Tritium Recovery Experiment from Li Ceramic Breeding Material Irradiated with DT Neutrons

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Abstract. Based on the standpoint of tritium breeding ratio in fusion reactors, the tritium recovery performance of breeder blankets should be closely examined. We have performed the first tritium recover experiment with DT neutron source in the world at the Fusion Neutronics Source facility in Japan Atomic Energy Agency. An experimental assembly simulating a solid breeder blanket was irradiated with DT neutrons and then the tritium production and recovery data were measured by means of the liquid scintillation method, separately. This experiment shows the tritium recovery ratio of the assembly is 1.05 ± 0.08 at 873 K, which indicates that the design of Japanese solid breeder blanket promises a good prospect of tritium recovery.

1. Introduction

A nuclear fusion reactor has to breed the fuel tritium for itself in its blanket. The loss of produced tritium in the blanket and the recovery system must be minimized because it is difficult to breed sufficient tritium in current designs of fusion blankets [1, 2]. In Japan, the water-cooling solid breeder blanket is a prime candidate, and the helium sweep gas with hydrogen will be used for the tritium recovery [3]. However, no DT neutron irradiation experiment for investing the tritium recovery properties for the solid blanket has been conducted and the tritium recovery ratio is one of urgent technical issues for the development of the fusion breeding blanket system. Therefore we have conducted an experiment with DT neutron irradiation in order to obtain the tritium recovery rate.

2. Experiment

2.1. DT neutron irradiation

Figure 1 shows the experimental arrangement. The DT neutron irradiation has been performed at the Fusion Neutronics Source (FNS) in Japan Atomic Energy Agency (JAEA). A cylindrical bulk (ϕ 630 x 457.2 mm^3) with beryllium blocks, which is a neutron multiplier , was used in order to simulate the neutron spectrum in the solid breeder blanket design. A stainless steel rectangular container which contained 67 g of natural abundance lithium titanate (Li_2TiO_3) pebbles was inserted at about 203 mm depth point from the beryllium bulk front surface.

Figures 2 and 3 show the overview of the container and the set up of breeding material pebbles, heater and insulator in the container, respectively. The container with a wire heater can increase temperature of the Li_2TiO_3 pebbles up to 1273 K. In order to remove tritium produced by neutron irradiation, it can also flow the sweep gas from the inlet to the outlet thought the breeding material. The total number of DT neutrons generated at the source was the order of 10^{15}, which was deduced with a measurement of associated 3.5-MeV alpha particle, and its error was about 2%. As the first experiment, we have carried out two irradiations at room temperature in order to examine the tritium recovery ratio. One was for the measurement of tritium production data and the other was for the measurement of tritium recovery data.

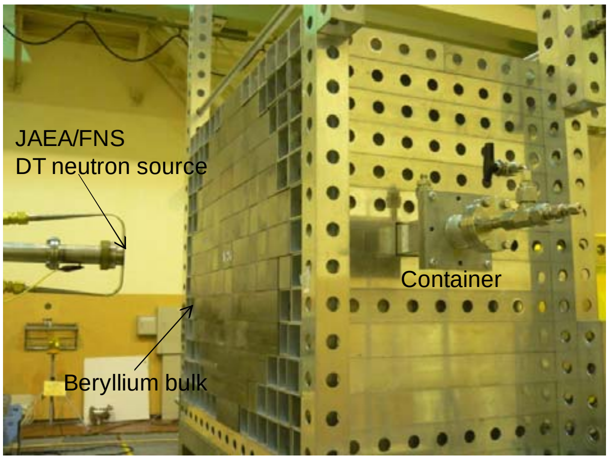

FIG. 1 DT neutron irradiation arrangement.

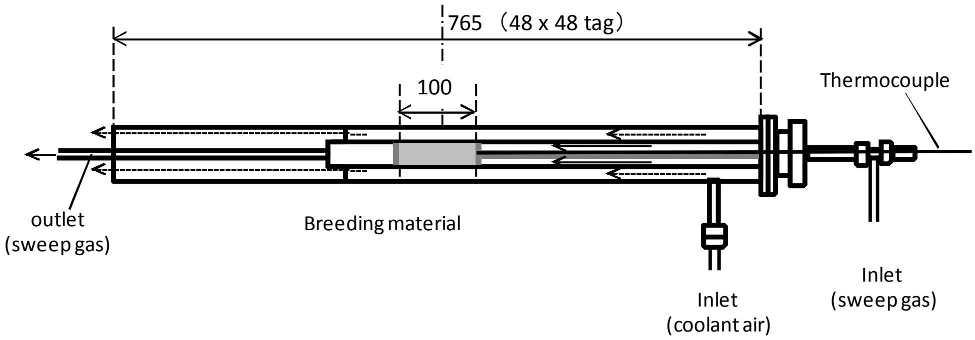

FIG. 2 Overview of container for irradiation.

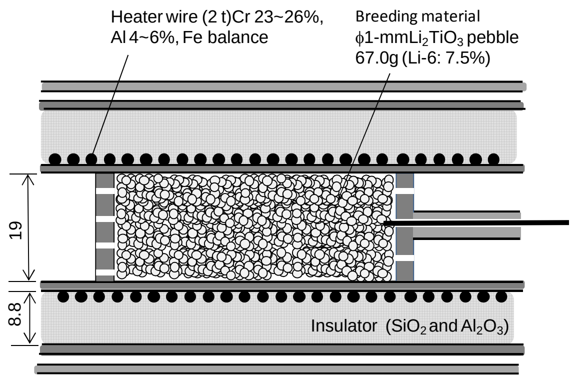

FIG. 3 Set up of breeding material pebble, heater and insulator in container.

2.2. Tritium production measurement

The procedure of tritium production measurement is shown in Fig. 4. After the neutron irradiation, a part of Li_2TiO_3 pebbles ejected from the container was kept in liquid HCl (0.2 N) to dissolve. The supernatant tritiated water (HTO) turned with the chemical reaction

$Li_2TiO_3(T) + HCl \rightarrow 2LiCl + H_2O(T) + TiO_2$, was mixed with a liquid scintillator. The tritium activity was measured with a liquid scintillation counter (LSC).

Figure 5 shows the experimental apparatus of the tritium recovery operation and measurement. Each part is connected with a stainless pipe of 7mm in inner diameter. The length from the stainless steel assembly with the sample to the first bubbler is about 1.5 m. The irradiated pebble was heated to 873 K with the heater in the container, where helium gas with 1% H_2 gas was flowed through the irradiated Li_2TiO_3 pebbles. The 100 standard cc/min gas was regulated with mass flow controllers. Tritium in the flow gas was oxidized to HTO with the CuO bed and the HTO was collected in the bubblers. During the operation, the temperatures of the stainless pipe and the CuO bed were kept at 423 and 773 K, respectively. Tritium radioactivity in the bubblers was also measured with the same LSC.

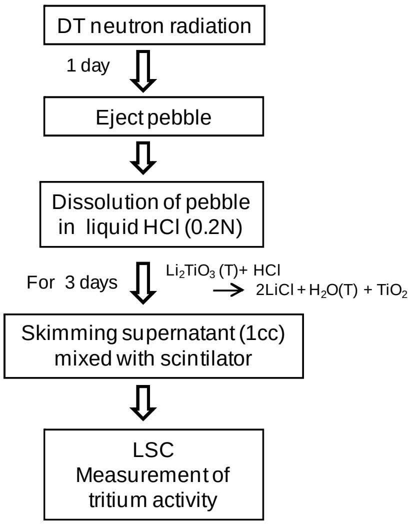

FIG. 4 Procedure of tritium production measurement.

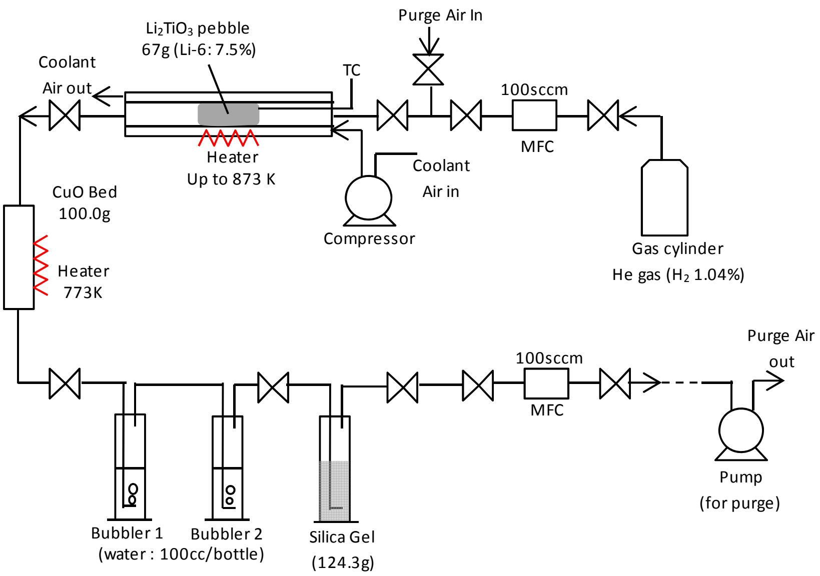

FIG. 5 Procedure of tritium recovery operation and measurement.

3. Results and discussion

TABLE I shows key points and results of the tritium measurement, tritium production and recovery data with the experimental error. The error of the tritium production data is higher than that of the recovery data because of complex treatment and counting statistical error. Finally the tritium recovery ratio of the assembly is 1.05 ± 0.08 at 873 K and the design of Japanese solid breeder blanket will promise a good prospect of tritium recovery up 873 K.

TABLE I: KEY POINTS AND RESULT OF TRITIUM PRODUCTION AND RECOVERY.

	Production	**Recovery**
Tritium measurement method	LSC	LSC
Sample material	$^{nat}Li_2TiO_3$ pebble	$^{nat}Li_2TiO_3$ pebble
Weigh g	10.23	67.00
Tritium sampling	Dissolution	Heating at 873 K
	HCl liquid 125g	Bubbler water 100g
	2.4 g	1 g
Scintirator	Ultima Gold^{TM} AB	Clear sol-II
Activity Bq/sample	32.0	84.6
DT neutron yield	1.75×10^{15}	1.61×10^{15}
Activity Bq/g/DT neutron	$P = 7.46 \times 10^{-16}$	$R = 7.84 \times 10^{-16}$
Total tritium measuring error %	7.2	2.3
DT neutron yield	2	2
Treatment procedure	4	<1
Measurement efficiency	1	<1
Tritium counting statistics	5.5	0.1
Tritium recovery ratio (R/ P)	**1.05 (± 0.08)**	

4. Conclusion

In order to clarify the tritium recovery property of solid breeder blanket designed by Japanese blanket team, we have performed the first tritium recover experiment with DT neutrons in the world at FNS in JAEA. From the measured tritium recovery ratio, a good prospective is obtained for the solid breeder blanket at 873 K. In order to progress the investigation of tritium recovery property for the ITER test blanket module and DEMO blanket designs, we are going to examine the dependency of the temperature and sweep gas with DT neutron source as the next step.

5. References

[1] TOBITA, K., et al., “Compact DEMO, SlimCS: design progress and issues”, Nuclear Fusion **49** (2009) 075029.

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[3] KAWAMURA, Y., et al., “Research and development of the tritium recovery system for the blanket of the fusion reactor in JAEA”, Nuclear Fusion **49** (2009) 055019.