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R&D activities of tritium technologies on Broader Approach in Phase 2-2

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HIGHLIGHTS

- R&D activities of tritium technology on BA were introduced.
- Representative results in each task were explained.
- Future plan of this activity was introduced.

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ABSTRACT

Activities on Broader Approach (BA) were started in 2007. In Phase 2-2, many R&Ds, development of tritium accountancy technology, development of basic tritium safety research and tritium durability test, were implemented successfully by JAEA and Japanese Universities. In Phase 2-3, new collaborative study for tritium measurement and new R&D activities for JET ILW are started. R&D activities on BA have continued in Phase 2-3 (2014–2016).

1. Introduction

Activities on Broader Approach (BA) were started in 2007 on the basis of the Agreement between the Government of Japan and the EURATOM. The period of BA activities consist of Phase 1 and Phase 2 dividing into Phase 2-1 (2010–2011), Phase 2-2 (2012–2013) and Phase 2-3 (2014–2016). Tritium technology was chosen as one of important R&D issues to develop DEMO plant. R&D activities of tritium technology on BA consist of four tasks. Task-1 is to prepare and maintain the multipurpose facility (DEMO R&D Building) in Rokkasho BA site in Japan. The licensing of DEMO R&D Building was finished in 2011 and 7.4 TBq of tritium can be stored. Details of the facility were described in elsewhere [1]. Tasks 2, 3 and 4 are main R&D activities for tritium and these focused on the following:

Task-2: Development of tritium accountancy technology
Task-3: Development of basic tritium safety research
Task-4: Tritium durability test

Task-2 is aiming to assess the feasibility of advanced tritium analysis methods and to confirm its reliability, use range and sensitivity. Task-3 has two objects. One is to construct database regarding hydrogen isotopes behavior in materials considering to be used in DEMO and ITER. The other is to assess the solution of the issues in DEMO, such as decontamination, permeation of tritium and estimation of inventory. Task-4 is aiming to confirm the soundness of materials, stainless steel and polymer, in the period of service. Results of the above in past Phase were reported [1–3].

These tasks have been implemented not only at DEMO R&D Building but also at facilities of universities. In the R&D activities of Phase 1, the collaborative research programs between Japan Atomic Energy Agency (JAEA) and Japanese universities were started and R&D have been carried out also in Universities.

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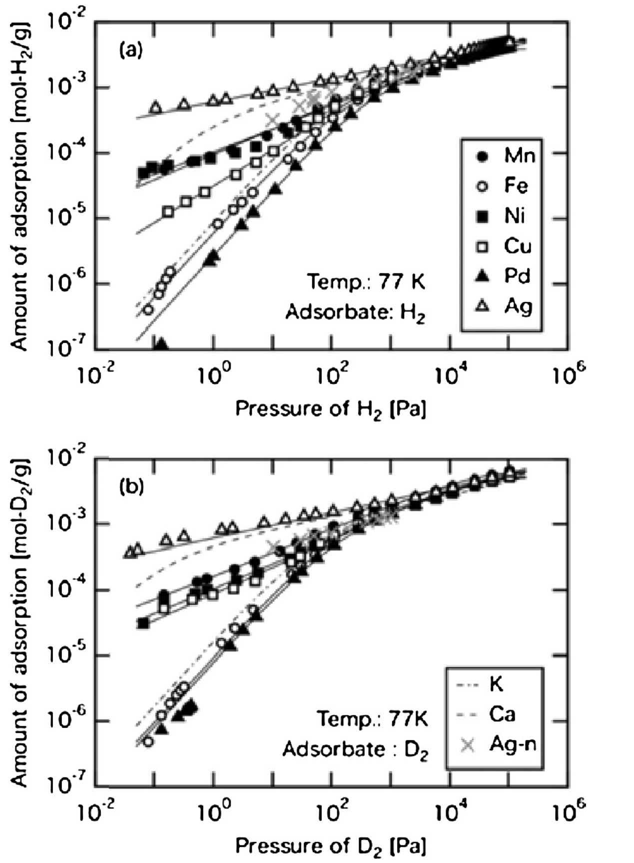

Fig. 1. Adsorption capacity of (a) H_2 and (b) D_2 on various cation-exchanged MOR [6].

Phase 2-2 already finished in 2013 and new Phase 2-3 is starting in 2014. This paper describes the representative results of each task during Phase 2-2 and shows the experimental plan in Phase 2-3.

2. Results of each task and future plan

2.1. Task-2: Development of tritium accountancy technology

In contrast to ITER, a series of continuous operation is required for the main fuel and the blanket tritium loop for a DEMO reactor. R&Ds for monitoring and analysis of tritium in this loop are required from viewpoint of the control of the loops. For this purpose, some basic experiments have been carried out to develop advanced tritium analysis methods.

Basic studies have been started to develop a real time monitor for isotope compositions in hydrogen gas. The micro GC (gas chromatography) is one of candidate method for DEMO analysis system. New materials for column which could separate hydrogen in shorter time have been studies. Zeolite easily exchanges its cation to another. And the pore size changes easily. Cation exchanged mordenite type zeolite (MOR) has been reported to have comparatively good capability of isotope separation at more than 77 K [4,5]. However, the correlation between adsorption capacity, adsorption rate, cation kind and cation exchange ratio have not been thoroughly investigated so far. Several MORs with their cation exchanged with transition metal ion were prepared, and adsorption capacities of hydrogen isotopes on them at 77 K were investigated experimentally [6]. Fig. 1 shows adsorption capacity of (a) H_2 and (b) D_2 on various cation-exchanged MOR. It was found that Ag-MOR has indicated fairly large adsorption capacity of hydrogen isotopes at 77 K in comparison with Ca-MOR. However, isotope effect on adsorption capacity became small.

In order to develop a technique for non-destructive evaluation methods, studies on BIXS (beta-ray induced X-ray spectroscopy) and Imaging plate (IP) have been carried out by the collaborative study with University of Toyama and Tohoku University. In this study of Phase 2-2, IP method was applied to the evaluation of tritium depth profile by using of different thickness of Cu films as X-ray absorbers. In addition to this, IP method has been applied to other measurements of tritium in materials. In collaborative studies with Kyushu University, IP was applied to determination of hydrogen diffusion coefficients at lower temperatures and to measurement of hydrogen solubility in ceramics [7,8]. The data obtained by IP were in good agreement with the extrapolation from the data taken at higher temperatures from the literature.

2.2. Task-3: Development of basic tritium safety research

The components in a DEMO reactor are continuously exposed to tritium for a long period in contrast to ITER. It is supposed that an appreciable amount of tritium permeates to a cooling water system. Processing of this tritium water should have a large impact on the DEMO reactor. Some new materials will also be used in the DEMO reactor. It is required to obtain basic data on the interaction between tritium and the new materials. In the blanket system of the DEMO reactor, it is required to obtain the data on tritium release form breeding materials and on the tritium/material interaction in some main components of the system. It is also quite significant to obtain the data on tritium retention within the wall of the vacuum chamber to evaluate its large tritium inventory. The above basic data should also contribute to enhance safety designs and public acceptance for the DEMO reactor.

In Task-3 regarding basic data on the interaction between tritium, the materials to be used are W, F82H and stainless steels. The influence of impurity and a trap site in the materials and the influence of plasma exposure with materials are major concerns in this subject. The blistering phenomena on the tungsten surface have been studied. A liner plasma generator apparatus has been set up at the BA site of JAEA to collect basic data on plasma surface interaction. To understand plasma surface interactions, tungsten (W) samples were exposed to low-energy (38 eV/D), high flux (10^{22} $D^+/m^2/s$) deuterium (D) plasma at 495, 545 and 550 K to a fluence of 10^{26} D/m^2. After the plasma exposure, tritium was introduced into the samples by exposure to deuterium–tritium gas mixture at 473 K. Tritium distribution on the W surface was examined by the techniques of autoradiography. Fig. 2 shows representative autoradiographic patterns and indicates that tritium was concentrated on the grain boundary and blisters [9]. In the collaborative study with Osaka University, D retention in TFGR W (W-TiC and W-TaC) was studied. Deuterium permeation experiment and thermal desorption spectroscopy (TDS) for D_2^+ or C^+ irradiated W were performed to estimate deuterium permeation and retention behavior at Shizuoka University [10]. In collaborative study with Hokkaido University, the F82H steel and the tungsten were irradiated with inert gas ions to introduce the surface damage, with the introduction of the surface impurity layer. After the irradiation, the deuterium retention and desorption behaviors were evaluated. With regard to tritium release form breeding materials, an experimental apparatus for a forced convection flow of $Li_{15.7}Pb_{84.3}$ eutectic alloy was set up in the energy chemical engineering laboratory of Kyushu University, and hydrogen mass transfer in the Li–Pb flow was investigated [11]. The development of decontamination for tritium from fusion reactor materials using a hydrothermal treatment has been carried out with collaborative study with University of Toyama. A preliminary study of tritium permeation mechanism in erbium oxide coatings has been