Design study of large superconducting coil system for JA DEMO

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Generally, DEMO requires larger toroidal field (TF) coils than ITER, resulting in two major difficulties, the tolerance in TF coil fabrication and installation and an increased inductance. This paper presents the possible solutions these on the basis of the design study on Japan’s DEMO (JA DEMO). It was confirmed that, in the case of adopting a mitigated tolerance by a factor of 2.5-5 compared with that of ITER, the resulting error field of TF coils is correctable to an acceptable level in terms of locked mode avoidance. The validity of a fast discharge scheme of TF coil current at the discharge time constant of less than 30 sec was confirmed, leading to a reasonable terminal voltage of each TF coil and the consistency with the electromagnetic forces acting on the vacuum vessel due to the induced eddy current. The problems resulting from large-scale TF coils commonly used in DEMO can be resolved with these approaches.

1. Introduction

Japan’s DEMO (JA DEMO) requires larger toroidal field (TF) coils than ITER to attain higher fusion output (about 1.5 GW) with larger plasma volume and to allow the installation of breeding blanket inside the TF coil bore. The adoption of such large TF coils results in two major difficulties, (1) the tolerance in TF coil fabrication and installation (2) an increased inductance. Main design parameters of the JA DEMO are a plasma major radius of 8.5 m, fusion output of 1.5-2 GW, the net electricity of 0.2-0.3 GW, and magnetic field on the plasma axis of 6 T [A]. The superconducting coil system of the JA DEMO consists of 16 toroidal field (TF) coils, a central solenoid (CS) and 7 poloidal field (PF) coils, as shown in figure 1. To demonstrate steady-state electric power generation in a power plant scale, a higher magnetic field strength and a 1.5 times larger TF coil bore than those in ITER are necessary [B].
2. TF coil tolerance issue and error field correction

The design concept of TF coil is basically similar to that of ITER, that is usage a radial plate, double pancake and cable-in-conduit conductor. One of the issues specific to the large-sized steady-state tokamak DEMO concept is the increasing technical difficulties due to the larger toroidal field coils than those in ITER. The fabrication tolerance has an effect on engineering difficulties of large coil system. JA DEMO adopts the design strategy of TF coils of mitigating the error field requirement and the error field is corrected by using an EFCC if needed to avoid locked modes. The target error field is set at $\delta B_{TMEI}/B_T \sim 10^{-4}$ in JA DEMO design. Here, $\delta B_{TMEI}$ is three mode error index (TMEI) defined by Fourier components $B_{1/1}$, $B_{2/1}$, and $B_{3/1}$ of poloidal/toroidal mode number $m/n = 1/1$, $2/1$ and $3/1$ on $q = 2$ surface. The $\delta B_{TMEI}$ of 0.6 mT is targeted for an error field correction with respect to the TF coil, PF coil and CS coil error fields in DEMO. The manufacturing and assembly tolerances could produce non-axisymmetric magnetic fields, leading to error fields. The relation between the tolerances and the resulting error fields have been calculated with a Monte-Carlo approach. The previous calculation results showed the mitigated target of error field of about $\delta B_{TMEI}/B_T \sim 10^{-4}$ provides a mitigation in the tolerance in coil fabrication and installation by a factor of 2.5-5 compared with that of ITER, contributing to realize the fabrication of larger coils for DEMO. In order to determine the EFCC currents, we have applied the least square method to minimize three error field components of $m = 1, 2$ and $3$ with $n = 1$ at the $q = 2$ surface. In the estimations, anti-series connections between coils located at the opposite side toroidally are assumed. In order to avoid interference with maintenance ports and NBI ports, the EFCCs has two types which are different toroidal angle ($67.5^\circ$ and $45^\circ$) located on plasma outboard side, where would be an effective position for error field mitigation are newly adopted, as shown in figure 2. As a result, the EFCC current required to correct $\delta B_{TMEI}$ up to 0.1 mT was 200 kAT even when a set of EFCCs are arranged outside the vacuum vessel in a toroidally non-periodic manner. The result indicates that the error field caused by mitigated tolerances of TF coil fabrication and installation can be corrected with ex-vessel EFCCs at realistic coil currents.
3. Fast discharge scheme of TF coil current
The large TF coil size of the JA DEMO gives a 3 times greater self-inductance if the conductor current and the magnetic field strength are the same magnitudes as those of ITER and then generates a 3 times terminal voltage that is quite capable of losing reliability of TF coils. Focusing on issues related to increase in the coil self-inductance due to increase in the TF coil size, identification of discharge time constants and fast discharge scenario [C] that match other reactor systems were investigated. The increasing in the discharge time constant reduces the terminal voltage, however the temperature acceptable limit of the conductor must be not exceeded in the coil quench event. In order to evaluate the upper limit of discharge time constant, the time evolution of conductor temperature on the fast discharge event is calculated, as shown in figure 3. It is found that the discharge time constant requires less than 30 seconds, which is consistent with a vacuum vessel design (less than allowable stress of SUS316L, 143 MPa@100°C), as shown in figure 4. The TF coils are divided into serially connected segments that are electrically isolated from each other and only the coil segment having a failed coil is rapidly discharged. The circuit current analysis indicates that this discharge scheme enables to reduce the terminal voltage with a factor of 0.6 or less and which would contribute to ensure reliability of the turn insulations. These results shown that the validity of a fast discharge scheme of TF coil current at the discharge time constant of less than 30 sec was confirmed, leading to a reasonable terminal voltage of each TF coil and the consistency with the electromagnetic forces acting on the vacuum vessel.
Figure 3: Time evolution of peak temperature of helium coolant, radial plate and conductor jacket, and helium peak pressure on TF coil fast discharge event at discharge time constant of 20, 30 and 40 seconds.

Figure 4: Tresca stress distribution of vacuum vessel on TF coil fast discharge event at discharge time constant of 30 seconds.
References
[A] Y. Sakamoto et al., 27th IAEA Int. Conf. on Fusion Energy (2018) FIP/3-2

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