§6. Design of the Blanket for FFHR

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Various fundamental studies on the design of helical type fusion reactor were conducted. The blanket uses molten salt flibe (Li_2BeF_4) as the breeding material and coolant and ferritic steel as the structural material. Key issues about the feasibility of this blanket are compatibility of the structural materials with flibe, tritium control and hydrology of the flibe as a coolant. The objective of the present study is to reflect the results of recent studies including JUPITER-II to the design of the blanket and to clarify the remaining but important subjects. The activities in 2001 are summarized as follows.

i) Compatibility of Structural Materials with Flibe [1,2]

Thermodynamic calculation and static experiments using pot studied compatibility of structural materials with flibe. Thermodynamic calculation for ferritic steel revealed that Cr_2O_3 can coexist in the system and it might work as a protective layer and that addition of metallic Be has a positive effect on preventing corrosion. By thermodynamic calculation for vanadium alloy, it was found that titanium is attacked and Be-addition has a considerable effect on preventing corrosion.

By corrosion experiments for type 430 stainless steel, it was found that stable oxide of Cr was formed on the surface. This oxide was considered to act as a barrier for corrosion. However, in the case of pure vanadium, destructive corrosion occurred on the surface. This indicates that stable film was not formed on the surface. In order to avoid severe corrosion, the followings were considered: (1) to control the atmosphere in a reducing condition by Be metal addition in Flibe and (2) to control the composition of V-alloy.

ii) Tritium Recovery from Flibe[3]

In order to clarify the tritium recovering process from flibe, in-situ tritium release experiments have been conducted using the research reactor YAYOI at the University of Tokyo. The tritium release was found to be composed of unit processes such as exchange reaction of T^+ with H_2, dissolution and desorption of H_2, desorpton of HT and TF. Among these processes, little knowledge on TF desorption makes it difficult to evaluate the total tritium recovery system. Hence, in order to obtain the kinetic date of TF desorption, non-radioactive experiment was conducted using HF as a simulant to TF. In addition to obtain the kinetic data, the desorption mechanism of HF desorption was also clarified.

iii) Tritium Behavior and Redox Study of Flibe

In flibe/chemistry/tritium studies in JUPITER-II, important experiments at the first stage are purification of flibe and redox control by using pot at INEEL. In Japan, supporting studies have been conducted to help these experiments. They include the effectiveness evaluation of gas purging for flibe purification, measurement of residual tritium by hydrogen isotope exchange reaction, preliminary evaluation of redox control by beryllium and design studies of tritium recovery by permeation method.

Remaining subjects for blanket design using Flibe are tritium control, integration with heat removal system, effectiveness evaluation of redox control by beryllium and its application to blanket system, and design integration of the blanket system. These subjects will be studied by combining each design activity, collaborative experimental studies, and JUPITER-II project.

Reference

1) Terai, T., et al., Fusion Technology **39**, (2001) 784.
2) Nishimura, H., et al., will be published in J. Nucl. Mater.
3) Terai, T., Suzuki, A., Tanaka, S., Fusion Technology **39**, (2001) 769