§1. Conceptual Safety Design Studies towards LHD-type Helical DEMO Reactors

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On the basis of a steady progress in the LHD experiment, a lot of achievements have been made in terms of refinement of the database, physics analysis, and engineering R&D for the helical system. This study advances conceptual design activity of the helical DEMO reactor FFHR-d1 by utilizing these achievements and wide-ranged researches including the core plasma physics and the reactor technology through cooperative researches in NIFS. This study also aims at establishing an engineering basis that enables engineering demonstration of the helical DEMO and contributing to a progress in nuclear fusion research by clarifying issues and prospects of each research field.

This study has been conducted under the Fusion Engineering Research Project, launched at the beginning of FY2010. Conceptual design activity of FFHR-d1 and related engineering R&D have been conducted by 3 research groups (superconducting magnets, in-vessel components, reactor system design) that consist of 13 task groups with a cooperative research between universities and other institutes. In this fiscal year, we focused on the study of the construction/maintenance method and the operational mode of FFHR-d1. Several ideas to enhance safety were proposed.

In the consideration of the construction method, which is regarded as the third round of the conceptual design, discussions were mainly focused on the segmentation method of blanket modules. For the shielding blanket modules, horizontal segmentation was proposed in view of ease of construction and ensuring the accuracy of installation. In connection with this proposal, a new idea for the formation of vacuum boundary by welding of the gap between the shielding blanket modules was proposed. It enables the formation of vacuum boundary after the winding of helical coils without the work in the narrow space between the shielding blanket modules and the helical coils. Consequently, the use of the winding core for the winding of the helical coils similar to the case in the LHD construction was selected as the first option. For the breeding blanket modules, a simple segmentation in a plane with a constant toroidal angle, which enables the replacement of breeding blanket modules only by a combination of uniaxial movements and poloidal rotation, was proposed[1)] (Fig. 1). For the divertor system, advanced liquid divertor concept named REVOLVER-D, which uses multiple vertical free surface flows of molten tin and can handle extremely high heat load with ~100 MW/m^2, was proposed[2)] (Fig. 2).

Relating to the multi-path design strategy[3)] introduced in the FY2013, improvement of the 1-D integrated physics analysis code was advanced by the design integration task group through the collaboration with the Numerical Simulation Reactor Research Project. Analyses of the plasma operation control scenario including the effect of impurity ions and alpha energy loss was conducted. In the design of the superconducting magnet, detailed examination of the sub helical coils, which locate outside of the main helical coils and generate inverse currents against the main coils to enlarge the space for the blanket modules, was started[4)]. The shape and position of the sub helical coils is being optimized by considering the effect on the magnetic field structure and the consistency with the above-mentioned molten tin divertor system. We make a continuous effort on the establishment of a more attractive design concept with putting more focus on the construction and maintenance method by means of the multi-path design strategy including these new ideas.

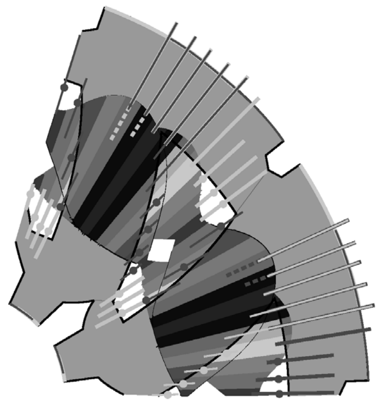

Fig. 1. Schematics of the breeding blanket module with a simple segmentation in a plane with a constant toroidal angle.

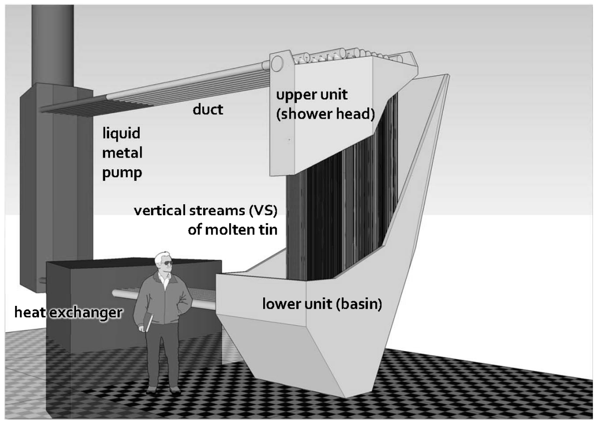

Fig. 2. Schematics of the liquid metal divertor REVOLVER-D

1) Goto, T. et al.: Plasma Fusion Res. **11** (2016) 2405047.
2) Miyazawa, J. et al: Proc. of First IAEA Technical Meeting on Divertor Concepts, P-7, Vienna, Austria, Sep.29 - Oct.2, 2015.
3) Sagara, A. et al.: Fusion Eng. Des. **89** (2014) 2114.
4) Yanagi, N. et al: Plasma Fusion Res. **11** (2016) 2405034.