Innovative Liquid Blanket Design Activities in Japan

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After selecting and proposing the molten-salt Flibe blanket in the LHD-type helical reactor FFHR in 1995 [1], intensive design studies on Flibe blanket have been continued and expanded into wide R&D areas, including materials compatibility, tritium disengager system, advanced thermofluid and MHD effects, and heat-exchanger system, in the NIFS collaborations under the Fusion Research Network in Japan Universities with international research programs such as JUPITER-II. The main purposes of these activities are to clarify key engineering issues and to enhance key R&D activities required for advanced system integration of D-T power plants. Recently an improved long-life Flibe blanket has been proposed, and the self-cooled Li/V blanket design has started, based on intensive R&D works and results. This paper mainly focuses on these liquid blanket designs and related R&D activities in Japan.

New design approaches are proposed for the Flibe / RAFS (reduced activation ferritic steel) blanket in FFHR to solve the major issues of blanket space limitation and replacement difficulty. For the neutron wall loading of $1.5MW/m^2$ as adopted so far in FFHR designs, an innovative concept of replacement-free blankets is possible in the reactor life of 30 years, using carbon armor tiles as ISSEC (Internal Spectral Shifter and Energy Converter), in which the tritium breeding and tiles cooling were key issues [2]. In our new design of the Spectral-shifter and Tritium breeder Blanket (STB) of Flibe in the limited thickness of about 1m, where the Be layers in the both C and Flibe zones are optimized, the fast neutron flux at the first wall of RAFS is reduced to about 1/3 the original flux. At the same time the local TBR of 1.3 is kept, and the fast neutron fluence to the super-conducting magnets is reduced to $8x10^{22}$ n/m^2. According to a preliminary thermal analysis, the surface temperature of the carbon armor is about 1,600°C under conditions of an effective thermal conductivity of 100W/m/K for C-Be-C bonded armors and heat transfer coefficient of $6,000W/m^2/K$ for the mechanical contact using a graphite sheet.

As for heat transfer issues in Flibe, the recent R&D results in the TNT (Tohoku-NIFS Thermofluid) loop experiments using HTS (Heat Transfer Salt) are quite promising. In fact it is clarified that the heat transfer enhancement using packed-bed tube is superior to that of turbulent heat transfer under the same flow-rate condition [3]. The key R&D issues to develop the STB concept, such as radiation effects on carbon and enhanced heat transfer of Flibe, are elucidated.

From the aspects of high thermal conductivity and tritium self-sufficiency without neutron multiplier Be, a liquid Li blanket is another attractive concept. As for MHD pressure drop, the relation between coating material electro-conductivity and allowable crack fraction has been made clear using 2D modeling [4]. Encouraged with intensive R&D progress in MHD coating and high purity V fabrication, optimization studies of Li/V blanket designs are in progress [5].

[1] A. Sagara et al., Fusion Engineering and Design, 29 (1995) 51.
[2] R.W. Conn, G.L. Kulchinski et al., Nucl. Technology, 26 (1975) 125.
[3] S. Chiba, H. Hashizume, A. Sagara et al., submitted to 16th TOFE.
[4] H. Hashizume et al., submitted to ISEM 2003, Versailles, France.
[5] T. Tanaka, T. Muroga, A. Sagara, submitted to 16th TOFE.