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Analysis of the thermal-hydraulic effects of a plasma disruption on the DTT magnet system

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The power exhaust represents a major challenge for the EUropean fusion reactor DEMOnstrator, asking for a new, robust design of the divertor, beyond the solution currently pursued for ITER. For this reason, the construction of a satellite fusion experiment, the Divertor Tokamak Test (DTT) facility, is being pursued in Italy. This compact tokamak, which must be very flexible in terms of plasma configurations, will test several DEMO-relevant divertor solutions.
The DTT superconducting magnet system includes 18 Toroidal Field (TF) magnets, 6 modules constituting the Central Solenoid (CS) and 6 Poloidal Field (PF) coils. All of them are cooled by forced-flow supercritical helium at 4.5 K. In the case of a plasma disruption, the fast reduction of the plasma current causes a magnetic field variation, inducing on one hand a sudden variation of the current in the PF and CS coils, and on the other hand AC losses in the conductors and eddy currents in the bulky steel structures of the TF coils. The former effect may increase the current in the coils above its nominal value, while the latter causes a heat deposition that could erode the available temperature margin. Both the effects can initiate a quench, requiring then a fast discharge of the coils.
The detailed thermal-hydraulic model of the DTT magnet system, developed using the 4C code, is applied here to simulate the effects of the plasma disruption. The heat load to the coil casing and to the winding pack, as well as the current evolutions, are used as input to compute the temperature margin during the transient and to assess if a quench could be initiated.
The results will give important feedbacks to the design of the protection system, such as the option of triggering a fast current discharge right after the disruption.

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