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Two Conceptual Designs of Helical Fusion Reactor FFHR-d1A Based on ITER Technologies and Challenging Ideas & Development of Remountable Joints and Heat Removable Techniques for High-temperature Superconducting Magnets & Lessons Learned from the Eighteen-Year Operation of the LHD Poloidal Coils Made from CIC Conductors

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A. The Fusion Engineering Research Project (FERP) in NIFS is conducting the conceptual design activity of the LHD-type helical fusion reactor FFHR-d1A. Recently, two options of "basic"and "challenging"have been newly defined. Conservative technologies including what will be demonstrated in ITER are chosen in the basic option, while new ideas that would possibly be beneficial for making the reactor design more attractive are boldly included in the challenging option, aiming at "early construction, easy maintenance and high thermal efficiency", in particular, for helical structure. In the basic option, the SC magnet coils adopt cable-in-conduit conductors with Nb3Sn strands cooled by supercritical helium at 4.5 K. The helical coils are wound by the "react and winding"method using a large-scale winding machine. The divertor system is the water-cooled tungsten monoblock divertor with cooling pipes made of Cu alloy. The blanket system is the water-cooled ceramic breeder blanket. The key technologies needed for the basic option are already well established in LHD or will be established through the R&D activities for ITER. However, we need to develop the maintenance schemes for these helical divertor and blanket with complicated 3D structures. In the challenging option, on the other hand, new technologies of the high-temperature superconductor (HTS), the liquid metal ergodic limiter/divertor, and the molten salt (FLiNaBe mixed with metal powders) breeder blanket are adopted. The "joint winding" based on the mechanical lap joint technique are applied to fabricate the helical coils by connecting segmented HTS conductors. The cooling scheme is simplified using helium gas at 20 K. A new liquid metal limiter/divertor has been proposed, where 10 units forming the molten tin shower jets stabilized by chains inside each jets are installed in the inboard side of the torus. Neutral particles are evacuated through the liquid metal shower. The blanket system using the metal powder mixed FLiNaBe is also the challenging option. The hydrogen solubility is effectively increased by adding powders of hydrogen storage metal such as Ti. Although the new technologies adopted in the challenging option can significantly ease the construction difficulties in the basic option, we have already started R&D arrangements to demonstrate them as fast as possible.

B. This study addresses development of mechanical joints and a heat removable technique for the remountable HTS magnet. We carried out (i) Optimizing structure and fabrication procedure of mechanical joints, and (ii) Analyzing heat transfer performance of metal porous media inserted channel to be applied to thermal analysis of joint. The developments and discussion will be taken into account in design of the remountable magnet. The "remountable" HTS magnet has been proposed for both tokamak and heliotron-type fusion reactors, which is assembled from coil segments with mechanical joints. Our recent study successfully developed a bridge-type mechanical lap joint of 100-kA-class HTS conductors consisting of simple stacking of REBCO HTS tapes. The joint achieved a joint resistance of 1.8 $n\Omega$ at 100 kA, 4.2 K. For the local heat removal, we have proposed a metal porous media inserted channel and experimentally evaluated its heat transfer performance with liquid nitrogen (LN2) to show heat transfer coefficient of 10 $\mathrm{kW/m^2K}$ at low mass flow rate. However, it took over a half day to fabricate the joint because the joint piece was not integrated but just individual REBCO tapes. In addition, the fabricated joints all had straight geometry. Furthermore the joint resistance obtained with large-scale conductor joint was larger than predicted value based on small-scale conductor joint and varied largely due to non-uniform contact pressure distribution on the joint surfaces. Therefore, an integrated joint piece was newly introduced to shorten the fabrication process. Furthermore, we apply heat treatment during fabrication of the joint to reduce the joint resistivity from 25 pΩmˆ2 to 8 pΩmˆ2. This means the heat treatment is promising to be applied to large-scale conductor joints. For the local heat removal, we need to predict heat transfer coefficient of various cryogenic liquid coolants such as liquid helium (LHe), liquid hydrogen (LH2) and liquid neon (LNe) with a metal porous media inserted channel, due to operating temperatures of <30 K. At the constant pump power and each coolant's saturated temperature, LH2 and LNe show almost the same heat transfer coefficient and DNB point. The DNB point for LH2 and LNe is about 10 times larger than that for LHe.

C. The Large Helical Device (LHD) superconducting magnet system consists of two pairs of helical coils and three pairs of poloidal coils. The helical and poloidal coils use composite conductors with pool cooling by liquid helium, and cable-in-conduit (CIC) conductors with forced cooling by supercritical helium, respectively. The poloidal coils were first energized with the helical coils on March 27, 1998. Since that time, the coils have experienced 50,000 h of steady cooling, 10,000 h of excitation operation, and seventeen thermal cycles. During this period, no superconducting-to-normal transition of the conductors has been observed, even during fast current discharge. The subsequent experience gained from eighteen years of operation has also provided further useful information regarding preventive design and maintenance of peripheral equipment and long-term changes in electromagnetic and hydraulic characteristics. First, the poloidal coil system has experienced events that have interrupted plasma experiments due to the malfunction of a quench detection system and an insulating break used in cryogenic piping. The malfunction of the quench detection system was caused by the noise from the plasma heating devices and the coupling current, which is an intrinsic property of a composite superconductor. This suggested that the noise for quench detection should be estimated before magnet operation. During the sixteenth cool-down, one of the breaks suddenly leaked helium, which was caused by the cracking of a plastic adhesive material between the FPR and the stainless-steel pipes. Further investigation is needed to clarify the age degradation and the creep behavior of plastic adhesive materials at cryogenic temperature. Second, the long-term monitoring of the electromagnetic and hydraulic characteristics of the coils has also been performed. Even though the AC losses slightly decreased during the first three years, the losses have remained unchanged in the fifteen years since. The pressure drops of coolant showed a tendency to decrease over the campaigns. The sudden increase in friction factor in the 15th campaign suggested that the compressor oil is a potential source of impurity gasses in helium coolant. These experiences would help in the design and maintenance of preventive measures for fusion magnets, including ITER and DEMO.

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