§ 12. Investigation of Tritium Behavior at D-D Burning Experiment in LHD and Study on Recovery, Treatment and Disposal Methods of Tritium

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Deuterium-deuterium burning experiments at the Large Helical Device (LHD) are being planed by the National Institute for Fusion Science (NIFS). Tritium will be produced by the D-D fusion reaction during the experiments, though the amount is estimated to be small. A part of tritium produced in the vacuum vessel of LHD will be retained in the plasma facing components and structural materials, and the rest will be exhausted to the tritium recovery system via the vacuum systems. It is also anticipated that a part of tritium will be transferred to the cooling water after permeation through the wall of cooling devices. Therefore, this research group is formed to understand tritium behavior inside of the LHD vessel and to establish the feasible methods of recovery, treatment and disposal of tritium produced in the LHD facility from the view point of technology and the social acceptance.

The main purposes of the present investigation are

1. to develop the quantitative and qualitative analysis method for estimation of the amount of tritium in various places of LHD,
2. to develop the tritium measuring method at various places of LHD,
3. to develop the tritium recovery method from various places in LHD,
4. to develop the method to recover tritium transferred to the vacuum system and cooling system of LHD,

and

5. to develop the safety treatment and disposal methods of recovered tritium feasible from the view point of technology and social acceptance.

In this year the tritium concentration on graphite materials were measured applying various methods as the imaging plate technology. It was confirmed that tritium included in the re-deposited layer formed on the first wall material was measurable. It was also concluded in the course of this research that concrete, the third barrier against tritium leak from a fusion reactor, can trap a large amount of tritium because it has fair amount of adsorption capacity and large amount of isotope exchange capacity.

Some results were reported in

1.Isotope Effect in Hydrogen Isotope Exchange Reaction on First wall Materials, K. Katayama, M. Nishikawa and J. Yamaguchi, J. Nucl. Sci. and Tech., vol. 39, 371-376(2002)
2.Study on behavior of tritium In cement materials, S. Satake, T. Motoshima, M. Nishikawa and T. Takeishi, 15th Japan-Korea Symposium on Chemical Engineering, Kumamoto, December7-8, 2002
3.Tritium Removal from JT-60U graphite tile, K. Katayama, T. Takeishi and M. Nishikawa, ibid.

In the group of this research, following subjects were also discussed;

1) tritium handling experiment and achievements at the University of Tokyo,
2) design study of processing systems at NIFS for tritiated exhaust gas and liquid during D-D experiments in LHD and their future plan,
3) results of experiments about tritium-Flibe research done at the Idaho National Environmental and Engineering Laboratory in the frame work of the Jupiter 2 US-Japan co-operation program,
4) recent research in Hokkaido University about PWI problems,
5) experiment about tritium permeation through the concrete wall performed at the Kyushu university, and
6) present situation of the ITER program.

It was recognized from these discussions that the database and experiences related to tritium traceability and management in the nuclear and accelerator facilities is very useful for those in the LHD facility during D-D experiments and fusion reactors.

It was concluded through the discussion that the research and development planning of tritium gaseous and liquid wastes processing systems proposed by NIFS was appropriate to perform early and safely D-D experiments at LHD with help from various tritium handling facilities in Japan.