§2. Development of 3-D Neutronics Calculation System for Helical Reactor Design

Tanaka, T., Sagara, A., Muroga, T.

Four types of advanced self-cooled liquid blanket systems (Flibe+Be/JLF-1, Flibe-cooled STB, Li/V-alloy and Flibe/V-alloy) have been studied in the design activity of the helical reactor FFHR2 [1-3]. Feasibilities of the blanket systems from the neutronics aspects have been investigated previously with a cylindrical or simple torus geometry assuming that the blanket layers completely covered DT core plasma, i.e. a geometry with no neutron leakage. Results of the investigations indicated that all of the four systems would achieve the compatibility between tritium self-sufficiency and neutron shielding ability within the limited blanket space of 1.2 m. However, for further understanding of the neutronics characteristics in the FFHR2, detailed investigations with a 3-D helical geometry have been required, since DT core plasma is covered with complicated blanket layers running helically in the troidal direction. Construction of a 3-D neutron transport calculation system has been started in the present study.

The 3-D neutron transport calculation system is especially focusing on quick feedback between neutronics evaluations and design modifications of the blanket systems. Cross-sections of the helical blanket components are divided into quadrangular meshes on the design drawings as shown in Fig. 1 (a). The coordinates of the meshes are input data of the calculation system. Vertices of 3-D geometry data are calculated according to the coordinates of the meshes and numerical equations defining the helical structures as shown in Fig. 1 (b). The geometry data are converted to an input file for the Monte-Carlo neutron transport code MCNP-4C. After the Monte-Carlo transport calculation, the results and 3-D geometry data are processed and passed to post-process software for 3-D visualization.

Tritium breeding abilities of the Flibe+Be/JLF-1 and Li/V-alloy blanket systems in the FFHR2m design were investigated with the present calculations system. The cross-section of the geometry for the 3-D calculation is shown in Fig. 2 (a). The geometry data consisted of ~3,000 cells for simulation of the full torus. A uniform torus-shaped neutron source of 1.5 m in diameter was assumed in the investigation.

The total TBRs (Tritium Breeding Ratios) for the Flibe+Be/JLF-1 and Li/V-alloy blanket systems were 0.82 and 0.81, respectively. Comparison with the local TBRs calculated for a simple torus model indicated that the effective coverage of the original FFHR2m blanket configuration was 60-70% due to wide opening between the helical structures around the divertor pumping area. Based on the results, the blanket configuration has been modified to enhance the total TBRs and also improve the shielding performance for the coils as shown in Fig. 2 (b). For the modified blanket configuration, the effective coverage increased to ~80% and the total TBRs of 1.08 and 0.98 could be obtained. Both of the blanket systems would achieve adequate tritium breeding performance in the FFHR2m by design optimization such as blanket dimensions, neutron reflectors etc.

Evaluation of tritium breeding ability for the Flibe-cooled STB system, which has a few more additional layers, has been underway by increasing the number of the cells. Import of geometry data from CAD system is also important function to be installed in the present calculation system for simulation of non-helical structures such as supporting structures, vacuum vessels etc. Plasma distribution will be simulated to examine the relation with neutronics performances. As to another important issue of neutron shielding performance for the superconducting coils, understanding of neutron flow through the helical structures is expected by division of the vacuum areas into small cells and 3-D visualization of the results.

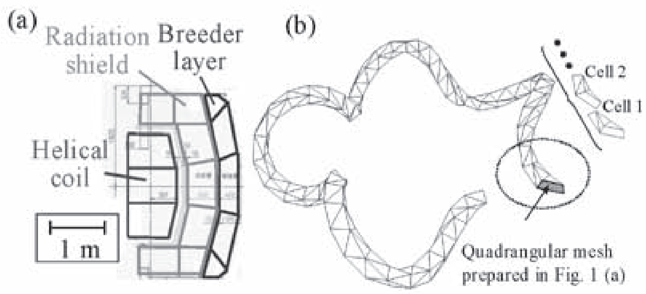

Fig. 1. (a) Division of cross-sections of helical structures into quadrangular meshes. (b) Generation of 3-D helical geometry data according to numerical equations.

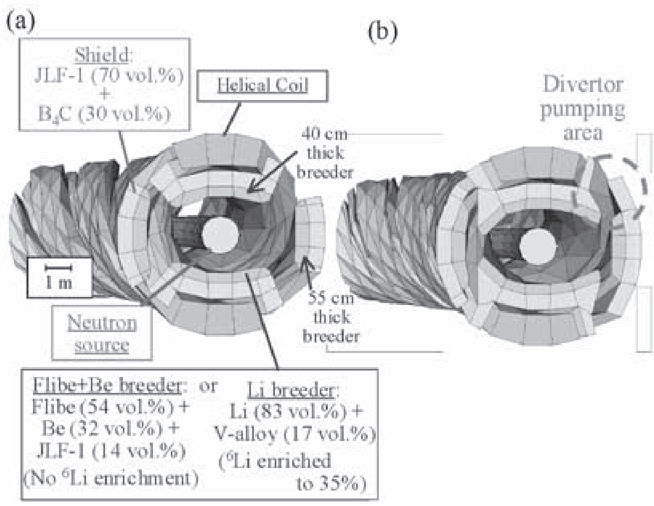

Fig. 2. (a) Cross-section of helical blanket geometry for neutron transport calculations. (b) Cross-section of modified blanket geometry. Breeding and shielding layers have been expanded around divertor pumping areas to improve tritium breeding and neutron shielding performances.

References
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