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Thermal Hydraulic Analysis of Toroidal Field Coil of CFETR

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Abstract-China Fusion Engineering Test Reactor (CFETR) is the next device in the roadmap for the realization of fusion energy in China, which aims to bridge the gaps between the fusion experimental reactor ITER and the demonstration reactor (DEMO).
The toroidal field (TF) coils play a major part in the tokamak, which provide the main magnetic field to confine the plasma, and one TF coil has been design in ASIPP since October, 2019, and planned to be completed in 2025.
CFETR consists of 16 TF coils and the length of major and minor radii are R=7.2 m and a=2.2 m. The TF magnet system will provide a 6.5 T magnetic field at the plasma center (at 7.2 m position) to confine the plasma. The plasma scenarios are characterized by a pulse length of 5000 s or more, with a plasma current 14 MA. In order to evaluate the feasibility of superconducting magnets used in CFETR, the thermal-hydraulic state of the coils is analyzed (e.g., normal operation, plasma disruption, quench, fast current discharge). The inlet and outlet pressure of helium cooling loops and operational temperature of the magnet is designed. The temperature margin of the superconducting magnets under different operation conditions are estimated.

Index Term- CFETR, totoidal field, thermal-hydraulic, temperature margin

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