1-2. Device Engineering and Cooperative Development Research

(1) Physics and Engineering of LHD Torus System

Research and development with regard to the LHD torus system has been executed for an upgrade of LHD and a consequent scenario to a fusion reactor. In particular, efficient pumping/fueling schemes and exploration of plasma wall interaction are emphasized since it is directly related to confinement improvement in the present and future experiment on LHD. Since installation of the closed divertor system is planned in the upcoming experimental campaigns, design study has been accelerated. A cluster of the related studies are motivated by the recognition that particle control is a critical issue in a fusion energy reactor. A comprehensive predictive code has been also developed to study operational scenario of a reactor in this category.

Heliotron magnetic field which LHD employs has intrinsic divertor configuration. However, the present arrangement in the plasma vacuum vessel is so called open divertor. Edge plasma control by the Local Island Divertor (LID) has been very successful in suppression of ambient neutrals and consequently has led to discovery of Internal Diffusion Barrier (IDB) which realizes very high density exceeding $1\times10^{21}m^{-3}$. This finding strongly motivates the closed helical divertor which is promising in improvement of plasma confinement and sustainment of a high-performance long pulse discharge. Since the expected wetted area on the divertor plate is 2 m^2 which is much larger than that in LID, the closed helical divertor has potential to cope with a high-power long pulse discharge. The physical design of the closed helical divertor has progressed based on the measurement of neutral pressure in the present LHD experiment and numerical calculation by EIRENE. The Hα emission measurement as well as the numerical simulation indicates that major neutrals are plugged in the inboard side on the horizontally elongated cross section in the case of the standard configuration with the magnetic axis position of 3.6 m. Therefore, the closed divertor system will be installed from the inboard side of the torus. Arrangement of baffle and new divertor plates is considered so as to fulfill the request to compress neutrals to 0.1 Pa. The divertor plates are made of iso-graphite and mechanically jointed to the cooling pipe, which are tolerable to the heat load of 1.5 MW/m^2 in steady state. Details of the specification of the closed divertor system, assessment of pumping efficiency for a variety of magnetic configuration and capability of new divertor plate against high heat flux are reported in the following pages. Installation of the closed helical divertor system will be done step-by-step and the first partial modification of the plasma vacuum vessel is planned in 2010.

Although the present and the next divertor plates are made of carbon in LHD, high-Z metal material, in particular tungsten, is examined in the research and development study as well as the experiment in LHD. Several divertor plates in LHD have been replaced by the plate with tungsten coating via plasma spray technique. Tungsten is one of promising candidate material for plasma facing materials in a fusion reactor because of its advantages of low sputtering yield, good thermal properties and suppression of retention of tritium. Together with concern about low ductile-brittle transition temperature, characteristics of tungsten should be quantified to assess its feasibility to a fusion reactor. Besides the LHD experiment, evaluation of tungsten coated carbon and deuterium retention has been done by using the electron beam irradiation facility and a linear-divertor-plasma simulator (NAGDIS-II in Nagoya University).

Fueling is also a critical issue in a fusion reactor, nonetheless, it still remains as a Cinderella issue. In addition to extended experimental approach using injection of solid hydrogen pellets, innovative fueling techniques have been being developed. Compact Toroid (CT) injection is an advanced fueling method. Recently, as a new approach to effective fuelling, injection of extremely super-high speed neutral particle flow by using a CT injector has been proposed. The CT injector of SPICA (SPeromak Injector using Conical Accelerator) provides the experimental platform to examine injection of CT and super-high speed neutral particle flow for LHD. The target is to efficiently produce low energy and high particle flux with a high speed of 300 km/s (equivalent to about 1 keV) and a high density of 10^{21} m^{-3} through charge-exchange between CT plasma and neutral gas in the neutralizer cell. The particle flow injection is expected to have deeper penetration and higher fuelling efficiency than a super-sonic gas puffing. SuperSonic cluster beam (SSCB) injection method is also being developed as a new fueling method for LHD. SSCB is an improved version of cluster jet injection or the supersonic gas injector. SSCB ejects high-pressure hydrogen gas cooled down to less than 77 K by a GM refrigerator through a fast solenoid valve with a Laval nozzle.

Besides the above mentioned engineering subjects, a useful code is being developed to prospect the future. A toroidal transport linkage code TOTAL (Toroidal Transport Analysis Linkage) copes with both tokamak and helical system, which is verified in terms of ITB and impurity transport.

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