Status of Fusion Blanket Irradiation Study in JAERI

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An experimental reactor such as ITER is planning various tests using several kinds of blanket designs in addition to demonstrating the physics of burning D-T plasmas. The data of neutron irradiation performance of a blanket is needed for the fusion blanket design. Studies for an in-pile functional test of a blanket mock-up and the development of blanket materials are being continued in the JMTR of Oarai Establishment in JAERI. The present status of these studies is briefly described in this report.

1. Introduction

Tritium release experiments of tritium breeding materials have been carried out to evaluate the temperature dependency of tritium release[1-4]. However, the engineering data (temperature control characteristics, transient behavior on pulse operation and tritium leakage to coolant) for blanket design were not obtained in these experiments. To get these engineering data, studies on the in-pile functional tests of a blanket mock-up and development of blanket materials are being continued in JMTR (Japan Materials Testing Reactor) in JAERI.

Schedule of in-pile functional test is shown in Fig. 1. In-pile functional tests of "phase 1" are planned for ITER driver blanket and "phase 2" for ITER test module (for DEMO). "Phase 1" will be carried out under the conditions of helium sweep gas. "Phase 2" will be carried out under the conditions of high temperature and pressure water cooling with helium sweep gas. Condition of "phase 1 and 2" and ITER test module for DEMO are shown in Table 1. Data of these in-pile functional tests will contribute for the design of ITER or DEMO blanket. Present status of an in-pile functional test "Phase 1" and the development of blanket materials are described in this report.

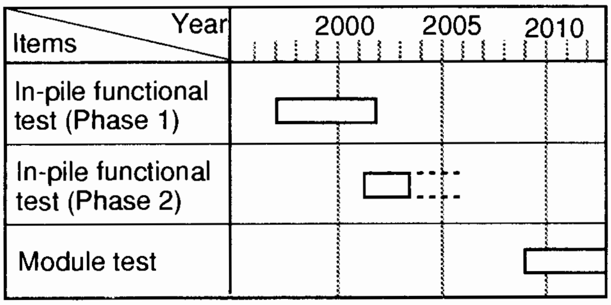

Fig. 1 Schedule of in-pile functional test.

Table 1 Irradiation conditions of in-pile functional test with JMTR and module test with ITER

Items	In-pile functional test (Phase 1)	In-pile functional test (Phase 2)	Module test
Sweep gas	He $+H_2$	He $+(H_2$ or H_2O or $O_2)$	He $+H_2$
Coolant	He gas 0.2MPa 300°C	Water ~15MPa 300°C	Water ~15MPa 300°C
Structure material	SUS316	F82H*	F82H*
Capsule diameter (mm)	65	140	-

* Low activation ferritic steel

2. In-pile functional test

The system of in-pile functional test "phase 1" is shown in Fig. 2. In-pile functional test "phase 1" will be executed with a blanket mock-up and a sweep gas system.

2.1 Blanket mock-up

There are three types of mock-ups which are prepared for the in-pile functional test "phase 1" : a mock-up simulating the ITER blanket structure which has been proposed by Japan site (multi-layered pebble-bed mock-up), a mock-up which simulates the ITER operation pattern (pulse-operation simulating mock-up) and a mock-up loading a lot of breeder material pebbles (kg-loading mock-up). The cross sections of these mock-ups are shown in Fig. 3. The multi-layered pebble-bed mock-up is considered to investigate the temperature control characteristic. The pulse-operation simulating mock-up is considered to investigate the tritium release property under pulse-operation simulating. The kg-loading mock-up is considered to investigate the mass transfer property.

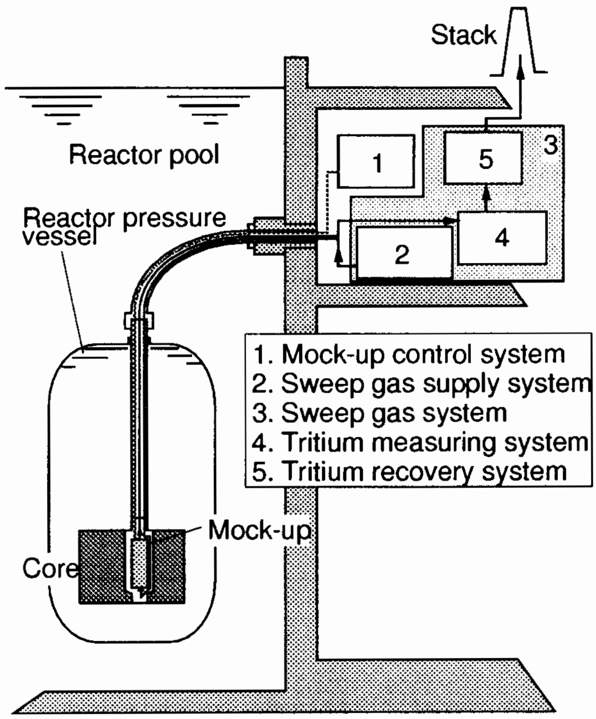

Fig. 2 The system of in-pile functional test "phase 1".

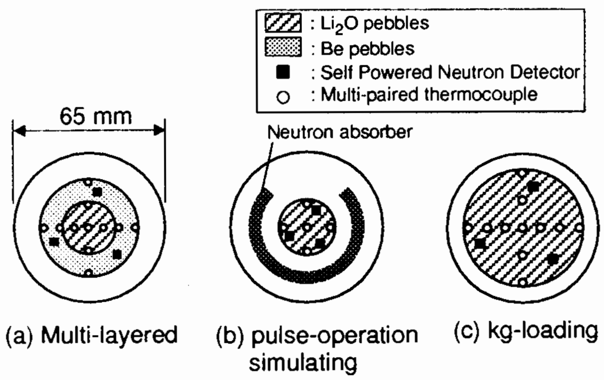

Fig. 3 The cross sections of mock-ups.

The conceptual design of multi-layered pebble-bed mock-up for vertical cross section is shown in Fig. 4. Multi-paired thermocouples are arranged in the pebble-bed. Multi-paired thermocouples have 3 hot junctions for vertical direction, and also measure the vertical direction temperature distribution in the pebble-bed in detail. Multi-layered pebble-bed mock-up will be irradiated by using L-4 irradiation hole of JMTR. ^{6}Li enrichment is selected as 1.0 at.% because of simulating the depression of thermal neutron and the restrict of tritium treatment. Tritium production rate of multi-layered pebble-bed mock-up is about 140 GBq/d (3.8 Ci/d).

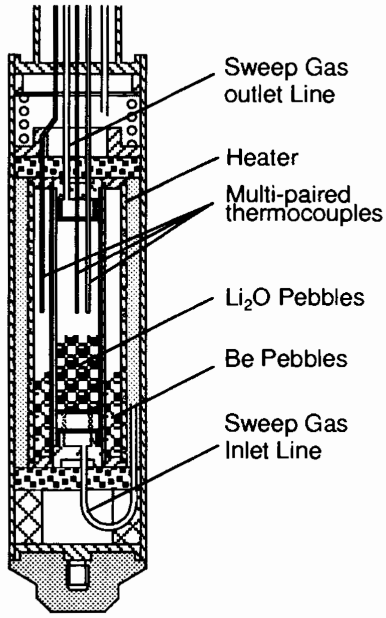

Fig. 4 The conceptual design of multi-layered pebble-bed mock-up for vertical cross section.

Preliminary test using capsule had been executed to design the pulse-operation simulating mock-up. This capsule consists of the Hf neutron absorber with a window, stepping motor made by MI (Mineral Insulator) cable for rotating neutron absorber and SPND (Self Powered Neutron Detector). Hf neutron absorber was rotated by the stepping motor in the core of JMTR. Output of SPND is shown in Fig. 5. This result shows that simulation of pulse operation is possible by using stepping motor and Hf neutron absorber.

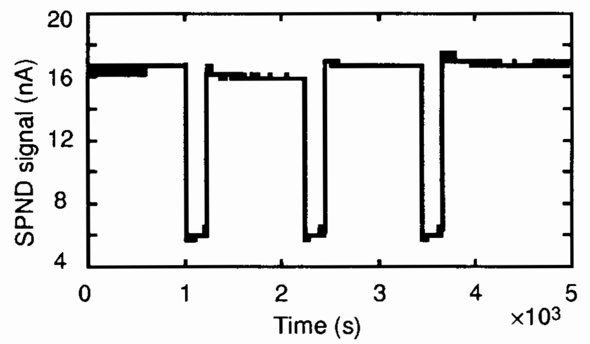

Fig. 5 SPND signal of pulse-operation simulating mock-up.

2.2 Sweep gas facility

The sweep gas facility is a system to measure and recover the tritium released from the blanket mock-ups[5]. The facility mainly consists of four systems: a sweep gas supply system, a tritium measuring system, a tritium recovery system and a clean up system of a glove box.

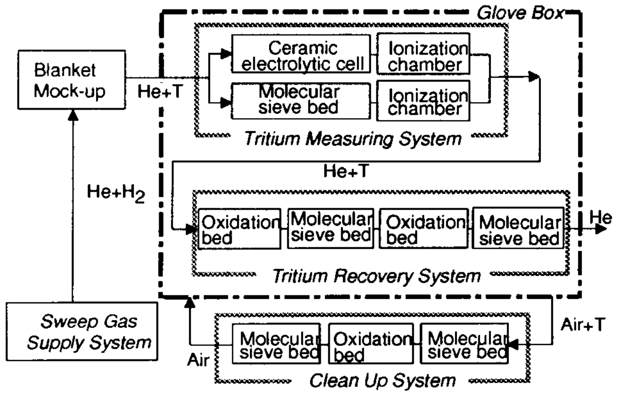

Fig. 6 Block diagram of the sweep gas system.

The block diagram of these systems is shown in Fig. 6, and the main design conditions are listed in Table 2.

The sweep gas supply system is for providing the sweep gas to the blanket mock-up. The sweep gas flow rate can be changed from 10 to 1,000 cc/min. Hydrogen can be annexed in the sweep gas. The hydrogen concentration in the sweep gas can be changed from 10 to 10,000 ppm by controlling a mass flow meter.

Table 2 Design conditions of He sweep gas system

Items	Design condition
Operation method	Once through
Sweep gas	Helium
Hydrogen concentration	10~10,000 ppm
Flow rate of sweep gas	10~1,000 cc/min
Oxidation method	Pd-catalyst
Reducing method	Ceramic electrolytic cell
Total leak rate	5×10^{-6} atm-cc/s
Exhaustion of tritium	$< 1\times10^{10}$ Bq/y

In the tritium measuring system, the total tritium concentration, the tritium concentration of gaseous species and the water species/gaseous species ratio of tritium released from the mock-up are able to be measured on-line. The sweep gas introduced into this system from the blanket mock-up is divided into two lines. One line is to measure the elapsed change of the total tritium concentration. In this line, the tritium in the sweep gas is transformed into tritium gaseous species by a ceramic electrolytic cell and the tritium concentration is measured by an ionization chamber. Two ceramic electrolytic cells are installed into this line so that the transformation from water species to gaseous species is done perfectly and the increase of background by the adsorption of tritium water species on the ionization chamber is prevented.

The other line is to measure the elapsed change of tritium gaseous species. After tritium water species in the sweep gas is removed by a molecular sieve bed, tritium gaseous species is also measured by an ionization chamber. The size of the molecular sieve bed has been determined carefully so that the time lag between the above two lines does not occur. The water species/gaseous species ratio of the released tritium can be obtained from the measurement results of two ionization chambers.

In the tritium recovery system, the tritium in the sweep gas is transformed into water species by an oxidation bed with Pd-catalyst and is recovered by a molecular sieve bed. The reason why the molecular sieve bed is used to recover the tritium is easiness to store without apprehension of tritium permeation. One molecular sieve bed has a sufficient capacity of recovering 37 TBq which corresponds to JMTR operation period for one year. Two sets of the oxidation bed and the molecular sieve beds are installed in series, because the tritium exhaust from the stack is restricted less than 10 GBq/y.

The tritium measuring system and the tritium recovery system are set up in a glove box. The total leak rate from the valves and the connections between pipes and apparatuses in the glove box is suppressed less than 5×10^{-6} atm-cc/s so that the internal exposure is suppressed less than 1 mSv/week. However, the clean up system of the glove box is set up to prepare for the worst. The maximum tritium release rate from the blanket mock-up is defined to be 180 GBq/d. When tritium of 180 GBq, corresponding to released tritium in one day, is assumed to be spreaded out in the glove box, this system has ability of reducing the tritium concentration in the glove box to 7.0×10^{-1} Bq/cm^3 within eleven hours.

3. Development of blanket materials

For the development of blanket materials, tritium breeder and neutron multiplier are studied on the points of the production process, the evaluation of characteristics under the neutron irradiation and reprocessing.

To evaluate the characteristics under the neutron irradiation and reprocessing, facility for the post irradiation examination of neutron irradiated blanket materials was constructed in the hot laboratory of JMTR [6]. This facility named "Beryllium PIE facility", consists of the four glove boxes, dry air supplier, tritium monitoring and removal system, storage box of neutron irradiated samples. Maximum tritium handling are 7.4 GBq/d (200 mCi/d). Ventilation system of beryllium PIE facility is shown in Fig. 7. Tritium release apparatus, laser flush apparatus of thermal properties measurement and mechanical test apparatus are already installed in glove box No.1 (GB-1), No.4 (GB-4) and No.5 (GB-5), respectively. Apparatus of reprocessing will be installed in glove box No.2/3 (GB-2/3) in near future.

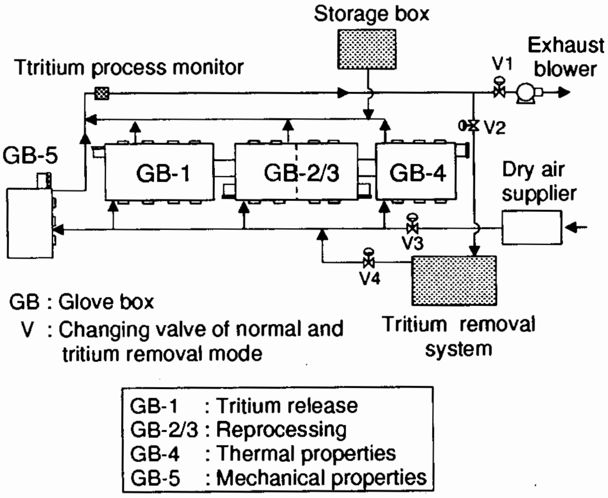

Fig. 7 Ventilation system of beryllium PIE facility.

3.1 Tritium breeder

Lithium-containing ceramics have been considered as candidates for solid breeder materials for fusion reactors. In lithium ceramics, lithium oxide (Li_2O) is one of the best tritium breeders from the standpoint of high lithium density and high thermal conductivity. Several studies have been carried out on the fabrication of small spherical forms to reduce the induced thermal stress in the breeder. Melting granulation method [7,8], and rotated granulation method [9] have been already established in fabrications of ϕ 1 mm Li_2O pebbles. From the point of Li reprocessing, new fabrication method are studied by using sol-gel method [10]. The sol-gel method is best for fabrication of Li_2O pebbles from the reprocessing solution with lithium. And the cost of the fabrication of pebbles will decrease by this method. In the present work, preliminary fabrication tests of Li_2O pebble are carried out by the sol-gel method.

The fabrication process is shown as follows:

a) Fabrication of gel-pebbles
The liquid mixture of Li_2CO_3 and poly vinyl alcohol (PVA) is dropped in cooled acetone through a nozzle and the gel-pebbles is fabricated.

b) Calcination of gel-pebbles
PVA in the gel-pebbles is removed and the Li_2CO_3 pebbles are fabricated in this process.

c) Thermal decomposition and sintering
The Li_2CO_3 pebbles are heated in vacuum and then transformed to Li_2O pebbles and sintering.

From the result of this test, bright prospects are obtained concerning the basic fabrication process of Li_2O pebbles by sol-gel method.

Table 3 Characterization of Li_2O pebbles

Methods	Diameter (mm)	Density (% T.D.)	Grain size (mm)	Strength (kgf)
Melting	0.965	73.9	8~38	2.39
Rotating	0.898	84.3	10~50	1.59
Sol-Gel	1.09	71.0	-	1.03

- : not measured

Table 4 Experimental results of lithium ceramics dissoluted with H_2O and various acids

	Solution			
	H_2O	HNO_3	H_2SO_4	HCl
Li_2O	G	G	G	G
$LiAlO_2$	I	G	D	D
Li_2ZrO_3	I	I	D	P
Li_4SiO_4	D	G	G	G
Li_2TiO_3	I	G	N	N

G :Good dissolution and handling
D :Disslution but handling is not easy
P :Poor dissolution
I :Insoluble solids
N :No experiments

The characteristics of lithium ceramics pebbles fabricated by these method were shown in Table 3. The density of Li_2O pebble fabricated by rotating granulation method was 84.3% and the density of Li_2O pebble fabricated by three methods was above 70%. Recently, fabrication tests of Li_2TiO_3 as a secondary candidate material were also started. Produced lithium-contained ceramics will be irradiated at high fluence by using Phenix reactor of the irradiation test for international collaboration.

Reprocessing technology development for tritium breeder materials has been proposed to recover lithium for effective resource use and to remove radioactive isotopes[11]. Five potential ceramic breeders were investigated to evaluate the dissolution and recovery properties of lithium. Experimental results of lithium ceramics dissoluted with H_2O and various acids is shown in Table 4.

3.2 Neutron multiplier

For the neutron multiplier, production technology of beryllium pebbles are almost established by using the rotating electrode method [12], and the evaluation of characteristics under the neutron irradiation and reprocessing is carrying out using Beryllium PIE facility.

In the tritium release experiments from neutron-irradiated beryllium, the tritium release rate was strongly affected by an oxide layer of beryllium surface. A tritium release model was constructed with taking the growth of surface oxide layer into account [13], and calculational results were compared with the experimental data. Fairly good agreement was obtained between the calculational results and experimental data, and the determined tritium diffusion coefficients in beryllium and beryllium oxide showed almost the same values as in literatures [14-17]. Diffusion coefficients of hydrogen isotopes in beryllium are shown in Fig. 8. "Disk" and "pebble" in this figure show effective diffusion coefficient. "Be" and "BeO" in this figure were used as diffusion coefficients of the bulk beryllium and surface oxide layer in the calculation of surface oxide layer growth model. Neutron fluence and irradited temperature of measured samples were total fast neutron fluence (E>1MeV) of about 10^{20} n/cm^2 and about 200°C in helium.

Fracture strength of un-irradiated beryllium is affected by the grain size, anisotropic slip and impurities (especially O, Al etc.). For the neutron irradiated beryllium, it is clear that fracture strength has a peak of fast neutron fluence at about 1×10^{21} n/cm^2 and decreases by temperature[12,18]. And Takeda's data [19] shows impurity of Fe affects fracture strength.

By using mechanical test apparatus of Beryllium PIE facility, neutron-irradiated beryllium with different grain size had been tested at room temperature. Fracture stress of neutron irradiated beryllium as a function of grain size is shown in Fig. 9. Dependence of grain size was still remain even in neutron irradiation, and shows Hall-Petch relation. Neutron fluence and irradited temperature of measured samples were total fast neutron fluence (E>1MeV) of about 1.3×10^{21} n/cm^2 and about 327°C in helium.

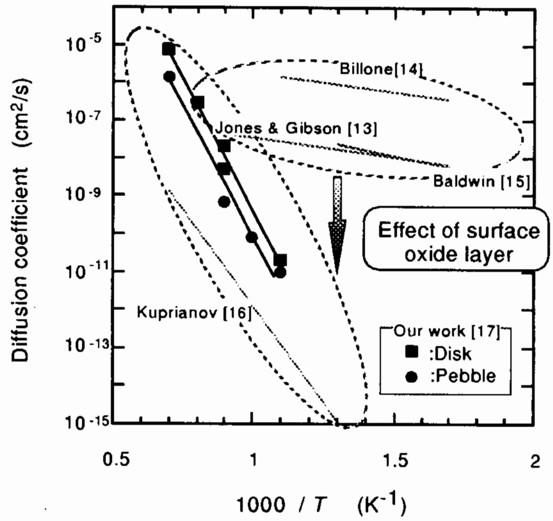

Fig. 8 Diffusion coefficient of hydrogen isotopes in beryllium.

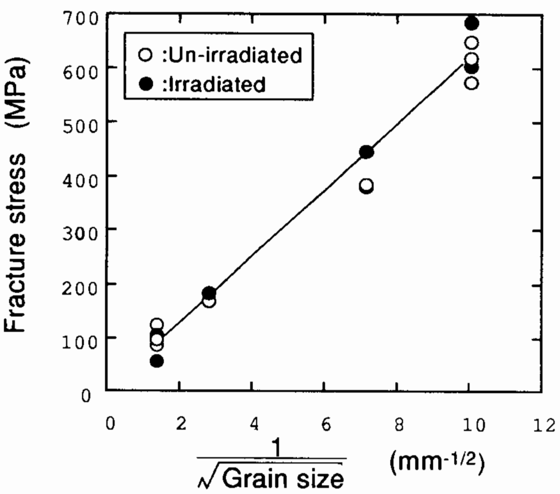

Fig. 9 Fracture stress of neutron-irradiated beryllium as a function of grain size.

Thermal properties of neutron-irradiated beryllium are the first data of the world, and were measured with swelling sample by using laser flush apparatus in Beryllium PIE facility [20]. Swelling samples of beryllium were formed by heating from neutron irradiated beryllium. Thermal conductivity were obtained by measured thermal diffusivity and specific heat. Results of thermal conductivity is shown in Fig. 10. In this figure, 78 and 61 %TD show the data of swelling sample. Thermal conductivity slightly decreased by neutron irradiation, and noticeable decreasing was induced by swelling.

Neutron fluence and irradiated temperature of measured samples were total fast neutron fluence (E>1MeV) of about 4.5×10^{20} n/cm^2 and about 200°C in helium.

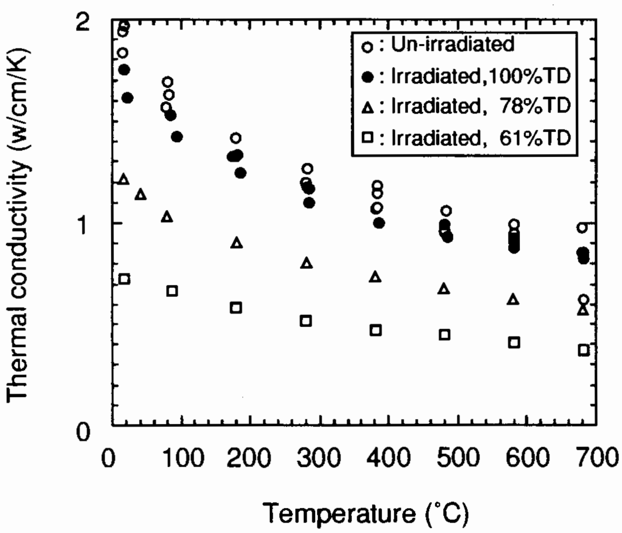

Fig. 10 Measurement results of thermal conductivity .

For the reprocessing technology of neutron irradiated beryllium, it is necessary to develop for the utilize of effective resource and removing of radioactive isotopes. A heat treatment method under the chlorine gas flow diluted with Ar gas was developed for reprocessing beryllium [21]. Neutron irradiated beryllium recovered as beryllium chloride, and impurities like ^{60}Co and tritium were separated. Separation efficiency of ^{60}Co and tritium in preliminary tests were 96 and 95%, respectively.

4. Summary

Status of the irradiation technology development for an in-pile functional test "Phase 1" on fusion blanket and the development of blanket materials are briefly described in this report. Bright prospects of an irradiation test "Phase 1" were obtained by development of mock-ups and establishment of facilities. Irradiation test "Phase 1" will start on March in 1997, and original engineering data for fusion blanket will be obtained by this test.

For the blanket materials, development on suitable production method for mass productions like ton scale will be continued for DEMO blanket. Some irradiation data of beryllium were obtained with Beryllium PIE facility. These indispensable efforts will contribute for the design of fusion blanket.

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