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DESIGN OF A FUSION ENGINEERING TEST FACILITY*

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ABSTRACT

The Fusion Engineering Test Facility (ETF) is being designed to provide for engineering testing capability in a program leading to the demonstration of fusion as a viable energy option. It will combine power-reactor-type components and subsystems into an integrated tokamak system and provide a test bed to test blanket modules in a fusion environment.

Because of the uncertainties in impurity control two basic designs are being developed: a design with a bundle divertor (Design 1) and one with a poloidal divertor (Design 2). The two designs are similar where possible, the latter having somewhat larger toroidal field (TF) coils to accommodate removal of the larger torus sectors required for the single-null poloidal divertor.

Both designs have a major radius of 5.4 m, a minor radius of 1.3 m, and a D-shaped plasma with an elongation of 1.6. Ten TF coils are incorporated in both designs, producing a toroidal field of 5.5 T on-axis.

The ohmic heating and equilibrium field (EF) coils supply sufficient volt-seconds to produce a flat-top burn of 100 s and a duty cycle of 135 s, including a start of 12 s, a burn termination of 10 s, and a pumpdown of 13 s. The total fusion power during burn is 750 MW, giving a neutron wall loading of 1.5 MW/m^2.

In Design 1 all of the poloidal field (PF) coils except the fast-response EF coils are located outside the TF coils and are superconducting. The fast-response coils are located inside the TF coil bore near the torus and are normal conducting so that they can be easily replaced. In Design 2 all of the PF coils are located outside the TF coils and are superconducting.

Ignition is achieved with 60 MW of neutral beam injection at 150 keV. Five megawatts of radio frequency heating (electron cyclotron resonance heating) is used to assist in the startup and limit the breakdown requirement to 25 V.

MASTER

1. INTRODUCTION

In September 1978 the U.S. Department of Energy (DOE) issued a policy statement for fusion energy[1], outlining a strategy to develop fusion energy as an economically attractive and environmentally acceptable energy option. This strategy involves three phases: scientific feasibility, engineering testing, and reactor demonstration.

It is anticipated that scientific feasibility will be demonstrated with the Tokamak Fusion Test Reactor (TFTR) in the mid-1980's. The Engineering Test Facility (ETF) is being designed for the engineering testing phase. It will combine power-reactor-type components and subsystems into an integrated tokamak system and provide a test bed to test blanket modules in a fusion environment. Reactor demonstration will be accomplished with an Engineering Prototype Reactor (EPR) and/or a Commercial Demonstration Reactor (DEMO).

The Office of Fusion Energy (OFE) within DOE has established an ETF Design Center at Oak Ridge National Laboratory (ORNL) to prepare the design of a tokamak ETF. The ETF Design

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Center is unique in that it combines the efforts of the four major tokamak laboratories (General Atomic, Massachusetts Institute of Technology, Oak Ridge National Laboratory, and Princeton Plasma Physics Laboratory) and several major industrial suppliers (General Electric, Grumman Aerospace, McDonnell Douglas, and Westinghouse) in an integrated team arrangement[2]. In addition, on-site support is provided by other laboratories (Hanford Engineering Development Laboratory and Idaho National Engineering Laboratory) and the A&E contractors Bechtel and Burns and Roe.

In February 1979 a workshop was convened in Knoxville, Tennessee, to provide input for a mission definition for the ETF[3,4]. Subsequent to this, the basic objectives of the ETF were established by OFE.

The primary purpose of the ETF is to serve as the means for developing fusion engineering technologies and thereby to demonstrate the practicality of fusion energy. To achieve this purpose the facility will be designed to focus fusion technology development on practical issues. It will generate sufficient thermonuclear energy and provide sufficiently flexible testing capabilities to allow meaningful testing of systems and components of relevance to practical fusion power reactors.

The primary requirements that must be met to achieve these objectives include the following:

(1) Demonstrate the successful operation of superconducting magnets of sufficient strength and size to be representative of a commercial power reactor.
(2) Incorporate the means for testing different blanket, first-wall, and shield modules. Ease of changeout of test items is a design requirement. Provide for the testing of reactor-relevant blankets for breeding tritium fuel. The capability for testing reactor-relevant electricity and synfuel production modules should also be incorporated.
(3) Demonstrate the capability of heat removal systems operating at power-producing temperatures.
(4) Demonstrate the use of systems and techniques for radioactive maintenance.
(5) Establish and apply the means to ensure public and operator safety in the conduct of machine operation.

Potential additional items will be added only to the degree justified by cost analysis of each added functional objective. Such items include

(1) plasma physics experimentation,
(2) materials testing,
(3) the small-scale generation of electricity.

Although an official schedule has not been adopted for the ETF, the Design Center is proceeding with the design based on the assumption that authorization to initiate the detail design and construction on October 1, 1983, will be received. On that basis the machine would become operational at the end of 1990.

2. OVERALL DESIGN

A major concern in the design of the ETF is the impurity control system. Two basic divertor impurity control systems, which appear to have some potential for power reactor applications, have been proposed: the bundle divertor and the poloidal divertor. Although the bundle divertor is more attractive with respect to physical installation, it presents substantial problems and uncertainties with respect to impact on plasma operating conditions. The poloidal divertor, on the other hand, is more attractive with respect to plasma conditions but is very unattractive in terms of physical installation. Thus, it is not at all apparent which design approach is the more favorable overall. Accordingly, two ETF machine designs are being developed: Design 1 with a bundle divertor and Design 2 with a poloidal divertor[5].

The basic design parameters for the two machines are listed in table I. When possible, the parameters have been kept the same. In some areas, such as the bore size of the toroidal field (TF) coils, however, it has been necessary to apply different values to account for the impact of using the different impurity control systems.

2.1 Design 1

The general arrangement of Design 1 is shown in figs. 1 and 2. The plasma chamber is assembled by the insertion of ten 36° shield sectors into a spool support structure. The face edges of the sectors are sealed with the spool support structure, forming a vacuum-tight enclosure for the plasma chamber.

Access and ripple considerations led to the selection of a ten-coil arrangement for the TF coil system. The TF coils, which have a bore 7.5 m by 10.8 m, require a field of 11.4 T at the coil in order to produce 5.5 T on-axis. It has not yet been decided whether these coils will be Nb_3Sn or superfluid-cooled NbTi.

The poloidal field (PF) coil system is installed mainly in the poloidal bore and outside the TF coils, but a limited number of low-current, fast-response coils are located in the toroidal bore of the TF coil assembly. Those in the poloidal bore and outside the TF coils are superconducting NbTi whereas those inside the toroidal bore are normal copper and are segmented so that they can be replaced if necessary.

The TF coils, superconducting PF coils, and bucking cylinder for the TF coils are all enclosed in a common dewar. The dewar is a dome structure that envelopes the top and bottom sections of the TF coils and the inboard region of the TF coil toroidal bore with a surface of revolution. The outboard legs of the TF coils are enclosed with individual extensions of the common dewar, providing ten bays for access to the torus.

The plasma is heated by an installation of four neutral beam lines, each with six ion sources. With three beam lines or a minimum of 17 of the 24 sources operating, the nominal injected power of 60 MW at 150 keV is achieved.

The plasma chamber is evacuated by an installation of four pairs of compound cryosorption vacuum pumps tied into the neutral beam injector (NBI) ducts at the NBI shutter shield. These pumps employ cryosorption panels for pumping helium as well as cryocondensation panels for pumping hydrogen isotopes. They are sized to reduce the plasma chamber pressure from 3×10^{-6} to 3×10^{-5} torr in 13 s.

A radio frequency (rf) system is provided for startup assist. This system, which supplies a total of 5 MW of power at 140 GHz, is used for ionization, plasma initiation, and supplemental heating of the plasma during the early phases of startup.

The bundle divertor is installed in one of the bays in such a manner that it can be removed by radial extraction. This divertor has two sets of coils: the primary divertor coils and a set of expansion coils. Being normal-conducting copper coils, they require about 100 MW of power. The diverted plasma impinges on a water-cooled target with tungsten tiles. Pumping of the divertor target is accomplished with two sets of three vacuum pumps, one set operational while the second set is being regenerated.

A combination pellet injection and gas puffing system is used for fueling. Three injectors are installed, one for tritium-rich pellets, one for deuterium-rich pellets, and one for pellets with a 50-50 mixture of tritium and deuterium.

The neutral beam injectors occupy four of the bays between adjacent TF coils, the bundle divertor one bay, and the fueling injectors half of a bay. The remaining 4 1/2 bays are available for diagnostics, instrumentation, and test modules.

2.2 Design 2

Design 2 is similar to Design 1 in many respects (figs. 3 and 4), but there are some notable differences that are congruent with the installation of a poloidal divertor. In order to minimize the space required to accommodate the torus, a single null divertor was adopted. With this approach the plasma axis had to be displaced upward 0.5 m to permit installation of the divertor collector at the bottom of the torus. The net effect was an increase in the overall torus height of about 1 m. In addition, in order to facilitate the replacement of the divertor collector modules, the truncations at the top and bottom of the outboard corners of the torus were eliminated. These changes in configuration necessitated enlarging the TF coils so that the horizontal bore is about 1 m larger than that for Design 1.

Because the poloidal divertor has a toroidally continuous divertor collector that must be replaceable, it is divided into ten modules that together with their pumping ducts can be independently removed from the torus. The pumping ducts each extend to a pair of cryosorption pumps that can be valved off to permit regeneration of one while the other is on-line. These pumps, which are sized to handle the divertor gas load, are more than adequate to pump down the plasma between burns.

Because of the larger size of the TF coils, the current ratings and physical sizes (cross section as well as length) of the PF coils must be considerably larger. In addition, large superconducting coils are required for creating the separatrix to direct plasma into the divertor collector.

With Design 2 the space in the lower part of the plasma chamber is occupied by the divertor collectors, severely limiting the space available for diagnostics installed in a vertical orientation and also reducing the space available for test modules. On the other hand, there is an additional bay available for instrumentation, diagnostics, and test modules. Therefore, the total area available to accommodate these requirements is approximately the same.

3. MAJOR DESIGN ISSUES

A number of design considerations have dominated the development of the ETF designs. Many of these issues are interrelated and cannot be resolved independently. Therefore, the design process is inherently an iterative one in which certain design decisions must be made on an interim basis assuming certain features that may be subject to change as the design develops. This implies that the interim design decisions must be re-examined as the features upon which they were made are changed.

The following discussion reflects some of the major issues for which interim design decisions have been made or for which the potential solutions have been more clearly defined.

3.1 Plasma chamber access

In deriving a design concept for access to the plasma chamber first wall and other internal attachments, the concerns can be put into two basic categories: (1) inspection and light-duty maintenance and (2) major replacement. Light-duty manipulators installed through penetrations in the torus can be used to accomplish inspection and light-duty maintenance. For major replacement of the first wall and plasma chamber armor, however, it is desirable to provide for the removal of large sections of the torus to a lot cell where the more intricate and extensive operations can be performed more efficiently.

Two concepts of torus segmentation were examined (fig. 5): (1) the number of sectors equal to the number of TF coils and (2) the number of sectors equal to twice the number of TF coils.

If the machine is designed so that the number of sections is equal to twice the number of TF coils, the size of the coils can be minimized on the basis of ripple constraints. Those sectors in the plane of the TF coils, however, have to be rotated out of that plane prior to radial extraction, leading to substantially greater complication in remote maintenance equipment and operations.

With the concept involving the same number of sectors as TF coils, which was adopted for the ETF design, the sectors can be removed with a single radial extraction. In order to minimize the impact of the sector removal on the size of the TF coils, a frame structure that includes only a small part of the outer part of the bulk shield is left in place. It is expected that these frames, which are not exposed to high wall loading, can be designed for the full life of the ETF. If necessary, however, they can be removed for replacement or repair.

3.2 TF coil arrangement

Substantially better access to the torus can be achieved by reducing the number of TF coils below an apparent cost optimum in the 16-20 range (fig. 6). However, initial scoping studies indicated that for a given TF ripple, only a small penalty is incurred for reducing the number of TF coils to about 12.

Accordingly, layouts were made for both Design 1 and Design 2 with a 12-TF-coil arrangement. In the case of Design 1 it was found that the 12-coil configuration did not provide sufficient space for a bundle divertor between adjacent TF coils and was marginal with respect to accommodating the neutral beam injectors, which require a minimal angle of 35° to limit ion loss. In the case of Design 2 a problem was encountered in providing space to extract the poloidal divertor module in 12 segments.

When the bore sizes of the TF coils were increased to provide the clearances required, it was found that the coils for the 12-coil installations had to be at least as large as they would be for a 10-coil installation. This, of course, implied that the 12-coil arrangements would cost more than the 10-coil arrangements when the required access considerations were included. Therefore, 10-coil arrangements were adopted for both Design 1 and Design 2.

High out-of-plane loads are imposed on the TF coils because of the interaction of the TF coil current with the poloidal fields. Various potential solutions were considered, including the installation of intercoil links, a space frame, and shear panels. An arrangement utilizing box beams in the wedge-shaped apertures at the top and bottom of the coil assembly was adopted (fig. 7). This design permits unobstructed access to the torus through ten bays between adjacent TF coils laterally and between the box structures vertically.

3.3 Installation of bundle divertor

Two major problems developed in accommodating the installation of the bundle divertor. One involved providing sufficient room for the bundle divertor between the TF coils, and the other concerned reacting the high radial loads (35 MN) due to the interaction of the magnetic field produced by the divertor with the machine's toroidal field.

The first concern was solved when the umber of TF coils was reduced to ten to permit shield sector removal.

The radial force problem was solved at least conceptually by mounting the divertor coils in a box structure and supporting the box structure on the adjacent TF coils (fig. 8). Because the magnetic forces acting on the adjacent TF coils are opposite and nearly equal, the net load from the system is nearly zero. The support struts between the divertor and TF coils must be designed to minimize the heat leak from the room temperature divertor structure to the cryogenic temperature TF coil structure.

3.4 Poloidal divertor arrangement

Both single- and double-null poloidal divertors were considered for the ETF. The double-null divertor is certainly preferred with respect to plasma physics characteristics because the double null provides vertical symmetry. With regard to installation and maintenance considerations, however, it presents obvious access problems and would necessitate the use of very large TF coils. This implies substantial cost penalties. Accordingly, the single-null design was adopted.

The question of whether to locate the divertor at the top or bottom of the torus was addressed. By locating the divertor collector at the bottom, the pump ducts can be placed at that location, thereby providing maximum access to diagnostics and test modules installed at the torus midplane. In addition, seismic effects on the pumps and pump ducts can be minimized. Also, although the installation of the divertor collector at an elevation below that of the neutral beam injectors poses some difficulty in removing the divertor collectors at those locations, in general, the lower elevation should facilitate the removal and replacement of these modules.

Providing pumping for the inboard divertor targets poses a problem. If pumping is provided, space has to be available for a pumping channel. High-density shielding, approximately 60 cm thick, has to be provided between the pumping channel and the TF coils to protect the electrical insulation in the coils from radiation damage. The plasma radius, then, is dictated by the pumping channel requirements. Although it is recognized that the divertor performance with the single-side pumping arrangement is less effective than it would be with both sides pumped, the single-side scheme was adopted to avoid the reactor size and cost penalties.

Various arrangements for replacing the divertor collector plates were also examined. By splitting each major segment into three smaller segments, it would be possible to remove the divertor target through the pumping duct with the aid of a suitable retrieval device. Alternatively, by dividing the target into two smaller segments, it would be possible to remove the target without impacting the size of the TF coils. However, because the more components that have to be removed, the greater the impact on the machine availability, it was concluded that the target should be designed for removal as one module between each pair of adjacent TF coils.

Alternative locations of the poloidal divertor coil were also considered. Various schemes for locating the divertor coil inside the bore of the TF coils were examined. Several methods for incorporating series of coils that provide toroidal continuity but can be individually removed were considered. Unfortunately, the structural support of such arrangements becomes very difficult. It was concluded that the divertor coil should be located outside the TF coil bore so that it could be a single continuous superconducting coil.

3.5 PF coil arrangement

Two basic approaches have been considered for the PF coil arrangement: normal-conducting coils inside the TF coil toroidal bore and superconducting coils outside the TF coil toroidal bore. It has generally been concluded that replacing a superconducting coil located inside the TF coil toroidal bore poses extremely difficult problems.

Placing the PF coils inside the toroidal bore results in a fast-response system, but the power requirements of a normal-conducting coil system are very large. The power requirements are greatly reduced with the external superconducting coils, but the system response is seriously degraded. It was concluded that the most acceptable arrangement is one that uses large superconducting coils outside the TF coils to produce the basic poloidal field and small normal-conducting copper coils inside the toroidal bore to respond to the plasma shifts and other short-time-scale motions.

A similar question arises relative to the placement of the ohmic heating (OH) coil solenoid. If the OH coils are placed inside the TF coil toroidal bore, the OH coil volt-seconds can be more easily produced; however, the replacement of the OH coils becomes highly questionable. Accordingly, the OH solenoid was placed inside the poloidal bore of the TF coil assembly.

The question then arises as to whether the solenoid should be placed inside the bore of the TF coil bucking cylinder or in recesses located in the outer surface of the bucking cylinder. Again, volt-second considerations favor locating the OH solenoid turns outside the bucking cylinder, but access considerations favor locating the solenoid inside. At present, the solenoid is located inside the bucking cylinder.

3.6 Shielding requirements and maintenance procedures

The reactor shield must be designed to limit the radiation exposure to the electrical insulation in the TF coils, limit the resistivity changes in the copper matrix of the superconductor due to neutron damage, and limit the refrigeration requirements for the TF coil cryogen due to radiation heating of the coil. For sizing the inboard shield the dose limit of 10^9 rads for the epoxy fiberglass used as the TF coil insulation is critical. Eighty centimeters of stainless steel and borated water is used to achieve the required attenuation.

It will be necessary to use remote maintenance techniques whenever the torus is separated or whenever a piece of equipment such as the neutral beam injector, which is subject to direct radiation from the plasma, is accessed. The question of maintenance philosophy and shielding provisions, however, determines whether or not remote maintenance is used in areas outside the bulk shield.

Remote maintenance techniques typically take longer to accomplish but can be started sooner than contact procedures. Also, they require special design features and maintenance equipment. Moreover, a major concern is that it is virtually impossible to design for all possible modes of failure. In particular, a large fraction of operational problems with current machines involves vacuum and coolant lines, electrical connections, and diagnostics adjustments that are difficult to predict and often difficult to access.

An additional consideration is that for major operations requiring remote maintenance procedures, substantial savings in downtime can be achieved by setting up the remote maintenance equipment and preparing the component to be removed using contact procedures. For example, in removing a shield sector the welded seal and bolted brackets between the sector and the support spool can be disjoined and the air bearing transport pallet can be installed using contact procedures.

These considerations led to the adoption of a requirement that the outboard shield be designed to limit the activation dose to 2 mrem/h 24 h after shutdown. This resulted in a basic shield requirement of 80 cm of stainless steel and borated water and 5 cm of lead sheathing.

3.7 Vacuum topology

3.7.1 Plasma chamber

Three basic options were considered for the location of the plasma chamber vacuum boundary for the ETF (fig. 9). Options I and II utilize a secondary vacuum boundary that relieves the leakage constraints on the boundary at the plasma chamber. Option I provides for a mechanical seal between torus sectors and a load-bearing vacuum boundary at the envelope of the TF coils (fig. 9a). Option II also employs a seal between adjacent sectors but has the load-bearing vacuum boundary at the reactor building walls (fig. 9b). Option III has a single leak-tight vacuum boundary at the envelope of the torus assembly (fig. 9c).

The principal advantage of Options I and II is that they permit the use of a somewhat imperfect mechanical seal between adjacent torus sectors, thereby facilitating the disassembly and reassembly of the torus for maintenance purposes. Option I has the principal disadvantage of having a large number of diagnostic and service penetration through the load-bearing vacuum closures at the TF coils. The main concern with Option II is that a leaky seal between torus sectors and vacuum conditions in the reactor building will result in free-molecular-flow leakage of tritium into the reactor building. This could result in substantial tritium contamination of the reactor building walls and equipment surfaces. Mainly because of these concerns the concept with a leak-tight seal at the torus boundary was adopted.

The vacuum boundary at the outside of the torus was judged to be preferable to one at the inside of the torus. It was recognized that an external vacuum seal would result in longer chamber pumpdown time due to virtual leaks. Locating the boundary at the inside of the torus, however, would require access through the torus to remove any part of it to replace the first wall or make major repairs. Such access would have a major impact on the availability of the machine and therefore was rejected in favor of locating the vacuum seal at the outside of the torus.

3.7.2 Magnetic coils

Several options are available for the vacuum vessel(s) for the superconducting coils. One approach is to provide separate dewars for the TF coil assembly, the OH solenoid and colocated EF coils, and the individual outboard OH and EF coils. Another approach is to enclose all of the superconducting coils in a common dewar. There are, of course, intermediate solutions.

The design with a common dewar around all coils (fig. 1) was adopted. It has the major advantage that the support structures for the individual coils are integrated into members at cryogenic temperature with only a limited number of structures carrying loads from the coils to ambient temperature supports. Also, the dewar construction is simplified, reducing the probability of leaks and enhancing the accessibility for repair.

3.8 Plasma chamber vacuum integrity

As indicated in Sect. 3.1, a decision was made to provide the vacuum boundary at the outside of the torus. Two basic approaches to providing the seal between adjacent sectors have been considered: a welded seal and a mechanical seal.

In both cases the seal has to be bakeable because the sector faces between adjacent sectors have to be thoroughly conditioned prior to operation of the machine. Also, it is desirable to decontaminate the surfaces exposed to tritium prior to disassembling any part of the torus to minimize the tritium contamination of the reactor building. This implies a metallic mechanical seal and probably involves a staged seal to tolerate imperfections in the individual seals. The best bakeable seals available require very high bolt loads[7], which severely compromise the objective of minimizing the downtime for repair or replacement.

In view of this concern emphasis has been placed on the development of a welded-seal concept. The vacuum boundary for the top, bottom, and inboard surfaces is provided by an all-welded support spool (fig. 10). This spool incorporates ten radially oriented frames. The torus sectors are inserted into the apertures at the outboard rim, and the vacuum boundary is completed by welding bellows between the torus sectors and the support spool (fig. 11). The structural loads are transmitted from the sectors to the spool through electrically insulated brackets bolted for ease of disassembly.

3.9 Plasma chamber electrical characteristics

In order to permit control of the plasma the plasma chamber must have a time constant of about 100 ms or less for a low-voltage startup. This implies an electrical resistance of 0.1 mΩ or more in the toroidal direction.

Two basic approaches were considered: a high-resistance torus assembly and a torus assembly with a dielectric break and provisions shorting the dielectric break to prevent damage to the poloidal coils during a plasma disruption. The design of a dielectric break that can be sealed against vacuum leaks and readily accessed or removed for leak repair is a difficult problem. The break would probably have to be installed in one of the spool frames, which are located in the planes of the TF coils. In order to remove it the frame would have to be cut away from the spool, and to replace it the frame would have to be rewelded to the spool.

In view of these difficulties it was decided to attempt to design the torus with a high resistance. The spool frames (fig. 10) and individual shield sectors (fig. 1) constitute electrical shorts in the assembly. The toroidal resistance, therefore, is determined by the resistance of the sector seal bellows, the spool cylindrical member, and the interframe panels that make up the top and bottom flanges of the support spool.

An initial attempt was made to design the entire shield and support spool assembly using stainless steel. However, analysis indicated that the resistances would be low by a factor of 2-3. It was decided, therefore, to use Inconel for the high-resistance components. Inconel has the advantages of both higher strength and higher electrical resistance that provide the requisite toroidal resistance.

3.10 Plasma chamber wall protection

The plasma chamber wall must be designed to accommodate the thermal, charged particle, and charge exchange neutral particle loads during normal burn cycles. In addition, it is expected that runaway electrons and trapped helium ions will impact the upper and lower facets of the outboard wall during normal operation. The chamber wall will also have to be protected from plasma disruptions.

Stainless steel tubular panels appear to be an effective method of accommodating the thermal and particle loads to the wall during normal operation. The tube wall has to be relatively thick to accept the particle erosion, but the thickness is limited by fatigue considerations to about 1 cm (fig. 12). Aluminum is also a candidate material for this application because with the higher thermal conductivity, the cooled surface can be made thicker and therefore more tolerant of the erosive conditions. Aluminum is less well characterized, however, and is less tolerant of temperature-overshoot conditions. In addition, for a given design the higher electrical conductivity leads to higher electromagnetic forces resulting from induced currents.

Runaway electrons, although poorly characterized at the present time, have been known to cause severe damage because of their deep penetration. To protect the upper facet of the outboard wall against runaway electrons and helium ions impinging on the mirror surface, graphite-armored, water-cooled panels are installed in 10° segments between the radial planes of the TF coils.

Although it is expected that an operating regime relatively free of major disruptions can be mapped early in the initial phases of operation, it is also expected that major disruptions will still occur because of plasma anomalies and equipment failure. For design purposes the frequency of these major disruptions has been estimated to be 10^{-3} per burn, or about 500 for the life of the reactor. It has been predicted that the major fraction of thermal energy from these disruptions will impact either the inboard, top, or bottom walls of the chamber on a time scale of approximately 20 ms. Analysis indicates that a metal tube wall would withstand only a few of these disruptions. Accordingly, an armored-wall concept was developed. Graphite tiles 3 cm thick, as limited by maximum temperatures to prevent acetylene generation, and 15 cm on the sides are attached to studs installed on the inboard wall. Similar tiles 2.5 cm thick are attached to the top and bottom walls where cross-chamber radiation cooling is less effective.

4. CONCLUSIONS

The two ETF conceptual designs being developed appear to meet the requirements of the fusion program to serve as the means for developing fusion engineering technologies and thereby to demonstrate the practicality of fusion energy. The basic tokamak systems can be integrated and operated in such a way as to provide the confidence needed to proceed with the more advanced EPR and/or DEMO reactors.

Now that basic designs have been developed for the two divertor concepts, more systematic evaluations will be made of the design alternatives on a subsystem and component level. For example, an aluminum first wall will be evaluated and compared with the tentatively adopted stainless steel concept. Alternative arrangements of various components such as the shield sector internal support will also be examined.

Alternative impurity control systems such as the pumped limiter will also be considered. Continuing theoretical and experimental results should lead to better design definition in many areas and possibly to improved design concepts in some.

Although the concepts have yet to be integrated with the ETF designs, space is available in several bays to accommodate first-wall, blanket, shield, and materials test modules. It is expected that in the case of the first-wall and blanket test modules, extensive non-nuclear testing and screening will precede installation in the ETF so that the impact on availability due to premature failure of the test components will be minimized.

The remote maintenance system design will receive more attention. Conceptual designs for special purpose equipment, such as neutral beam injectors and shield sector transporters, as well as general purpose machines, will be developed. It is expected that substantial research and development will be required to develop the machines with the capacity and versatility needed for the ETF.

Cost is another area that will receive increased effort. As the machine becomes better defined, better cost estimates can be made. The total cost estimate will then lead to a re-examination to confirm or redefine the mission of the ETF in the light of the payoff from the project.

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FIGURE CAPTIONS

Fig. 1. ETF Design 1 — elevation view.

Fig. 2. ETF Design 1 — plan view.

Fig. 3. ETF Design 2 — elevation view.

Fig. 4. ETF Design 2 — plan view.

Fig. 5. Torus segmentation concepts.

Fig. 6. Reactor cost as a function of number of TF coils.

Fig. 7. TF coil intercoil support structure.

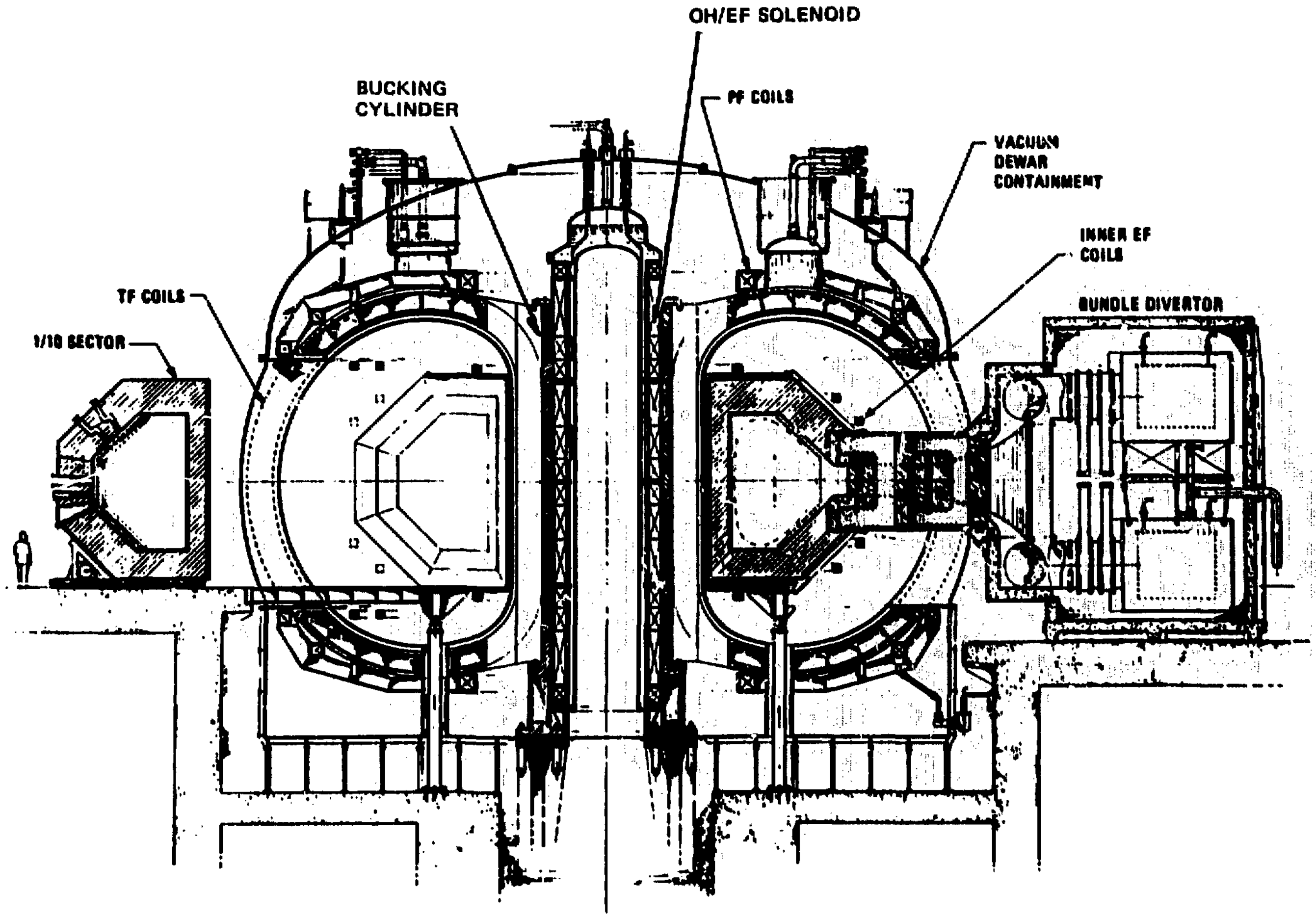
Fig. 8. Bundle divertor support system.

Fig. 9. Vacuum topology options.

Fig. 10. Spool structure.

Fig. 11. Sector vacuum seal.

Fig. 12. First-wall life.



OH/EF SOLENOID
BUCKING CYLINDER
PF COILS
VACUUM DEWAR CONTAINMENT
INNER EF COILS
TF COILS
1/10 SECTOR
BUNDLE DIVERTOR

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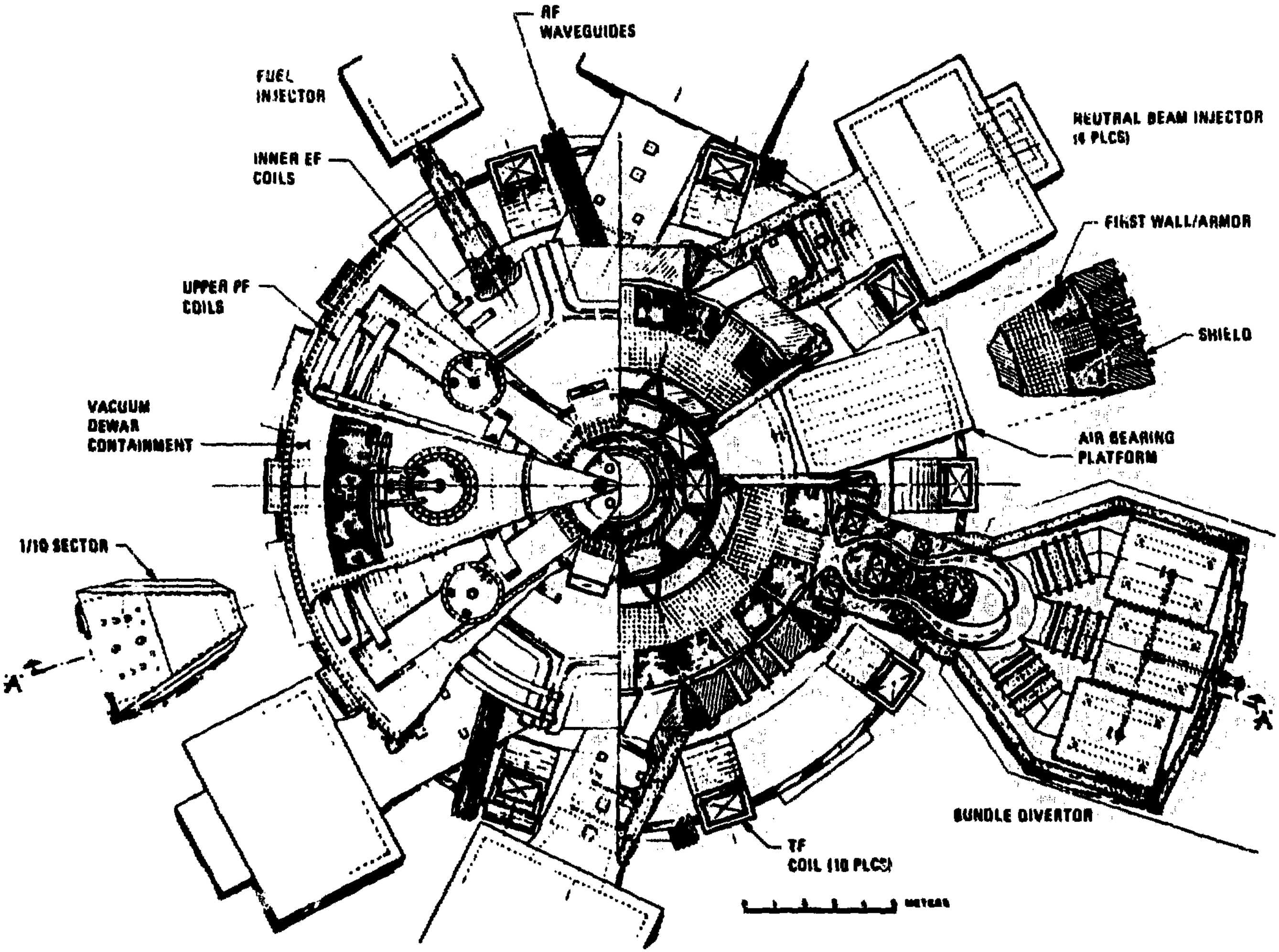

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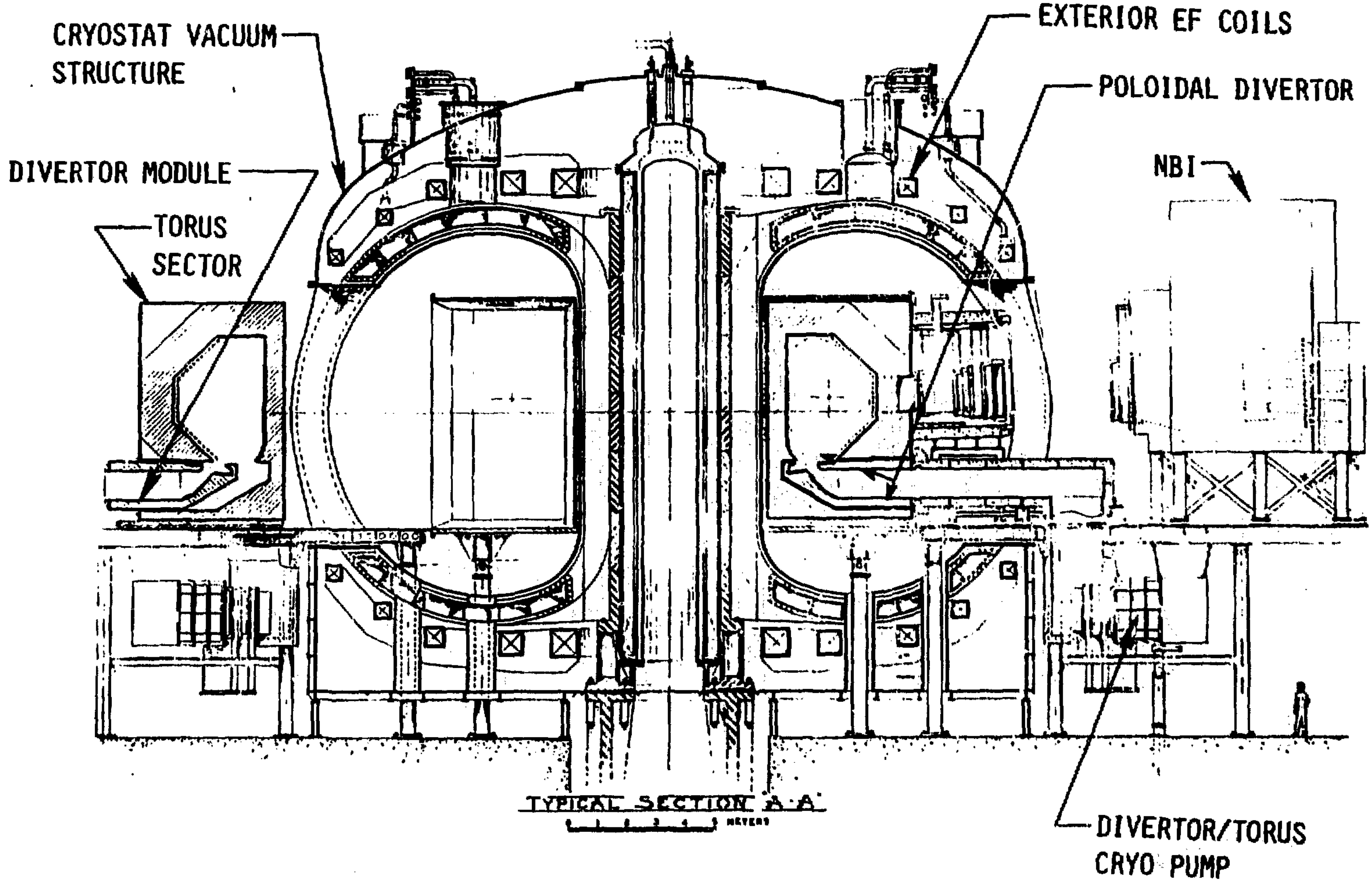
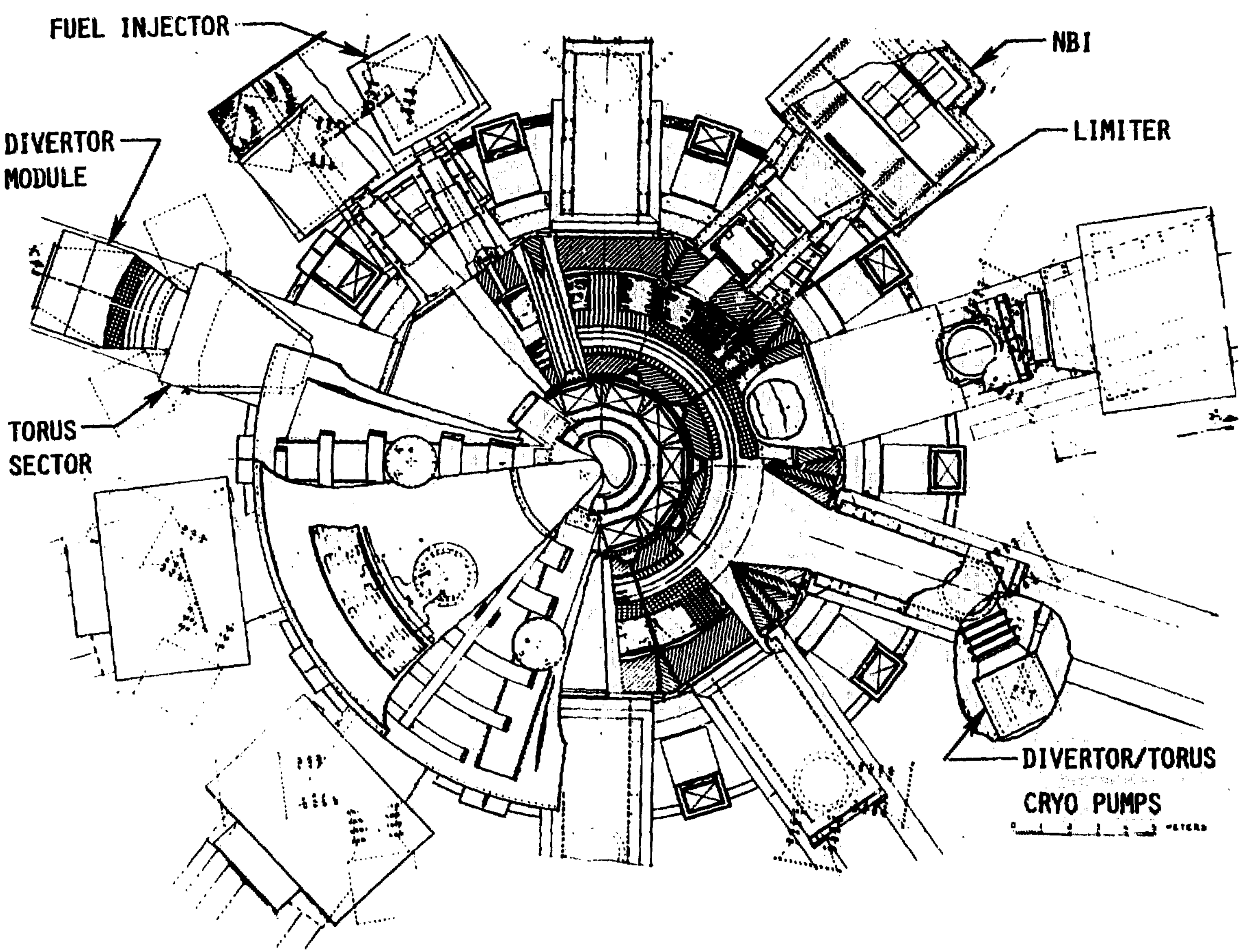
DESIGN 2 - ELEVATION VIEW



TYPICAL SECTION 'A-A'

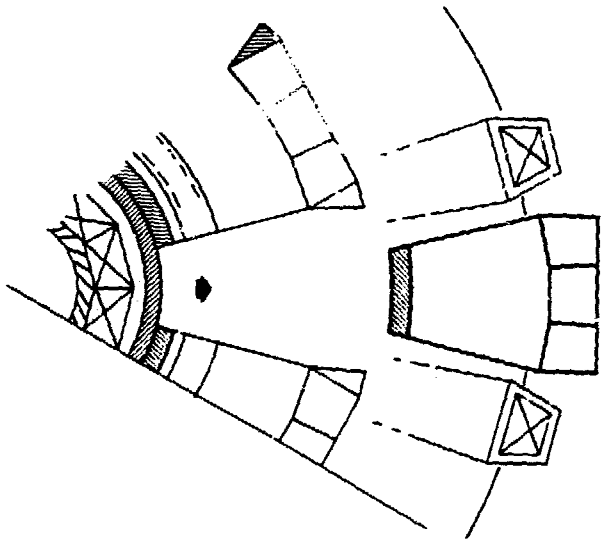
ORNL-DWG 80-3106 FED

DESIGN 2 - PLAN VIEW

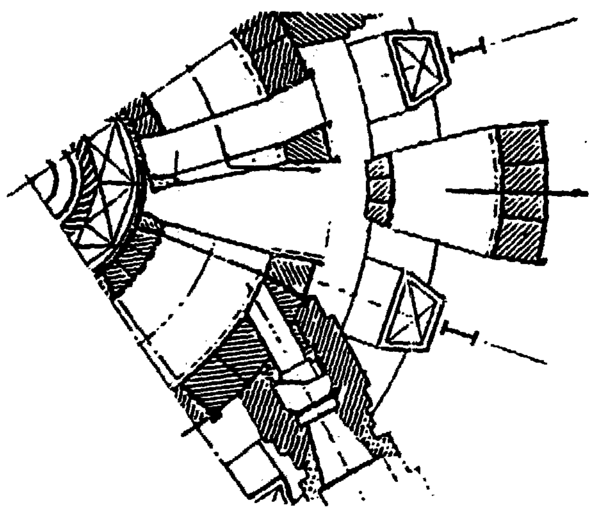


TORUS SEGMENTATION APPROACHES

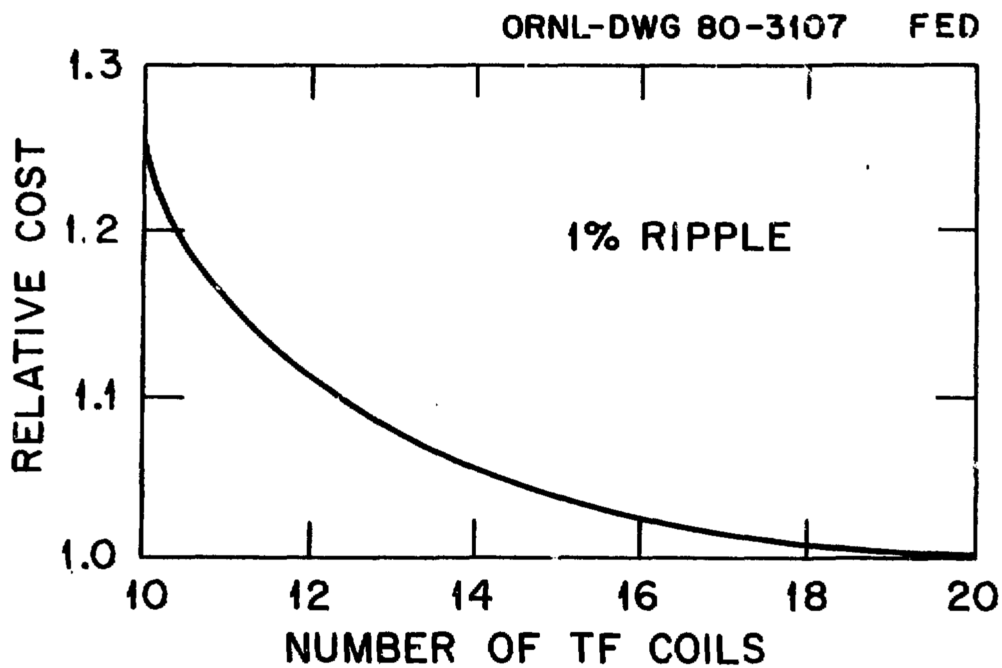
FED/VU 80-300



- TORUS SEGMENT EQUALS THE NUMBER OF TF COILS



- TORUS SEGMENT EQUALS A MULTIPLE NUMBER OF TF COILS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1.3
1.2
1.1
1.0
RELATIVE COST
1% RIPPLE
10
12
14
16
18
20
NUMBER OF TF COILS

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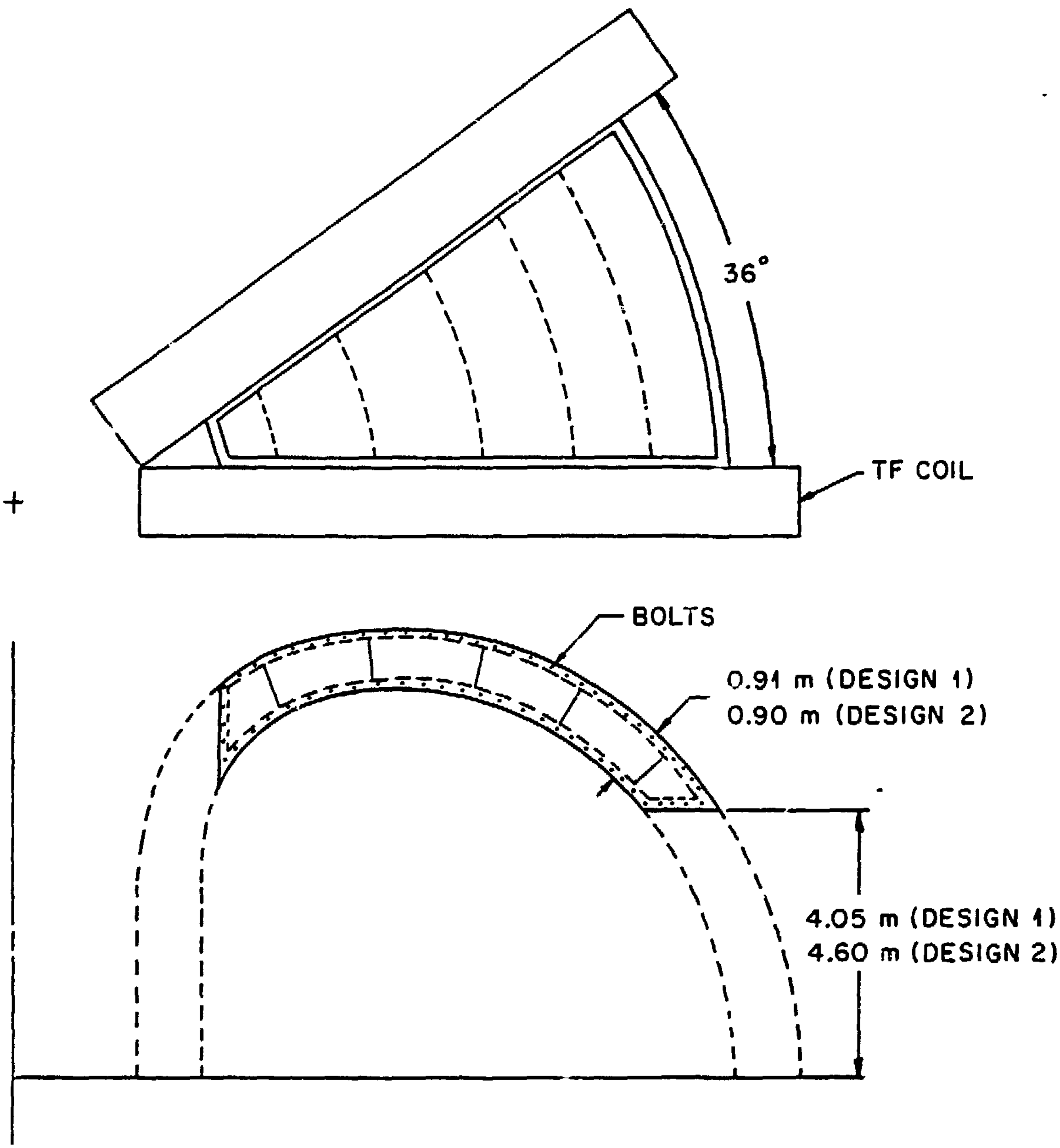
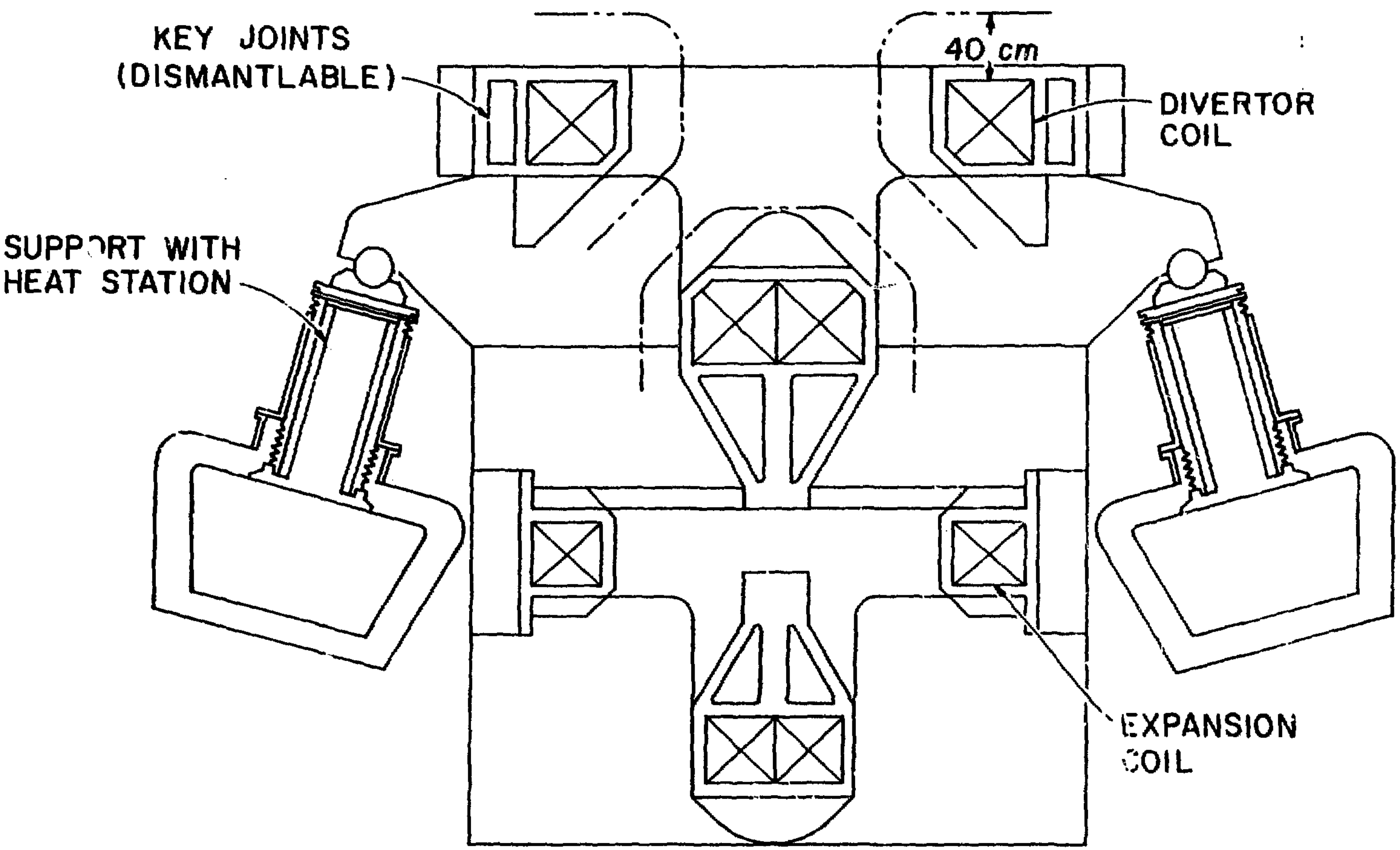
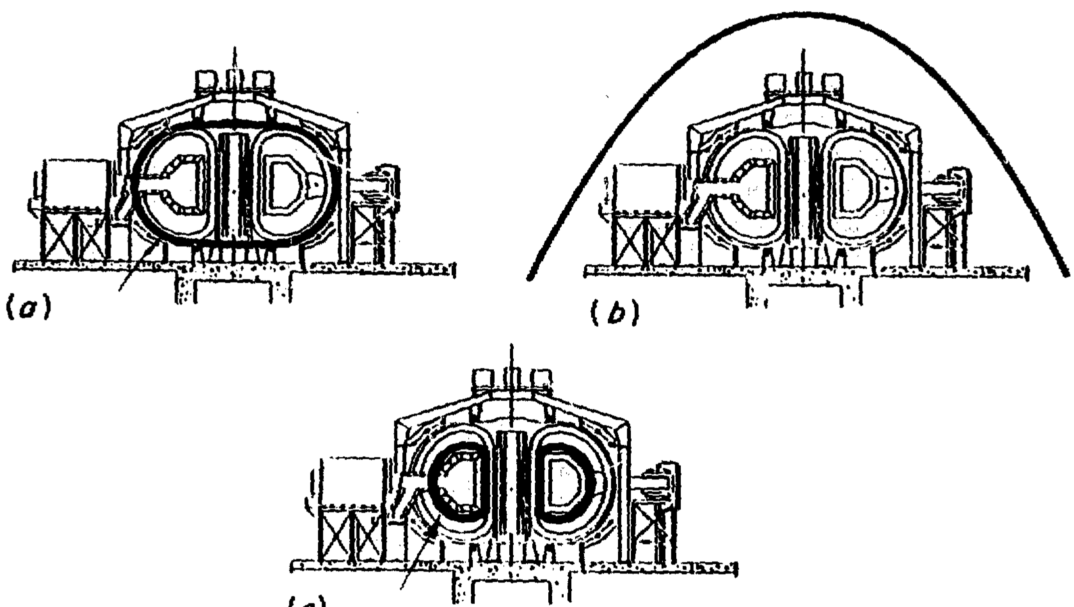


ORNL-DWG 80-3109 FED





(a) (b) (c)

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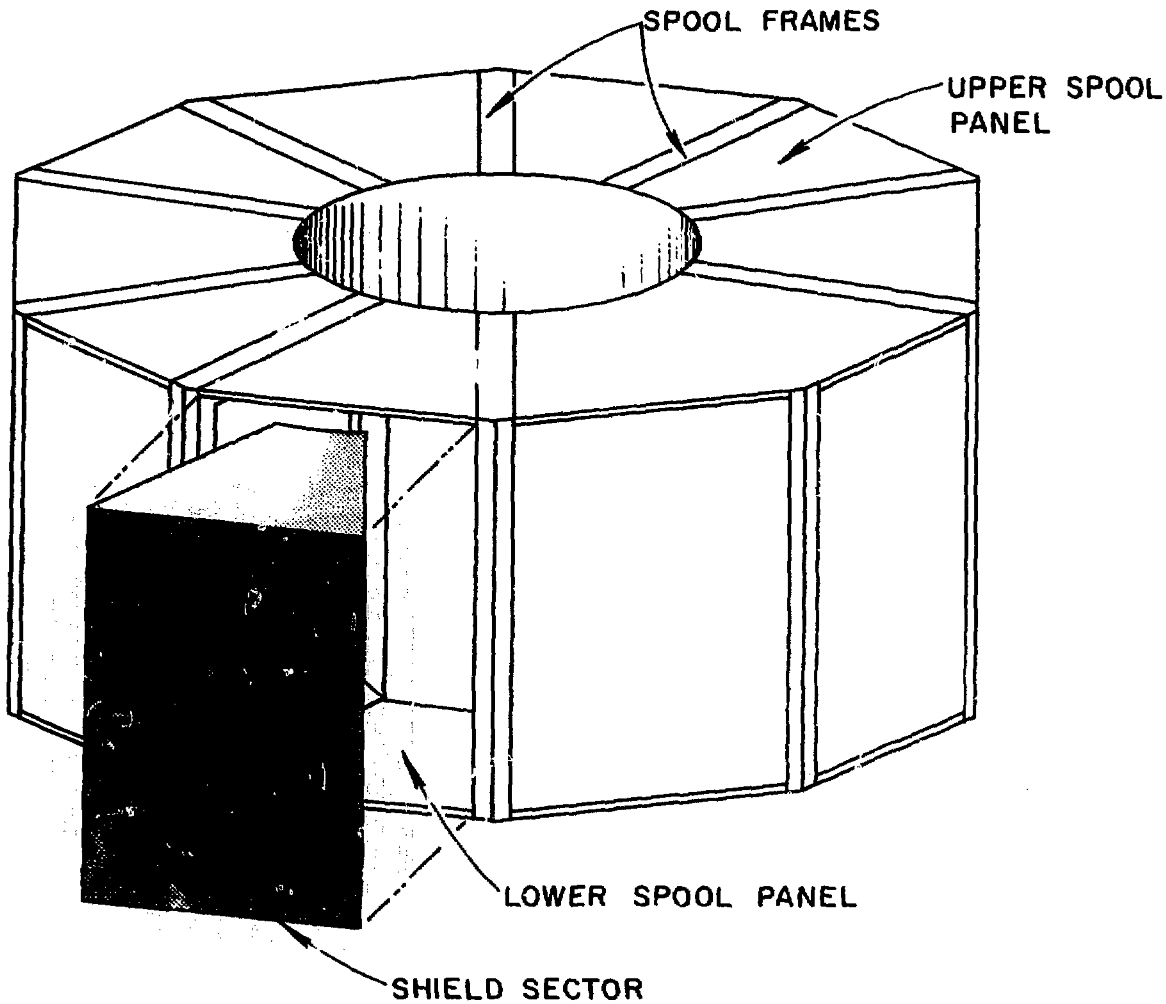
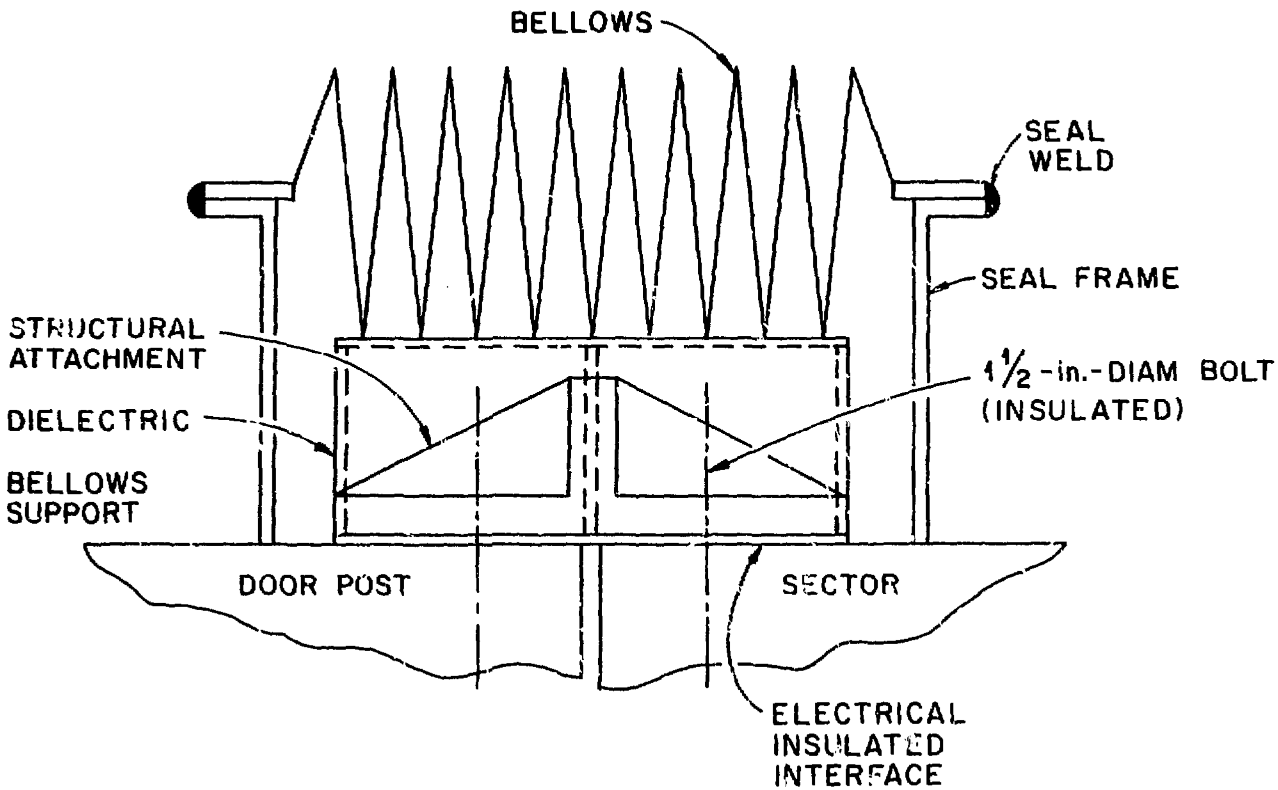
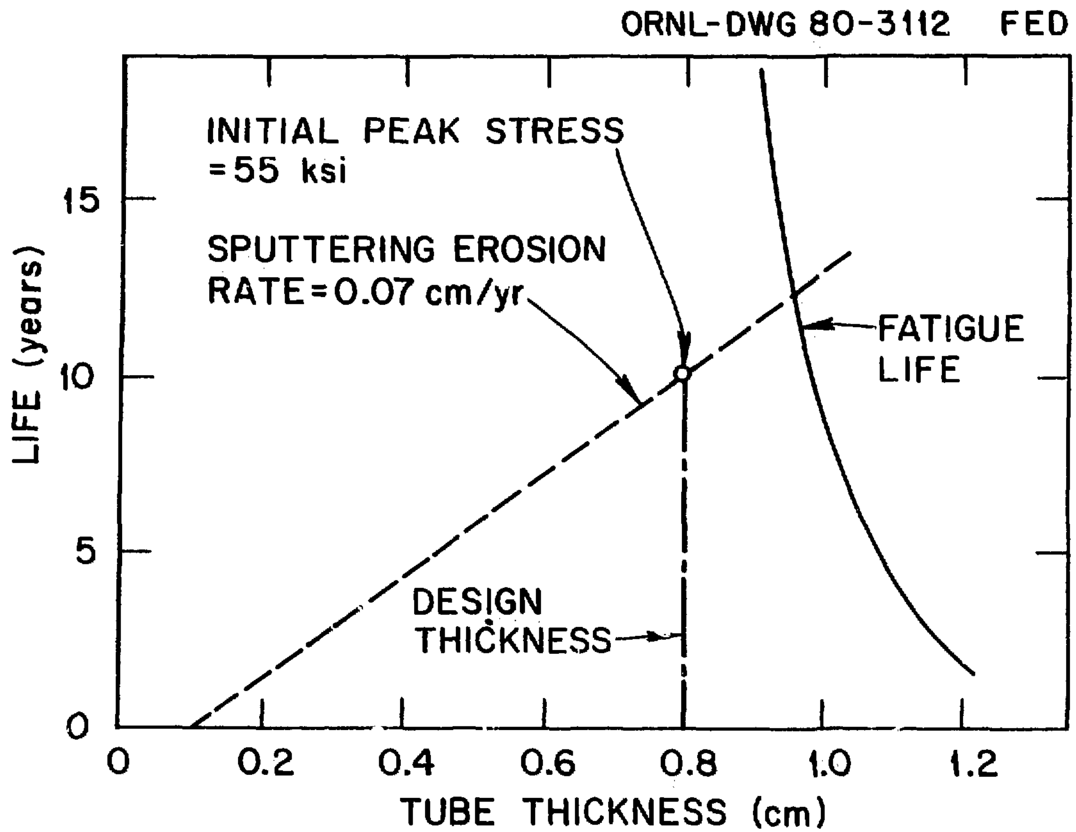


ORNL-DWG 80-3111 FED

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INITIAL PEAK STRESS
=55 ksi
SPUTTERING EROSION
RATE=0.07 cm/yr
FATIGUE
LIFE
DESIGN
THICKNESS
LIFE (years)
TUBE THICKNESS (cm)
0
5
10
15
0
0.2
0.4
0.6
0.8
1.0
1.2

TABLE I

ETF Design Parameters[a]

Plasma major radius, R	5.4 m
Plasma elongation, δ	1.6
Plasma minor radius, a	1.3 m
Plasma volume, V_p	289 m^3
Plasma current, I_p	6.1 MA
Neutron wall loading, L_w	1.5 MW/m^2
Total fusion power, P_{tot}	750 MW
Fusion power density, n	2.6 MW/m^3
Number of TF coils	10
TF coil vertical bore	10.8/12.6 m[b]
TF coil horizontal bore	7.5/8.6 m[b]
Field at TF coil, B_m	11.4 T
Field on-axis, B_T	5.5 T
Steady-state burn time	100 s
Total cycle time	135 s
Total volt-seconds	85
Neutral beam power, P_{inj}	60 MW
Neutral beam energy, E_{inj}	150 keV
Injection time, τ_{inj}	6.0 s
Microwave power (startup)	5 MW
Microwave frequency	140 GHz

[a]Reference 6.

[b]Design 1/Design 2.