§24. Study on Tritium Behavior in Liquid Blanket System of Laser Inertial Fusion Reactor

Nishikawa, M. (Graduate School of Eng. Sci., Kyushu U.),
Fukada, S. (Graduate School of Eng., Kyushu U.),
Katayama, K. (Graduate School of Eng. Sci., Kyushu U.)

In designing of the liquid blanket system for a laser inertial fusion reactor, it is necessary to have the effective tritium recovery system for assurance of tritium self-supply and to have the tritium safety confinement system for certification of tritium radiation safety. Then, 1t is required to know the tritium transfer properties in liquid breeder materials because the wetted wall system is considered to protect the first wall in design of the inertial fusion reactor of Osaka University. At present, use of Lithium Lead is considered as the breeder material though only a small amount of reports have been made on tritium transfer properties. It is anticipated that a large part of the bred tritium may permeate to the outer circumstances because the solubility of hydrogen in lithium lead is considered to be so small. Accordingly, the object of the present researchers is to make the measurement of tritium diffusivity and solubility in lithium lead and lithium as a part of the cooperative research program lead by the Institute of Laser Engineering, Osaka University.

The following advancements are made in this year.

(1) Neutronics calculation estimated using the ANISN code shows that thickness of 1m is enough to obtain the effective tritium breeding ratio when LiPb is chosen as the blanket material as shown in Fig. 1.

(2) The solubility of hydrogen into LiPb obtained in this study is compared in Fig. 2 with solubility from other researchers. It is necessary to have more data because no good agreement of data of each study is obtained as can be seen from Fig. 2.

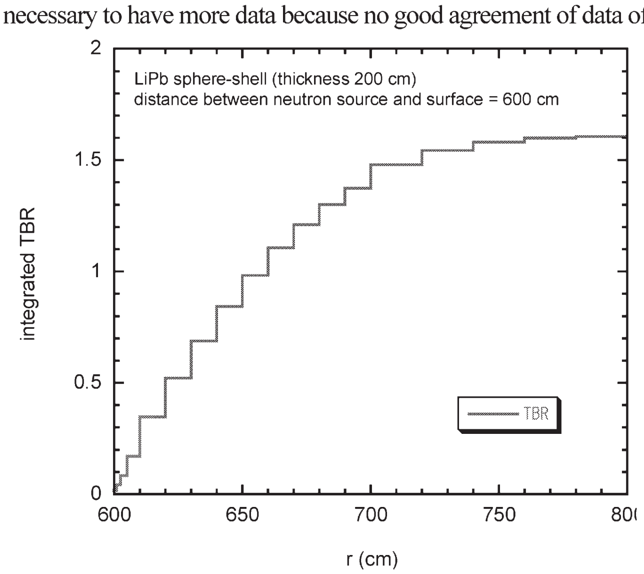

Figure 1 Tritium breeding ratio estimated for LiPb blanket

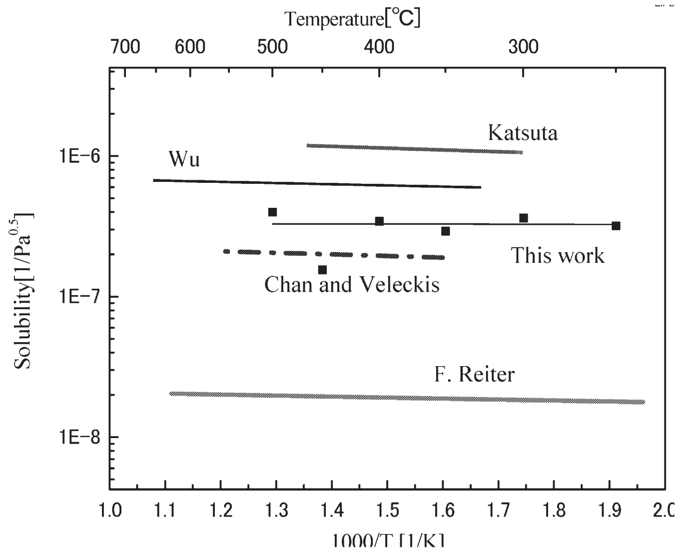

Figure 2 Solubility of hydrogen in LiPb

(3) The tritium release behavior from solid breeder blanket materials are also performed in this cooperative research works. It is assured in this study that the model constructed by the present authors can predict the tritium release behavior from solid breeder materials. Almost all parameters required to estimate the tritium behavior using this code have been quantified in the series of experiment by the present authors.

[Reports published in 2005 related to this study]

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3)Recovery of retained tritium from graphite tiles of JT-60U, T. Takeishi, K. Katayama, M. Nishikawa, N. Miya and K. Masaki, Fusion Sci. and Tech., vol. 48, 565-568(2005).

4)Release behavior of bred tritium from irradiated Li4SiO4, T. Kinjyo, M. Nishikawa, K. Katayama, T. Tanifuji, M. Enoeda and S. Beloglazov, Fusion Sci. and Tech., vol. 48, 646-649(2005).

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6) Diffusion coefficient of tritium through molten salt flibe and rate of tritium leak from fusion reactor system, S. Fukada, R.A. Anderl, A. Sagara and M. Nishikawa, Fusion Sci. and Tech., Vol.48, 666-669(2005).