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Recent results of R&D activities on tritium technologies for ITER and fusion reactors at TPL of JAEA

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ABSTRACT

At Tritium Process Laboratory (TPL) of Japan Atomic Energy Agency (JAEA), tritium technologies for a fusion reactor have been carried out up to date. The design studies of Air Detrtiation and Vent Detritiation System (ADS/VDS) of ITER have been carried out in JAEA as a contribution of Japan to ITER. For the tritium processing technologies, our efforts have been focused on the R&D of the tritium recovery system of ITER test blanket system. A ceramic proton conductor has been studied as an advanced blanket system. A series of fundamental studies on tritium safety technologies for ITER and for fusion DEMO plants has also been carried out. The main R&D activities in this field are the tritium behavior in a confinement and its barrier materials, monitoring, accountancy, detritiation and decontamination, etc. Especially, for the fundamental studies for a DEMO plants, a part of the studies will be carries out a new facility at Rokkasho in Aomori in the Broader Approach (BA) program in coming 10 years. For this purpose, a design study of the facility at Rokkasho has first been started.

1. Introduction

The facility of Tritium Process Laboratory (TPL) in Japan Atomic Energy Agency (JAEA) is an unique laboratory for fusion technology where can handle more than 1 g of tritium in Japan [1]. Maximum storage amount of tritium is licensed 22.2 PBq. Now, 13.7 PBq of tritium (at March 2007) is stored in the radiological controlled area of TPL and used safely in a multiple confinement system for the following R&D. The recent R&D activities in TPL are mainly (1) ITER ADS/VDS design, (2) investigation of tritium recovery system from breeding blanket, and (3) fundamental studies on tritium safety enhancement.

In this paper, the above recent activities on tritium technologies at TPL in JAEA are summarized for ITER and future fusion demo reactor. Recently, it has been decided that a part of fundamental studies for a DEMO plant would be carried out at a new facility at Rokkasho in Aomori in the Broader Approach (BA) program. The present status of the BA program for tritium technologies is also reported in this paper.

2. Tritium processing technology development for breeding blanket

In the research and development of Blanket Tritium Recovery system, the design of the system for ITER–test blanket module (TBM) was reviewed from the viewpoint of the preparation of a safety report. An outline design of the Blanket Tritium Recovery system for the ITER–TBM system was carried out [2] as an advanced system using an electrochemical hydrogen pump with ceramic proton conductor. In addition, our attention was focused on the effect of electrode property on the hydrogen transportation. The sputtered electrode was first prepared and their conductivities were studied by the impedance measurement. However, it was not confirmed that the conductivity was caused by transportation of proton. The hydrogen transportation experiments were then carried out using the proton conductor ceramic membrane with the sputtered electrode. Fig. 1 shows a comparison of the current–voltage property curves among various electrodes. The concentration of H_2 in a feed stream was 0.1%. The transportation number was almost 1.0. Therefore, it was confirmed that the observed current was caused by the transportation of proton. When the Pd sputtered electrode was used, the amount of hydrogen transportation was about four times as large as that for the case of usual Pt pasted electrode. It is expected that the Pd sputtered electrode is fairly dense membrane. Hydrogen can pass through membrane, but oxygen and water cannot pass through the membrane. The proton concentration in the

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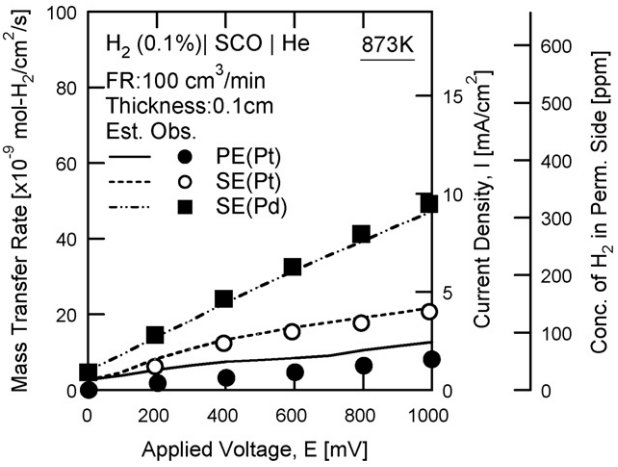

Fig. 1. Comparison of the current–voltage property curves among various electrodes (PE: pasted electrode, SE: sputtered electrode).

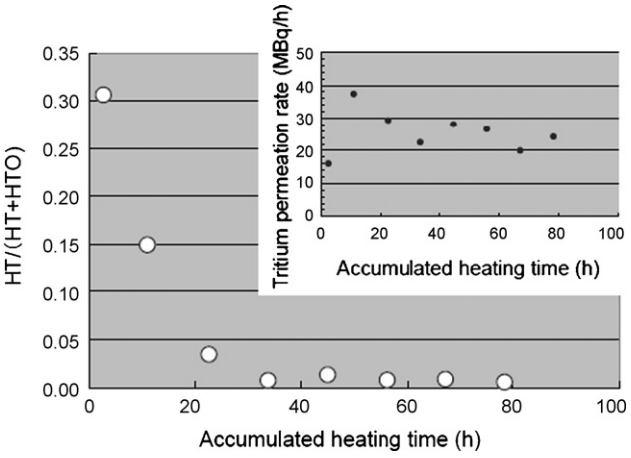

Fig. 2. Total permeated tritium amount and its hydrogen gas fraction.

ceramic is kept at a high value, so that the proton conductivity becomes large. The hydrogen transportation capability is increased by using the sputtered electrode.

An amount of high concentration tritium water is expected to be produced in the bleeding blanket and in the vacuum vessel by the oxygen baking. The electrolytic reduction method using a solid ceramic electrolyte was studied to process the above tritium water. It was found that the efficiency of the water reduction reaction was drastically increased by adding ceria (CeO_2) to an electrode. From the surface observation and the impedance analysis of the electrode, it was found that ceria adding is effective not only for the formation of the porous and fine surface structure of electrode, but also for the decrease the electrode resistance of reduction [3].

The fixed-bed adsorption process has also been studied as one of the promising processes for the recovery of tritiated water in the field of nuclear fusion. A series of experiments was carried out in the H_2O–HTO binary system by using NaX and NaY zeolite [4]. The ratio of SiO_2/Al_2O_3 was changed from 2.0 to 10.0 in the experiments. A large difference in isotherm was observed by changing the SiO_2/Al_2O_3 ratio from 7.0 to 10.0. The desorption energy range at 50–60 kJ/mol predominates for NaA2.0, NaX2.0, NaX2.5 and NaY3.0. On the other hand, the range at 42–50 kJ/mol predominates for NaY5.0, NaY7.0 and NaY10.0. The removable water capacity of NaY10.0 at around room temperature is approximately four times larger than that of NaA2.0 even though both zeolites have nearly the same absorbable capacity of H_2O nearly at $P/P_0 = 1.0$. The weaker interaction generates the isotope effect on desorption. HTO dehydration ratio has strong interrelation with accumulated amount of purge gas. It is independent upon flow rate of purge gas. The size of water absorber such as dryer or dehumidifier can be downscaled by the adoption of NaY10.0 as adsorbent. The consideration of heating and cooling period necessary for temperature swing desorption process is not required in the column design.

3. Basic study on tritium behavior

In order to establish a proper control method of the DT fuel isotope balance in ITER, isotopic composition of hydrogen, which was delivered rapidly from ZrCo bed (1/10 scale of ITER fuel storage and delivery system) by vacuum pump, was investigated as functions of initially stored H/D ratio (H:D = 9:1, 1:1, 1:9) and pre-heated temperature (573–623 K). The equilibrium pressure (P) of hydrogen–metal system has large isotope effect such as $P_{H2} < P_{D2} < P_{T2}$ for ZrCo, however, the maximum difference of H,D isotope fractions was within about 5%, during rapid delivery of about 90% hydrogen gases at 623 K and initial H:D of 1:1. In cases of initial H:D of 9:1 or 1:9, the differences of H,D isotope fractions were rather small of a few percent. The isotope fraction was continuously changed. It was not so large difference in comparison with that of equilibrium pressures. However, even if the fluctuation of the isotope ratio is less than 5%, depending on the requirements from plasma physics experiments and fuel accountancy of tritium plant, batch fuel delivery from adequate gas tanks after isotope composition adjustment will be preferable to direct rapid delivery from storage bed [5].

In order to accumulate data on tritium transferred to cooling water of a fusion reactor, a series of experiments of tritium permeation into water jacket pressurized to 0.8 MPa by He gas was performed through pure iron piping, which contained about 1 kPa of pure tritium gas at 423 K. Chemical forms of tritium permeated into water were monitored periodically under continuous purging water jacket by He. Observation of metal surface was also carried out periodically by SEM and XRD analysis. The main results are as follows:

(1) The permeation amount through pure iron to water jacket was about 1/5 of the calculated value in this experimental condition as seen in Fig. 2. Even if surface oxide layer (fine and porous magnetite) grew as shown in Fig. 3, tritium permeation rate to water was not changed drastically.
(2) HT fraction of permeated tritium was 30% initially, then, it decreased drastically to about 1/50 (see Fig. 2). It suggested the relation between the HT fraction decrease and oxide layer growth.
(3) Permeated species and amounts were not affected clearly by dissolved hydrogen amount difference between pure He and 1% H_2 in He balance purging under 0.8 MPa at 423 K.

From the above results, it suggests that the tritium in pure iron would transfer to water by isotope exchange reaction through surface hydroxyl species of magnetite. On the other hand, if surface hydroxyl density is small, the tritium would permeate to water as hydrogen gas after surface recombination [6].

In order to develop for the optimal decontamination technique, the decontamination experiment was carried out as a function of water vapor concentration in the purge gas (N_2) for epoxy paint, acrylic resin and butyl rubber. As the result, the desorption rates