Study on Maintenance, Recycle, and Radioactive Waste Management of Fusion Reactor
Part III. Accumulation and removal of tritium
核融合炉の保守・リサイクル・バックエンド対策に関する検討
3. トリチウムの蓄積と除去

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Retention and release of hydrogen isotopes by tungsten depsotion layer and neutron-irradiated tungsten are dicussed in this presentation to obtain better understanding of tritium accumulation and removal in/from first walls and divertor components of DEMO reactors. The results of laboratory-scale experimens showed that tritium retention up to relatively high cocnetration is possible under expected reactor conditions. Evaluation of decay heat induced by neutron irradiation and temerature change due to it are important to predict tritium release behavior from first wall and divertor components during maintenance and disposal processes.

1. Background

Tungsten is currently recognized as a primary candidate of armor materials for first-walls and divertor components of DEMO reactors due to favorable physical properties such as high melting point and high thermal conductivity. It is, however, known that tungsten deposition layer can accommodate hydrogen isotopes up to relatively high concentration (e.g. H/W = 0.7 [1]) depending on substrate temperature and contents of impurities such as oxygen and He [1-3]. In addition, retention of hydrogen isotopes in the bulk of tungsten increases with neutron irradiation due to trapping effects by defects. The results obtained in Japan-US Joint Project TITAN showed that deuterium retention in tungsten at 773 K increased from 50 appm to 2000 appm after neutron irradiation to 0.025 dpa in a fission reactor [4-6]. Simulation studies by ion irradiation indicated that concentration of hydrogen isotopes in damaged tungsten can reach 1-2 at.% [7,8]. Therefore, in spite of low tritium solubility in well-annealed tungsten, tritium retention in armor materials could be highest among fusion reactor materials except getters for tritium storage and supply (U or ZrCo).

Temperature of armor materials is expected to be relatively high even after termination of plasma due to decay heat induced by extensive neutron irradiation [9]. Hence, tritium release from hot armor materials could be a critical safety issue during the maintenance and disposal processes. From these viewpoints, retention and release of tritium by tungsten deposition layer and neutron-irradiated tungsten are discussed in this presentation.

2. Tritium Retention

Fujiki et al. [1] reported that hydrogen retention in tungsten deposition layer formed by hydrogen RF plasma reached H/W = 0.7 at ambient temperature and oxygen content of O/W = 1.69. Reduction in deuterium retention with increasing substrate temperature was observed by Temmerman and Doerner [2] and Alimov et al. [3]. The extent of reduction, however, was strongly dependent on vacuum conditions and presence or absence of He. The deuterium retention at 623 K was D/W = 2×10^{-4} when the deposition layer was prepared in poor vacuum conditions with pure D plasma, but it was 4×10^{-2} under high vacuum conditions and addition of He. Detailed understanding of retention mechanisms is necessary for the evaluation of tritium retention under reactor conditions. It was found in thermal desorption measurements that deuterium desorption continued up to 900 K.

In TITAN Project Task 2-1, disk-type specimens of pure tungsten were prepared in Japan and irradiated in High Flux Isotope Reactor (HFIR), Oak Ridge National Laboratory (ORNL) at 50 ^{o}C to 0.025 dpa. Then the specimens were shipped to Idaho National Laboratory (INL) to expose high flux deuterium/tritium plasma with the linear plasma machine called Tritium Plasma Experiment

(TPE). Figure 1 shows thermal desorption spectrum of deuterium thus obtained for neutron-irradiated tungsten after exposure to TPE plasma at 473 K to fluence of 2×10^{25} D m^{-2} together with spectrum from unirradiated specimen [4-6]. The desorption of deuterium from neutron-irradiated tungsten continued up to 1000 K, while that from unirradiated specimen was completed at 700 K. In addition, the total deuterium retention in the former was significantly larger than the latter. Measurements of depth profiles of D after exposure to plasma at 773 K showed that deuterium retention was 2000 appm in neutron-irradiated tungsten and 50 ppm in unirradiated specimen as described above [6]. These observations indicated the formation of strong trapping sites by irradiation. The trapping energy was evaluated to be 2 eV from the desorption spectrum, and this value corresponds to the trapping by voids [7,8].

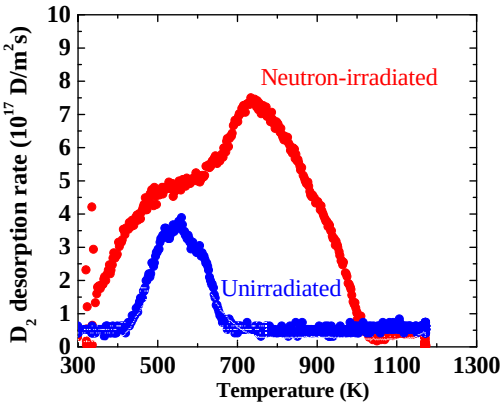

Fig.1. Thermal desorption spectra of deuterium from unirradiated and neutron-irradiated (0.025 dpa) tungsten after exposure to high flux plasma at 473 K.

3. Issues Needs To Be Clarified

The above-mentioned observations indicated that tritium retention in armor materials could be relatively high under reactor conditions. During replacement and decomposition of blanket and divertor components, circulation of coolant should be stopped. Then temperature of armor materials should increase by decay heat [9], and thermally-activated tritium release would start. For the prediction of release behavior of tritium, precise evaluation of temperature change during maintenance and disposal processes is necessary in addition to the detailed understanding of retention mechanisms in deposition layer and radiation defects.

On the other hand, controlled temperature rise by using decay heat may allow passive and/or active tritium removal treatments such as pumping and glow-discharge cleaning at high wall temperatures before opening of vacuum chamber. Such treatments at high temperatures would help greatly to remove tritium not only from surfaces but also bulk and significantly reduce uncontrolled tritium release during maintenance and disposal processes. Precise evaluation of decay heat and temperature change is again required to design such decontamination scenarios.

Acknowledgments

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References

[1] T. Fujiki, K. Katayama, S. Kasahara, S. Fukada and M. Nishikawa, Fusion Eng. Design, **85** (2010) 1094.

[2] G. De Tmmerman and R. P. Doerner, J. Nucl. Mater., **389** (2009) 479.

[3] V. Kh. Alimov, J. Roth, W. M. Shu, D. A. Komarov, K. Isobe and T. Yamanishi, J. Nucl. Mater., **399** (2010) 225.

[4] M. Shimada, Y. Hatano, P. Calderoni, T. Oda, Y. Oya, M. Sokolov, K. Zhang, G. Cao, R. Kolasinski, J. P. Sharpe, J. Nucl. Mater., in press. (doi:10.1016/j.jnucmat.2010.11.050)

[5] Y. Oya, M. Shimada, M. Kobayashi, T. Oda, M. Hara, H. Watanabe, Y. Hatano, P. Calderoni and K. Okuno, Phys. Scr., *Comparison of deuterium retention for ion-irradiated and neutron-irradiated tungsten*, in press.

[6] M. Shimada, G. Cao, Y. Hatano, T. Oda, Y. Oya, M. Hara and P. Calderoni, Phys. Scr., *Deuterium depth profile in neutron-irradiated tungsten exposed to plasma*, in press.

[7] M. Poon, A. A. Haasz, J. W. Davis, J. Nucl. Mater., **374** (2008) 390.

[8] O. V. Ogorodnikova, B. Tyburska, V. Kh. Alimov, K. Ertl, J. Nucl. Mater., **415** (2011) s661.

[9] F. Andritsos and M. Zucchetti, J. Nucl. Mater., **212-215** (1994) 662.