§2. System Design of the Helical DEMO Reactor FFHR-d1

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i) Design window analysis for Helical DEMO Reactor FFHR-d1

On the basis of the findings in the experiment at the LHD[1)], conceptual design activity of the helical DEMO reactor FFHR-d1 has been conducted by the Fusion Engineering Research Project in NIFS since FY2010[2)]. As the next generation reactor after the LHD, FFHR-d1 aims at an early demonstration of maintainability, tritium self-sufficiency and net electric-power generation. The design of FFHR-d1 is based on the engineering knowledge base established by the past FFHR series[3)] but focuses more on the certainty of the extrapolation from LHD. In the first step of the conceptual design process, design window analysis was conducted using the system design code HELIOSCOPE[4)]. Here a new approach to determine the core plasma profile by direct extrapolation from the LHD experimental results, Direct Profile Extrapolation (DPE) method[5)], was used. A typical plasma profile obtained by pellet fueling and neutral beam heating with the LHD standard configuration was selected. In this study, the same constraint for the stored magnetic energy, $W_{\rm mag} \leq 160$ GJ, is assumed as an engineering constraint. In the LHD experiments, the edge electron density n_{ea} is limited by the Sudo density scaling[6)]. In this respect, another boundary of the design window is defined by $n_{ea}/n_{\rm Sudo} \leq 1$. The remaining constraints are confinement improvement factor $\gamma_{\rm DPE}$ and beta enhancement factor f_β. At present, there is no clear finding to determine the upper limit of these values. Thus we assume $\gamma_{\rm DPE} = 1.3$ and $f_\beta \leq 5$ (which corresponds to $\beta_0 \leq 10\%$ in this case).

Figure 1 shows the contours of these design parameters. The design window without shading corresponds to the region that satisfies all the design constraints. The design point having the maximum space for the blanket $\Delta_{\rm c\text{-}p}$ within this design window was selected as a candidate for FFHR-d1; $R_{\rm c} = 17$ m and $B_{\rm t,c} = 4.7$ T (an average toroidal field strength on the helical coil winding center), and $\Delta_{\rm c\text{-}p} = 89$ cm. In order to confirm the validity of this design point, additional analysis was conducted by related task groups.

ii) Mathematical expression of the shape of in-vessel components

After the decision of the main design parameters, the detailed design of the in-vessel components (vacuum vessel, blankets and divertor components) is in progress. The basic shape of the in-vessel components is set using the result of the magnetic field line tracing calculation. Similar to the vacuum vessel design of LHD, the shape of vacuum vessel and blankets around the helical coil is defined by mathematical formulae (trigonometric functions of the toroidal angle on the helical-coil vertical cross-section). The shape of vacuum vessel and blankets at the other location are defined as an arc on the poloidal cross-section. In order to utilize the helical divertor configuration that can pull out the divertor strike point at the point distant from the core plasma, we consider to place divertor components behind the blanket in order to avoid direct exposure by the fusion neutrons. An elliptic arc instead of a circular arc is adopted to enlarge the space behind the blanket as shown in Fig. 2.

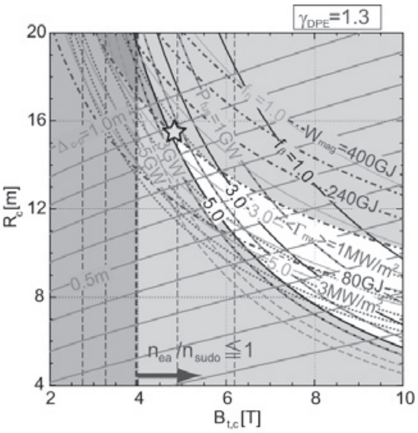

Fig. 1: Result of design window analysis for FFHR-d1. Star indicates candidate design point for FFHR-d1.

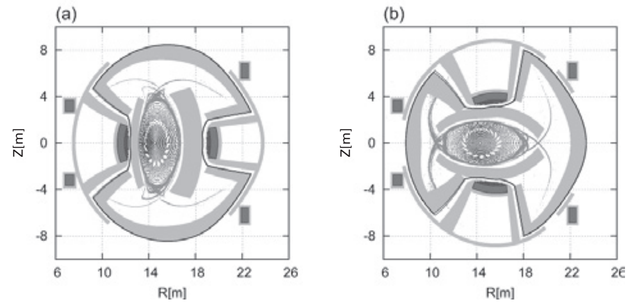

Fig. 2: Proposed basic shape of in-vessel components on (a)vertically- and (b)horizontally-elongated cross-section.

1) A. Komori *et al.*, Fusion Sci. Technol. (2010) vol. 58 pp.1-11.

2) A. Sagara *et al.*, Fusion Eng. Des. (2012), in press.

3) A. Sagara *et al.*, Fusion Eng. Des. (2006) vol. 83 pp.1690-1695.

4) T. Goto *et al.*, Nucl. Fusion (2011) vol. 51 pp.083045.

5) J. Miyazawa *et al.*, Fusion. Eng. Des. (2011) vol. 86 pp.2879-2885.

6) J. Miyazawa *et al.*, Fusion. Sci. Technol. (2010) vol. 58 pp.29-37.