram, a suppression tank is to be included to avoid an excessive pressure increase in case of in-TBM water leak.

(2) Cooling System for Helium Cooled TBM

Design conditions of the cooling system for the JA helium-cooled TBM are summarized in Table 4.4.2-1. The first wall area facing the plasma is 1.49m × 0.91mH with 20 mm-wide gap between the TBM and the common frame. Thermal power (removal heat) of the TBM is 1.61 MW with maximum 0.5 MW/m^2 (0.3 MW/m^2 average) of surface heat flux and nuclear heating due to neutron wall loading of 0.78 MW/m^2. Primary coolant conditions are 300 °C and 500 °C at TBM inlet and outlet, respectively, and pressure of 8 MPa. The flow rate of the primary coolant is 1.8 kg/s. About 0.2 % of the primary coolant flow is bypassed and circulated through a purification system. The thermal power of the TBM is finally transferred to the ITER secondary coolant water of 35/75 °C at a heat exchanger inlet/outlet, respectively, and 0.5 MPa. For the common frame, water coolant of the ITER first wall/blanket will be used.

The flow diagram of the cooling system for the helium-cooled TBM is shown in Fig. 4.4.3-2. Major components in the main loop are a main heat exchanger, a circulator and a heater. The high-pressure primary coolant flows inside tube and low-pressure secondary coolant outside tube. The heater about 200 kW is equipped to warm-up the system and also to adjust the TBM inlet temperature during operation. The flow diagram of the helium purification system is also shown in Fig. 4.4.3-3.

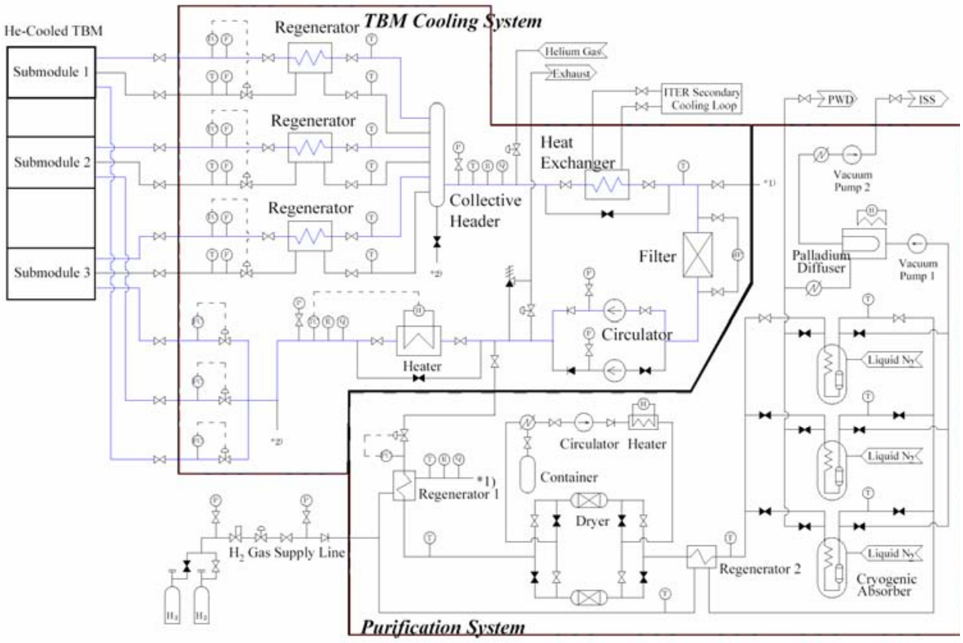


Fig. 4.4.3-2 Flow diagram Cooling System of Helium Cooled TBM

(3) Layout Plan of Cooling Systems

A layout plan of the primary cooling systems for JA Water Cooled and He Cooled TBMs is shown in Fig. 4.4.4-3. As seen from the figure, its footprint is about 7 m x 6.861 m with 5 m height for all cooling systems. Though the space is very tight resulting in complicated pipe routing and only 0.5 m-wide space left around each cooling system for access during installation and maintenance, the cooling systems could be installed into the TCWS vault. It should be noted that the drain/surge tank (1.5 mOD x 2.0 mH) and the suppression tank (10 m^3) are not included in the above layout. Therefore, the space for these tanks is required, probably in a lower space in the building.

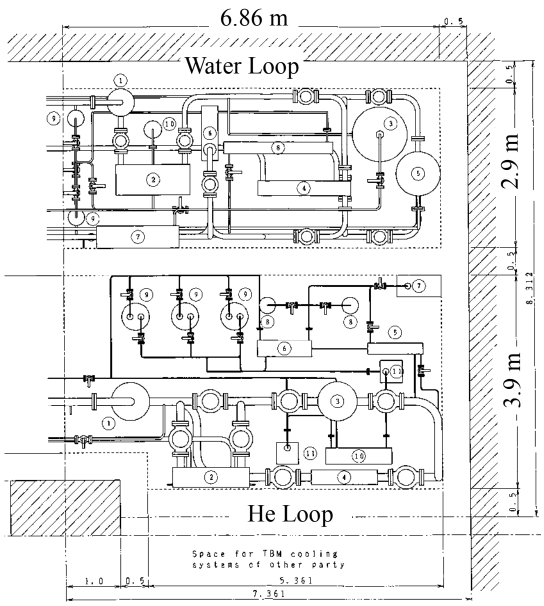


Fig. 4.4.3-3 Layout plan of cooling systems of Water Cooled and He Cooled TBMs

(4) Tritium Recovery and Measurement Systems

Tritium recovery system for the TBM has very important roll to recover tritium from helium purge gas of the TBM, clean up humidity or vapor from helium purge gas, and supply clean purge gas to the TBM. At the same time, it is required to transfer recovered tritium in the most suitable form to ITER tritium plant with minimal impact to ITER tritium plant systems.

The major functions required to the tritium recovery system of the TBM are,

1) to measure gas composition of helium purge gas for the purpose of the evaluation of TBM function,
2) to recover H_2 and HT, H_2O and HTO from the helium purge gas,
3) to cleanup purge gas (humidity and vapor) and condition.

JA water-cooled and helium-cooled TBM's utilize similar helium purge system to recover generated tritium. Here, one common tritium recovery system for both TBM's will be installed with multi-point gas analysis system for obtaining tritium release data and system control. However, some of gas analysis equipments are to be installed within a transfer cask (about 1 m^3) in front of the test port bioshield plug of each TBM to obtain the data independently.

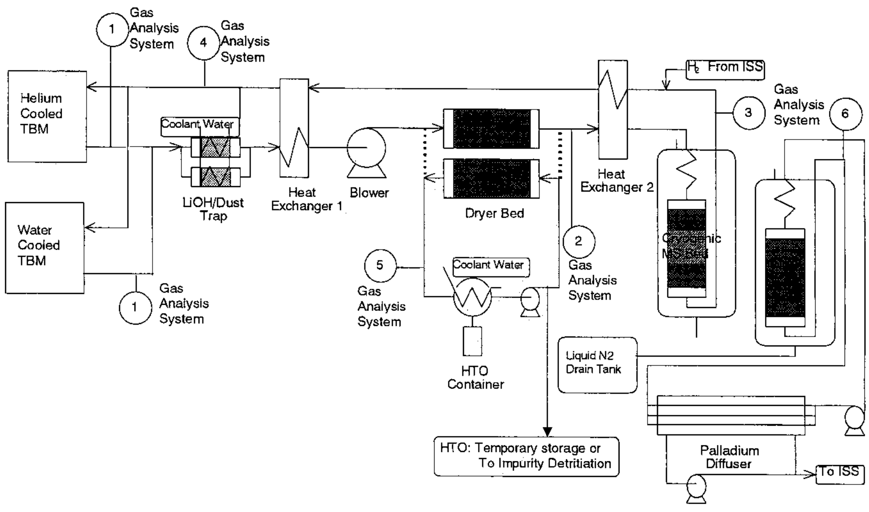


Fig. 4.4.3-4 Schematic flow diagram of JA TBM tritium recovery system

Schematic flow diagram of JA TBM tritium recovery system is shown in Fig. 4.4.3-5. The tritium recovery by helium purge gas separate from the main coolant is the same for the water-cooled and helium-cooled TBM's. The tritium producing capability, thus the purge gas conditions, are also similar for both TBM's. The tritium recovery system consists of LiOH/LiOT vapor trap, purge gas cooler, cryogenic molecular sieve bed, palladium diffuser, purge gas heater, transfer pump and gas analysis systems. The schematic diagram of gas analysis system is shown in Fig. 4.4.3-6. The gas analysis system equipped at the outlet of each TBM (in front of the bioshield plug) consists of moisture detector, ion chamber, gas chromatography and small dryer bed. These detectors will be set to identify H_2, HT, H_2O and HT concentration, separately. Also, appropriate detectors will be set to the important analysis points of tritium recovery system components for monitoring of the TBM tritium recovery system performance.

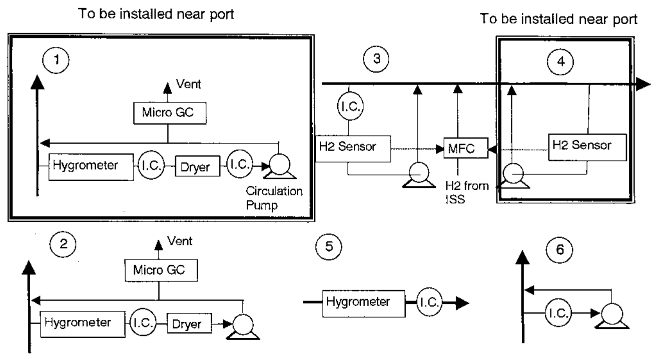


Fig. 4.4.3-5 Gas analysis system in JA TBM tritium recovery system

(5) Layout Plan of Tritium Recovery and Measurement System

Components of Tritium Recovery System are contained in glove boxes and installed in Tritium Building. Tritium Measurement System for TBM inlet and outlet purge gas is planed to be installed in a temporary transfer cask (1m × 1m × 1m) in the port cell area.

4.4.4 TBM Analysis

(1) One Dimensional Neutronics and Thermal Analysis

Figures 4.4.4-1 and 4.4.4-2 shows one dimensional TBR and temperature analysis for Water Cooled and He Cooled TBMs. As described in Sub-chapter 4.4.2, the basic configuration of module structure is DEMO blanket structure. In the case of TBM W-4 and H-4 (thermo-mechanical TBM), the highest temperature of structural material of FW, breeder layer and multiplier layer are decided to be the same as the DEMO blanket, 550, 600, 900 °C, respectively. To achieve the similarity of such temperature distribution to DEMO blanket, thickness of each layers were studied as design parameters. Figures 4.4.4-1 and 4.4.4-2 are obtained using decided set layer thickness. Total TBR of 1.42 and 1.47 are obtained for Water Cooled TBM and He Cooled TBM by one dimensional calculation.

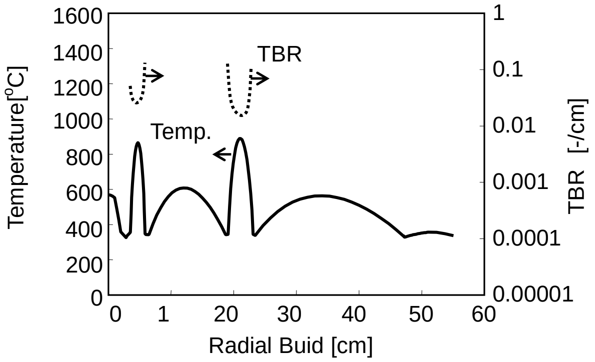


Fig. 4.4.4-1 TBR and Temperature Distributions in Radial Direction By One Dimensional Analysis for Water Cooled TBM

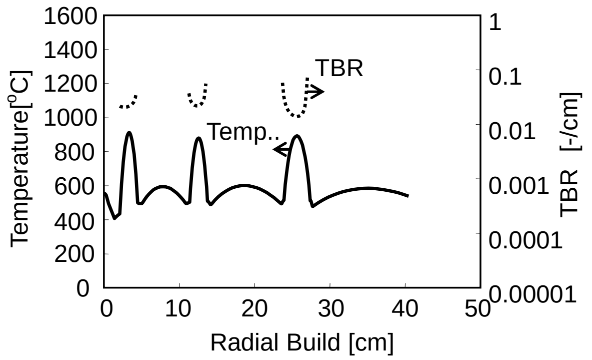


Fig. 4.4.4-2 TBR and Temperature Distributions in Radial Direction By One Dimensional Analysis for He Cooled TBM

(2) Two Dimensional Neutronics Analysis and Induced Activity and Decay Heat Analysis

Nuclear analyses were performed on Water Cooled Solid Breeder TBM and He Cooled Solid Breeder TBM by 2D calculation, by using DOT3.5, FUSION-40 (JENDL2.1 edition) for neutron transportation, APPLE-3 for the evaluation of nuclear reaction rates and ACT-4, CROSS-LIB, CHAIN-LIB and GAMMA-LIB ('90 edition) for the evaluation of induced activation rates[4.4.4-1]. Geometry of calculation model consists of TBM and its internal structure and common frame. By the calculation results, it became clear that neutron flux exists to the direction from TBM to common frame side. Consequently, integrated value of TBR is 1.13 and 1.23 for Water Cooled TBM and He Cooled TBM. As can be seen in the former section, TBR is evaluated to be decreased compared to one dimensional calculation, because of absorption of neutron by SS/Water structure of common frame. The decay heat after one year is about one order of magnitude smaller than after one day. The decay heat of structural material, F82H, at the front part of the first wall is about $2x10^{-2}$ W/cm^3. The decay heat values for first breeder layer and the first Be layer are about $1x10^{-2}$ W/cm^3 and $5x10^{-5}$ W/cm^3. Induces activity reduces in exponential function of radial location. The plotted value of induced activity of the first wall is for the fluence of about 0.3 MWa/m^2. The induced activity is lower than the DEMO condition, where the neutron fluence is about 7.5 MWa/m^2, however, the dose rate is relatively high and the necessity of remote access handling and shielding is carefully evaluated further.

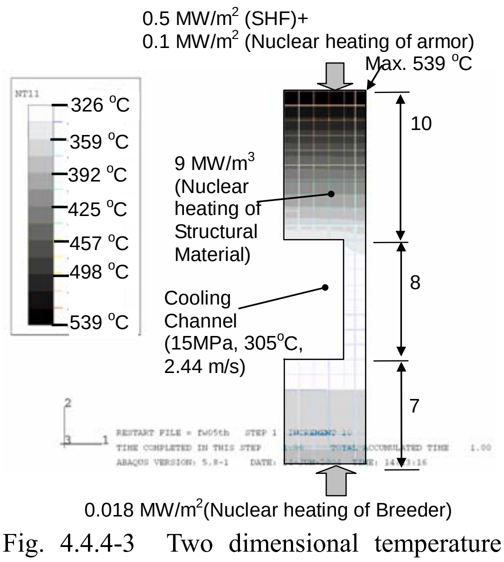


Fig. 4.4.4-3 Two dimensional temperature distribution in the cross section of first wall of Water Cooled TBM

(3) Thermo-mechanical Analysis

The structure design showed sound progress to establish detailed drawings with consideration of coolant route, fabrication and assembly procedures of modules including pebble packing. Thermo-mechanical integrity was evaluated by FEM analysis. Thermo-mechanical endurance is one of the most important test issues. Figure 4.4.4-3 shows the temperature distribution in the first wall of Water Cooled TBM evaluated by two dimensional thermo-mechanical analysis. To obtain the similarity of temperature on the first wall structure, the front part thickness was determined to be 10 mm as seen in Figure 4.4.4-3. Consequently, the highest temperature of the structural material, 539 ^{o}C, which satisfies the F82H design window, appeared at the most distant part of plasma side surface from cooling channel. Figure 4.4.4-4 shows the stress distribution in the first wall of Water Cooled TBM evaluated by two-dimensional thermo-mechanical analysis. By stress analysis, it was shown that the stress range was within elastic range. The highest TRESCA stress 359 MPa appeared at the same place as the highest temperature appeared. This stress value was evaluated to satisfy 3Sm value for F82H. Figures 4.4.4-5 and 4.4.4-6 show temperature and stress distribution in the first wall of He Cooled TBM evaluated by two dimensional thermo-mechanical analysis. In the case of He Cooling TBM, heat transfer coefficient of coolant He flow in the first wall channel is small. Consequently, the front part thickness of the first wall structure need to be 4 mm as seen in Fig. 4.4.4-5, therefore, peak TRESCA stress is smaller value 270 MPa than Water Cooled TBM case.

(4) Analysis on Tritium Behavior in TBMs

Tritium behavior in TBM module is one of most important issues for both of evaluation of tritium production function and tritium management of TBM. Tritium behavior in TBM consists of tritium generation by nuclear reaction, tritium release and inventory in solid breeder and tritium permeation to coolant water or helium through cooling tube wall. For tritium generation, nutronics calculation clarified the tritium generation rate distribution in breeder layers which is seen in Figures 4.4.4-1 and 4.4.4-2 and two-dimensional calculation result. Based on these result, tritium release and inventory performance were analyzed by using the calculation model stated by Nishikawa et al.[4.4.4-2]. By the analysis, it was estimated that tritium inventory in the first breeder layers of Water Cooled TBM and He Cooled TBM, become saturated in 3 or 4 pulse operation, which means the tritium residence time is

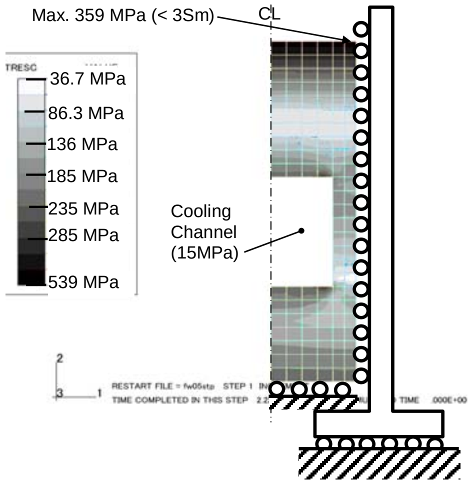


Fig. 4.4.4-4 Two dimensional stress distribution in the cross section of first wall of Water Cooled TBM

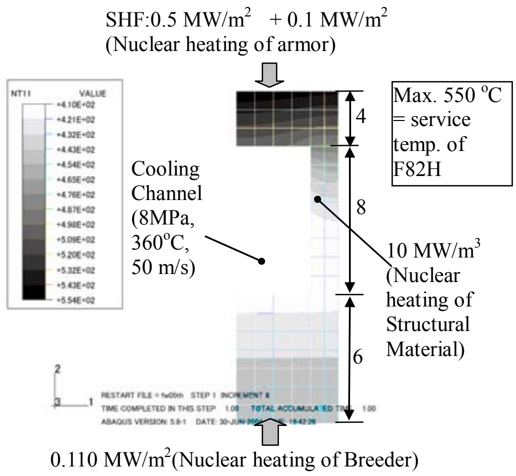


Fig. 4.4.4-5 Two dimensional temperature distribution in the cross section of first wall of He Cooled TBM

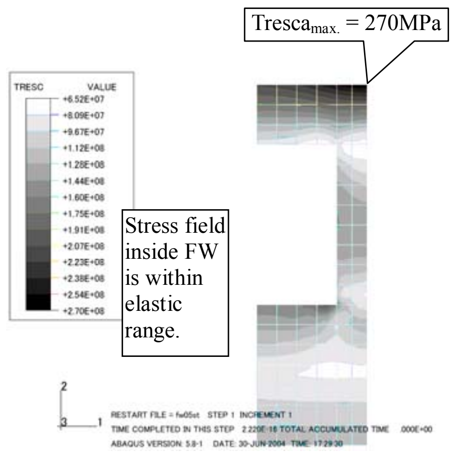


Fig. 4.4.4-6 Two dimensional stress distribution in the cross section of first wall of He Cooled TBM

less than 1200 to 1600 sec. The similar behavior is estimated in the second breeder layer for both TBM. Tritium permeation was also evaluated to be negligible if the permeation barrier with decontamination factor of about 100 was applied.

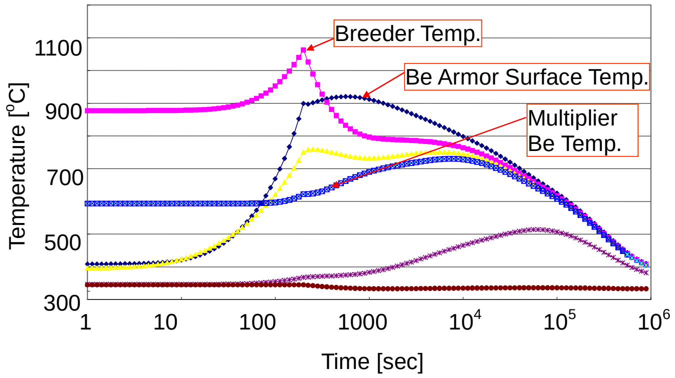


Fig. 4.4.4-7 Temperature evolution of TBM in the case where Be armor fall occurs at 900°C and plasma shut down occurs.

(5) Safety Analysis

Objective of this safety analysis is to evaluate the substantial safety of the Water Cooled TBM in such aspects as establishment of post accident cooling in the TBM, hydrogen gas generation by Be-steam reaction, and pressure increase and spilled water amount by Ingress of Coolant Event (ICE). The evaluation was performed in conservative conditions to show the upper bound of consequences in significant events, which can be assumed by the similarity of the safety analysis of the ITER shielding blanket. It is important to perform systematic identification of safety issues for detailed safety analysis on the TBM. The reference events applied in this safety analysis are specified in the ITER reference[4.4.4-3]. Most important safety analysis scenario can be summarized as follows.

(1) Loss of water coolant flow without pipe break
(2) Plasma operation and TBM heating without cooling
(3) Over-heating of structural material and pipe break
(4) Water ingress to over-heated Be pebble bed
(5) Temperature increase, with simultaneous Be - water reaction
(6) When FW temperature reaches the critical temperature, plasma operation is stopped.

Assumption of the safety analysis can be summarized as follows.
(1) Heat loss through radiation from the module wall to common frame.
(2) Reaction heat of water- Be pebble reaction is incorporated in temperature analysis.
(3) Nuclear heating rate data by 1D calculation were used.
The temperature evolution on the TBM is shown in Fig. 4.4.4-7. By the results, it was shown that temperature evolution could be converged, if the plasma is stopped by the infuse or Be armor tile fall when the FW surface temperature is 900 °C. Also, if the 0.3% cooling remains, temperature evolution is converged. Total production of hydrogen is 0.063 [mol/m^2-first wall], which is small compared to the ITER limitation of 10 kg per event.

References

[4.4.4-1] D. Tsuru, et al., "Proceedings of eleventh international workshop on ceramic breeder blanket interactions, December 15-17, 2003, Tokyo Japan", *JAERI-Conf 2004-012*, pp35-39 (2004)
[4.4.4-2] T. Kinjyo, et al., "Estimation of tritium release behavior from solid breeder materials under the condition of ITER test blanket module", to be presented in ICFRM12 (2005).
[4.4.4-3] Accident Analysis Specifications for NSSR-2 (AAS-2) Version 2.1 (June 25, 1997)

4.4.5 Supporting R&D and Supporting Activities to other Parties TBMs

(1) Strategy of Blanket Development of Japan

As described in the sub-chapter 4.4.1, ITER TBM testing is regarded as one of the most

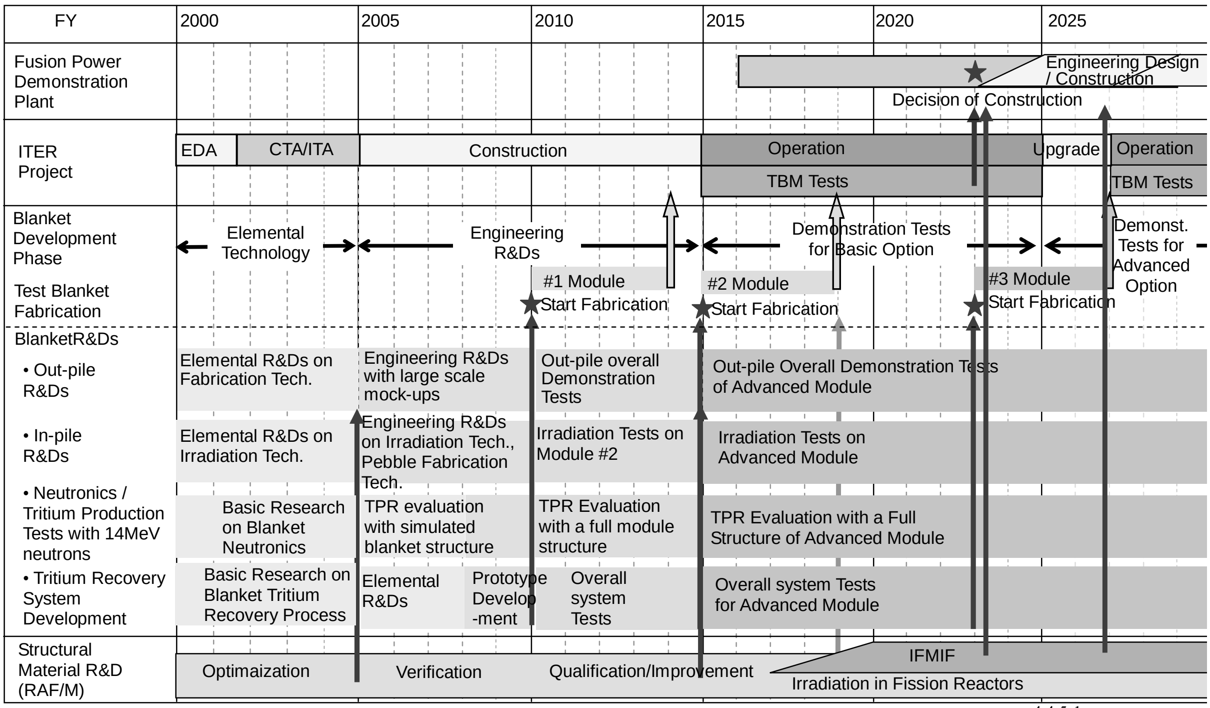


Fig. 4.4.5-1 R&D plan of solid breeder blanket and material development for DEMO[4.4.5-1].

important milestones, by which integrity of candidate blanket concepts and structures are qualified, together with material development and qualification of irradiation performance by IFMIF. Japan is investigating the possibility of testing all types of TBMs and contributing to all WSGs under the framework of TBWG with involvements of all of JAERI and universities and NIFS[4.4.5-1].

With respect to the development of the primary candidate blanket for the fusion power demonstration plant, solid breeder test blankets made of RAFM are being developed by JAERI with cooperation of universities, according to the stepwise development plan consists of elemental technology development phase, engineering R&D phase and ITER TBM test phase[4.4.5-2]. In all essential issues of blanket development, elemental technology development has been almost completed and is now stepping further to the engineering R&D phase, in which scalable mockups of solid breeder test blanket modules will be fabricated and tested to justify the total structure integrity and to certify the final fabrication specification of TBMs in the next 5 years[4.4.5-2].

With respect to the development of the advanced blankets, key issues have been addressed and critical technologies are being developed for high temperature solid breeder blanket with SiC_f/SiC structure, He cooled liquid LiPb breeder blanket with SiC inserts and its dual-coolant option, liquid Li self-cooled blanket with V alloy and molten salt self-cooled blanket with RAFM structure by universities and NIFS. The development of advanced blankets is showing steady progress, taking into account the ITER TBM testing program[4.4.5-1].

The development of blankets in Japan is showing sound and steady progress on both of solid and liquid breeder blankets under coordinated domestic development strategy, for both of primary and advanced options.

(2) R&D Achievement of Solid Breeder TBMs (WSG-1 and WSG-3)

Figure 4.4.5-1 shows the long term R&D program of the solid breeder blanket development toward the fusion power demonstration plant. To achieve the ITER module testing, the blanket development is programmed to consist of stepwise phases, Elemental Technology phase, Engineering R&D phase and Demonstration Test phases for basic options and advanced options of blankets. As the achievement of the Engineering R&D phase, the manufacturing specification and the safety demonstration data are expected[4.4.5-1].

Essential issues of the solid breeder blanket development are Out-pile R&D, In-pile R&D, Neutronics and Tritium Production Tests with 14 MeV Neutrons and Tritium Recovery System Development. Out-pile R&D consists of the development of blanket module fabrication and the development of thermo-mechanical and chemical compatibility design database of breeding region of the blanket. In-pile R&D consists of the development of irradiation technology for partial blanket mockups in fission reactor, the development of fabrication technology for breeder and multiplier pebbles, and irradiation tests of breeder and multiplier pebble beds. Neutronics tests by 14 MeV neutron source consists of the precise evaluation of neutronics characteristics of the blanket materials and tritium production rate data with real blanket materials and mockups. The development of tritium recovery system consists of the development and basic research on the processes of blanket tritium recovery system.

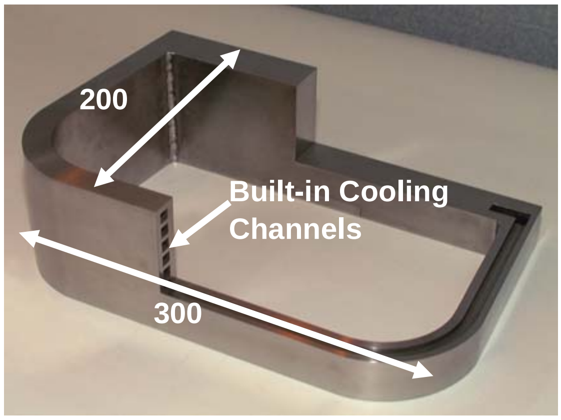


Fig. 4.4.5-2 Fabricated Mockup of Box Structure with Built-in Cooling Channel.

Fabrication technology of the blanket box structure

Reduced activation martensitic ferritic steel, F82H (8Cr-2W-V-Ta) has been developed for fusion application[4.4.5-3]. Material improvement and property data accumulation have been performed by material development group of JAERI and universities[4.4.45-4, 4.4.5-5]. In fabrication technology development of the DEMO blanket in JAERI, a hot isostatic pressing (HIP) bonding method, especially for the first wall structure with built-in cooling tubes has been proposed. Preliminary screening tests have been performed to obtain suitable conditions of HIP bonding and after HIP heat treatment[4.4.5-6]. By using investigated conditions, first wall panel mockup has been fabricated to be tested by high heat flux tests. Under the heat load of 2.7 MW/m^2 up to 5000 cycles, the fatigue performance of the high heat flux tests showed soundness of the fabrication of the first wall mockup, compared with the data of the base metal certified by IEA round robin tests[4.4.5-7]. The box structure mockup with the wall with built in cooling channels was also fabricated for investigation of the manufacturing procedures as shown in Fig. 4.4.5-2. Due to the grain coarsening by HIP heat treatment, the reduction of fracture toughness of HIP joints in the first wall was detected. However, the further investigation clarified the heat treatment conditions for recovery of ductility and the improvement of HIP condition[4.4.5-8]. By the result, heating above 1100 °C and consequent normalizing process was needed to cancel the previously coarsened grain size, which

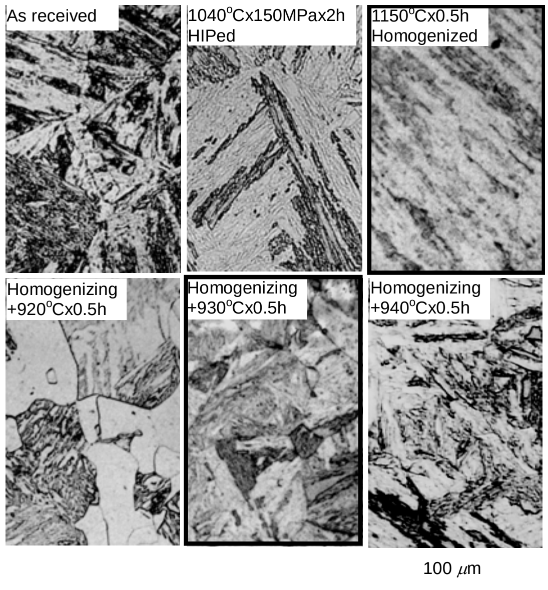


Fig. 4.4.5-3 Typical microstructure of post hip heat treated F82H[4.4.5-8].

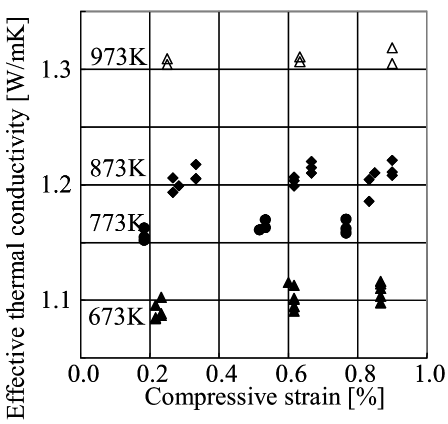


Fig. 4.4.5-4 Effective thermal conductivity of a compressed Li_2TiO_3 pebble bed[4.4.5-14].

temperature is also applicable for HIP treatment. Post HIP heat treatment (PHHT) is also investigated. In order to obtain fine prior austenite grain, it is required to normalize just above the temperature (910ºC) where the structure consists wholly of austenite. By these results, it was suggested that the HIP process (HIP at 1150 ºC + PHHT at 930 ºC + Tempering) could improve both the joining properties and the fracture toughness[4.4.5-8], as shown in Fig. 4.4.5-3. Also, the development of first wall armor joining to the first wall of RAFM has shown progress. As one of the candidate armor material for fusion power demonstration plant, solid state bonding of tungsten and F82H was studied by using Spark Plasma Sintering (SPS) method. According to the results of trial bonding and destructive observation, W and F82H could be joined by the solid-state-bonding without any insert material[4.4.5-9]. For the first wall armor of TBMs, beryllium is recommended. For Be armor joining for TBMs, the further R&D is needed based on the technique of HIP joining of Be and Cu alloys[4.4.5-10].

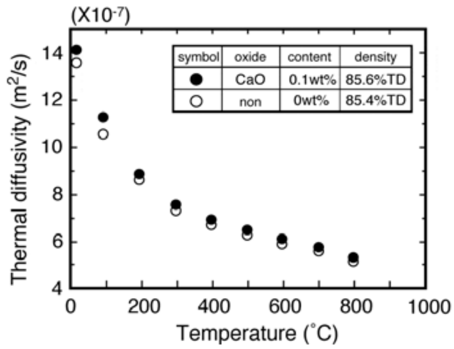


Fig. 4.4.5-5 Thermal diffusivity of Li_2TiO_3 added with CaO, and Li_2TiO_3 without additive[4.4.5-16].

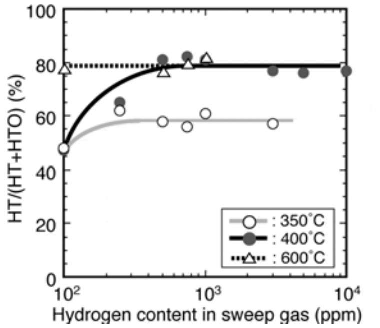


Fig. 4.4.5-6 Influence of hydrogen content in the sweep gas on the fraction of HT; (the sweep-gas flow rate was 200 cc/min)[4.4.5-19].

Investigation on Thermo-mechanical Characteristics of Breeder and Multiplier Pebble beds

As a most important base data, effective thermal conductivities of breeder pebble beds were researched by using hot wire method under the frame of the IEA-IA fusion nuclear technology collaboration[4.4.5-11, 4.4.5-12]. From the results, fitting parameters for correlation of thermal conductivity was determined to represent the observed value of major candidate breeder pebbles. By measured results of effective thermal conductivity of single and binary packed beds of Li_2TiO_3 pebbles fabricated by wet method, it was shown that the estimated values by using the same fitting parameter obtained by single packing bed showed fair agreement with the obtained data of the binary packing bed[4.4.5-13]. Also, effective thermal conductivity of the bed was measured under compressive load up to 10MPa at temperatures ranging from 673K to 973K. As can be seen from Fig. 4.4.5-4, at all temperatures, increases of effective thermal conductivity due to the compressive deformation were confirmed. According to the observation, the change of effective thermal conductivity was not significant. When successive loads, heating-cooling or compression, worked on the bed, effective thermal conductivity increased according to promotion of the compressive deformation[4.4.5-14]. Therefore, further investigation is planned to evaluate pebble bed integrity under cyclic and long term compression in high temperature.

Based on the achievements stated above, the out-pile R&D with engineering scale mockups is foreseen as the next step.

Development of Tritium Breeder Material

Lithium titanate (Li_2TiO_3) has been selected as the first candidate material for the fusion power demonstration plant from viewpoints of effective tritium release and high chemical stability at the operating temperatures. A wet production process for Li_2TiO_3 was developed, which is suitable for mass production and recycling of ^6Li. This process successfully supplied ^6Li-enriched Li_2TiO_3 pebbles which satisfied the target values of the specifications: namely, density of 80-85%TD, diameter of 0.85-1.18mm and grain size <5μm, as well as a good sphericity value of smaller than 1.1[4.4.5-15]. Furthermore, Li_2TiO_3 with oxide additive (CaO, ZrO_2, Sc_2O_3) has been being developed recently in order to control the growth of the grain size and to keep the stability of its crystal structure at high temperatures. Pellet density measurement of Li_2TiO_3 with the additives showed that these oxide additives were effective in controlling the grain growth. Thermal diffusivities for the CaO-added Li_2TiO_3 are slightly higher (at maximum by 5% from room temperature to 400˚C) than those for Li_2TiO_3 without additive, as shown in Fig. 4.4.5-5[4.4.5-16]. Similar results were obtained for ZrO_2 and Sc_2O_3

addition. The overall results suggest that the oxide addition is effective not only in controlling the growth of the grain size but also in improving the thermal properties.

Development of Neutron Multiplier Material

Beryllium (Be) metal is a reference material for neutron multiplier, and semi-industrial fabrication technology of beryllium pebbles was established by a rotating electrode method. However, Be has temperature limit of 600˚C, because of high chemical reactivity and large swelling of Be metal. Therefore, Be pebble bed design tends to limit the performance of the breeding blanket of the fusion power demonstration plant which is operated with high temperature coolant up to high neutron dose (20,000 atomic ppm He and 50 dpa).

Recent R&D is, therefore, focused on Be alloys, in particular Be-Ti alloys[4.4.5-17], because they are promising as advanced multiplier materials due to their superior properties, such as high melting points and high chemical stability at high temperatures.

Trial fabrication tests of Be alloy pebbles were performed in the rotating electrode method to reduce the brittleness of the stoichiometric $Be_{12}Ti$ (Be-7.7at%Ti). As a result, it was revealed that a two phase structure of $Be_{12}Ti$ and αBe was effective in reducing the brittleness. A preliminary fabrication of a small amount of Be-5at%Ti and Be-7at%Ti pebbles were successfully performed without electrode break by thermal stress[4.4.5-18]. Characterization of the Be-Ti alloys produced was performed in cooperation with universities in Japan[4.4.5-19].

Development of Irradiation Technology for In-pile Functional Tests

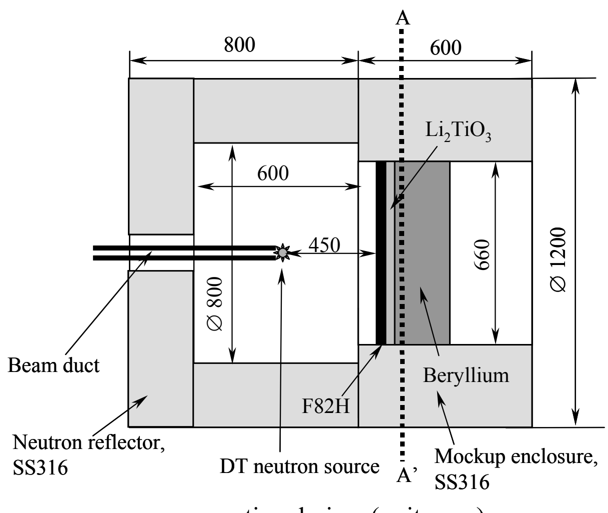
Various irradiation techniques have been developed for in-pile functional tests of tritium breeding materials. Main items are: 1) pulse irradiation technique by changing the neutron flux with a neutron absorber (hafnium) window rotated by a radiation-resistant small motor, 2) multi-paired thermocouples for measuring temperatures at many points, 3) a highly sensitive and responsive self-powered neutron detector (SPND), and 4) ceramic coating for reducing tritium permeation through the structural material to enable a reliable in-pile tritium release experiment.

An integrated in-pile test is being performed in the Japan Materials Testing Reactor (JMTR) by using above techniques[4.4.5-19]. Measurement of tritium release from a Li_2TiO_3 pebble bed revealed that the fraction of HT of the total amount of tritium, HT/(HT+HTO), increased with increasing the hydrogen content in the sweep gas, when the center temperature of the pebble bed was 400˚C[4.4.5-19, 4.4.5-20]. On the other hand, the fraction was almost constant independent of the hydrogen contents, when the center temperature was 600˚C (see Fig. 4.4.5-6). These results suggest that the tritium release was controlled by a reaction at the surface of the Li_2TiO_3 pebble, because the surface reaction is influenced by the hydrogen content as well as the pebble bed temperature.

Neutronics / Tritium Production Tests with 14MeV neutrons

Neutronics integral experiments have been conducted with small partial mockups relevant to the ITER test blanket module proposed by JAERI using DT neutrons at FNS of JAERI[4.4.5-21]. Figure 4.4.5-7 shows the schematic view of the experiment assembly. Small partial mockups of the ITER test blanket modules were installed at about 450 mm distant from the DT neutron source. The neutron reflector applied in this experiment relates to the effect of the incident back-scattering neutron current.

Numerical analyses were conducted by using the Monte Carlo neutral particle transport code MCNP-4C and the fusion evaluated nuclear data library FENDL-2. Figure 4.4.5-7 shows the ratio of the calculated value to the experimental value



cross-sectional view (unit: mm)

Fig. 4.4.5-7 Schematic view of experiment assembly of neutronics test of blanket mockup[4.4.5-21].

(C/E) distribution. The C/Es are 0.96 –1.08 and 1.03 – 1.18 for the experiments without and with the reflector, respectively. The calculation results of the integrated tritium productions agree well with the experimental data within 2 %, i.e. 7 % of the experiment error, and 11 % for the experiments without and with the reflector. From this study, it was clarified that the integrated tritium productions could be accurately predicted for the experiment without the reflector. Uncertainties of the experiment with the reflector are larger than those without one. It is reasoned that this occurs due to uncertainties of the cross section data about the back-scattering neutrons from the SS316. Thus, in conclusion, the prediction accuracy is in the range of 2 – 11 % on the calculation of the tritium breeding ratio (TBR) in the blanket design at this point. Further investigation will be performed to improve the prediction accuracy.

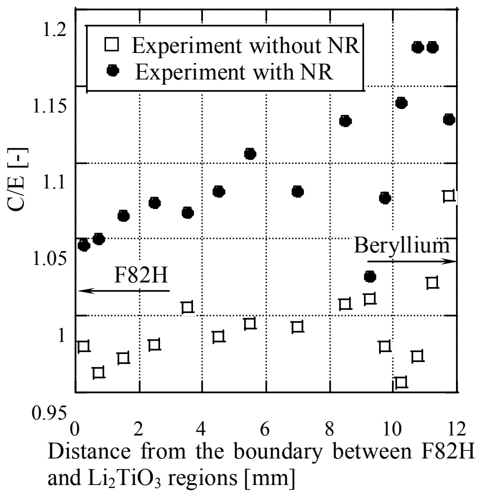


Fig. 4.4.5-8 Ratio of the calculation result to the experiment one (C/E) about the TPR from 6Li for the experiments with and without the neutron reflector (NR)[4.4.5-21].

Tritium Recovery System Development

In the present plan, bred tritium is taken out by passing of H_2-added helium sweep gas, because of the reduction of tritium inventory and the enhancement of tritium release. The kind/quantity of additive for sweep gas and the sweep gas condition at the blanket outlet are important information for development of breeding blanket interface system (BBI)[4.4.5-22]. Generally, the concentration of H_2 in the sweep gas is assumed to be 0.1-1.0%. The chemical species to be processed in BBI are HT, H_2, HTO and H_2O. So, the processing with separation by chemical form is reasonable.

In JAERI, development of cryogenic molecular sieve bed (CMSB) has been carried out for processing of HT and H_2[4.4.5-23]. This is the packed bed of porous adsorbent such as molecular sieve 5A, and is used at 80K. Recently, system integration proof test of CMSB, fuel cleanup system (FCU) and isotope separation system (ISS) has been carried out. Hydrogen isotopes containing tritium are adsorbed on CMSB, and CMSB is vacuumed and heated up to release adsorbed gas in regeneration step. Released gas is sent to ISS after purification by Pd diffuser in FCU. Figure 4.4.5-9 shows the dynamic behavior of simulated fuel processing and CMSB system in regeneration step. The amount of hydrogen isotopes supplied to CMSB agreed well with the amount of hydrogen isotopes recovered in ISS, and the demonstration of the integrated system has succeeded.

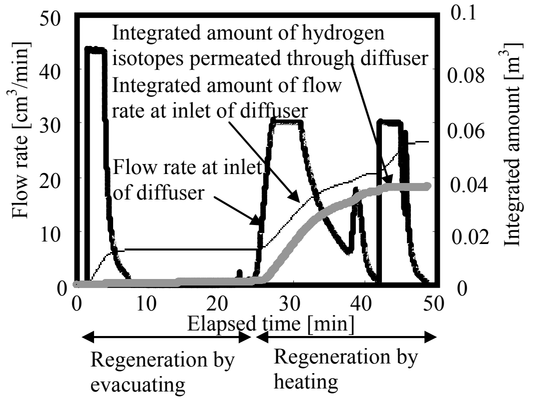


Fig. 4.4.5-9 Dynamic behavior of simulated fuel processing and CMSB systems in regeneration step.

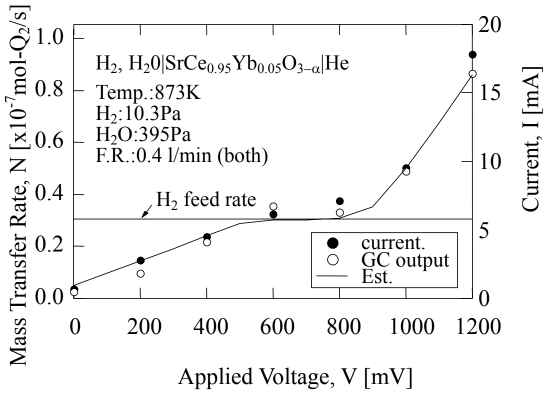


Fig. 4.4.5-10 An example of ionic hydrogen transportation property of hydrogen pump[4.4.5-24].

In DEMO plant, large amount of high temperature sweep gas should be processed effectively. So, in JAERI, the electrochemical hydrogen pump using proton conductor membrane has been proposed as the trade-off system, and its study is going on. The driving force of hydrogen transportation is a potential difference, and hydrogen transportation from low pressure side to high pressure side is possible. This property is suitable for the blanket sweep gas condition. Figure 4.4.5-10 shows an example of ionic hydrogen transportation property of the hydrogen pump using $SrCe_{0.95}Yb_{0.05}O_{3-\alpha}$ membrane. When the applied voltage exceeds 0.8V at 873K, hydrogen transportation via water decomposition also appears. BBI using hydrogen pump may become very efficient system.

When H_2-added sweep gas is applied, tritium leakage via permeation to coolant can not be ignored. In the case where water is used as the coolant, the increase of the load of water detritiation system (WDS) in the tritium plant is not avoided. In ITER design, the chemical exchange combined electrolysis (CECE) method is used. In DEMO plant, it is expected that the amount of water which is processed in WDS increase. So, the reduction of the amount of water by concentration of tritium is necessary to apply CECE to DEMO plant. In JAERI, the pressure swing adsorption (PSA) method by synthetic zeolite packed bed has been proposed to concentrate tritium in water. By observation of breakthrough curve of H_2O and HTO, it was shown that separation of HTO by PSA method is possible[4.4.5-24].

(3) Unit Testing Concept using a Sub-module of He Cooled TBM (WSG-1)

Japanese Helium Cooled Solid Breeder TBM applies integrated structure of three sub-modules. By using one of sub-modules, there is a possibility of testing different concept of internal configuration of the solid breeder blankets. Figure 4.4.5-11 shows a typical concept of testing unit of high temperature solid breeder / SiC_f/SiC blanket cooled by He. One of Japanese commercial fusion plant uses solid breeder blanket with SiCf/SiC structure cooled by high temperature He. In this testing concept, it was proposed to insert a test article of a solid breeder blanket surrounded with the thermal insulation wall of SiCf/SiC inside the sub-module box structure made by RAFM. By adjusting the flow rate of He coolant to the test article, the operation temperature of the test article is raised to higher temperature than 550 oC, for the purpose of testing the thermo-mechanical characteristics. The detailed structure including the support structure of the test article in the sub-module box will be investigated in future. In the latter 10 year period of ITER operation, the testing of TBM made of SiCf/SiC first wall may be also considered, depending on the development progress of materials and module fabrication technology and the result of the test article testing.

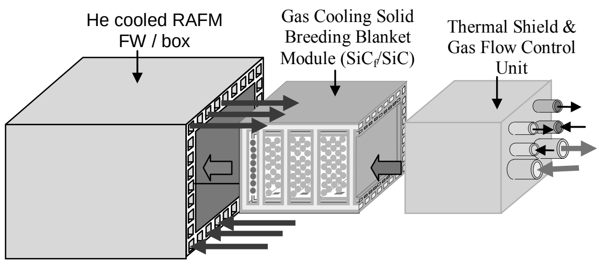


Fig. 4.4.5-11 The Concept of SiC_f/SiC Blanket Unit Tests in RAFM Structure.

(4) He Cooled LiPb blanket (WSG-2)

Liquid LiPb blanket is one of major option of the TBM that attracts interests of all 6 parties. Helium Cooled Lithium Lead (HCLL) is intended to be tested to aquire maximum information for DEMO design in EU, and this TBM option is developed by European leadership with all other parties involvement in SWG2. However Lithium-lead has an improved option of Dual Coolant Lithium Lead (DCLL) concept for higher temperature operation. The original EU design of the HCLL module is made of RAFM vessel filled with liquid lithium lead, and cooled with cooling panel where high pressure helium is circulated. With SiC insert that separates Lithium Lead from RAFM structure, Lithium Lead can be circulated at higher temperature and flow rate because of heat and electrical insulation. Metal surface is protected from corrosion or erosion. Lithium lead works as heat transfer media, and provides an attractive option for high temperature blanket above the limit of RAFM with ITER/TBM. Japan showed the interest and possibility of technical contribution on the investigation of SiC incert and design of DCLL option. Identified subjects to be studied includes SiC/LiPb and RAFM/LiPb compatibility, evaluation of MHD effect, and the development of SiC insert. Kyoto university recently started the operation of a small LiPb loop as a collaboration with JAERI, and will pursue above technical issues to be combined with their strong technical capability of

SiC research. Through the expected results with international efforts, original HCLL will elevate the operation temperature gradually as DCLL.

It is pointed out that many of the advanced reactor design including Japanese applies LiPb-SiCf/SiC blanket to provide high grade heat above 900 °C, and this DCLL option provides practical and realistic technical approach toward them starting from HCLL day-1 TBM as a conservative design. Because Japan has a strategy to develop economical fusion reactor with single stape of DEMO following ITER, such a multiple generations of blanket to gradually and steadily improve the plant performance is important.

(5) Self-cooled Li/V blanket (WSG-4)

A Design Description Document was presented from Russia based on Li/Be/V blanket concept, in which Be was used for neutron multiplying purposes[4.4.5-25]. Japanese WSG-4 is technically supporting the Russian design but independently examining a Li/V test module, in which Be is not used[4.4.5-26]. The Li/V concept has some advantages over Li/Be/V concept such as simple blanket structure, being free from the issues of natural resource limit and handling safety concerning beryllium and no need for periodic replacement of blanket because of the lifetime of Be. Elimination of Be multiplier in the breeding blanket will give benefits of cost reduction and safety enhancement. This aspect is also the case for solid breeder TBMs, however, it is difficult to eliminate Be multiplier in the solid breeder blankets with limited thickness of radial build, without avoiding insufficient tritium breeding performance. Recent neutronics calculations showed enough tritium self-sufficiency of Li/V blanket in tokamak[4.4.5-26] and helical[4.4.5-27] systems.

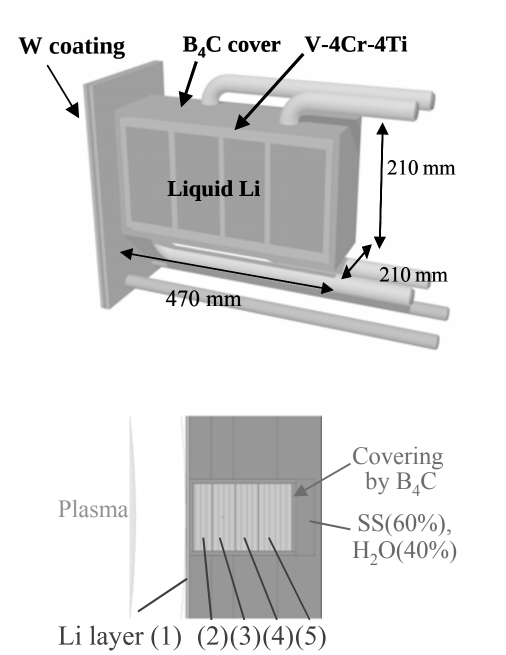


Fig. 4.4.5-12 The schematic view and cross section of the Li/V test module[4.4.5-26].

The primary purpose of the Li/V module test was defined as validation of the tritium production rate predicted based on the neutron transport calculation. For this purpose the module was designed to be composed of sectioned thick boxes which accommodate slow Li flow. The schematic view and cross section of the module is given in Fig. 4.4.5-12. This system enables to measure the tritium production rate as a function of the distance from the first wall. The size of the four boxes was limited (~0.027m^3) so as to satisfy the introduction limit of liquid lithium into the ITER test port. The module is covered with a B_4C layer for the purpose of shielding thermal neutrons. Figure 4.4.5-13 shows the neutron spectra for ITER first wall, Li/V TBM with B_4C cover of 7.5 mm thick and V/Li full blanket[4.4.5-26]. With the B_4C cover, the flux of low energy neutrons decreases and the spectrum approaches that of the Li/V full blanket. The coating with W was shown not to influence TBR[4.4.5-27]. The plasma-facing surfaces of the module would be covered with W coating. The effect of tungsten coating on the tritium production performance was investigated for a Li/V DEMO blanket[4.4.5-27], which showed a positive effect of the coating thickness on the tritium production rate in the case using 35% enriched ^{6}Li. The

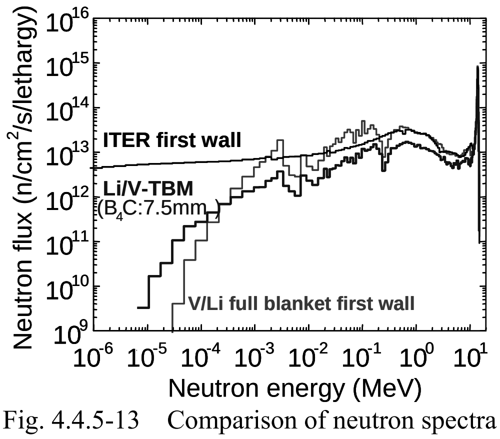


Fig. 4.4.5-13 Comparison of neutron spectra for ITER first wall, Li/V TBM with B_4C cover of 7.5 mm thick and V/Li full blanket[4.4.5-26].

feasibility of the plasma spray coating of W on V-4Cr-4Ti was demonstrated[4.4.5-28]. Significant progress has been made in fabrication technology of vanadium alloyed with focus on V-4Cr-4Ti alloys including fabrication of large V-4Cr-4Ti ingot, manufacturing into plates, sheets, wires, rods and thin tubes, laser welding and so on[4]. Thus manufacturing the test module with high quality is thought to be feasible for V-4Cr-4Ti structures. In Li layer (1), MHD insulator coating is necessary, and the test of the coating is one of the objectives of the layer. Current options of the coating would be (a) PVD coating of Er_2O_3 or Y_2O_3, (b) two-layer coating with Er_2O_3 (or Y_2O_3) covered by vanadium alloys, and (c) in-situ coating of Er_2O_3 by reaction of pre-doped Er in Li and pre-doped O in V-4Cr-4Ti structural materials[4.4.5-29]. As to the tritium recovery from Li, feasibility of gettering tritium by yttrium was demonstrated[4.4.5-30]. Although tritium recovery technology from tritiated yttrium is not verified, this method seems to be feasible to the module test where limited amount of tritium needs to be recovered.

(6) Flibe Blanket (WSG-5)

One of the candidate liquid breeder blankets applies molten-salt Flibe as the self cooling breeder[4.4.5-31]. It has been designed for Force-Free Helical Reactor, FFHR, which is a demo-relevant helical-type D-T fusion reactor based on the great amount of R&D results obtained in the LHD project. It features the reduced activation ferritic steel, JLF-1, or Vanadium alloy, V-4Cr-4Ti, as the structural material and Be pebble bed as the neutron multiplier and redox controller to reduce corrosive F radicals. To enhance the shielding ability, C and B_4C are placed in the rear side region of the Flibe blanket.

Investigation of key technologies of the Flibe blanket have been performed on the thermo-fluid study using Tohoku-NIFS Thermo-fluid Loop for molten salt (TNT Loop) built in Tohoku University (Fig. 4.4.5-14)[4.4.5-32], redox control technology to reduce F radicals, tritium inventory and disengaging technology of molten-salt Flibe partly in the frame of Japan-US collaboration program, JUPITER-II. Flibe chemistry experiments and TBM neutronics calculations are on going in Universities and NIFS. In parallel, the development of structural materials and structure fabrication technology are being performed.

Design and development of Flibe TBM is mainly being performed by US. Japanese universities and NIFS are cooperating its design activity in the frameworks of TBWG WSG and JUPITER-II program. The technology of thermo-hydraulics of Flibe can be covered by the scale of TNT Loop. The further R&D achievements are expected to be obtained by TNT Loop.

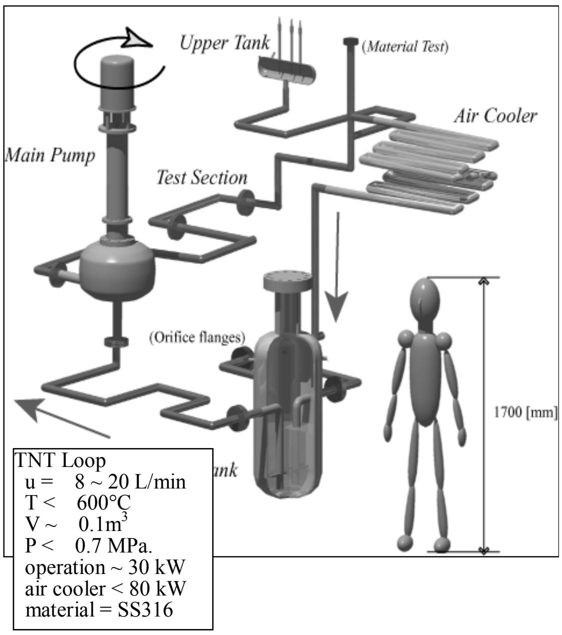


Fig. 4.4.5-14 Tohoku-NIFS Thermo-fluid Loop for molten salt (TNT Loop) built in Tohoku University[4.4.5-32].

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4.4.6 Validation Program prior to the installation in ITER

ITER blanket module testing is the first unique test bed of blanket module in a fusion environment although the load conditions on the TBM in ITER are not fully relevant to the fusion power demonstration plant. By the testing, it is necessary to qualify the methodology of blanket design and integrated performances and reliability of the blanket system under the integrated load conditions. Specific issues to be qualified are:

(a) tritium production consistent with neutron spectrum
(b) high grade heat generation relevant to the electricity generation
(c) thermo-mechanical behavior of the module with pebble beds under surface heat load and neutron irradiation
(d) tritium release characteristics from the pebble bed
(e) tritium permeation to the coolant
(f) corrosion of the cooling channel and activated corrosion products
(g) integral behavior of all elements in the blanket system
(h) reliability of the blanket system

As described in Sub-chapter 4.4.5, prior to the ITER blanket module testing, Out-pile R&D, In-pile R&D, Neutronics / Tritium Production Rate (TPR) tests with 14 MeV neutron source and Tritium Recovery System Development will be carried out together with the material development. By the Engineering R&D, which consists of above mentioned four major R&Ds, integrity of TBMs should be evaluated by using scalable TBM mockups to show the relevancy of TBMs to the module testing and the finalization of specification of manufacturing of TBMs (TBM W-1 and TBM H-1) in 5 years. Also, by using the prototype TBM mockup, relevancy to ITER safety standard will be demonstrated.

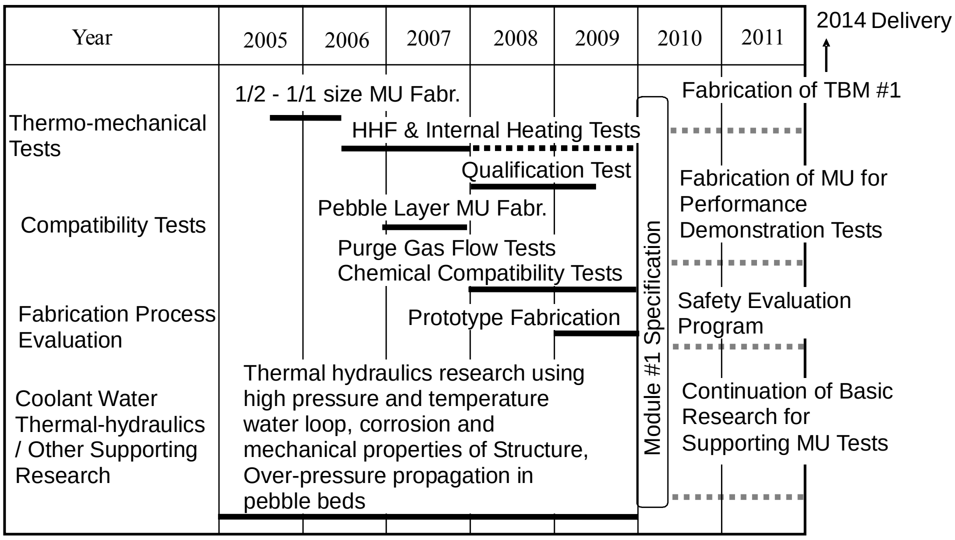

Fig. 4.4.6-1 Time schedule of development and validation of Water Cooled and He Cooled Ceramic Breeder TBM.

Figure 4.4.6-1 shows the time schedule of validation of Water Cooled and He Cooled Ceramic Breeder TBM. The validation program is planned for 10 years. In the former 5 year period, which is expected to start from 2005, half size to full size mockups for validating thermo-mechanical performance and purge gas chemical stability will be fabricated and tested under TBM relevant conditions. In the thermo-mechanical validation tests, the loading conditions of surface heat flux will be given using ion beam irradiation. Simultaneously, the volumetric heating will be simulated by insert panel heater in the pebble bed of the mockup. Schematics of the thermo-mechanical validation test is shown in Fig. 4.4.6-2. For the coolant, high pressure and temperature water (25 MPa and 450 ^{o}C) will be used. The

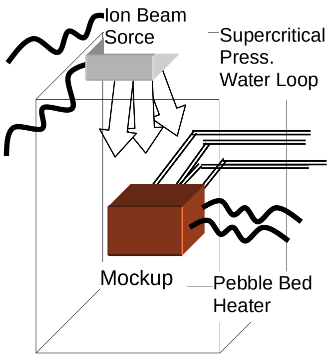

Fig. 4.4.6-2 Concept of Qualification Test of Water Cooled and He Cooled TBM.

supercritical water loop is already constructed in the existing high heat flux test facility in JAERI. After thermo-mechanical validation tests and chemical stability validation tests, qualification tests should be performed to show justification of the capability of fabrication and delivery of the real TBMs on time. In the final stage of the former five year period, the fabrication of prototype TBM will be performed to identify the detailed technical conditions of the fabrication process. As the goal of the former period of validation program, the detailed fabrication specification will be decided. In the later five year period, the validation activity is focused on the evaluation of the safety performance of the TBMs. Basic safety related performance is planned to be investigated in the former five year period, and they will be extended in the later five year period. In the course of later five year period, the detailed fabrication specification of each TBMS for DT operation tests on time.

4.4 Japan Proposal To TBWG

4.4.1 DEMO Study and Testing Strategy in ITER

(1) Roadmap and Strategy of DEMO Blanket Development in Japan

The Fusion Council of Japan has established the long-term R&D program of the blanket development in 1999. The development of blankets is performed according to this program in Japan. In the program, JAERI has been nominated as a leading institute of the development of solid breeder blankets, in collaboration with universities, as the primary candidate blanket for the fusion power demonstration plant[4.4.1-1], while, universities and NIFS are assigned mainly to develop advanced blankets for future blanket options of the fusion power demonstration plant and commercial power plants.

Figure 4.4.1-1 shows the roadmap of blanket development in Japan. ITER blanket module testing is regarded as the most important milestone, by which integrity of candidate blanket concepts and structures are qualified. In parallel with the blanket development, material development is being performed on major candidate materials. The blanket of the fusion power demonstration plant will be decided by the qualification data of the test blanket module (TBM) by ITER blanket module testing and irradiation data by International Fusion Material Irradiation Facility (IFMIF). The fusion material development is being performed on reduced activation ferritic/martensitic steel (RAFM)[4.4.1-2], as the primary candidate structural material, and silicon carbide composite (SiC_f/SiC)[4.4.1-3], vanadium alloy[4.4.1-4], ODS ferritic steel as the advanced structural materials, and also on functional materials such as tritium breeder, neutron multiplier, insulation materials and so on. The blanket development has been, and is being performed under interactive relationship with material development.

As the primary candidate blanket for the fusion power demonstration plant, solid breeder test blankets made of RAFM cooled by helium and water are being developed by JAERI with cooperation of universities. In all essential issues of blanket development, elemental technology development has been almost completed and is now stepping further to the engineering R&D phase, in which scalable mockups of solid breeder test blanket modules will be fabricated and tested to clarify the total structure integrity for final specification decision of test blanket modules.

As the advanced blanket options, solid breeder blanket module with SiC_f/SiC structure cooled by high temperature helium gas, liquid LiPb breeder with SiC inserts cooled by helium and its dual-coolant option, liquid Li self-cooled blanket made by V alloy and molten salt self-cooled blanket made by RAFM are under development by universities and NIFS with cooperation. Key issues have been addressed and critical technologies are being developed.

The development of blankets in Japan has shown sound progress on both of solid and liquid breeder blankets under coordinated domestic development programs, for both of primary and advanced options. Japan is investigating the possibility of testing all types of blankets under Test Blanket Working Group (TBWG) framework with both of JAERI and universities/NIFS involvements.

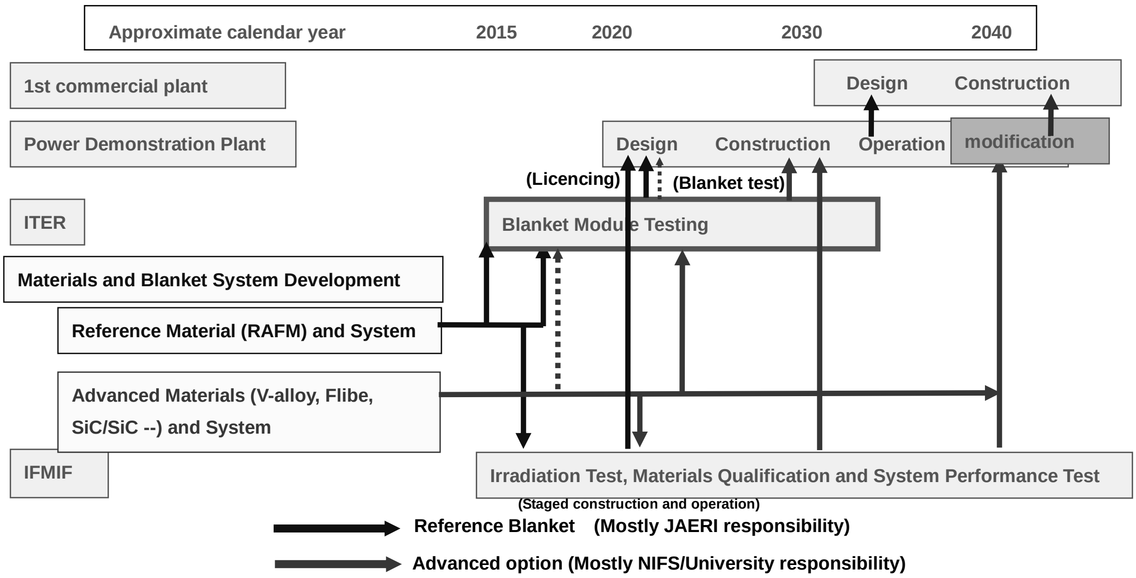

Fig. 4.4.1-1. Road map for materials and blanket development in Japan.

(2) DEMO Blanket Design as the Development Target of TBMs

One of conceptual designs of the DEMO reactors has been performed by JAERI, aiming at the achievement of similar plasma performance to SSTR[4.4.1-5], such as fusion power, Q value, and neutron wall load with more economical merits[4.4.1-6]. In line with the reactor design, DEMO blanket design has been conducted[4.4.1-1], as the advanced solid breeder blanket for DEMO, cooled by the supercritical water. The basic structure of this blanket design is regarded as common structure for basic coolant option of high pressure water (280 – 320°C, 15 MPa) or backup coolant option of high pressure Helium (300 – 500°C, 8 MPa). Major design parameters of the DEMO blanket are summarized in Table 4.4.1-1. Load conditions and applied materials are similar to those of SSTR.

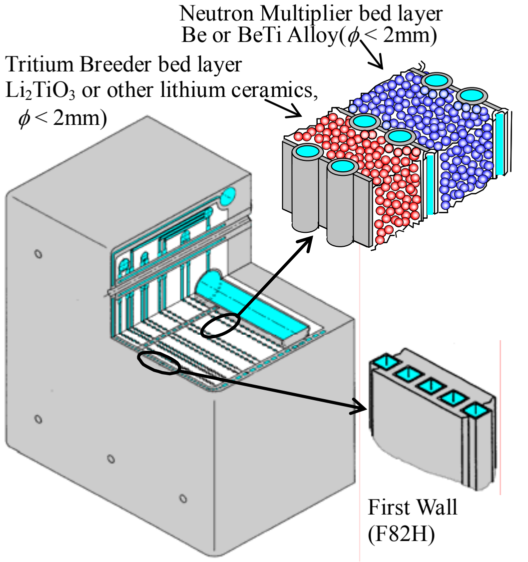

Fig. 4.4.1-2. Schematic structure of one of cadidate DEMO blanket.

Conceptual layout of the blanket module is shown in Fig. 4.4.1-2. Dimension of the blanket module is about 2 m high, 2 m wide, and 0.6 m thick. Reduced activation ferritic steel, F82H, which is currently under development by JAERI, was selected as the structural material. Ceramic breeder and beryllium neutron multiplier are packed in a form of small pebbles in a layered structure as shown in the figure. Li_2TiO_3 was selected as the primary candidate tritium breeder material in this design work and other lithium ceramics are regarded as the backup candidate breeder material. With respect to the neutron multiplier material, Beryllium was selected as the primary candidate and inter-metallic compounds, such as $Be_{12}Ti$, are regarded as the advanced neutron multiplier materials.

Table 4.4.1-1. Typical Design Parameters of DEMO Blanket.

Item	Value
Surface heat flux	0.5 (peak 1) MW/m^2
Neutron wall load	3.5 (peak 5) MW/m^2
Neutron Fluence	>10 MWa/m^2
Coolant Material	Supercritical water
Coolant Pressure	25 MPa
Inlet / Exit Temperature	280 /510 °C
Tritium Breeding Ratio	>1.05
Structural Material	RAFS* (F82H)
Tritium Breeder	Li_2TiO_3 or other Li ceramics
Neutron Multiplier	Be or BeTi alloy

* Reduced activation ferritic steel

Table 4.4.1-2. Estimated TBR for DEMO Blanket.

Materials	Li_2O / Be		Li_2TiO_3 / Be		Li_2TiO_3 / $Be_{12}Ti$				
^{6}Li Enrichment	30%	90%	30%	90%	90%	30%	90%	30%	90%
Packing Structure	Breeder / Multiplier Separate					Breeder + Multiplier Mix			
Temperature Limits	Breeder 900°C				900°C	600°C		900°C	
	Multiplier 600°C				900°C				
Local TBR	1.53	1.56	1.41	1.52	1.37	1.24	1.35	1.35	1.43
Coverage Requirement*	69%	67%	74%	69%	77%	85%	78%	78%	73%

* Required coverage fraction of the plasma facing surface of the breeding region of the blanket in the total area of the plasma facing surface, to achieve net TBR, 1.05.

Net TBR is required to be more than 1.05 for stable operation and accumulation of startup fuel for next fusion plant[4.4.1-7]. Table 4.4.1-2 summarizes estimated values of TBR with major candidate options of materials, ^{6}Li enrichment and structure. Li_2TiO_3 and $Be_{12}Ti$ are expected to have better compatibility with water in high temperature than Li_2O and Be. Even in case where Li_2TiO_3 or $Be_{12}Ti$ are applied, net TBR satisfied more than 1.05[4.4.1-1]. As the design study of water cooled solid breder blankety for DEMO, thermo-mechanical analyses of blanket module structure, tritium inventory analysis, evaluation of tritium permeation[4.4.1-8], configuration and thermal Efficiency evaluation of power plant have been conducted.

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4.4.2 TBMs Design

(1) Design of TBMs

The primary candidate blanket is supposed to be solid breeder blankets for DEMO which will be planed next to ITER. Therefore, Japan is proposing to deliver Japanese solid breeder TBMs from the first day of ITER operation and to perform module testing. Therefore, design work has been performed on solid breeder TBMs intensively. As described in subchapter 5.4, Japan is planning to have 4 TBMs, for system check and environment measurement (TBM W-1, H-1), neutronics test (TBM W-2, H-2), tritium production test (TBM W-3, H-3) and thermo-mechanical test (TBM W-4, H-4). TBM W-1 and H1 have the same structure as TBM W-4 and H4. Dimensional variation will be made to TBM W-2, H-2, W-3 and H3, based on the design of TBM W-4 and TBM H-4. Therefore, detailed design of TBMs is mainly performed for TBM W-4 and TBM H-4, TBMs for thermo-mechanical performance test. Design of solid breeder TBMs have been performed under the design conditions stated in Chapter 2, to represent major structure features of the design of DEMO blanket described in the previous sub-chapter[4.4.2-1].

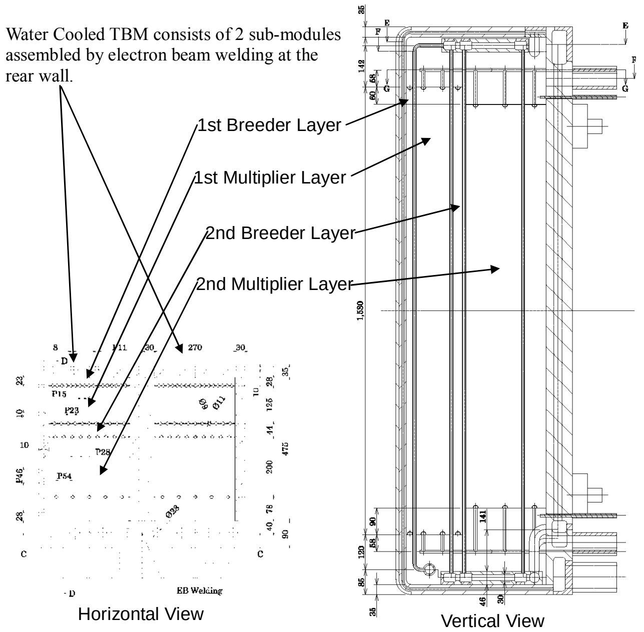

Fig. 4.4.2-1 Structure of Typical Cross Sections of Water Cooled TBM[4.4.2-2]

The design of Japanese solid breeder TBMs has the following features.

(a) First wall and side walls are fabricated in near-net-shape by Hot Isostatic Pressing (HIP) using RAFM, for realizing built-in cooling channel structure.

(b) Vertical slots was adopted to split the blanket module into smaller sub-modules, in less than 50 cm intervals, for the purpose of reduction of electro- magnetic force in vertical disruption event (VDE) and increasing the endurance to internal over-pressure in the case of coolant ingress in the module[8]. Sub-modules are integrated at rear wall by welding.

(c) For Helium Cooled TBM, By-pass flow, which is merged to the outlet flow after temperature homogenization by heat exchanger, is planned to adjust the TBM operation temperature to DEMO relevant condition. In addition, coolant manifold for each sub-module is independent to each other. Correspondingly, the heat exchanger for by-pass flow of each sub-module can be independent. This design allows the flexibility of testing different concepts of sub-modules.

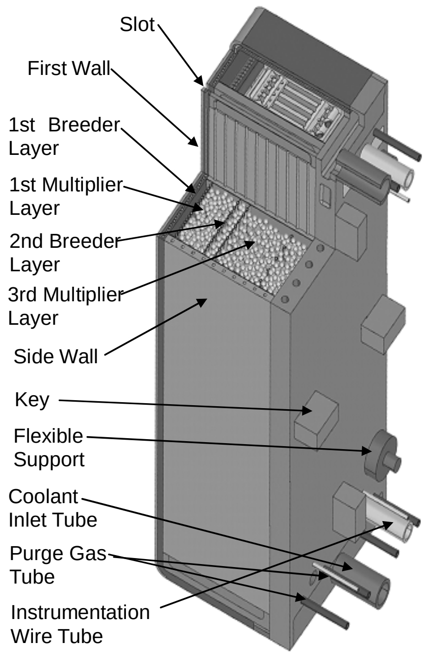

Fig. 4.4.2-2 Three dimensional view of the structure of Water Cooled TBM[4.4.2-1]

(d) Breeder and multiplier are packed in layered pebble beds whose partition walls are integrated with cooling pipes. The internal structure is designed according to the same concept as the breeding blanket for fusion power demonstration plant[1].

Table 4.4.2-1 summarizes the major specification of Water Cooled and He Cooled Solid Breeder TBMs proposed by Japan. Detailed structure design has been performed on TBMs for thermo-mechanical tests[4.4.2-2].

Figure 4.4.2-1 shows the typical drawing of vertical and horizontal cross sections of Water Cooled TBM[4.4.2-2]. In the water-cooled TBM, two sub-modules have same box structures and internal structures. The first wall made of F82H has built-in rectangular cooling paths. As for internal structure, it has multi-layer pebble beds structure same as the DEMO blanket. Breeder and neutron multiplier formed by small pebbles are packed separately in inner box structure made of F82H thin plates, which is separated into four layers by cooling panels. The cooling panel consists of F82H tubes, which are the inner diameter of 9mm and the thickness of 1.5mm, and thin plates connecting adjacent tubes. The inner box structure is welded to the first wall and the back plate. The thickness of each layer and pitches between tubes at each cooling panel were optimized to experience similar level of temperatures and possibly stresses as those in the DEMO blanket according to the transient performance analyses of temperature evolution and tritium generation / release performance. Figure 4.4.2-2 shows the schematic three dimensional drawing of

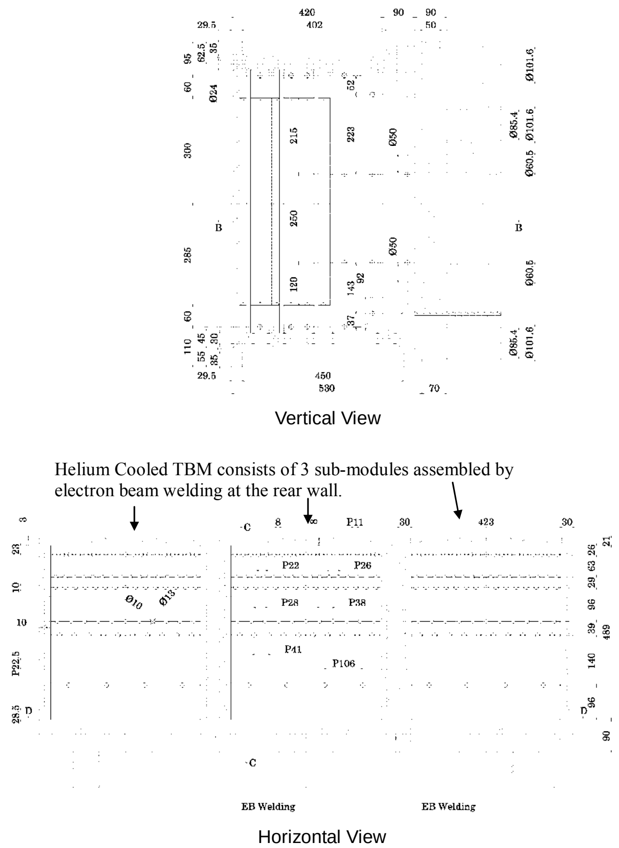

Fig. 4.4.2-3 Structure of Typical Cross Sections of Helium Cooled TBM[4.4.2-2]

Table 4.4.2-1 Major Specification of Water Cooled and Helium Cooled Solid Breeder TBMs[4.4.2-2]

Items		Unit	Water Cooled: Reference	Water Cooled: Advanced	Helium Cooled
Structural Material			F82H		F82H
Coolant			Pressurized Water	Supercritical Water	Helium Gas
Multiplier	Primary		Be		Be
	Temperature Limit	°C	< 600		< 600
	Advanced		$Be_{12}Ti$ or BeTi alloy		$Be_{12}Ti$ or BeTi alloy
Breeder	Primary		Li_2TiO_3		Li_2TiO_3
	Temperature Limit	°C	< 900		< 900
	Backup		Other Li Ceramics		Other Li Ceramics
Area of First Wall		m^2	0.68 × 1.94		1.49 × 0.91
TBM Thickness		m	0.6		0.6
Surface Heat Flux		MW/m^2	0.3		0.3
Nuetron Wall Load		MW/m^2	0.78		0.78
Total Heat Deposit		MW	1.55		1.61
Total Tritium Production		g/FPD	0.156		0.180
Coolant Pressure		MPa	15.5	25.0	8.0
Coolant Inlet Temperature		°C	280.0	360.0	300.0
Coolant Outlet Temperature		°C	325.0	380.0	500.0
Coolant Flow Rate		kg/s	6.15	6.53	1.80
Coolant Bypass Flow Rate		kg/s	—	—	0.51

Water Cooled Solid Breeder TBM composed by established drawings[4.4.2-1].

Figure 4.4.2-3 shows the typical drawing of vertical and horizontal cross sections of He Cooled TBM[4.4.2-2]. As for the helium-cooled TBM, one or two sub-modules may have different structures to the Japanese sub-module(s), which are designed and developed by other parties. The sub-module proposed by Japan has almost same box structure and multi-layer internal structure as that of the water-cooled TBM. It is noted that there is a space of 96mm thickness in front of the back plate where neutron multiplier pebbles are not packed to lay coolant connecting pipes, helium purge gas connecting pipes and cable conduits for instrumentations. The thickness of each layer and pitches between tubes at each cooling panel were also optimized same as the water-cooled TBM. Table 4.4.2-2 shows major materials and weight which compose ceramic breeder test blankets of Japan. Total weight is about 2.5 tons in both TBMs. As the important information on interface condition with the Shield Plug of the Common Frame, specifications of penetration pipes through the Shield Plug is listed in Table 4.4.2-3 and 4.4.2-4 fro Water Cooled and He Cooled Ceramic Breeder TBM. Both interface conditions meet the limitation of the penetration pipe numbers and dimensions from the design of the Shield Plug. In the case of He Cooled Ceramic Breeder TBM, there is a possibility to install more numbers of penetration pipes for independent sub-module cooling. However, in this case, the pipe diameters are limited to smaller size than 35 mm^{OD} to meet the Shield Plug requirement.

Table 4.4.2-2 Major materials and weight of Japan Ceramic Breeder TBMs

Material	Water Cooled Ceramic Breeder TBM	He Cooled Ceramic Breeder TBM
F82H	2041 kg	1961 kg
Be Armor	23 kg	7 kg
Li_2TiO_3 pebble	91kg	104 kg
Be pebble	430 kg	305 kg
Coolant	38 kg	0.5 kg
Total	2624 kg	2378 kg

Table 4.4.2-3 Interface pipe specifications of Water Cooled Ceramic Breeder TBM to Shield Plug.

	Coolant		Purge gas		Cable Conduit	
	Inlet	Outlet	Inlet	Outlet	Thermocouple Strain gauge	Micro-fission chamber
Fluid	Water	Water	Helium	Helium	-	-
Press.	15.5MPa	15.5MPa	0.1MPa	0.1MPa	-	-
Temp.	280^{o}C	325^{o}C		450^{o}C	-	-
Pipe size	101.6mm^{OD} 76.2mm^{ID}	101.6mm^{OD} 76.2mm^{ID}	25.4mm^{OD} 18.4mm^{ID}	25.4mm^{OD} 18.4mm^{ID}	76.3mm^{OD} 57.3mm^{ID}	13.8mm^{OD} 7.8mm^{ID}
Number	1	1	1	1	4	4
Dest.	TBM cooling system		TBM tritium recovery		Control panel	

Table 4.4.2-4 Interface pipe specifications of He Cooled Ceramic Breeder TBM to Shield Plug.

	Coolant		Purge gas		Cable Conduit	
	Inlet	Outlet	Inlet	Outlet	Thermocouple Strain gauge	Micro-fission chamber
Fluid	Helium	Helium	Helium	Helium	-	-
Press.	8.0MPa	8.0MPa	0.1MPa	0.1MPa	-	-
Temp.	300^{o}C	472^{o}C		450^{o}C	-	-
Pipe size	101.6mm^{OD} 85.4mm^{ID}	101.6mm^{OD} 85.4mm^{ID}	25.4mm^{OD} 18.4mm^{ID}	25.4mm^{OD} 18.4mm^{ID}	60.5mm^{OD} 50.0mm^{ID}	13.8mm^{OD} 7.8mm^{ID}
Number	1 (3*)	1 (3*)	3	3	6	6
Dest.	TBM cooling system		TBM tritium recovery system		Control panel	

* Depending on the number of independent sub-modules, there is a possibility of more pipe numbers only with smaller diameter (about 35mm^{OD}).

References

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4.4.3 TBM Systems

(1) Cooling System for Water Cooled TBM

Design conditions of the cooling system for the JA water-cooled TBM are summarized in Table 4.4.2-1. The first wall area facing the plasma is 0.68m × 1.94mH with 20 mm-wide gap between the TBM and the common frame. Thermal power (removal heat) of the TBM is 1.55 MW with maximum 0.5 MW/m^2 (0.3 MW/m^2 average) of surface heat flux and nuclear heating due to neutron wall loading of 0.78 MW/m^2. Primary coolant conditions are 280 °C and 325 °C at TBM inlet and outlet, respectively, and pressure of 15.5 MPa. The flow rate of the primary coolant water is 6.15 kg/s. About 5 % of the primary coolant flow is bypassed and circulated through a purification system (CVCS: chemical and volume control system). The thermal power of the TBM is transferred to the ITER secondary coolant water of 35/75 °C at a heat exchanger inlet/outlet, respectively, and 0.5 MPa. Demonstration of electricity generation utilizing the power from the water-cooled TBM is planned. When the electricity generation is demonstrated, the electricity generation system is connected, via the steam generator, as an intermediate loop to the ITER secondary cooling system. For the common frame, water coolant of the ITER first wall/blanket will be used.

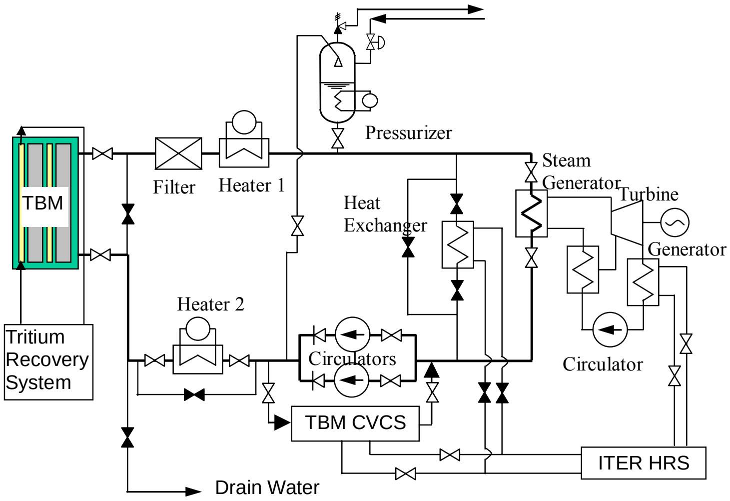


Fig. 4.4.3-1 Flow Diagram of Cooling System of Water Cooled TBM

The flow diagram of the cooling system for the water-cooled TBM is shown in Fig. 4.4.3-1. Major components in the main loop are a steam generator, a main heat exchanger, a circulation pump, a pressurizer, a heater 1 and a heater 2. The high-pressure primary coolant flows inside tube and low-pressure secondary coolant outside tube. For circulation pump system, two circulation pumps are planned for redundancy in case of a pump trip accident was reduced to one pump because of the limited layout space. The pressurizer is designed to accommodate the volumetric change of water coolant due to its temperature rise from room temperature (20 °C) to 300 °C. The heater 1of the same power as the TBM, 1.17 MW, is equipped between the TBM outlet and the steam generator inlet and used during the demonstration of electricity generation. Namely, the heater 1 will provide the power to compensate the power reduction of the TBM during the dwell time, thus to keep the water temperature at steam generator inlet constant and to avoid excessive load change for the turbine system. The heater 2 of 450 kW is equipped to warm-up the system by temperature rising rate at about 50 °C/h and also to adjust the TBM inlet temperature during operation. Though it is not shown in the diagram, a suppression tank is to be included to avoid an excessive pressure increase in case of in-TBM water leak.

(2) Cooling System for Helium Cooled TBM

Design conditions of the cooling system for the JA helium-cooled TBM are summarized in Table 4.4.2-1. The first wall area facing the plasma is 1.49m × 0.91mH with 20 mm-wide gap between the TBM and the common frame. Thermal power (removal heat) of the TBM is 1.61 MW with maximum 0.5 MW/m^2 (0.3 MW/m^2 average) of surface heat flux and nuclear heating due to neutron wall loading of 0.78 MW/m^2. Primary coolant conditions are 300 °C and 500 °C at TBM inlet and outlet, respectively, and pressure of 8 MPa. The flow rate of the primary coolant is 1.8 kg/s. About 0.2 % of the primary coolant flow is bypassed and circulated through a purification system. The thermal power of the TBM is finally transferred to the ITER secondary coolant water of 35/75 °C at a heat exchanger inlet/outlet, respectively, and 0.5 MPa. For the common frame, water coolant of the ITER first wall/blanket will be used.

The flow diagram of the cooling system for the helium-cooled TBM is shown in Fig. 4.4.3-2. Major components in the main loop are a main heat exchanger, a circulator and a heater. The high-pressure primary coolant flows inside tube and low-pressure secondary coolant outside tube. The heater about 200 kW is equipped to warm-up the system and also to adjust the TBM inlet temperature during operation. The flow diagram of the helium purification system is also shown in Fig. 4.4.3-3.

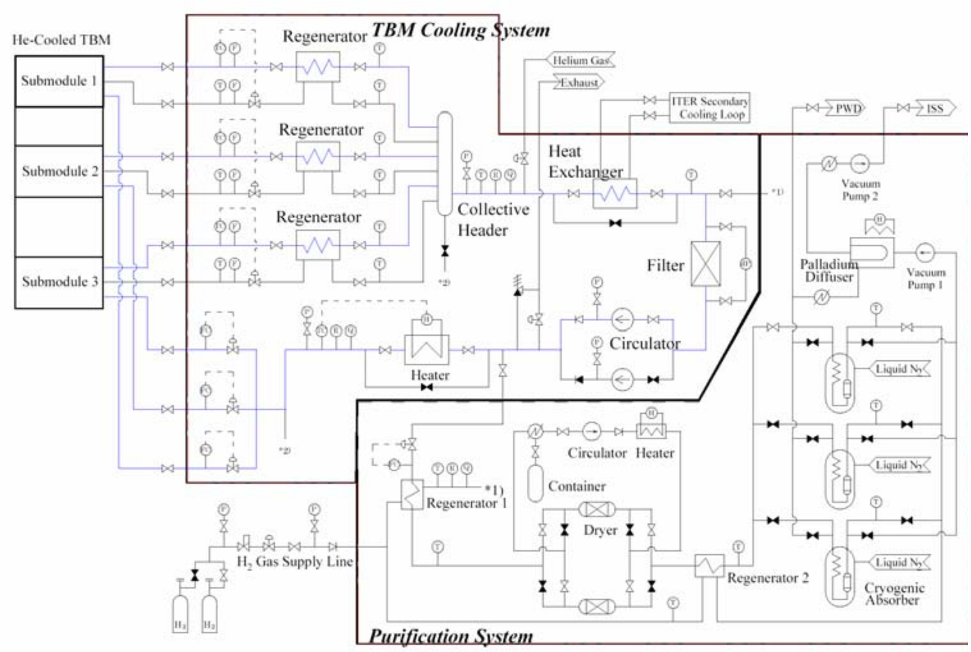


Fig. 4.4.3-2 Flow diagram Cooling System of Helium Cooled TBM

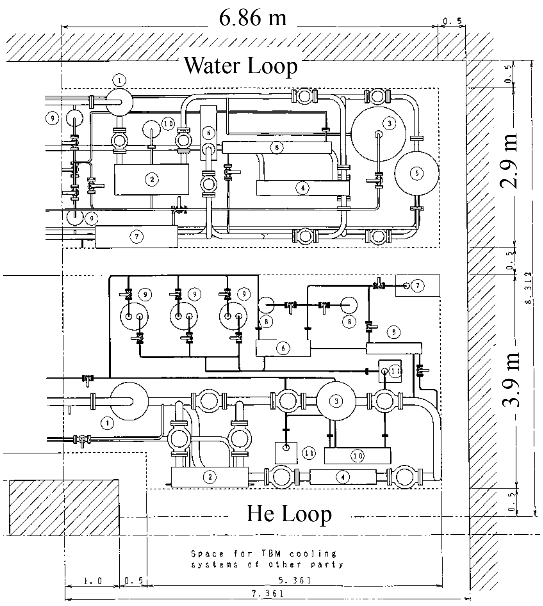


Fig. 4.4.3-3 Layout plan of cooling systems of Water Cooled and He Cooled TBMs

(3) Layout Plan of Cooling Systems

A layout plan of the primary cooling systems for JA Water Cooled and He Cooled TBMs is shown in Fig. 4.4.4-3. As seen from the figure, its footprint is about 7 m x 6.861 m with 5 m height for all cooling systems. Though the space is very tight resulting in complicated pipe routing and only 0.5 m-wide space left around each cooling system for access during installation and maintenance, the cooling systems could be installed into the TCWS vault. It should be noted that the drain/surge tank (1.5 mOD x 2.0 mH) and the suppression tank (10 m^3) are not included in the above layout. Therefore, the space for these tanks is required, probably in a lower space in the building.

(4) Tritium Recovery and Measurement Systems

Tritium recovery system for the TBM has very important roll to recover tritium from helium purge gas of the TBM, clean up humidity or vapor from helium purge gas, and supply clean purge gas to the TBM. At the same time, it is required to transfer recovered tritium in the most suitable form to ITER tritium plant with minimal impact to ITER tritium plant systems.

The major functions required to the tritium recovery system of the TBM are,

1) to measure gas composition of helium purge gas for the purpose of the evaluation of TBM function,
2) to recover H_2 and HT, H_2O and HTO from the helium purge gas,
3) to cleanup purge gas (humidity and vapor) and condition.

JA water-cooled and helium-cooled TBM's utilize similar helium purge system to recover generated tritium. Here, one common tritium recovery system for both TBM's will be installed with multi-point gas analysis system for obtaining tritium release data and system control. However, some of gas analysis equipments are to be installed within a transfer cask (about 1 m^3) in front of the test port bioshield plug of each TBM to obtain the data independently.

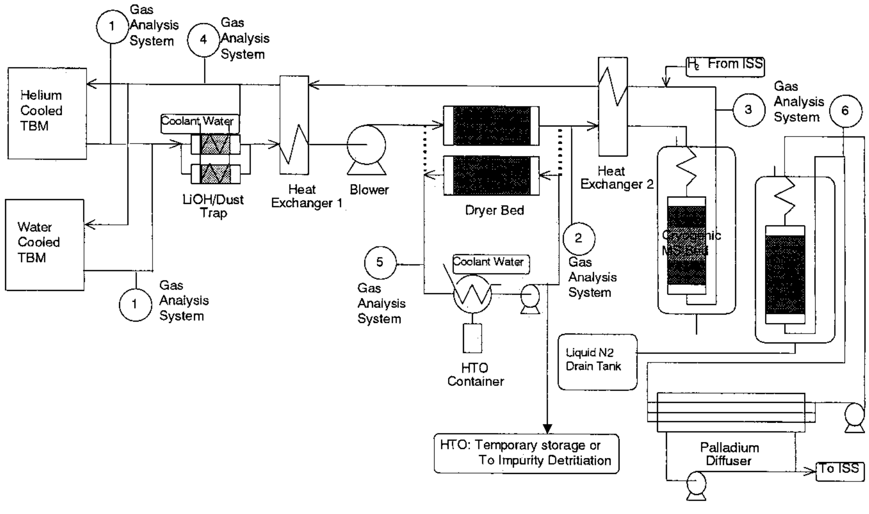


Fig. 4.4.3-4 Schematic flow diagram of JA TBM tritium recovery system

Schematic flow diagram of JA TBM tritium recovery system is shown in Fig. 4.4.3-5. The tritium recovery by helium purge gas separate from the main coolant is the same for the water-cooled and helium-cooled TBM's. The tritium producing capability, thus the purge gas conditions, are also similar for both TBM's. The tritium recovery system consists of LiOH/LiOT vapor trap, purge gas cooler, cryogenic molecular sieve bed, palladium diffuser, purge gas heater, transfer pump and gas analysis systems. The schematic diagram of gas analysis system is shown in Fig. 4.4.3-6. The gas analysis system equipped at the outlet of each TBM (in front of the bioshield plug) consists of moisture detector, ion chamber, gas chromatography and small dryer bed. These detectors will be set to identify H_2, HT, H_2O and HT concentration, separately. Also, appropriate detectors will be set to the important analysis points of tritium recovery system components for monitoring of the TBM tritium recovery system performance.

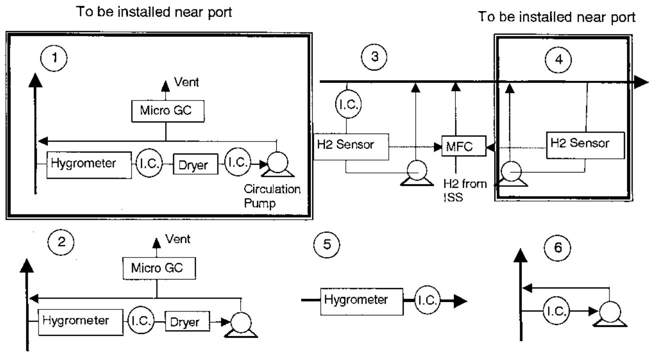


Fig. 4.4.3-5 Gas analysis system in JA TBM tritium recovery system

(5) Layout Plan of Tritium Recovery and Measurement System

Components of Tritium Recovery System are contained in glove boxes and installed in Tritium Building. Tritium Measurement System for TBM inlet and outlet purge gas is planed to be installed in a temporary transfer cask (1m × 1m × 1m) in the port cell area.

4.4.4 TBM Analysis

(1) One Dimensional Neutronics and Thermal Analysis

Figures 4.4.4-1 and 4.4.4-2 shows one dimensional TBR and temperature analysis for Water Cooled and He Cooled TBMs. As described in Sub-chapter 4.4.2, the basic configuration of module structure is DEMO blanket structure. In the case of TBM W-4 and H-4 (thermo-mechanical TBM), the highest temperature of structural material of FW, breeder layer and multiplier layer are decided to be the same as the DEMO blanket, 550, 600, 900 °C, respectively. To achieve the similarity of such temperature distribution to DEMO blanket, thickness of each layers were studied as design parameters. Figures 4.4.4-1 and 4.4.4-2 are obtained using decided set layer thickness. Total TBR of 1.42 and 1.47 are obtained for Water Cooled TBM and He Cooled TBM by one dimensional calculation.

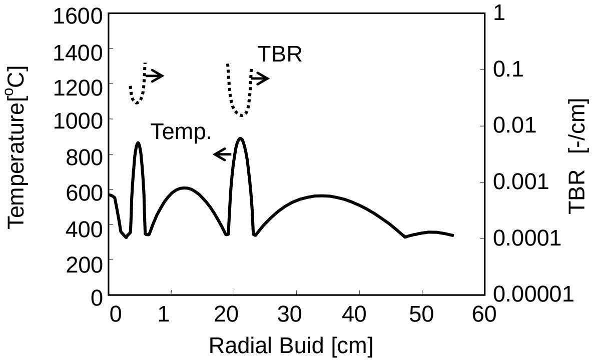


Fig. 4.4.4-1 TBR and Temperature Distributions in Radial Direction By One Dimensional Analysis for Water Cooled TBM

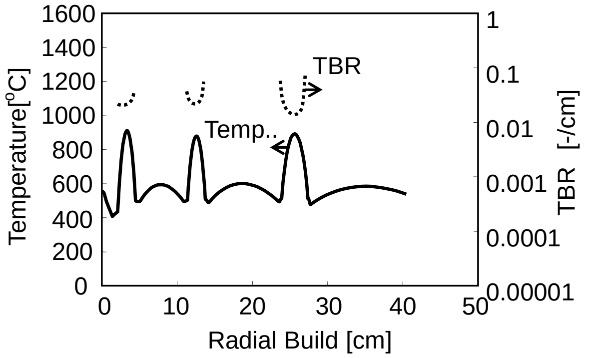


Fig. 4.4.4-2 TBR and Temperature Distributions in Radial Direction By One Dimensional Analysis for He Cooled TBM

(2) Two Dimensional Neutronics Analysis and Induced Activity and Decay Heat Analysis

Nuclear analyses were performed on Water Cooled Solid Breeder TBM and He Cooled Solid Breeder TBM by 2D calculation, by using DOT3.5, FUSION-40 (JENDL2.1 edition) for neutron transportation, APPLE-3 for the evaluation of nuclear reaction rates and ACT-4, CROSS-LIB, CHAIN-LIB and GAMMA-LIB ('90 edition) for the evaluation of induced activation rates[4.4.4-1]. Geometry of calculation model consists of TBM and its internal structure and common frame. By the calculation results, it became clear that neutron flux exists to the direction from TBM to common frame side. Consequently, integrated value of TBR is 1.13 and 1.23 for Water Cooled TBM and He Cooled TBM. As can be seen in the former section, TBR is evaluated to be decreased compared to one dimensional calculation, because of absorption of neutron by SS/Water structure of common frame. The decay heat after one year is about one order of magnitude smaller than after one day. The decay heat of structural material, F82H, at the front part of the first wall is about $2x10^{-2}$ W/cm^3. The decay heat values for first breeder layer and the first Be layer are about $1x10^{-2}$ W/cm^3 and $5x10^{-5}$ W/cm^3. Induces activity reduces in exponential function of radial location. The plotted value of induced activity of the first wall is for the fluence of about 0.3 MWa/m^2. The induced activity is lower than the DEMO condition, where the neutron fluence is about 7.5 MWa/m^2, however, the dose rate is relatively high and the necessity of remote access handling and shielding is carefully evaluated further.

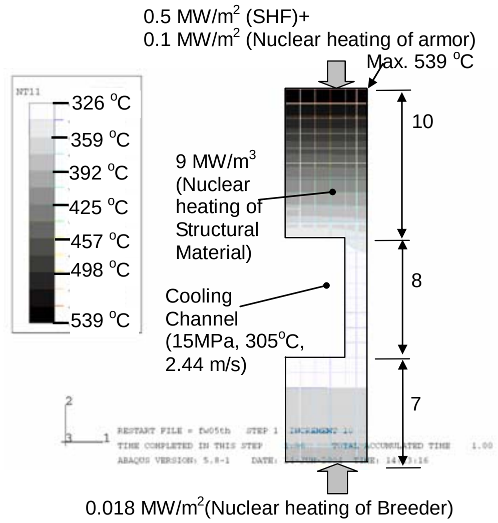


Fig. 4.4.4-3 Two dimensional temperature distribution in the cross section of first wall of Water Cooled TBM

(3) Thermo-mechanical Analysis

The structure design showed sound progress to establish detailed drawings with consideration of coolant route, fabrication and assembly procedures of modules including pebble packing. Thermo-mechanical integrity was evaluated by FEM analysis. Thermo-mechanical endurance is one of the most important test issues. Figure 4.4.4-3 shows the temperature distribution in the first wall of Water Cooled TBM evaluated by two dimensional thermo-mechanical analysis. To obtain the similarity of temperature on the first wall structure, the front part thickness was determined to be 10 mm as seen in Figure 4.4.4-3. Consequently, the highest temperature of the structural material, 539 ^{o}C, which satisfies the F82H design window, appeared at the most distant part of plasma side surface from cooling channel. Figure 4.4.4-4 shows the stress distribution in the first wall of Water Cooled TBM evaluated by two-dimensional thermo-mechanical analysis. By stress analysis, it was shown that the stress range was within elastic range. The highest TRESCA stress 359 MPa appeared at the same place as the highest temperature appeared. This stress value was evaluated to satisfy 3Sm value for F82H. Figures 4.4.4-5 and 4.4.4-6 show temperature and stress distribution in the first wall of He Cooled TBM evaluated by two dimensional thermo-mechanical analysis. In the case of He Cooling TBM, heat transfer coefficient of coolant He flow in the first wall channel is small. Consequently, the front part thickness of the first wall structure need to be 4 mm as seen in Fig. 4.4.4-5, therefore, peak TRESCA stress is smaller value 270 MPa than Water Cooled TBM case.

(4) Analysis on Tritium Behavior in TBMs

Tritium behavior in TBM module is one of most important issues for both of evaluation of tritium production function and tritium management of TBM. Tritium behavior in TBM consists of tritium generation by nuclear reaction, tritium release and inventory in solid breeder and tritium permeation to coolant water or helium through cooling tube wall. For tritium generation, nutronics calculation clarified the tritium generation rate distribution in breeder layers which is seen in Figures 4.4.4-1 and 4.4.4-2 and two-dimensional calculation result. Based on these result, tritium release and inventory performance were analyzed by using the calculation model stated by Nishikawa et al.[4.4.4-2]. By the analysis, it was estimated that tritium inventory in the first breeder layers of Water Cooled TBM and He Cooled TBM, become saturated in 3 or 4 pulse operation, which means the tritium residence time is

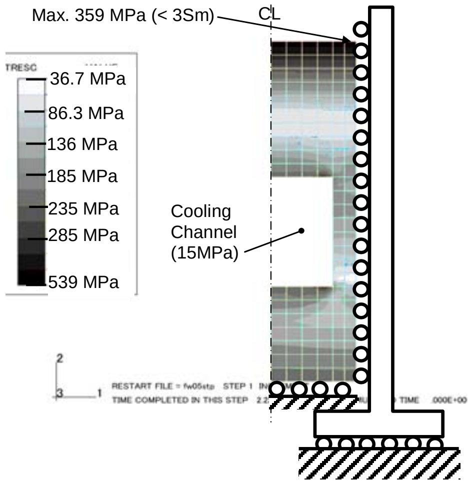


Fig. 4.4.4-4 Two dimensional stress distribution in the cross section of first wall of Water Cooled TBM

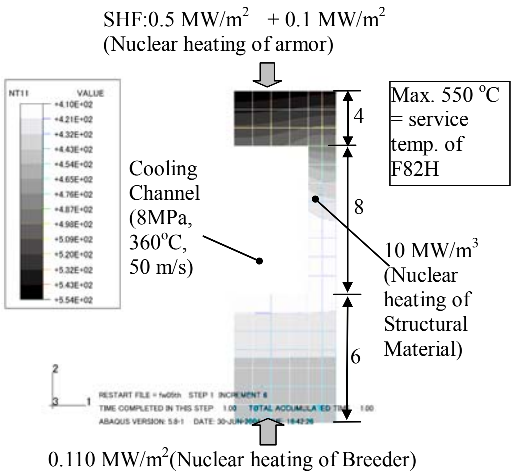


Fig. 4.4.4-5 Two dimensional temperature distribution in the cross section of first wall of He Cooled TBM

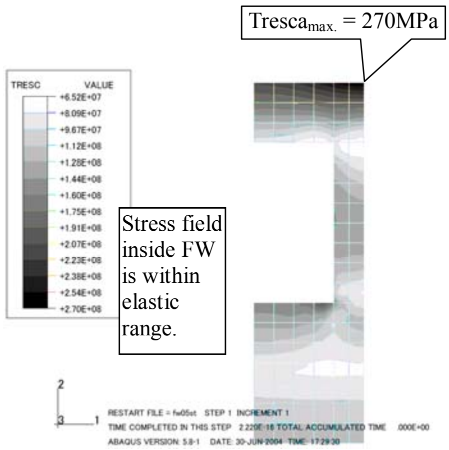


Fig. 4.4.4-6 Two dimensional stress distribution in the cross section of first wall of He Cooled TBM

less than 1200 to 1600 sec. The similar behavior is estimated in the second breeder layer for both TBM. Tritium permeation was also evaluated to be negligible if the permeation barrier with decontamination factor of about 100 was applied.

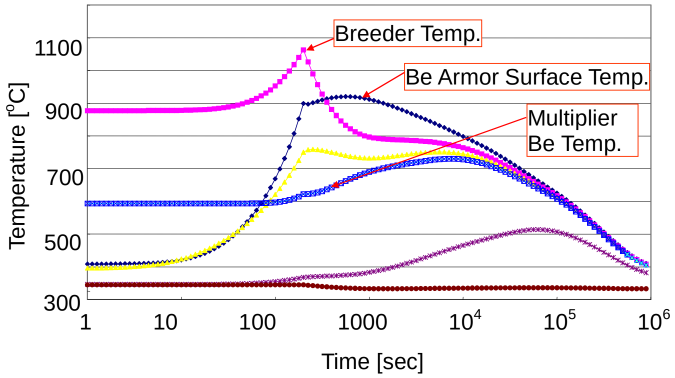


Fig. 4.4.4-7 Temperature evolution of TBM in the case where Be armor fall occurs at 900°C and plasma shut down occurs.

(5) Safety Analysis

Objective of this safety analysis is to evaluate the substantial safety of the Water Cooled TBM in such aspects as establishment of post accident cooling in the TBM, hydrogen gas generation by Be-steam reaction, and pressure increase and spilled water amount by Ingress of Coolant Event (ICE). The evaluation was performed in conservative conditions to show the upper bound of consequences in significant events, which can be assumed by the similarity of the safety analysis of the ITER shielding blanket. It is important to perform systematic identification of safety issues for detailed safety analysis on the TBM. The reference events applied in this safety analysis are specified in the ITER reference[4.4.4-3]. Most important safety analysis scenario can be summarized as follows.

(1) Loss of water coolant flow without pipe break
(2) Plasma operation and TBM heating without cooling
(3) Over-heating of structural material and pipe break
(4) Water ingress to over-heated Be pebble bed
(5) Temperature increase, with simultaneous Be - water reaction
(6) When FW temperature reaches the critical temperature, plasma operation is stopped.

Assumption of the safety analysis can be summarized as follows.
(1) Heat loss through radiation from the module wall to common frame.
(2) Reaction heat of water- Be pebble reaction is incorporated in temperature analysis.
(3) Nuclear heating rate data by 1D calculation were used.
The temperature evolution on the TBM is shown in Fig. 4.4.4-7. By the results, it was shown that temperature evolution could be converged, if the plasma is stopped by the infuse or Be armor tile fall when the FW surface temperature is 900 °C. Also, if the 0.3% cooling remains, temperature evolution is converged. Total production of hydrogen is 0.063 [mol/m^2-first wall], which is small compared to the ITER limitation of 10 kg per event.

References

[4.4.4-1] D. Tsuru, et al., "Proceedings of eleventh international workshop on ceramic breeder blanket interactions, December 15-17, 2003, Tokyo Japan", *JAERI-Conf 2004-012*, pp35-39 (2004)
[4.4.4-2] T. Kinjyo, et al., "Estimation of tritium release behavior from solid breeder materials under the condition of ITER test blanket module", to be presented in ICFRM12 (2005).
[4.4.4-3] Accident Analysis Specifications for NSSR-2 (AAS-2) Version 2.1 (June 25, 1997)

4.4.5 Supporting R&D and Supporting Activities to other Parties TBMs

(1) Strategy of Blanket Development of Japan

As described in the sub-chapter 4.4.1, ITER TBM testing is regarded as one of the most

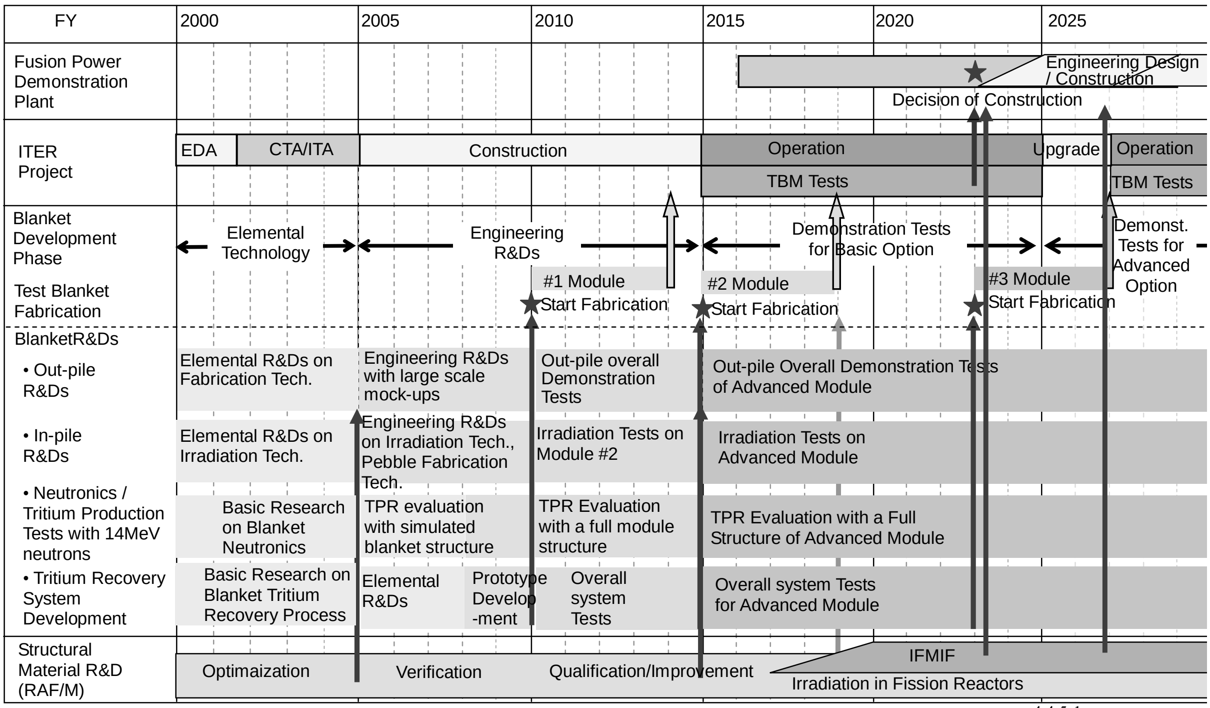


Fig. 4.4.5-1 R&D plan of solid breeder blanket and material development for DEMO[4.4.5-1].

important milestones, by which integrity of candidate blanket concepts and structures are qualified, together with material development and qualification of irradiation performance by IFMIF. Japan is investigating the possibility of testing all types of TBMs and contributing to all WSGs under the framework of TBWG with involvements of all of JAERI and universities and NIFS[4.4.5-1].

With respect to the development of the primary candidate blanket for the fusion power demonstration plant, solid breeder test blankets made of RAFM are being developed by JAERI with cooperation of universities, according to the stepwise development plan consists of elemental technology development phase, engineering R&D phase and ITER TBM test phase[4.4.5-2]. In all essential issues of blanket development, elemental technology development has been almost completed and is now stepping further to the engineering R&D phase, in which scalable mockups of solid breeder test blanket modules will be fabricated and tested to justify the total structure integrity and to certify the final fabrication specification of TBMs in the next 5 years[4.4.5-2].

With respect to the development of the advanced blankets, key issues have been addressed and critical technologies are being developed for high temperature solid breeder blanket with SiC_f/SiC structure, He cooled liquid LiPb breeder blanket with SiC inserts and its dual-coolant option, liquid Li self-cooled blanket with V alloy and molten salt self-cooled blanket with RAFM structure by universities and NIFS. The development of advanced blankets is showing steady progress, taking into account the ITER TBM testing program[4.4.5-1].

The development of blankets in Japan is showing sound and steady progress on both of solid and liquid breeder blankets under coordinated domestic development strategy, for both of primary and advanced options.

(2) R&D Achievement of Solid Breeder TBMs (WSG-1 and WSG-3)

Figure 4.4.5-1 shows the long term R&D program of the solid breeder blanket development toward the fusion power demonstration plant. To achieve the ITER module testing, the blanket development is programmed to consist of stepwise phases, Elemental Technology phase, Engineering R&D phase and Demonstration Test phases for basic options and advanced options of blankets. As the achievement of the Engineering R&D phase, the manufacturing specification and the safety demonstration data are expected[4.4.5-1].

Essential issues of the solid breeder blanket development are Out-pile R&D, In-pile R&D, Neutronics and Tritium Production Tests with 14 MeV Neutrons and Tritium Recovery System Development. Out-pile R&D consists of the development of blanket module fabrication and the development of thermo-mechanical and chemical compatibility design database of breeding region of the blanket. In-pile R&D consists of the development of irradiation technology for partial blanket mockups in fission reactor, the development of fabrication technology for breeder and multiplier pebbles, and irradiation tests of breeder and multiplier pebble beds. Neutronics tests by 14 MeV neutron source consists of the precise evaluation of neutronics characteristics of the blanket materials and tritium production rate data with real blanket materials and mockups. The development of tritium recovery system consists of the development and basic research on the processes of blanket tritium recovery system.

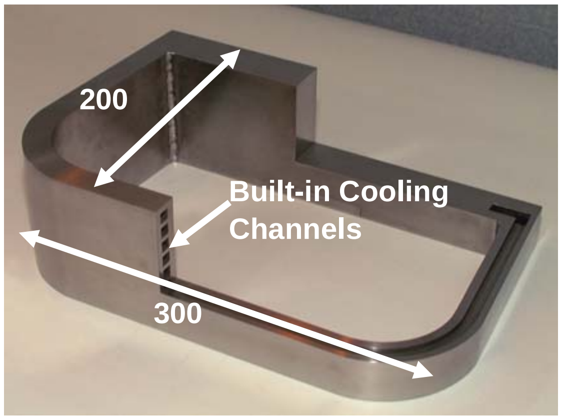


Fig. 4.4.5-2 Fabricated Mockup of Box Structure with Built-in Cooling Channel.

Fabrication technology of the blanket box structure

Reduced activation martensitic ferritic steel, F82H (8Cr-2W-V-Ta) has been developed for fusion application[4.4.5-3]. Material improvement and property data accumulation have been performed by material development group of JAERI and universities[4.4.45-4, 4.4.5-5]. In fabrication technology development of the DEMO blanket in JAERI, a hot isostatic pressing (HIP) bonding method, especially for the first wall structure with built-in cooling tubes has been proposed. Preliminary screening tests have been performed to obtain suitable conditions of HIP bonding and after HIP heat treatment[4.4.5-6]. By using investigated conditions, first wall panel mockup has been fabricated to be tested by high heat flux tests. Under the heat load of 2.7 MW/m^2 up to 5000 cycles, the fatigue performance of the high heat flux tests showed soundness of the fabrication of the first wall mockup, compared with the data of the base metal certified by IEA round robin tests[4.4.5-7]. The box structure mockup with the wall with built in cooling channels was also fabricated for investigation of the manufacturing procedures as shown in Fig. 4.4.5-2. Due to the grain coarsening by HIP heat treatment, the reduction of fracture toughness of HIP joints in the first wall was detected. However, the further investigation clarified the heat treatment conditions for recovery of ductility and the improvement of HIP condition[4.4.5-8]. By the result, heating above 1100 °C and consequent normalizing process was needed to cancel the previously coarsened grain size, which

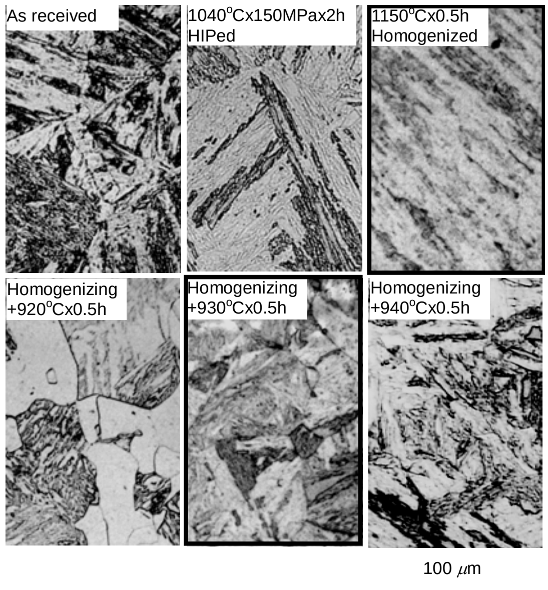


Fig. 4.4.5-3 Typical microstructure of post hip heat treated F82H[4.4.5-8].

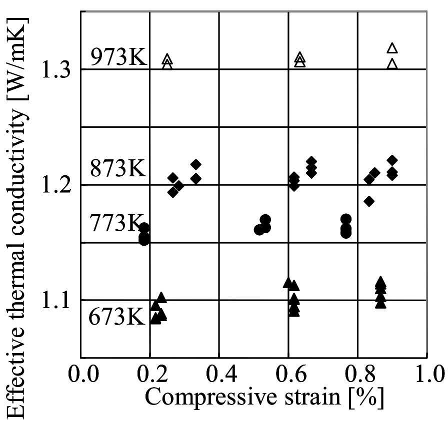


Fig. 4.4.5-4 Effective thermal conductivity of a compressed Li_2TiO_3 pebble bed[4.4.5-14].

temperature is also applicable for HIP treatment. Post HIP heat treatment (PHHT) is also investigated. In order to obtain fine prior austenite grain, it is required to normalize just above the temperature (910ºC) where the structure consists wholly of austenite. By these results, it was suggested that the HIP process (HIP at 1150 ºC + PHHT at 930 ºC + Tempering) could improve both the joining properties and the fracture toughness[4.4.5-8], as shown in Fig. 4.4.5-3. Also, the development of first wall armor joining to the first wall of RAFM has shown progress. As one of the candidate armor material for fusion power demonstration plant, solid state bonding of tungsten and F82H was studied by using Spark Plasma Sintering (SPS) method. According to the results of trial bonding and destructive observation, W and F82H could be joined by the solid-state-bonding without any insert material[4.4.5-9]. For the first wall armor of TBMs, beryllium is recommended. For Be armor joining for TBMs, the further R&D is needed based on the technique of HIP joining of Be and Cu alloys[4.4.5-10].

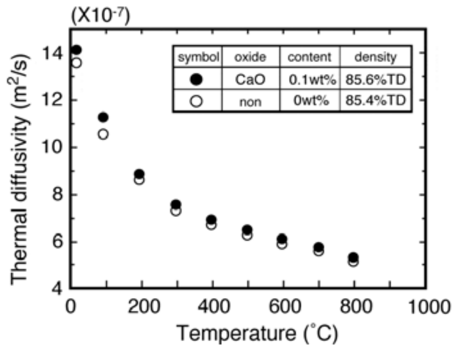


Fig. 4.4.5-5 Thermal diffusivity of Li_2TiO_3 added with CaO, and Li_2TiO_3 without additive[4.4.5-16].

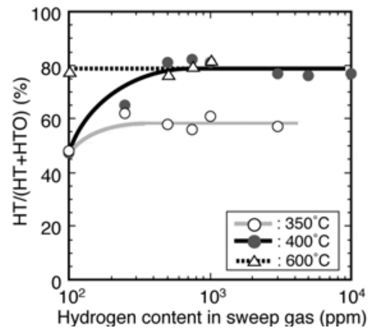


Fig. 4.4.5-6 Influence of hydrogen content in the sweep gas on the fraction of HT; (the sweep-gas flow rate was 200 cc/min)[4.4.5-19].

Investigation on Thermo-mechanical Characteristics of Breeder and Multiplier Pebble beds

As a most important base data, effective thermal conductivities of breeder pebble beds were researched by using hot wire method under the frame of the IEA-IA fusion nuclear technology collaboration[4.4.5-11, 4.4.5-12]. From the results, fitting parameters for correlation of thermal conductivity was determined to represent the observed value of major candidate breeder pebbles. By measured results of effective thermal conductivity of single and binary packed beds of Li_2TiO_3 pebbles fabricated by wet method, it was shown that the estimated values by using the same fitting parameter obtained by single packing bed showed fair agreement with the obtained data of the binary packing bed[4.4.5-13]. Also, effective thermal conductivity of the bed was measured under compressive load up to 10MPa at temperatures ranging from 673K to 973K. As can be seen from Fig. 4.4.5-4, at all temperatures, increases of effective thermal conductivity due to the compressive deformation were confirmed. According to the observation, the change of effective thermal conductivity was not significant. When successive loads, heating-cooling or compression, worked on the bed, effective thermal conductivity increased according to promotion of the compressive deformation[4.4.5-14]. Therefore, further investigation is planned to evaluate pebble bed integrity under cyclic and long term compression in high temperature.

Based on the achievements stated above, the out-pile R&D with engineering scale mockups is foreseen as the next step.

Development of Tritium Breeder Material

Lithium titanate (Li_2TiO_3) has been selected as the first candidate material for the fusion power demonstration plant from viewpoints of effective tritium release and high chemical stability at the operating temperatures. A wet production process for Li_2TiO_3 was developed, which is suitable for mass production and recycling of ^{6}Li. This process successfully supplied ^{6}Li-enriched Li_2TiO_3 pebbles which satisfied the target values of the specifications: namely, density of 80-85%TD, diameter of 0.85-1.18mm and grain size <5μm, as well as a good sphericity value of smaller than 1.1[4.4.5-15]. Furthermore, Li_2TiO_3 with oxide additive (CaO, ZrO_2, Sc_2O_3) has been being developed recently in order to control the growth of the grain size and to keep the stability of its crystal structure at high temperatures. Pellet density measurement of Li_2TiO_3 with the additives showed that these oxide additives were effective in controlling the grain growth. Thermal diffusivities for the CaO-added Li_2TiO_3 are slightly higher (at maximum by 5% from room temperature to 400˚C) than those for Li_2TiO_3 without additive, as shown in Fig. 4.4.5-5[4.4.5-16]. Similar results were obtained for ZrO_2 and Sc_2O_3

addition. The overall results suggest that the oxide addition is effective not only in controlling the growth of the grain size but also in improving the thermal properties.

Development of Neutron Multiplier Material

Beryllium (Be) metal is a reference material for neutron multiplier, and semi-industrial fabrication technology of beryllium pebbles was established by a rotating electrode method. However, Be has temperature limit of 600˚C, because of high chemical reactivity and large swelling of Be metal. Therefore, Be pebble bed design tends to limit the performance of the breeding blanket of the fusion power demonstration plant which is operated with high temperature coolant up to high neutron dose (20,000 atomic ppm He and 50 dpa).

Recent R&D is, therefore, focused on Be alloys, in particular Be-Ti alloys[4.4.5-17], because they are promising as advanced multiplier materials due to their superior properties, such as high melting points and high chemical stability at high temperatures.

Trial fabrication tests of Be alloy pebbles were performed in the rotating electrode method to reduce the brittleness of the stoichiometric $Be_{12}Ti$ (Be-7.7at%Ti). As a result, it was revealed that a two phase structure of $Be_{12}Ti$ and αBe was effective in reducing the brittleness. A preliminary fabrication of a small amount of Be-5at%Ti and Be-7at%Ti pebbles were successfully performed without electrode break by thermal stress[4.4.5-18]. Characterization of the Be-Ti alloys produced was performed in cooperation with universities in Japan[4.4.5-19].

Development of Irradiation Technology for In-pile Functional Tests

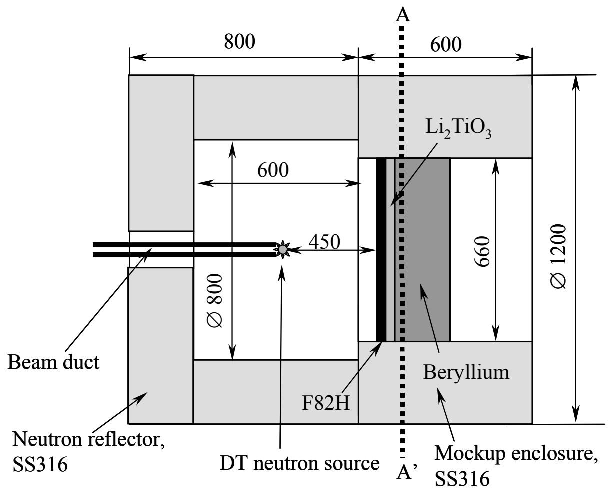
Various irradiation techniques have been developed for in-pile functional tests of tritium breeding materials. Main items are: 1) pulse irradiation technique by changing the neutron flux with a neutron absorber (hafnium) window rotated by a radiation-resistant small motor, 2) multi-paired thermocouples for measuring temperatures at many points, 3) a highly sensitive and responsive self-powered neutron detector (SPND), and 4) ceramic coating for reducing tritium permeation through the structural material to enable a reliable in-pile tritium release experiment.

An integrated in-pile test is being performed in the Japan Materials Testing Reactor (JMTR) by using above techniques[4.4.5-19]. Measurement of tritium release from a Li_2TiO_3 pebble bed revealed that the fraction of HT of the total amount of tritium, HT/(HT+HTO), increased with increasing the hydrogen content in the sweep gas, when the center temperature of the pebble bed was 400˚C[4.4.5-19, 4.4.5-20]. On the other hand, the fraction was almost constant independent of the hydrogen contents, when the center temperature was 600˚C (see Fig. 4.4.5-6). These results suggest that the tritium release was controlled by a reaction at the surface of the Li_2TiO_3 pebble, because the surface reaction is influenced by the hydrogen content as well as the pebble bed temperature.

Neutronics / Tritium Production Tests with 14MeV neutrons

Neutronics integral experiments have been conducted with small partial mockups relevant to the ITER test blanket module proposed by JAERI using DT neutrons at FNS of JAERI[4.4.5-21]. Figure 4.4.5-7 shows the schematic view of the experiment assembly. Small partial mockups of the ITER test blanket modules were installed at about 450 mm distant from the DT neutron source. The neutron reflector applied in this experiment relates to the effect of the incident back-scattering neutron current.

Numerical analyses were conducted by using the Monte Carlo neutral particle transport code MCNP-4C and the fusion evaluated nuclear data library FENDL-2. Figure 4.4.5-7 shows the ratio of the calculated value to the experimental value



cross-sectional view (unit: mm)

Fig. 4.4.5-7 Schematic view of experiment assembly of neutronics test of blanket mockup[4.4.5-21].

(C/E) distribution. The C/Es are 0.96 –1.08 and 1.03 – 1.18 for the experiments without and with the reflector, respectively. The calculation results of the integrated tritium productions agree well with the experimental data within 2 %, i.e. 7 % of the experiment error, and 11 % for the experiments without and with the reflector. From this study, it was clarified that the integrated tritium productions could be accurately predicted for the experiment without the reflector. Uncertainties of the experiment with the reflector are larger than those without one. It is reasoned that this occurs due to uncertainties of the cross section data about the back-scattering neutrons from the SS316. Thus, in conclusion, the prediction accuracy is in the range of 2 – 11 % on the calculation of the tritium breeding ratio (TBR) in the blanket design at this point. Further investigation will be performed to improve the prediction accuracy.

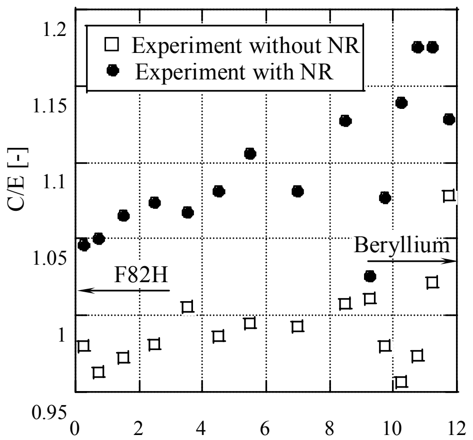


Fig. 4.4.5-8 Ratio of the calculation result to the experiment one (C/E) about the TPR from 6Li for the experiments with and without the neutron reflector (NR)[4.4.5-21].

Tritium Recovery System Development

In the present plan, bred tritium is taken out by passing of H_2-added helium sweep gas, because of the reduction of tritium inventory and the enhancement of tritium release. The kind/quantity of additive for sweep gas and the sweep gas condition at the blanket outlet are important information for development of breeding blanket interface system (BBI)[4.4.5-22]. Generally, the concentration of H_2 in the sweep gas is assumed to be 0.1-1.0%. The chemical species to be processed in BBI are HT, H_2, HTO and H_2O. So, the processing with separation by chemical form is reasonable.

In JAERI, development of cryogenic molecular sieve bed (CMSB) has been carried out for processing of HT and H_2[4.4.5-23]. This is the packed bed of porous adsorbent such as molecular sieve 5A, and is used at 80K. Recently, system integration proof test of CMSB, fuel cleanup system (FCU) and isotope separation system (ISS) has been carried out. Hydrogen isotopes containing tritium are adsorbed on CMSB, and CMSB is vacuumed and heated up to release adsorbed gas in regeneration step. Released gas is sent to ISS after purification by Pd diffuser in FCU. Figure 4.4.5-9 shows the dynamic behavior of simulated fuel processing and CMSB system in regeneration step. The amount of hydrogen isotopes supplied to CMSB agreed well with the amount of hydrogen isotopes recovered in ISS, and the demonstration of the integrated system has succeeded.

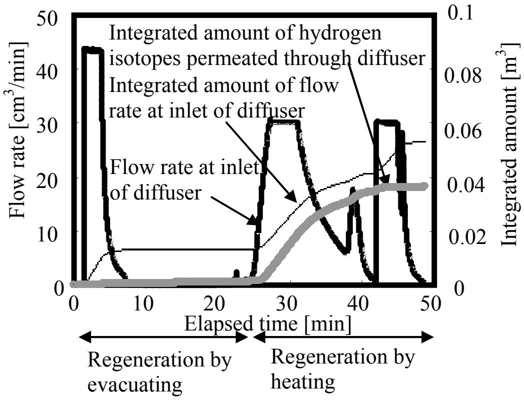


Fig. 4.4.5-9 Dynamic behavior of simulated fuel processing and CMSB systems in regeneration step.

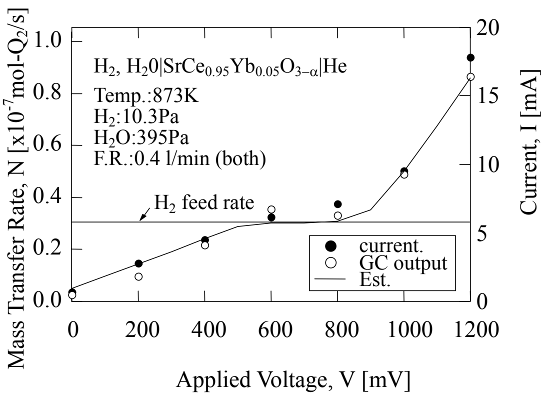


Fig. 4.4.5-10 An example of ionic hydrogen transportation property of hydrogen pump[4.4.5-24].

In DEMO plant, large amount of high temperature sweep gas should be processed effectively. So, in JAERI, the electrochemical hydrogen pump using proton conductor membrane has been proposed as the trade-off system, and its study is going on. The driving force of hydrogen transportation is a potential difference, and hydrogen transportation from low pressure side to high pressure side is possible. This property is suitable for the blanket sweep gas condition. Figure 4.4.5-10 shows an example of ionic hydrogen transportation property of the hydrogen pump using $SrCe_{0.95}Yb_{0.05}O_{3-\alpha}$ membrane. When the applied voltage exceeds 0.8V at 873K, hydrogen transportation via water decomposition also appears. BBI using hydrogen pump may become very efficient system.

When H_2-added sweep gas is applied, tritium leakage via permeation to coolant can not be ignored. In the case where water is used as the coolant, the increase of the load of water detritiation system (WDS) in the tritium plant is not avoided. In ITER design, the chemical exchange combined electrolysis (CECE) method is used. In DEMO plant, it is expected that the amount of water which is processed in WDS increase. So, the reduction of the amount of water by concentration of tritium is necessary to apply CECE to DEMO plant. In JAERI, the pressure swing adsorption (PSA) method by synthetic zeolite packed bed has been proposed to concentrate tritium in water. By observation of breakthrough curve of H_2O and HTO, it was shown that separation of HTO by PSA method is possible[4.4.5-24].

(3) Unit Testing Concept using a Sub-module of He Cooled TBM (WSG-1)

Japanese Helium Cooled Solid Breeder TBM applies integrated structure of three sub-modules. By using one of sub-modules, there is a possibility of testing different concept of internal configuration of the solid breeder blankets. Figure 4.4.5-11 shows a typical concept of testing unit of high temperature solid breeder / SiC_f/SiC blanket cooled by He. One of Japanese commercial fusion plant uses solid breeder blanket with SiCf/SiC structure cooled by high temperature He. In this testing concept, it was proposed to insert a test article of a solid breeder blanket surrounded with the thermal insulation wall of SiCf/SiC inside the sub-module box structure made by RAFM. By adjusting the flow rate of He coolant to the test article, the operation temperature of the test article is raised to higher temperature than 550 oC, for the purpose of testing the thermo-mechanical characteristics. The detailed structure including the support structure of the test article in the sub-module box will be investigated in future. In the latter 10 year period of ITER operation, the testing of TBM made of SiCf/SiC first wall may be also considered, depending on the development progress of materials and module fabrication technology and the result of the test article testing.

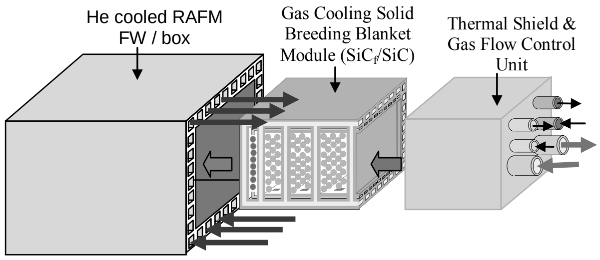


Fig. 4.4.5-11 The Concept of SiC_f/SiC Blanket Unit Tests in RAFM Structure.

(4) He Cooled LiPb blanket (WSG-2)

Liquid LiPb blanket is one of major option of the TBM that attracts interests of all 6 parties. Helium Cooled Lithium Lead (HCLL) is intended to be tested to aquire maximum information for DEMO design in EU, and this TBM option is developed by European leadership with all other parties involvement in SWG2. However Lithium-lead has an improved option of Dual Coolant Lithium Lead (DCLL) concept for higher temperature operation. The original EU design of the HCLL module is made of RAFM vessel filled with liquid lithium lead, and cooled with cooling panel where high pressure helium is circulated. With SiC insert that separates Lithium Lead from RAFM structure, Lithium Lead can be circulated at higher temperature and flow rate because of heat and electrical insulation. Metal surface is protected from corrosion or erosion. Lithium lead works as heat transfer media, and provides an attractive option for high temperature blanket above the limit of RAFM with ITER/TBM. Japan showed the interest and possibility of technical contribution on the investigation of SiC incert and design of DCLL option. Identified subjects to be studied includes SiC/LiPb and RAFM/LiPb compatibility, evaluation of MHD effect, and the development of SiC insert. Kyoto university recently started the operation of a small LiPb loop as a collaboration with JAERI, and will pursue above technical issues to be combined with their strong technical capability of

SiC research. Through the expected results with international efforts, original HCLL will elevate the operation temperature gradually as DCLL.

It is pointed out that many of the advanced reactor design including Japanese applies LiPb-SiCf/SiC blanket to provide high grade heat above 900 °C, and this DCLL option provides practical and realistic technical approach toward them starting from HCLL day-1 TBM as a conservative design. Because Japan has a strategy to develop economical fusion reactor with single stape of DEMO following ITER, such a multiple generations of blanket to gradually and steadily improve the plant performance is important.

(5) Self-cooled Li/V blanket (WSG-4)

A Design Description Document was presented from Russia based on Li/Be/V blanket concept, in which Be was used for neutron multiplying purposes[4.4.5-25]. Japanese WSG-4 is technically supporting the Russian design but independently examining a Li/V test module, in which Be is not used[4.4.5-26]. The Li/V concept has some advantages over Li/Be/V concept such as simple blanket structure, being free from the issues of natural resource limit and handling safety concerning beryllium and no need for periodic replacement of blanket because of the lifetime of Be. Elimination of Be multiplier in the breeding blanket will give benefits of cost reduction and safety enhancement. This aspect is also the case for solid breeder TBMs, however, it is difficult to eliminate Be multiplier in the solid breeder blankets with limited thickness of radial build, without avoiding insufficient tritium breeding performance. Recent neutronics calculations showed enough tritium self-sufficiency of Li/V blanket in tokamak[4.4.5-26] and helical[4.4.5-27] systems.

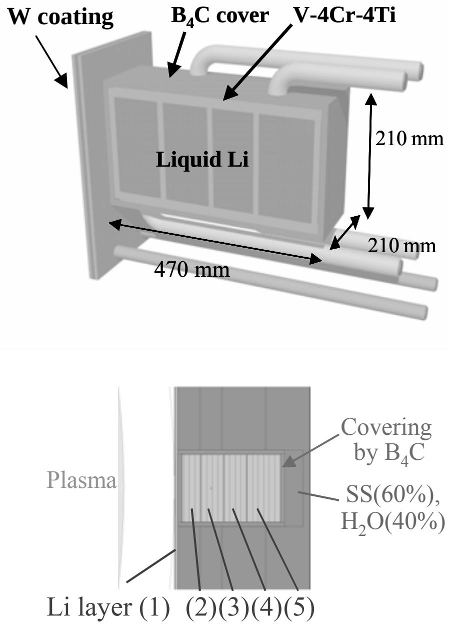


Fig. 4.4.5-12 The schematic view and cross section of the Li/V test module[4.4.5-26].

The primary purpose of the Li/V module test was defined as validation of the tritium production rate predicted based on the neutron transport calculation. For this purpose the module was designed to be composed of sectioned thick boxes which accommodate slow Li flow. The schematic view and cross section of the module is given in Fig. 4.4.5-12. This system enables to measure the tritium production rate as a function of the distance from the first wall. The size of the four boxes was limited (~0.027m^3) so as to satisfy the introduction limit of liquid lithium into the ITER test port. The module is covered with a B_4C layer for the purpose of shielding thermal neutrons. Figure 4.4.5-13 shows the neutron spectra for ITER first wall, Li/V TBM with B_4C cover of 7.5 mm thick and V/Li full blanket[4.4.5-26]. With the B_4C cover, the flux of low energy neutrons decreases and the spectrum approaches that of the Li/V full blanket. The coating with W was shown not to influence TBR[4.4.5-27]. The plasma-facing surfaces of the module would be covered with W coating. The effect of tungsten coating on the tritium production performance was investigated for a Li/V DEMO blanket[4.4.5-27], which showed a positive effect of the coating thickness on the tritium production rate in the case using 35% enriched ^{6}Li. The

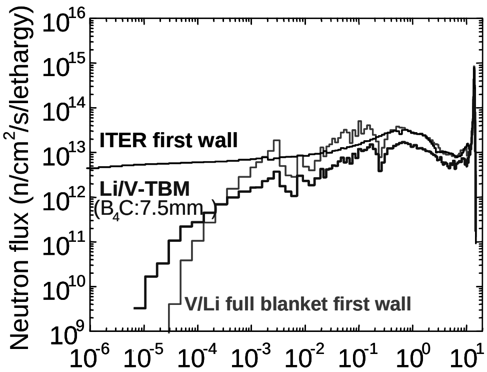


Fig. 4.4.5-13 Comparison of neutron spectra for ITER first wall, Li/V TBM with B_4C cover of 7.5 mm thick and V/Li full blanket[4.4.5-26].

feasibility of the plasma spray coating of W on V-4Cr-4Ti was demonstrated[4.4.5-28]. Significant progress has been made in fabrication technology of vanadium alloyed with focus on V-4Cr-4Ti alloys including fabrication of large V-4Cr-4Ti ingot, manufacturing into plates, sheets, wires, rods and thin tubes, laser welding and so on[4]. Thus manufacturing the test module with high quality is thought to be feasible for V-4Cr-4Ti structures. In Li layer (1), MHD insulator coating is necessary, and the test of the coating is one of the objectives of the layer. Current options of the coating would be (a) PVD coating of Er_2O_3 or Y_2O_3, (b) two-layer coating with Er_2O_3 (or Y_2O_3) covered by vanadium alloys, and (c) in-situ coating of Er_2O_3 by reaction of pre-doped Er in Li and pre-doped O in V-4Cr-4Ti structural materials[4.4.5-29]. As to the tritium recovery from Li, feasibility of gettering tritium by yttrium was demonstrated[4.4.5-30]. Although tritium recovery technology from tritiated yttrium is not verified, this method seems to be feasible to the module test where limited amount of tritium needs to be recovered.

(6) Flibe Blanket (WSG-5)

One of the candidate liquid breeder blankets applies molten-salt Flibe as the self cooling breeder[4.4.5-31]. It has been designed for Force-Free Helical Reactor, FFHR, which is a demo-relevant helical-type D-T fusion reactor based on the great amount of R&D results obtained in the LHD project. It features the reduced activation ferritic steel, JLF-1, or Vanadium alloy, V-4Cr-4Ti, as the structural material and Be pebble bed as the neutron multiplier and redox controller to reduce corrosive F radicals. To enhance the shielding ability, C and B_4C are placed in the rear side region of the Flibe blanket.

Investigation of key technologies of the Flibe blanket have been performed on the thermo-fluid study using Tohoku-NIFS Thermo-fluid Loop for molten salt (TNT Loop) built in Tohoku University (Fig. 4.4.5-14)[4.4.5-32], redox control technology to reduce F radicals, tritium inventory and disengaging technology of molten-salt Flibe partly in the frame of Japan-US collaboration program, JUPITER-II. Flibe chemistry experiments and TBM neutronics calculations are on going in Universities and NIFS. In parallel, the development of structural materials and structure fabrication technology are being performed.

Design and development of Flibe TBM is mainly being performed by US. Japanese universities and NIFS are cooperating its design activity in the frameworks of TBWG WSG and JUPITER-II program. The technology of thermo-hydraulics of Flibe can be covered by the scale of TNT Loop. The further R&D achievements are expected to be obtained by TNT Loop.

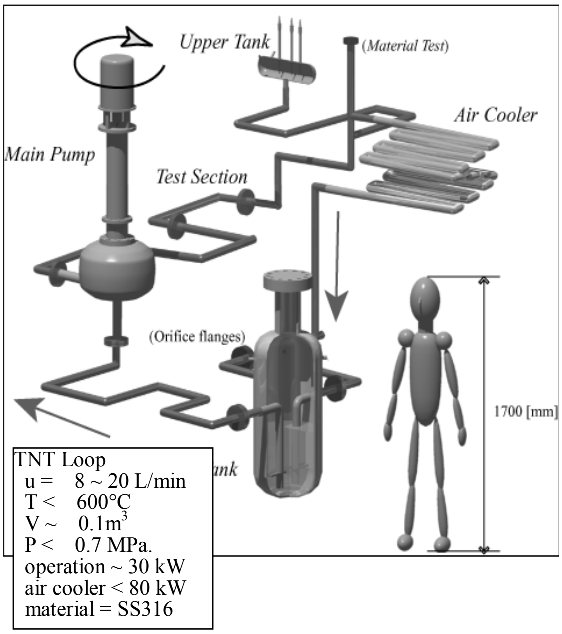


Fig. 4.4.5-14 Tohoku-NIFS Thermo-fluid Loop for molten salt (TNT Loop) built in Tohoku University[4.4.5-32].

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4.4.6 Validation Program prior to the installation in ITER

ITER blanket module testing is the first unique test bed of blanket module in a fusion environment although the load conditions on the TBM in ITER are not fully relevant to the fusion power demonstration plant. By the testing, it is necessary to qualify the methodology of blanket design and integrated performances and reliability of the blanket system under the integrated load conditions. Specific issues to be qualified are:

(a) tritium production consistent with neutron spectrum
(b) high grade heat generation relevant to the electricity generation
(c) thermo-mechanical behavior of the module with pebble beds under surface heat load and neutron irradiation
(d) tritium release characteristics from the pebble bed
(e) tritium permeation to the coolant
(f) corrosion of the cooling channel and activated corrosion products
(g) integral behavior of all elements in the blanket system
(h) reliability of the blanket system

As described in Sub-chapter 4.4.5, prior to the ITER blanket module testing, Out-pile R&D, In-pile R&D, Neutronics / Tritium Production Rate (TPR) tests with 14 MeV neutron source and Tritium Recovery System Development will be carried out together with the material development. By the Engineering R&D, which consists of above mentioned four major R&Ds, integrity of TBMs should be evaluated by using scalable TBM mockups to show the relevancy of TBMs to the module testing and the finalization of specification of manufacturing of TBMs (TBM W-1 and TBM H-1) in 5 years. Also, by using the prototype TBM mockup, relevancy to ITER safety standard will be demonstrated.

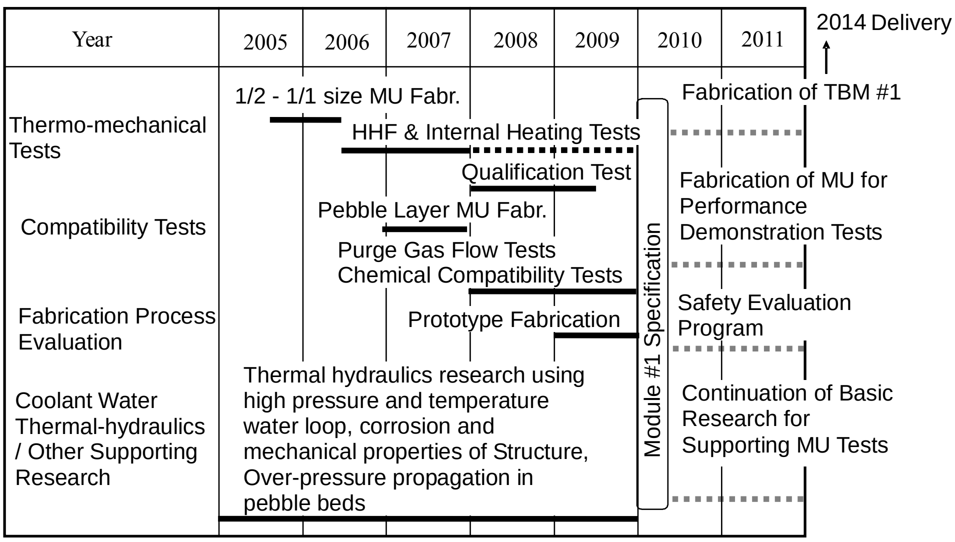


Fig. 4.4.6-1 Time schedule of development and validation of Water Cooled and He Cooled Ceramic Breeder TBM.

Figure 4.4.6-1 shows the time schedule of validation of Water Cooled and He Cooled Ceramic Breeder TBM. The validation program is planned for 10 years. In the former 5 year period, which is expected to start from 2005, half size to full size mockups for validating thermo-mechanical performance and purge gas chemical stability will be fabricated and tested under TBM relevant conditions. In the thermo-mechanical validation tests, the loading conditions of surface heat flux will be given using ion beam irradiation. Simultaneously, the volumetric heating will be simulated by insert panel heater in the pebble bed of the mockup. Schematics of the thermo-mechanical validation test is shown in Fig. 4.4.6-2. For the coolant, high pressure and temperature water (25 MPa and 450 ^{o}C) will be used. The

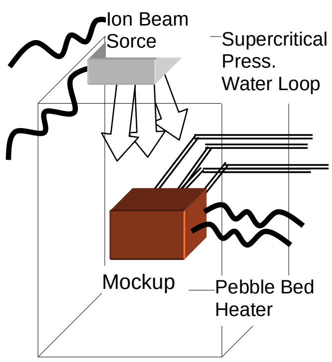


Fig. 4.4.6-2 Concept of Qualification Test of Water Cooled and He Cooled TBM.

supercritical water loop is already constructed in the existing high heat flux test facility in JAERI. After thermo-mechanical validation tests and chemical stability validation tests, qualification tests should be performed to show justification of the capability of fabrication and delivery of the real TBMs on time. In the final stage of the former five year period, the fabrication of prototype TBM will be performed to identify the detailed technical conditions of the fabrication process. As the goal of the former period of validation program, the detailed fabrication specification will be decided. In the later five year period, the validation activity is focused on the evaluation of the safety performance of the TBMs. Basic safety related performance is planned to be investigated in the former five year period, and they will be extended in the later five year period. In the course of later five year period, the detailed fabrication specification of each TBMS for DT operation tests on time.

4.5 Republic of KO Proposals

4.5.1 DEMO Studies and Testing Strategy in ITER

One of the key missions of ITER is to validate the design concepts of tritium breeding blankets relevant to a power producing reactor like a DEMO. The ITER should demonstrate the feasibility of the breeding blanket concepts that would lead to a tritium self-sufficiency and the extraction of a high grade heat, suitable for an electricity generation in a fusion reactor.

KO defined the nuclear technology roadmap of fusion, as shown in Fig. 4.5.1 [4.5-1]. The development strategy consists of several major programs: KSTAR for the study of a long-pulse, advanced tokamak operation, ITER for the burning plasma experiment, DEMO for the demonstration of producing net electricity from a fusion reactor, and a commercial fusion reactor. Material testing and integral testing of the fusion reactor components needs to be done using IFMIF and CTR (Component Test Reactor). To accomplish the mission of each program, not only plasma physics and fusion engineering for the construction and operation of facilities, but also the fusion reactor technologies need to be developed in parallel, by making use of the expertise existing in the KO fission program which shares key technologies with a fusion reactor. In addition to electricity production, the fusion technology developed through this strategy is expected to support applications such as a transmutation of high-level nuclear waste and a hydrogen production.

DEMO is regarded as the last step before the development of a commercial fusion reactor and the major requirements of DEMO are:

- It should demonstrate a net electric power generation.
- It should demonstrate a tritium self sufficiency.
- The blanket system should have a reasonably high thermal efficiency to show an extraction of high-grade heat and positive evidence of a low cost for the electricity.
- It should demonstrate the safety aspect of a power plant and it should be licensable as a power plant, i.e., even in the case of the worst accident, no evacuation plan is necessary.

With the above requirements and a limited extension of the improved plasma physics and technology from the 2nd phase of the ITER operation (EPP phase), various DEMO concepts can be defined depending on the breeding blanket concept and the mission of DEMO. Detailed studies on the scope of the technical parameters of DEMO are underway, but the major technical parameters of DEMO can be summarized as follows:

- Fusion power is about 2 GW for a net electricity generation
- Neutron wall loading is above 2 MW/m^2
- Maximum FW heat flux is less than 1.0 MW/m^2
- Low-activation structural material is used
- Thermal efficiency is above 30 %

The breeding blanket concept has to be defined as a priority in any DEMO study since it is essential for characterizing DEMO. Breeding blankets proposed for testing in ITER should demonstrate the key functions of a DEMO blanket and provide sufficient information which validate the design concepts and the operational feasibility for the DEMO blanket. KO proposes two DEMO relevant blanket concepts for testing in ITER. One is the Helium Cooled Solid Breeder (HCSB) blanket and the other is the He Cooled Molten Lithium/FS (HCML) blanket.

The Helium Cooled Solid Breeder (HCSB) blanket uses He as a coolant, Beryllium as a neutron multiplier, and Ferritic/Martensitic Steel (FMS) as a structural material. The ceramic breeder is a Li-based compound, Li_2SiO_4 in a pebble-bed form and Be is also used in a pebble-bed form. The He coolant is at a static pressure of 8 MPa with an inlet temperature of 300°C and an outlet temperature up to 500°C depending on the operating conditions. He with 1% hydrogen is used as a purge gas for the tritium extraction from the lithium containing ceramic(Li_2SiO_4).

The He Cooled Molten Lithium (HCML) blanket uses He as a coolant at an inlet temperature of 250-350°C and an outlet temperature up to 550°C and Li is used as a tritium breeder. Its potential advantages are as follows: virtually no concern for a tritium permeation into the coolant system; simplified high-performance system with a He-direct cycle; alleviated material problems due to a very slow Li flow speed; no concern for a Li fire in an inert gas environment; marginal MHD (Magneto-Hydro-Dynamics) effects due to a very slow Li flow; no Po-210 & Hg-204 generation; Li loop as a redundant cooling circuit in the case of the He loss accident. With one or two layer(s) of a graphite reflector inserted in the breeder zone, the TBR and the shielding performances can be increased. The graphite reflector thickness and ^{6}Li enrichment need to be optimized for a self-sufficient TBR [4.5-2].

To assess the performances of the DEMO relevant blanket concepts in ITER, the testing strategy is:

- For HCSB TBM, the functional tests of a small size sub-module will be performed from a day-one operation of the ITER and independent TBM will be tested from a later phase (D-T phase) of the ITER operation.
- HCML TBM will be tested from a day-one operation of the ITER if its acceptability for an installation is proved by positive R&D results from its technical feasibility and safety validation

In addition to the above 2 TBMs, KO has also interests in the R&D progresses of other TBM families and wants to contribute to the development of those TBMs since the blanket concept as well as the DEMO concept needs to be continuously updated based on the R&D achievements. International collaboration on the R&D and the testing of the TBMs are important due to a shortage of testing space in the ITER. KO supports fully a coordinated test program for space and time sharing, and the development of common TBMs under the ITER framework.

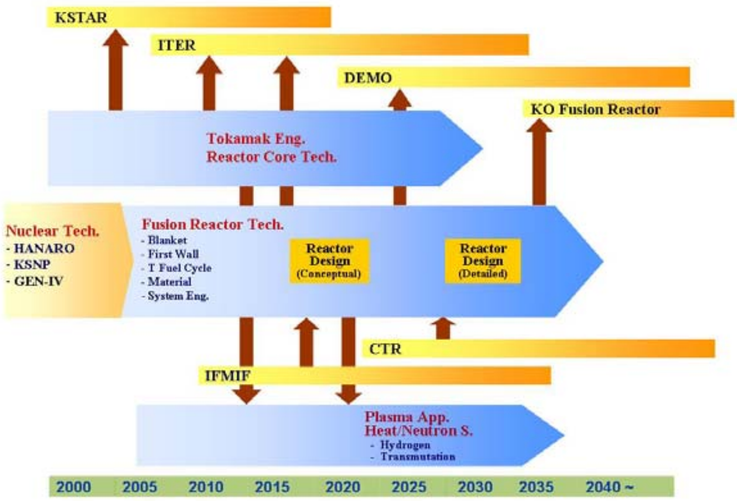
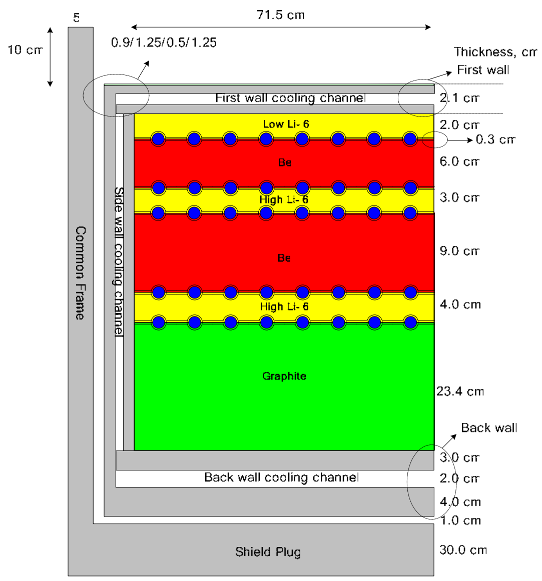


Fig. 4.5.1 The technology roadmap of fusion in KO

4.5.2 TBMs design and analysis

4.5.2.1 Helium Cooled Solid Breeder (HCSB)

Lithium ceramics have been utilized as the breeding materials in a breeder design. Because Lithium provides a limited Tritium Breeding Ratio (TBR) performance, Be is usually used as a neutron multiplier. Be has a high multiplication efficiency and is regarded as one of the best neutron reflector materials. However, the natural resource of Be is very limited and hazardous to human being as well. Consequently, the cost of Be is very high and it is highly reactive with water, generating hydrogen gas. Therefore, in the KO HCSB TBM design, the amount of Be is reduced by replacing some of it with graphite as a reflector [4.5-3].



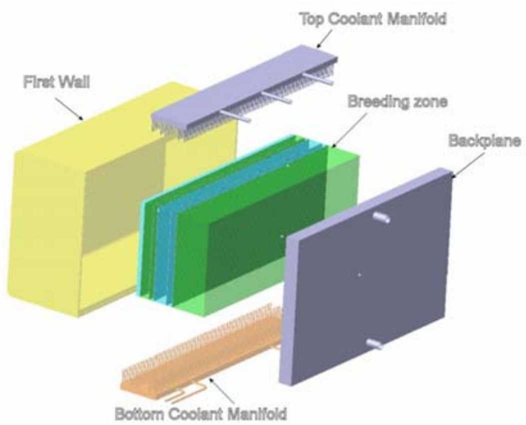


Fig. 4.5.2 The KO HCSB TBM concept

Figure 4.5.2 shows the schematic concept of the KO HCSB TBM. As a breeder, Li_4SiO_4 with a 97 % TD and 62 % packing fraction is used and the ^{6}Li enrichment is 40%. As for the Be multiplier, a 95 % TD and 80 % packing fraction are used. The graphite is also used in a pebble-bed form in order to accommodate any possible geometrical changes during a neutron irradiation. The packing fraction of the graphite is assumed to 85% in this design. The thick graphite reflector has an advantage in that it can play a role of a heat sink in the case of a loss of coolant accident.

Neutronics

For the HCSB TBM, a 3-D Monte Carlo analysis was performed for the evaluation of the neutronic performances by using the MCCARD code. In addition, in order to assess the activation of the TBM, a depletion analysis was also done with the ORIGEN2 code for a 110 day full power operation. In Fig. 4.5.3, the activation of the TBM is shown. Table 4.5.1 shows the major nuclear performance of the TBM. The zone-wise power distribution is used in the thermal-hydraulic analysis which is described in the following sections. The local TBR is quite low since the common frame makes up a significant fraction of the first wall.

Table 4.5.1 Design data and nuclear performance of HCSB TBM

Structural material	Eurofer
Coolant	He gas
Breeder	Li_4SiO_4
Multiplier/Reflector	Be/Graphite
SHF (avg.), MW/ m^2	0.3
Average NWL, MW/ m^2	0.78
FW are, m2	1.49x0.91
Heat deposition, MW	1.17
T production rate, g/FPD	0.253
Local TBR	1.02

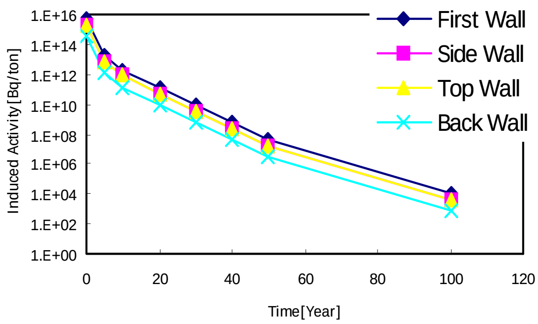


Fig. 4.5.3 Activation of the structural materials of the HCSB TBM

Thermal-Hydraulic Analysis

Thermal-hydraulic analysis was performed in order to calculate the temperature distribution of the first wall and the breeding zone. The HCSB cooling is designed to use the helium coolant coming from the common manifold in the rear wall. The coolant flows through the first wall and then the breeding zone in a radial and poloidal direction at a static pressure of 8 MPa. The inlet and outlet temperatures of the coolant are assumed to be 300 ^oC and 500 ^oC. The coolant flow rate for the HCSB with a thermal power of 1.455 MW and a coolant temperature rise of 200 ^oC is 1.4 kg/s. The neutronic calculation provided the thermal power distributions in the first wall and the breeding zone. The thermal-hydraulic design parameters of the KO HCSB are summarized in Table 4.5.2. The heat transfer coefficient is estimated by using a correlation for a turbulent duct flow.

Table 4.5.2 Design parameters of the KO HCSB

Parameter	Value
Thermal power (MW)	1.455
Surface heat flux from plasma (MW/m^2)	0.3
TBM_HCSB dimension (m) Height Width Depth	 0.91 1.49 0.495
Coolant inlet/outlet temperature (^{O}C) First wall Breeding zone	 300/405 405/500
Coolant mass flow rate (kg/s)	1.40
Cooling system pressure (MPa)	8.0
Heat transfer coefficients (W/m^2K) First wall channel Breeding zone channel	 3950 1250

The temperature distributions of the HCSB were calculated using a computational fluid dynamics (CFD) code, CFX-5, for two computational models. The first CFD model is a two-dimensional model which includes the first wall and the breeding zone. The second one models only the first wall including the beryllium armor in the plasma side and it is used to perform the thermo-mechanical analysis. Figure 4.5.4 shows the first CFD model and the predicted temperature distribution. It is noted that the current CFD model did not include the thickness of the cooling channel in the breeding zone and its support structure. The average surface heat flux of 0.3 MW/m^2 from a plasma is applied to the surface of the beryllium armor. The helium coolant temperatures used in this analysis are 390 ^{o}C and 500 ^{o}C in the first wall channel and the breeding channels, respectively. The peak temperature of 899 ^{o}C was predicted to occur in the third breeder (BR3), which is lower than the limiting value of 900 ^{o}C. However, the temperature of the support structure for the cooling channel in the breeding zone increases up to approximately 610 ^{o}C which is higher than the temperature of the limit of 550 ^{o}C. Therefore, a further optimization of the HCSB design will be made to reduce the peak temperature of the support structure. Future work is also required to evaluate the pressure drop of the coolant in the TBM.

Figure 4.5.5 shows the second CFD model and the calculated temperature distribution for use in thermo-mechanical analysis. The surface heat flux from a plasma is assumed to be a local maximum, i.e., 0.5 MW/m^2. The local maximum temperatures in the beryllium armor and in the first wall were predicted to be 566 ^{o}C and 548 ^{o}C, respectively.

Thermo-Mechanical Analysis

The finite element model for the thermal-mechanical analysis was created using a structure analysis code, ANSYS Version 9.0. The 4-node shell element was used for the 2-D analysis. The nodal temperature was modelled based on the temperature output at all the nodes for the first wall model which was obtained through the thermal-hydraulic analysis. All the boundary layer nodes of each material were merged with a coincident node.

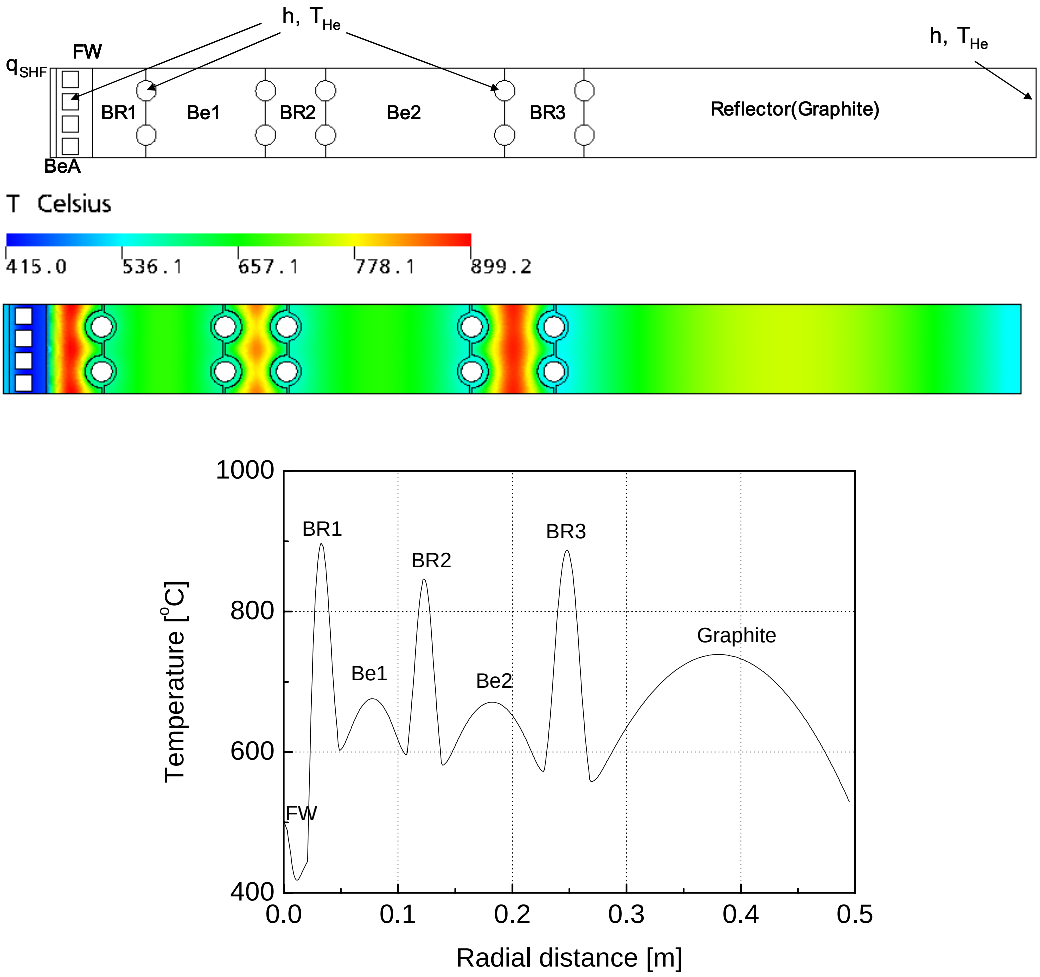


Fig. 4.5.4 Computational model and temperature distribution for the KO HCSB

The applied boundary condition was chosen for the real boundary condition. The middle layer nodes were constrained at the z-directional degree of freedom (dof). The left vertical edge (in y-directional) was constrained in the horizontal (x-directional) dof. The bottom edge was constrained in the y-directional dof. All of these boundary conditions were considered as a thermal expansion between the TBM materials and the common frame since there was some empty gaps between these parts.

As shown in Fig. 4.5.6, the maximum von Mises equivalent stress of the first wall showed at 101 MPa and the maximum deformation of it was 0.305 mm. The maximum stress based on this analysis is much lower than the maximum allowable stress. Of course, this TBM design will be verified using the global 3-D analysis model in the future.

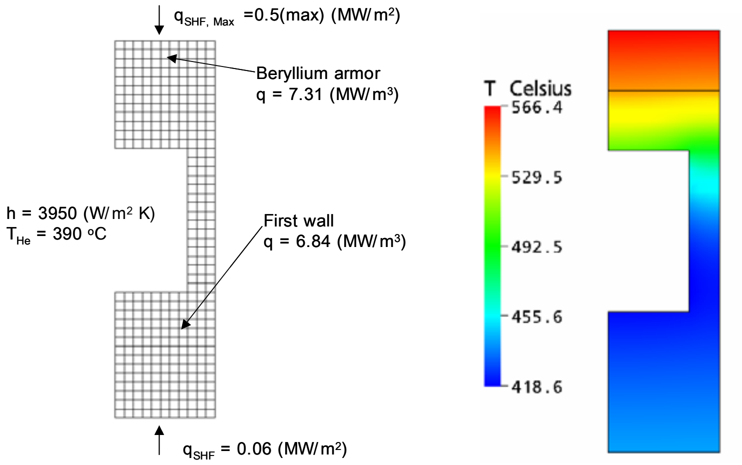


Fig. 4.5.5 Computational model and temperature distribution for the first wall of the KO HCSB

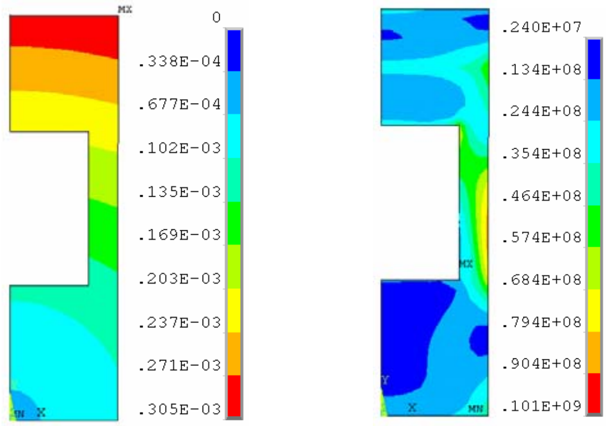


Fig. 4.5.6 Thermal deformation(left) and stress(right) distributions for the first wall of the KO HCSB

4.5.2.2 Helium Cooled Molten Lithium (HCML)

Figure 4.5.7 depicts the design of the KO HCML TBM concept. The whole TBM is cooled only by the He coolant and the molten Li is used as the tritium breeder. It is well known that liquid Li is compatible with the steel up to 550 °C. Due to the low speed of the molten Li, there are no serious MHD and material corrosion issues. With the HCML concept, the heat exchanger design is relatively simple since the liquid Li is not involved in the heat removal.

As in the HCSB concept, a graphite is also used as a reflector in order to minimize the neutron leakage from the TBM. Based on the neutronics analysis, the graphite reflector is placed such that the TBR is maximized: a thick front region and a thin back breeder. In ITER, the Li inventory is limited due to the safety reason, so the amount of Li in the HCML TBM is about 28 liters, which satisfies the Li limit. The ^{6}Li enrichment is 90 wt%, corresponding to an optimal value in terms of the TBR and the Li speed is very slow, less than a few mm/sec.

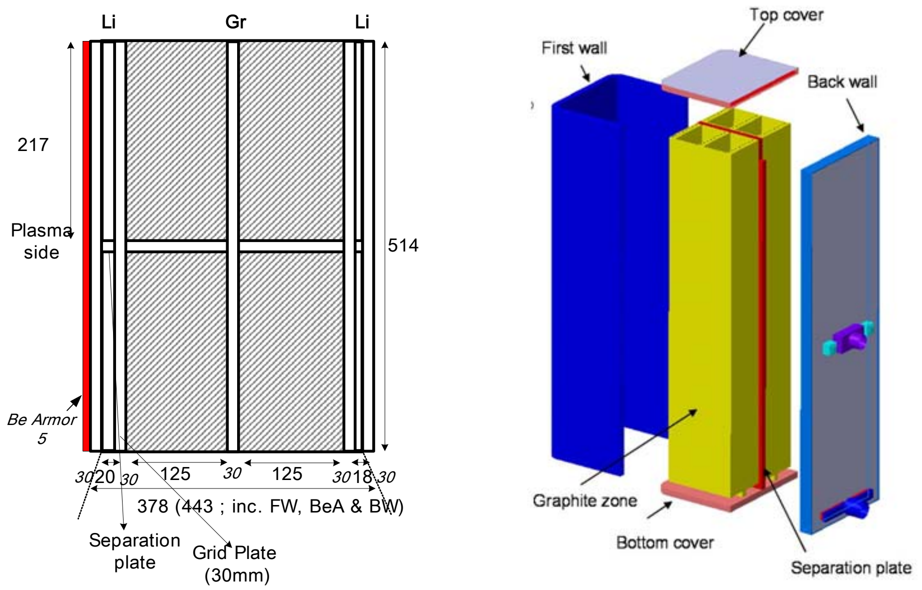


Fig. 4.5.7 The KO HCML TBM concept

Neutronics

For the model in Fig. 4.5.7, a 3-D Monte Carlo analysis was performed with the MCCARD code. The design data and nuclear performance of HCML are shown in Table 4.5.3 and the heat generation rate of each component is shown in the Figure 4.5.8. The total heat deposition is substantially lower than in the HCSB case since the HCML TBM does not contain any neutron multiplier. Also, the TBR is less than unity since the Li inventory is limited in the ITER. In the DEMO-like design, the front Li breeder region could be significantly expanded for a higher value of the TBR. A low TBR does not matter in the TBM because the main purpose of the TBM is to validate the design concepts and the operational feasibility for the DEMO blanket As a result, the tritium production rate is relatively small, too.

Table 4.5.3 Design data and nuclear performance of HCML TBM

Structural material	Eurofer
Coolant	He gas
Reflector	Graphite
SHF (avg.), MW/m^2	0.3
NWL, MW/ m^2	0.78
FW are, m2	0.514x1.72
Heat deposition, MW	0.793
T production rate, g/FPD	0.038
Local TBR	0.45

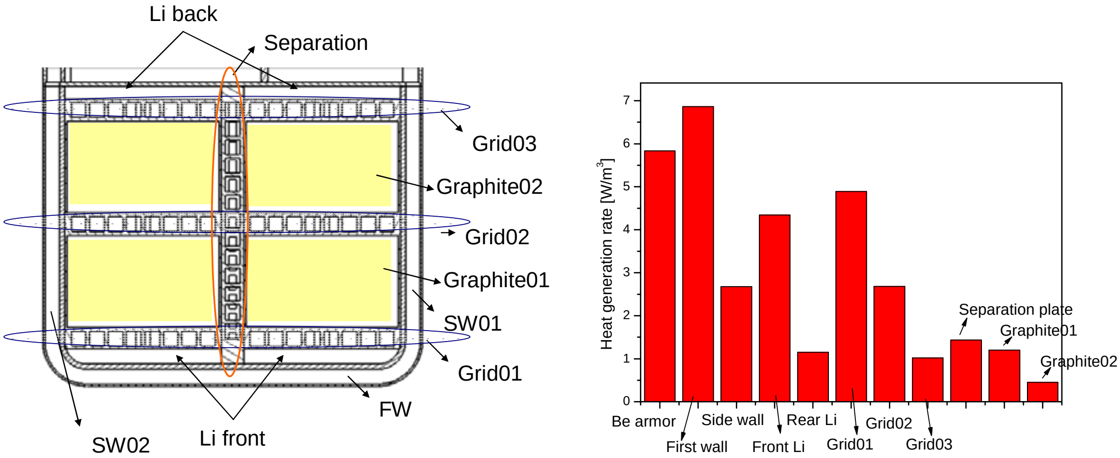


Fig. 4.5.8 Heat Generation rate for each component of the KO HCML TBM

Thermal-Hydraulic Analysis

Thermal-hydraulic analysis was also performed in order to calculate the temperature distribution of the first wall and the breeding zone using the CFD code, CFX-5.7. Figure 4.5.9 shows the He flow paths in the entire TBM and first wall. The helium coolant flows through the first wall and then the half amount of inlet He flows the breeding zone in a poloidal direction at a static pressure of 8 MPa. When the inlet temperature is assumed to be 300 ^{o}C, the He temperatures is 338.1 ^{o}C at the first wall exit. The coolant flow rate with a thermal power of 0.77 MW and the coolant velocity of 45 m/sec is 2.41 kg/s. The thermal-hydraulic design parameters are summarized in Table 4.5.4.

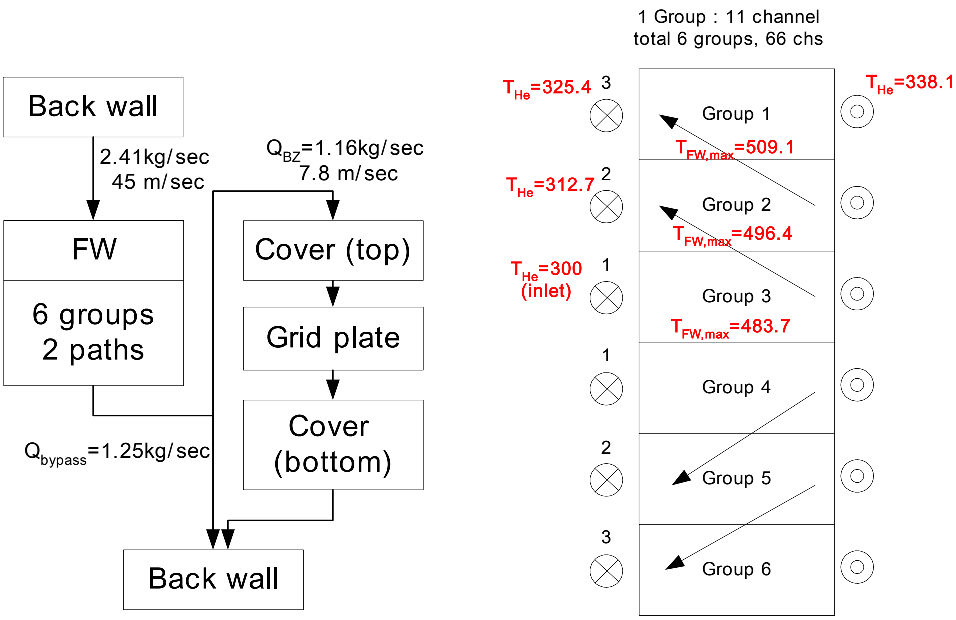


Fig. 4.5.9 He flow scheme and its temperature at each position

Table 4.5.4 Design parameters of the KO HCML

Parameter	Value
Thermal power (MW)	0.793
Surface heat flux from plasma (MW/m^2)	0.3
TBM_HCML dimension (m) Height/Width/Depth	1.72/0.514/0.443
Coolant inlet/outlet temperature (^oC) First wall Breeding zone	 300/338 338/393
Coolant mass flow rate (kg/s)	2.41
Cooling system pressure (MPa)	8.0

The temperature distributions of the HCML were calculated using the 3D model of first wall and breeding zone separately. The average surface heat flux of 0.3 MW/m^2 from plasma is applied to the surface of the beryllium armor. The helium coolant temperatures used in this analysis are 325.4 ^oC and 338.1 ^oC in the first wall channel and the breeding channels, respectively. Figures 4.5.10 and 4.5.11 show the calculated temperature distributions at each region. The peak temperatures were calculated to be 508.7 ^oC in the first wall contacting with Be armor and 700 ^oC in the front graphite reflector of breeding zone, respectively. The pressure drop of the coolant in the HCML will be evaluated in the future.

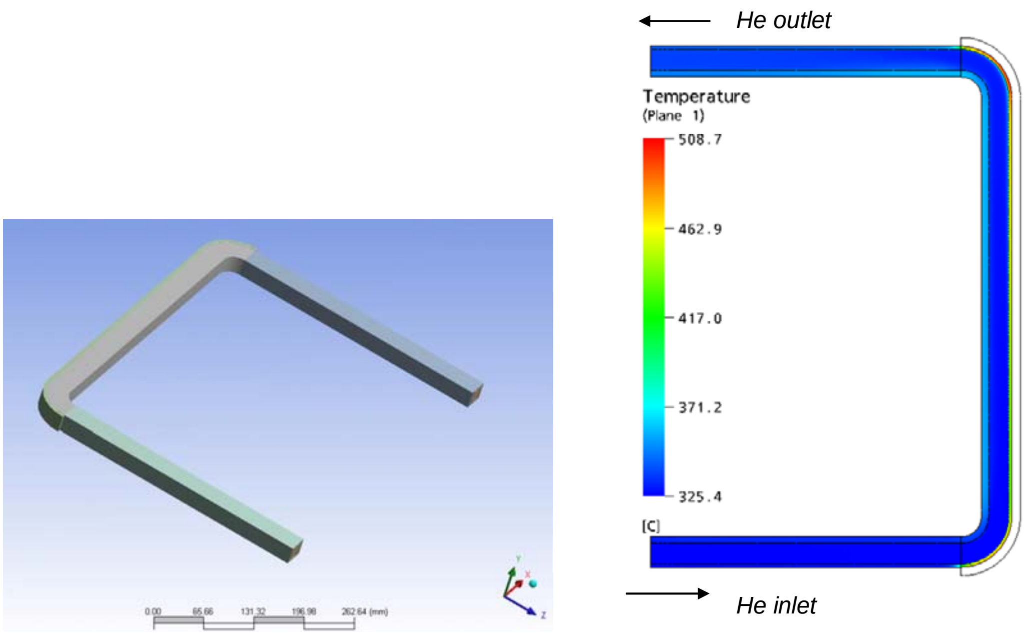


Fig. 4.5.10 Computational model and temperature distribution for the first wall of the KO HCML

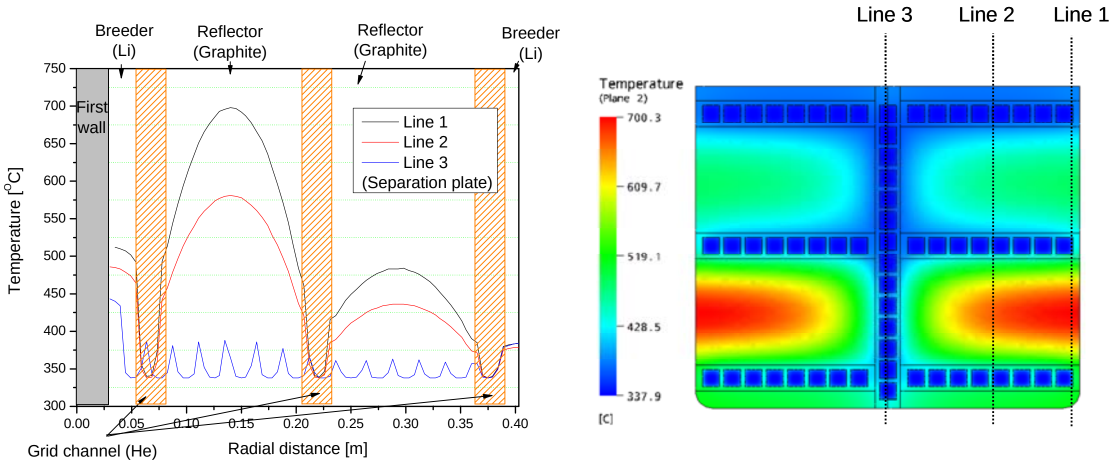


Fig. 4.5.11 Calculated temperature distribution for the KO HCML

Thermo-Mechanical Analysis

Using the CFD model of the first wall for the HCML, a finite element model for the thermal analysis was created by ANSYS Version 9.0. The boundary conditions are determined from the results of CFX-5.7 analysis. Figure 4.5.12 shows the calculated thermal deformation and stress distributions. The maximum von Mises equivalent stress of the first wall is 123 MPa which is lower than the maximum allowable stress, and the maximum deformation of it was 3.73 mm.

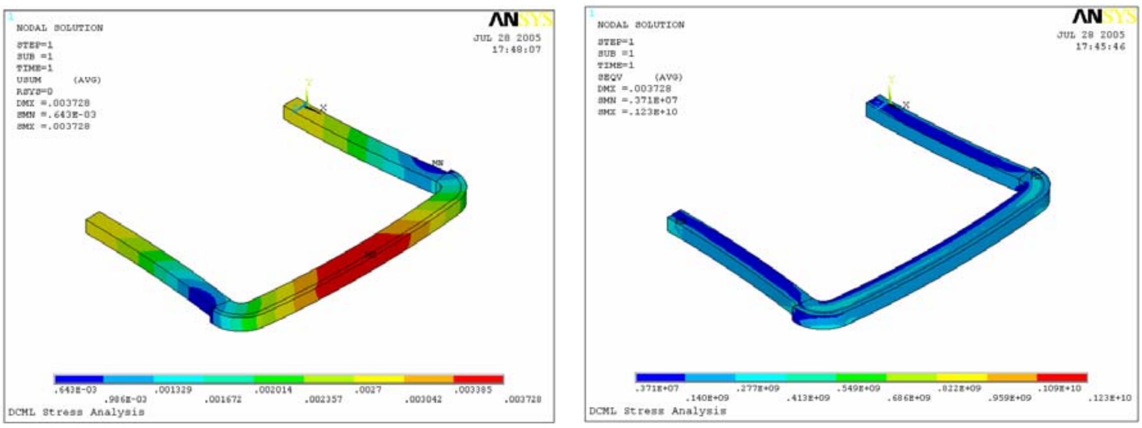


Fig. 4.5.12 Thermal deformation(left) and stress(right) distributions for the first wall of the KO HCML

4.5.3 TBMs systems

4.5.3.1 HCSB Systems

The proposed KO sub-module and TBM for the helium-cooled solid breeder concept are not to have independent ancillary equipment but rather to have a partial or complete sharing of other parties' helium lines and tritium extraction system. KO plans to collaborate with other parties on the development and installation of helium cooling and tritium extraction systems. This will help to reduce the number of helium and tritium lines and the corresponding equipment in the TWCS vault. However, the KO HCSB sub-module needs some independent systems such as a helium coolant conditioning system, tritium measurement system and a neutron measurement system.

The helium coolant conditioning system is housed in the piping integration cask located behind the bioshield plug. The purpose of this coolant conditioning system is to divide the main coolant into a number of coolant streams and regulate the temperature according to the flow conditions required for the sub-modules.

The tritium measurement system is installed in the port cell area to measure the tritium concentration and compositions in the purge gas stream before the purge gas proceeds to the tritium extraction system. KO wants to share TES with other parties but will have its own tritium measurement system. The tritium measurement system consists of a dryer, hygrometer, ionization chambers, residual gas analyzers and associated Turbo and backing pumps. The measurement system measures the total tritium concentration as well as the tritium concentration of the HT and HTO forms. The concentrations of HT and HTO will be measured at the first ionization chamber; the water is then removed by the dryer and the concentration of HT will be measured again at the second ionization chamber.

The neutron measurement system, designed to perform dedicated measurements of the tritium production and neutron fluxes and spectra, will be installed in the port area.

4.5.3.2 HCML Systems

Study on the HCML TBM has been mainly focused on the TBM itself and most of the required ancillary systems are not designed as yet. The following ancillary systems will be designed and developed for the HCML TBM.

- He coolant system
- Tritium recovery system including the Li maintenance
- Neutron and tritium measurement system

4.5.4 Supporting Activities for other Parties TBMs

4.5.4.1. Supporting Activities to Other Parties' HCSB TBMs

KO is not to have an independent helium-cooling system but can support other parties in designing and fabricating the system.

KO will participate in the international R&D activities and make efforts to contribute to solving key issues in cooperation with other parties. The key remaining issues might involve;

- pebble bed thermal conductivity measurements,
- modeling of the mechanical behavior of a breeder pebble bed,
- high fluence irradiation of ceramic breeder pebble beds,
- design database for solid breeder blankets and test modules, and
- tritium control in solid breeder blankets including the purge gas conditions.

Since these efforts need to be conducted in a collaborative nature, KO will support other parties' TBM R&D including sharing a database, design and analysis, and participation in joint development.

4.5.4.2 Supporting Activities to Other Parties' HCML TBMs

The HCML TBM concept is quite unique among the proposed TBM concepts from other parties in that molten Li is used as the tritium breeder. Meanwhile, it shares some important technologies with the self-cooled Li/V concept from RF, e.g., tritium recovery from a Li and Li-loop maintenance etc, which are core technologies in the concept. KO looks forward to an international collaboration for the development of the HCML TBM. If the collaboration is specifically materialized, the HCML concept could be a kind of an international TBM and both the TBM and required ancillary systems might be developed together with the collaboration parties.

4.5.5 Supporting R&D and Validation Program prior to the installation in the ITER

4.5.5.1 Supporting R&D and Validation Program prior to the Installation in ITER for the KO HCSB

Research and development tasks for the solid breeder blanket concept are necessary for all the areas; sub-module and TBM design and analysis, ceramic breeder and beryllium pebble fabrication and characterization, pebble bed thermo-mechanical interaction and compatibility, characterization of tritium release, retention, and permeation, structural material fabrication techniques and thermo-mechanics, and the burnup effects on a ceramic breeder performance. These tasks will be performed internationally and domestically.

KO is considering sub-module tests and TBM tests. The test modules will be qualified and validated following the criteria prior to an ITER installation. Mockup and prototype TBMs should be fabricated and tested before the ITER installation.

Sub-module design and analysis. The sub-module design will be developed over the next few years, based on our TBM design concept and the known database. Our design includes graphite pebbles as a reflector, so our design work will focus on graphite pebbles and the database.

Pebble fabrication. Li-ceramic pebble, Be pebble and graphite pebble will be fabricated on a lab scale over the next few years, and then the fabrication technology will be upgraded on a large scale in cooperation with the industry. International collaboration is anticipated on the fabrication and characterization.

First wall fabrication. The key fabrication technique of the first wall, HIP joining, will be developed step by step. The hipping technique will be tried using a small specimen and then applied to a large size specimen or full size first wall.

Material Development. The R&D on the pebble and structural materials should include material behaviors under a neutron fluence, a database evaluation and a code development. This effort should include incorporating the thermo-physical and mechanical properties of the welds and joints into the state-of-the-art finite element thermal and structural codes to ensure that the performance of the critical areas of the design is adequately addressed. Irradiation effects should also be considered at the later stage of the development.

Pebble bed development. The purpose is to develop a database that can predict the thermo-mechanical behaviors of pebble beds – Li-ceramic, Be, graphite. The thermo-mechanical behaviors include the cyclic effect on the integrity of the pebbles and the dimensional stability at the interface.

4.5.5.2 Supporting R&D and Validation Program prior to the Installation in ITER for the KO HCML (TBD)

Research and development tasks for the HCML are necessary for all the areas similarly to the HCSB; TBM design and analysis, characterization of tritium release, retention, and permeation, and structural material fabrication techniques and thermo-mechanics.

References

[4.5-1] “Nuclear Technology Roadmap”, MOST, KOSEF, KNS, 2005 (in Korean).

[4.5-2] B.S. Han et. Al., “Optimal Configuration of Neutron Reflector in He-Cooled Molted Li Blanket’, 7th International Symposium on Fusion Nuclear Technology, May 22-27, 2005, Tokyo, Japan.

[4.5-3] Y. Kim et al, “Li-6 Enrichment Zoning in Solid Breeder Blanket”, 7th International Symposium on Fusion Nuclear Technology, May 22-27, 2005, Tokyo, Japan

4.7. UNITED STATES PROPOSALS

4.7.1. DEMO Studies and Test Strategy

With the U.S. rejoining ITER and in light of the new R&D results from the U.S. and world programs over the past decade, a study was initiated [4.7-1] to select two blanket options for the U.S. ITER Test Blanket Module (TBM). For consistency in blanket concept comparison, a set of DEMO-like parameters was selected with a maximum neutron wall loading of 3 MW/m^2 and a maximum first wall (FW) surface heat flux of 0.5 MW/m^2. Three key conclusions were reached early in the study: (1) Selection between solid and liquid breeders cannot be made prior to fusion testing in ITER and the U.S. should be engaged in testing both options at some level. (2) While some liquid breeder options do have potential for higher performance, they also have serious feasibility issues requiring more assessments. (3) In general, solid breeder is proposed by all parties, yet performance issues remain. During 2004, the U.S. Plasma Chamber Community team focused on the assessment of critical issues of liquid breeder concepts with strong participation from the U.S. materials, safety and plasma facing component programs. Examples of issues that were evaluated included magnetohydrodynamic (MHD) insulators, MHD effect on heat transfer, tritium permeation, corrosion, SiC_f/SiC flow channel insert (FCI) viability and compatibility [4.7-1 and 4.7-2].

As a liquid blanket option, the dual-coolant liquid breeder (DCLL) blanket concept, with the potential for self-cooled design, was selected. The design and development is underway for a liquid breeder TBM series with the flexibility to test a helium-cooled reduced activation ferritic steel (FS) FW and structure design with self-cooled Pb-17Li breeder zone that uses SiC_f/SiC composite FCI as MHD and thermal insulator [4.7-1]. A detailed description of this work is presented in the DCLL DDD report [4.7-3]. Also evaluated was a helium-cooled FS structure with low melting-point molten salt blanket option, which due to the low electrical and thermal conductivity of the molten salts required no FCI. Since many critical issues remain for the molten salt, the DCLL design was selected in place of the molten salt option which is still a possibility. A detailed description of this work is also presented in the appendix of the DCLL DDD report [4.7-3].

For a solid breeder blanket option, a helium-cooled solid breeder concept with FS structure and beryllium neutron multiplier was selected. A detailed description of this work is presented in the helium-cooled solid breeder DDD report [4.7-4].To focus the effort, the U.S. will not propose an independent TBM for this option. Instead the strategy is to support EU, Japan and China using their TBM designs, structure and ancillary equipment, where the U.S. will contribute unit cell and sub-module test articles that focus on particular technical issues that can utilize unique U.S. expertise that will be of interest to all parties. These test blanket unit cells will be designed and inserted into the helium-cooled ceramic breeder test port. The decision to test the unit cell or sub-module test articles will be made in a few years depending on the international test program and the U.S. budgetary situation.

4.7.1.1. Liquid Breeder Blanket Options. The still evolving U.S. 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 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 U.S. 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 and 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 resulted in the most productive and cost effective strategy for developing scientific understanding and technological systems needed for Pb-17Li breeding blankets.

The current U.S. strategy for ITER testing is to progress from basic structural, hydraulic and MHD performance to more integrated test modules in concert 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 provide 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 is 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 flowrates 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 ferritic steel 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 DT 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_f/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 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.

4.7.1.2. Solid Breeder Blanket Options. In this particular concept, the unit cell approach incorporates consistent interface conditions that the host party (in this case EU) 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 19.5×21.1 cm^2. 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 DT-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. A submodule TBM would take up a testing space of a quarter port 73×91 cm^2 and has its own structural box.

These test blanket units will be designed and inserted into the helium-cooled ceramic breeder test port in concert with the ITER operation. Three sequential phases can be envisioned: (1) FW structural thermomechanics and transient electro-magnetic (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 DT-phase, and (3) tritium breeding, release and thermomechanics explorations (TM) tests during the DT-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. Collected data can then be used to optimize the configuration aspect of the solid breeder blanket design.

4.7.2. TBM Design and Analyses

4.7.2.1. Dual Coolant Liquid Breeder Option. In support of the ITER TBM program, the U.S. has been focusing on the dual coolant Pb-17Li liquid breeder (DCLL) blanket design, a concept that has been explored extensively in the U.S. [4.7-1, 4.7-2, 4.7-3, 4.7-5] and by the European Union [4.7-6]. A detailed description of this design is presented in the US-DCLL DDD report. Reduced Activation Ferritic Steel (FS) is selected as the structural material, and helium is selected as the first wall and blanket structure coolant. We propose to test the concept in one half of a designated test port as shown in Fig. 4.7.2.1-1. It is mounted inside a water-cooled frame designed to hold two different test modules. The front surface of the module is 64.5 cm wide and 194 cm high. The total radial depth of the TBM is 41.3 cm followed by a 30 cm thick inlet/outlet piping zone. A separate 316SS/H_2O shield plug is used behind the TBM. A 2 mm-thick beryllium layer is utilized as a plasma facing component (PFC) material on the FS first wall.

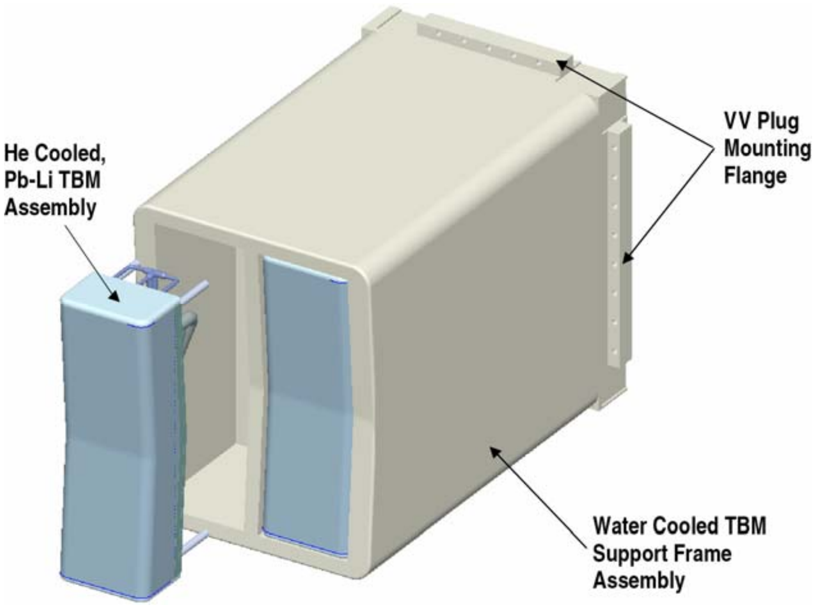


Fig. 4.7.2.1-1. DCLL TBM assembly installed in one of the half ports.

A drawing of the DCLL sub-assemblies is shown in Fig. 4.7.2.1-2. The sub-assemblies will form the box structure of the TBM. The support key as shown in Fig. 4.7.2.1-2 will be inserted into matching slot in the shielding block located behind the TBM, and the four positioning pins will be inserted into the shielding block and used to set the TBM into the proper position and provide the radial support needed at the top and bottom during operation. The TBM was designed to accommodate the two fluid flows internally and maintain the total separation between them. Also it is designed to withstand the maximum He pressure in case of an internal leak from the He into the Pb-Li chambers.

Helium is used to cool the first wall and all FS structure, and the self-cooled breeder Pb-17 is circulated slowly in the poloidal direction as shown in Fig. 4.7.2.1-3 for the integrated module design (I TBM), which has been the focus of design work so far. This module is to be used during the last High Duty Cycle D-T phase during the first 10 years of ITER operation. For the DCLL design, this blanket concept has the potential of satisfying the design limits of FS while allowing the feasibility of having a high Pb-17Li outlet temperature of 650 to 700^{o}C, which can lead to high thermal efficiency for fusion power reactor design.

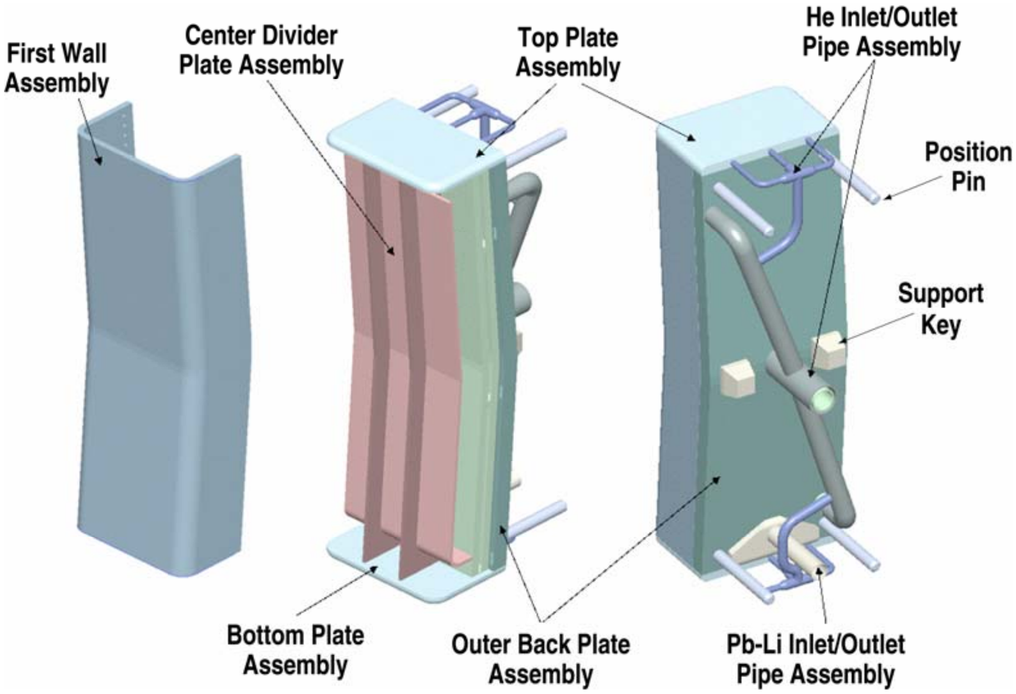


Fig. 4.7.2.1-2. DCLL TBM sub-assemblies.

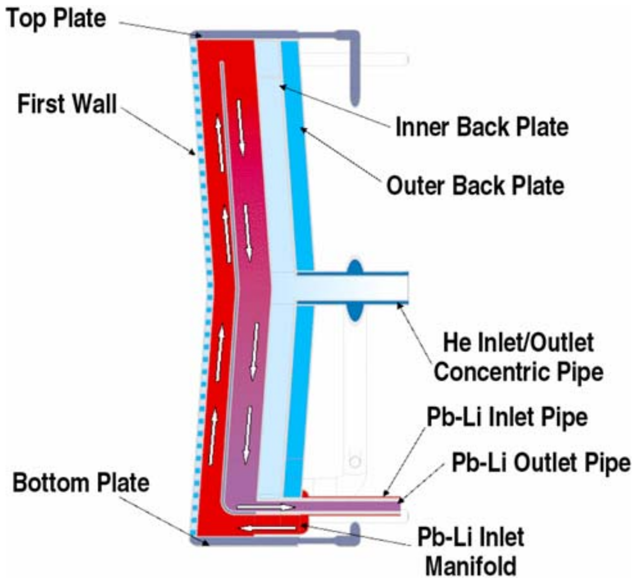


Fig. 4.7.2.1-3. 2-D schematic of the Pb-17Li circuit in the TBM.

Neutronics. Neutronics calculations were performed to determine the relevant nuclear performance parameters for the DCLL TBM. These include tritium breeding, nuclear heating, radiation damage, and shielding requirements. The neutron wall loading at the TBM is 0.78 MW/m^2. The Pb-17Li is enriched to 90% Li-6. SiC FCIs are used (5 mm-thick) at the inside walls of all Pb-17Li flow channels. The calculated local tritium breeding ratio in the DCLL TBM is 0.741 because of the relatively small radial thickness, which is limited by the amount of Pb-17Li that we can use due to safety reason. During a D-T pulse with 500 MW fusion power, tritium is produced in the DCLL TBM at the rate of 3.2x10^{17} atoms/s. Nuclear heating profiles in the different blanket constituent materials were determined for use in the thermal hydraulic and structural analysis. The total nuclear heating in the TBM is 0.982 MW. The radioactivity inventory and afterheat in the TBM were assessed at shutdown and at several post-irradiation times. At shutdown, they are as low as 2.4 MCi and 0.02 MW, respectively. The waste disposal rating (WDR) of the F82H structure, the Pb-17Li breeder and

SiC insert were found to be 1.3×10^{-2}, 9×10^{-3}, and 2×10^{-4}, respectively. These values are far below unity and thus the impact on safety and waste disposal is minimum and well within ITER regulatory guidelines.

MHD Analysis. One unique design feature of the DCLL design is the slow moving Pb-17Li flowing in the poloidal direction as shown in Fig. 4.7.2.1-3. An FCI made of silicon carbide composite (SiC_f/SiC) is used to perform the functions of MHD and thermal insulation. To address the MHD pressure drop and flow features in the poloidal channels with FCI, numerical computations were performed. The liquid moves slowly (about 10 cm/s depending on the desired outlet temperature for a given experimental series) upward through the front channels, and then downward through the return (back) channels. The cross-sectional dimensions of the front and back channels are slightly different. The mathematical model used assumes fully developed, laminar flow. The flows in the channels are considered separately, without taking into account possible electromagnetic coupling between them. However, the effect of electromagnetic coupling between the bulk flow and the flow in the thin gap of the same channel is considered. The computer code solves the governing equations in the domain that includes the bulk flow region, FCI, gap flow region, and the ferritic wall. Parameters for high outlet temperature tests are the following: Pb-17Li inlet temperature is 400°C; Pb-17Li outlet temperature is 650°C. The electrical conductivity of SiC-composite is 20 S/m, and the FCI thickness is 5 mm. The magnetic field (toroidal) is 4 T. The MHD pressure drops in other blanket elements were calculated using other MHD codes, empirical correlations and analytical solutions for flows where 3-D effects due to axial currents are dominant. All calculated MHD pressure drops are summarized in Table 4.7.2.1-1. Table 4.7.2.1-1 also shows the pressure drop for each component as a fraction of the total MHD pressure drop. One can see that the MHD pressure losses in the poloidal channels are much smaller than the 3-D MHD pressure drops associated with the manifolds and the fringing magnetic field. Uncertainties on the manifolds and the fringing magnetic field pressure drop calculation could be increased by a factor of two. A scheme to enforce flow balancing in parallel poloidal channels will likely be needed.

Table 4.7.2.1-1
MHD Pressure Drops in the DCLL TBM

Flow	ΔP_i (MPa)	$\Delta P_i/\Delta P$ (%)
Front channels	0.384×10^{-3}	0.13
Return channels	0.485×10^{-3}	0.16
Concentric pipe (internal, uniform B-field)	15.4×10^{-3}	5.1
Concentric pipe (annulus, uniform B-field)	0.0286	9.5
Concentric pipe (internal, fringing B-field)	0.0585	19.3
Concentric pipe (annulus, fringing B-field)	0.0585	19.3
Inlet manifold	0.070	23.2
Outlet manifold	0.070	23.3
Total	0.302	100

Helium Thermalhydraulics. For the helium flow circuit computational fluid dynamic calculations were performed using the commercial software FLUENT [4.7-7], were performed to determine the helium gas flow distribution in the channels of the first wall. Results of the calculations were used to evaluate and optimize the design of the headers for the FW helium flow circuits. Cooling of the FW is achieved with two counter-flowing helium circuits. Each circuit consists of eight channels making five passes of the FW. Helium transitions from one pass to the next through headers. The configuration of the headers determines the uniformity of channel flow distribution from one pass to the next. This defines the necessary dimension of the back manifold in order to attain flow uniformity.

Total pressure drop in the helium circuit is then estimated as 0.81 MPa, which amounts to 10.1% of the circuit inlet pressure of 8 MPa as shown in Table 4.7.2.1-2. This is acceptable for a testing component and can be much reduced for a power production device like the DEMO.

Table 4.7.2.1-2
Summary of Pressure Drop in the Helium Circuit

Circuit Part	Pressure Drop (MPa)	Fraction of Inlet Pressure (%)
Inlet/outlet pipes	0.32	4
Flow distribution	0.14	1.7
First wall	0.351	4.4
Total	0.81	10.1

The FLUENT code was also used to evaluate the thermal performance of the FW, specifically, the maximum first wall temperature, helium outlet temperature and the heat transfer coefficients in the channels. Results show that the maximum FW temperature of 523^{o}C occurs locally along the pass in the fifth channel of each circuit.

Structural Analysis. Heat conduction and elastic stress analyses of the DCLL were carried out using the finite element program ANSYS [4.7-8]. The structural material for the DCLL is F82H steel. A 3-dimensional CAD model of a 5-channel section of the DCLL ITER-TBM was used for the analysis. This section is located at the top of the TBM, where the highest FW temperatures are expected because of the Pb-17Li breeder temperature rise inside the TBM. Heat conduction in the poloidal direction was considered by applying the temperature conditions of adjacent coolant channels above and below the section. The heat condition analysis of the 5-channel section results in a slightly higher ΔT across the FW than would be the case if poloidal conduction were ignored. Steady state temperature distributions were calculated. The maximum temperatures are 559^{o}C and 557^{o}C at the FW and the Back Plate, respectively, slightly higher than the design limit of 550^{o}C. The corresponding stress analyses were conducted using 8-node brick elements. The average primary membrane stress was taken as the average across the plasma-facing side of the FW, and the primary bending stress was taken as ~2/3 of the peak stress based on the ITER structural design criteria definition [4.7-9]. Results show that all the primary plus secondary stress limits are satisfied within the 5-channel section of the TBM. High temperature ITER Structural Design Criteria provide conservative but simple rule to prevent progressive deformation (cyclic creep-ratcheting) on the basis of elastic analysis by using either the 3 Sm or the Bree-diagram rule. These rules were applied and results show that the maximum high temperature time-independent ($P_L+P_b<KS_m$) and time-dependent ($P_L+P_b/K_t<S_t$) design rules for both for primary membrane and membrane plus bending stresses are satisfied. Structural analysis of the entire module revealed that the maximum Von Mises stress occurs not at the FW but at the sharp interface corner between the internal support structure and the side of the FW. Using more realistic rounding of such interface corners will reduce these high stress concentrations. Preliminary steady-state structural analysis was performed and results show that with minor design modification, the DCLL design can withstand the loss of coolant accident condition without exceeding the ultimate design limit of F82H. Details are reported in the US DDD.

Tritium Extraction. The U.S. Participant Team is developing an ITER TBM based on a dual-cooled lead-lithium (DCLL) DEMO blanket concept [4.7-10]. A key technology issue regarding the success of this DEMO blanket concept is the extraction of tritium from the Pb-17Li to a level that avoids high tritium inventories in the Pb-17Li-helium heat exchangers and unacceptable permeation rates from the primary and secondary systems into the confinement building during normal operation. Because the Pb-17Li temperature entering these heat exchangers is around 700^{o}C, the materials most likely to work in this environment as heat exchanger tubes are refractory metal alloys. Alloys of niobium and tantalum are presently under consideration. Unfortunately, because the solubility of tritium is high and because the diffusivity of tritium is rapid for these alloys, the partial pressure of tritium above the Pb-17Li breeder entering the heat exchanger must be kept below ~0.05 Pa in order to maintain DEMO heat exchanger inventories below 400 g-T [4.7-10] and permeation rates below the allowed operational release guidelines for DEMO fusion facilities of 1 g-T/a as HTO [4.7-11].

An extraction method that appears promising for the DEMO is the vacuum permeator. This component contains a bank of small diameter, thin wall, niobium alloy tubes through which the entire Pb-17Li primary coolant flows at 5 m/s. At the outside of the tube is a high vacuum region. Because the Pb-17Li is turbulent at this velocity, the mass transport of tritium to the Pb-17Li/tube interface is greatly enhanced over ordinary diffusion. Once at this surface, the tritium readily diffuses through the niobium tube walls and into the vacuum region where it is pumped away in the molecular form to the tritium processing plant. Because there are significant material development issues associated with the proposed permeator for the DEMO, a niobium-tube permeator is not being proposed for the ITER DCLL TBM system. Instead, a prototype permeator made of martensitic steel is being proposed for the DCLL TBM as the primary component of one possible TBM tritium extraction system (TES) design. Results show that at the end of the pulse, the tritium pressure into the permeator is 1.8 Pa and the pressure out of the permeator is 1.36 Pa with an efficiency of ~24%. At the end of the dwell time, these pressures drop to 0.75 Pa and 0.71 Pa, respectively. The area of this martensitic steel permeator surface is about 3.14 m^2. This extraction scheme also matches the Pb-17Li bypass flow scheme described in Section 4.7.3.1. In order to estimate the tritium inventory and permeation rates for the DCLL TBM cooling systems, a TMAP code [4.7-12] model of the entire TBM system was developed. The TBM is simulated in this TMAP model as five enclosure volumes and seven diffusion structures or walls. Three of five enclosures are Pb-17Li volumes that represent the combined volume of the breeder zones of the TBM (~0.28 m^3). These volumes simulate the core TBM flow, the Pb-17Li flow in the core, and the flow in two types of gaps between the SiC inserts and TBM walls. Results show that of the 2.33 g-T produced annually in the DCLL TBM, 79.3% permeates through the permeator, and 10.5% permeates into the VV. Of the remaining 10.2%, most of this tritium (98%) resides in the austenitic steel piping of the helium cooling systems. The tritium permeation rate into the ITER heat removal system through the TBM helium-water heat exchangers is ~0.06 mg-T/a. The annual guideline for TBM releases is the sum of airborne plus water releases. Since the total TBM annual release target is 1 mg-T/a as HTO, then the 0.06 mg-T/a does not appear to be a problem when compared to the release guideline, or when compared to the TBM airborne release. As a backup tritium extraction method, the bubble column extraction method being developed by EU will also be assessed.

Disruption Analysis. The DCLL TBM design was subjected to plasma disruption scenarios using the OPERA code. Two ITER vertical disruption cases were analyzed. In both cases, the plasma moved toward the bottom of the machine during the disruption. In one case, the plasma current decayed exponentially with an 18 ms time constant while in the second case, the plasma current decayed linearly to zero in 40 ms. The disruption model included the vacuum vessel, the mid-plane port nozzle, the TBM frame (10 cm thick), and the TBM module. The electrical resistivity of various parts of the TBM was adjusted to account for the helium cooling channels. The magnetic properties of the ferritic steel in the TBM were not included in the analysis because the material will be saturated by the toroidal field in ITER. The maximum induced current was in the 40 ms disruption case at 20 ms after the start of current decay. The eddy current forces cause torque about the center of the TBM and try to twist the top of the TBM in one direction while twisting the bottom in the opposite direction. The integrated magnitude of the forces is in the range 0.1 to 1 MN. The eddy current forces were used as input for a static finite element analysis to determine the stresses in the TBM structure and to determine the necessary supports within the frame.

Safety. For this work we have performed a safety assessment on the design, which provided guidance to our TBM design and 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 the potential tritium loss from the breeder to the vicinity led us to the use of the helium intermediate heat transport loop. We also recommended that concentric pipes be used to connect the liquid breeder between the TBM and the breeder/helium heat exchanger. To minimize tritium permeation in the FW-coolant loop, aluminum tubes are recommended for the He/water heat exchanger. The helium-coolant inlet and outlet piping should have permeation barrier like alumina coating or Al outside sleeve. Results from the safety assessment for the ancillary equipment for the two FW/blanket concepts show that ITER safety criterion can be met provided that we take care of controlling the amount of breeder used in the system and the reduction of tritium permeation loss from the FW coolant loop and from the liquid breeder loop.

Assessment of the safety impact on ITER of a DCLL TBM concept shows that the anticipated radiological inventories are small in comparison to those produced in the ITER VV due to normal operation of ITER. Possible hydrogen sources were examined in this assessment and the conclusion was drawn that the maximum quantities produced during accident conditions should be less than the ITER limit of 2.5 kg. Pressurization of the ITER VV, TBM test cell and TCWS vault by the helium coolant from the TBM ancillary system does not pose a serious threat to these confinement structures. Passive decay heat removal is assured, resulting in long term TBM temperatures less than 300^{o}C after 5 days. However, at this time these are preliminary conclusions that are based on assumptions about the DCLL TBM. More rigorous analyses will be required once more design details become available for the DCLL TBM concept.

4.7.2.2. Helium-Cooled Ceramic Breeder Blanket Options. Two distinct design approaches [details in references 1, 2, 3, 4 and 13–15 for this section] have been considered to fulfill the testing objectives of the helium-cooled ceramic breeder blanket: (1) design the unit cell/submodule for low temperature operation, or a look-alike approach; and (2) refer to a reactor blanket design and use engineering scaling to reproduce key parameters under ITER wall loading conditions, so that phenomena under investigation can be measured at their reactor-like level. The two approaches result in two different sets of operational parameters, the low temperature scenario being used for neutronics assessment or NT unit cell (Fig. 4.7.2.2-1) and the high temperature scenario or TM unit cell (Fig. 4.7.2.2-2) for thermo-mechanic and tritium release performance evaluation.

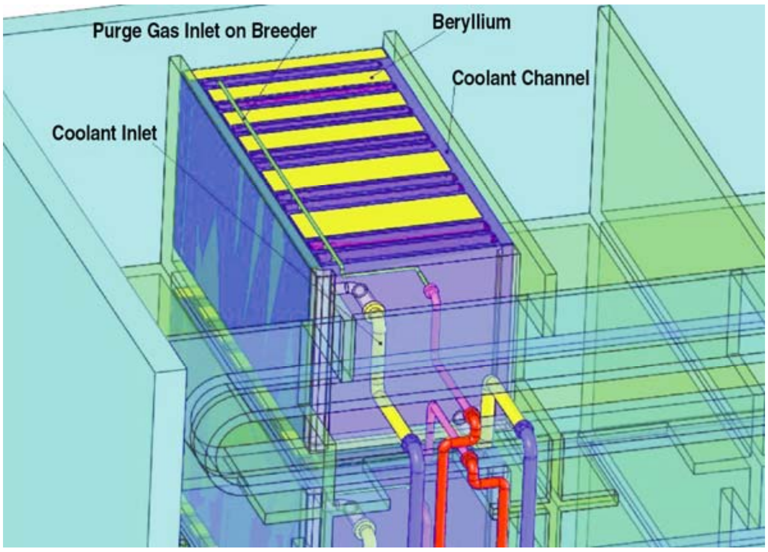


Fig. 4.7.2.2-1. Example EM/S and NT unit cell test article design housed inside the EU's HCPB structural box.

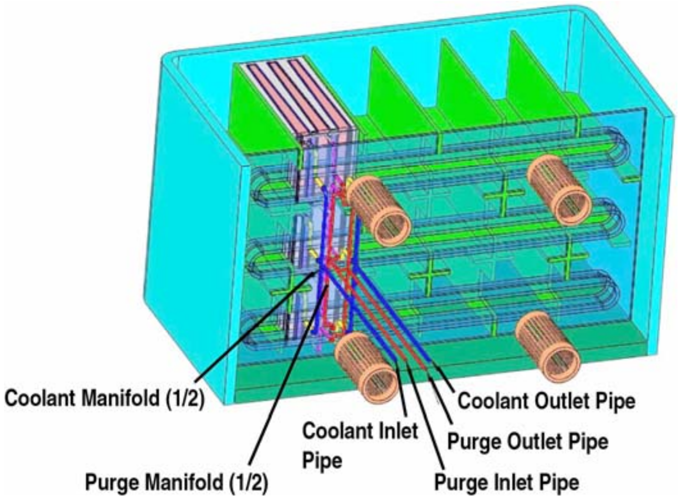


Fig. 4.7.2.2-2. Schematic view of the three TM unit cell test articles housed inside the EU's HCPB structural box.

The unit cell to be considered for the first TBM electro-magnetic tests will have a configuration similar to that of the neutronics unit cell. A stream of coolant from the primary loop will be fed into the unit cell array common manifold located at the back manifold region of the structural box. The coolant is subsequently divided into three paths for cooling three unit cells (Fig. 4.7.2.2-3). The helium coolant entering into the unit cell may be coming directly from the supply line of the helium loop, and thus it may be necessary to raise its temperature from 300^{o}C to 350^{o}C (for TM unit cell) using an external heater located in the port cell area in order to reproduce coolant operating temperatures and replicate prototype breeder temperature levels such that the exit temperature reproduces a typical prototype helium outlet temperature of 500^{o}C. The total heat generated inside a unit cell is about 35.8 kW, which is removed by a coolant flow rate of 0.046 kg/s. This implies that a total flow rate of 0.138 kg/s is needed to cool the proposed U.S. unit cell test array.

An analysis has been performed for a breeder/beryllium unit representing a sub-unit found in the edge-on configuration as in the TM unit cell design using a finite element code MARC. As shown, the

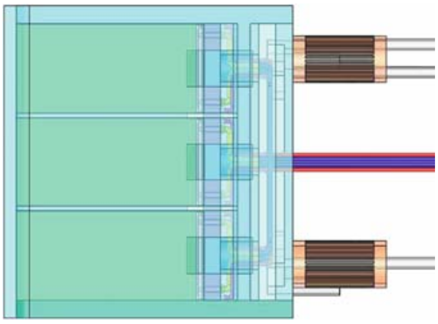


Fig. 4.7.2.2-3. Helium flow distribution scheme for three unit cell test articles.

unit cell is to be inserted into the EU helium-cooled pebble bed (HCPB) structural box to address the issue associated with the pebble bed thermomechanical integrity. The design incorporates features of an edge-on blanket configuration with an attempt to minimize the use of beryllium by increasing the breeder width as it moves toward the back of the blanket region. Specifically, an engineering scaling has been applied to reproduce prototypical ceramic breeder pebble bed thermo-mechanics behavior. Since ITER neutron wall load (0.78 MW/m^2) is much smaller than that of a prototype fusion power reactor (i.e. 3 MW/m^2), the breeder unit dimension is scaled up by a factor of roughly the square root of the ratio of the neutron wall load between the scale and prototype models to correctly duplicate the DEMO blanket temperature profiles. The calculated Von Mises stress profile of this sub-unit, resulting from a combined effect of temperature gradient, differential thermal expansion and structural constraint, shows a maximum stress level of 1.2 MPa located inside the beryllium pebble bed (Fig. 4.7.2.2-4).

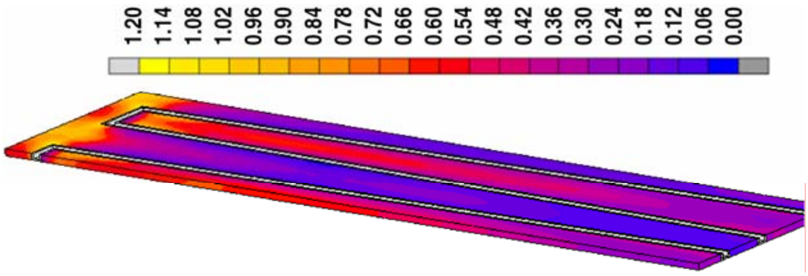


Fig. 4.7.2.2-4. Calculated Von Mises stress profile in a TM unit cell (maximum stress of 1.2 MPa occurs inside the Be bed).

In the submodule design, the breeding zones are housed behind a ferritic steel U-shaped FW structural box, as shown in Fig. 4.7.2.2-5, with dimensions of a toroidal width of 73 cm and a poloidal height of 91 cm. These dimensions are based on a frame structure design with a 10 cm width. Two look-alike submodules are planned to be inserted into ITER at year 1 and year 5, respectively. Since the EM forces depend upon the specific design, the structural features including structural support and fabrication techniques of the EM/S submodule should be similar to that of the later NT and TM submodules in order to validate the structure's ability to withstand off-normal tests, including thermal impulse and EM forces from disruption, and also to measure the effect of the FS on perturbation of the local magnetic fields. It is possible that the breeding materials such as beryllium will not be used in the EM/S if safety is more of a concern due to the uncertainties involved in the plasma operations. The EM/S should, however, have electrical characteristics similar to those of more integrated TBMs as well, so that the induced eddy current and its distributions are simulated.

The helium coolant is routed toroidally through the first and side walls in alternating directions. The overall FW thickness is 28 mm, including a coolant channel of 16×14 mm and a front wall thickness of 5 mm (Fig. 4.7.2.2-5). The pitch between the coolant channels is 18 mm. The FW is designed to remove a total deposited heat of 0.307 MW, based on the contribution of the average surface heat flux of 0.3 MW/m^2 and nuclear heating deposition on the front and side walls of the FW structures with a neutron wall load of 0.78 MW/m^2. Because a relatively high velocity is needed to ensure an adequately high heat transfer coefficient for removing a maximum surface heat load of 0.5 MW/m^2 locally, the FW design features a reduced coolant flow area by grouping five coolant flow channels in a series into one coolant flow path. In the low temperature scenario the 8 MPa helium coolant enters the submodule at a rate of 0.755 kg/s and a temperature of 100°C and is subsequently distributed into 2 paths to remove the 0.45 MW heat generated in the breeder region. In the low temperature operation design, the helium flows first in the breeder region channels and then in the first wall since the goal is to keep the breeding material temperature low. In the high temperature design the scheme is reversed, since the main challenge becomes the cooling of the first wall structure. The 8 MPa helium coolant enters the submodule at a rate of 0.9 kg/s at a temperature of 300°C and is subsequently distributed into 10 first wall cooling paths for surface heat removal. About 10% of this flow is by-passed away from the breeding zones to achieve a typical outlet temperature of 500°C. The remaining coolant in the submodule is divided into four paths for cooling upper and lower caps and two breeding configurations. 2-D transient thermal analysis has been performed to study temperature characteristics under an ITER 400 s burn cycle. As shown in Fig. 4.7.2.2-6, temperatures near the front, less than 5 cm behind the first wall, have reached equilibrium values, while temperatures near the back are about to reach equilibrium values.

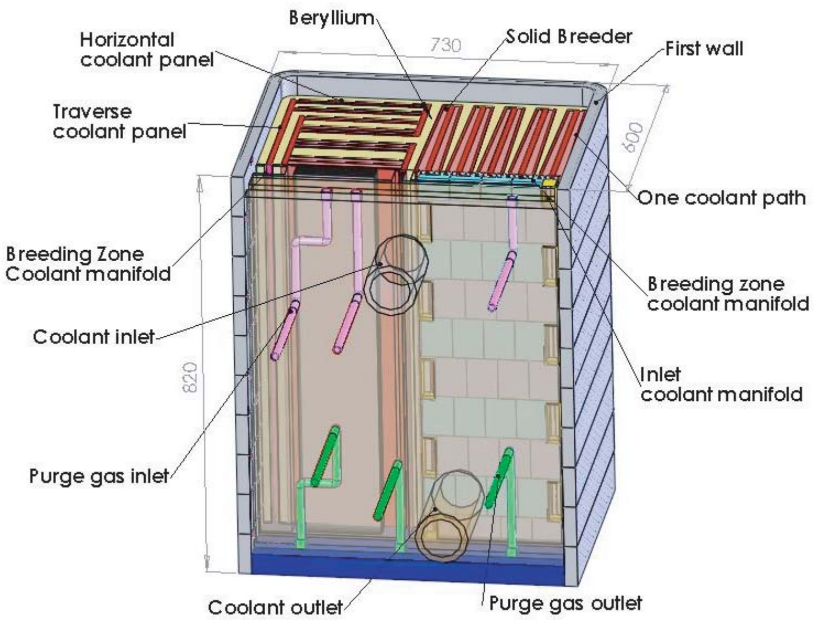


Fig. 4.7.2.2-5. Example design configuration for EM/S and NT submodules.

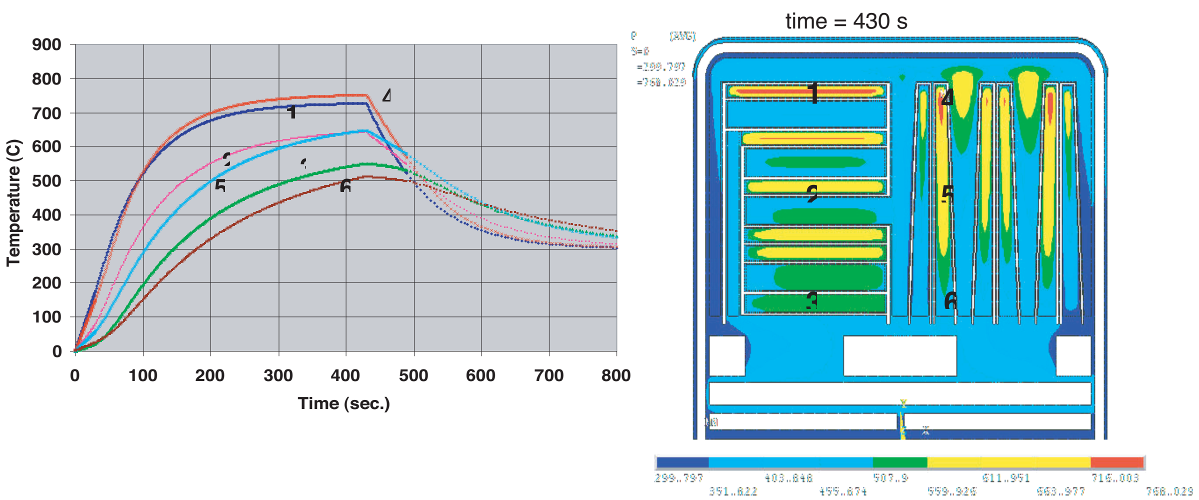


Fig. 4.7.2.2-6. 2-D temperature profile at the end of a burn (right); temperature histories during a burn cycle at several breeder locations (left).

Besides the 8 MPa high pressure helium coolant, a 0.1 MPa helium purge gas is used to purge tritium out of the breeding zones. Three (or two) purge gas streams enter the submodule through the pipes connected at the back manifold region. Each purge gas is then directed to the upper end cap purge gas manifold and is sub-distributed into different breeding units including beryllium pebble beds. The purge gas passes through the packed bed region, collected at the bottom end cap manifold and is then directed into the purge gas outlet pipe.

There are three outlet purge gas pipes: one carries all the tritium generated in the layered configuration breeding zones, one carries all the tritium generated in the edge-on configuration breeding zones, and the third carries all the tritium generated in the beryllium zones. The purge gas flow scheme and associated manifold are illustrated in Fig. 4.7.2.2-7.

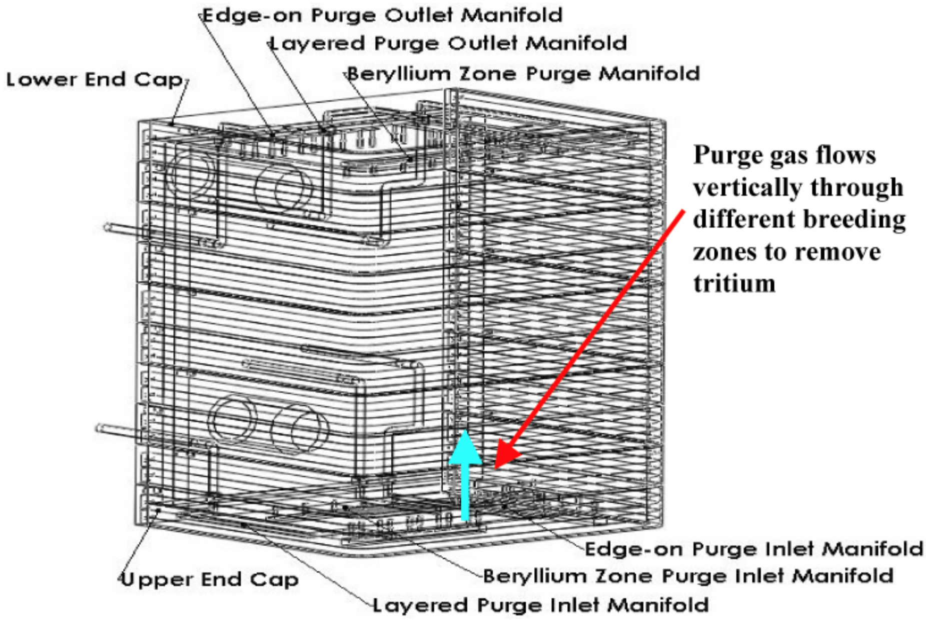


Fig. 4.7.2.2-7. Purge gas is fed into breeding zones through purge gas nozzles located at the upper end cap.

4.7.3. TBMs SYSTEMS (ANCILLARY CIRCUITS)

4.7.3.1. DCLL System. For the DCLL design, there are two coolant systems. The first one is the helium-cooled system removing the surface heat and nuclear power from the first wall and blanket structure. The second is the self-cooled liquid breeder system removing the nuclear power from the breeder. Detailed descriptions of the coolant loop systems, ancillary equipment necessary to support the DCLL and corresponding systems and equipment at the TCWS vault are presented in the DCLL DDD report. Design input parameters for the ancillary systems design are given in Table 4.7.3.1-1.

First Wall Helium Loop. The first wall and structure helium loop will be a self-contained loop including heat transport, tritium extraction, helium purification, and heat exchange to the TCWS plant cooling water. This system is designed to extract about 54% of the total power generated in the one-half module during normal operation in the desired temperature window.

Intermediate Loop Between Liquid Breeder and Water System. To avoid long liquid breeder pipes running from the TBMs to the heat exchanger in the TCWS, we decided to utilize a helium coolant intermediate loop. A liquid breeder-to-helium heat exchanger is located close to the test module to minimize the amount of liquid breeder and the corresponding loss of tritium to the surroundings. The liquid breeder transport loop is designed to extract 100% of the total power generated in the one-half module. This allows the possibility for the testing of a complete self-cooled liquid breeder design option. To avoid the use of advanced materials for the handling of high temperature Pb-17Li, a bypass loop system is proposed. Hot Pb-17Li returning from the TBM is mixed with the bypassed cold Pb-17Li at the bypass section, resulting in only a warm stream going to the tritium extraction and heat exchanger systems. In this way, the high temperature features of the TBM, especially the function of the SiC_f/SiC FCI as a thermal insulator at high temperature, can be tested in ITER without requiring high temperature materials in the tritium extraction and heat exchanger systems.

Table 4.7.3.1-1
DCLL Design Ancillary Equipment Input Parameters

	He	Pb-17Li
Average neutron wall loading, MW/m^2	0.78	
Average surface heat flux, MW/m^2	0.3	
Blanket M	1.01	
Thermal power, MW	0.73	1.36
Fraction of blanket power, %	54	100[a]
T_{in}/T_{out}, ^oC	380/460	340/440
Coolant pressure, MPa	8	2
Mass flow rate, kg/s	1.76	72
Volume flow rate, m^3/s	0.343	$7.75x10^{-3}$
Input max. flow speed, m/s	100	2
Tritium breeding ratio	0.741	

[a]This allows the possibility of testing a complete liquid metal self-cooled blanket option

General Space Allocations. The equatorial ports allocated for the test blanket modules have limited space and limited access through the port plugs. Space is also limited outside the port area to accommodate all the ancillary equipment needed to service and run the TBM. This space must be shared between the parties occupying the port. Furthermore, ITER safety, remote handling and remote operation requirements must be met when designing the TBM module and all of its support equipment. Figure 4.7.3.1-1 shows the standard test port arrangement where the TBM is connected to the transfer cask situated outside the Bio-Shield port opening. The TBM module is mounted inside the shielding frame, which is water-cooled, providing the support of the TBM and proper shielding behind the TBM as required by ITER. The TBM module along with the shielding frame and the VV port plug forms one complete assembly called the VV port plug. A special transporter will be utilized to carry and install the port assembly into the VV plug. Once the installation process is completed, the transporter will be moved out, making room for the transfer cask, which houses the supporting equipment for the TBM as shown in Figs. 4.7.3.1-1 and 4.7.3.1-2.

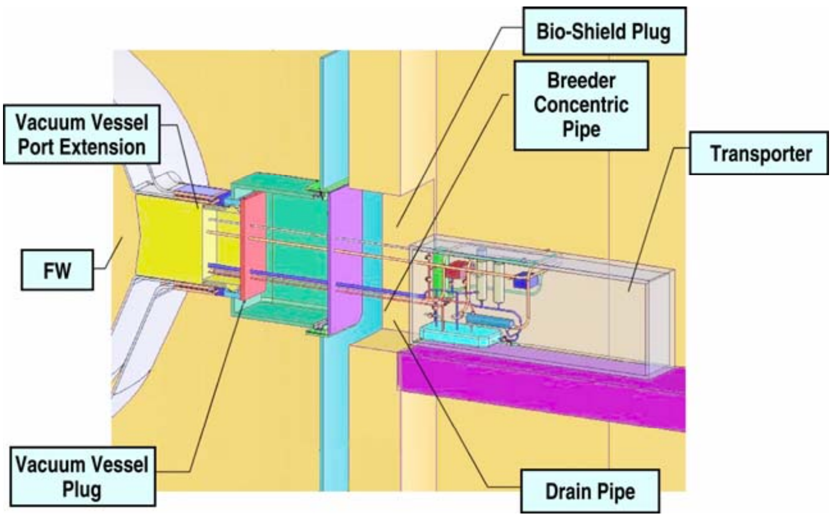


Fig. 4.7.3.1-1. Test port general arrangement.

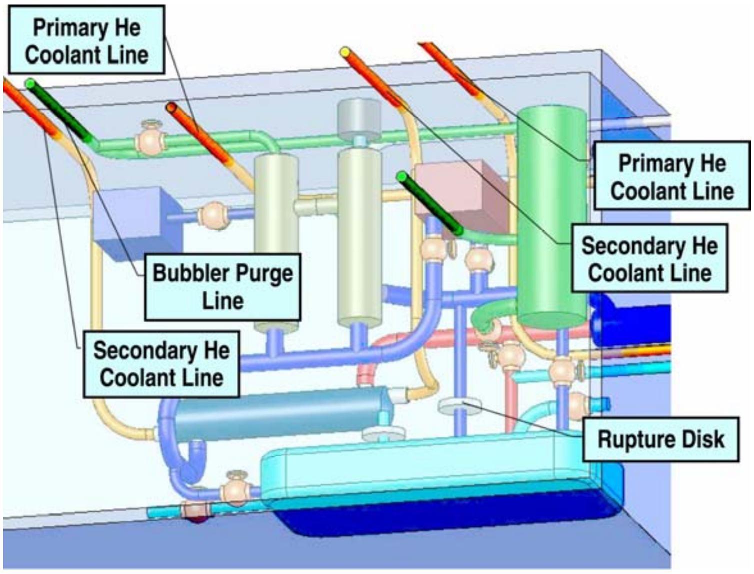


Fig. 4.7.3.1-2. DCLL TBM equipment in the transporter.

The TBM is connected to the transporter through a series of pipes providing all the needed service mainly for cooling and diagnostics. These pipes must penetrate two barriers as they are routed between the transporter and the TBM. The U.S. TBM design relies on the two coolant loops and purge lines to provide all the operational services needed. There are six pipes running between the transporter and the TBM. One concentric pipe carries the Pb-17Li breeder, with the hot liquid running in the inner pipe. The breeder energy is then transferred to two helium inlet and outlet pipes via the Pb-17Li/He heat exchanger. Therefore two He coolant lines are used for cooling the first-wall/structure and the breeder, a drain pipe is used for draining the liquid from the TBM and the VV plug assembly prior to removal or in case of emergency, and finally a gaseous product purge line is used at the top of the TBM module as shown in Fig. 4.7.3.1-2.

TCWS Building Layout and Space. A space of 16.6 × 7.3 m with a clear height of 5 m is assigned in the south end of the TCWS building for all TBM cooling system. This space is shared with all the parties to house the corresponding cooling systems. The DCLL TBM design requires primary and secondary He coolant loops that are located in this area. Necessary equipment including heat exchangers, helium heating unit, pressure control sub-systems, tritium extraction sub-system and various flow meters were specified, as shown in Fig. 4.7.3.1-3. The two helium loops will share some equipment such as the pressure control system and the tritium processing system. The coolant lines will be connected directly to He-Water heat exchangers. The water pipes from the heat exchanger will be routed to the ITER heat removal system. He circulators located in the TCWS area will circulate the He back into the test module assembly. Figure 4.7.3.1-3 shows the general arrangement of the cooling equipment located in the TCWS area. The tritium processing and extraction system for both primary and secondary coolant loops are also located in the TCWS area. Based on the current design, the total area requirements for the FW/structure and breeder helium cooling systems in the TCWS building should not exceed $20 m^2$ for the U.S. DCLL TBM design. This requirement may be revised based on additional design details and changes. The layout currently occupies an area 6.7 m long by 3 m wide.

Services running from the TCWS area to the general ITER plant include the water coolant for the primary and secondary loops and the He purification bypass lines. The final pipe runs for these lines will be determined later based on more design details.

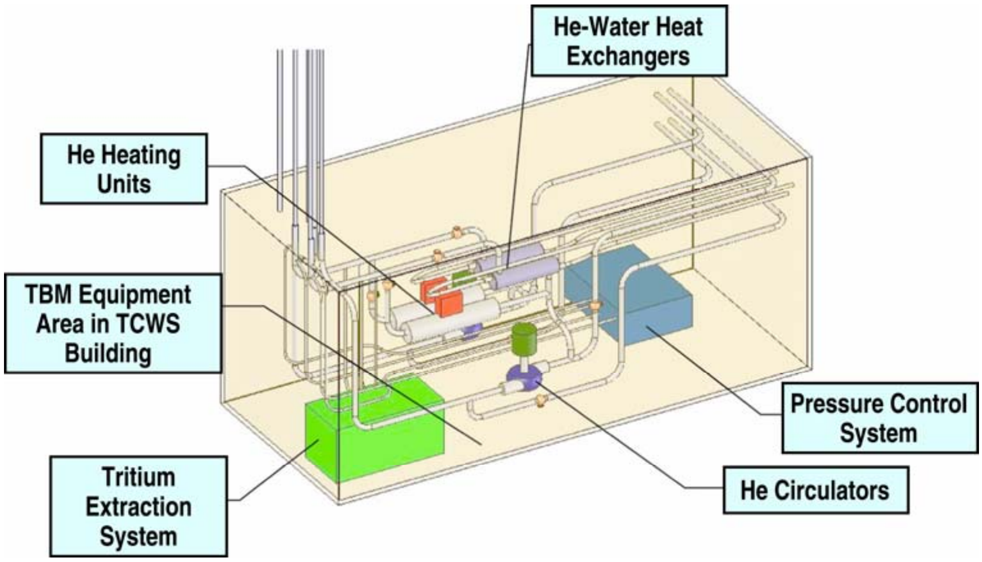


Fig. 4.7.3.1-3. U.S. TBM primary and secondary coolant loops in TCWS.

4.7.3.2. HCPB Systems. The proposed ITER TBM for the helium-cooled solid breeder concept with ferritic steel structure option is not to have U.S. independent ancillary equipment but rather to have a partial or complete sharing of other parties' helium lines and auxiliary systems. Detailed descriptions of the helium-cooled solid breeder TBM is given in the U.S. helium-cooled solid breeder DDD report [4.7.1-2]. The sharing of the auxiliary system also includes tritium extraction subsystems. This implies that the U.S. plans to collaborate with EU and JA on the development and installation of helium cooling and tritium extraction systems although details of such collaboration are yet to be defined. Technically, this will help to reduce the number of helium and tritium lines and corresponding equipment in the TWCS vault. However, an effective integrated scheme is needed to ensure that each party's needs are taken into consideration. For example, it is desirable to have independent coolant temperature control as well as tritium measurement from the purge lines of the U.S. test blanket unit cells/sub-module, which calls for helium coolant conditioning components and tritium measurement systems to be installed in the port cell area (or a modular solution). The principal components in the U.S. helium cooled solid breeder test blanket system include:

1. Test Blanket Unit Cell/Sub-module in the TBM and associated auxiliary lines;
2. Helium Coolant Conditioner System;
3. Tritium Measurement System (installed after year 3 of ITER operation);

In addition, a neutronics measurement system, designed to perform dedicated measurements of tritium production and neutron fluxes and spectra, will be installed in the port area.

The helium coolant conditioning system includes valves, a heater, and a mixer. The system is housed in the piping integration cask located behind the bioshield plug. The purpose of this coolant conditioning system is to divide the main coolant into a number of coolant streams and regulate the temperature according to the flow conditions required for the subunits. In addition, a by-pass pipe has been proposed by injecting some excess amount of flow to cope with the uncertainties in the surface thermal loading condition in ITER. This excess amount of flow bypasses the breeder zones and is removed after the first wall cooling. The temperature of the by-pass line can be about 150K lower than that of the outlet. One of the ideas is to mix this by-pass helium flow with the normal outlet flow in the mixer located in the port cell area rather than to run two pipes into the TCWS building. However, this idea may cause flow instability resulting in fatigue loadings to the pipes and pipe junctions. Further investigation on this and other schemes to mix by-pass low temperature stream with high outlet temperature stream is underway.

The tritium measurement system is installed in the port cell area to measure tritium concentration and compositions in the purge gas stream before the purge gas proceeds to the tritium extraction system. The U.S. plans to share with EU and/or JA the tritium extraction system but have its own tritium measurement system. The tritium extraction is achieved with the help of a helium purge gas containing up to 0.1 % vol. H_2; the addition of hydrogen is needed to facilitate the tritium release by isotopic exchange. Removal of tritium and excess hydrogen from the helium carrier gas is performed in the extraction systems installed in the proposed glove box (4m x 1.2m x 5.5m) in the Tritium Plant. The tritium measurement system consists of dryer, hygrometer, ionization chambers, residual gas analyzers and associated Turbo and backing pumps. The measurement system measures total tritium concentration as well as tritium concentration of HT and HTO forms. The concentrations of HT and HTO will be measured at the first ionization chamber; the water is then removed by the dryer and the concentration of HT will be measured again at the second ionization chamber. The tritium measurement system will take up a space of ~ 1×1×1 m^3 at port cell area.

4.7.4. Supporting Activities to Other Parties' TBMs

4.7.4.1. Supporting Activities for the EU-HCLL and Other Lead-Lithium-Based Blanket Systems. The U.S. pre-ITER DCLL TBM R&D strategy seeks to encourage close collaboration with the worldwide effort and interest in Pb-17Li systems. Currently, this effort is mostly concentrated in the EU for the HCLL, but China is beginning work on the Pb-17Li blanket as well. While no agreement has been formalized, a shared, coordinated and mutually beneficial R&D (including instrumentation development) and ITER testing program on the HCLL and DCLL is the most productive and cost effective strategy for developing scientific understanding and technological systems needed for Pb-17Li breeding blankets. The DCLL concept shares several development issues with the EU-HCLL concept:

- Ferritic Steel (FS) thermal physical and irradiated properties, production, joining and fabrication technologies.
- Pb-17Li corrosion with FS (EUROFER and F82H).
- Pb-17Li chemistry control of corrosion and transmutation products.
- Tritium permeation into, and removal from, helium coolant.
- Tritium removal from Pb-17Li.

These are all issues requiring R&D for the HCLL and the DCLL. The U.S. would like to involve people in the research underway in these areas to contribute as much as possible to the R&D efforts.

In addition, liquid metal MHD analysis of geometries relevant to the DCLL using simulation tools and experimental facilities in the U.S. is one area where collaboration is possible. Comparison to EU simulations and experiments would be mutually beneficial and increase confidence in simulation results used to design respective TBMs and ancillary systems.

Test blanket modules must be diagnosed to monitor the performance during operation in order to provide information to the main ITER control systems (sensing potentially dangerous off-normal or accidental situations) as well as acquire data related to many phenomena being studied by the specific TBM experiment. It seems likely that all the TBMs and the ITER shield blanket will share many common needs for basic instrumentation, and any development of such instrumentation specifically for the fusion environment should be performed in coordination with interested ITER partners and the international team. Much of the instrumentation being used in operating tokamaks can be considered, with the added requirement of operating in radiation environment. Selection and integration of suites of diagnostics for individual U.S. TBMs should also proceed in conjunction with more detailed TBM designs.

4.7.4.2. Supporting Activities to Other Parties' Helium-Cooled Ceramic Breeder TBMs. Major research and development tasks for the solid breeder blanket concept include ceramic breeder and beryllium material development and characterization, material system thermomechanics interaction and compatibility, characterization of tritium release, retention, and permeation, structural material fabrication techniques and thermomechanics, and burnup effects on ceramic breeder performance. Most of these tasks have been partially accomplished, and one may argue that its R&D has been matured to a state that an integrated testing in a fusion environment is essential to further verify the solid breeder blanket concept feasibility as a viable blanket option for fusion power reactor applications.

The efforts of the U.S. in the development of the helium-cooled ceramic breeder blanket concept have been international in scope. The activities under the IEA collaboration form the major part of these efforts. Presently, the U.S. is contributing to the following IEA topical working groups: (1) ceramic breeder packed bed interface conductivity measurements, (2) benchmark experiment and modeling of the mechanical behavior of a breeder pebble bed, (3) high fluence irradiation of ceramic breeder pebble beds, (4) design database for solid breeder blankets and test modules, and (5) tritium control in solid breeder blankets including purge gas conditions. Specifically under Working Groups 1 and 2 the plans are, in collaboration with international partners, to complete the current phase of model development, taking into account thermal creep behavior, and complete the beginning-of-life (BOL) thermomechanics assessment including cyclic effects for a solid breeder blanket in order to satisfy licensing requirements. Since these efforts are conducted in a collaborative nature, the commitment of the U.S. to support other parties' TBM R&D includes sharing a database, computer models and analysis and participating in joint experiments. The progression (realization) of the ITER solid breeder test program provides an additional avenue where contributions can be augmented toward advancing our understanding of blanket behavior in an integrated fashion. A possible extension of the U.S. contribution can be the development of an integrated performance code for design and result analysis. Such a code development could utilize input from all parties and from different areas including electromagnetics, neutronics, thermal hydraulics, thermomechanics and stress analysis, material characterization, tritium analyses, radioactivity analyses, and safety analyses. The use of fission reactors to simulate ITER operation for a scaled test article, including burn parameters, may become important at the later stage of ITER testing as the impact of neutron fluence on acceptable performance may have to be addressed even prior to ITER testing. The U.S. could provide the fission reactor test facilities and be involved in any of the main design, fabrication, and analysis activities. However, such an activity can only become possible though international collaboration (e.g., for cost sharing…etc.). Alternatively, the U.S. could provide expertise to help identify and establish an in-pile test reactor facility for the other party.

With the involvement of parties whose R&D has just begun, a channel to effectively engage their contributions should be established through international agreement under IEA auspices. These newly joined parties can benefit and advance from the available results, and the U.S. is ready to facilitate the establishment of such a channel.

Furthermore, there are helium loop facilities available at U.S. National Laboratories (e.g., SNL and PPPL). As ITER construction progresses, one could consider modifications of these facilities to meet the specific environmental conditions expected for small-scale mockup testing. In particular, the testing results can be made available to the international community.

4.7.5. Supporting R&D and Validation Program Prior to the Installation in ITER

Supporting R&D and validation program prior to the installation in ITER is essential for the success of the TBM program. The necessary R&D and validation programs for the DCLL and HCPB are presented in the following sections.

4.7.5.1. Supporting R&D and Validation Program Prior to the Installation in ITER for the DCLL. For the DCLL concept the following common and unique R&D and validation programs are needed.

Common R&D are in the area of material development, like ferritic steel fabrication technology, material compatibility studies, tritium extraction from Pb-17Li and generic diagnostics; as well as necessary R&D in the development of Pb-17Li based breeder systems and supporting ancillary systems and components.

The DCLL option also has unique issues that will require near term and parallel R&D in concert with the testing program in ITER.

FCI. The function of SiC_f/SiC as an FCI is an integral part of the DCLL concept. Any flaws and failures of the FCIs will impact the MHD and thermal behavior in significant ways. The FCI material (SiC composite or some surrogate) must be fundamentally compatible with Pb-17Li at high temperature in a dynamic, non-isothermal system with FS structure and ancillary equipment thermophysical at lower temperature. The material must be stable in a neutron environment with well-characterized thermophysical properties meeting the needs of the MHD, thermal and tritium permeation issues. SiC composite properties and fabrication techniques will need to be developed including some compatibility experiments with Pb-17Li in static and dynamic tests.

MHD Flow Testing. Circulating Pb-17Li for power removal requires higher mass flow rates than the HCLL concept; correspondingly, MHD and heat transfer analysis of TBM flows in poloidal channels and manifold regions in a simulation facility will be needed. Initial experiments can possibly be done with Pb-17Li MHD simulants like lower melting temperature gallium alloys or Pb to look at normal operation pressure drop and flow balancing issues. But soon, tests will need to be conducted in a prototypic Pb-17Li flow loop including effects of elevated temperature compatibility, mass transport and stress to investigate FCI failure modes. Continued development and application of corresponding simulation tools are also needed, and planning for required MHD experiments in the U.S. and international MHD facilities is underway.

Integrated Helium Flow Test. The DCLL structure with poloidal Pb-17Li channels and complex helium flow channels needs careful design and optimization in order to ensure adequate wall cooling with no "hot" spots. This system will need to operate within the temperature limits of the ferritic steel, and will need to maintain sufficient strength during accidental pressurization of the Pb-17Li channels in the loss-of-coolant accident scenario. A dedicated integrated testing facility, including the helium flow loop, will be needed. Corresponding tests should include simulation of high pressure helium, electromagnetic effects and thermal loads from normal and off-normal plasma operation, including startup and shutdown. It is possible that such a facility may be available or developed internationally.

Advanced Materials. For higher performance applications like power reactors, the use of refractory alloys, SiC-composite, and corrosion-barrier-coated super-alloys are all being evaluated for high temperature tritium permeator and heat exchanger tubes, as are direct contact tritium stripers and heat exchanger technologies. These are presently planned as complementary R&D programs to be implemented as much as possible in parallel with the ITER TBM program.

A detailed planning and costing exercise to determine required R&D, testing facilities and resources needed to bring the DCLL concept to ITER for testing is just now beginning in the U.S. community. In the near term, necessary R&D for the EM/S TBM module will be emphasized in order to meet deadlines for the initiation of construction and prototype testing.

During this period, contact with the HCLL community in EU and the Pb-17Li design development in China is needed in order to investigate the possibility of collaboration, which is necessary to get the optimum use of limited resources.

It has been estimated that it will take about 36 months to complete the fabrication and testing of a 0.5 m^3 module starting from the engineering design. Therefore our R&D plan calls for initiating this process about 4 years before the target date of the ITER installation, allowing one year for the integration test before installation in ITER.

4.7.5.2. Supporting R&D and Validation Program prior to the Installation in ITER for Helium Cooled Ceramic Breeder Pebble Bed Blankets. Considering that the environmental conditions are highly uncertain during the initial years of ITER operation, the U.S. believes that a joint testing program for the HCPB concept is particularly effective during this period of ITER operation. This period could be covering the H-H, D-D, and early D-T phases. Since the U.S. does not plan to provide an independent test module, we will optimize the collaboration with the EU and Japan, and the sharing of their ancillary equipment. Presently, the U.S. plans to contribute unit cells and submodule test articles that focus on technical issues of interest to all parties. The decision to test one or both of these two options will be made in a few years, in concert with the ITER program and the U.S. budgetary situation.

Transient Testing. It is proposed that a joint task of off-normal tests, including disruption EM forces, should be performed to demonstrate the capability of the TBM to survive off-normal events in ITER. However, since the resulting forces depend upon specific designs, this test can be scheduled towards the end of the R&D when the specific design details are available.

Material Development. Near-term R&D on the structural material should focus on the issues of fabrication and bonding, database evaluation and engineering design code development. This effort should include incorporating thermal-physical and mechanical properties of the welds and joints into the state-of-the-art finite element thermal and structural codes to ensure that the performance of critical areas of the design is adequately addressed. Irradiation effects should also be considered at the later stage of development.

Neutronics. The neutronics R&D should focus on the development of measurement techniques and instrumentation needs. This should be supported by sophisticated 3-D neutronics calculations to help investigate the effects of the boundary conditions on the TBM designs in the understanding of the computational results.

Thermomechanical. The near-term focus on the thermomechanics module R&D is the continued development of a predictive capability to address the cyclic effect on the integrity of the pebbles and dimensional stability at the interface and to model the inter-relationship between the formation of the gap and subsequent temperature and stress responses.

Helium Flow Loop. To complete the design, helium flow stability analysis in the complex distributing and collecting manifolds is needed, and a small scale manifold test in a helium flow loop should be performed in order to benchmark the calculations. As a part of the R&D program, an integrated computer code which models the integrated behavior will be developed for design and analysis.

It appears that the R&D for the next 3-5 years should continue to focus on the development of the database necessary for TBM design and fabrication and performance prediction. In the meantime, efforts will be made to coordinate with the WSG-1 community to develop qualification criteria and technology needed for the test program of the helium-cooled ceramic breeder blanket concepts. For the remaining years prior to ITER testing, R&D should focus on the fabrication and testing of the TBM as well as the preparation of the auxiliary systems needed for the testing. During this later period, it is the U.S.' intention to involve industrial teams to play a lead role on the TBM fabrication and testing.

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5 - TESTING PROGRAM BY BLANKET CONCEPTS

5.1 Recall of the Agreed Approach

Some Parties envisage focusing on confirmation tests. For this kind of tests, TBM size of half a port appears to be the best compromise between obtaining sufficient measurement sensitivity and maximizing of the number of TBMs that can be tested simultaneously taking into account the space limitations and shield efficiency. Other Parties prefer to focus on parallel and sequential, functional tests, which require smaller size sub-modules, each of them designed using engineering scaling.

In all cases, taking into account the reduced FW loads compared to DEMO, the agreed TBM testing approach is to have, for each blanket concept, a series of different TBM designs, each one devoted to specific testing objectives, starting from the initial TBM in the H-H phase where no nuclear heat is present to the one installed at the end of the D-T phase where pulse-length longer than 1000 s and a large number of back-to-back pulses could be expected.

In relation with the Parties proposals, five blanket design families have been defined:

1) He-cooled ceramic/Be blankets
2) liquid eutectic Lithium-Lead blankets
3) water-cooled ceramic/Be blankets
4) liquid Lithium blankets
5) molten salt blankets

For each blanket family, a Working Sub-Group (WSG) has been created by TBWG at the beginning of 2004 in order to evaluate the status of the TBM design and R&D activities, to propose a development strategy agreed among interested Parties, and to define an appropriate testing program. The WSG on molten salt blanket has been put on hold because no significant activity could be performed. Each of the other 4 WSGs has prepared a test plan "as common and coherent as possible" and tried to identify interest and/or need of collaboration between Parties to possibly avoid duplication.

Again, in order to allow some flexibility on the final port and space allocation, the various test programs have been prepared independently from each other. However, because of the expected lack of space in ITER test ports, but also in port cell and in TWCS vault, each WSG has been requested to minimize as much as possible the required space and to promote collaboration between Parties. Potential of time sharing strategy for TBMs testing of different Parties could also be investigated.

5.2 Assumed ITER scheduled maintenance shutdowns

It has been stated that TBMs test program should not jeopardize the ITER base machine operations and availability. This requirement means that TBM replacements have to be done during ITER scheduled shutdowns for maintenance.

Therefore, in order to define TBMs testing sequence and/or time sharing between different TBM designs, it is necessary to know the frequency and length of the ITER scheduled maintenance shutdowns.

At present, the detailed ITER maintenance schedule is not yet fully defined. However, a preliminary schedule has been assumed in the 2001 TBWG Report [**5-1**].

For the present TBWG activities, it has been agreed to keep the same assumptions, as recalled hereafter. The assumed ITER scheduled maintenance plan for the first ten years of operations foresees a one-month scheduled shutdown at the end of each year and a three-month shutdown before the beginning of the D-T phase. The possibility of unscheduled shutdowns is not taken into account.

5.3 Solid Breeder Helium-Cooled (SBHC) blankets

5.3.1 Introduction

Solid Breeder Helium-Cooled (SBHC) blanket concepts have been proposed for testing in ITER by all of the 6 International Parties (IPs). While some of the Parties have developed their test plans according to their projected SBHCs for the DEMO reactor (as described in Section 4), the testing strategy, and therefore the proposed TBM, varies according to each Party's view on the DEMO development plan. These proposed SBHC TBMs share similar features:

a) Use of high pressure helium of ~ 8 MPa operating between 300°C and 500 °C. The helium flows in small channels (or tubes) embedded in the structure, and removes the surface heat coming from the plasma to the FW and the volumetric heating from the breeding/multiplier/structural materials.
b) Use of a ferritic/ferritic-martensitic steel as structural material. In general, the design temperature limit of this material (550°C) dictates the maximum operating temperature of helium.
c) Use of a ceramic pebble bed breeder, including a single-size or a binary-size pebble bed of Li ceramic breeder such as Li_4SiO_4 or Li_2TiO_3.
d) Use of Be as a neutron multiplier in the form of a pebble bed (a single-size or a binary-size) or solid porous block.
e) Use of a low pressure (0.1-0.2 MPa) helium purge gas to extract tritium produced in both the breeder and Be zones.

However, different configurations have been proposed by each IP. The major differences are summarized below:

- Japan, China and Korea present designs of parallel bed configurations in which the ceramic breeder (CB) and Be pebble beds are arranged parallel to the FW, including single-size CB with a packing of about 63% and binary Be beds with a packing of about 80%.
- Korea proposes to fill part of its TBM bed with graphite pebbles (or small blocks) as a reflector, with the objective of reducing the amount of beryllium use.
- The US proposes the testing of two design configurations, including a parallel and a perpendicular configuration. The goal is to evaluate the effects of the configurations on tritium production and pebble bed integrity.
- EU presents a blanket design that is characterized by a robust box and an arrangement of the breeder/multiplier in breeder units with perpendicular beds. Both beryllium and ceramic breeder beds are single-size.
- RF proposes a breeder-in-tube (BIT) configuration with CB binary-size beds. Beryllium in the form of a porous body fills the space surrounding the helium/breeder tubes.

5.3.2 ITER Testing Strategies

The testing strategies vary according to the different Parties' views on DEMOs development; especially notable are the differences between the US and EU test plans. EU proposes a strategy oriented to a "fast track" approach assuming that ITER is the unique step to the DEMO reactor. In this strategy, the ITER tests have a particular meaning: they have to provide as much information as possible on the behavior of the selected system, and for this reason, any technical expedient is adopted in order to reduce the extrapolation gap between ITER and DEMO. The consequence of this strategy is the necessity for both a large TBM (about 1 m^2 of surface exposed to the plasma) with relevant geometrical similitude and the simulation of DEMO relevant values for the primary testing parameters.

On the other hand, the US considers ITER an important fusion testing device for performing initial fusion "break-in" tests, including calibration and exploration of the fusion environment. Part of the fusion environment exploration is the screening of a number of configurations. In addition, the US believes a dedicated fusion component test facility is necessary to reduce the high risk of initial DEMO operation. The US does not propose to test independently a specific configuration, but rather to evaluate several options of blanket arrangement in co-operation with other parties. The other strategies proposed by the Parties fall between these two positions.

Nevertheless, the proposed testing program (number of TBMs, time schedule, the objective of each test, etc.) is very similar. In general, it follows the suggestions of the final document of TBWG for the period of extension of the EDA [**5-2**], and 3 or 4 different TBMs have been proposed by each IP to be inserted successively in ITER to test different aspects of performance issues for the respective concepts.

The first conclusion of that document was that tests of any relevant blanket concept cannot be performed with a simple look-alike mock-up because the major parameters of ITER (such as neutron wall load, duty cycle) are too far from the assumed reactor conditions. One way to compensate for deficiency is to use many TBMs, with each dedicated to a class of experiments. To ensure that the test results of the reduced-scale conditions such as ITER can be meaningfully extrapolated to DEMO conditions, act-alike TBMs have been proposed. Specifically, since temperature plays a crucial influence on both thermomechanical and tritium release performance, several look-alike design parameters, including breeding zone pebble bed thickness, ^{6}Li enrichment, and/or gas composition in the beds can be modified to produce DEMO-like operating temperatures and conditions. For other tests such as neutronics, it will be more important to maintain a similar geometry and material composition, since parameters like temperature or power level will be less important if the objective is to verify the tritium breeding capability of the concept.

A second important conclusion was the necessity of integrating testing strategy in concert with the first 10 years of ITER plasma-phase operation, or a phased approach in which testing is to progress from basic structural, hydraulic and neutronic performance to a more integrated test in the later years of ITER operations. Of course the most valuable tests are the integrated tests performed in the last 2 years of the first 10 years of ITER operation, in which long pulses with a burn time longer than 1000s may be possible. This allows the TBMs to achieve DEMO-like equilibrium temperature operating conditions to make test data more meaningful. Nevertheless, several tests can already be made in the first couple of years of ITER operations. For example, neutronic investigation requires shorter pulses with neutron presence, and this will be possible beginning the 4th year of ITER operation. The behaviour of TBMs under electromagnetic (EM) transients can be tested during the H-H plasma phase. The H-H phase, although without neutron, will be extremely important for testing of the main systems and operation procedures, which allows the progressive qualification of the systems, eliminates defects, and increases the reliability of the TBM system for the later phases.

The testing program can be summarized (following the terminology used for the EU HCPB) as: 1) an EM-TBM that is inserted during the H-H phase which is dedicated to investigate the impact of EM transient forces and to measure response to such forces and test the overall systems; 2) a NT (neutron and tritium) TBM that is tested starting to the D-D phase and earlier DT phase with the objective of demonstrate neutron performances and T production; 3) a TM (thermo-mechanics of breeder zone) TBM that is used to test thermo-mechanical behavior of Be and ceramic breeder (mostly in form of a pebble beds); 4) a PI (Plant Integration) TBM that will test heat removal and tritium extraction during high performance plasma operations. **Table 5.3-1** summarizes a list of tests that are foreseen in each plasma phase during the first 10 years of ITER operations.

Table 5.3-1: List of objectives of tests during the different phases of ITER operations

Plasma phase	Test objectives
H-H	- Integrity of structure and supports during EM forces - Functionality, safety, thermal kinetics of the ancillary systems. - Measurement of electro-mechanical effects of EM transient (including ferromagnetic effects) in TBM and surrounding structures. - Heat extraction from the FW and verification of H-H surface heat data.
D-D	Test of neutronic measurement systems and first measurement campaign
Low duty D-T	- Heat extraction from the FW and from the breeding zone. - Measurement of thermo hydraulic parameter for verification of TBM performances. - Neutron flux and spectra measurement; code validation. - T production and inventory measurements. Comparison with computational results.
Low duty/ High duty D-T	- Test on thermo-mechanics of pebble beds, for code validation - Test on T permeation in the system for code validation. Effect of the chemistry variation of the coolant Helium on the T control.
High duty D-T	- Measurement of temperature and stress distribution for code validation - Long term performances - Verification of previous measurements, reproducibility
High duty D-T (long pulses)	- High grade heat generation - test of the systems with production of Helium at the design outlet temperatures (500°C). - On-line tritium recovery. Test of the extraction of T from the TBM at different conditions of the purge flow chemistry.

5.3.2 A Co-ordinated test programme

The number of independent TBMs that can be present in one port contemporaneously strictly depends on the space availability (see Section 6.4) in the port cell area, where spaces are necessary not only for TBM pipe penetrations but also for remote handling tools, frame coolant pipes, and auxiliary systems. It is further limited by space availability in the TCWS building, Tritium building, and Hot Cell building. Here an independent TBM is defined as a TBM that has its own interface with the port frame and TBM supporting auxiliary system; an independent TBM also includes those TBMs that are made up of several sub-modules and share a common back plate as an interface to the port frame and TBM supporting auxiliary system. The assessment of the space required for performing any independent TBM testing concludes that only 3 independent TBMs can be tested simultaneously if it is assumed that three half ports are dedicated to this line of blanket testing (two half-port positions in port A plus a half-port position in port C.)

A possible coordinated test program to accommodate all IPs requirements under the assumption of three independent TBMs is a time-sharing approach. In this approach, each IP would receive half of the overall testing time. Whether or not such an approach is economically effective as well as technically sound should be evaluated by the individual party; An alternative approach is to develop a common yet well-coordinated test program by reducing the number of independent TBMs. In such an approach, not only will the testing time for a given independent TBM be maximized but also a larger resource can be available for analysis, diagnostics development, and PIE, which makes testing more meaningful. The concern as to whether or not this collaborative approach would be compatible with Party's own DEMO development strategy should be evaluated by the individual party and taken into consideration when developing this international coordinated test program.

The different perspectives concerning collaboration are summarized below to help further development of this international coordinated test program:

1) An aggressive, international co-operation with the objective of having an international program of blanket development. This is of course the most effective method, but also the most difficult to implement. It requires a broader official agreement among the involved IPs, with the goal of developing this line of blanket components up to the DEMO reactor.

2) Partial international agreements for co-operation especially during the first couple of years of ITER operations. The objective can be to design common TBM objects to produce data that can be used for the development of later independent TBMs or DEMO blanket concepts. This can be envisaged in tests on the neutronic and thermo-mechanical performance of the material in the breeding zone or in the testing of tritium control. It should be checked, whether this strategy will be fully compatible with the respective national program and if it is compatible with a safe and reliable operation of ITER (which requirements should fulfil a TBM with a completely new design to be accepted in a later phase of ITER operation). Some proposals in this direction have been already presented by EU, JA and KO. The US strategy is oriented in this direction (see Section 4).

3) Time sharing of a testing place, allowing the IPs to test their own concepts independently. This strategy allow the testing of more concepts, but has as the consequence of reducing the testing time available for each IP. It is clear that the most valuable tests from ITER will be possible only with the advanced plasma performances envisaged in the last two to three years, and the use of this time will be critical for many concepts. A common effort is in any case necessary to develop and operate part of the fix equipment like the HCS or common interfaces.

An example of a coordinated test program among the 6 IPs in the SBHC line based on three half-ports is shown in **Table 5.3-2**. In this table it has been assumed that all 6 IPs intend to test an independent TBM, at least in the later phase of ITER operations. The standard dimension of an independent TBM is assumed to be half a port. Modules with common helium feeds, but containing sub-modules of different party's design, or designed in co-operation with different parties, are considered as example for position 1 and 2. In position 3 an example is shown by assuming that only home TBM will be tested (consequently, a full time sharing approach.) Of course, several intermediate scenarios can be imagined.

Table 5.3-2: Example of port sharing with final test of 6 SBHC independent concepts based on the availability of 3 x (1/2 port) positions.

	1	2	3	4	5	6	7	8	9	10
	H-H			D_D						
Position 1	EM(P1+P2)			NT(P1)		NT-P2	TM-P1	TM-P2	PI(P1)	PI(P2)
Position 2	EM(P3+P4)			NT(P3+P4)			TM(P3+P4)		PI(P3)	PI(P4)
Position 3	EM(P5)		EM(P6)	NT(P6)		NT(P5)	TM(P6)	TM(P5)	PI(P5)	PI(P6)

As discussed in Section 6, this example assumes the sharing of 3 HCSs (or a unique HCS with the possibility to regulate independently the inlet He characteristic for 3 TBMs), that can be used by different parties in different times. A similar agreement could be necessary also for the Tritium Extraction Systems in the Tritium Building.

5.3.3 Requirements in performing the proposed test programs

A standard interface is required in order to implement test port sharing effectively. This interface standardization involves a common layout between TBM and the shield plug frame, a common layout for pipes, auxiliary components and measuring systems in the port cell area, and a common remote handling technique. The standardization can place constraints on operation and design of a single testing program; whether these constraints impact any national testing program has yet to be evaluated. For example, the radial dimension of any TBM would be fixed; does this impact any tests? The amount of diagnostics may be reduced due to even more stringent space availability; does this reduce the value of the tests? The need to achieve a better integration within the available port cell area to accommodate different operating scenarios (temperature, flow rate, etc.) using additional auxiliary components and associated piping would set a limit on the capability, dimensions, and numbers of the proposed system and its testing program. It is believed that a precise list of the needs and accompanying system with respect to space and time should be made available in advance to ensure that a coordinated test program is effectively developed. This concern also applies to the sharing of the helium cooling system (HCS) in the TCWS as well as the tritium extraction system in the tritium building. For such systems, design parameters can be defined according to the operational conditions requested by the most demanding Party or Parties (see § 4.3.2) with built-in flexibility to compensate all the TBM systems that are envisaged for testing. If the space availability requires only a single helium cooling system to be accommodated in the ITER building for all SBHC TBMs, a design solution and component layout to allow flowing helium gas with different chemical compositions (i.e. addition of H_2 or H_2O as proposed by several IPs) to different TBMs as well as tritium monitoring before merging the helium coolant for tritium permeation evaluation study should be studied. On the other hand, the maximum allowable tritium partial pressure in the loop (a design parameter) should be commonly decided.

While it seems not technically insoluble, this list of concerns is not exhaustive. Further study of all the issues connected to the proposed example integrated testing program (**Table 5.3-2**) should be conducted as soon as possible in order to finalize the proposal. On the other hand, JCT can help facilitate the advancement of this international coordinated test program by finalizing designs that concerning machine interface issues such as remote handling scheme, shield plug design, integration of TBM systems in ITER, etc.

5.3.4 International cooperation

International co-operation remains the key for the success of the blanket test program in ITER. In conclusion, the main lines of collaboration in the SBHC blanket testing will be summarised:

1) Cooperation in bilateral or international frameworks (i.e. IEA, IAEA, etc.) for the development of basic technologies for the SBHC blanket and TBM testing in ITER. Some agreements (i.e. in the framework of IEA for RAFM or ceramic breeder) are already operating; all the IPs are interested in continuing and expanding to continue and enlarge this kind of collaboration. Principal areas can be:

    a. Development of ceramic breeders and beryllium in the forms suitable for blanket applications
    b. Development of RAFM steel
    c. R&D in the helium technology for fusion reactors-
    d. Neutronics data base for design and calculations
    e. Tritium control and extraction-
    f. Fabrication technology and diagnostics for the TBM-
    g. Out-of-pile and in-pile-testing for the qualification of the TBM design-

2) Cooperation in ITER for the use of common equipments. This will be mandatory also if a completely independent development of a TBM line is envisaged (as discussed above). This implies:

   a. Standardization of the TBM/frame interface
   b. Common use of the port cell equipment (integrated layout);
   c. Common use of one or more helium coolant systems; in addition, many different systems must be supplied over the 10 years of experimentation. Independent of future agreements on the procurement of these systems, a strong co-ordination in the design, construction and testing phases of the HCSs will be necessary. In the case that space availability becomes an issue, this collaboration could be extended to the realization of a unique "big" helium facility to supply the entire SBHC program.

3) Cooperation in the design/manufacturing/testing of "common" TBMs. This, as already mentioned, can be made for part or all the TBMs in ITER. Some proposals in this direction have already been identified; extension of this collaboration is encouraged. This implies:

   a. Preparation of common tests that can be useful for different program; design of common equipment and sharing of results;
   b. Use of sub-modules inside the testing program of an other IP; this will allow testing of different concepts in smaller test objects, but will require strong cooperation among partners in the design of a common supporting structure.

4) Cooperation (of two or more IPs) in the development of a particular SBHC design. This implies an agreement on a common design and the definition of a common testing program, and maybe also of a common strategy for the development of DEMO blankets.

The aforementioned four points lay out different levels of international cooperation, while point 2) has been addressed in more detail during the WSG-1 work and recognized as mandatory in order to achieve an integrated test program in ITER. Points 3) and 4) are currently under discussion between the IPs accompanying with technically practical proposals identified. After the official start of the ITER undertaking, international agreements have to be negotiated among the IPs to establish the international framework for this cooperation.

5.4 Helium-Cooled & Dual-Coolant Lithium Lead (LL) blankets

5.4.1 Introduction

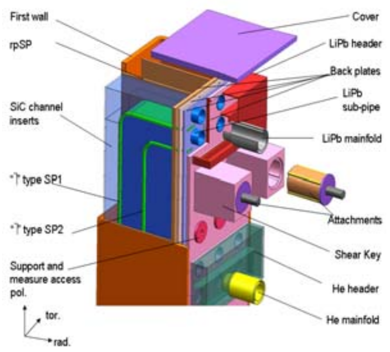
In the framework on the Working Sub-Group 2 (WSG2), 6 Parties are interested in developing (or collaborating on the development) and testing LiPb-LAFS-He concepts (LL blankets) in ITER.

Three Parties have proposed their own TBM design with the associated test program in ITER:

- CH with the Dual Functional Lithium-Lead (DFLL) concept (see section 4.2),
- EU with the Helium-Cooled Lithium-Lead (HCLL) concept (see section 4.3),
- US with the Dual-Coolant Lithium-Lead (DCLL) concept (see section 4.7).
-

Three other Parties (JA, KOR, RF, see respectively sections 4.4, 4.5 and 4.6) are proposing to collaborate on tests & R&D.

The designs of the proposed TBM concepts are recalled in **Figure 5.4-1**.

<table>
<tr><th>CH proposal</th><th>EU proposal</th><th>US proposal</th></tr>
<tr>
<td>
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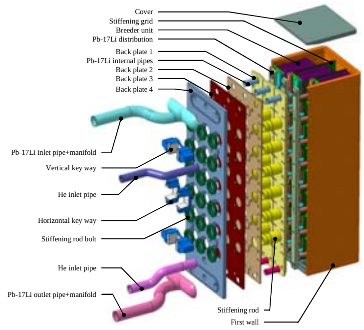
<td>
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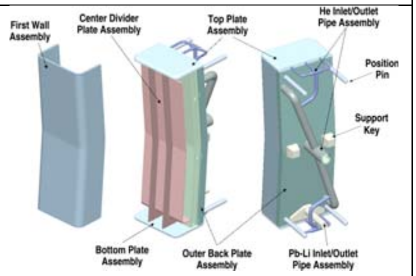
<td>
</td>
</tr>
<tr><td>Single-He / Dual coolant</td><td>Single-He coolant</td><td>Dual coolant</td></tr>
</table>

Figure 5.4-1: view of the TBMs proposed for the ITER testing in the WSG2 framework

Each TBM has the size of a vertically-divided ITER half-port. Without any collaboration, the LL blanket testing would require the use of 3 half-ports. The objective of the WSG2 work is to try to improve collaboration between Parties in order to reduce to 2 the half-ports needed to achieve the test objectives of the LL concepts.

5.4.2 Testing strategy

Besides the general TBM testing objectives described in section 3, and based on a common approach of the overall blanket testing objectives, the specific LL concepts objectives are:

- Validate structural integrity of TBM's for operation in ITER under integrated action of thermal, mechanical, and electromagnetic loads. Potential extrapolation of experimental results to DEMO conditions. This objective is common to all TBMs but it is repeated here in order to stress that it is specific to each TBM design.
- Study of PbLi flow and assessment of MHD effects.
- Validation on the accuracy of tritium generation rate calculations for general tokamak configuration and the DEMO LL blanket designs.
- Validation of thermal calculation results, and neutronics models, including nuclear libraries used in ITER and DEMO analyses especially for the prediction of tritium generation rate, nuclear heat deposition, neutron multiplication and shielding efficiency.
- Study of the tritium recovery process efficiency (from PbLi and, if required, from the He coolant system), temperature dependence of residual tritium inventory in the blanket, and T-permeation from LL towards the main He-coolant stream.
- Validation of irradiation effects studied in fission reactor spectrum with the aim to check out the impact of the neutron spectrum at least for low fluence irradiation on LAFS structures, design-specific steel-steel welds and joints.
- Study interaction and compatibility among various materials (including structure, insulators, permeation barriers, etc.) with LL in an integrated environment including representative temperature and flow fields, irradiation, and impurities.
- Demonstration of LL DEMO blanket ability to generate high temperature heat suitable for electricity generation.

To reach these objectives, a common strategy has been adopted, based around the following lines:

a) TBMs shall be inserted since day-1 of ITER;
b) TBMs shall include all DEMO relevant technologies, which will be progressively included provided they are meaningful for the concerned test;
c) A progressive TBM qualification and testing program adapted to different ITER operation phases;
d) Four TBM types are envisaged to cover the first 10 years of ITER operation.
e)

All these four points, which have a strong impact on the testing strategy and program and on the conclusions of the WSG2 group, have been agreed by all Parties and justified in details in respective DDD reports.

5.4.3 Test programs

Test programs developed by each Party proposing TBM testing in ITER are detailed in [**5-3**, **5-4**, **5-5**]. They are summarized in the following **Tables 5.4-1** to **5.4-3**.

Table 5.4-1: CHN DFLL test program

#	Test Description	Test Requirements	TBM	Min. Dur
1	EM and MHD effect	H-H	EM	1 year
2	Coating Tests, anti-corrosion Tests	H-H	EM	1 year
3	heat load tests on FW	H-H	EM	Entire Exp
4	Neutron measurement, instrument validation	D-D	NT	6 months
5	Neutron flux measurement, code and data validation	D-D	NT	1 year
6	Tritium production rate and inventory measurement	Low Duty D-T	TT	TBD
7	Tritium permeation and extraction	Low Duty D-T	TT	TBD
8	Stress and temperature field distribution for code validation	Low Duty D-T	TT	1 year
9	Integrated function	High Duty D-T	IN	Entire Exp
10	MHD and FCI integrated effects	High Duty D-T	IN	TBD
11	Radiation damage and Reliability	High Duty D-T	IN	Entire Exp

Table 5.4-2: EU HCLL test program

#	Test Description	Test Requirements	TBM	Min. Dur
1	Installation of TBM, leak tests, RH tests	Prior to day-1	EM	1 month
2	Resistance against EM forces		EM	Whole BPP
3	Functionality, safety, thermal kinetics of ancillary circuits	Vacuum in plasma chamber	EM	3 months
4	Heat extraction from FW, verification of H-H surface heat flux data	During plasma pulses	EM	10 days (10 pulses)
5	MHD pressure drop as a function of PbLi flow-rate	During plasma pulses	EM	4 weeks
6	H/D permeation from Pb-17Li into coolant, evaluation of D inventory	Vacuum in plasma chamber	EM	6 months
7	Calibration of T source, test of extractor w/ D	D-T Plasma	NT,TT	10 pulses
8	Deuterium permeation into FW coolant, verification of D-D surface heat flux data	D-D Plasma	NT	3 months
9	T inventory (experimental conditions TBD)	D-T Plasma	NT	3 months
10	Temperature fields for code validation	D-T Plasma	TT	10 pulses
11	Stress distribution for code validation	D-T Plasma (pulse > 100 s)	TT	10 pulses
12	T permeation into coolant: - no PbLi circulation, no extraction	D-T plasma w/ pulse alap & frequent repetition time	TT	30 pulses consecutiv
13	- with PbLi circulation, no extraction	" Idem	TT	3 months
14	- with PbLi circulation, with extraction	" Idem	TT	6 months
15	Reproducibility of preceding tests, reliability	" Idem	In	As long ap

Table 5.4-3: US DCLL test program

#	Test Description	Test Requirements	TBM	Min. Dur
	Magnetic Field Perturbations / Shakedown	Prior to day 1	EM/S	6 months
1	Be Erosion	H-H	EM/S	Entire Exp
2	Disruption loads	H-H	EM/S	Entire Exp
3	LM filling / freezing / remelt / thermal constants	H-H	EM/S	6 months
4	Hot FW tests	H-H	EM/S	6 months
5	Flowing LM tests and initial FCI tests	H-H	EM/S	6 months
6	Neutron Gamma flux	D-D	NT	TBD
7	Tritium production characterization	Low Duty D-T	NT	TBD
8	Nuclear heating	Low Duty D-T	NT	TBD
9	MHD flow and heat transfer and FCI function	Low Duty D-T	T/M	1 year
10	Tritium permeation and removal	Low Duty D-T	T/M	1 year
11	Integrated function	High Duty D-T	I	Entire Exp
12	Radiation damage and long term FCI performance	High Duty D-T	I	Entire Exp
13	Reliability	High Duty D-T	I	Entire Exp

5.4.3 Potentials of a coordinated test programs

5.4.3.1 Common tests

As the test programs present some common test features, and due to the fact that basic materials and systems used in the 3 concepts are the same (use of LAFS, PbLi, He, He and PbLi systems), it is in principle possible to fulfill for all TBMs some of the objectives listed above with tests realized in less than three TBM. This common tests could concern for example the codes validation, neutronics, PbLi flow and MHD features, tritium management (extraction from PbLi and He, measurements), compatibility between materials, interaction of systems (PbLi and He loops, instrumentation), ...etc. With a good collaboration on the preparation of the tests and a strong interaction in the exploitation of the results, test performed on 1 or 2 TBMs could give some pertinent results for all 3 TBMs.

5.4.3.2 Integrated TBM

Due to the difference in the DEMO option and blanket design of each party, it seems difficult to envisage the integration of one concept to another one (i.e., two concepts integrated in a single TBM). In particular, the main differences concern the structure itself, the PbLi flowing path and PbLi flow parameters, the fraction of power extracted by helium, and also the type of steel used (CLAM for China, Eurofer for EU, F82H for US). The requirement to have DEMO relevant TBM testing makes difficult the possibility to converge towards a unique representative structure. This is particularly true due to the need to validate the structural integrity of the TBM under relevant loads.

5.4.3.3 Time sharing

Taking into account the needed times for each type of test, one can see that theoretically the full period of 3 years is not necessary for reaching the testing objectives. For example, with a preliminary synthesis based on the planned program, the EM-TBM should in principle reach its objectives within 18 months. However, this time, which is a strict minimum, does not take into account:

- any margin,
- any repetition of test,
- the fact that operational events (for example all type of disruptions and ELM events) could not occur during this time,
- the need of repair (for example if some failure of instrumentations does not allow to reach a test objective).

Moreover, the first year of operation in each ITER phase is crucial in order to be able to integrate modifications in the design and fabrication of the following TBMs.

Taking into account the functional constraints, and the fact that in principle TBM replacement are possible only once a year (during ITER planned shutdowns), the main conclusion is that from an operational point of view, time sharing could reduce the possible fulfilment of test objectives.
However, if an agreement is reached on common tests in the first period of each phase of operation, it could be in principle possible to condense into a shorter period the tests, which can not be shared, such as the validation of the box behaviour.

In order to ease the TBM replacement in the short time allowed (1 month of ITER shutdown), it is necessary to tend towards standardized interfaces between TBM and frame. It may be difficult to achieve standardization in terms of size of connecting pipes due to the different piping geometries (concentric tubes for US, simple tubes for others), but standardization could occur in the case of a back-shield directly linked with the TBM.

5.4.4 Collaborations

All parties have expressed a strong interest to collaborate on tests & R&D. The main items identified for possible international or bi-lateral collaborations are:

- Instrumentation (in particular neutronics, tritium related tasks, MHD, PbLi pressure and flow),
- SiC insert development,
- PbLi loop experiments (compatibility with LAFS and SiC, corrosion, impurities),
- Tritium cycle (permeation, extraction from PbLi and He).

Larger detail on the envisaged collaborations for each Party is given in section 4.

5.4.5 Conclusions

Concepts from the LL field present several basic differences and it appears difficult to combine them in an integrated structure TBM (common TBM featuring two different DEMO concepts). This will lead to a loss of representativeness and of DEMO relevancy. The time-sharing strategy could offer a solution for reducing the need in term of testing space provided it is linked to a common test program with useful results obtained for all 3 TBMs in each initial TBM test. However, the potential of the time sharing strategy has to be analyzed in more details taking into account the required replacement time. However, it appears clear that only with a strong co-operation between Parties can the loss of testing objectives achievement be minimized. In particular, some common test objectives have been found, and with a coordinated elaboration and exploitation of the results the test of each TBM could provide some results of common interest.

5.5 Water-Cooled Ceramic/Be (WCCB) blankets

5.5.1 Test Strategy and Schedule

Module tests in ITER will be dedicated mainly for investigating blanket characteristics under 14 MeV neutron irradiation, under huge electromagnetic field and with their synergetic effects including high surface heat flux to the module first wall, and finally for demonstrating the blanket performance to be applied to the fusion power demonstration plant. For these investigation and demonstration, a series of tests in ITER are planned as shown in **Figure 5.5-1** with testing goals summarized in **Table 5.5-1**.

Description of each test is given below.

(a) System check-out
At the beginning of the ITER operation, overall system check-out will be performed including TBM ancillary, remote handling, instrumentation and data acquisition systems. Additional system check-out for neutronics relating systems will also be conducted at the start of low duty D-T operation.

(b) Characterization (basic behavior test) without neutron irradiation
Following the first system check-out, therefore even prior to the low duty D-T operation, in the former 10-year period of ITER operation, a TBM will be installed in ITER to investigate the structural integrity against the induced electromagnetic loads and high surface heat flux from the plasma, the heat removal capability for the surface heat flux, and the effect of the ferromagnetic ferritic steel structural material on the plasma controllability.

(c) Remote handling test
Before the TBM and the ITER basic machine are activated by D-T operations, remote maintainability of the TBM will be demonstrated more than once, e.g., on an occasion TBM exchange.

(d) Neutronics test
At the beginning of low duty D-T operation, or even from D-D operation stage, neutronics test will be started. The objective of this test is to characterize the neutron environment and examine the accuracy in predicting key neutronics parameters including tritium production rate, nuclear heating, induced radioactivity and decay heat. The TBM could be divided into sub-modules which would individually incorporate representative parts of the candidate blankets. Neutron fluence required for these tests will be very low such as several tens of shots at most. Since long burn time will not be necessary either, some tests can be conducted using sub-modules without cooling but only with simulated coolant tubes/panels for investigating an internal neutronics environment of candidate blankets more precisely.

(e) Performance test

After the characterization of neutronics environment, the performance test will be conducted. The objective of this test is to evaluate tritium generation/extraction performance including the capability of continuous in-situ tritium recovery, heat generation/removal performance including the extraction of high quality energy for electricity generation, thermal, mechanical and hydraulics behavior under thermal (surface heat flux and nuclear heating) and electromagnetic loads. This test will be performed up to the end of the former 10-year period of ITER operation. During this period, the TBM could be changed, possibly 1 ~ 2 times, depending on required modification and improvement based on the test results. In addition, when successful and promising results are obtained from the out-of-ITER R&D's, e.g., materials development, they could be also incorporated.

(f) Advanced performance test

Further performance test with advanced types of TBMs will be planned in the latter 10-year period of ITER operation. Since two-year change-out from the ITER shielding blanket to the ITER breeding blanket could be planned, more sufficient examination of the irradiated TBM would be carried out, then more comprehensive design improvement could be achieved during this period. Based on this investigation and improvement, a modified TBM would be installed as long as possible to investigate the effect of neutron fluence on the blanket performance, e.g., due to irradiation damage of materials, and to enhance the reliability, but remaining the possibility of its changing to further improved modules, up to 1 ~ 2 times, incorporating newly developed technologies. An electricity generation will be demonstrated, possibly in the late period of this stage, using the thermal power extracted from the test module by high temperature coolant.

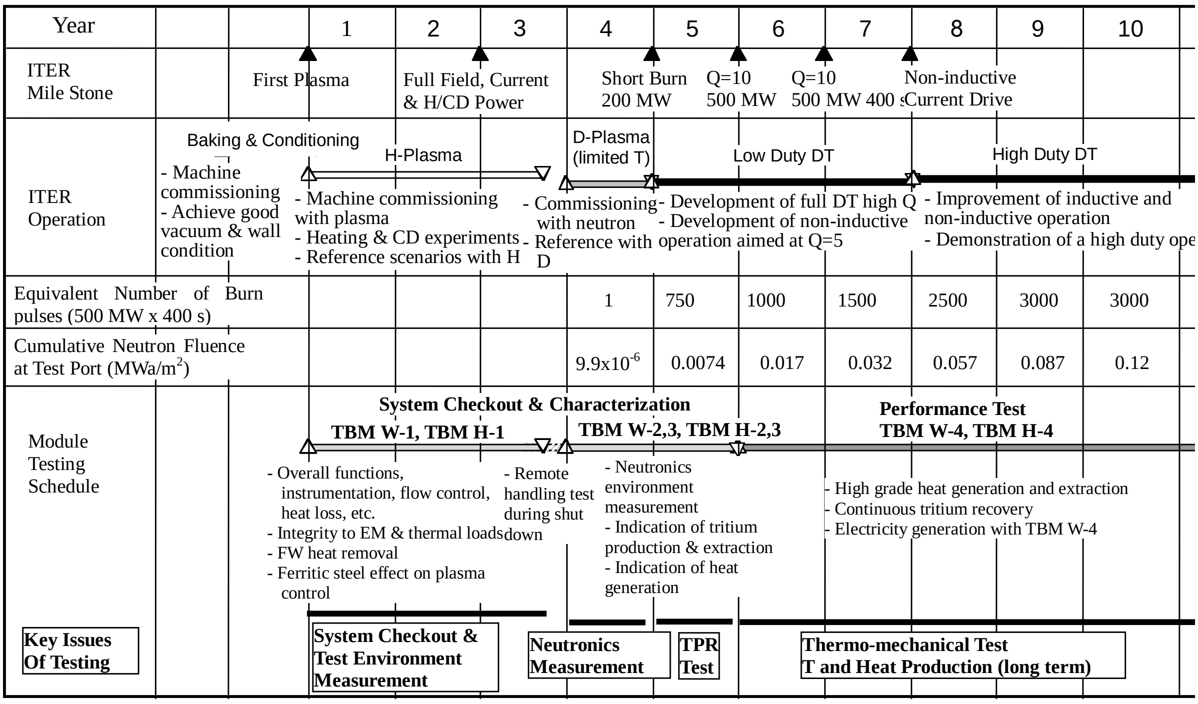


Figure 5.5-1: Schedule of ITER blanket module testing.

5.5.2 Extrapolation to the Fusion Power Demonstration Plant

Major roles of a DEMO blanket are: 1) tritium production and recovery for fuel self-sufficiency, 2) extraction of high-grade heat for electricity generation, and 3) a part of radiation shielding for super-conducting magnets and workers. In addition, the DEMO blanket itself should withstand against thermal (surface heat flux from plasma and nuclear heating), mechanical (coolant pressure and purge gas pressure), and electromagnetic (especially during plasma disruption) loads under severe neutron irradiation. Therefore, in the tests using TBM's in ITER, their performance to fulfill these basic functions should be investigated and finally confirmed. However, since most device parameters of ITER as an experimental reactor are likely to be lower than those of a DEMO reactor, some ways to extrapolate the TBM performance to the DEMO blanket performance are needed.

(a) Tritium production and recovery

Tritium production in the DEMO blanket can be estimated by analysis once the analytical predictability is confirmed and/or methods to obtain the prediction accurate enough are established through the in-ITER module tests. Tritium recovery process including the helium purge gas conditions for tritium extraction from the blanket can be also investigated and demonstrated by the in-ITER module tests. On the other hand, tritium release from solid breeder materials is strongly related to operational temperatures of the breeder materials. Namely, the breeder materials should be maintained within an appropriate temperature range to enhance the tritium release. Temperature gradient in the materials is caused by nuclear heating which is proportional to the neutron wall load. Since the neutron wall load of ITER is lower than that of a DEMO reactor, the breeder temperatures become lower than the above appropriate range if the test module is designed to be "look alike". Therefore, the test module should be designed as an "act-alike" module, by modifying the coolant channel arrangement and the thickness of breeder and multiplier regions, so as to simulate the breeder temperatures in the DEMO blanket and demonstrate the tritium release from the breeder materials.

(b) Electricity generation

Possibly electricity generation or at least the extraction of high-grade heat can be demonstrated by using high temperature coolant. To reach thermal steady state, or at least quasi-steady state, of blanket materials and outlet coolant temperature, plasma burn times longer than 400-500 s for blanket front part and 3000-5000 sec for blanket back part are required. Heat removal performance and thermal-hydraulic characteristics can be also investigated and demonstrated with some supplementary analyses for overall dimensional difference between the TBM and the DEMO blanket.

(c) Shielding performance

Shielding performance of the DEMO blanket can be predicted accurately enough once analysis methods are established and data on nuclear responses are obtained through the in-ITER module tests.

(d) Structural integrity

As for the structural integrity, synergetic effects of thermal, mechanical and electromagnetic loads can be investigated with the TBM. The same coolant and purge gas pressure loads as of the DEMO reactor will be applied to the TBM. Supplementary analyses can compensate the dimensional difference between the TBM and the DEMO blanket. Electromagnetic loads are dependent on plasma parameters including the method of stability control, which are not well-defined for the DEMO reactor at present. In any case, prediction methods for the induced electromagnetic loads and the mechanical response of the blanket structure are expected to be established through the in-ITER module tests and will be applied to the DEMO blanket. Thermal stresses are caused by temperature gradient across the material. The temperature gradient is caused by heat loads such as surface heat flux and nuclear heating. Since the heat loads in ITER are lower than those in the DEMO reactor, an "act-alike" design of the test module can be considered to simulate thermal stresses of the DEMO blanket. For the "act-alike" design, modification of material thickness and cooling conditions can be taken into account.

(e) Irradiation effects

Irradiation effects on the materials properties and the blanket performance are a most critical issue for the testing. Some changes in thermo-physical properties of non-metals, e.g., thermal conductivity, will occur below neutron fluence of 0.1 MWa/m^2. Several important effects become activated in the fluence range of 0.1-1 MWa/m^2. These effects include solid breeder sintering/radiation cracking and possible onset of breeder/multiplier swelling. Therefore, the blanket behavior relating these effects and their interactions can be investigated by the in-ITER module tests. However, irradiation effects, especially on structural material characteristics including DBTT change, helium embrittlement and swelling, will appear in the fluence range of 1-3 MWa/m^2 or higher. Therefore, these irradiation effects on structural materials need to be evaluated separately in out-of-ITER R&D program, such as IFMIF.

5.6 Li-based blankets testing program

Three Li-cooled TBM concepts are actively developed by the ITER project-participating countries: self-cooled Li concepts (RF and Japan) and He-cooled Li-breeder concept (Korea). China and the US are expressing their interest in testing these concepts. The RF and recently Korea have planned to start their TBM testing on day one of the ITER plasma, while Japan – at the later stages of ITER operation.

All participants to the Li-based Blanket Working Subgroup agreed on the overall objectives of blanket tests in ITER and on the testing goals and preliminary testing program of TBMs in ITER as described below.

In line with the above stated common testing strategy, the tests of a number of DEMO "act like" TBMs, though retaining all DEMO blanket materials, are planned. Some of the modules are of the diagnostic type for measuring ITER magnetic field topography inside the TBM elements, for neutronics testing (code and nuclear data validation during the D-phase and the early DT-phase for the combination of the materials used in TBM), for transient electromagnetic tests during the H-phase (measurement of magnetic fields, eddy currents, forces and moments acting on the TBM's during fast plasma transients).

Specific objectives of Li-cooled TBM tests in ITER-FEAT are the following:

- Demonstration of electro-insulating barriers/coating performance under combined effect of neutron irradiation, high magnetic field, temperature gradients and liquid metal dynamic flow.
- Demonstration of engineering technique for self-healing of electro-insulating coatings and for on-line removal the impurities from Li.

These tests may be started from the very beginning of ITER operation, since at first they require just the magnetic field. It is even preferred to have different values of toroidal magnetic field only, then of poloidal magnetic field, and finally, a combination of different values of toroidal and poloidal magnetic fields acting together.

The other test objectives are specified in the stated above common aspects of testing strategy.

The test modules will differ in materials (with or without Be multiplier, WC or TiC shielding layer, some types of electro-insulating barriers/coatings on the interface of Li/structure material, etc.), slightly in design (to simulate different characteristics of DEMO blanket). In general, the testing will proceed in the following sequence: i) TBM systems checking, ii) basic characteristics and performances testing, iii) short term functional tests, and finally iv) long term functional tests.

Table 5.6-1. Possible scenarios of TBM tests in ITER-FEAT

ITER-FEAT operation stages	Years from First Plasma	Number of pulses	TBM types
H-plasma D-plasma (limited amount of T)	1…3 4	–	Installation of 1 to 4 TBM-0 modules, replacement - ~1 module per year
D-T-plasma with low duty cycles	5 6…7	750 (1000…1500) per year	Installation of 1 to 3 diagnostic modules TBM-1 and 2 modules TBM-2 to be replaced in (7…12) months
D-T-plasma with high duty cycles	8…10	(2500…3000) per year	TBM-3 (1 to 2 modules) to be replaced in 1…2 years

Tests description:

- **TBM-0 (1 to 4 modules) - modules for diagnostics and preliminary tests (first four years of ITER operation)**

Test objectives:

- Checking of all auxiliary systems functioning;
- Magnetic field topography investigation;
- Investigation of magneto-hydrodynamic and hydraulic characteristics of the module and the liquid metal cooling system elements (may be started from the 1st year of ITER operation);
- Checking of functioning and preliminary tests of the electro-insulating barriers/coatings (may be started from the 1st year of ITER operation);
- Investigation of thermal characteristics (use of the FW with additional electric heater with specific power up to 50÷80 W/cm^2, TBD);
- Checking of the thermodynamic characteristics of the tritium extraction system (TES) with the use of pure lithium;
- TES degassing modes investigation in operation with hydrogen plasma;
- Investigation of FW stresses and deformations, arising as a result of electro-dynamic loads.

Boundary conditions:

- Operation both without plasma and magnetic field and with hydrogen and deuterium plasma and magnetic field;
- All ancillary systems are required.

Order of magnitude of tests duration:

- Separate tests duration some hours;
- Total duration of the tests some months.

- **TBM-1 (1 to 3 modules) - diagnostic module (neutron flux characteristics) and TBM-2 (2 modules) – short term functional tests (5th to 7th years of ITER operation)**

Test objectives:

- Neutron characteristics investigation at different options of the shielding and multiplier (WC, TiC, with or without Be);
- Short-time tests of TBM and ancillary systems with optimum shielding option (demonstration of high-grade heat extraction and high temperature integrated TBM performance is achieved with lithium volume flow rate and inlet temperature regulation);
- TES tests.

Boundary conditions:

- Low duty DT plasma;
- All ancillary systems are required.

Order of magnitude of tests duration:
- Separate tests duration some (minutes…hours);
- Total duration of the tests some months.

- **TBM-3 (1 to 2 modules) – functioning tests**

Test objectives:
- Long-term tests of TBM and ancillary equipment with the most acceptable options of design, chosen during the previous tests.

Boundary conditions:
- DT plasma;
- High duty cycles;
- All ancillary systems are required.

Order of magnitude of tests duration:
- Tests total duration (1…3) years.

Some modification/additions from the participants to the specific objectives of Li/V TBM tests, as well as to the TBM types design might be possible later. Recent Korean proposals for testing of Helium-cooled Li-breeder TBM will require further coordination of test plans. Participants are well aware that they should stay inside the restrictions imposed by Li ancillary systems and by port/space sharing requirements or replacement frequency. It means that their TBMs should have the same interface with the Frame/Port Plug to have a standardized Frame/Port Plug for all TBMs of Li family (or every TBM should use its own Frame/Port Plug), use the same ancillary systems or, at any rate, fitted into the VV Port Extension and near-port transporter.

5.7 Dual-Coolant Molten Salt (DCMS) blankets

In the Working Subgroup of the TBWG the US and Japan are expressing their interests in planning the testing program on Dual-Cooling Molten Salt (DCMS) for the TBM. These interests are mainly based on the R&D activities in the US-Japan joint program JUPITER-II, which mainly focuses on liquid breeder blanket in terms of advanced concepts, including molten salt, and supplies key databases relevant to fusion blanket engineering such as molten salt purification, materials compatibility, tritium chemistry, thermofluid and MHD effects.

Standing on these R&D activities, the reference design of DCMS has been reported by the US activities. The main concepts are the followings:
1) Use of ferritic steel for the structure material.
2) First wall He loop to the tokamak cooling water system (TCWS)
3) Intermediate He loop between the liquid breeder and the TCWS.
4) Concentric pipes for the liquid bleeder access tubes.
5) Centrifugal pumping system for molten salt.
6) Detritiation bypass system with the He relief line and a Pd/Ag permeator.
7) REDOX control for free F or TF with Be.

The preliminary assessment of the safety impact on ITER of a DCMS concept shows that the anticipated radiological inventories, maximum hydrogen quantities produced during accident conditions, and tritium permeation from the ancillary system are found to be below the ITER guidelines.

The activities in Japan Universities can support technical issues on a molten salt loop, which is already under operation in Tohok-NIFS Thermofluid (TNT) loop with the centrifugal pumping system and the He relief line. REDOX control with Be is very promising according to the recent JUPITER-II experiments in INEL. On the other hand, the R&D issues on concentric pipes and detritiation bypass system are common to the dual-coolant lithium-lead (DCLL) blanket.

5.8 Discussion and Conclusions on testing program

The previous chapters have shown possible coordinated TBMs test plans for each blanket family and given preliminary ideas of possible collaboration between two or more Parties.

Proposed TBM test plans make use of space sharing and/or time-sharing. Space sharing is limited by the number of available test ports, by the number of allowed penetration through the shield, by the limited volume of the port cells and by the limited dedicated volume available in the TWCS vault. Time-sharing is limited by the length of time required by each test and by the constraint that TBM replacement can occur only once a year during the planned ITER shutdowns.

If all proposed TBMs will have to be tested, it clearly appears that, despite the effort made for minimizing the space requirement, it will not be possible to perform all the expected tests during the first ten years of ITER operations.

This conclusion indicates that, in any case, strong collaboration between Parties is compulsory. Therefore, if agreements cannot be reached on technical basis, some collaboration rules will have to be negotiated and established at a political level in the near future.

5.9 References

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6 – TEST PORT CELL BOUNDARY CONDITIONS AND OPERATIONS

6.1 Enhanced Flexibility Approach

At present, the ITER Parties have proposed several independent DEMO-relevant TBMs that cannot all be tested simultaneously on day-one. Space limitation is not only due to the space available in the test ports but also to the limited space available in the port cells, in the vertical shafts, and in the TCWS vault. The situation will become even more difficult during the D-T phase because additional TBMs are proposed. The optimization of space and time-sharing among TBMs belonging to the same blanket family has confirmed that it will not be possible to perform all proposed tests and a selection will be required.

Such a selection can be performed in a few years from now and it is technically desirable to allow a large flexibility and give today only the minimum necessary requirements. In fact, in order to be able to integrate the TBM test in the ITER machine design and procurement, it is required to fix, for each port, the number and the dimension of the connection lines crossing the vertical shaft between port cells and TCWS vault and other ITER buildings. Of course, the number of connecting lines has to be minimized because a limited space is available in the vertical shaft.

Looking at the Parties proposals, it appears that most TBMs need He-coolant connection lines (with the word “line” is intended one inlet and one outlet pipes). In particular, HCCB, HCLL and HCLi TBMs need one He-line, DCLL and DFLL need two He-lines. WCCB need one water-line, while SCLi does not need coolant connection line because Li heat exchanger can be located in the port cell.

After optimization on the use of the available space, the TBWG has decided to have the following coolant lines:

- two He-lines for port nb. 16 (also called Port A),
- one water line and two He-lines for port nb. 18 (also called Port B),
- three He-lines for port nb. 2 (also called Port C).

He-coolant pipes characteristics have been standardized (same diameter, operating pressure, and material). Other service lines, such as those for heat rejection system and He purge gas, are also required. **Table 6.1-1** summarizes the required lines for each port.

Table 6.1-1: Number and type of connection lines in each port

System type	Test Port #16	Test Port #18	Test Port #2
He-coolant	2 lines	2 lines	3 lines
H_2O-coolant	-	1 line	-
ITER heat rejection system	available	available	available
He purge gas	4 lines	4 lines	4 lines
ITER component cooling system	1 line	1 line	1 line

The available connection lines at each port will become a further constraint on future testing plans together with the space availability in the port cell and in the TCWS vault. On the other hand, such constraints still allow sufficient flexibility on the choice of the TBM types to be tested in each port that will have to be performed few years before TBM commissioning.

6.2 Arrangement of Test Port Cells and Pipes Connections

6.2.1 Rationale

TBM systems includes circuit and components of the primary heat transfer system (PHTS), secondary coolant circuits and components, Tritium management components, liquid breeder loop, instrumentation packaging and control system, safety-relevant detection systems and valves.

Its complexity indicates that integration in ITER machine and building is a quite difficult operation and that TBM tests will require considerable preparation.

To maximize measurement efficiency, Tritium-related components have to be located in the port cell. The extracted Tritium is directed toward the ITER Tritium system located in the Tritium building.

Assuming two TBMs present in each port, the remaining space available in the port cell is mostly used for piping and is not sufficient for hosting the heat extraction components which will then be located in the TCWS vault, with the exception of the liquid metal heat exchanger which, because of their relatively limited dimension, may be possibly located in port cell.

The pipes connections available in each port are given in **Table 6.1-1**.

6.2.2 Port Cell A

In the frame of the general approach to have a flexible system and not compromise later agreements on port sharing, the Port Cell A has to be designed to accommodate two sets of piping connection from the TCWC and Tritium Building capable to support the contemporaneous presence of 2 independent Helium Cooled TBM objects of the type SBHC with dimensions of ½ a port each. In addition it has been proposed to standardise this lines in order to allow the sharing of the positions in the related TBM-Port Plug among the 6 parties that are interested in the testing of SBHC concepts. This standardisation is possible because the operational conditions of the 6 proposed systems are compatible (pressure, temperatures, etc.). Each of this line should be dimensioned for the removal of the heat produced in an half a port (about 1.3 MW); the dimensions of the Helium lines have been chosen as envelopment of the proposal of the different parties.

The standardisation mentioned up to now should be applied also for helium purge line that will connect the port cell to the Tritium building. At the moment a full agreement is not completed, but taking into account the present information, two independent lines for each ½ port position is tentatively envisaged. As consequence of that, four similar He-purge-lines are proposed.

In addition one line of the HRS and one of the CCWS should be foreseen for the cooling of additional equipment in the Port Cell. **Table 6.2.2-1** summarise the proposal of piping allocation in Port Cell A.

Table 6.2.2-1: Specification of pipes interfaces at the boundary of the Port Cell-A

Item	No. of pipes	Pipe size (ID/OD) [mm]	Insulation thickness [mm]	Note
He line-1	2 (*)	……/139.8	Hot: 120 Cold: 85	From SBHC-TBM to HCS located in the TCWS
He line-2	2 (*)	……/139.8	Hot: 120 Cold: 85	From SBHC-TBM to HCS located in theTCWS
He purge line-1	2	--/36	TBD	From SBHC-TBM to TES in Tritium Building
He purge line-2	2	--/36	TBD	From SBHC-TBM to TES in Tritium Building
He purge line-3	2	--/36	TBD	From SBHC-TBM to TES in Tritium Building
He purge line-4	2	--/36	TBD	From SBHC-TBM to TES in Tritium Building
HRS line	2	25/33.2	50	
CCWS line	2	25/33.2	50	

(*) under the assumption that HXs or thermal sleeves necessary for the by-pass line can be located in the Port Cell, otherwise at least 3 pipes per lines are requested.

Some consideration should be made for the integration of the piping in the Port Cell. Only the 2 He lines will be routed in the vertical shaft, the purge line are routed elsewhere and the HRS pipes penetrate the space above the port cell door. These main Helium line are the most critical ones; in fact they are rather large (about 380 mm DO including the insulation) and can reach maximum temperature of 500°C. These lines will be plugged in the vertical shaft wall and then should be routed in the Port Cell area, cross the Bio-shield and reach the TBM connection position at the flange of the TBM-Port Plug. The standard proposal of ITER [**6-1**] (see also section 2.7.1) considers only the possibility of a fix assembly of pipes connecting directly the two interfaces in the port cell. However, most of the proposed TBM system concepts foresee ancillary systems located in the Port Cell (e.g. bypass valve and thermal sleeves) and these systems should be connected to the main Helium pipes. For this reason an alternative proposal to the standard assembly studied by the ITER Team has been investigated in an EU study for the integration of the HCPB system. This integration schema [**6-2**] is illustrated in **Figure 6.2.2-1** to **6.2.2-4**.

As the major requirements for the layout are the restricted space and the necessity to realise a quick exchange of the components, the use of an integrated device - the so called Piping Integration Cask (PIC), which combines all the piping, the additional equipment and part of the bioshield into a movable car - allows the pipes to be cut and re-welded only at the boundary interfaces of the Port Cell (Interface 2 and 3), removing all the intermediate components as a block. The lay-out of Helium pipes from the interface 2 to 3 is dictated mainly by the necessity to compensate the thermal expansion of the pipes (maximum design values are 500°C and 8 MPa); large radius bends have been designed to accommodate this expansion. The accommodation of a second analogous system to support the TBM in the lower port is still possible if a common PIC is used for the integration of both systems.
A last consideration about the implication of the Port Cell A lay out on the TCWS. To feed the 2 independent He pipes foreseen in this Port Cell, two independent HCSs should be allocated in the TCWS.

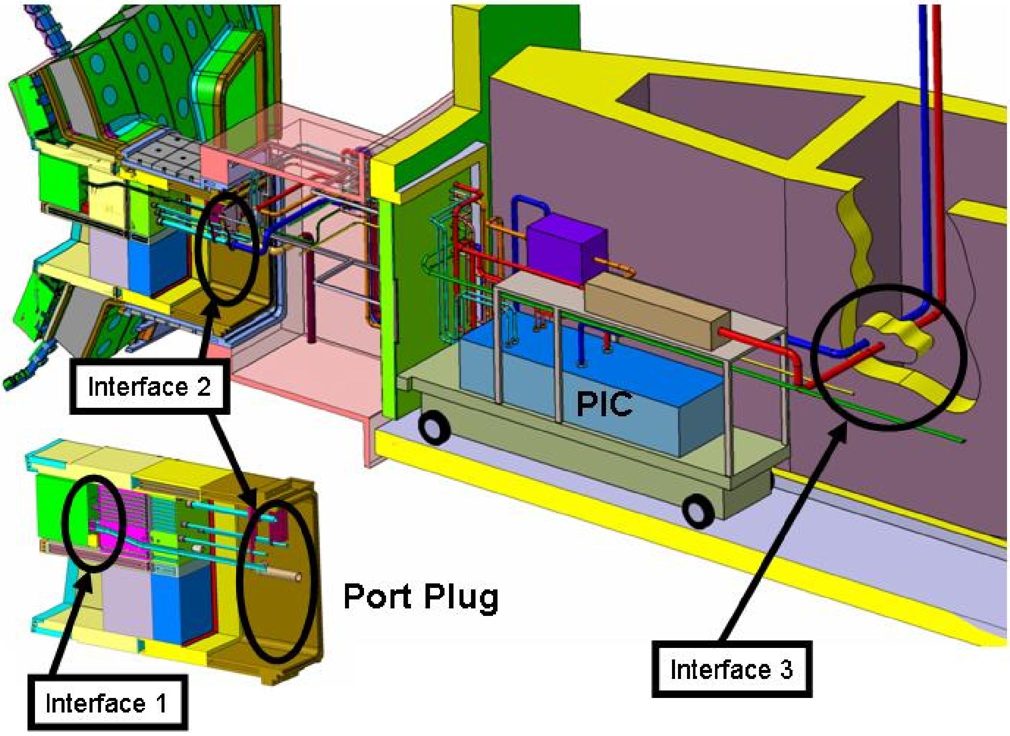


Figure 6.2.2-1: Integration scheme proposed for the HCPB TBM system in Port Cell A (PIC: Piping Integration Cask)

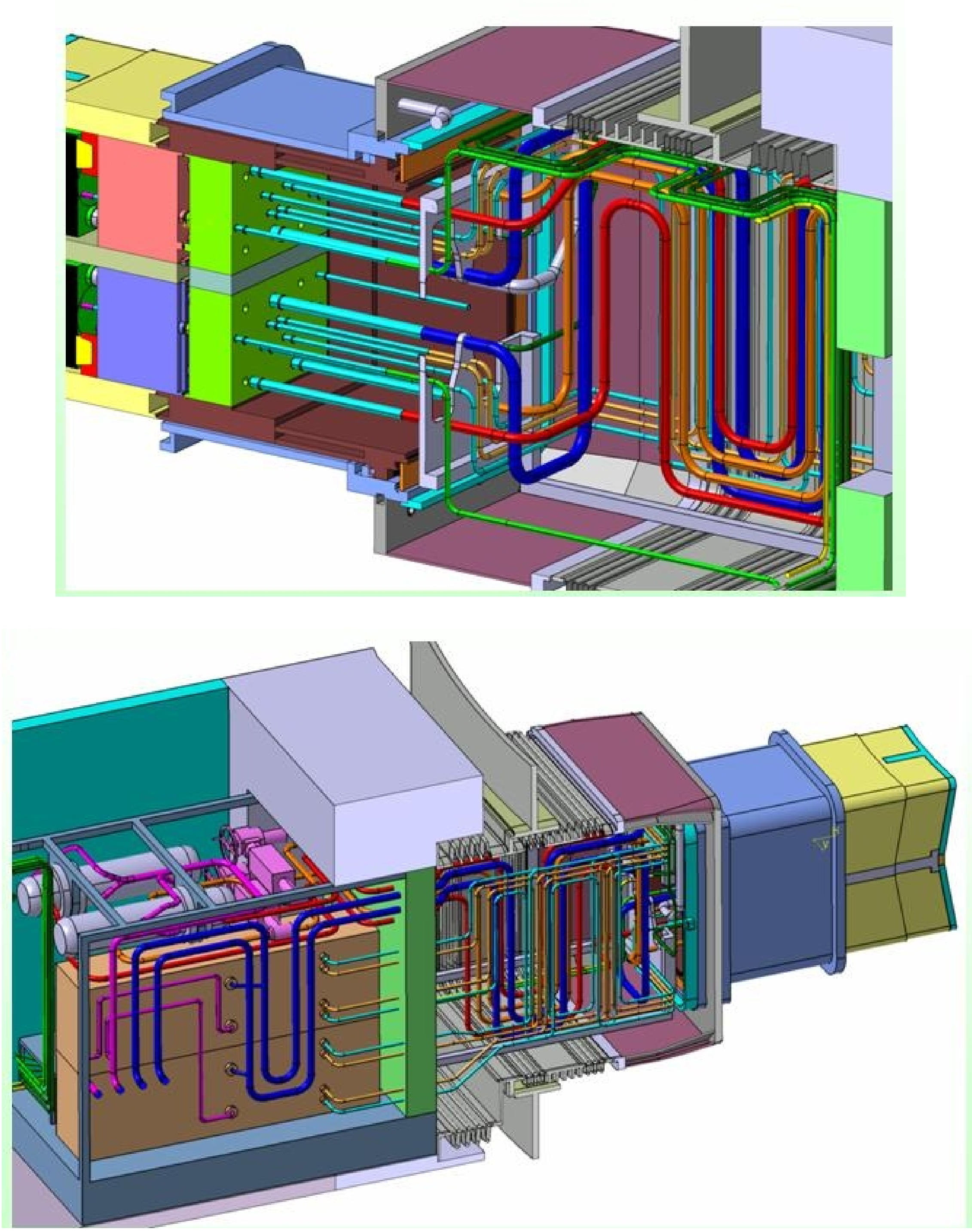
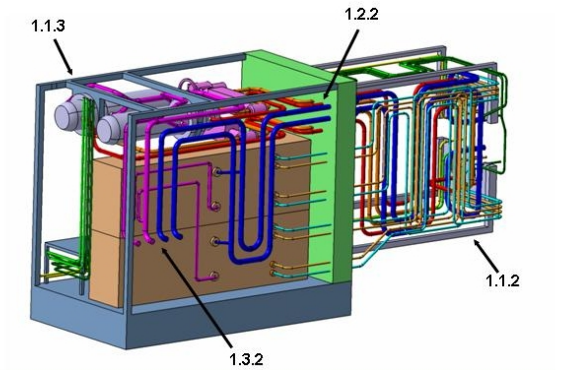
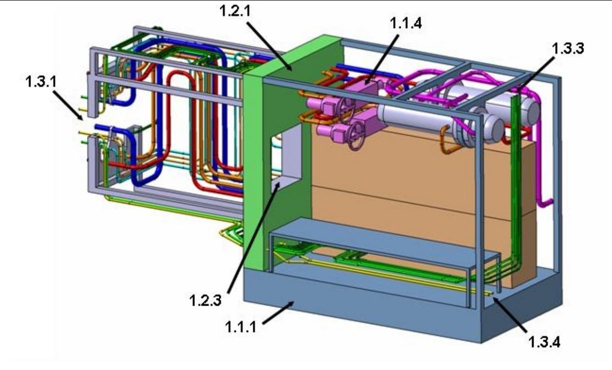


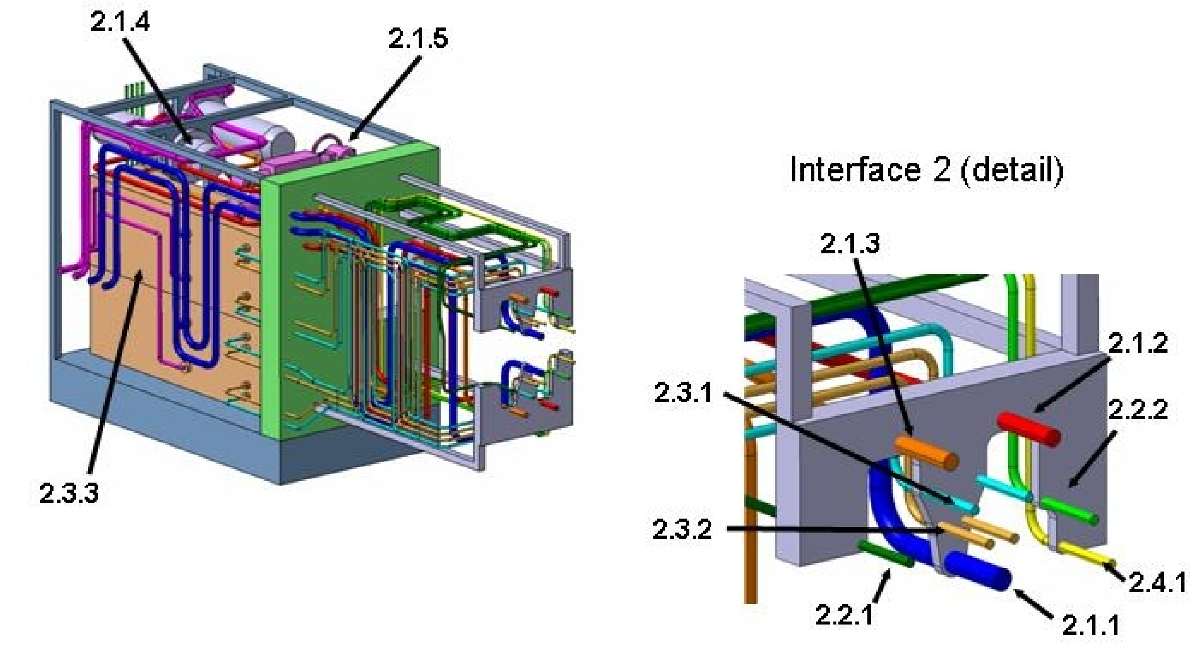
Figure 6.2.2-2: Example of integration of two systems with similar interfaces (based on the HCPB design)





Common systems for both TBMs **1.1 Structure components of PIC** **1.1.1 Pallet system** **1.1.2 Mechanical support for pipes** **1.1.3 Mechanical support for heat exchangers** **1.1.4 Mechanical support for valves**	**1.2 Bioshield** **1.2.1 Concrete wall (400mm)** **1.2.2 Penetrations for pipes** **1.2.3 Tool access corridor (with door)** **1.3 Interfaces** **1.3.1 Interface 2 (port plug / PIC)** **1.3.2 Interface 3 Helium** **1.3.3 Interface 3 Tritium** **1.3.4 Interface 3 Instrumentation**

Figure 6.2.2-3: List of the main components presented in Fig. 6.2.2-2 (Part 1)



2	**Equipment for upper TBM**
2.1	**Helium Cooling System**
2.1.1	**Helium inlet**
2.1.2	**Helium outlet**
2.1.3	**Helium bypass**
2.1.4	**Heat exchanger / gas mixer**
2.1.5	**Valve (Mass flow control bypass)**
2.2	**Tritium System**
2.2.1	**Purge line inlet**
2.2.2	**Purge line outlet**
2.3	**Measurement system (TM-Module, separate BU supply)**
2.3.1	**Helium inlet (2x)**
2.3.2	**Helium outlet (2x)**
2.3.3	**Container for measurement equipment**
2.4	**Instrumentation**
2.4.1	**Pipe for sensing lines**

Figure 6.2.2-4: List of the main components presented in Fig. 6.2.2-2 (Part2)

6.2.3 Port Cell B

6.2.3.1 Pipe Interfaces

In Test Port #18 (Port B), it is planned to have available one water coolant line and two Helium coolant lines. In order to evaluate the need of space in the port cell for the TBMs systems, it is assumed to have two independent TBM systems, one related to a Water-Cooled Solid Breeder (WCSB) TBM from WSG3 and another one related to a Helium-Cooled Lithium-Lead (HCLL) TBM from WSG2. The possibility of installing a DCLL system or a DFLL system is also briefly discussed. Other possible combinations (see section 6.5) are not addressed here.
In the case of a HCLL TBM and of a WCSB TBM, the specification of pipe interfaces at the boundary of Port Cell B is summarized in **Table 6.2.3-1**. Two He lines, one water line and two He purge lines are requested for pipe interfaces between the port cell and the shaft in Port Cell B.
HCLL TBM applies Helium cooling for the first wall of the module and internal LiPb region. In the case of He cooling, a bypass flow is envisaged for the purpose of adjustment of coolant temperature of internal LiPb regions to DEMO relevant condition. The bypass flow will exits the TBM after the cooling of the FW and will be mixed to the main coolant exit flow inside the port cell. He Cooling System is planned to be installed in TCWS Vault, therefore one He line is requested for HCLL. As the interface condition, two He lines are requested for better flexibility of testing. For tritium extraction from LiPb, LiPb loop in the port cell space. Therefore, there is no LiPb pipe interface at the boundary of the port cell and shaft.

WCSB TBM uses one pair of water line (inlet and outlet) for cooling. The Water Cooling System is planned to be installed in the TCWS vault, therefore, one water line is requested for pipe interface. For tritium extraction of WCSB TBM, He purge gas is introduced into the TBM and bred tritium is carried to the Tritium Recovery System (TRS). For release of tritium generated in Be multiplier pebbles, He purge route will be established and lead to the TRS. TRS is planned to be installed in the Glove boxes in the Tritium Building of ITER. Therefore, two pairs of He purge lines for tritium management of breeder zone and multiplier zone.

6.2.3.2 Space Requirement for Ancillary Systems

Table 6.2.3-2 summarizes the space requirements of ancillary systems of Port B. As described in the above, the HCS for WSG2 and WCS for WSG3 are requested to be installed in the TCWS Vault. As for tritium system, the TRS for WSG3 is requested to be installed in the Tritium Building. For the port cell space, LiPb Loop for tritium extraction of WSG2 and Tritium Measurement System for WSG3 are requested to be installed independently in the port cell space.

6.2.3.3 Conclusion

This section summarized the possible requirements of pipe interfaces and space for allocating proposed WSG2 TBMs and WSG3 TBM for Test Port #18 (Port B). In this test port, it is necessary to have LiPb Loop unique to each type of TBM in the port cell. Therefore, as far as the available space in the port cell is limited to the current design, it is difficult to plan more than two half port modules even by using quarter size port modules.

Table 6.2.3-1: Specification of pipes interfaces at the boundary of the Port Cell-B

Item	No. of pipes	Pipe size (ID/OD) [mm]	Insulation thickness [mm]	Note
He line-1	2	95.4/108	80	From HCLL(DFLL, DCLL)-TBM to He Cooling System located in the TCWS (Inlet / Outlet)
He line-2	2	95.4/108	80	From DFLL, DCLL-TBM to He Cooing System located in the TCWS (Inlet / Outlet)
Water line-1	2	76.2/101.6	80	From WCSB-TBM to WCS located in theTCWS
He purge line-1	2	18.4/25.4	TBD	From WCSB-TBM to TRS in Tritium Building
He purge line-2	2	18.4/25.4	TBD	From WCSB-TBM to TRS in Tritium Building
HRS line	2	25/33.2	50	
CCWS line	2	25/33.2	50	

Table 6.2.3-2: Space Requirements of Ancillary Systems of Port B

(1) Cooling Systems

WSG	TBM	Coolant	System	Foot Print	Location
WSG2	HCLL	He, 8MPa, 300 – 500 °C	All Systems	2.4m x 7.36m x 5m(H)	TCWS Vault
	DFLL		All Systems	3.6m x 6m x 5m(H)	TCWS Vault
	DCLL		All Systems	3m x 7m x 5m(H) or maximum allowable in TCWS Vault	TCWS Vault
WSG3	WCSB	Water, 15MPa, 285 - 323 °C	Main Cooling Loop	20 m^2 x 5m(H)	TCWS Vault
			Turbine Loop (TBD)	5m x 8m x 5m(H)	TBD

(2) Tritium Systems

WSG2	HCLL	LiPb	LiPb circuit for tritium extraction	1.6m x 2.19m x 2.315m	Transporter in Port Cell
	DFLL		LiPb circuit for tritium extraction	6.5m(L) x 1.3m(W) x 2.65m(H)	Transporter in Port Cell
	DCLL		LiPb circuit for tritium extraction	~5 m(L) x ~2 m or half transporter width x ~4 m(H)	Transporter in Port Cell
WSG3	WCSB	He Purge Gas	Tritium Recovery System	GB: 1.2m x 4m x 3m(H) x 2	Tritium Building
			TMS #1	0.5m x 1m x 0.5m(H)	Transporter in Port Cell

6.2.4 Port Cell C

The majority of TBM concepts require the placing of some cooling ancillary systems in Tokamak Cooling Water System (TCWS) Vault, the piping from the Port areas to TCWS Vault to be placed in the Vertical Shafts. In order to provide for the most possible flexibility of TBMs testing in Port C, piping/cooling arrangement (from Port area to TCWS vault) was investigated [**6-1**] for the following combination of cooling/purge lines (**Table 6.2.4-1**):

- three He lines (1 lines for He-cooled TBM and 2 lines for dual-coolant TBM);
- one He pressure relief pipe
- three He purge lines for two TBMs;
- Heat Rejection System (HRS) lines for the total heat load of two TBMs occupying half of the Test Port space each;
- 1 line from Component Cooling Water System (CCWS) for cooling of ancillary systems components of TBMs with demineralized water.

Table 6.2.4-1: Specification of pipes inside the port Port-C cell

Item	No. of pipes	Pipe size (ID/OD) [mm]	Insulation thickness [mm]	Note
He line-1	2	--/88.9	TBD	Dual Coolant TBM (prim.); FW --- TCWS vault
He line-2	2	--/88.9	TBD	Dual Coolant TBM (2nd); Transporter---TCWS vault
He line-3	2	--/108	TBD	He-cooled TBM; TBM --- TCWS vault
He pressure relief line	1	100/110	TBD	He-cooled TBM in port; Transporter---TCWS vault
He purge line-1,2	2	--/10		From transporter to Tritium Plant
He purge line-3	2	--/36		He-cooled TBM; from TBM to Tritium Plant
HRS line	4 or 2	65/76 or ~100/114.3	50	for 2 loops in port cell; Transporter---assembly hall
CCWS line	2	25/33.2	50	for Li-cooled TBM ; Transporter---utility tunnel

Since the insulation thickness is not fixed yet, 300 mm of OD including insulation is assumed for all He-cooling TBM pipes, and 50 mm of insulation thickness for HRS and CCWS is assumed. Only He cooling pipes are routed through the Vertical Shaft. No guard pipe around the TBM pipes for the confinement purpose is assumed for this investigation. Feed and return pipes for the frame (FW/BLK PHTS) are routed inside the cryostat; the He purge pipes do not use the Vertical Shafts and their size is relatively small (~10 mm –36 mm OD). The HRS lines for the components inside the port cell come from the assembly hall, and the pipe routing from gallery area penetrating the space above the port cell door is considered. Space for piping and their configuration are governed by pipes thermal stresses. The pipes are straight in the Vertical Shafts and all the bends are in the Port Cell area and in TCWS vault.

As a conclusion, seven pipes 300 mm in outer diameter (insulation included) may be placed permanently in the Vertical Shaft, their penetrations through the Shaft wall may be plugged from the Port Cell side (see **Figure 6.2.4-1**). Piping on the Port cell side wall (**Figure 6.2.4-2**) is arranged after placing the TBMs to be tested inside the Port and removed after tests are finished. The maximum number of He cooling pipes (four) is needed while testing Dual Coolant TBM and He cooled TBM simultaneously. All other combinations of TBMs require less number of piping to TCWS vault, while Li self-cooled TBM does not require any connections to TCWS Vault.

Such arrangement of piping (dismantling them together with TBMs to be tested) provides enough space in Port cell to reach the front side of the near-port transporter with some ancillary systems for hands-on operations.

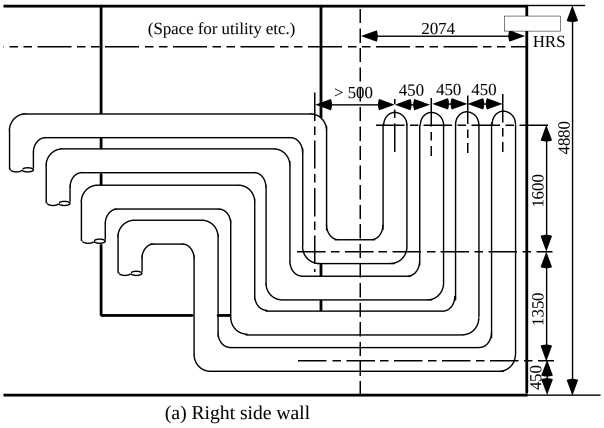


Figure 6.2.4-1: Configuration of penetration at the vertical shaft wall

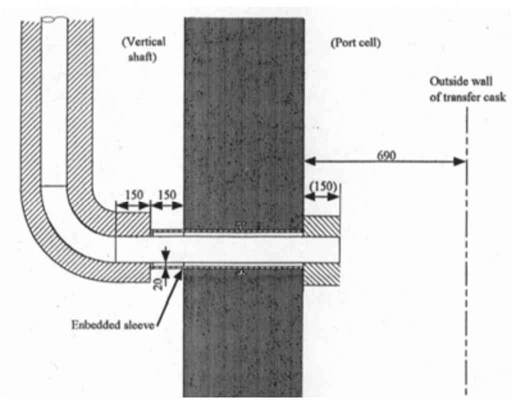


Figure 6.2.4-2: TBM pipes layout on the side wall of the port cell

6.3 Auxiliary Systems from ITER

6.3.1 Heat rejection system

The heat generated in TBMs and in PHTSs system is transferred to the heat rejection system (HRS) provided by ITER. A water circulating system (WCS) is designed for cooling all the TBM systems. The maximum temperature for feed and return in HRS is designed to be 35 ˚C and 75 ˚C, respectively. The flow rate is determined for this temperature condition and the total heat load of all PHTSs is estimated to of the order of 6 MW (to be updated according to TBMs design and operating conditions). The design pressure in HRS is less than 1.0 MPa..

6.3.2 Tritium Systems

The tritium pipes between the port cells and the ITER Tritium Plant would be routed through the gallery area at the level of EL 0.0 m, and an integration strategy for the pipes is to be determined. The interface with the ITER Tritium Plant has yet to be determined in detail. For the time being, space has been reserved at the level of EL 0.0m to install eight glove boxes (including two spare) of the following dimensions; GB: 3m (L) x 1m (W) x 2.7m (H) with pass box: ∅ 0.75m x 0.75m (L). This is believed to be more than will be required, so that in principle some additional equipment could be placed here. It must be stressed, however, that this equipment, under fault condition, must not in any way threaten the integrity of Tritium Plant. This does exclude the installation of any large size pressurized equipment in the Tritium Plant. **Figure 6.3.2-1** shows a plan view of the tentative space allocation for glove boxes, associated with TBMs, in the Tritium Plant. Another interface item is the routing of tritium bearing lines between the TBM ports at the VV (Ports #2, #16 and #18) and the glove boxes at the level of EL 0.0m, which is fixed temporally. This will be undertaken when the requirements have been fully defined by the Parties.

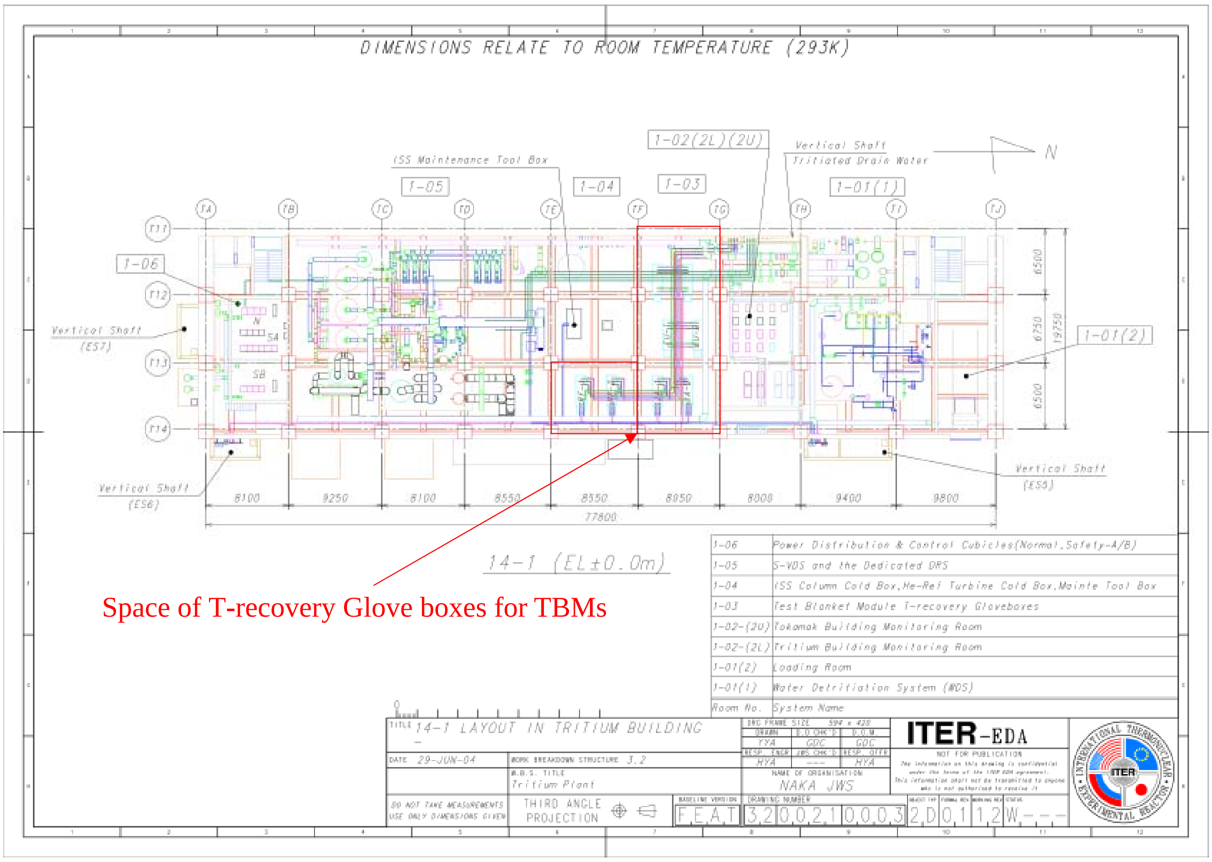


Figure 6.3.2-1: Tentative room space allocation for TBM Tritium Recovery Glove boxes in the ITER Tritium Plant Building

Available floor space: ~ 19.5m^L x ~ 8m^W + ~ 6.5m^L x ~ 8.5m^W
Available clear height: ~ 6.5m including piping route between TBM ports to GBs

6.4 Constraints due to space limitations

As already mentioned in section 5, space constraints will strongly limit the number of independent TBM systems that could be tested at the same time in ITER. These space constraints are analyzed in the following sections.

6.4.1 Space inside the frame (TBM Port Plug)

The number of TBM inside a Port Plug is strongly limited by the interface needs; hydraulic, mechanical, grounding and instrumentation connections to the Port Plug are required. A typical independent TBM will require at lest 2 (3 in almost of the proposed case) connections to the main coolant system. The dimensions of these pipes are in the range of 50-80 mm ID for helium cooled systems. Additional pipes are required for the purge of the CB and Be beds or for the re-circulation of the liquid breeder: at least 2 pipes of 20-30 mm ID. The mechanical attachment should be robust to cope with the EM mechanical load as consequence of major disruptions. Typical ITER designs are adopted that make use of flexible and shear keys. Also if more compact systems can be envisages especially for modules of reduced dimensions, it is clear that all these systems require large space. Present studies [**6-3**] suggest that the integration of more than 2 independent modules per Port is questionable.
The interface design between TBM and Port Plug has not yet been defined in detail. Previous integration systems (see § 4.3) ware based on the use of in-bore tools for the handling of the interface in the hot cell from the rear side of the blanket. Recently a new proposal of the JCT envisages the

possibility that a part of the Port Plug shielding will be connected to the TBM, moving the interface TBM-ITER inside the shielding region; this proposal is discussed in section 2.6.2.

This concept can give more flexibility, allowing some changes in the interface lay-out. However, it is difficult today to give a definitive conclusion about the compatibility with the TBM programme. This design alternative will be investigated in the next time.

6.4.2 Space in TCWS vault

According to the proposed lay-out of the 3 Port Cells, coolant systems should be foreseen in TCWS to feed the lines foreseen for each Port Cell. Taking into account the different strategy proposed by the party, the space requirement in the TCWS vault increases according to **Table 6.4.2-1**.

Table 6.4.2-1: Potential space requirement in the TCWS vault for TBM ancillary circuits expected in ITER according to the available information from the parties (sub-modules not included)

	Type of coolant systems	Footprint	Height	Note
1	HCSB System	25 m^2 *	5 m	Based on the EU design
2	HCSB System	27 m^2	5 m	Based on the JA design
3	HCSB System	~ 13 m^2	5 m	Based on the RF design
4	HCSB System	18 m^2	5 m	Based on the CH design
5	WCSB System	20 m^2	5 m	Based on the JA design
6	HCLL System	25 m^2 *	5 m	Based on the EU design
7	DCLL – HC System	20.1 m^2	5 m	Based on the US design
8	DFLL – HC System	21.6 m^2	5 m	Based on the CH design
9	HCML System	> 20 m^2	5 m	Based on the KOR design

* including the Coolant Purification System but not the Pressure Control System (with Helium storage tanks).

In the past a dimensioning of these systems has been done on the basis of the following assumptions: an area of a footprint of 16.6 m x 7.3 m and a high of 5m is available for the integration of the systems in the TCWS (space for access should be integrated as well). This place was considered for the location of four independent Coolant systems (2 helium and 2 water systems of the capacity of circa 1MW heat reaction each). In respect of the situation in 2001, the space for the installation of the components of the HCS is limited in vertical direction by pipes of the Heat Rejection System (HRS) and there are additional space restrictions caused by the requirement to have an emergency escape at the side walls of the Vault which surround the position of the EU-He-Room (as in the 2001 report). In particular the envisaged possibility to use a crane for the assembly and maintenance operations of the Coolant Systems is precluded by the presence of the HRS piping (**Figure 6.4.2-1**). This produced further constrains in the design of the HCS that have to provide other way for the assembly and maintenance of the components.

Considering the decreased space available in the TCWS and the increase of space request for additional Cooling systems (also if it should be mentioned that some of these systems described in **Table 6.4.2-1** can be shared among the parties because only a limited number of TBM can be tested simultaneously), the place available in the TCWS is largely insufficient for the needs of the Blanket testing programme.

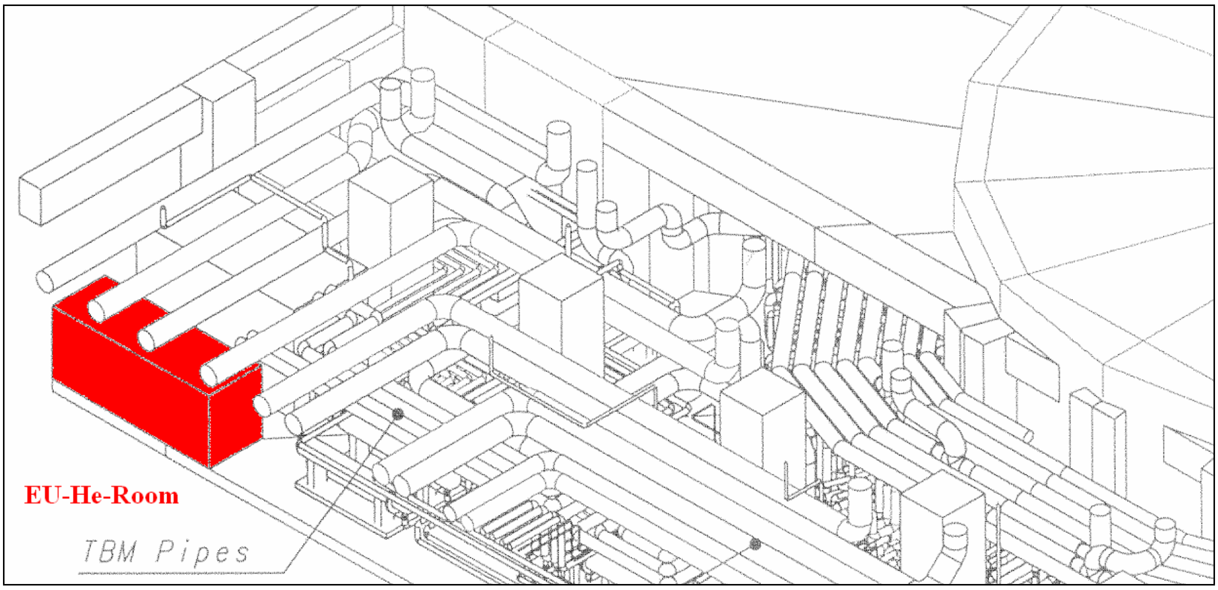


Figure 6.4.2-1: Present arrangement of the place in the TCWS vault based on the 2001 position for the EU loops

A proposal to overcome this issue could be to investigate the possibility to build only one large Helium Loop to feed all the HC-TBMs that are in operation at the same time. It has to be checked if this solution is still compatible with the respective testing planes. The availability of additional place for auxiliary equipments (like the storage tanks) should be checked, as well.

6.4.3 Space in Port Cells

Most of the TBM systems proposed by the ITER Parties require the location of equipment in the Port Cell both for special measurement stations and for entire or part of the primary cooling loops that cannot find place in the TCWS.

Taking in account the study shows in section 6.6.2, the space available in the port cell is critical. Most of the space will be used for routing of the main Helium pipes. The limitation of space in the Port Cell (in addition to the constraints in the TWCS) is one of the principal reasons for the limitation of "independent" TBM systems that can be tested at the same time in one horizontal Port.

Furthermore, additional space should be provided in the ITER building to park the Port Cell equipment during the maintenance operation. E.G. the PIC should be entirely removed from the Port Cell to allow the docking of the RH transporter for replacement of the Port Plug. This point has to be addressed in the definition of the TBM System – ITER interface.

Table 6.4.3-1: Potential space requirements in the TBM testing Port Cells expected in ITER according to the available information from the parties.

Type	System in Port Cell	Dimensions	TBM size
EU - HCPB	By-pass control * (1 valve and 1 thermal sleeve)	0.5 m (h) x 2.0 m (b) x 2 m (l)	½ port TBM
	Neutron & Tritium Measurement System, Helium Conditioning System**	1.2 m (h) x 0.8 m (b) x 4 m (l)	½ port TBM
JA – SBHL	By-pass mixing heat exchanger	0.8 x 2.2 x 0.5	One for all sub-modules, For all the testing phases
RF – CHC	He to water heat transfer system	6.5(L) x 1.8(W) x 2.65(H)	½ port TBM
CH – HCSB	He to water heat transfer system	4(L) x 1.2(W) x 4(H)	¼ port TBM
US-SBHL	Neutron Measurement System Tritium Measurement System Helium Conditioning System **	TBD	TBM submodule
	By-pass control * (1 valve and 1 thermal sleeve)	TBD	TBM submodule
KOR – HCSB	Neutron & Tritium Measurement System	TBD	TBM sub module
EU – HCLL	PbLi loop	1.6m (W) x 2.19m (L) x 2.315m (H)	½ port TBM
	By-pass control * (1 valve and 1 thermal sleeve)	0.5 m (h) x 2.0 m (b) x 2 m (l)	½ port TBM
JA – SBWC	TBD	TBD	½ port TBM
RF – Li/V	Heat transfer system (intermediate to heat rejection) and inert gas supply system	6.5(L) x 0.8(W) x 2.65(H)	½ port TBM (to be shared with JA?)
KOR - HCML	He to water heat transfer system and Li circuit	TBD	½ port TBM
US-DCLL	Liquid breeder to He heat transfer system	6.5(L) x 1.3(W) x 2.65(H)	½ port TBM
CHN-DFLL	LiPb ancillary circuit and detritiation unit	6.5(L) x 1.3(W) x 2.65(H)	½ port TBM

* Part of the primary HCS
** System connected to the primary HCS

6.4.4 Space in the Tritium Building

According to the present proposals several independent TESs (that requires 1-2 glove boxes each) will be located in the Tritium Building. Up to now the integration of these systems has not be analysed in detail.

6.4.5 Space in Hot Cell to Refurbish the TBM Port Plugs

The flexibility of the testing programme will be also limited by the number of TBM exchange that is allowed for a TBM. At the present it has been assumed that the TBMs can be exchanged every years during the scheduled maintenance time, i.e. one TBM can be exchanged 1 time per year and the operation requires 1 month. This statement assumes implicitly that ITER has the capability to refurbish the 3 TBM Port Plugs during the month of scheduled maintenances and this is still compatible with the other refurbishment operations that could be performed in Hot Cell. This assumption should be checked; also if the space available in the Hot cell seems largely insufficient for these type of maintenance. Enlargement of the Hot Cell should be considered to allow this kind of operation. Other maintenance strategies can be proposed to accelerate the replacement. A way to solve this problem can be to replace the old port plugs with new ones, already prepared in the hot cell. This maintenance schema can reduce the replacement time of each TBM Port Plug, as it doesn't require hot cell operation in the critical time. The Hot Cell design should be checked for this type of operations. More details on the Hot Cell operations and requirements are given in Chapter 7.

6.5 Examples of TBMs installation plans in the initial H-H phase

Parties have proposed several TBMs for testing since the first day of ITER H-H operations. In order to illustrate the flexibility allowed by the agreed testing strategy, taking into account the constraints of pipes connections in each port and the space availability in port cells and TWCS vault, some example of port allocation are given in **Figure 6.5-1**. With "one line" it is meant "two pipes", one inlet pipe and one outlet pipe.

As shown in the previous chapters, it is recalled that the candidate TBMs for installation on ITER day_one, at present, are the following:

- He-Cooled Ceramic Breeder (HCCB) TBM: there are 4 TBMs proposed, HHCB-1 to HCCB-4, all of them requiring one He-line; it is assumed that HCCB-1 needs horizontally divided half-port, HCCB-2 and HCCB-3 can accept either horizontal or vertical half-port, HCCB-4 needs vertical half-port. Proposed TBMs with size of ¼ port are included in the half-port size envelop;
- He-Cooled Lithium-Lead (HCLL) TBM: there is one proposal that requires one He-line and a vertical half port; LiPb circuit components are located in the port cell;
- Dual-Coolant Lithium-Lead (DCLL) TBM: there is one proposal that requires two He-lines and a vertical half port; LiPb circuit components are located in the port cell;
- Dual-Functional Lithium-Lead (DFLL) TBM: there is one proposal that requires two He-lines (when operating as DCLL) and a vertical half port; LiPb circuit components are located in the port cell;
- Water-Cooled Ceramic Breeder (WCCB) TBM: there is one proposal that requires one water-line and a vertical half port;
- He-Cooled Lithium (HCLi) TBM: there is one proposal that requires one He-line and a vertical half port; Li circuit components are located in the port cell;
- Self-Cooled Lithium (SCLi) TBM: there is one proposal that does not require coolant line connections because all Li circuit components (including heat-exchanger) are located in the port cell.

Because of the constraints of vertical vs. horizontal TBM geometry and of number and type of coolant lines connections requirement, not all combinations among the proposed TBMs are possible. The 5 examples shown in Figure 6.5-1 take into account these constraints. In order to keep a maximum of flexibility and not to anticipate future selection, it may be noticed that all TBMs are present in 3 out of the 5 cases.

Considering the uncertainties present today on each TBM development, as discussed in the previous chapters, the risk that one or more TBMs will not be ready on time for installation in ITER on day_one is relatively high. Therefore, the flexibility illustrated by Figure 6.5-1 allow to better guarantee ITER that a sufficient number of TBMs will be delivered on-time for filling the three test ports.

However, in order to be sure that ITER can start operation without delay, it is likely that dummy TBMs, designed and fabricated as the main ITER shielding blanket, may be anyway required.

	Port Nb. 16 *(2 He-lines)*	**Port Nb. 18** *(1 H_2O & 2 He-lines)*	**Port Nb. 2** *(3 He-lines)*
Case 1	**HCCB-1** **HCCB-2**	**WCCB** **HCLL**	**SCLi** **DCLL**
Case 2	**HCCB-1** **HCCB-3**	**WCCB** **DFLL**	**HCLi** **DCLL**
Case 3	**HCCB-1** **HCCB-2**	**WCCB** **HCLL**	**HCCB-4** **DFLL**
Case 4	**HCCB-2** **HCCB-3**	**HCLi** **HCCB-4**	**SCLi** **DFLL**
Case 5	**HCCB-3** **HCCB-4**	**SCLi** **HCLL**	**HCLi** **DCLL**

Figure 6.5-1: Example of possible ITER test port allocation on day_one assuming half-port size TBM (1/4 port size TBMs can be seen as sub-modules)

6.6 Conclusions and recommendations

This chapter indicates that the present Parties TBM test program cannot be entirely performed in ITER and that priorities have to be agreed by Parties either through collaborations leading to common test programs or through detailed technical assessment of benefit from common tests.

As expected, space availability is the main constraint. An important conclusion made by the TBWG is that the space availability in ITER test ports, and the consequent test port allocation, is not the only space availability constraint. In fact, the space availability in port cells, vertical shaft and TWCS vault put even more restrictive constraints.

For instance, typical TBM size has been fixed to half a port leading to the possible simultaneous test of 6 TBMs (2 in each port). Arrangements of a larger number of TBMs in each port frame (3 or even 4) could be found but it appears clear that 4 independent TBMs systems (coolant + tritium recovery system) cannot be located neither in the corresponding port cell, neither in the TWCS vault (if TBMs are present in the other 2 ports). Moreover, no sufficient space is available for pipes connection in the vertical shaft, and too many pipes would have to cross the test port shield.

Therefore, if one wants to test more than 2 TBMs in each port, it is required to agree to share, at least partially, the cooling systems and probably the Tritium extraction system. In any case, strict collaboration on TBM test program between Parties is required.

Time sharing may improve the situation, however the possibility are limited by the complex operation required by TBM replacement which can only be performed once a year. Moreover, testing priorities have in any case be established and agreed and some partial temporal sharing of the same TBMs systems is desirable.

The agreed flexibility on the choice of TBM port allocation leaves sometime for launching R&D collaboration among Parties that is highly recommended. This collaboration could be facilitated by the R&D results which will become available in the near future (which could lead reduce confidence or increase on some the present TBM proposal).

A further recommendation is addressed to the ITER Team and concerns the space availability in the test port cell and in the ITER TWCS vault. The possibility of increasing the room available in these two crucial regions for TBM testing should be assessed in details. As far as the TWCS vault is concerned, the accessibility of the TBMs systems located there should be guaranteed and system components replacement procedure should be established in details as soon as possible.

6.7 References

[6-1] Y. Kataoka, “Preliminary layout of TBM piping inside the port cell and the vertical shaft”, Report N 26 MD 49 04-09-29 W 0.1, JWS Naka, Japan, 2004.

[6-2] H. Neuberger, L.V. Boccaccini, R. Meyder, “Integration of the EU HCPB Test Blanket Module System in ITER”, presented to the ISFNT-7 in Tokyo, May 2005.

[6-3] L.V. Boccaccini, J.-F. Salavy, R. Lässer, A. Li Puma, R. Meyder, H. Neuberger, Y. Poitevin, G. Rampal, “The European Test Blanket Module Systems: Design and Integration in ITER”, invited paper to ISFNT-7, Tokyo, May 2005.

7 - REMOTE HANDLING & HOT CELL ISSUES

7.1 Remote Handling Tools

The RH tools required during TB plug replacement are basically the same as those utilized for other ITER equatorial plugs replacement. These consist of the following:

a. Transfer cask (equator port type cask, **Figures 7.1-1, 7.1-3, 7.1-4**) payload capacity : 48 ton,
b. Cask internal handling equipment (plug handling tractor) (**Figure 7.1-2**).

Additional, hands-on tools will be required for the disconnection of all pipes and cables from the rear of the TB plug and for the removal of the TB equipment located inside the port cell.

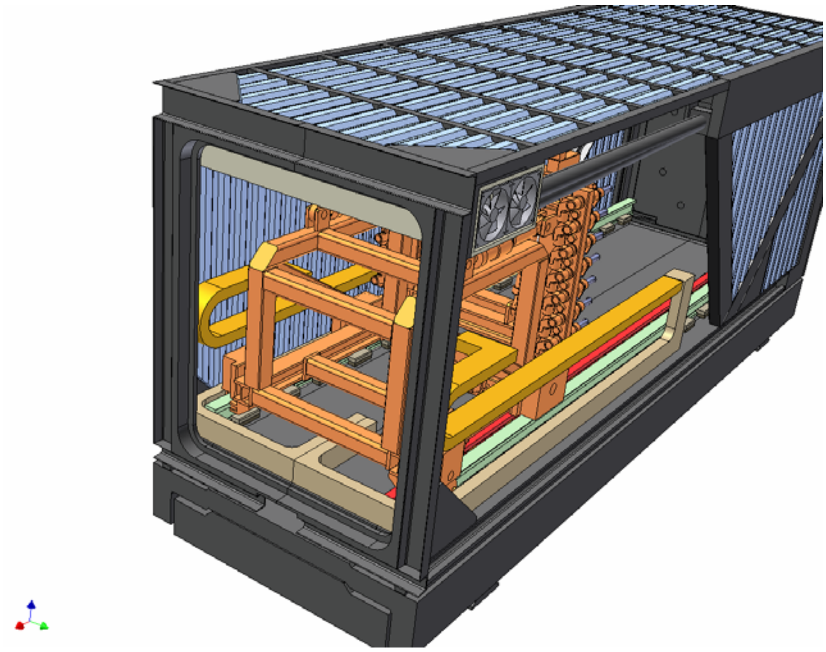


Figure 7.1-1: Equator port plug cask (rear view) showing plug handling tractor

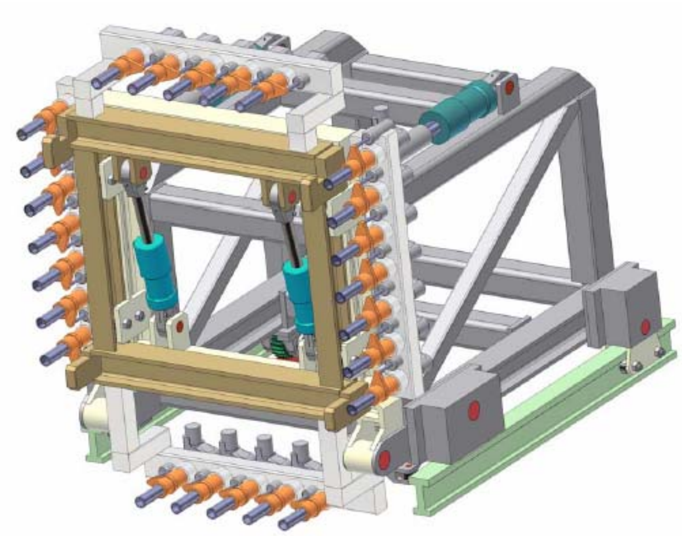


Figure 7.1-2: Equator port plug handling tractor (concept)

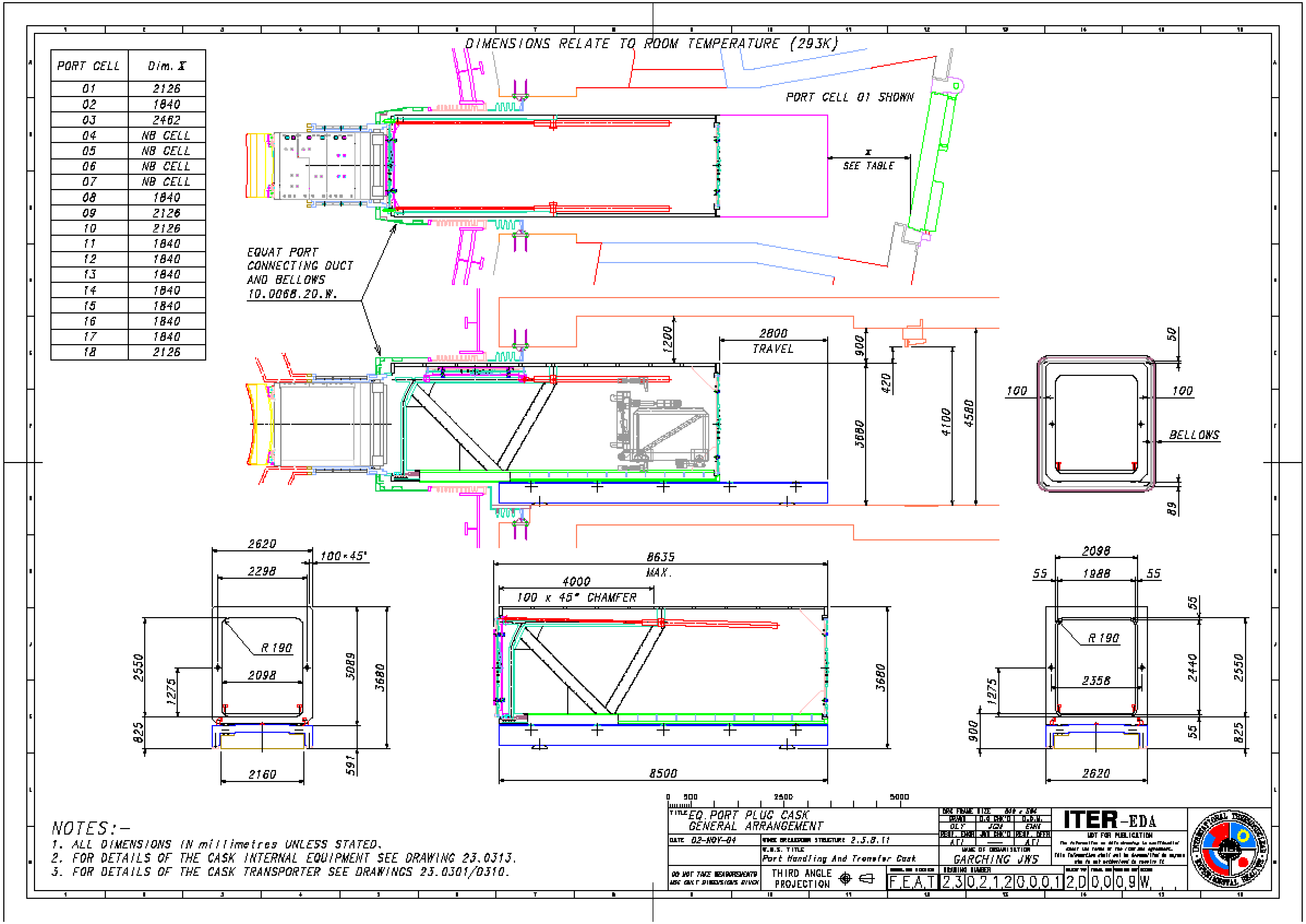


Figure 7.1-3: Equator port plug handling cask design

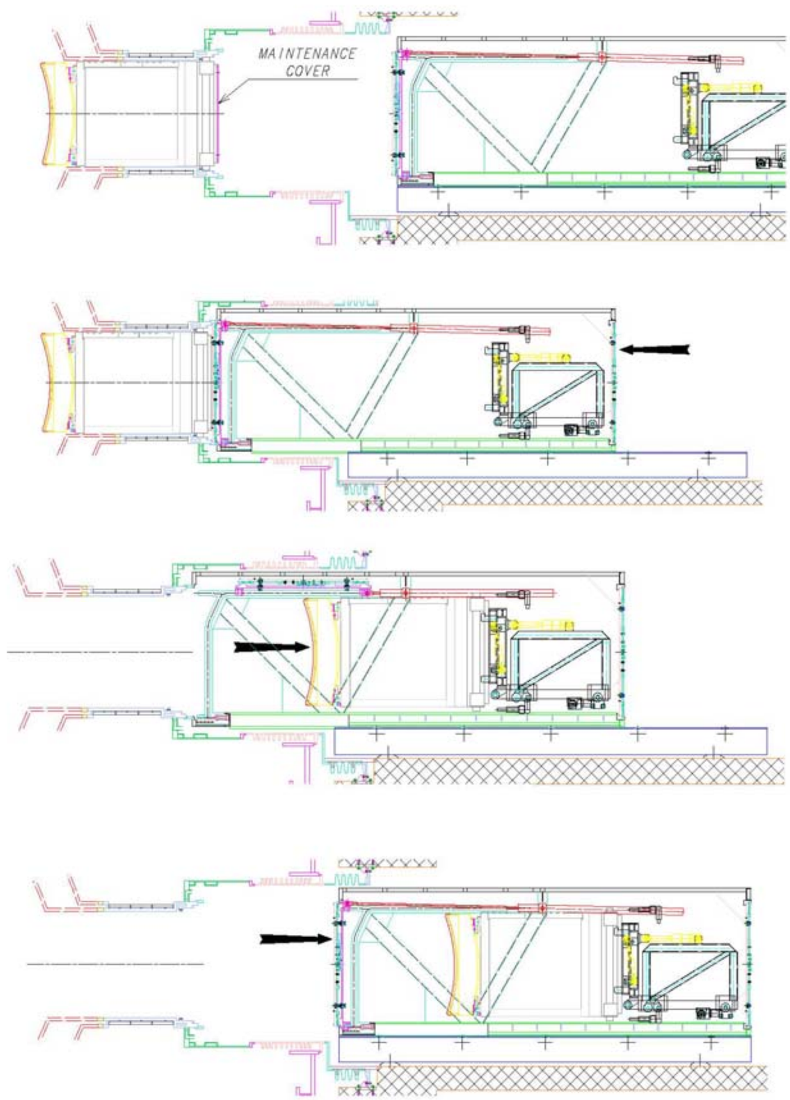


Figure 7.1-4: Typical equator port plug removal sequence

7.2 Remote Handling Operations

Replacement of TB plugs is a RH class 1 operation, i.e. a regular operation that must be fully planned in advance. Because it is a planned operation that occurs several times during the lifetime of the ITER machine, the impact on the operational program must be minimized. The design of TB modules and the frame must therefore be fully compatible with the remote replacement operations. The procedures as well as the associated hands-on and RH equipment must be designed and demonstrated prior to start of operations.

The basic sequence required for TB plug removal is outlined below:

a. the port cell is cleared of all TB systems and related equipment using hands-on (note: temporary parking space inside the Building for TBM service cask has yet to be identified)
b. removal of any bioshield structure located in the Port Cell
c. procedures and tools (to be defined by the TB plug responsible officer-RO)
d. if required, the TB plug is drained of its fluids (liquid metal)
e. the vacuum lip seal is cut and removed
f. part of the mechanical attachment flange is removed
g. the above operations are carried out in a radiation environment < 100 μSv/h.
h. the RH cask is maneuvered remotely inside the port cell and docked to the VV port flange
i. the TB plug is completely disconnected from the VV flange and lifted inside the transfer cask
j. the transfer cask double seal door is closed, leaving on the VV port a maintenance door (for temporary sealing of the VV) and the transfer cask with the TB plug travels to the HC
k. the transfer cask docks to the HC and the TB plug is delivered inside the HC refurbishment area.

After replacement of the TB module in the HC, the procedure is repeated in reverse order, replacing cutting by re-welding and testing. After an overall pressure and leak test, the hydraulic lines are evacuated and re-filled with coolant, and the purge lines reconnected to the purge stream.

7.3 Operations in Hot Cell

The ITER HC design has been revisited in 2004-05 and it is undergoing an internal review prior to final approval. A plan of the ground floor (including the areas where the ITER in-VV components refurbishment is processed for repair and/or disposal) is shown in **Figure 7.3-1**.

There is no major difference in the approach to generic maintenance operations carried out inside the HC on the TB plugs and on other equatorial plugs. These are: minor repairs, sub-components replacement (provided that these are not too difficult to reach and replace remotely) and testing. Any more complex operations need to be assessed for feasibility. It is the TB's RO responsibility to verify ahead of the TB plug installation and operation on the ITER machine that any required RH activity on the TB plug inside the HC can indeed be carried out in terms of technical feasibility, tooling and general support equipment, including radwaste processing and disposal.

To date, the HC operations foreseen for the TB plug consist in the dismounting and re- mounting of the front TB shield module from plug, to include:

- cut/welding/inspection (*) of hydraulic pipes
- disconnection and re-connection of mechanical/hydraulic joints (**)
- disconnection and re-connection of grounding and diagnostic cables (**)

– small repairs (tbc) (**)

(*) testing procedures and test values tbc
(**) techniques and procedures yet to be evaluated from a RH operations standpoint

All the above operations must be assessed for remote feasibility and demonstrated in hands-on fashion before they are included into the ITER HC work flow. It should be stressed that the possibility to maintain the TBM remotely must be taken into account at the TBM's design and manufacturing stage. For example, the TBM and backside shield may need to be delivered pre-assembled, tested and ready for installation into the plug frame, inside the HC.

The tight TBM's turnaround time required during a machine shutdown may eventually require the procurement of six TBM plugs, with three complete frames and TBM assemblies ready for machine installation and three frames being refurbished (off-line) inside the HC. Spare plug(s) may also be needed in case a TBM plug is not ready for installation on the machine.
The above considerations may have an impact on the HC ability to store TBM plugs. The current HC storage capacity is for threes TBM plugs.

At present, the HC features neither the facilities nor the procedures for liquid metal draining from the TBM. In this respect, requirements, technical aspects and solutions will be provided and discussed with the IT in due course. After replacement, each TB plug will not be returned to the Party but be disposed as radwaste. If return is required, packaging, transport and waste disposal licensing have to be established by the Party in agreement with the ITER Host Organization and with receiving Party nuclear authorities.

TBM's Post Irradiation Examination (PIE) is an important part of the TB testing program (i.e: Tritium inventory for reconstruction of Tritium balance, etc.). However, the current ITER HC does not include any PIE facility. TBM components' PIE samples cutting, preparation for inspection and inspection facilities are currently outside the scope of the ITER HC building and equipment supply. If possible, easy to remove, pre-manufactured PIE samples (coupons) should be used. If this is not possible or sufficient, a detailed information on the PIE operations, equipment, space and facilities required, testing requirements and procedures will need to be presented and discussed with ITER.

Based on the current knowledge and assumptions, the following TB plug related operations will/will not take place inside the HC:

- TB plugs and modules in excess of those foreseen for installation onto the ITER machine (3 plugs with relevant TB modules) will not be stored, for storage space reasons, inside the HC
- if not re-used, TB plugs will be processed and disposed as radwaste (a detailed material list breakdown including amount and material type is required)
- TB plug and relevant sub-systems metrology will not be carried out, except for external TB plug dimensions, using existing HC measuring equipment
- Tritium recovery from the TBM's will be carried out, most likely after coarse cutting of the plug and modules
- PbLi (or other liquid metals) and Be recovery will not be attempted (but may eventually be necessary, prior to TB plug cutting for radwaste disposal, techniques to be provided)
- In-HC PIE is not currently planned (except on small test pieces if easily removable, but space has yet to be allocated, if available)

- any information on space requirements, existing and additional equipment required in relation to TB plug and module handling, maintenance, PIE & disposal needs to be provided by the relevant TBM supply Party

Prior to acceptance of a TB plug inside the ITER HC, the TB's RO is expected to prepare and submit a set of technical documents. These are:

- the Plant Data Description (PDD) document. It provides a technical description of the system, including drawings, physical properties, size, material, radiation conditions, mechanical, electrical, hydraulic interfaces as well as all the foreseeable RH maintenance operations (see sample in **Figure 7.3-2**, three sheets)
- the RH Task Data Sheet which describes in detail which operations should be carried out inside the HC (based on the available space and equipment) (see sample in **Figure 7.3-3**, refers to another component, the Limiter Plug)
- the Operation Sequence Description Document (see sample in **Figure 7.3-4**, case of limiter plug)

Such documents will allow the HC operators to evaluate if and how a TB plug can be accepted for refurbishment inside the HC. Also, to establish which of the available HC equipment (or any new equipment) needs to be deployed and what procedures need to be implemented, including acceptance testing and criteria for submission of the refurbished TB module prior to re-installation in the VV.

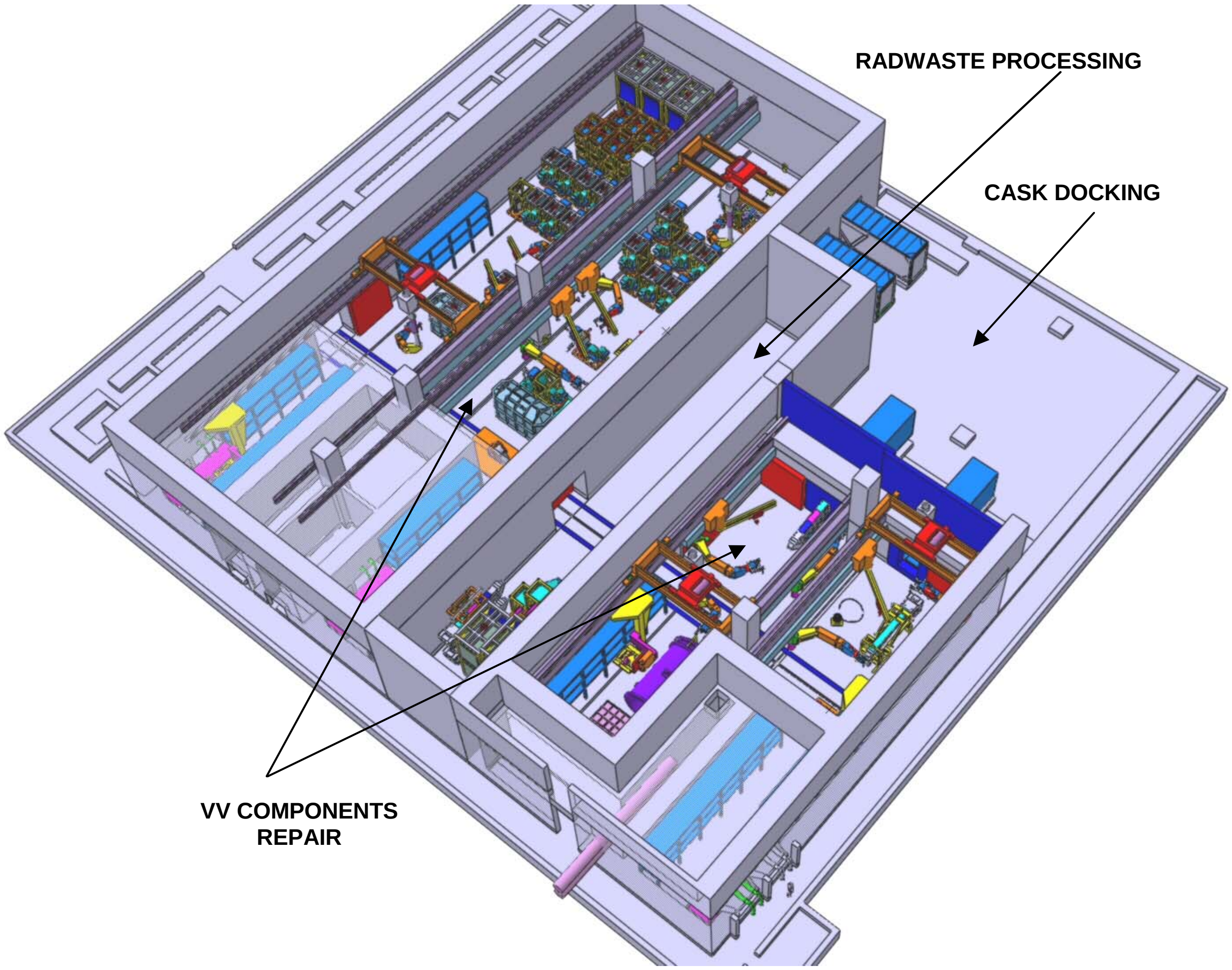


Figure 7.3-1: Plan view of the ITER Hot Cell (ground floor)
(based on January 2005 design proposal)

EFET EWIV

European Fusion Engineering and Technology EWIV

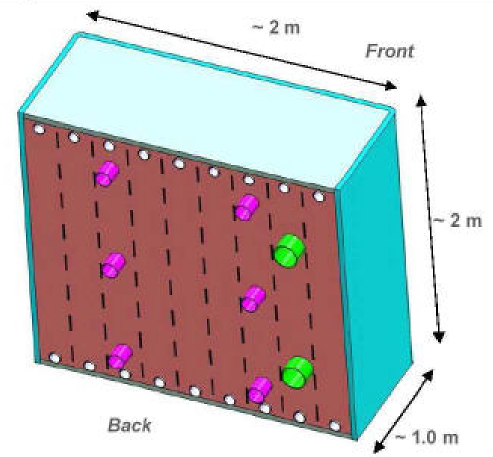
IDENT.	ITR-300-PDF-100-A
Rev.: 004	Date: 21-May-2004
Author: R.Gottfried	Affiliation: Framatome ANP

REMOTE HANDLING PLANT DATA SHEET 1 - DESCRIPTION

Plant Item: Test Blanket Module:

Description:
There will be 3 Test Blanket modules facing the plasma, mounted on 3 equatorial port plugs, recessed by 50mm from the other blanket modules. Each of these 3 Test Blanket modules will be designed differently, probably one by Russia, one by Japan and one by the EU.
The 3 Test Blanket modules serve as prototypes for a future breeding blanket, providing an experimental platform for the first wall blanket module design needed for a commercial fusion reactor. In particular the aim is to demonstrate the breeding of Tritium from Lithium. The Lithium might be in solid form (ceramic pebbles or powder) or in liquid form (with lead).
A Test Blanket Module consists of a stainless steel box, which acts as the outer shell and contains many smaller cells. Helium gas at 80 bar acts as a coolant. Each cell has a He pipe coming in and going out. In the proposed EU design (HCLL), liquid PbLi will be pumped into the top most cell, meanders down a cascade of inner plates and is pumped out of the bottom cell.
Test Blanket modules are classified as RH class 1 (→ changed annually). They are not refurbished, but replaced. To achieve this the entire equatorial port plug is inserted into the port plug test tank, where the TBM is removed. The TBMs will then be posted to and further studied in a dedicated Post Irradiation Experiment (PIE) Hot Cell.
The mechanical attachment of the TBM to the plug structure is likely to be similar to that of the generic equatorial port plug, i.e. via multi-link type pins.

General Assembly:



Page: 1 of 3

EFET EWIV

European Fusion Engineering and Technology EWIV

IDENT.	ITR-300-PDF-100-A
Rev.: 004	Date: 21-May-2004
Author: R.Gottfried	Affiliation: Framatome ANP

Test diagnostic and instrumentation	(tbd)
Final Inspection	Visual and dimensional

Residual Active Components (Rad-Waste):

Description	• various (3) different types of Test blanket modules, all containing Lithium in some form
Mass (kg)	• > 1 ton
Size (m)	• 2.1 x 1.7 x 0.7
Material(s)	• StSt → housing • Lithium → breeder material • Be → PFW tiles
Contact Radiation dose rate	• 400 (Gy/h) after 12 days

Sources of Data:
Antonio Cardella
EFET/TR/NR179 GE
HCLL Demo Blanket, Y. Poitevin, 8 July 2003
Drawings 23.0278.0001 to .0010 (14 June 2001)
Drawings 23.0279.0001 to .0010 (14 June 2001)

Level of confidence in the source data:
High. However, as yet undecided questions are marked: tbd (to be defined)

Revision History:

001	Bernhard Haist	12-01-04	First Draft
002	Bernhard Haist	16-01-04	Input from J. Palmer
003	R. Gottfried	08-04-04	Masses changed
004	R. Gottfried	21-05-04	Sources of Data completed

Page: 3 of 3

Figure 7.3-2: PDD Document sample for TBM system (information recorded here are not up-to-date)

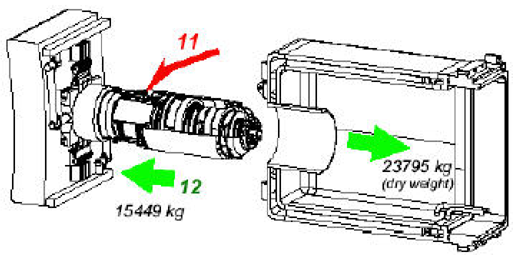
EFET EWIV European Fusion Engineering and Technology EWIV	IDENT.	ITR-300-TDF-103
	Rev.: 000	Date: 07-April-2004
	Author: Roland Gottfried	Affiliation: Framatome ANP

REMOTE HANDLING TASK DATA SHEET 1 – TASK OVERVIEW

Task : Port Limiter Plug: **Removal of the Limiter Module with Container and Adjustable Supporting Units (ASU)**
Reference procedure

Task Objective: Removal of the Limiter Module with Container and ASU
Target plant: • ITR-300-PDF-0101 Port Limiter Plug
Start Point: • The Port Plug is received from the cask, installed at the Port Plug Repair and Test area
End Point: • The Limiter Module with container and ASU is replaced
Assumptions:
Main Issues:

Removal of the Limiter Module with Container and ASU



Page: 1 of 4 COMMERCIAL-IN-CONFIDENCE

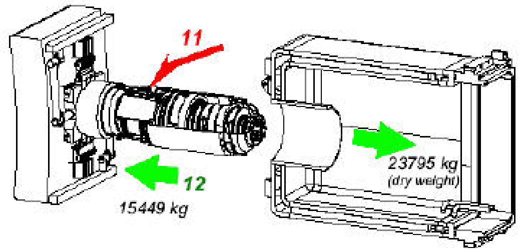
Figure 7.3-3: RH Task Data Sheet Document (SAMPLE, case of limiter plug)

EFET EWIV European Fusion Engineering and Technology EWIV	IDENT.	ITR-300-OSD-103-A
	Rev.: 001	Date: 14-Sep-2004
	Author: Bernhard Haist	Affiliation: Oxford Technologies

OPERATIONS SEQUENCE DESCRIPTION

Task : replacement of Limiter Module (PFW) of Port Limiter Plug including the container and adjustable support units

Sequence Description
Task Objective: • Replace the entire Limiter Module including the container and the ASU (Adjustable Support Unit)
Plant: • ITR-300-PDF-101 Port Limiter Plug
Assumptions: • Port Limiter module clamp incorporates alignment features
Start Condition: • Port Limiter Plug is flanged into its adapter plate at the cask docking port and protrudes into the port plug refurbishment cell
Remote Operations Sequence:



1. Manipulator: cleans plasma facing components with vacuum brush (ITR-300-OSD-101)
2. Manipulator: inspects and measures structure (ITR-300-OSD-001 Visual Inspection and possibly ITR-300-OSD-002 Dimensional Inspection)
3. Manipulator: attaches swarf collection tray
4. Manipulator: installs clamp to Limiter Module to avoid movement of PFW versus structure
5. Main crane: supports clamp
6. Manipulator: deploys cutting tool
7. Cutting tool: removes weld no. 11 of the outer bellow (from backside)
8. Main crane: removes old Limiter Module with container and ASU and disposes of it (ITR-300-OSD-007 Disposal of Radwaste)

Page: 1 of 2 COMMERCIAL-IN-CONFIDENCE

Figure 7.2-4: Operation Sequence Description Document (SAMPLE, case of limiter plug)

7.4 Post-Irradiation Examination Needs

Parties believe that the post-irradiation examination (PIE) of the test blanket module (TBM) is an important part of the testing program. This kind of investigation will be mostly a complementary part of experiments of ITER testing. It provides additional information to validate TBM performance, optimize operating parameters, assess the condition of the TBM, evaluate the root causes of unanticipated failures (if any), reconstruct the tritium balance in conjunction with the results of direct measurements of T in the TBM materials, etc. In particular, with progressive testing of different TBMs during different operational phases of ITER, it is essential to perform PIE to gather information on TBM performance so that design and fabrication procedures can be modified to achieve greater reliability during succeeding ITER operational phases, and to project possible performance for DEMO.

It is commonly assumed that transportation of an activated TBM back to the country of origin will be extremely difficult and expensive. It has been recommended that an Annex be included to the ITER Hot Cell for TBM PIE and be located close to ITER HCB or near ITER. It is also assumed that facilities to perform detailed PIE on miniature specimens or on small sections of a TBM may not be needed at the ITER site, since each Party already maintains extensive facilities for specimen preparation, examination and testing. To avoid the expense and difficulty of shipping large radioactive components, the main function of the TBM PIE Hot Cell should be to perform optical and other nondestructive examinations, extraction of specimen cassettes, and sectioning the TBM into smaller, more manageable pieces and preparation of samples for shipment. Although it has been argued that a detailed PIE facility may already be available near the ITER site, in such a case an international PIE facility can be arranged, which minimizes transportation of any radioactive items. The TBWG has highlighted that such an enterprise can be a collaborative effort between the parties and encouraged a prior agreement and official framework concerning the items of intellectual property to be established.

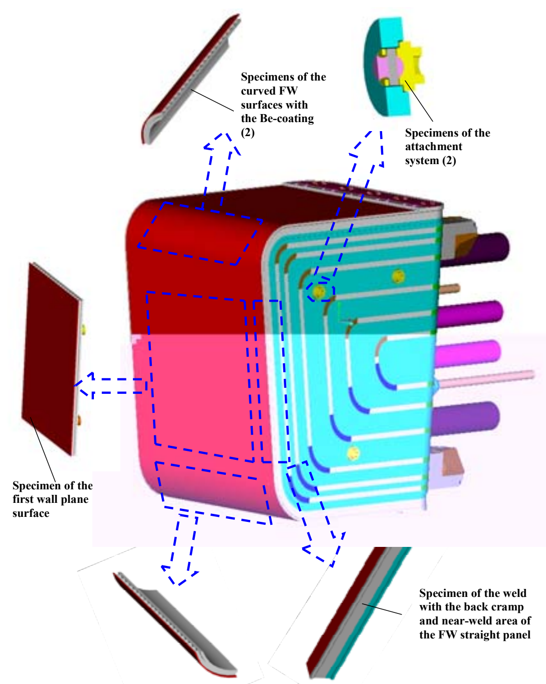


Figure 7.4-1: Example specimens to be cut-out from a TBM

After a period (to be determined) of storage in which the decay heat is reduced to a level to be determined, the PIE process begins. However, during the time that the decay heat is still substantial, it is necessary to monitor and record any released tritium from various parts of TBM to allow reconstructing the tritium balance. The PIE starts with a preliminary visual inspection of the TBM box and a metrology for global box deformation evaluation, which then proceeds to sectioning and extracting specimen from the TBM. It is desirable to know in advance what parts/subassembly of a TBM would be sectioned to recover for more detailed PIE and to allow easy specimen preparation. Example subassemblies/ parts that will be retrieved after ITER testing are illustrated in **Figures 7.4-1** and **7.4-2** and listed in **Table 7.4-1**. A combined description of PIE needs is illustrated in **Table 7.4-2**. Additionally, specimen cassettes that contain miniaturized test specimens for determining the integrated effects of the ITER environment on TBM materials can be designed into the test blanket modules, which can be readily removed without any cutting.

To recover any PIE specimens/parts, the TBM PIE Hot Cell needs to be equipped with:

1) Equipment and glove box tritium measurement facilities to recover and measure any potential tritium release while the TBM is still hot.
2) Remote manipulators for handling the TBM, sections removed from the TBM, and specimen cassettes.
3) Remote visual inspection and recording to examine and record the condition of the TBM and sections of a TBM. Such equipment is essential for guiding subsequent cutting operations.
4) Ultrasonic and eddy current nondestructive inspection of the TBM and sections removed from the TBM.
5) Metrology to characterize dimensional changes with an accuracy of ±1 mm (the structural analysis shows that a toroidal width of 1 m TBM will deform ~7 mm under ITER operating conditions).
6) Equipments (to be determined) to remove and store any residual liquid metals from liquid breeder TBMs
7) Computer controlled cutting tools such as laser and argon-arc torches, mechanical cropping equipment and electro discharge machining capability for careful sectioning of the TBM and subcomponents into more manageable pieces.
8) The capability to perform simple mechanical tests such as tensile or Charpy impact testing of pre-machined specimens over a range of temperatures and possibly under vacuum or inert gas environment.
9) RF or thermocouple monitoring to track the temperature of the TBM and TBM sections.
10) Radiation monitoring for radioactive decay of the TBM, TBM sections, and test specimens.
11) Provisions must be made to collect and pick up any waste parts from the sectioning.
12) The capability to load sections of a TBM and test specimens into appropriate activated material shipping casks for shipment.

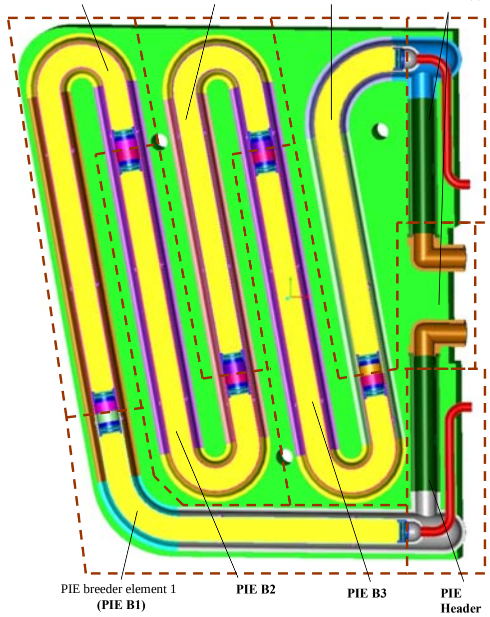


Figure 7.4-2: Example test blanket zone specimens for PIE

Table 7.4-1: Types of TBM PIE Samples

First wall structural panel, joints, stiffing plate, coolant header
Special subcomponents such as shear keys, flexible supports, attachment units, FCI, permeation barrier
Breeder cooling units, cooling plates
Breeder unit, ceramic breeder pebbles
Beryllium pebbles
Liquid breeders (in solid forms)

PIEs can be categorized according to types of materials. Example PIE tests for ceramic breeder and beryllium pebbles include measurement of residual tritium by annealing, examination of microstructure changes, measurement of helium production and swelling, and measurement of thermal-physical and mechanical properties. Example PIE tests for structural material specimens are microstructure, chemical analysis and mechanical properties of HIP joints and welding parts, metallographic examination of the diffusion bonding areas, inspection of corrosive products and cracks, identification of leaks, and mechanical tests such as resilience and traction.

Detailed technical parameters and corresponding spatial requirements will need to be specified in the future. Taking advantage of the H-H, D-D and then D-T operational phases of the ITER program, the implementation of remotely handled and irradiation tolerant equipment in the TBM Hot Cell can also be installed in a progression corresponding to the successive phases.

Table 7.4-2: Example Description of TBM (HCLL) PIE Needs

Operation Type	Object	Description/related issue	Comment
Visual inspection	TBM	Preliminary inspection	From external
Metrology	TBM	Check of box deformation	From external
Cutting (& RH)	TBM	Removal He collector from TBM (Open box) Access to internals	Cut at back plate (BP)/ FW welds Cut at tubes/BP welds
Cutting (& RH)	He collector	Dismantle He collector for inspection	Cut all BP/tube welds No PbLi residuals
Cutting (& RH)	TBM without He collector	Removal of Breeder Cooling Units (BCU) for access to internals	Cut spot welds Extraction could be difficult due to residual PbLi (local heating required?)
PbLi removal from steel surface	BCU	Removal of residual PbLi from BCU steel surfaces in view of PIE (corrosion, crack initiation, etc.)	Process to be defined
Pressure test (He)	BCU	Identification of leaks	Requires flange welding and He (static) at high pressure, temperature
Visual inspection	BCU	Identification of zones for PIE (corrosion, crack)	
Cutting	BCU	Cutting specimens out of Cooling Plates(CP) for mechanical tests	Small specimens (few mm thick) Prepare "V" notch (resilience tests)
Metallographic examinations	CP specimens	Examination of diffusion bonding area	
Mechanical test	CP specimens	Evaluation of mechanical properties	(Traction, resilience, etc.)
PbLi removal from steel surface	Box (FW+SG+ cover assembly)	Preparation of box for inspection (corrosion, cracks initiation, etc.)	Process to be defined
Visual inspection	Box (FW+SG+ cover assembly)	Identification of zones for PIE (corrosion, crack)	
Cutting	FW+SG+ cover assembly	Cutting of specimens out of Stiffening Plates (SP) and FW for mechanical tests	Small specimens (few mm thick) Prepare "V" notch (resilience tests)
Metallographic examinations	SP, FW specimens	Examination of the diffusion bonding area	
Mechanical test	SP, FW specimens	Evaluation of mechanical properties	(Traction, resilience, etc.)
Disposal	All remaining subcomponents, PbLi residuals, effluents	Storage of contaminated materials	

7.5 Comments

Remote handling of test blanket plugs is a planned operation which must be handled routinely and efficiently. The design of the TBMs and the design of tools for their remote handling inside the ITER Building and Hot Cell must be fully coordinated.

The TBM design needs to be further developed with assembly and maintenance requirements set as a priority. A high level of co-ordination between TBM designers and assembly/maintenance engineers is essential for the success of this project.

ITER remote handling and Hot Cell were designed to accommodate handling of TBMs proposed by 3 parties with minimum cost. There are no provisions for post irradiation examination, no storage space for more than 3 plugs and relevant TB modules, no possibility to refurbish TBMs. It is expected that remote operations inside the Hot Cell will be limited to the disconnection of a complete TB module sub-assembly and the installation of a new one. To accommodate TBM testing requirements proposed by 6 parties the current Hot Cell system is inadequate. To satisfy these requirements existing and available facilities at the construction site must be found or else the ITER Hot Cell must be significantly increased in size. Technically this is possible. Parties must confirm their commitments to TBM testing and recognize necessary cost increases in the ITER budget.

7.6 Conclusion

TBM replacement occurs in the ITER hot cell, where the whole TBMs/shield plug system is remotely transported in a standard ITER transfer cask.

The addition of a hot cell cask docking port, the modification of the TBM replacement procedure (including the provision of three additional TBM plugs for off-line refurbishment) and the expansion of the hot cell storage area may be required if the simultaneous replacement of the 3 test-port plugs has to be carried-out during a ITER planned shut down (1 month per year).

Because of the large number of TBM components present in the port cell, parking spaces need to be added in order to allow RH operation with the transport cask.

It is not yet clear if the irradiated TBMs (or part of them) will be sent to the Party which has manufactured them or if they will be left in the ITER site. In principle, valuable information (often necessary to the test results comprehension and interpretation) can be obtained from TBM post-irradiation examinations.

Current ITER Hot Cell scheme considers the irradiated TBMs as a waste object. Therefore, the ITER hot cell is designed to be used for replacing irradiated TBMs, but it is not designed to allow TBM repair and/or Post-Irradiation Examinations.

The TBWG recommends to assess in details the issues of TBMs replacement and PIE facility design and procurement. Information on requirements by Parties should be given to ITER Team in order to evaluate the possibility and the cost of a modification of the present TBMs procurement and replacement procedures and of the hot cell design. In both case additional space is likely to be required.

The presence of hot-cell facilities on the site selected for construction has also to be taken into account could help to solve this issue. Storage needs shall also be re-evaluated.

8 - GENERAL SAFETY CONSIDERATIONS

This section summarizes parts of the Generic Site Safety Report GSSR, parts of Section 3 of the Plant Integration Document [**8-1**] and the Safety Analysis Data List [**8-2**]. These documents are annexes and will reflect future changes in design and operation of ITER.

8.1 Safety objectives

The TBM are subjected to special safety requirements and shall not compromise the safety objectives, principles, requirements and guidelines of ITER [**8-1**]. Therefore the ITER safety is shortly introduced. The general safety objectives of protection of individuals, society and the environment are assured by GSSR-I [**8-3**]:

- Confinement and control of the radioactive and energy sources to keep radiation exposure and radiological releases below specified guidelines for normal operation, (including maintenance) and anticipated incidents and accidents.
- Prevention of accidents with high confidence and ensure that possible consequences are bounded.
- Ensuring that consequences of more frequent events, if any, are minor.
- Minimization of radioactive waste hazards and volumes.
- Demonstration that the favourable safety characteristics of fusion and appropriate safety approaches limit the hazards from internal accidents such that there is, for some countries, technical justification for not needing evacuation of the public.

8.2 Safety Principles

In the following extract from GSSR-I [**8-3**], the word “shall” is used to denote a firm requirement, the word “should” to denote a desirable option and the word “may” to denote permission, i.e. neither a requirement nor a desirable option.

8.2.1 <u>Defence-in-Depth</u>

All activities are subject to overlapping levels of safety provisions. A failure at one level would be compensated by other provisions, which follow the principles of redundancy, diversity and independence. Priority shall be given to preventing accidents. Protection from and mitigation of the consequences of postulated accidents shall be provided, including successive barriers for confinement of hazardous materials. Consistent with the "defence in depth", there are multiple approaches and prevention programs, such as administrative guidelines, procedures and inspections.

8.2.2 <u>As Low as Reasonably Achievable</u>

As a basic principle, exposures to hazards shall be kept as low as reasonably achievable (ALARA), economic and social factors being taken into account.

8.2.3 <u>Passive Safety</u>

Passive safety shall be given special attention. It is based on natural laws, properties of materials, and internally stored energy. Passive features, in particular minimisation of hazardous inventories, help assure ultimate safety margins in addition to fusion’s favourable safety characteristics.

8.2.4 Consideration of ITER Safety Characteristics

The safety approach shall be driven by a deployment of fusion's favourable safety characteristics to the maximum extent feasible. Relevant characteristics are:

- the fuel inventory in the plasma is always below 1g so that the fusion energy content is small.
- plasma burn is terminated inherently when fuelling is stopped and self-limiting with regard to power excursions, excessive fuelling, and excessive additional heating due to the limited confinement by the plasma of energy and particles;
- plasma burn is passively terminated by the ingress of impurities under abnormal conditions (e.g. by evaporation or gas release or by coolant leakage);
- large heat transfer surfaces and big masses exist and are available as heat sinks;

8.3 Design Principles

Safety design requirements are listed in the TBM section of GSSR-II [**8.3**]. General requirements are:

- plasma facing components should be excluded from safety functions. Consequences: The ex-vessel part of the primary cooling system is the first barrier for in-vessel radioactivity.
- emphasise passive safety in the design;
- maximise simplicity, fail-safe and fault-tolerant design, redundancy and diversity (wherever appropriate), independence, and testability.

8.4 Inventory Guidelines

The following inventory guidelines are explained and evaluated in GSSR-I & -III and listed in the PID [**8-1**].

8.4.1 Radioactive Materials

The inventories of tritium and activation products constitute a radiological hazard and hence shall be reduced as much as possible. Guidelines for these inventories are provided in the table A2a and b of the appendix.

8.4.2 Reactive Materials and Potential Hydrogen Production

Relevant for the TBM are the chemical energy especially the potential for hydrogen production. The final pressure in the VV caused by the combustion of 4 kg of hydrogen reaches the design pressure of the VV, i.e. 200 kPa. Therefore 4 kg of hydrogen is used as acceptance criteria for all reference accidents. The 1000 g-T in the VV correspond to 670 g-H_2 and the total NB-cryo-pump inventory to 350 g-H_2, together with the partly spilled hydrogen in the PFC water coolant (total ~1 kg) for radiolysis-suppression, would yield around 1.5 kg H_2 (equivalent) in the VV. This leads to the administrative guideline for hydrogen production of 2.5 kg resulting in limits for dust on hot surfaces (Table A2b of the appendix) and reactive materials in the TBMs (see §1.6).

8.4.3 Coolant enthalpy

Water and helium coolant (high temperature and pressure and cryogenic) have the potential of pressurizing confinements. The acceptable inventories have to be evaluated with accident analysis and according requirements and protection the confinement.

- He spills to the VV shall be limited to <45 kg (controlled by inventory limits or isolation valves) to assure reliable functioning of the VVPSS.

8.5 Confinement

Confinement is a fundamental safety requirement to protect against the release of radioactive and or toxic materials exceeding project release guidelines (table A1 of the appendix) for normal operation including maintenance, incidents, accidents and hypothetical events.

To provide margin against uncertainty in source term and mobilization as well uncertainty in modeling, the assessed releases shall not exceed 10% of the guideline for accidents. Releases from incidents shall be prevented. Incidents should not contaminate systems normally not contaminated and should not cause additional exposure to operators.

Confinement is implemented by sets of successive physical envelopes and systems monitoring and controlling pressures in, radioactive releases from and to these envelopes and systems protecting them.

The design basis for the confinement envelops and systems shall take into account the loads and environment of all events identified by the safety assessments including possible protection of the confinement by systems such as (decay) heat removal, pressure relief and control of energies etc.

Consideration should also be given to the mitigation of consequences from confinement degradation by, failures beyond design base accidents, i.e. by hypothetical sequences.
Systems implementing confinement functions and requirements are listed in PID [**8-1**] tables 3.1-3 to **-**7.

8.5.1 Confinement provided by ITER for the TBMs

These confinements are listed in table A5 of the appendix describing the confinement requirements and performance, the control of detritiated and filtered release, depression systems and protection of the confinement.

8.5.2 Confinement to be provided by TBMs

They are listed in the **Table 8-1** below.

Table 8-1: Confinement functions of TBM components (GSSR II, [8-3])]

Confinement	Confinement Nr. for source	Control and Protection
Ex-vessel parts of TBM-FW coolant loop	- 1st barrier for in-VV source terms (no safety functions (SIC) for PFC)	- 2nd confinement for separation between air and reactive coolants and or control of permeation/leakage.
Ex-vessel parts of breeder/multiplier coolants and purge-gases circuits in the Tritium Extraction System (TES)	- 1st barrier for activated/tritiated breeder, multiplier, corrosion products and dust (ceramics) - 1st barrier for in-VV source terms if TBM-box not SIC (see below)	- 2nd confinement for separation between air and reactive material and or control of permeation/leakage. - 2 redundant isolation valves avoid pressurization of the TES (2 bar design pressure of tritium systems) in case of a TBM FW coolant leak into the TBM-box or - pressure relief to keep p<2 bar (not to VV)
TBM in-vessel box	in ITER no safety function for in-VV components	- designed for in-box coolant break or pressure relief to protect VV (H_2 production) - passive (decay) heat removal (radiation, conduction to colder parts of the Tokamak
TBM-port plug (part of VV)	- 1st barrier for in-vessel material and TBM-materials (TBM in-vessel box is not SIC)	- passive (decay) heat removal by VV-HTS
TBM-assembly-cask with auxiliaries (stays in place during operat.)	- potential 2nd barrier for TBM-coolants and auxiliaries	- possibility of inertisation, detritiation, pressure relief to port cell etc.

8.5.3 Confinement of Chemical Energy

Confinement shall prevent/limit chemical reactions between coolant water, air and breeder/multiplier material, so that the confinement function is not threatened.
The design should ensure that two barriers exist between air and hydrogen, which may exist during normal operation or may be generated by accidents. A design with a single barrier between air and hydrogen shall be justified and agreed by IT.

8.5.4 Additional requirements for the confinement barriers

Confinement barriers and related systems shall meet the following requirements:

- Capable of returning the contained volume to sub-atmospheric pressures within 24 h after accidents.
- Valves implementing confinement shall close within specified period after detection of accidents. The confinement isolation valves shall fail closed on loss of power.
- Emergency power shall be provided to TCWS vault coolers, room depression systems, vent detritation systems (VDSs), VVPSS, suppression tank vent system (ST-VS) and safety chillers and safety related instruments and monitors.

8.6 Special Safety related Requirements for Test Blankets

Strictly the only top level safety requirements (for a generic site) are the release guidelines given in table A1 of the appendix and the occupational dose guidelines in table A3 of the appendix. For a consistent strategy to assure these top level requirements, ITER defines additional sets of top level project guidelines for safety design, inventories and requirements for components and systems e.g. for the confinements.

For consistency with the general ITER safety approach, the TBM design shall consider the following requirements:

- All ex-vessel parts of cooling and other auxiliary system are part of the first confinement barrier.
- Decay heat removal should be achieved by thermal radiation to basic machine.
- Chemical reactions between coolant, air and breeder/multiplier material shall be limited so that confinement function is not threatened. Self-sustaining chemical reactions shall be precluded by designing for sufficient heat transfer to colder parts of the machine. The 2.5 kg limit for additional hydrogen production should not be exceeded.
- Special consideration for Li fires in local test module confinement shall be made.
- Intermediate cooling loops are necessary for liquid Li system.

For the safety analysis of the TBM's the accidents defined in §8.9 ("Assessment") need to be considered and the following assumptions **shall** be taken into account:

- limit potential hydrogen production to 2.5 kg for each independent module (no common cause failure) by
  - Liquid Li shall be limited to less than 35 liters
  - PbLi should be limited to 0.28 m^3. Alternatively, detailed analysis of water/PbLi interaction should be performed.
  - Beryllium of the first wall of test module should be limited to 10 kg (in addition to breeder multiplier inventory potentially producing hydrogen).
  - Potential beryllium-steam reactions in pebble-bed breeder designs need to be addressed (Hydrogen, Heat of reaction etc.).

- Test blanket port cells shall enhance confinement function for radiological inventories.
- The test blanket assembly cask (which stays in place during operation) should have a confinement function.
- Test blanket module shall be recessed 50 mm from the first wall of basic machine
- *Deviations from these assumptions must be justified by detailed accident analysis and agreed with ITER.*

8.7 Normal Operation

Releases (leakage, permeation, maintenance) according

- annual release guideline (For one TBM about two orders of magnitude below guideline in table A1 of the appendix)
- reduce radioactivity such that effluents are ALARA;
- monitor the effluents.
- zoning (DAC level, direct radiation). Port cell defined as GREEN zone (see table A4 of the appendix)

To assure that the radiological requirements are met, through the entire life cycle of ITER, a radiation protection program (RPP) shall be developed and implemented.

8.8 Occupational Safety

Guidelines for individual and collective worker doses are listed in PID [**8-1**] and table A3 of the appendix allow to assign priorities for the process of reducing occupational radiation exposure according ALARA.

8.9 Assessment

Postulated initiating events (PIEs) should be systematically identified. A set of reference events shall be selected that encompass the entire spectrum of events. Hypothetical sequences should be used to investigate the ultimate safety margins. The intent is to demonstrate the robustness of the safety case with regard to the project's radiological requirements.

8.9.1 Component Classification

Systems, barriers, structures and components important for personnel and public safety (classified as Safety Important, i.e. SIC) shall be identified and appropriate requirements set. The identification and setting of requirements shall be based on the consequences of failure as determined by safety assessments. The PID [**8-1**] section 3.1.3 the lists SIC components in ITER and table 3.1-11 lists guidelines related to SIC.

Only components and systems classified as SIC can be taken into account in the safety analysis.

8.9.2 General accident analysis guideline

8.9.2.1 Event Sequence

The postulated initiating event (PIE) starts the event sequence along the line of defense for each safety function required. The development of the event trees stops when it leads to sequences defined hypothetical. An aggravating failure (AF) of one required SIC component needs to be included in the sequence. Simultaneously a loss of off site power (LOOP) of long (32 hr) and short (1 hr) duration needs to be analyzed, where there is an adverse effect on plant safety and to define emergency power requirements. Further a loss of all AC power, referred to as "station black out" for 1 hour needs to be considered as PIE. An AF is specified as a single failure of an active and passive component during

required change of state (AAG, [**8-4**]). Specific event sequences for a selected PIE (AAS, [**8-5**]) may override the above rule of: **PIE + AF+ the most unfavorable of (LOOP (1 hr to 32 hr).**
The PIE selected for the TBM assessment are listed in **Table 8-2**.

8.9.2.2 *Operator* Actions

Operator actions can be credited after:
> 1 hr for simple actions and > 8 hr for clearly identifiable actions
>32 hr if an assessment is required but a defined procedure exist, else >72 hours including the functioning
or use of non-SIC components if justified.

Table 8-2: **Selected PIE and required demonstrations for compatibility with the ITER safety approach**

PIE	Assessment of	Comments
1) Loss of coolant into VV	• small pressurisation of the first confinement (i.e.VV) • passive removal of decay heat • Limited chemical reactions and hydrogen formation	passive plasma shut down by coolant spill no damaging plasma disruption
2) Loss of coolant into breeder/ multiplier zone	• pressurisation of the module and purge gas sysstem. • Limited chemical reactions and hydrogen formation • subsequent in-vessel leakage	no active plasma shut down
3) Ex-vessel loss of coolant into port cell (vault)	• pressurisation of the port cell, vault, assembly cask • Limited chemical reactions and hydrogen formation	no active plasma shut down (possible passive plasma shut down by melting of the Be-TBM-FW)

Consequences resulting from subsequent damage to the tokamak (e.g. via heavy disruption) are beyond the scope of the assessment for TBMs fulfilling the ITER TBM requirements of §1.6 and ITER guidelines. Earlier safety analyses of TBMs are reported in GSSR [**8-3**] Volume VII, Appendix A.

8.10 Earthquake/Fire

Seismic requirements during and following a SL-2 earthquake are listed in the PID [**8-1**] table 3.1.-10.

In case of fire ITER shall be designed to assure that the:

- required safety functions are maintained, through a combination of fire prevention, fire detection and suppression, and mitigation of adverse effects on components important to safety (SIC);
- propagation of fire consequences that may impair safety functions are limited by spatial separation, redundancy, diversity, etc.

8.11 Decommissioning and Waste

The design shall support decommissioning as appropriate for an experimental device by:

- shielding to reduce induced activation of ex-vessel components during operation.
- use of modular components to simplify dismantling and reduce waste;
- use of remote handling equipment and procedures developed for normal operation;
- The design shall reduce the quantities of radioactive liquid waste.
- The design shall further incorporate means to reduce the volumes and radio-toxicity of materials which may remain as long-term waste after decommissioning by:
- re-use of components to the extent practical.
- limiting impurities in the materials to allow their clearance as early as practical;

8.12 Licensing

For licensing, a safety dossier (Gor03 [**8-6**]), similar to the ITER generic site safety report (GSSR) is required. There is a possibility that separate regulatory approval will be needed for the proposed TBMs.
The GSSR is structured into volumes as listed in the references.

8.13 References

[8-1] Project Integration Document, Section 3, http://www.iter.org/bl
[8-2] Safety Analysis Data List, http://www.iter.org/safety
[8-3] GSSR, Generic Site Safety Report, http://www.iter.org/bl
GSSR Volumes:
I. Safety Approach;
II. Safety Design;
III. Radioactivity and Energy Source Terms and other Hazardous Materials;
IV. Normal Operation;
V. Waste (Radioactive and Hazardous) and Decommissioning Provisions;
VI. Occupational Safety;
VII. Analysis of Reference Events; (Appendix A: Safety Assessment of the ITER Test Blanket Modules (TBM).
VIII. Ultimate Safety margin,
IX. External Events (Earthquake, Fire, etc.);
X. Sequence Analysis (to justify eference Events);
XI. Safety Models & Codes (verification and validation);
[8-4] AAG, Accident Analysis Guideline, http://www.iter.org/safety
[8-5] AAS, Accident Analysis Specification, http://www.iter.org/safety
[8-6]: Gor03, Test Blanket Module: Content of Safety Dossier, G 81 MD 13 03-10-21 W0.1 or ftp://ftp.itereu.de/tbwg/File_Exchange/safety/

Appendix 8-A

Table 8-A-1: Project Release Guidelines (non-site specific) (GSSR I, [**8-3**])

Events (Sequences) or Conditions	Goal	Dose	g-Tritium		activated products[1)]	
			in HTO	in HT	g-AP	g-ACP
Normal Operation including faults as a result of the ITER experimental nature.	Reduce releases to levels as low as reasonably achievable	Normal operation **0.02 mSv** 1% of annual natural dose	**0.1**	**1**	**1**	**5**
Incidents likely to occur one or more times during the life of the plant.	Reduce likelihood and magnitude of releases with the aim to prevent releases	Chronic with ingestion **0.1 mSv**	**0.1**	**1**	**1**	**1**
Accidents not likely during the life of the plant.	Reduce likelihood and magnitude of releases	Chronic w/o ingestion **5 mSv**	**5**	**50**	**50**	**50**
Hypothetical	demonstration of ultimate safety margin	Early dose[2)] **50 mSv** (no-evacuation criteria)	**90**	**900**	**20000**	**N/A**

[1)] in-vessel dust (AP), corrosion products in primary coolant (ACP), TBM materials need separate evaluation
[2)] 1 week without ingestion for the MEI (most exposed individual) at 1 km distance from a 1 hour release.

Table 8-A-2a: TER guideline for Tritium Inventories: (Total site inventory < 3000 g-T) (GSSR I, [**8-3**])

Typ of inventory	g-Tritium	Safety assessment values account for uncertaineties
Mobilizable in-vessel (in PFC, dust, etc.)	330 [a), b)]	
Cryopumps open to VV	120 [b)]	
Subtotal in-vessel	450 (ITER administrative guideline)	1000 g-T
Subtotal fuel cycle tritium inventory	450 (ITER administrative guideline)	700 g-T
Long-term storage	<450 g in each independent storage area	
Hot cell and waste processing	<250 g	

[a)] Not counting bred tritium in beryllium: 125 g (immobile for T<600C)
[b)] 5% of the in-vessel tritium shall be assumed as carbonised molecules C(DT)

Further Tritium Guidelines

Tritium Concentrations in Tokamak Cooling Water Systems
In-vessel components heat transfer system: <0.005 g/m^3-water
Vacuum vessel heat transfer system: <0.0001 g/m^3-water

Table 8-A-2b: ITER guideline for Activated Products

Activated Corrosion Products (ACP) / primary HTS-loop		**radiological source term**	**on hot surfaces (H_2 production)**
		10 kg (assessment value)	-
in Vessel Dust (AP) Administrative Limit	Carbon	200 kg	6 kg
	Beryllium	100 kg	6 kg
	Tungsten	100 kg	6 kg

Table 8-A-3: ITER guideline occupational exposure (GSSR VI [**8-3**], PID [**8-1**])

Dose Limits		
ICRP limit for annual individual worker doses	20 mSv/year averaged over 5 years, not > 50 mSv/year	
Project Guidelines		
Annual individual worker dose	< 5 mSv	
Individual dose for any given shift	< 0.5 mSv/shift	
Collective annual worker dose	< 0.5 person-Sv (averaged over operational life time)	
ALARA threshold for dose rates	100 μSv/h	An 'ALARA threshold' triggers a systematic review during the ITER design phase. This does not imply that ALARA reviews will not be performed when the design is below the thresholds.
ALARA threshold for collective worker dose to operate and maintain a system for a year	30 pers-mSv	
ALARA threshold for collective worker dose for a task performed less often than annually	30 pers-mSv	

Table 8-A-4: Area Classifications and Radiation Access Zones (GSSR VI [**8-3**], PID [**8-1**])

Zones/Access Limitations	**Airborne / Total Dose Rate / Area Contamination Characteristics**
Zone A Non-Supervised Area Unlimited Access.	• No airborne contamination. Dose rate < 0.5 μSv/h; • WHITE contamination control zones only: No surface or airborne contamination and no reasonable possibility of cross-contamination.
Zone B Supervised Area Limited Access for NRW. (a) Unlimited Access for RW. (a)	• Total dose rate (internal + external) < 10 μSv/h; • GREEN contamination control zones acceptable: No loose contamination tolerated. May be subject to temporary surface or airborne cross-contamination, airborne should not exceed 1 DAC.
Zone C Controlled Area Limited Access for all workers.	• < 100 DAC and < 1 mSv/h; • AMBER contamination control zones acceptable: Airborne and loose surface contamination must be controlled and shall be maintained ALARA.
Zone D Controlled / Restricted Area, shall have physical barriers to prevent inadvertent entry. Entry only with a high level of approval.	• Airborne >100 DAC or external dose rate > 1 mSv/h; • RED contamination control zones are only tolerated in Zone D. These areas have permanent or higher than AMBER levels of contamination.

(a) Personnel performing work requiring exposure to radiological hazards will be designated as Radiation Workers (RW). All other personnel, including non-designated visitors, will be treated as Non-Radiation Workers (NRW).

Notes: DAC = Derived Air Concentration: unprotected exposure to 1 DAC = 10 μSv/h
For internal dose rate, hazard defined in DAC of airborne contamination. For external dose rate, hazard defined as μSv/h

Tritium (HTO)	Tritium (HT)	Be (ativated imp.)	Stainless Steel	Copper	Tungsten W	co-deposited T Carbon	ACP (5xTungsten)
DAC [Bq/m^3 air]		DAC [μg/m^3 air] 1 day after shutdown					
$3.1 \cdot 10^5$	$2.0 \cdot 10^{10}$	10	0.2	0.4	0.2	0.05	1

Table 8-A-5: Confinement provided by ITER for TBM's, Data from SADL [**8-2**]

Confinement # for - free Volume - *Design pressure [bar abs.]* - Assessed leak rate	**Control** of - detritiated and filtered release - sub-atmospheric pressure, - face flow >1m/s across leak	**Protection from**
VV and extensions 1st for in-VV sources 1600 m^3 *2 bara* 1vol.%/day at 2 bara	- VVPSS-bleed line + ST-VDS at p >0.9 bar in VV (3 min for ST-VDS startup) + N-VDS-1 - maximum window or break size 0.02 m^2 - S-ADS by operator (air leak incident)	- VVPSS, rupture disk at p>1.5 bar - Isolation valves for cryogenic lines (He-spill < 45 kg) - VV-HTS natural circulation - Administrative limits on dust on hot surfaces (hydrogen) - passive plasma shut-down **1)** for T_{FW}>1080 ^oC and spills - Alternating sectors cooled by different HTS loops enhances radiative cooling to non-affected sectors and protect against magnet failures
Port cell enhances confinement for in-VV sources 175 m^3 *1.6 bara* 100vol%/day at Δp=300Pa	- N-VDS-2 (not SIC) because of S-VDS serving the gallery	- pressure relief into vault at Δp>20kPa, re-closes at Δp<1kPa - vacuum breaker - coolant bypass with isolation valve (small diameter reduces break size) during baking
TCWS-vault+pipe shafts 2nd for in-VV sources 46000 m^3 *2bara* 100vol%/day at 2 bara	- vault dryer + N-VDS-1 (plasma burn+baking) - HVAC (maintenance) - HVAC isolation + S-VDS	- vault cooler 1.2 MW and leak detection for LBB - active plasma shut down for p >1.05 bar**2)** - relief to cryostat, p> 1.8 bar (cold cryostat during baking) - relief to environment, p>2bar (hypothetical)
Gallery ultimate for in-VV sources 62000 m^3 *1.1 bara* 100vol%/day at Δp=300Pa	- HVAC (8Vol.change/day) - Dust filter 95% efficient) - HVAC-isolation and automatic S-VDS initiation	- monitoring

1) Be TBM-FW does not shut down passively the plasma below melting temperature because of the small area compared to ITER FW

2) No signals from Test Blankets to activate the safety classified Fusion Shut down System

Gallery and rooms housing the VVPSS and drain tanks as well as the tritium building act as ultimate line of defense with HVAC isolation and S-VDS start up.

ST-VDS: Suppression Tank Vent Detritiation System, 150 m^3/hr (SIC), 99.9% efficient for dust and tritium removal
N-VDS-1: Normal Vent Detritiation System: 500 m^3/hr, dust filter efficiency 99.9%, detritiation efficiency 99% (SIC)
N-VDS-2: 700 m^3/hr, provides negative pressure in port cells relative to gallery (no SIC)
S-VDS: Standby VDS (SIC), 3000 m^3/hr for TCWS-vault (maintenance), Gallery, VVPSS- and drain tank rooms and Tritium building (connection to port cells under discussion). Final release through N-VDS-1
S-ADS: Standby Air Detritiation System 4500m^3/hr for VV maintenance and enhanced detritiation in Tritium and Tokamak building (no SIC).

The S-VDS is automatically triggered at 0.1 μg-tritium/m^3 or 0.8 g-activation products (4 g-ACP) released into room. Room isolation 30 s after detection. No flow for 5 minutes, then detritiation efficiency of 95% for 2 hours and 99% after. Dust filter efficiency 99.9%.

The depression systems of the HVAC's control the sub-atmospheric pressure also in case of fire (HVAC isolation, cooler etc) and combined fire and tritium release by switching to a detritiation system.

9 - TECHNICAL SPECIFICATIONS, QA AND ACCEPTANCE TESTS FOR TBMS

This section addresses the issues associated with procurement of TBMs and outlines the general requirements for integrity assurance of TBMs, including QA and acceptance test. To facilitate this, the basic approach is discussed for assuring structural integrity of the ITER components, together with key technical elements that could be referred for procurement of TBMs.

9.1 Basic approach for integrity assurance of ITER components

9.1.1 Structural design criteria (SDC) for ITER engineering design

ITER has unique and attractive safety features, such as no nuclear excursion, inherent safety shutdown of plasma, low decay heat density and limited radiological hazard potential equivalent to 50-kW research reactors. This could imply that the level of reliability to be attained for the mechanical components would be equivalent with ASME Sec. VIII with respect to radiological safety.

On the other hand, the ITER tokamak components are to be designed to meet the operational requirements so as to assure structural integrity and leak tightness under electromagnetic and thermonuclear loads. In addition, the ITER tokamak components require different technology from the existing codes and standards because of their unique construction as typically shown below:

1) Design by analysis for complex structures under electromagnetic loads
2) Special welding/bonding such as partial penetration welding, HIP and brazing
3) Different welding joint category from pressure vessel
4) Superconducting magnets and structures operated at cryogenic temperature
5) Limited access for inspection
6) Non-metallic parts

To facilitate the above uniqueness, the ITER project has developed the Structural Design Criteria (SDC, [**9-1**]) in collaboration with the code experts of the participant Parties. SDC has been applied for structural design of the ITER mechanical components for development of the engineering design to date. SDC doesn't contain the requirements for QA, fabrication, operation and maintenance, since these depend on the industrial base and practice in each country, and should be formulated for construction.

SDC is basically composed of the following three criteria:

(1) SDC-MC for superconducting magnets and structures: this includes special structural materials for cryogenic use, design by analysis for electromagnetic loads considering unique superconducting characteristics, etc.
(2) SDC-IC for in-vessel components: this includes neutron irradiation influence on structural materials, design by analysis for electromagnetic loads, etc.
(3) Existing industrial codes for all other components: this is based on the US codes and standards, such as ASME for mechanical components, and ACI (American Concrete Institute) and AISC (American Institute of Steel Construction) for buildings and structures.

9.1.2 Codes and standards for procurement and licensing

The ITER Organization, a newly established international legal entity, will be fully responsible for construction, operation and exploitation of the ITER facility. All ITER components will be procured by the ITER Organization in support of the participant Parties. For procurement, the ITER Organization will provide the technical specifications and requirements, including codes & standards (ITER C&S) to assure structural integrity of mechanical components. The ITER C&S will be formulated with well-known codes & standards, such as ASME and RCC-MR, based on SDC and additional requirements for fabrication, operation and maintenance.

The ITER C&S should be established to ensure license acceptance as well as to assure quality and reliability of the components that are procured among various Parties. For this, the followings requirements and provisions are to be incorporated:

1) Conformity to the regulatory requirements for assurance of safety function
2) Identification of duty and responsibility, considering international procurement
3) Details of technical requirements specific to the ITER structures, systems and components, together with their justification or authorization
4) Provision of criteria or principle to identify equivalency with supplier's own QA and conformity assessment requirements

The basic requirements of QA can be referred to the international standards such as ISO or other well-recognized standards. The regulatory authority will confirm proper QA activities so as to assure and maintain the required quality and reliability of the components procured. The conformity assessment will be basically implemented by independent review, inspection, accreditation and audit. The detailed procedures of QA and conformity assessment can be in accordance with supplier's own administration if it is equivalent with the basic requirements prescribed in the ITER C&S.

9.2 General requirements for procurement of TBMs

Based on the above mentioned approach for the ITER components, the general requirements can be highlighted below, together with key technical elements to be considered for the procurement of TBMs and associated structures and auxiliaries.

9.2.1 Basic approach

9.2.1.1 General

TBMs are installed for experimental purpose and are not safety-related components. However, high standards of quality assurances are required as the ITER component for safe and reliable ITER operation. Therefore, at least, the structural integrity and nuclear shielding capability of TBMs, structures and auxiliaries are to be assured in accordance with the ITER technical requirements. In addition, the TBM components that form the primary boundary, such as port closure plates and external auxiliaries with containing radioactive material, are to be licensable as safety-related components.

9.2.1.2 Licensing

For the TBMs and structures (non safety-related TBM component), it is required to ensure no mechanical impact on the ITER primary boundary due to their operation. This requires justification of the applied codes and standards specific to TBMs and structures. The comprehensive safety analysis is also required to address all possible events and show their consequences within the ITER safety envelope. The construction application of TBMs and structures should include at least the basic design information, such as materials, structures with dimensions and weight, for estimating the amount of source terms due to activation of TBMs and structures generated by neutron irradiation. For the safety-related TBM components that form the primary boundary, all activities for construction, operation and maintenance have to be approved by the regulatory process as same as other safety-related ITER components. The Inspection Organization should be authorized in each country and the QA activities are audited and confirmed by the ITER Organization and by the Regulator where necessary.

9.2.1.3 Construction and maintenance

The construction and maintenance of TBMs, structures and auxiliaries are to be based on adequate technical requirements, including codes and standards, and administrative requirements to control and administer all processes. The technical requirements specific to TBMs should be justified, including identification of the equivalency with the ITER C&S. The administrative requirements should include QA/QC and conformity assessment, and their equivalency with the ITER criteria should be identified.

9.2.1.4 Acceptance test

For non safety-related TBM components, it is required at least to confirm design and fabrication reports, examination and testing records, dimension measurement, external inspection, and pressure and leak testing. In case of the safety-related TBM components, all reports and records should be provided in accordance with the administrative requirements (QA/AC and conformity assessment) as same as other safety-related ITER components.

9.2.2 Key technical elements

9.2.2.1 Technical specification documents

Technical specification documents (TSD) are to be prepared for the procurement of TBMs, structures and auxiliaries. TSD should cover the general requirements and technical requirements, and will include at least (i) contract stages and schedule, (ii) responsibilities, (iii) QA basic requirements, (iv) applicable codes and standards, (v) acceptance criteria, (vi) free issued hardware, and (vii) detailed technical specifications and requirements unique to TBMs, such as materials, design, fabrication (including welding), examination and testing, maintenance and inspection, and packaging and transportation. Note that specific requirements such as tolerances, surface roughness and outgas rate should be also included in the technical requirements.

9.2.2.2 Material specifications

Material Specifications are to be prepared in close contact with possible manufacturers. Special attention should be paid on additional requirements specific to TBMs or ITER, such as optimization of chemical composition for reduction of the cobalt and niobium for induced radioactivities and characterization of electromagnetic properties. When non code-qualified material is used, detailed specifications should be provided, together with qualified data for defining the allowable strength.

9.2.2.3 Structural design

TSD should clarify the codes and standards or structural technical standards applied to TBMs for assuring the structural integrity: these should give specific technical rules for material, design, fabrication, examination and testing, inspection, repair and replacement, packaging and transportation. The technical rules should be based on unique features of TBMs such as non-conventional material, design and fabrication.

As mentioned, the structural design criteria for the ITER in-vessel components (SDC-IC) have been prepared by using existing codes and further development is under way for formulation with additional technical requirements for construction, operation and maintenance. Typical examples are the technical rules for non-standard material, unique welding/bounding joints and their joint efficiency, Non Destructive Test (NDT) including surface defect inspection. Assuming the TBMs environmental conditions similar to those of the ITER in-vessel components, some of design criteria or approaches will be used for structural design of TBMs although their applicability should be justified.

It also should be noted that the loading events and combinations can be classified into several categories, depending on the frequency of event occurrence. Correspondingly, the allowance or damage limit for the structural design is defined with different safety factor for different event categories. For example, in case of infrequent and extremely unlikely events, the safety factor of 1.2 and 2.0 can be applied, respectively.

Considering complex structure and electromagnetic loads, the structural design codes should allow the choice of “Design by Analysis” as well as “Design by Formula” or “Design by Test”. In case of Design by Analysis, elastic and inelastic analysis including limit analysis should be applied to prevent the following failure modes:

- gross plastic deformation,
- incremental plastic collapse (ratcheting),
- fatigue, creep and brittle fracture,
- Buckling.

The elastic analysis requires certain procedures to categorize the calculated stresses into different stress classification: (i) Primary Stress, (ii) Secondary Stress and (iii) Peak Stress. This makes inconvenient for complex structures under electromagnetic loads. On the other hand, the inelastic analysis doesn’t require the stress classification. There are basically two types of analysis, limit analysis and plastic analysis. The limit analysis is used to calculate the limit load of the structure, based on small deformation theory and an elastic-perfectly plastic material model. When the limit load is calculated, this procedure will be much simpler than the elastic analysis stress categorization procedure. However, the effects of plastic strain concentrations in localized areas of the structure must be assessed in the light of possible fatigue, ratcheting (incremental plastic collapse) and buckling failure. The plastic analysis is used to determine the plastic collapse load. The analysis is based on a model of the actual material stress-strain relationship.

9.2.2.4 Welding and NDT methods

Specific points of welding and NDT methods are to be noted because these would give an impact on the TBM design and the fabrication methods. As a general rule, TBM welded joints qualification and inspection/testing will be carried out based on the TSD standards.

(a) <u>100 % NDT tests and requirement of high quality welds</u>: It is expected that 100 % surface and volumetric tests will be requested on welds to confirm high quality welds (for mechanical structural integrity and to avoid risk of vacuum leaks).

(b) <u>One-sided welds:</u> One-sided welds cannot be avoided in all-welded box structure. They will require qualification of the welding procedure and sufficient testing to ensure accurate statistics on the strength and fatigue life. Based on this testing, knock down factors on the strength or the fatigue curves may be required.

(c) <u>Volumetric tests on one-sided welds:</u> Radiographic testing (RT) cannot be used on welds with one-sided access. Ultrasonic testing (UT) is to be used as a substitute. It is proposed to keep a minimum distance between welds to achieve acceptable S/N ratio and required resolution. UT usually requires 2 wave launching angles (usually 60 and 45 degrees). Dye Penetrant Test (PT) can also be considered as a substitute (to be confirmed by the TSD) and it is expected that the weld joint efficiency is 0.85, as shown in **Table 9-1** and **Table 9-2**.

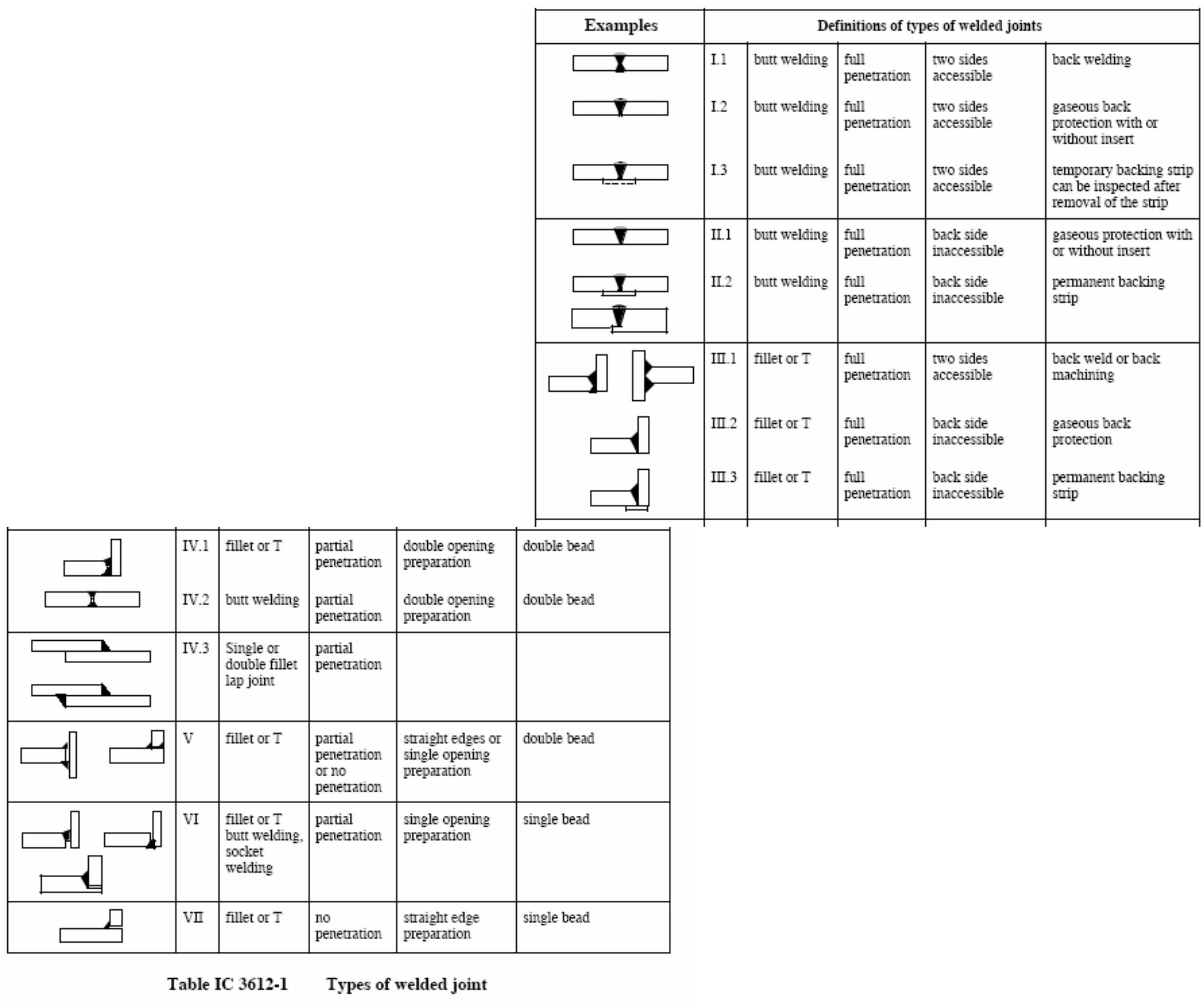
(d) Surface NDT by liquid penetrant at welds: Liquid penetrant inspection is required usually to inspect the surfaces. To avoid contamination of the high quality vacuum, adequate high-purity liquid penetrant must be selected. A substitute method is also proposed to eliminate outgas issues. The photothermal camera method has been developed for this purpose.

(e) T-joint welds are used but cross welds are to be avoided (prohibited in existing design codes).

(f) Requirements for qualification tests by using mock-ups or proto-types are to be defined. It is proposed that fabrication methods of non-conventional materials/structures are to be demonstrated by the qualification tests.

(g) Weld joint efficiency is defined depending on the type of welded joint and NDT method as shown in **Tables 9-1** and **9-2**.

Table 9-1: Types of welded joints



Examples	Definitions of types of welded joints				
	I.1	butt welding	full penetration	two sides accessible	back welding
	I.2	butt welding	full penetration	two sides accessible	gaseous back protection with or without insert
	I.3	butt welding	full penetration	two sides accessible	temporary backing strip can be inspected after removal of the strip
	II.1	butt welding	full penetration	back side inaccessible	gaseous protection with or without insert
	II.2	butt welding	full penetration	back side inaccessible	permanent backing strip
	III.1	fillet or T	full penetration	two sides accessible	back weld or back machining
	III.2	fillet or T	full penetration	back side inaccessible	gaseous back protection
	III.3	fillet or T	full penetration	back side inaccessible	permanent backing strip
	IV.1	fillet or T	partial penetration	double opening preparation	double bead
	IV.2	butt welding	partial penetration	double opening preparation	double bead
	IV.3	Single or double fillet lap joint	partial penetration		
	V	fillet or T	partial penetration or no penetration	straight edges or single opening preparation	double bead
	VI	fillet or T butt welding, socket welding	partial penetration	single opening preparation	single bead
	VII	fillet or T	no penetration	straight edge preparation	single bead

Table IC 3612-1 Types of welded joint

Table 9-2: Weld joints efficiency

Type of welded joint / Examinations	I.1, I.2, I.3, III.1	II.1, III.2	II.2, III.3, IV, V, VI, VII
Volumetric examination[1] + Surface examination[2] after welding (both sides)	1.0		
Surface examination[2] during welding + Surface examination[2] after welding (one side)	0.85		0.5
Surface examination[2] after first pass + Surface examination[2] after welding (one side)	0.7		0.5
Surface examination[2] after welding (one side)	0.5		0.5

1 radiography or ultrasonic

2 liquid penetrant (for components not exposed to high vacuum) or magnetic particle

9.2.2.5 Leak testing (vacuum testing) and Pressure testing

Test conditions, procedures and acceptance criteria should be defined in TSD for leak testing and pressure testing.

(a) Pressure testing
For example, test pressures are defined in SDC-IC, as follows:
- Hydrostatic test: 1.25 P_D x Sm(Tt)/Sm(Td)
- Pneumatic test: 1.15 P_D x Sm(Tt)/Sm(Td)

The test pressures are defined based on the design pressure (P_D) of each component. Sm(Tt)/Sm(Td) is a correction factor according to the temperature difference of the pressure testing. The test pressures can be different according to design codes. For example, according to RCC-MR, higher value (1.43 P_D) might be requested for the vessel.

(b) Leak testing (vacuum testing)
It will be requested that leak testing is completed for each TBM before its transportation to the site. “Hot leak testing” might also be requested (to be defined in the future).

9.3 Reference

[9-1] G 74 MA 8 01-05-28 W0.2, “ITER Structural Design Criteria for In-vessel Components”.

10 - LICENSING PROCESS

This chapter recalls the main constraints that have to be satisfied by TBMs and associated systems in order to be included in the general licensing procedure of ITER. A general approach for experimental devices and an overview of the information that the ITER site team would need to conduct the Licensing Process are provided.

10.1 General approach

The ITER Joint Assessment of Specific Sites has found that ITER documentation is consistent with licensing guidelines at all proposed sites, allowing the construction, safe operation and decommissioning of ITER.

For the TBMs, ITER DRG has assumed that they appear as a normal VV plug, do not compromise the safety and reliability of operation, and are replaced, at the most, once a year during ITER planned shutdown by remote maintenance.

Test Blanket design, mounting, operation, waste production and dismantling needs to obey the ITER design requirements and guidelines.

As far as they are available, safety relevant characteristics on TBMs have to be provided within the framework of ITER documentation, i.e. the Preliminary Safety Report & the relevant sections of all the other mandatory documents, and safety ITER documents. Nevertheless, considering that some of them could be not sufficiently described at this licensing stage, another approach may be to include TBMs in the more general frame of "experimental devices".

The global approach to deal with experimental devices and programmes in facilities is provided in next section.

10.2 Approach for Experiments and Programmes

The links and interactions between ITER Experimental programme (H, D, T phases and TBM programme) and Safety operational limits are going to be studied and defined in the framework of an EISS task in 2005-2006. See annex A of EISS-5 contract with EFDA where in the task definition SL53.2 it is pointed out that *"The design must take into account new experiments to be added to the device as TMB, NBI, diagnostics, i.e…"*.
And according to recommendations and commitments made by the safety authorities after the "dossier d'Options de Sûreté" (Safety Option report) evaluation:
"It must be shown for example that maximum foreseen fusion power value is in accordance to power loads to be supported by vacuum vessel cooling systems. The consistency of this value with the experimental program must be shown. It must be verified that waste generated by new experiments are inside the given margins for the different categories of waste and that a depository will exist for all the produced radio-nuclides. The principles for respecting that an extension of any experimental program with higher fusion power or neutron fluence will not exceed the foreseen amount of waste taking into account the uncertainties will be presented"

Issues from this study will offer a valuable frame to define an experiments integration policy.
Nevertheless general trends may be sketched: when considering that along the whole life of ITER some new devices or programmes may appear, the safety process should be consequently affected and the three main following arrangements are possible, according to three generic situations:

- Experiments considered and analysed in Initial Safety Files
- Experiments with no extensive description in Initial Safety Files

- Unforeseen Experiments, Equipments or Tokamak operating conditions

10.2.1 Experiments considered and analysed in Initial Safety Files

This is the best situation. Safety and Environmental safety relevant information on TBMs or any new experimental items would be provided in the Preliminary Safety Report (RPrS) and other mandatory documents. Complementary information would be included necessary in the next licensing step: the Provisional Safety Report (RPS).

In this case it should be checked that the required information on TMB is included in the corresponding chapters of RPrS and other safety documents table of contents or in a dedicated chapter on experiments. Additional constraints from ITER "operator" set out in the Design requirements and Guidelines have also to be taken into account.

For example the following points should be tackled:

– Safety Principles
  - General Safety Goals
  - Compliance of Safety Functions
  - Rules for event analysis and components classification

– Operation conditions
– Event analysis, Calculation codes
– Nuclear/non-nuclear risks
– Internal hazards analysis
  - Fire, explosion, chemical risk , load drop, flooding

– External hazard analysis
  - Earthquake, aircraft crash ...

– Justification of the design for SIC
– Impact of the facility on workers, public and neighbouring environment (in any operation conditions + ALARA)
– Normal + abnormal operation / effluents and releases
– Quality Assurance (with regard to Safety)
– Tests (related to Safety) for start-up + In Service Inspection
– Waste (tritium inventory and materials could be a problem) and Decommissioning

Remark about TBMs waste production & management and Decommissioning:
Decommissioning requirements will have minimal effects on the TBMs (provided they satisfy safety and operational requirements, e.g. on remote maintenance and rules for hot-cell management.
From the point of view of waste disposal the TBMs materials choice must comply with national regulations on nuclear waste that could limit the use of specific materials or would need special studies and/or agreements for acceptance of some radioisotopes or chemical materials. In any case a complete radiological inventory with the characteristics of the expected activated material is required. In particular the amount of Beryllium to be managed during maintenance and dismantling should be specified.
Compliance with IAEA regulations on transportation of radioactive materials regulations will have to be satisfied.

Disposal site will be either host country or country of origin, depending on ownership of TBMs and on negotiations to be held on this issue in the near future.

An example of requirements on structural material information is given § 10.3.
For checking the compatibility of the experiment characteristics with the regulatory framework outlined above an internal commission could be established for validation of the conformity to initial description and analysis inside the envelop case and for comparing experimental operation conditions to the pre-defined parameters

10.2.2 Experiments with no extensive description in Initial Safety Files

It could happen that some TBMs will not be defined in sufficient detail to timely provide all information required in the Preliminary Safety Report. This could also be the case for other experimental components and/or diagnostics. One possibility is to define an “envelope” case or a limited number thereof. From the regulatory standpoint, for each new experimental component a demonstration showing that it falls within on these envelopes should be provided.

Nevertheless the following constraints will have to be taken into account:

- Highly recommended to:
  - anticipate the "type" of experiments in the Creation Decree
  - include the related wastes and releases in the initial documents
- Compulsory to respect host country Quality Insurance Order from the very beginning (design)
- No major changes in envelope situations
- To present specific Safety Report or Information to Safety Authority (take into account the corresponding time constraints in planning).

10.2.3 Unforeseen Experiments, Equipments or Tokamak operating conditions

In that case, or if significant changes in some envelope situations are needed, the process is heavier.

Roughly two cases are possible:

- A limited impact could lead to a new limited Licensing Process (modifications of Decrees without public enquiries)
- A significant increase of impact would induce a complete Licensing Process.

10.2 Example of required Data

All information must be present in the DDD of each proposed TBMs. As an example, the detailed elements and materials inventories for each TBMs are given in **Tables 10-1** and **10-2**.
These tables must be completed with the activation data at the end of foreseen life for each experiment in order to evaluate ORE consequence if any and waste management.

Table 10-1: Structural materials composition *(without impurities)*

Name	Elements and fraction wt %
Eurofer	Fe, Cr (9.40), C (0.11), Mn (0.50), Si (0.05), W (1.0), V (0.25), Ta (0.08), N (0.03), B (0.005)
F82H	Fe, Cr (8.0), C (0.10), Mn (0.50), Si (0.20), W (2.0), V (0.2), Ta (0.04), N (<0.01), B (0.003)
RF-FS (9Cr MoVNb)	Fe, Cr (8.6-10), C (0.12), Mn (0.3-0.6), Si (0.17-0.34), Ni (<0.5), Mo (0,6-0.8), V (0.1-0.2), S (<0.025), P (<0.03)
CLAFM (China)	Fe, bal, Cr(9%), Mn (0.45%), C(0.1%), W(1.5%) ,Ta(0.15%) V(0.2%)
V-4Cr-4Ti (RF)	V (90%), As (<6.5%), Ti (4%), Cr (4%), Si (4.0E-2%), Al (2.0E-2%), Mo (2.0E-2), O (1.5E-2%), Fe (1.2E-2%)

Table 10-2: Materials masses in a half-port size TBM*

	Structural material		Breeder(s)		Multiplier		Coolant		Functional material(s)	
TBM type	Name	Mass [kg]	Name(s)	Mass [kg]	Name	Mass [kg]	Name	Mass [kg]	Name(s)	Mass [kg]
China HCSB	Eurofer	1326	Li_4SiO_4	51	Be pebbles	106	He	0.2	None	-
EU HCSB	Eurofer	1920	Li_4SiO_4 or Li_2TiO_3	97	Be pebbles	396	He	~0.6	None	-
Japan HCSB	F82H	1961	Li_2TiO_3 or others	104	Be or $Be_{12}Ti$ pebbles	305	He	0.5	None	-
Korea HCSB	Eurofer	833.8	Li_4SiO_4	85.3	Be pebbles	126.7	He	0.095	Graphite pebbles (n reflector)	305.3
RF HCSB (for one submodule)	FS (9Cr MoVNb)	524 x 2	Li_4SiO_4	35.5 x 2	Be (porous blocks)	310 x 2	He		None	-
US HCSB (for 3 unit cells)	F82 H or Eurofer	144	Li_4SiO_4 or Li_2TiO_3	38.5	Be pebbles	22	He		None	-
Japan WCSB	F82H	2041	Li_2TiO_3 or others	91	Be or $Be_{12}Ti$ pebbles	430	H_2O	38	Al_2O_3 (T permeation barrier)	xx
China DFLL	CLAFM	1193	Pb-17Li	2196	-		He	0.76	SiC/SiC (flow channel insert)	105
EU HCLL	Eurofer	1784	Pb-17Li	2863	-		He	0.95	Al_2O_3 (T permeation barrier)	< 5
US DCLL	F82H	763	Pb-17Li	1824	-		He	1.05	SiC/SiC (flow channel insert)	49
Korea HCML	Eurofer	802	Li	13.9	-		He	0.283	Graphite pebbles (n reflector)	333.7
RF SCLi	V-4Cr-4Ti	315	Li	11	Be (porous blocks)	20	Li (same as breeder)	Included in breeder column	1) Er_2O_3 or AlN (insulating coating) 2) WC inserts	< 1 150

* All TBMs FW are expected to be covered by a Be protection layer (thk. 2mm, TBD)

11 – GENERAL CONCLUSIONS, RECOMMENDATIONS AND FUTURE WORK PROPOSALS

This chapter summarizes the major findings of the TBWG activities, including the identification of the work that has yet to be performed by Parties in support of their TBMs proposals and that to be performed jointly by Parties and ITER Team in support of the integration of TBMs systems in the ITER design and procurement to allow TBM testing to take place.

11.1 TBWG Main Conclusions

The TBWG has concluded that the tests of DEMO-relevant TBMs in ITER will give essential information to accomplish such a development, although a part of the required R&D results will have to be obtained in other facilities, namely the high dose irradiation effects on blanket materials, materials interactions, and synergistic effects.

Besides the need of checking TBM compatibility with ITER operations, TBM testing in the initial H-H phase is essential both to demonstrate structural integrity and safety-related performances of TBMs before starting the D-T operations and to validate remote operations on the on the TBM systems.

The preliminary TBM integration work in ITER has shown that blanket testing in ITER will be very complex and very lengthy. It is therefore required to make most of the performance tests, if feasible, in dedicated out-of-pile facilities prior to the installation in ITER.

Present TBM designs are dictated by testing objectives and are performed to assess the TBM behavior under ITER operating conditions. However, to recover the required data, the development of appropriate instrumentation and data acquisition systems is necessary with high priority.

In order to maximize the information obtained by TBM testing in ITER, it is essential to develop and improve the corresponding DEMO blanket designs in order to have a coherent basis of comparison and interpretation for the obtained TBM results and to have the possibility to evaluate the impact on the DEMO blanket designs of the observed TBM performances. All Parties have identified, at least, two kinds of DEMO-relevant blanket for testing them in ITER in support of their breeding blankets development programs.

Significant steps need to be performed prior to the TBMs installation in ITER, such as materials fabrication routes and irradiation resistance, out-of-pile tests of mock-ups and associated systems up to the test of prototypical TBM systems, and RH equipment validation.

Taking into account the reduced FW loads compared to DEMO, the agreed TBM testing approach is to have, for each blanket concept, a series of different TBM designs, each one devoted to specific testing objectives, starting from the initial TBM in the H-H phase where no nuclear heat is present to the one installed at the end of the D-T phase where pulse-length longer than 1000 s and a large number of back-to-back pulses could be expected.

At present, the ITER Parties have proposed several independent DEMO-relevant TBMs that cannot all be tested simultaneously on day-one. Space limitation is not only due to the space available in the test ports but also to the limited space available in the port cells, in the vertical shafts, and in the TCWS vault. The situation will become even more difficult during the D-T phase because additional TBMs are proposed.

However, although at different levels, all proposed TBMs need further specific R&D before proving their acceptability. It is likely that some proposals will be abandoned either for technical or financial reasons and, therefore, it is important to allow some flexibility on the final choice of TBM concepts to be installed in each port. For this reason, the TBWG has asked to each Party to prepare a DDD for each proposed TBM system, independently from port allocation or space availability in ports, in port cells and in TWCS vault. This information will allow to understand the detailed requirements for each TBM and to prepare the necessary actions in order to not prevent their possible test in the ITER test port. The final choice of the TBMs to be tested can be postponed of few years from now.

In order to allow such flexibility without disturbing the ITER construction phase, the TBWG decided to fix, for each port, the number and the dimension of the connection lines crossing the vertical shaft between port cells and TCWS vault and other ITER buildings. After optimization on the use of the available space, the TBWG has decided to have the following coolant lines: two He-lines for port nb. 16, one water line and two He-lines for port nb. 18 and three He-lines for port nb. 2. Pipes characteristics have been standardized (same diameter, operating pressure, and material).

The rules for the final TBM type selection will have to be defined in future when all required information about R&D results, manufacturing, performances, and funding availability will be known. The final selection of the TBM systems to be installed on day_one could be performed by an ad-hoc group charged with an appropriate mission.

The performed safety analyses, summarized in the report and detailed in the DDDs, do not show any major difficulties. However, as the port cells form a containment volume, overpressure events, such as the rupture of a cooling pipe, have yet to be assessed.

TBM replacement occurs in the ITER hot cell, where the whole TBMs/shield plug system is remotely transported in a standard ITER transport cask. Because of the large number of TBM components present in the port cell, parking spaces need to be added in order to allow RH operation with the transport cask. Either the addition of an hot cell port or the modification of the TBM replacement procedure is required if the simultaneous replacement of the 3 test-port plugs has to be possible during a ITER planned shut down (1 month per year).

The ITER hot cell may be used to replace irradiated TBMs, but it is not designed to allow TBM repair and/or Post-Irradiation Examinations. Current ITER Hot Cell scheme considers the irradiated TBMs as a waste object. The presence of hot-cell facilities on the site selected for construction has also to be taken into account could help to solve this issue. Storage needs shall also be re-evaluated.

Several interface issues between TBMs systems and ITER machine and buildings have been identified and need to be further investigated. The main ones are the following:

- the space and remote handling capabilities in the unique ITER Hot Cell appears clearly insufficient. In fact, TBMs have to be replaced relatively often and quickly and PIE has to be performed on the dismantled TBMs;
- the proliferation of component casks and large permanent devices in the port cell is of concern for ITER. They will require temporary parking space during port plug removal and may affect the number of ports that can be maintained simultaneously;
- the space in the TCWS has become today no more sufficient to install all TBM heat transfer equipments requested by Parties. There may be a similar problem in the T-Plant building.The use of common (shared) equipment could help to solve the issue;
- the impact of the confinement strategy for Port Cells recently adopted in ITER has to be checked.
- the port frames procurement is missing from the released ITER packaging. This point has to be clarified as soon as possible because it is an essential element for allowing the installation of TBMs at the beginning of ITER operation.

11-2 TBWG Recommendations

To meet the dead line of TBMs installation in ITER since the first day of ITER H-H operation, it is now urgent to define an appropriate "TBMs Test Program" organization, in strict collaboration with ITER or as part of it, which should deal with the following:

- to assess, to coordinate and to define "priorities" for test proposals,
- to monitor the corresponding activities performed by Parties and,
- to solve the interfaces issues (frame, space availability, hot cell operation, etc..).

In any case, duty and responsibilities of ITER and Parties concerning TBM Testing Program, should be clearly and urgently defined.

All ITER Parties have declared their interest in TBM program and their intention to deliver TBM systems (or sub-components of them) for the first day of ITER operation. Therefore, adequate funding and staffing should be committed by Parties on TBM program as soon as possible and on multi-annual basis. Moreover, ITER costs associated with the TBM testing program should be recognized and addressed by Parties. The Parties commitment should concern both domestic program and ITER organization.

These recommendations have been presented by the TBWG Chairman at the ITER Preparatory Committee Meeting held in Naka on May 16-17, 2005. The details of the presentation are given in Annex 2.

11-3 Future Work Proposals for TBWG

These proposals concern the near future activities that should be performed by the TBWG or another equivalent technical group. Two main activities appear essential for ensuring the possibility of TBM testing in ITER since the beginning of ITER operation:

1) verify that all the interfaces issues and specific TBWG requirements are integrated in the ITER design before starting construction. Design work in collaboration with ITER Team should be performed in order to find acceptable solutions which can easily be implemented in the ITER construction plan, including procurement. Main items to be further addressed are: 1) the test port frame/port plug design; 2) the need of RH equipment in the port cell and definition of RH strategy, 3) the design of hot cell and hot cell port plug in order to increase the availability of space (e.g., parking space of activated TBM and/or port plug) and of equipment (for PIE requirements).

2) increased interaction between Parties both for R&D collaboration and TBM design proposals assessment. It is expected that the number of TBM proposals can be reduced after detailed technical assessment of all proposals. The assessment could include the identification of the main reasons and justifications for each proposal, the evaluation the main issues associated with each proposal and verify if the various designs allow some, at least partial, convergence. The acceptability of particular choice by the various Parties should be discussed. The assessment should be based on the DEMO design corresponding to each TBMs.

ANNEX 1

Test Blanket Working Group (TBWG)

Terms of Reference for the period of

the ITER Transitional Arrangements (ITA) activities

Proposal to re-establish the Test Blanket Working Group (TBWG) and Terms of Reference for the TBWG work.

1 Proposal

PT/IT Leaders propose to the Preparatory Committee the re-establishment of the Test Blanket Working Group (TBWG) with the scope of activity defined at point 2. Reporting and Membership of the TBWG are defined at point 3 and 4.

2 Scope of TBWG activities

a. The TBWG will be charged to provide the design documentation for the assessment of the Test Blanket Modules (TBM) proposed by the Parties. For each TBM the work shall include:
- To insure the updating of the design documentation (drawings, supporting analyses and tests, etc.);
- To verify the integration of TBM's into the ITER machine, including the physical interfaces, auxiliaries, support facilities, machine operation, safety, waste management, reliability and maintenance requirements;
- To verify the compatibility of their design, R&D development, fabrication and testing with the ITER construction time schedule.

b. The TBWG shall promote the co-operation among the ITER Parties 'test blanket development programmes (including exchange of information and proposals of coordinated activities.)

c. The TBWG shall verify the integration of the TBM's in the safety and environmental evaluations made by each official candidate site. On request of the ITER parties and for any official candidate site, the TBWG shall provide, support for the design adaptation to the site requirements and for the relevant licensing activities.

d. The TBWG will further develop the co-ordinated blanket test programme. The test program must be consistent with the ITER mission as outlined in the ITER SWG Report to the IC on Task #1;
- "ITER should test tritium breeding blanket module concepts that lead in a future reactor to tritium self-sufficiency, the extraction of high-grade heat and electricity production."
- "The option for later installation of a tritium breeding blanket on the outboard of the device should not be precluded."

3 Organisation and Reporting

a. For the activities 2.a, c and d the TBWG shall report to the Interim Project Leader and the Preparatory Committee and inform the PTs leaders. For the activity at point 2.b the TBWG shall report to the PTs leaders.

b. For the scope of activities described at point 2.a, the final deliverable shall be a report to be submitted to the Interim Project Leader containing;
- Design description of the proposed designs for the TBM:
- Assessment of the integration in ITER;
- Assessment of design, R&D and fabrication time schedule;

c. For the scope of activities described at point 2.c, the final deliverable shall be a report to be submitted to the Interim Project Leader, and to the PT leaders indicating for each official candidate site the assessment of the impact compared to ITER base machine of each TBM on the safety and environment evaluation. The assessment will be based in the specific analyses made by the team in charge of each official candidate site.

d. For the scope of activities described at point 2.d the TBWG shall submit to the Interim Project Leader and to the PT Leasers a report defining a test programme in the machine, suitable to fully qualify the proposed TMB designs.

e. Intermediate reports shall be provided, as the necessity shall suggest.

4. Membership

a. The PT Leaders and the Interim Project leader shall designate their representatives, up to three from each Party and four from the ITER International Team.

b. In consultation with the PT Leaders, the Interim Project Leader shall appoint the Chair among the representatives of the Parties and the Co-chair among the representatives of the ITER International Team.

c. The Interim Project Leader and the PT Leaders can participate to the meetings so if so they wish. External experts can participate, by invitation of the Chair, acting in consultation with the Members. A representative of each official candidate site (and in particular the representatives of the “Preferred Site”) shall be invited to the meetings, when required.

5. Meetings

a. As a rule the TBWG will meet every 4 - 6 months.

b. Other meetings can be arranged by mutual agreement of the Chair and Co-Chair.

c. The Meetings should alternate among the ITER work sites; other locations can be selected by mutual agreement of the Chair and Co-chair depending on specific reasons of convenience.

6. Resources

The ITER International Team and the Participant Teams will allocate internal resources for the accomplishment of the TBWG scope of activities.

7. Duration of the TBWG

The TBWG is established for the duration of the ITER ITA activities.

ANNEX 2

Viewgraphs of the TBWG Chairman Presentation at the 4th ITER Preparatory Committee (IPC-4) meeting

(Naka, May 18-19, 2005)

ITER TBWG 1

ITER Preparatory Committee Meeting

Test Blanket Working Group (TBWG)
Main Achievements and Recommendations

L. Giancarli

TBWG Chairman

L. Giancarli, TBWG, Main Achievements and Recommendations Naka, May 19-20, 2005

ITER TBWG 2

Test Blanket Working Group (TBWG)

Tritium Breeding Blanket

Not present in ITER, it is a necessary nuclear component in DEMO (the machine after ITER), with the functions of producing all the Tritium required as fuel (~300 grams/day) and of recovering the heat for electricity production

Test Blanket Modules (TBMs)

It is recognized that the test of breeding blanket mock-ups in ITER is therefore essential

► ITER mission : *"ITER should test tritium breeding module concepts that would lead in a future reactor to tritium self-sufficiency, the extraction of high grade heat and electricity production."* (ITER SWG Report to the IC)

► The ITER TBM Program is a central element in the plans of all six ITER Parties for the development of tritium breeding and power extraction technology.

Consequent need of coordinating the Parties Test Programs and of integrating it in ITER plant and operations without significantly disturbing the main ITER R&D program

➔ Creation of the Test Blanket Working Group

L. Giancarli, TBWG, Main Achievements and Recommendations Naka, May 19-20, 2005

ITER TBWG 3

Test Blanket Working Group (TBWG)

TBWG Membership

► 4 members from the ITER Team (including the deputy director)
► 3 members from each of the six ITER Parties
► experts are invited when required

TBWG Missions

► provide the Design Description Document (DDD) of the Test Blanket Module (TBMs) systems proposed by ITER Parties and assess the interfaces with the ITER machine,

► promote co-operation among participants on the associated R&D programs,

► verify the integration of TBM testing in ITER site safety and environmental evaluations,

► develop and propose coordinated TBM test programs taking into account ITER operation planning.

Recall : TBMs have to be representative of a DEMO breeding blanket, capable of ensuring tritium-breeding self-sufficiency and of accommodating high-grade coolants for electricity production.

L. Giancarli, TBWG, Main Achievements and Recommendations Naka, May 19-20, 2005

ITER TBWG 4

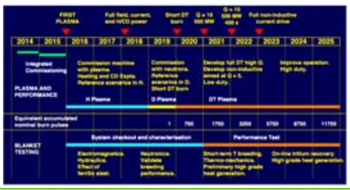
TBWG Main Achievements *(1/4)*

Agreed assessment from TBWG on the interest of TBMs testing program

► ITER is the unique opportunity for testing TBMs prior to DEMO construction,

► taking into account the ITER operation parameters, essential information and data on breeding blankets can be obtained from the tests in ITER,

► TBMs have to be installed since the first day of the H-H operation (to check interfaces & main operations, compatibility with ITER operations and to support to safety dossier)



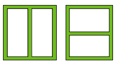
L. Giancarli, TBWG, Main Achievements and Recommendations Naka, May 19-20, 2005

ITER TBWG 5

TBWG Main Achievements *(2/4)*

► 3 ports of ITER are devoted to TBM testing

► typically, the reasonable size of a TBM is half a port *(horizontal/vertical configuration)*
➔ a maximum of 6 TBMs can be tested simultaneously



► several proposals of TBM has been made by ITER Parties:
- ☞ Helium-cooled Li-based Ceramic/Beryllium TBM (4 different designs)
- ☞ Helium-cooled liquid Lithium-Lead TBM (3 different designs)
- ☞ Water-cooled Li-based Ceramic/Beryllium TBM (one design)
- ☞ Liquid natural Lithium TBM (2 different designs)
- ☞ Li-based Molten Salt TBM (one design to be defined at later stage)

► for each of these blanket types a Working Sub-Group has been established in order to define coordinated test program, identify common technical issues and to promote collaboration between Parties

► uncertainties are present on most proposals *(technical, financial, time schedule,...)* therefore TBWG has decided to leave the maximum reasonable "flexibility" on the definition of the TBMs port allocation:
➔ cooling pipes connections at each ITER test port have been fixed taking into account all Parties proposals, however the selection of TBMs will be made later, few years before installation (reasonably, 4/5 years)

L. Giancarli, TBWG, Main Achievements and Recommendations Naka, May 19-20, 2005

ITER TBWG 6

TBWG Main Achievements *(3/4)*

TBM testing is a complex task requiring a strong integration effort in ITER machine and operations

► TBMs are installed in a common water-cooled steel frame, including shield, whose replacement has to be compatible with existing ITER remote handling facilities

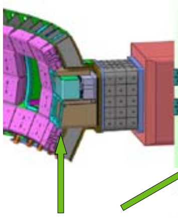
► TBM tests requires several TBM associated systems:
- coolant circuits with pumps and heat exchanger located in the Tokamak Water Coolant System (TWCS) vault
- Tritium extraction system located in the port cell with connection to the Tritium building
- maintenance/dismantling performed with ITER remote handling facilities
- replacement of TBMs performed in the ITER Hot Cell

► for each TBM type, different TBM designs are required *(act-alike TBMs)* for different testing objectives in different ITER phases
➔ need of several planned Port Plug replacements

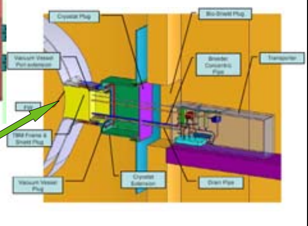
L. Giancarli, TBWG, Main Achievements and Recommendations Naka, May 19-20, 2005

ITER TBWG 7

TBWG Main Achievements *(4/4)*



TBM location in a ITER test port



View of a typical TBM test port cell arrangement

L. Giancarli, TBWG, Main Achievements and Recommendations Naka, May 19-20, 2005

ITER TBWG 8

TBWG General Recommendations

TBM Program Organization

❑ TBWG considers that in order to reasonably guarantee the feasibility of the TBMs program since the first day of ITER operation, TBM systems and interfaces have to be fully integrated in the ITER machine and buildings and the financial support corresponding to this integration has to be included in the negotiation and procurements packages.

❑ Because of the complexity of the task and the required integration in ITER, the planning of TBM testing and its execution requires a structured framework among the Parties either as a new arrangement under ITER Joint Implementation Agreement or as a part of it.

On this matter, the share of the rights and responsibilities between the Parties and ITER have to be clearly established.

Examples of possible cases:
- ITER seen as a "TBM user facilities" *(difficult, considered the complexity of the TBM system and the required level of integration)*
- TBM Test Program as "part of ITER" *(large effort from ITER Team required)*
- Mixture of the two previous cases

L. Giancarli, TBWG, Main Achievements and Recommendations Naka, May 19-20, 2005

9

TBWG Technical Assessment

Main identified technical issues needing immediate action for fulfilling the TBM test mission of ITER

- Design and manufacturing of TBM interfaces with ITER, such as test port plug (frame, shield) and system connections need to be finalized either by ITER Team or by Parties with appropriate funding (at present, not included in the ITER procurement packaging).
- TBMs have to be replaced several times, often simultaneously. At present, only one access port is available in the single ITER hot cell. Therefore, the "hot cell replacement" operation is a "bottle neck" for TBM testing.
TBM replacements have to be performed during the planned ITER shutdowns (1 month), therefore feasibility appears questionable without adaptation of the hot cell features.
- Presence of large TBM permanent components and piping in the test port cell could require temporary parking space during port plug removal (not available at present).
- To fully exploit the tests, TBMs need post-irradiation examinations (PIE). No facilities are at present available in the hot cell for these operations (not even for TBM cutting).
- A large R&D effort on TBM is required by Parties prior to TBM installation (functional demonstration, pre-testing, quality assurance and control, etc…) within the fixed deadline.

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Test Blanket Program Impact on ITER

Potential impact of the TBM Test Program on ITER team activities to be taken into account by Parties

- Assessment of the impact on ITER operations
  - check compatibility with plasma *(e.g., ferromagnetic materials, leakages)*
  - verify feasibility of interfaces *(i.e., frame, shield, remote handling, connections)*
  - verify TBMs acceptability *(i.e., safety, codes and standards, reliability)*
  - and, in general, verify that ITER performances are not degraded by TBM tests
- Ensure that TBM testing has no impact on ITER schedule
  - verify, through monitoring of TBMs development program, that appropriate TBMs will be available at the expected date
- Consequence on ITER cost (specific facilities, staffing, etc…)
  - ITER has to perform quality assurance and control of the TBM systems in all stages of engineering design/fabrication/TBM pre-testing and acceptance tests
  - ensure the availability remote handling facilities
  - ensure the availability of interfaces *(e.g., frame, connection, space in TWCS vault and in Hot Cell)*
  - as a minimum requirement, ITER has to ensure coordination between different TBMs test program *(use of test ports, space, transport to Hot Cell, Hot Cell utilization)*

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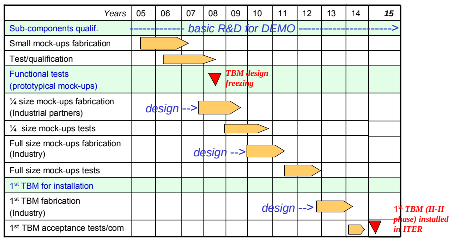
TBWG Assessment on Parties Contributions *(1/2)*

Time schedule and Cost effort for Domestic Program

- Typical (tight) schedule needed for a TBM system before installation in ITER on day_1:



- Typical cost: from EU estimation, about 80 M$ per TBM type (up to commissioning)

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TBWG Assessment on Parties Contributions *(2/2)*

Requirements to all Parties interested on participating to the TBM Test Program and potential impact on Domestic Program

- At present, only TBM conceptual designs are available, supported by R&D results at laboratory scale to proof "performances", therefore a large effort is still required
- Proposed schedule for delivery of TBMs & associated systems should be coherent with the proposal of TBM system installation in ITER at the beginning H-H operations
- Parties should guarantee the feasibility of the proposed time schedule and deadline with the commitment of the appropriate financial support and staffing level
- The establishment of framework for collaboration between ITER Parties *(two Parties or more)* is required (including the intellectual property issue) in order to reduce the number of TBM types to be tested and to reduce their cost and time schedule

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Test Blanket Working Group (TBWG)

Summary of Main Recommendation

- ITER is an unique opportunity to test breeding blankets and extremely valuable information can be obtained
- To meet the dead line of TBMs installation in ITER since the first day of ITER H-H operation, it is now urgent to define an appropriate "TBMs Test Program" organization, in strict collaboration with ITER or as part of it, that should deal with the following:
  i) to assess, to coordinate and to define "priorities" for test proposals,
  ii) to monitor the corresponding activities performed by Parties and,
  iii) to solve the interfaces issues (frame, space availability, hot cell operation, etc..).
  In any case, duty and responsibilities of ITER and Parties concerning TBM Testing Program, should be clearly and urgently defined
- All ITER Parties have declared their interest in TBM program and their intention to deliver TBM systems (or sub-components of them) for the first day of ITER operation.
  Therefore, adequate funding and staffing should be committed by Parties on TBM program as soon as possible and on multi-annual basis.
  Moreover, ITER costs associated with the TBM testing program should be recognized and addressed by Parties .

→ The Parties commitment should concern both domestic program and ITER organization.

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