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Research and development status on fusion DEMO reactor design under the Broader Approach

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HIGHLIGHTS

- Latest status of the DEMO design activity under the Broader Approach.
- Start points of DEMO parameter scoping study.
- DEMO divertor study aiming at detached plasma with impurity injection and magnetic flux tube expansion by advanced divertor configurations.
- Assessment of possible remote maintenance schemes for DEMO.

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ABSTRACT

The main objective of DEMO design activity under the Broader Approach is to develop pre-conceptual design of DEMO options by addressing key design issues on physics, technology and system engineering. This paper describes the latest results of the design activity, including DEMO parameter study, divertor and remote maintenance. DEMO parameter study has recently started with “pulsed” DEMO having a major radius (R_p) of 9 m, and “steady state” DEMO of R_p = 8.2 m or more. Divertor design study has focused on a computer simulation of fully detached plasma under DEMO divertor conditions and the assessment of advanced divertor configuration such as super-X. Comparative study of various maintenance schemes for DEMO and narrowing down the schemes is in progress.

1. Introduction

In the DEMO design activity under the Broader Approach (BA), the first task was to crystalize the joint design work to be implemented by EU and Japan (JA). After three-year preparatory activity (2007–2010) in which various views on DEMO were shared by EU and JA researchers in workshops, the details of the joint design work were agreed on and the joint work started in 2011.

Previous DEMO design study carried out before the BA [1–3] revealed critical design issues on huge power removal in the divertor, tritium self-sufficiency and power extraction, remote maintenance, integrated design including plasma control and so on. All these issues are not only difficult themselves but each one is closely related with the other. Therefore, in the design integration, it is required to understand trade-off relations and intertwined constraints underlying in these issues and resolve all of them systematically. For the purpose, the BA DEMO design has concentrated on assessment of critical design issues, which includes the consolidation of the knowledge base so far achieved and needed for the design, and analysis of key design issues and options. In the process, every possible option of each component or technology is investigated and assessed for narrowing down to feasible options. The assessment will contribute to developing pre-conceptual DEMO options and planning future technology development programs toward DEMO.

In parallel with the study on critical design issues, DEMO design parameters need to be investigated to envisage possible DEMO options. Benchmark of systems codes independently developed by CCFE (Culham Centre for Fusion Energy) and JAEA (Japan Atomic Energy Agency) has been carried out, confirming good agreement

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except several issues such as impurity radiation power, fast ion beta and bootstrap fraction [4,5].

This paper describes the main areas of the activity, this is, progress in DEMO design parameters, divertor and remote maintenance in the following sections. The plan for safety research for DEMO is briefly summarized in Section 5.

2. Parameter study of DEMO

The basic approach of the BA DEMO design is to develop possible DEMO options that are foreseen from ITER, other on-going and planned R&D programmes. In this context, projections of ITER and the present knowledge to DEMO should be based on a certain broadening dependent on conservative and optimistic views. Another aspect to note is that EU and JA have different requirements for DEMO. Consequently, multiple DEMO concepts are likely to be developed in the BA.

EU has explored pulsed and steady state power plants and actually a pulsed option was characterized as the most conservative model in the European Power Plant Conceptual Study (PPCS) [2]. In the BA activity, EU puts emphasis on a pulsed DEMO concept assuming small extension of ITER technology. Preliminary scoping study using an EU systems code is presently pursuing a pulsed DEMO generating net electricity (P_e) of 0.5 GWe and fusion power (P_{fus}) of around 2 GW at a major radius (R_p) of 9 m, aspect ratio of about 4 and pulse length of as short as about 2 h.

Previously, JA reactor study had focused on steady state DEMO concepts that were relatively compact and capable of producing plant-level electricity (~1 GWe) [1,3]. The previous reactor study reveals that such compact reactors have problems in power removal in the divertor, removal of residual heat of the in-vessel components in a loss-of-coolant accident (LOCA) and tokamak operation capability due to insufficient volt-second supply. Based on the result, JA currently considers a relatively large and lower power steady state concept with sufficient volt-second supply for operational development from pulsed (~0.5 h) to steady state, having parameters of $R_p \sim 8.2$ m or more, $P_{fus} = 1.3$–1.5 GW and $P_e = 0.2$–0.3 GWe.

In the scoping study, it is commonly recognized that a vertically stable plasma elongation under constraints of DEMO needs to be investigated, which means no use of in-vessel coils for vertical stabilization and the installation of stabilizing shells away from breeding blanket.

3. Divertor

Divertor is one of the most critical design issues giving a significant impact on DEMO design parameters. The problem is how to handle a huge power exhausted from the main plasma being several times as high as that in ITER and eventually to maintain the divertor heat flux at a tolerable level well below 10 MW/m^2. In order to find a possible solution for the problem, two approaches have been progressing. The first approach is to pursue a fully detached plasma condition based on divertor simulation [6,7], and the second is to seek for the possibility of modifying divertor configuration such as super-X and snowflake [8].

3.1. Divertor simulation

Admitting that existing divertor simulation codes have not satisfactorily been validated by experiments, computer simulation is practically the only approach to foresee DEMO because the DEMO conditions are substantially far away from existing experiments. For the JA divertor design study, a divertor simulation code SONIC was upgraded to appropriately deal with thermal and friction forces of impurity ions along the magnetic field lines. In the simulation, impurity transport and the resulting plasma detachment were simulated self-consistently for different detachment scenarios.

3.1.1. Impurity injection

Impurity seeding using different species (Ne, Ar and Kr) is anticipated to change radiation distribution in the scrape-off layer (SOL) and the divertor because of the Z-dependence of radiation loss rate coefficient. The divertor simulation indicates that selecting the impurity species can control the distribution of radiation power in that, with seeding higher-Z impurity such as Kr, a wider plasma detachment region and lower divertor heat flux were obtained at the outer divertor [7]. On the other hand, higher-Z impurity tends to be transported to the main plasma and to cause higher radiation in the main edge, which may deteriorate plasma confinement.

3.1.2. Long leg divertor

Previously, it was confirmed that the divertor geometry contributes to reducing divertor heat flux and that V-shaped corner is efficient to enhance particle recycling near the strike point and to produce detached plasma [9]. In order to reduce the divertor heat flux further, longer leg divertor was proposed and investigated for SlimCS [6]. The role of the long leg divertor is to decrease the divertor plasma temperature because of extended connection length to the target plate and to increase the heat receiving area of the divertor.

Recent simulation [7] indicates that, when Ar is seeded to the divertor region of the plasma with a fusion output (P_{fus}) of 3 GW, full detachment over the outer divertor is attained as shown in Fig. 1(a). Then, the radiation power fraction reaches 80% of the power exhausted to the SOL from the main plasma (500 MW) and the divertor electron and ion temperatures decrease to 2 eV. In the long leg divertor, the ionization front moves upstream (Fig. 1(b)), contributing to reducing the peak heat load on the divertor plate.

3.2. Alternative divertor configurations

Another approach for reducing the divertor heat load is a flux tube expansion in the divertor with "super-X" or "snowflake" configuration. Application of such configurations has been investigated by EU [10] and JA [8].

Fig. 2 depicts the arrangement of poloidal field (PF) coils for the super-X configuration and the required coil currents, where the original plasma configuration is based on SlimCS with a plasma current of 16.8 MA. The super-X configuration is numerically obtained using external PF coils. However, the reality of coil current required for Coil #9 of 180 MA seems unlikely because the cross section for the current is anticipated to be 4–6 m^2 when the conductor current is 60–100 kA. Development of high current conductor of 150–200 kA can allow the super-X with external PF coils but the leak field produced by such a high current coil may restrict the installation of peripheral equipment such as diagnostics, solenoid valves and pumps. By winding some of the PF coils in an inter-linked way against the toroidal field (TF) coils, the required coil currents are dramatically reduced as shown in Fig. 2(b). Inter-linked winding was originally proposed as a novel concept of central solenoid (CS) allowing a compact CS with a sufficient volt-second supply [11,12]. The winding needs to be carried out in situ in the middle of the TF coil installation and the vacuum vessel one. Although the engineering feasibility of the inter-linked winding must be verified from various aspects, the required coil currents have reality. Further study is under way from the points of view of the compatibility of the inter-linked coils and the neutron and thermal shield. Regarding the super-X divertor, it is noteworthy that the connection length along a magnetic field line near the super-X null point is extended. The connection length of the magnetic field line