# ENGINEERING FEATURES OF THE INTOR CONCEPTUAL DESIGN*

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## Summary

The INTOR engineering design has been strongly influenced by considerations for assembly and maintenance. A maintenance philosophy was established at the outset of the conceptual design to insure that the tokamak configuration would be developed to accommodate maintenance requirements. The main features of the INTOR design are summarized in this paper with primary emphasis on the impact of maintenance considerations.

The most apparent configuration design feature is the access provided for torus maintenance. Particular attention was given to the size and location of superconducting magnets and the location of vacuum boundaries. All of the poloidal field coils are placed outside of the bore of the toroidal field coil and located above and below an access opening between adjacent toroidal field coils through which torus sectors are removed. A magnet structural configuration consisting of mechanically attached reinforcing members has been designed which facilitates the open access space for torus sector removal.

For impurity control, a single null poloidal divertor was selected over a double null design in order to maintain sufficient access for pumping and maintenance of the collector. A double null divertor was found to severely limit access to the torus with the addition of divertor collectors and pumping at the top. For this reason, a single null concept was selected in spite of the more difficult problems associated with the required asymmetric poloidal field system and higher particle loadings.

Tokamak support systems and the reactor building and facilities are also important to the overall design evolution and were included in the conceptual design effort. However, this paper discusses only the primary tokamak systems.

## Introduction

The INTOR engineering design which evolved during the conceptual design phase represents a combined team effort by all four participating groups: Euratom, Japan, USSR, and the USA. Therefore, the design will be described with no attempt to identify the specific contribution made by each individual or group. Each participating group developed a national design[1,2,3,4] which was used as the basis for the detailed studies which subsequently led to the selection of the international design concept. The conceptual design report contains a complete description of the project including a discussion of major options considered, the rationale for selection, and supporting analyses.[5]

The key features of the design are illustrated in the perspective drawing shown in Figure 1. The principal engineering parameters are given in Table 1.

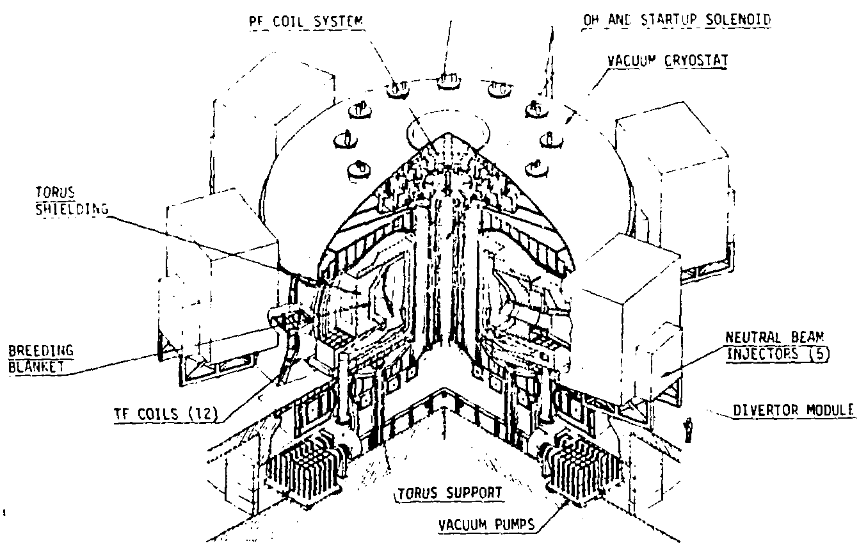


Figure 1. Perspective view of INTOR.



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Table 1. Principal INTOR Engineering Parameters

| | |
|---|---|
| Major radius | 5.2 m |
| Plasma radius | 1.2 m |
| Plasma elongation | 1.6 |
| Fusion power | 620 MW |
| Neutron wall loading | 1.3 MW/m$^2$ |
| Neutral beam heating power | 75 MW |
| Burn time | 200 s |
| Number of TF coils | 12 |
| TF coil bore size | 7.7 × 10.7 m |
| Field on axis | 5.5 T |
| TF coil maximum field | 11 T |
| Tritium breeding ratio | 0.65 |
| Stationary power supply | 241 MW |
| Pulsed energy storage | 22.5 GJ |
| PF system total flux | 110 V-s |
| Availability goal | 50% |

Maintenance considerations were established at the outset of the INTOR Design Study as fundamental to the development of the design configuration. The complex electromagnetic features of the tokamak device when coupled to the power reactor impact of component activation in the presence of tritium could lead to excessive downtime for machine repair. For this reason, a maintenance philosophy was established for the conceptual design to allow maintenance requirements to influence the design configuration. The maintenance philosophy is summarized as follows:

- The tokamak will be designed from the outset to be maintained and repaired by the use of existing or near-term technology for remote maintenance equipment such as manipulators, viewing systems, and transfer mechanisms.

- Certain systems must be designed and developed with very high reliability so that failure will not be expected within the lifetime of the device. Failure of these systems would require a major shutdown of the facility (six months to one year) for repair or replacement. Superconducting toroidal magnetic field (TF) and poloidal magnetic field (PF) coils, the inboard portion of the torus shield, and several major support structures have been identified as systems of this type and designated as semi-permanent installations.

- Sufficient radiation shielding will be provided in the torus and around penetrations to limit the shutdown dose level of components exposed to the reactor room. "Hands-on" maintenance will be considered for normal operations when the torus internals are not removed. The maximum dose rate anywhere in the room after twenty-four hours of shutdown is specified as 2.5 mrem/h.

- All systems will be designed for fully remote maintenance to cover cases of emergency.

Implementation of this philosophy has led to a modularized design concept, and designing to achieve the required access has had a significant impact on the design of the tokamak systems.

## Design Description

The main features of the INTOR engineering design are summarized in the following sections.

### Toroidal Field Coil Design

The most significant configuration driver is the access requirement for torus maintenance. The 12 TF coils have been sized with sufficient bore dimensions so that a complete torus sector, consisting of 1/12 of the total, can be withdrawn by a simple straight motion between the outer legs of the coils. These 12 torus sectors fit within a semi-permanent upper, inner and lower shield frame. For this design approach, the optimum number of TF coils is 12 since the coil size for the 0.75% ripple limit coincides with the coil size required for torus sector removal (see Figure 2).

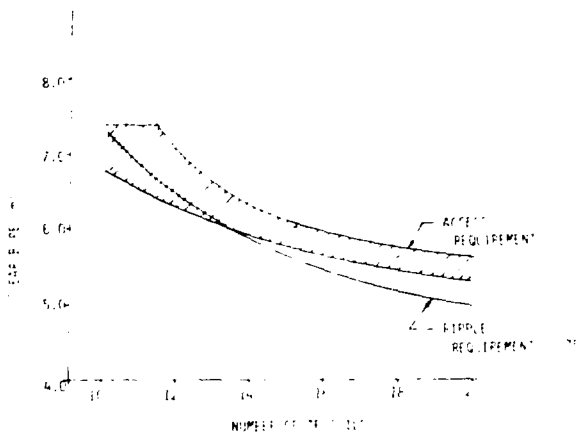

Figure 2. Size vs number of TF coils.

The TF coil configuration has been developed with sufficient flexibility to incorporate any one of the three major conductor concepts presently under development worldwide for operation up to 12 tesla maximum field. The three conductor concepts are: $Nb_3Sn$ conductor, liquid helium bath cooled at 4.2 K; NbTi conductor, superfluid liquid helium bath cooled at 1.8 K; and $Nb_3Sn$ conductor forced flow liquid helium cooled at 4.2 K.

### Poloidal Field Coil System

All of the PF coils have been placed outside of the bore of the TF coils. The PF coils can therefore all be superconducting since mechanical joints are not required for assembly. All of the PF coils have been located above and below the access opening between adjacent TF coils through which the torus sectors are removed. A small solenoidal, cryoresistive coil is placed within the ohmic heating solenoid to provide the breakdown voltage for plasma initiation.

### Vacuum Topology and Torus

Since all the PF coils external to the TF coil bore are superconducting, it was possible to design a single vacuum cryostat to contain all of the coils. The vessel includes individual enclosures for the outer TF coil legs as part of the common cryostat. With this feature, access to the torus is maintained without penetration of the cryogenic vacuum boundary. Another important feature of this design is that there is a complete separation of the cold and warm components, which eases the structural design requirements for thermal movements of the large structures.

The torus system, consisting of a first wall, blanket, shield and divertor collector, has been configured in two major parts; a semi-permanent shield and removable sectors (see Figure 3). The components exposed

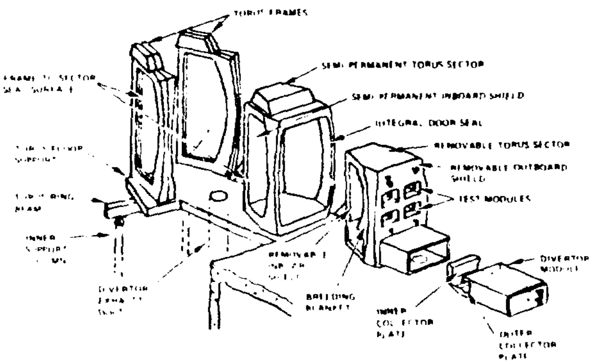


Figure 3. Torus configuration.

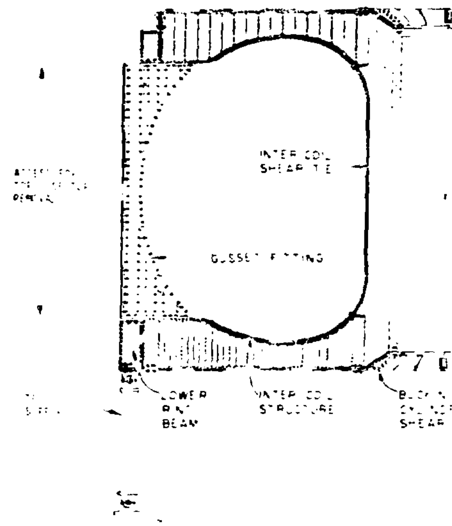

Figure 4. TF coil support structure - elevation view.

to the most severe damage from particle and heat loads (first wall and blanket regions) have been combined into a sector which can be removed separately from the torus shielding. More importantly, the vacuum seal for this sector is entirely on the outside of the torus. The seal weld is on a rectangular flange easily accessible between the TF coil outer legs. The other portion of the torus, consisting of the structural frames and semi-permanent shield modules, forms the primary vacuum boundary and is not removed for normal planned maintenance procedures.

The shielding thickness satisfies the goal for hands-on maintenance to the device externals after a 24 hour shut down period.

Single Null Poloidal Divertor

The divertor collector operates in the most severe environment of any torus component, and its design must include provisions for frequent repair. Modular divertor sectors have been designed which can be removed in a manner similar to that used for the main torus sectors. A double null divertor was found to severely limit access to the top of the torus. For this reason, a single null concept has been selected despite the more difficult design problems associated with the asymmetric PF system and higher particle flux per collector.

In the single null divertor system, the divertor chamber is located at the bottom of the toroidal plasma chamber. This results in a shift of the plasma upwards by 0.6 m relative to the TF coil horizontal centerline, in order to center the assembly of toroidal chamber and divertor region inside the TF coil system and to facilitate its maintainability. The configuration of the divertor region is indicated in Figure 3. It consists of 12 divertor modules, each located inside a removable torus sector. Each divertor module is provided with an exhaust duct from which it can be easily disconnected without dismantling the pumping system. The collector plates and the other divertor surfaces subject to high erosion are cooled by water flowing inside parallel cooling channels set one beside the other; they are protected by tiles of refractory metals attached to the surface. Each module, including its shielding plug, is extended toward the outer boundary of the torus system, ... can be easily disconnected from the outside.

Structural Support System

The large TF coils combined with the all-external PF coils impose very large pulsed magnetic forces on the TF system. Thermal considerations require that all the structural support be provided at cryogenic temperatures. A structural configuration consisting of mechanically attached reinforcing members has been designed which maintains the access space for torus sector removal (Figure 4).

The structural design was verified by a three-dimensional finite-element analysis. The calculational model included a representation of all 12 TF coils and their support structure. Local stress analyses were performed to investigate details of the design, including the local plate bending. The allowable design stress was based on conventional limits for steady state loads. Crack growth and fracture mechanics considerations were included for pulsed loads. An allowable cyclic stress of 200 MPa was adopted. However, due to the lack of fatigue data at 4°K, this is an area identified for research and development needs.

Another feature of the structural configuration is the gravity support system. The support has been placed entirely at the outside of the machine to provide access to the bottom of the machine (see Figure 4).

Tokamak Radial Build

The major radial dimensions of the INTOR are summarized in Figure 5. The radial build dimensions show space allocation for all components as well as the required gaps for assembly tolerance.

Dedicated Torus Sectors

The facilities layout is based upon the concept of a dedicated bay—the region between adjacent TF coils (Figure 6). Three bays are dedicated to testing—two for blanket and materials testing and one for plasma engineering hardware and diagnostics testing. Five bays are dedicated to the four active and one redundant neutral beam injectors; two bays are dedicated to instrumentation, diagnostics and control; and two bays for pellet fueling. This approach provides a straightforward interface between the tokamak device and the facility.

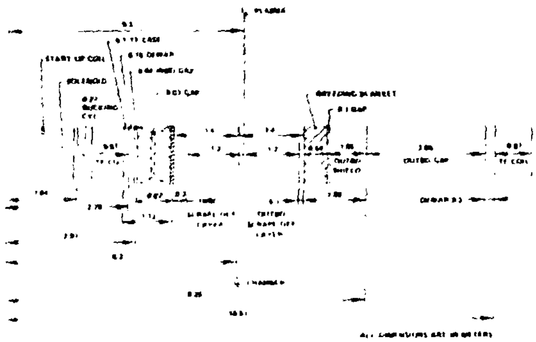

Figure 5. Radial build.

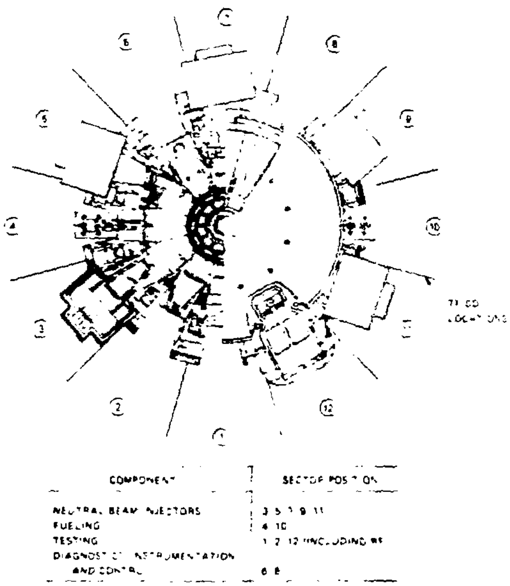


Figure 6. Dedicated sectors.

## Conclusions

The INTOR conceptual design effort has resulted in a concept which appears to be feasible from engineering considerations of design, fabrication and operations. The most significant accomplishments of the engineering design are summarized as follows:

- The design configuration provides significant access for maintenance of the tokamak systems. Torus and component shielding permits hands-on maintenance operations in the reactor building.
- A PF coil system has been designed which is totally external to the bore of the TF coils permitting the use of superconducting coils.
- The torus assembly concept provides a plasma chamber vacuum topology with an easily accessible seal and closure at the outboard region of the torus between adjacent TF coils.
- The structural design concept maintains open access and accommodates the high out-of-plane cyclic loading.
- The cryogenic vacuum topology provides a single containment for all superconducting coils and completely separates the cold and warm components.
- A modular design approach provides improved maintenance capability for those components with more frequent failures such as the first wall and blanket, divertor collectors, and neutral beam injectors.

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