Progress on the design of various magnetic sensors for ITER:

Diamagnetic and flux loops, high frequency coils, in-vessel and divertor coils, ex-vessel coils, and external Rogowskis

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The ITER magnetic diagnostics is required to be extremely reliable since it provides essential measurements for machine operation and protection, plasma control and for diagnostic interpretation. The magnetic measurements can be divided into local measurements of magnetic field components, in-flux measurements (circular and saddle loops) and measurements of currents (Rogowski coils). All measurements are based on field or flux changes with respect to time, which requires a precise integration of the signals. On the other hand, the diagnostic set also provides essential measurements of the MHD mode activities. For this reason, reliability is the dominant design requirement, especially in the area of wiring and connections and in choosing margin against radiation damage and anomalous electrical loads.

CRPP is responsible within the EFDA task program for the design, calculation and performance analysis of the high frequency coils, MHD saddles loops, diamagnetic loop system and divertor coils for ITER.

Inductive probes are the primary means of measuring the local magnetic field outside the plasma boundaries. Such probes may be used to measure both the equilibrium field and fluctuation related to plasma instabilities. In order to detect MHD activities, the high frequency response of the coils is of major interest. The HF coils are responsible for the measuring of low (m,n) MHD modes, sawteeth, disruption precursors, as well as high frequency macro instabilities (fishbones, TAE modes) in the main system and for determining the plasma current and plasma position and shape in the backups.

The main common design criteria for the HF coils and the saddle loops is related to the effect of the so-called θ^*-correction to the probe poloidal angle, as defined by their location on the vessel and the position of the magnetic axis. To obtain the poloidal mode numbers, the physical position of the magnetic probes, mounted on the vessel, has to be corrected for the curvature of the poloidal magnetic field lines at the mode radial location. The θ^*-correction is required to accurately reconstruct the poloidal mode structure from edge measurements due to contributions from the Shafranov shift and the plasma shape (elongation and triangularity). This correction depends on prior knowledge of the flux surfaces on which the modes are located, and it can have a significant effect on probes located on the low field side due to ballooning effects.

Since it is known that one of the basic assumptions of the magneto-hydro dynamical theory (MHD) is the helical symmetry of the magnetic perturbations, a poloidal set of coils is enough to measure $B\theta$ perturbations and to determine the poloidal harmonics (usually m< 10). The high frequency coil is located in the gap between blanket modules and the wall. They are just outside the direct view of the plasma to reduce nuclear heating. The high frequency coils are distributed along the poloidal contour in 6 sectors displaced by 60˚ toroidally. They are usable at up to 2 MHz. In order to cover up to m ~ 10, 20 high frequency coils were primary foreseen. In fact, only 18 high frequency coils are placed in each sector, due to the restriction to one/blanket module in the main chamber.

According to the recent ITER design, the high frequency coils are enclosed in a heat shield, providing both protection from the plasma and prevention from interfering with other circuits. The effective area of the coil is 0.075 m^2. All HF coils are replaceable.

The MHD saddles are responsible for the locked modes, sawteeth, disruption precursors and low (m,n) MHD modes in the main system and for the plasma position and shape in the backups. The MHD saddles can be used also as backup measurement for the equilibrium reconstruction, if necessary. In principles, fast measurements are required for the modes. The time response of the eddy currents in the vessel limits the usefulness of fast sampling of the radial field. For axisymmetric modes (n=0) the m = 3 eddy timescale is of the order of 100ms. For this reason a sampling time of 200Hz for these measurements is set.

The dedicated saddles are mounted on the inner wall of the vacuum vessel. MHD dedicated saddles exist on 9 machine sector pairs (40° apart toroidally). Poloidally, there are eight of these loops, mounted in each of the sector pairs, 72 in total. They are constructed in a similar way as the equilibrium flux loops. They are simply loops of mineral insulated cable attached to the vessel thermally and mechanically via resistance-welded clips at frequent intervals. They are made from 2mm MI cable. The saddle loops are permanent.

The diamagnetic loop system measures the toroidal magnetic flux expelled by the plasma, for the purpose of estimating the perpendicular thermal energy of the plasma. The relation between the diamagnetic flux and the thermal energy content depends on the plasma shape and equilibrium. Thus, the measured diamagnetic flux must be included in the measurement set used by the inverse equilibrium reconstruction to obtain a correct estimate of the perpendicular thermal energy. Typical maximum value for the diamagnetic flux on ITER is obtained for maximum plasma current and large β_p plasmas giving a figure of $\Phi_P = 3.6Wb$ for $I_p = 10MA$ and $\beta_p=3$.

The main poloidal loop is mounted on the inner wall of the vessel and has a surface area of $35m^2$. For redundancy, there are three loops separated by 120° toroidally. The material is MI cable with diameter of 2mm. The loop is attached to the wall by simple spot welded clips. In areas where the gap cannot be guarantied mechanical stops are used for protection. Two loops, wired in parallel are used in order to circumvent the obstacles.

The compensation of the screening effects of the vessel eddy current is important to increase the bandwidth of the diamagnetic measurement which would otherwise be limited by the characteristic fall-off time of poloidal vessel currents. Multi-turn compensation coils having 20 m^2 coil effective area are placed inside the vacuum vessel below the outer divertor support rail, halfway between ports. This effective area is required to ensure that the integration error after 3600s is similar to that of the tangential field coil circuit (~2 mVs). This corresponds to an effective offset of 540 nV. The compensation coil is located in a low nuclear heating area and is partly shielded from the vessel gussets.

The divertor is one of the components being exposed to the highest peak heat load in a fusion reactor. About 15 % of the energy produced by fusion reactions must be removed through the divertor, yielding a thermal peak load of up to 20 MW/m^2, which constitutes a significant technical challenge. The divertor has to absorb this extremely high heat fluxes and has to keep the plasma impurities at a reasonable low level. Divertor equilibrium coils are designed to supplement the in-vessel magnetics set for the measurement of separatrix-wall gaps and reconstruction of equilibria (plasma shape and position). They are found to improve the reconstruction accuracy near the X-point.

72 (6x6x2) coils are located on 6 divertor cassettes (ports 02, 04, 08, 10, 14, 16 (6 position). The divertor coil system is comprised of pairs of equilibrium coils normal and tangential to the mounting surfaces of selected cassettes (six coils with an axis perpendicular to divertor cassette elements, and six separate coils at equivalent positions with an axis parallel to divertor cassette elements). The construction of divertor coils will be similar to the inner vessel coils. Additionally, the cooling should be better (as the nuclear heating is higher), and the effective area somewhat higher, to reduce the sensitivity to junction and other parasitic

potentials in the connector. The apparent volume constrains (2x10x10x10 cm^3) are not as strict as by in-vessel coils (9x4.5x5 cm^3). Proposed coil effective area should be ~0.5 m^2 which corresponds to output voltage ~50 V.

Divertor cassette region is characterised with an extremely high non-homogeneity of neutron heat flux, which varies from 100 to 1 kW/m^2. The maximum nuclear heating up to 2.5 W/cc expected on the top region of divertor dome is ~4 times higher than the one of in-vessel tangential coils. More effective cooling of the coils located in the high heat flux regions (on dome) can be done by re-optimisation of the coil shape. The fact that electromagnetic screening effects on the divertor dome body could be smaller than for in-vessel coils which are located on the vacuum vessel allow us to use additional thermo-conductive elements (like copper plates) in the design of divertor equilibrium coils located on divertor dome.