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Magnetic Diagnostics in Alcator C-Mod

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ABSTRACT

The full package of magnetics diagnostics for Alcator C-Mod is now installed inside the vacuum vessel. This consists of four poloidal arrays of pickup coils for measuring B_p , three Rogowski coils, a set of full flux loops, two diamagnetic loops with compensation coils, a small 2-D array of mirnov coils to measure edge fluctuations, and some partial flux loops for detecting locked modes. All the coils have been absolutely calibrated to $\pm 0.5\%$ before installation, and an in-situ calibration of this system will also be carried out. Internal mounting enables fast time response despite the shielding effects of skin currents in the thick vessel wall. However, this complicates the design considerably due to constraints on physical size, vacuum compatibility, temperature extremes, protection from plasma damage, etc.

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INTRODUCTION

Alcator C-MOD differs substantially from its predecessors, Alcator A and C. Two of the most fundamental differences, at least as far as magnetics instrumentation is concerned, are: (a) the ability to achieve highly non-circular plasma shapes, and (b) the thick vacuum vessel wall^[1]. The Alcator C-MOD vacuum chamber is the primary load-bearing component for nearly all of the poloidal field coil system and therefore its walls must be relatively thick (15-25 mm), with no electrically insulating break. This effectively results in a highly conducting shell (of order $\sim 50 \mu\Omega$) surrounding the plasma, which presents difficulties in controlling and measuring plasma magnetics unless the B -field and flux sensors are located inside the vacuum vessel.

The capability to produce highly non-circular, diverted plasmas enormously expands the plasma operating parameter space which can be explored on high field, high density tokamaks. This added flexibility comes at the expense of far more complexity in the PF system and in the equilibrium control requirements. Therefore Alcator C-MOD has been equipped with a comprehensive set of magnetic field and flux sensors for both diagnostic as well as control purposes^[2].

GENERAL DESIGN CONSTRAINTS

Since the magnetics hardware is installed inside the vacuum chamber, it either has to be completely sealed in its own secondary vacuum system, or else all the materials and construction methods must be compatible with ultra-high vacuum (10^{-9} torr). This concern is primarily directed to the electrical wire insulation material. In order to reduce cost and complexity and avoid possible problems with vacuum leaks, we have chosen the second option. Using a specially built vacuum chamber, we measured the outgassing rates of various materials and found that, besides ceramics, teflon was the only commonly available wire insulation which was acceptable. (Ceramic-coated wire is available, but we found it to be mechanically and electrically unreliable, as well as prohibitively expensive.) Teflon is not very radiation resistant, but the total neutron and gamma fluence from lifetime operation of Alcator C-MOD is within its allowable dose limits when operated in

vacuum. The C-MOD vessel will be baked periodically at 150° C, but this is well under the maximum operating temperature of teflon (260° C).

The only surface to which the magnetics components can be attached is the actual stainless steel wall of the vacuum chamber, and the most reasonable method is by resistance welding, since all the instrumentation has to be retrofitted into the completed vessel. However, nearly half of the vessel wall area is covered by thermal protection tiles. Therefore 'channels' have been machined into the back of the tile hardware in order to accommodate the magnetics coils and wires (fig. 1). The tiles are only 2 cm thick, thus the channels cannot be much larger than 1 cm in diameter. This places a *severe* constraint on the physical size of the sensor components, especially the B_p and rogowski coils. The backs of the tiles are in close proximity to the magnetics components, and in areas of particularly high heat load, such as the inboard midplane and the divertors, they can reach a steady-state temperature of 500° C. (For reasons of complexity and reliability, there is no internal cooling system.) In order to protect the teflon wire insulation, several precautions have been implemented. First, allowance has been made for a small vacuum gap so that none of the magnetics components contact the tiles. This restricts the heat transfer to just infrared radiation. Secondly, stainless steel heat shield covers have been welded on top of all components and wires. These covers conduct heat to the chamber wall, which is actively cooled during plasma operation. Calculations and bench tests have shown that under worst case conditions, the temperature inside the heat shields should be no more than 125° above the vessel temperature. Unfortunately, the shield covers and surrounding vacuum gap further restrict the physical size of the magnetics sensors. In areas of the vessel wall which are not covered by tiles, the heat shields have been installed anyway to prevent exposure of the components to energetic plasma particles, UV, and soft x-rays, and also to mechanically protect the coils and wiring during in-vessel work.

DESCRIPTION OF COMPONENTS

Four poloidal arrays of 26 coils each have been installed to measure the local poloidal magnetic field parallel to the chamber wall (Fig. 2). Each of these B_p coils is wound on a ceramic spindle and has an active length of 3 cm and a diameter of 8 mm. The

teflon insulated wire is size AWG 30 and was used for all of the magnetics components. Tests of the wound coils showed no breakdown between layers at 3 KV. The wire leads are tightly twisted and are long enough (3–4 meters) to run all the way around the machine to the vacuum feedthrough ports, thereby eliminating trouble-prone connections inside the chamber. All of the B_p coils were absolutely calibrated to $\pm 0.5\%$ at frequencies up to 50 Khz using a Helmholtz coil pair prior to installation into the tokamak. The mean value for NA (# of turns \times area per turn) is 66.5 cm^2 ($V = -NA\dot{B}$).

To insure accurate positioning and alignment of the B_p coils, they are mounted on precisely machined stainless steel belts, and these belts are welded onto the vacuum chamber wall. The inboard side of the vessel was provided with precision alignment pins onto which the belts were closely fitted, so the placement error for coils in this region are $\leq 0.5 \text{ mm}$ in any direction and angular orientation error is $\leq 0.5^\circ$. The outboard side of the chamber has some appreciable weld distortion and so the positional errors of coils in this region are slightly larger. Ideally, the four poloidal arrays would be arranged at 90° intervals in the toroidal direction, however the Alcator C-MOD vacuum chamber has a 10-fold symmetry which precludes this spacing. Instead, the belts have been arranged in two diametrically opposite arrays separated by 72° (see fig. 3). An array of 16 similar, but smaller mirnov coils arranged in a 'T'-shape has also been placed near the inboard midplane at one toroidal location (port F in fig. 3) in order to measure modes with higher k -number (wavelengths down to 4 cm).

In between each B_p coil location there is a toroidal flux loop to measure the total vertical flux at 26 locations around the vessel cross-section (not visible on the scale of figure 2). The choice of 26 B_p and 26 flux measurements is based upon a theoretical study^[3] which shows that this number of sensors should be more than sufficient to permit accurate MHD reconstructions of any plasma equilibrium in C-MOD and still allow for substantial redundancy in case some sensors fail. The flux sensors are purposely arranged between B_p coils so that the intersection between toroidal and poloidal components does not present a problem. The flux loops are constructed of the previously described wire threaded into very small stainless steel tubing (0.8 mm O.D.), which is not a trivial task considering that some of the loops are nearly 7 meters long. Each flux wire makes one

revolution in the toroidal direction and then the leads are twisted and routed to the vacuum feedthroughs. These end joints are protected by the same stainless steel heat shield covers used throughout the magnetics assembly. The flux tubing is partially covered by lengths of thin stainless steel bands which are welded to the wall and hold the flux loops in place. The lengths of the flux loops (i.e. their circumference) is easily measured and therefore their radial location is known to ≤ 1 mm (better on the inboard side) and the error in loop area is $\leq 0.5\%$. There are some locations where full flux loops are not feasible because they would block port access. In these areas, partial flux loops (often called 'saddle' or 'picture-frame' loops) have been mounted on the outboard wall of the vessel between one pair of ports to measure the radial magnetic flux there. From these signals, the full poloidal flux can be calculated at these locations, although with considerably larger measurement errors. In addition, another set of partial flux loops consisting of two diametrically opposite pairs has been mounted on the outboard midplane. Each pair is wired to yield a signal equal to the difference in individual loops, in order to detect slowly growing non-axisymmetric (i.e. $n = 1$) 'locked' modes^[4].

The same construction is also used for the main diamagnetic loops, which are located at two toroidal locations (fig. 3), except that the loops make one poloidal circuit rather than toroidal. In order to measure the exceedingly small effect of the plasma on the toroidal magnetic flux, it is necessary to carefully null out the vacuum magnetic field, and in general this is quite difficult in a tokamak, primarily because of mechanical distortions, alignment errors, and eddy currents^[5]. These difficulties are reduced somewhat in Alcator C-MOD by placing the vacuum field compensation coils inside the conducting vessel (fig. 2). They consist of a dimensionally stable mandrel around which is wound two separate coils; the primary one has 48 turns and cancels out the main loop signal to within 2%, and is passively wired in series with it. The second coil has only 8 turns and is added to the composite signal using an op-amp circuit with adjustable gain. This multiple coil approach should reduce the effect of op-amp drift. It is recognized, however, that other errors, which are difficult to characterize until machine operation begins, will probably also have to be cancelled out.

Rogowski coils for measuring plasma current have been installed at three different toroidal locations. Two of them consist of upper and lower halves, while the third consists of inboard and outboard halves. They were fabricated in halves rather than complete rings because it was much easier to fit them in the machine, and also because the individual coil components might possibly be used to determine plasma in/out and up/down motion. Each segment is about 175 cm long and consists of teflon-insulated wire wound in a spiral groove around a teflon tube having a diameter of 6 mm. The winding pitch is ~ 8 turns/cm and is made by simply threading the tube with a die. The principal design difficulty which had to be solved is the enormous thermal expansion coefficient of teflon. At the vessel bakeout temperature of 150°C , an unconstrained coil would expand many centimeters. In order to prevent this, a thick copper wire (1.6 mm diam) is threaded and screwed down the center of the entire length of the teflon tube. The friction between the central copper rod and the surrounding teflon tube prevents any measurable expansion or contraction of the rogowski (beyond that of the copper). The central rod also acts as the cancellation return wire. All the rogowski coils were calibrated prior to installation and the resulting sensitivity of 2.53×10^{-8} volts/(dI/dt) is very close to the calculated value. In addition, the sensitivity of the rogowskis to external currents was also measured using the Helmholtz setup and found to be $<0.1\%$ of the internal value. Pickup of toroidal magnetic field was also found to be negligible.

All of the magnetics signals are proportional to time derivatives of the desired physical quantities such as magnetic fields and currents. Therefore all of the signals will be integrated using differential input op-amp integrators which have a sample-and-hold circuit to set and reset the baseline. The analog integration circuits have been tested and found to have ≤ 5 mv drift in 10 seconds and $\leq 1\%$ absolute error overall for the 3 second duration of a C-MOD plasma discharge. The integrated signals will be digitized by CAMAC and used to determine the plasma equilibrium and to study MHD instabilities. In addition, approximately 60 signals will also be used by the control system to provide realtime feedback control of the plasma shape and position.

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FIGURE CAPTIONS

Figure 1—The magnetics sensors are welded to the vessel wall within channels machined in back of the tile hardware. The available space is only ~ 10 mm in diameter.

Figure 2—diagram of the Alcator C-MOD vacuum vessel cross-section showing the location of B_p coils and diamagnetic compensation coils. Flux loops run toroidally between each of the B_p coils but are too small to see on this scale. For clarity, the thermal protection tiles which cover the inboard half of the vessel wall (left side of figure) have not been shown. All of the magnetics components and wires are protected by stainless steel shields, and these have also not been shown.

Figure 3—a top view of the vacuum chamber showing the toroidal locations of B_p coil arrays, rogowski coils, diamagnetic loops, etc. Note that the magnetics hardware is located midway between ports so as to not block any diagnostic access.

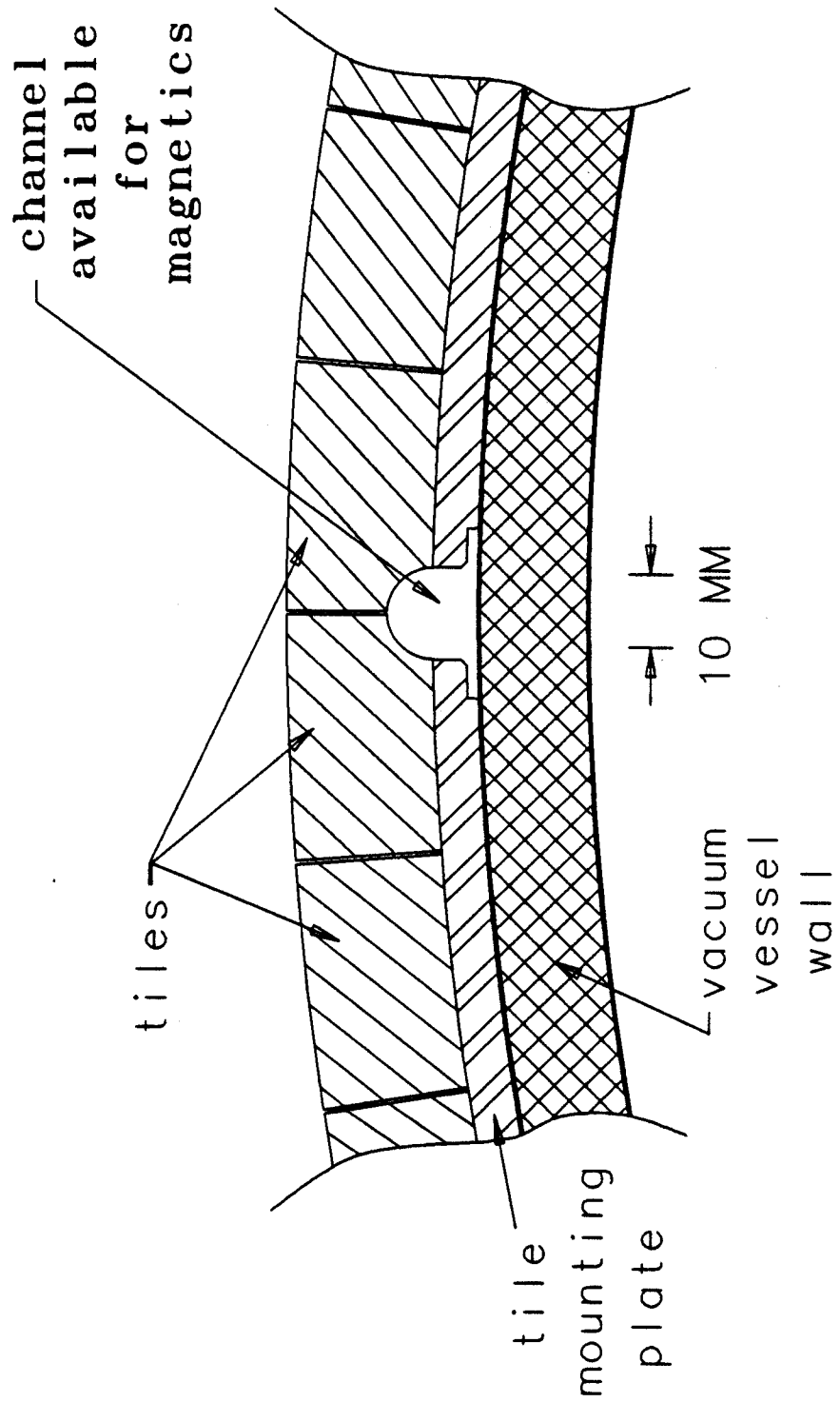


Figure 1

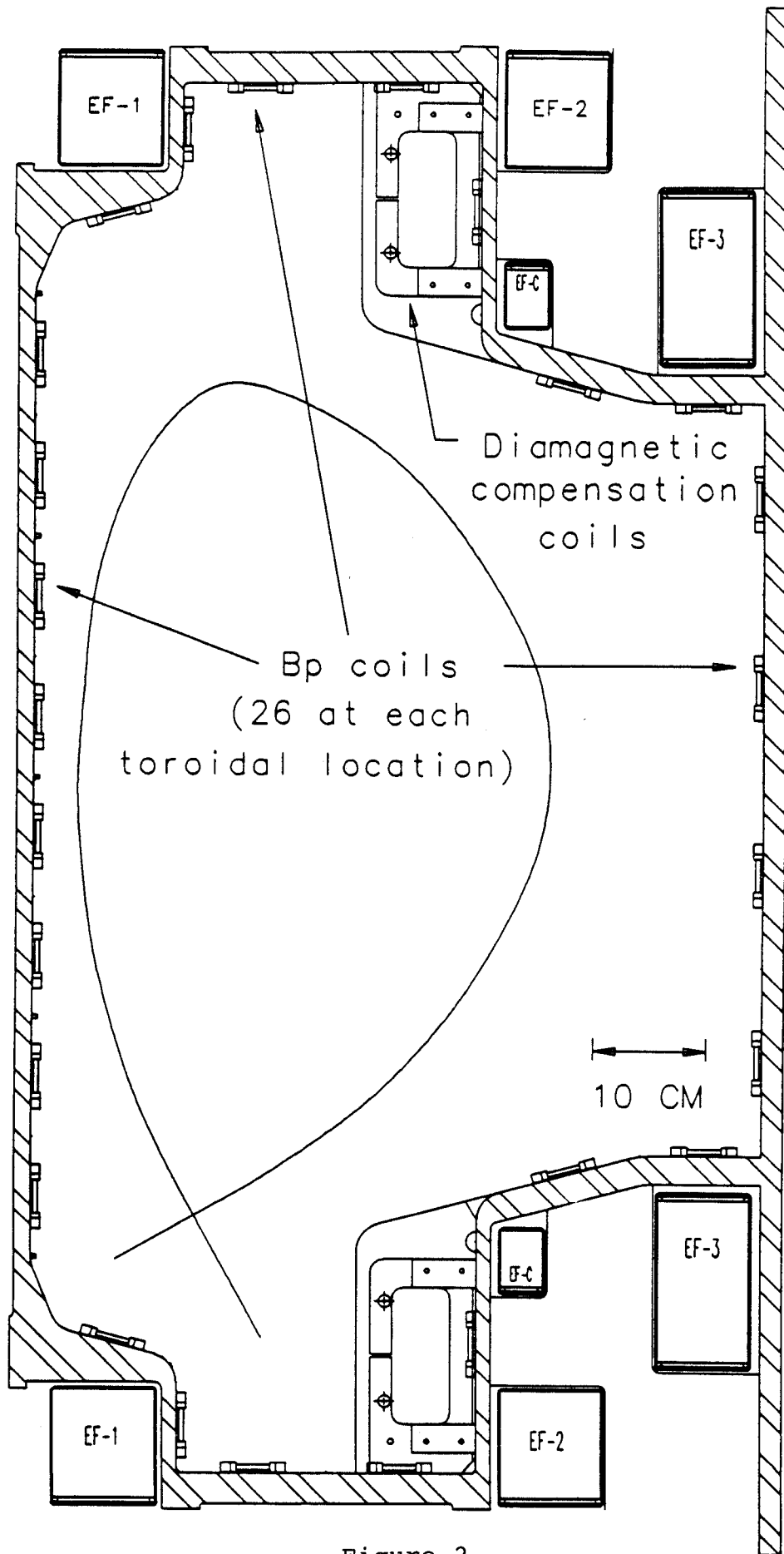


Figure 2

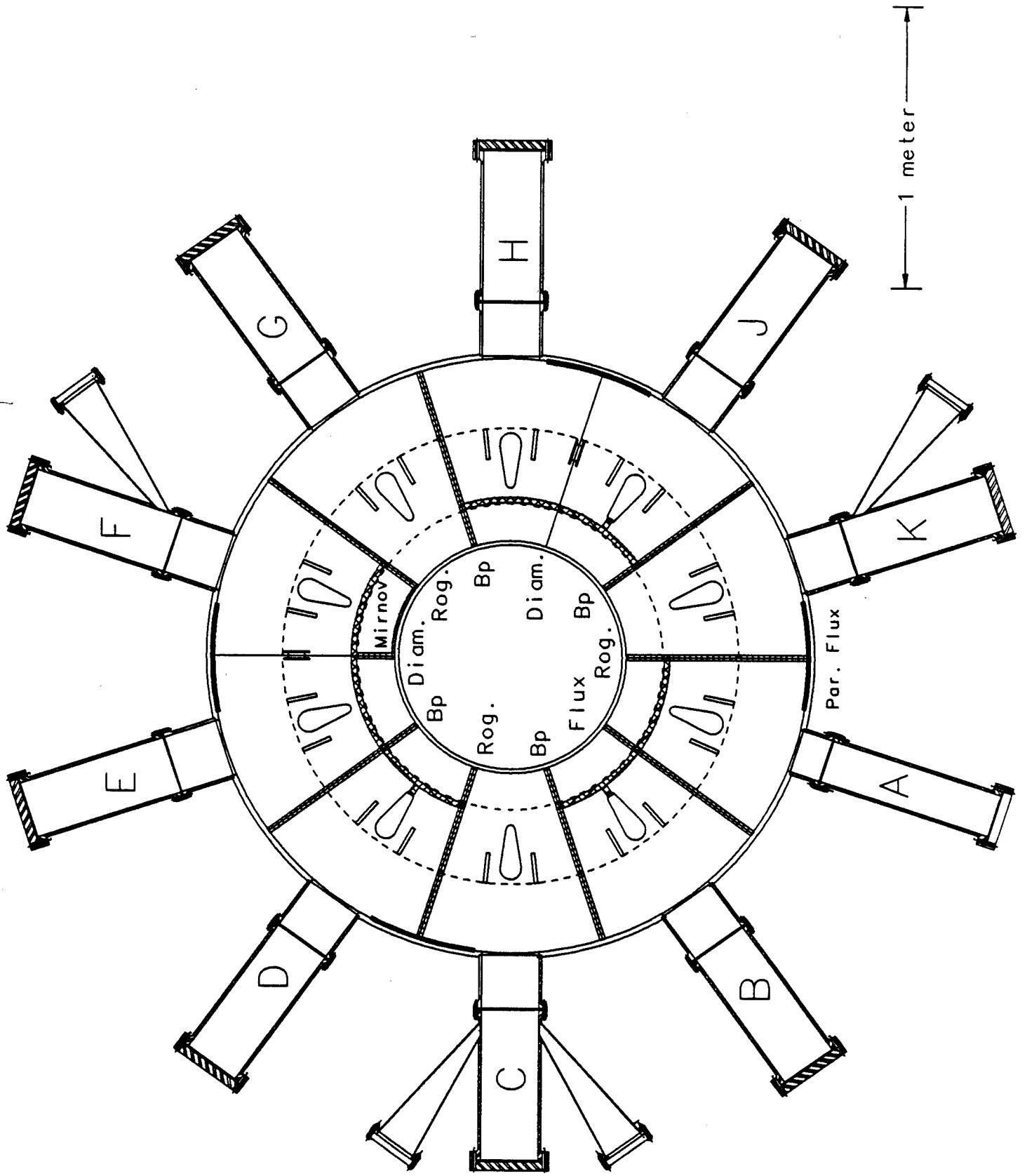


Figure 3