Test Strategy and Development Achievements of ITER Solid Breeder Test Blanket Modules in Japan

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Abstract. This paper presents progresses of the test strategy development, design and supporting R&Ds of solid breeder Test Blanket Modules (TBMs) in Japan. Japan is proposing to test its unique designs of Water Cooled Solid Breeder (WCSB) TBM and Helium Cooled Solid Breeder (HCSB) TBM from the beginning of the ITER operation. Water cooled solid breeder blanket has the better capability of heat removal and applicability to the more compact and economical fusion reactors. On the other hand, if it uses beryllium pebble beds as the neutron multiplier, it has the potential safety concern of thermo-chemical excursion of water and beryllium reaction in the case of the coolant water ingress in the beryllium pebble beds. Safety analysis of the WCSB TBM showed the possible preventive measures to such a case in TBM tests in ITER. Also, to void such potential safety concern, the advanced multiplier material, Be-Ti alloy, is under development in Japan, because it has very low reactivity with water even in high temperature. Helium gas coolant has less heat removal capability but higher safety to thermo-chemical excursion phenomena. As for the design work, structure design showed steady progress and clarified detailed structure taking into account the fabrication procedure. As for supporting R&Ds, the corrosion characteristics of the structural material by high temperature and pressure water was clarified as one of critical structure integrity issues. Also, important design data of the breeder pebble bed has been clarfied. Along with the development progress, the test strategy has been investigated to obtain the most effective results of TBM test program.

1. Introduction

The Fusion Council of Japan has established the long-term research and development program of the blanket in 1999. In the development program, Japan Atomic Energy Agency (JAEA) has been designated as a leading institute for developing a solid breeder blanket system, which is the first candidate blanket type for DEMO fusion plant in Japan. According to the development plan, JAEA is performing R&D and design development of DEMO blankets and Test Blanket Modules (TBMs). TBM test is the unique module scale tests of blankets in the fusion reactor environment using ITER. The development and testing of TBM in ITER are the most important milestones toward DEMO fusion power plant. This paper presents the latest achievement of the development of test strategy, design and technology development of solid breeder TBMs in Japan.

2. Test Strategy of Test Blanket Modules

In the development program of fusion blankets, the ITER TBM test is regarded as one of the most important milestones, by which integrity of blanket concepts and structures are qualified. Prior to the ITER TBM test, the out-of-ITER R&Ds should be extensively performed so as to construct the TBMs on a sound technical basis and to conduct testing in ITER efficiently with minimum impact to the ITER operation. Based on the above premise, the tests in ITER will be dedicated mainly to the investigation of blanket characteristics under 14 MeV neutron irradiation, strong electromagnetic field and their synergetic effects including high surface heat flux to the module first wall, and finally to the demonstration of the blanket performance to be applied to a DEMO reactor. A series of the ITER TBM tests are proposed, according to the ITER operation phases, namely, “system checkout and electro-magnetic test” during H-H phase, “neutronics test” during D-D and limited D-T phase, and “performance tests” during low duty and high duty D-T phases [1]. For all tests, dedicated module designs are proposed to obtain the most effective test results with the layered pebble bed structure concept which is proposed in the DEMO blanket design.

The primary candidate blanket is supposed to be solid breeder blankets for DEMO. Therefore, Japan is proposing to deliver two types of Solid Breeder TBMs, Water Cooled Solid Breeder (WCSB) and Helium Cooled Solid Breeder (HCSB) TBMs from the first day of ITER operation and to perform module testing. According to this plan, design work has been performed on both TBMs.

3. Design and Performance Analyses of Test Blanket Modules

3.1. Module Structure

The design of Solid Breeder TBMs of Japan has the following features.

(1) 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.

(2) Vertical slots were 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 plasma disruption and increasing the endurance to internal over-pressure in the case of coolant ingress in the module. Sub-modules are integrated at rear wall by welding.

(3) For HCSB TBM, a 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.

(4) 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 the DEMO blanket [2].

Table I summarizes the major specification of the WCSB TBM and the HCSB TBM proposed by Japan. *FIGURE 1* shows the schematic three dimensional drawing of the WCSB TBM composed by detailed drawings [3]. In the WCSB 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

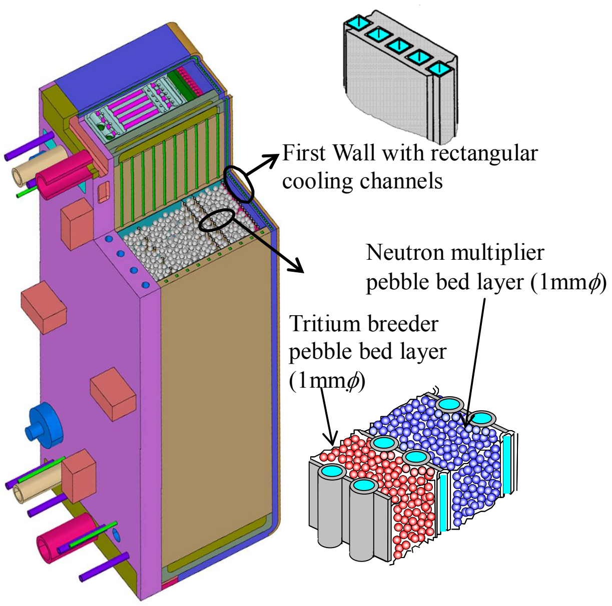


FIG. 1. Structure of Water Cooled Solid Breeder TBM [3].

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. The coolant of TBM will be supplied from the dedicated Water Cooling System for the WCSB TBM.

3.2. Water Cooling System for the WCSB TBM

Design conditions of the cooling system for the WCSB TBM are summarized in Table I. Thermal power (removal heat) of the TBM is 0.904 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 3.59 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.

Major components in the main loop are a steam generator, a main heat exchanger, a circulation pump, a pressurizer and two heaters. For circulation pump system, two circulation pumps are planned for redundancy in case of a pump trip accident. 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. One of compensation heater output of 0.904 MW is equipped between the TBM outlet and the steam generator inlet to provide the power to

TABLE I: DESIGN CONDITIONS OF SOLID BREEDER TBMS OF JAPAN

Items	Unit	WCSB	HCSB
Structural Material		F82H*	F82H*
Coolant		Pressurized Water	Helium Gas
Neutron Multiplier		Be (pebble)	Be (pebble)
Temperature Limit	ºC	< 600	< 600
Tritium Breeder		Li_2TiO_3 (pebble)	Li_2TiO_3 (pebble)
Temperature Limit	ºC	< 900	< 900
Area of First Wall	m^2	0.484×1.66	1.208×0.71
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	0.904	0.988
Total Tritium Production	g/FPD	0.134	0.157
Coolant Pressure	MPa	15.5	8.0
Coolant Inlet Temperature	ºC	280.0	300.0
Coolant Outlet Temperature	ºC	325.0	500.0
Coolant Flow Rate	kg/s	3.59	1.1
Coolant Bypass Flow Rate	kg/s	—	0.33

* Reduced activation ferritic steel

compensate the power reduction of the TBM during the dwell time. Another heater 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.

3.3. Helium Cooling System for the HCSB TBM

Design conditions of the cooling system for the HCSB TBM are summarized in Table I. Thermal power (removal heat) of the TBM is 0.988 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 1.1 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.

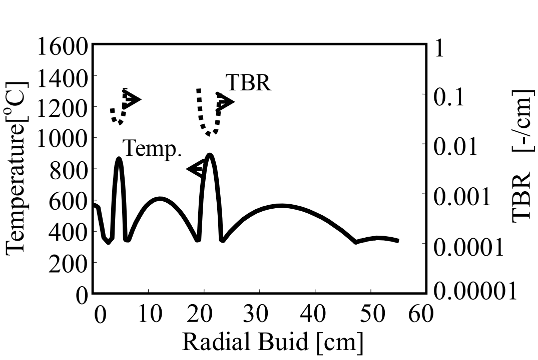
3.4. Tritium Recovery and Measurement Systems

The major functions required to the Tritium Recovery System of the TBMs 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.
One tritium recovery system for both TBMs is designed, with multi-point gas analysis system for obtaining tritium release data and system control. Some of gas analysis equipments are to be installed within a transfer cask (about 1 m^3) in the Port Cell area in ITER Tokamak building. The primary option of the tritium recovery process is the Cryogenic Molecular Sieve Bed (CMSB) system, which is well established process of hydrogen isotope recovery from He flow. 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 gas analysis system equipped at the outlet of each TBM (in front of the bio-shield 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. Components of Tritium Recovery System will be contained in glove boxes and installed in Tritium Building of ITER.
Since the cryogenic molecular sieve bed system require relatively large size of apparatus and batch wise operation, development of advanced tritium recovery system is also being performed, based on the electro-chemical hydrogen pump system, which require smaller component size and enables continuous operation. Depending on the progress of the development, the advanced option will be applied.

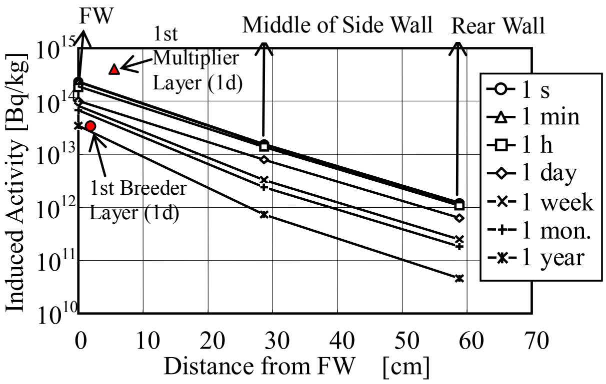
3.5. Performance Analyses

a) Neutronics analysis

One dimensional neutronics and thermal analysis were performed for initial determination of Tritium Breeding Ratio (TBR) and layer thickness using ANISN. *FIGURE 2 (1)* shows one dimensional TBR and temperature analysis for the WCSB TBM. In the calculation,

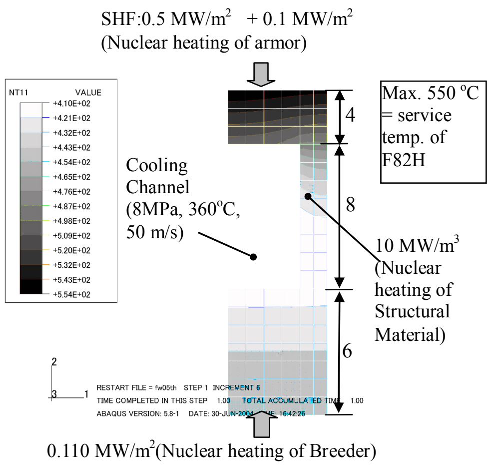


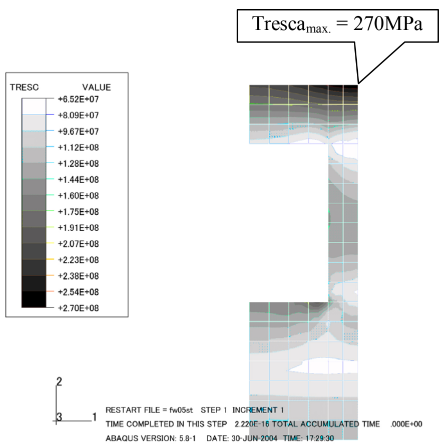
(1) TBR and Temperature Distributions in Radial Direction By One Dimensional Analysis for the WCSB TBM.



(2) Induced Activity of the WCSB TBM after 0.3 MWa/m^2.

FIG. 2. Results of Neutronics Analysis.





(1) Temperature distribution in the cross section of first wall of the HCSB TBM

(2) Stress distribution in the cross section of first wall of the HCSB TBM

FIG. 3. Results of Thermo-mechanical Analysis of the First Wall of the HCSB TBM.

thicknesses of breeder and multiplier bed were decided so that the highest temperature of structural material, breeder pebble bed and multiplier pebble bed are below 550, 600, 900 °C, respectively. Total TBR of 1.42 are estimated for the WCSB TBM.
Two dimensional neutronics analysis was performed for more detailed design data on the distributions of nuclear heating rate, tritium breeding ratio, induced activity and decay heat. The analysis model incorporated the common frame as the surrounding structure of the TBMs. For calculation of neutron transport, DOT3.5 with cross section library, FUSION-40 (JENDL2.1 edition) was employed. For calculation of nuclear reaction, APPLE-3 was used. For induced activation rate calculation, ACT-4 with CROSS-LIB, CHAIN-LIB and GAMMA-LIB ('90 edition) was applied. *FIGURE 2 (2)* shows the estimated induced activity after 0.3 MWa/m^2 ITER operation. The estimated induced activity is 1 order of magnitude lower than that of DEMO fluence, 10 MWa/m^2.

b) Thermo-mechanical analysis

Thermo-mechanical integrity of the TBM structure is one of the most important issues. The first wall needs to withstand the maximum surface heat flux, 0.5 MW/m^2. The temperature and stress distribution in the first wall of Water Cooled TBM have been evaluated by 2 dimensional thermo-mechanical analysis. The analysis result showed that temperature and stress of the structure satisfied the F82H design window [4]. In the case of the first wall of the

HCSB TBM, the highest temperature of the structural material, 550 °C, which satisfies the F82H design window, appeared at the most distant part of plasma side surface from cooling channel (see *FIG. 3 (1)*). *FIGURE 3 (2)* shows the stress distribution in the first wall of the HCSB TBM evaluated by two-dimensional thermo-mechanical analysis. By stress analysis, it was clarified that the highest TRESCA stress 270 MPa appeared at the same place as the highest temperature appeared. This stress value was evaluated to satisfy 3Sm value for F82H.
For the case when the coolant water ingresses into the TBM box, 3 dimensional FEM analysis has been performed. The result of elasto-plastic analysis for the WCSB TBM showed that it can be possible to prevent plastic collapse. The displacement and the stress values of the elastic analysis of the larger TBM dimension were more significant than the analysis result of updated dimension. Therefore, it can be expected that the total failure of the WCSB TBM module box is prevented even in the over-pressurization of 15 MPa at 300 °C.

c) Tritium release behavior

Behavior of bred tritium in the breeder pebble bed is affected by complex combination of mass transfer processes such as, diffusion in solid grain, humidity adsorption on surface, gas-liquid exchange reaction on the grain surface with adsorbed humidity and exchange capacity in solid. From the view point of tritium control, optimization of purge gas conditions, such as, water humidity, H_2 swamping concentration, purge gas velocity, need to be optimized. Preliminary analysis of tritium inventory and release of solid breeder pebble bed of JA solid breeder TBMs have been performed by using the integrated analysis model developed by Kinjo and Nishikawa et al.[5].

d) Preliminary safety analysis

Preliminary safety analyses of TBMs have been performed according to the hypothetical event sequences to identify ultimate safety margin in ITER. Three initiating events were given, (1) In-vessel pipe breaks of TBM and other first wall shielding blanket modules, (2) Large ex-vessel TBM coolant leak, (3) Pipe break in TBM. The safety analyses of the WCSB TBM and HCSB TBM have been performed according to the above mentioned 3 cases.
For the WCSB TBM, Be – steam reaction in high temperature is the most critical safety issue, because it is exothermic reaction and produces hydrogen. In Case (1), the total holdup of coolant is very small compared to the volume of the Vacuum Vessel. Therefore, the pressurization is small. Also, plasma can be expected to distinct by coolant ingress. Cases (2) and (3) lead to the most severe case, where contact of coolant water and Be pebbles and loss

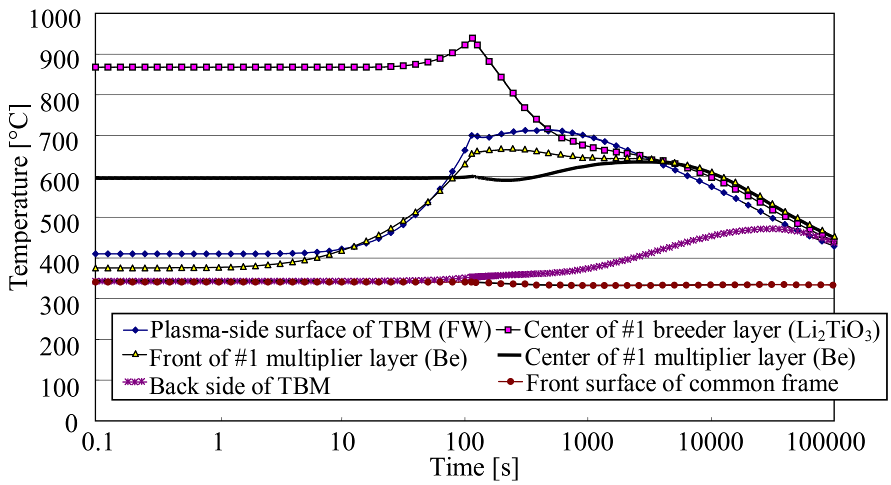


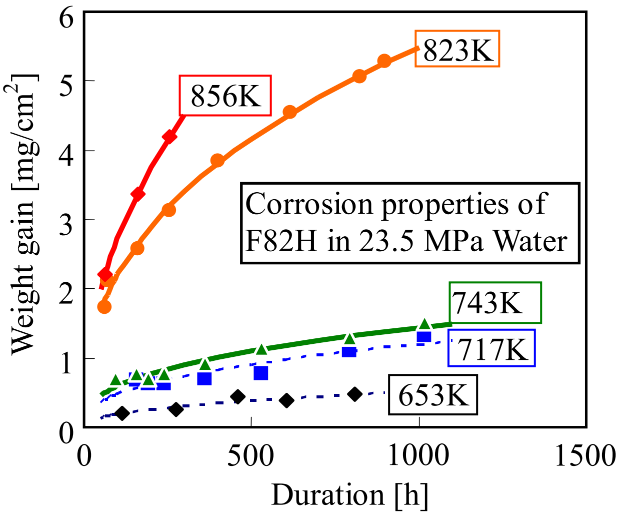
FIG. 3. Temperature transient of TBM, pipe break in TBM.

of cooling capability must be assumed simultaneously. Calculation was performed to evaluate the temperature transient and the hydrogen production. In calculation, one dimensional temperature analysis was performed for the case of loss of coolant of the TBM during the ITER operation, incorporating the reaction heat of Be and water. Calculation has been performed to investigate the time limit to terminate the plasma operation to avoid thermo-chemical reaction excursion in Be pebble bed. By the calculation, it was clarified that the plasma needs to be terminated in 115 sec, when the surface of the first wall reaches 700 °C. *FIGURE 3* shows calculated results of temperature transient in the WCSB TBM when plasma was terminated in 115 sec after the event started. By the calculation, it was shown that temperature of the TBM could be stabilized to about 450°C in 100,000 seconds. By the safety analysis of the HCSB TBM, it was shown that the temperature is stabilized safely, assuming that the plasma operation is stopped when the temperature of Be armor tile becomes 1150°C. To void such potential safety concern, the advanced multiplier material, Be-Ti alloy, is being developed in Japan, because it has very low reactivity with water even in high temperature [6].

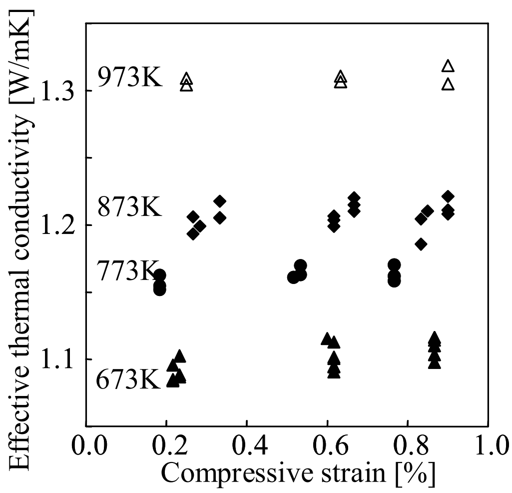
4. Supporting R&Ds

Essential issues of supporting R&Ds are categorized into four R&D subjects; Out-pile R&D, Tritium Recovery System Development, In-pile R&D including development of breeder/multiplier fabrication technology [6] and Neutronics and Tritium Production Tests with 14 MeV Neutrons [7]. The achievements of the latter two R&D subjects will be presented in this conference.

The Out-pile R&D has been performed mainly on the development of the blanket module fabrication and the development of thermo-mechanical design database of breeding region of the blanket. For fabrication of the first wall with embedded cooling channels, the fabrication method of rectangular cooling tubes was investigated and certified by applying cold rolling, for the first step of the fabrication of the first wall of the prototype TBM. Also, the corrosion rate of F82H in high pressure and temperature water was measured and prospect to the TBM operation was obtained, as seen in *FIG. 4 (1)* [8]. For the thermo-mechanical design of the breeder layer, the influence of the compressive stress on effective thermal conductivity of



(1)Weight changes of RAFM, F82H, in supercritical pressure water[8].



(2)Effective Thermal Conductivity of a Compressed Li_2TiO_3 Pebble Bed[9].

FIG. 4. Recent Results of Out-pile R&D.

breeder pebble bed was investigated. It was clarified that the relationship can be well correlated by the strain under various bed temperature [9]. Also, the relationship between the packing fraction and effective thermal expansion coefficient of Li_2TiO_3 pebble bed was measured. It was clarified that average thermal expansion coefficients are constant for the beds with different packing factors, which are in expected range of the packing fraction in the test blanket module.

For the Tritium Recovery System Development, the electrochemical hydrogen pump using the proton conductor membrane has been investigated as the process technology to recover tritium and tritiated water from large amount of blanket purge gas. It was shown that hydrogen isotopes can be recovered from the gas with very low hydrogen partial pressure and even from the humidity in the blanket purge gas flow [10]. Also, the basic operation characteristics of the electrochemical hydrogen pump were clarified [11].

5. Conclusions

(1) Test strategy of the Solid Breeder TBMs of Japan is defined, taking into account the ITER operation modes and expected data for extrapolation to DEMO blanket conditions.
(2) Design of TBMs covered major critical issues. Design integration is being continued.
(3) Technology R&Ds are showing steady progress in all essential areas, Out-pile R&D, Tritium Recovery System Development, In-pile R&D including development of breeder/multiplier fabrication technology and Neutronics and Tritium Production Tests with 14 MeV Neutrons.

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