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The Engineering Design of the Tokamak Physics Experiment (TPX)

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The Tokamak Physics Experiment (TPX) is designed to develop the scientific basis for a compact and continuously operating tokamak fusion reactor. TPX has a long pulse (1000s) capability, can accommodate high divertor heat loads, has a flexible poloidal field (PF) system, and auxiliary heating and current drive systems that make it an ideal test bed for development of attractive reactor concepts. The design incorporates superconducting magnets in both the toroidal field (TF) and poloidal field (PF) systems. Long pulse deuterium operation will produce 6×10^{21} neutrons per year requiring remote maintenance of the in-vessel hardware. This paper provides an overview of the TPX design with the emphasis on developments in the tokamak design since the Conceptual Design Review (CDR) in March, 1993.

1. OVERVIEW OF DEVELOPMENTS IN TOKAMAK DESIGN

Analyses performed since the CDR indicated that improvements in the TF coil design were necessary to meet specified design criteria at full field (4T) with full nuclear heating. Improvements were also required in the PF coil design to meet an extended range of flexibility in β_N-l_i space. These results prompted significant changes in the tokamak configuration.

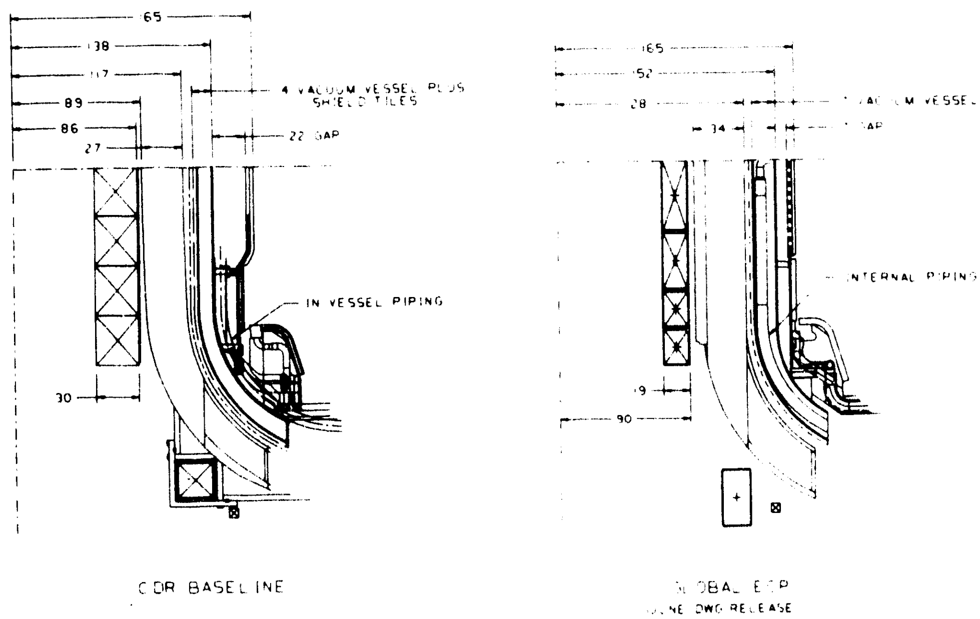
Analyses also indicated that additional conducting structure would improve passive stabilization of the external kink mode and allow β_N values greater than 3 to be achieved. This was clearly important for the TPX experimental objectives of achieving β_N of 4 to 5, so the design of the in-vessel passive stabilizer was revised.

A number of other design changes have been incorporated, most notably in the divertors. The inboard divertor target plate was repositioned in order to more effectively utilize space in this region. Specifications on the gap sizes between the target plates and the baffle plate were modified to optimize pumping performance. Design heat loads were reduced, consistent with dispersive divertor operation.

2. CONFIGURATION CHANGES FOR IMPROVED MAGNET PERFORMANCE

The CDR design featured 22cm between the back of the inboard limiter/passive stabilizers and the vacuum vessel. Repositioning the inboard divertor target plate opened up the possibility of significantly reducing this 22cm space. By moving all of coolant connections to the inboard limiter/passive stabilizers to the plasma facing side and routing the plumbing inside the double-wall vacuum vessel, the 22cm space could be reduced to 7cm. This allowed an extra 15cm to be allocated to the magnet and vacuum vessel envelopes in order to solve the magnet problems under more benign conditions (lower peak fields in the TF and CS magnets with reduced nuclear heating) with minimum cost impact. The revised inboard radial build is illustrated in Figure 1.

Figure 1
Revised Inboard Radial Build



2.1 Vacuum Vessel and Shield Design

The CDR design featured a double-wall vacuum vessel with leaded glass shield tiles (doped with boron carbide) mechanically attached to the outside surface. The space between the two walls of the vacuum vessel was filled with 150°C water during normal operation which also provided shielding.

Analyses indicated that the leaded glass tiles could be eliminated without compromising the shield performance by increasing the vacuum vessel envelope and borating the shield water. Eliminating the tiles has several desirable effects: it eliminates the need for lead, with its attendant environmental concerns and ultimately its disposal as a mixed waste; it eliminates the need for R&D associated with fabricating the tiles; it eliminates the time-consuming machine assembly task of installing the tiles on the compound curved vessel surfaces; and it eliminates a potentially worrisome failure mode (a cracked tile creating a thermal short between the 5K cold mass and the 423K [150°C] vacuum vessel or degrading the shielding effectiveness).

The new design eliminates the shield tiles and features a double-wall vacuum vessel filled with borated water. The shield water is borated with 110 grams per liter of boric acid. The primary issue related to borating the water is corrosion. The compatibility of the borated water with titanium at 150°C must be tested and the cooling loop materials must be screened to ensure no corrosion problems. MHD effects must be considered since the borated water is an electrolyte and will generate voltages that may enhance the corrosion rate.

2.2 Plasma Facing Components

The pipes to feed and return coolant to and from the inboard limiter and passive stabilizers are routed between the two walls of the vacuum vessel. Connections to these pipes must be made from the plasma facing surface on the inboard limiters and passive stabilizers.

The new design incorporates the inboard limiter module and upper and lower passive stabilizer modules into a single module, as shown in Figure 2. The design features a toroidal array of sixteen identical modules. Modules are joined to adjacent modules along the top and bottom to form conducting rings. A resistive break is provided to facilitate plasma initiation.

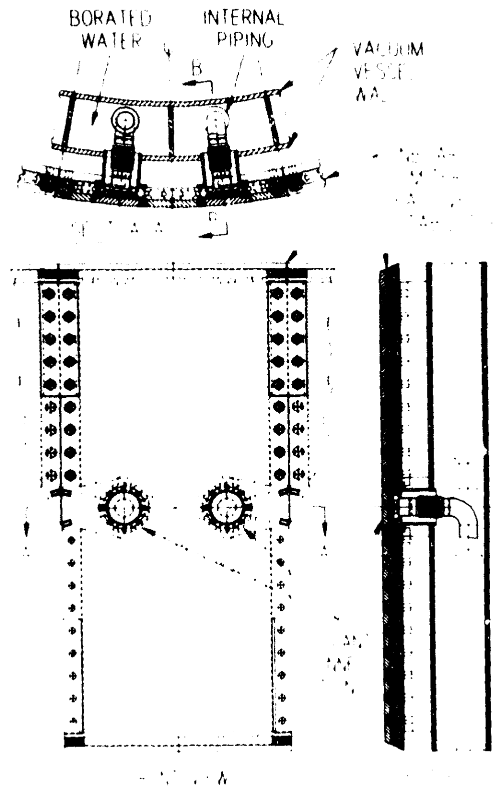
2.3 TF Coil Design

The inner leg of the TF coil has been moved 14cm further outboard resulting in a peak TF field which is lower by 0.5T, i.e., 8.4T in the new design versus 8.9T in the CDR design. In addition to lowering the peak field, the number of strands in the superconductor was changed form 405 to 486, thus improving its performance. In the new design, the TF can satisfy all design criteria at 4T if the peak temperature at the inboard leg is kept less than 6.0K. With a helium inlet temperature of 5.0K and inlet/outlet pressures of 5atm/3atm, the calculated peak bore temperature is 6.0K with a total flow rate through each winding pack of 28g/s. The nominal case cross-section has increased to 34cm x 41cm (from 25cm x 35cm). The outer leg of the TF has been moved back 5cm to provide space for the expanded TF case.

The new TF conductor design is based on a 486-strand conductor, with a 2.5:1 copper:non-copper ratio, whereas the CDR design was based on a 405 strand conductor with a copper:non-copper ratio of 3.5:1.

The TF coil set is electrically interconnected with two interleaves. This allows the terminal-to-terminal discharge voltage to be reduced from the CDR value of

Figure 2
Inboard Limiter/Passive Stabilizer Module

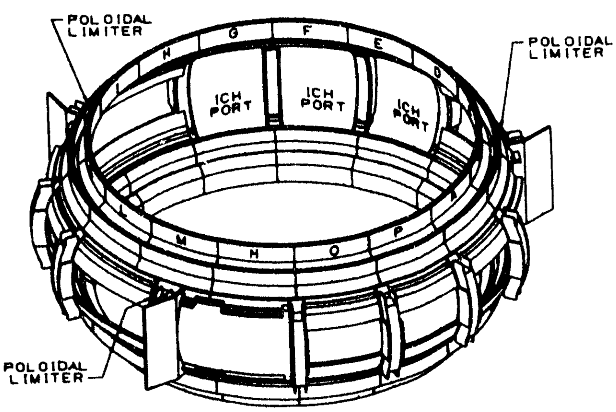


15kV to 7.5kV, which greatly reduces the electrical stress on the TF system.

2.4 PF Coil Design

In the PF system, the outer diameter of the central solenoid coils increased by 4cm. The larger cross-sectional area allows the flux swing requirement to be met at lower field which tends to decrease the cost of the system. A series of PF optimization studies were performed. These studies led to a final set of cost-optimized PF coils that meets all of the MHD equilibrium flexibility points, satisfies physics and engineering constraints on plasma initiation, and is capable of an

Figure 3
Passive Stabilizer Configured for External Kink Mode Stabilization



inductively driving the plasma to full current.

The new PF design resulted from a series of optimization studies performed over the past year. The new design features a larger outer diameter for the CS coils which allowed the CS coils to become thinner in the radial dimension. In addition, the CS coils were not constrained to be of equal height. The optimization also resulted in ring coils which are taller and narrower than in the CDR. A taller, narrower PF5 can be seen in Figure 1. Taller, narrower coils tend to have lower peak fields for a given number of ampere-turns than coils of squatter proportions. Overall, the size of the PF system did not change significantly. The proposed PF set is more capable than the CDR set in that it satisfies a more stringent set of engineering design criteria while providing more flexibility in β_N-l_i space.

3. KINK MODE STABILIZATION

An analysis method was developed to estimate the stability limits for 3D structures. Using this method, the performance of the CDR passive stabilizer design was found to provide passive stabilization of the external kink mode up to a β_N value of only 3. This is clearly below the TPX experimental objective of achieving β_N of 4 to 5.

The in-vessel passive stabilizer, originally provided to satisfy vertical position control requirements, was expanded by adding conducting elements. The additional elements include vertical conductors connecting the upper and lower toroidal conductors, and toroidal conductors that provide a wider toroidal current path over parts of the circumference. The new passive stabilizer design is shown in Figure 3. Analysis of the new design indicates that it provides passive stabilization of the external kink mode up to β_N values greater than 6 in the baseline configuration.

4. SUMMARY

Solutions have been developed to address the issues which arose during the conceptual design phase of TPX. These solutions provide design envelopes which we believe will be robust to remaining uncertainties in design and analysis.

5. ACKNOWLEDGMENTS

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