Mechanical and Thermal Design Aspects of the Fusion Engineering Device (FED)

B.A. Cramer
McDonnell Douglas Astronautics Company/Fusion Engineering Design Center, Oak Ridge National Laboratory, Oak Ridge, Tennessee 37830, U.S.A.

SUMMARY

This paper highlights major design features and associated mechanical and thermal problems being investigated in the Fusion Engineering Device (FED) design studies. Specific areas discussed include the toroidal field coil structural design, shield sector design, first wall design, and pumped limiter design.

The toroidal field (TF) coils are major structural components of the tokamak configuration. The pulsed nature of the tokamak operation has been identified as one of the most difficult structural design problems for the TF coils. Fatigue crack growth as a result of pulsing the poloidal fields is being considered in the FED TF coil design. The effect is significant. The design configuration results from a trade-off between the mission of the device (the number of cycles) and the overall placement and structural design of both the toroidal and poloidal field coils. Maintenance considerations and overall access to the machine are additional factors in the evolution of the coil design.

The plasma chamber vacuum closure concept allows a means of access to the plasma chamber of the tokamak which enhances the feasibility of maintenance operations. The design provides a vacuum seal at the outside of the shield. Access is gained between the TF coils. Each of the ten shield and first wall assemblies is removed through openings between the TF coils. Other considerations such as vacuum sealing, nuclear shielding, and electrical requirements are addressed.

Design of the plasma chamber first wall components is one of the most challenging engineering efforts in fusion reactor development. In the FED design the feasibility of using a pumped limiter for impurity control is being studied. This limiter defines the plasma edge and thus receives high heat and particle fluxes. The result is a shorter life because of erosion from physical sputtering, erosion from chemical sputtering, and vaporization from plasma disruption radiation. Because of the limiter, however, the remainder of the first wall receives modest heat loads, allowing for thick structures which provide margin for severe unexpected events.

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1. Introduction

The Congress of the United States passed legislation in September 1980 that established a national goal to operate a magnetic fusion demonstration plant at the turn of the twenty-first century. One of the first objectives of the program is to demonstrate the engineering feasibility of magnetic fusion with a Fusion Engineering Device (FED). Toward achieving this goal, a team of scientists and engineers has been assembled at the Oak Ridge National Laboratory to conduct conceptual design studies for the FED development. The study results presented in this paper represent contributions from the entire FED Design Center staff.

2. Fusion Engineering Device Design

Elevation and plan views of the current FED configuration (March 1981) are shown in Figures 1 and 2. These views show the major reactor components. The magnet system consists of toroidal field (TF) coils and poloidal field (PF) coils. Except for the PF coils between the shield and TF coils, all coils are superconducting and are enclosed in a large common cryostat. The torus shield structure also forms the vacuum vessel and contains the first wall components and pumped limiter. Associated with these are the vacuum pumping system. The primary plasma heating mechanism is ion cyclotron resonance heating (ICRH). Capabilities are also being considered for neutral beam heaters and bundle divertor options. Key parameters are listed in Table 1.

3. Toroidal Field Coil Structural Design

The TF coils are subjected to loads both in, and normal to, the plane of the coil. The in-plane load remains constant with time after the TF coil is energized and tends to dilate the TF coil. The out-of-plane pulsed load, from pulsing the PF coils, results in out-of-plane moments tending to overturn the TF coil about its horizontal midplane.

In pulsed tokamak reactors, stress limits imposed by fracture mechanics considerations tend to be very restrictive compared to conventional membrane and bending stress limits. This effect is demonstrated in Figure 3, where allowable stress of 316 LN stainless steel is plotted as a function of the number of design cycles. The plot includes a factor of 2 on stress or 4 on cycles, whichever is more conservative. It has been assumed that an initial flaw diameter of 0.5 cm can be detected by ultrasonic testing. The allowable stress for the FED TF coil case, at design values of 3.5×10^5 full-field pulses and 8 T field in the winding, is 110 MPa which is well below the S_m value of 460 MPa. The resulting coil case wall thicknesses are approximately 10 cm.

4. Shield Sector Design

The torus assembly includes a spool structure which forms the vacuum walls for the in-board, top and bottom of the reactor. The shield is divided into ten sectors, each of which can be installed and removed radially from the spool between the TF coils using wheeled or air flotation techniques. The outer vacuum seal is made between a sector flange, the spool at the top and bottom, and spool frames along the sector sides. An approach for removal of a shield segment is shown in Figure 4. This assembly has many challenging and often competing design requirements. For example, the vacuum sealing includes the following requirements:

- accessibility,
- inspectability,
- easy replaceability with remote techniques,

- separation from structure,
- tolerance to misalignment of shield segment,
- resistance to radiation damage,
- compatibility with bakeout techniques, and
- minimization of virtual leaks.

Electrical resistance is also an important consideration. Two general approaches are available for meeting the resistance requirements. They are: (1) provide high resistance continuous structure and (2) provide virtually infinite resistance through discontinuous structures (dielectric breaks). In the first case, the problem is to provide material thin enough while still satisfying load carrying requirements. An example of this is the problem of supporting thin shield walls for internal coolant pressure. In the second case, the problem is to maintain vacuum integrity at the dielectric break location. An additional consideration is providing electrical features for protection from arcing erosion during plasma disruptions. To accomplish this, the FED shield sectors are electrically connected at the wall nearest the plasma by compressible metal contacts.

Nuclear design of the tokamak bulk shield is a much more complex problem than the analogous design of a fission reactor shield. The competing requirements on the inboard shield are minimum thickness versus adequate protection of components. The shield is a box structure designed to achieve optimum proportions of steel and water. Several steels are being considered. Requirements include minimizing heat loads to magnets, radiation damage to magnet materials and shutdown dose rate for maintenance.

5. First Wall Design

The baseline first wall protection design for the FED shield considers three regions. These regions, shown in Figure 5, are (a) actively cooled stainless steel wall occupying the 45° and outboard shield facets, (b) passively cooled graphite armor occupying the top and inboard walls, and (c) passively cooled graphite armor bands on each side of the limiter for charge-exchange neutral protection. The graphite armor is coated with titanium carbide. Requirements for attachment of first wall tiles and coolant panels are based on thermal expansions and plasma disruption electromagnetic loads.

The design philosophy for the FED first wall is that components should be relatively thick (>1 cm) and be capable of accepting locally severe conditions. This approach provides a maximum degree of confidence in reactor operation for a reasonable length of time. Thus, no special protection is provided for runaway electrons.

Because of the relatively low heat loads, (7.5 W/cm^2 surface heat load) the stainless steel first surface is 1.25-cm thick with maximum structural temperatures of 200°C and associated maximum thermal stresses of 130 MPa. This thickness should provide capability for accepting a limited number of disruption loads and unexpected erosion.

Graphite tiles on the inboard and top reactor surfaces provide maximum protection capability for shock and heat load conditions associated with plasma disruption and neutral beam dumps. Additionally, graphite satisfies the desirability for materials with low atomic number over a significant portion of the first wall. The titanium carbide coating is intended to control chemical erosion as much as possible.

It is desirable to avoid extensive periods of time with graphite surface temperatures in the methane generation range (400°C to 800°C) and the acetylene generation range

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(>1400°C). For the baseline FED tile thickness of 5 cm, armored surfaces operate with maximum surface temperatures between 800°C and 1400°C. During cool-down between burns, inboard and top armor surface temperatures will drop within the methane generation range, thus causing some chemical erosion to occur.

Erosion rates for the first wall graphite and stainless steel surfaces are based on a four-phase operating scenario. Erosion of the graphite tiles covering the top and majority of the inboard wall are subjected to disruption vaporization from 2000 disruptions. In these areas the total erosion over the reactor life is 0.5 cm. Erosion rates are higher where the C-X neutrals impinge because of additional physical erosion. The total erosion in these regions is 2.6 cm.

6. Limiter Design

The limiter for the FED baseline design is located at the bottom of the chamber. The limiter extends around the entire torus. It is made up of ten removable segments, one for each shield sector. A particle collection/vacuum channel extends around the torus below the limiter. This continuous channel converges to ten separate vacuum ducts extending outward between the TF coils. Each removable limiter segment weighs approximately 15 tonnes.

The limiter design is shown in Figure 6. The limiter leading edge must extend far enough beyond the plasma edge to minimize heat and particle loads but close enough to allow adequate pumping. The structure is copper with integral coolant channels. The surface material is graphite. Tile thickness is 1.25 cm on the top and 0.25 cm on the bottom of the limiter. Overall limiter thickness is limited by pumping requirements. The copper substrate maximum temperature is maintained below 200°C. The maximum graphite tile temperature is 1200°C.

Loads on the limiter include internal coolant pressure and plasma disruption electromagnetic forces. Additionally, thermal stresses occur during operation and also result from fabrication processes such as brazing. The coolant pressure of 1.0 MPa is accepted by appropriate internal rib spacing. Thermal stresses in the copper structure near the limiter tip are approximately 70 MPa during reactor operation. Electromagnetic loads from plasma disruption are approximately 0.4 MPa, normal to the limiter surface. This load on the cantilevered limiter requires increasing limiter depths and structural thicknesses with increasing distance from the leading edge. Adequate load carrying capability can be achieved, however, for the 20 cm limiter depth at the shield support point.

Several erosion mechanisms combine to limit the life of the limiter. These mechanisms are physical sputtering from charged particles and charge-exchange neutral particles, chemical sputtering, and vaporization caused by the intense heating during plasma disruptions. For materials with low atomic number the limiter tile erosion rate results in a life of a few months at 10 to 20% availability. Use of materials with high atomic number would result in a life approximately equal to the FED operating life.

7. Concluding Remarks

In the course of the FED design effort, many key issues are being identified and studied. This effort is being directed toward development of an optimum feasible design space and machine cost. Issues include the ability to adequately inspect, repair, and replace large magnet and shield components. Also, the importance of controlling plasma edge conditions and developing an understanding of plasma disruption and erosion processes cannot be overemphasized. Continued studies toward developing an understanding of these and other key issues are important in meeting magnetic fusion goals.

Table 1. Key parameters for the FED configuration (March 1981)

Plasma major radius, R (m)	4.8
Plasma minor radius, a (m)	1.3
Plasma elongation, δ	1.6
Plasma current, I_p (MA)	5.4
Number of TF coils	10
Conductor	NbTi
Maximum field at TF coil, B_m (T)	8
Field on axis, B_t (T)	3.6
TF coil bore, m	7.5×10.9
ICRH plasma heating power (MW)	36
Neutron wall load (MW/m^2)	0.4
Total fusion power (MW)	180
Plasma volume, V_p (m^3)	267
Burn time, t_{BURN} (s)	100
Lifetime pulses	3.5×10^5

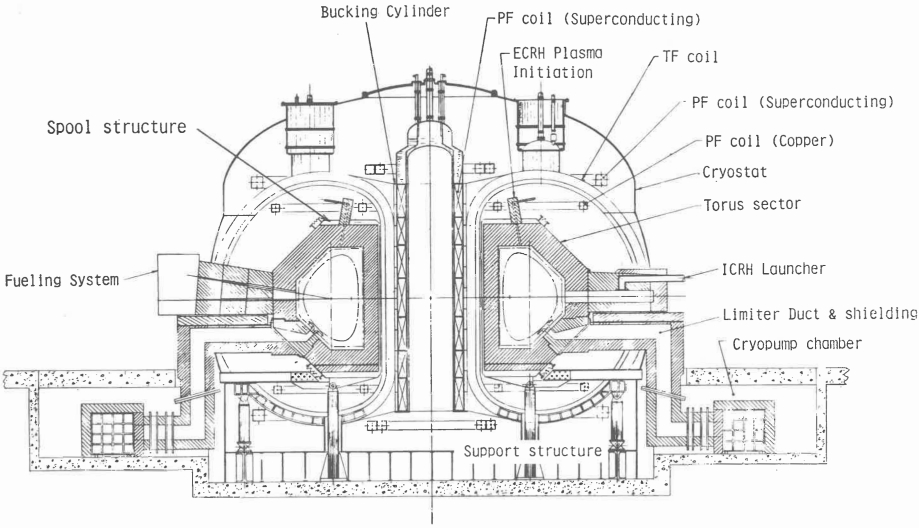

Figure 1. FED configuration elevation view (March 1981)

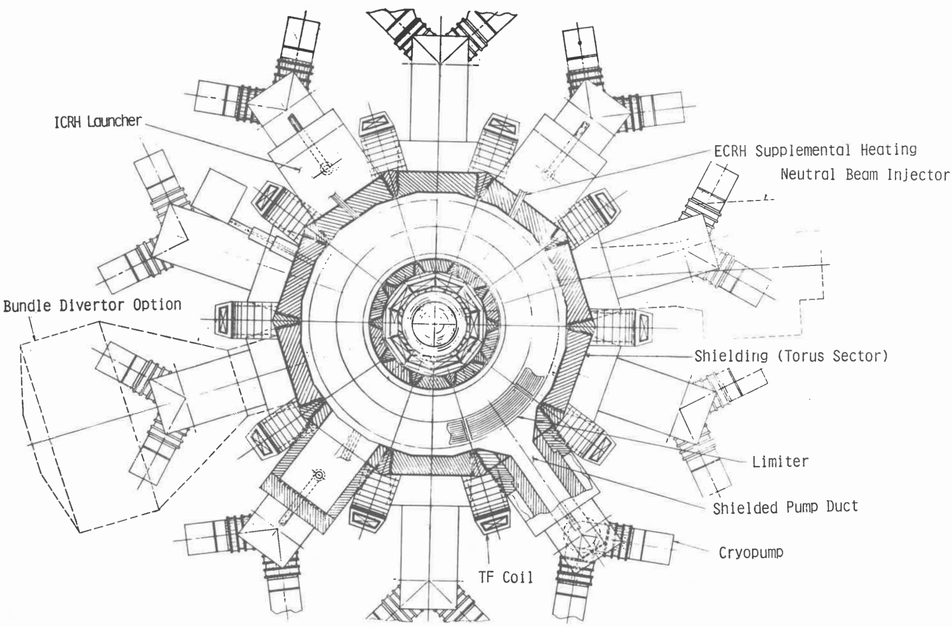

Figure 2. FED configuration plan view (March 1981)

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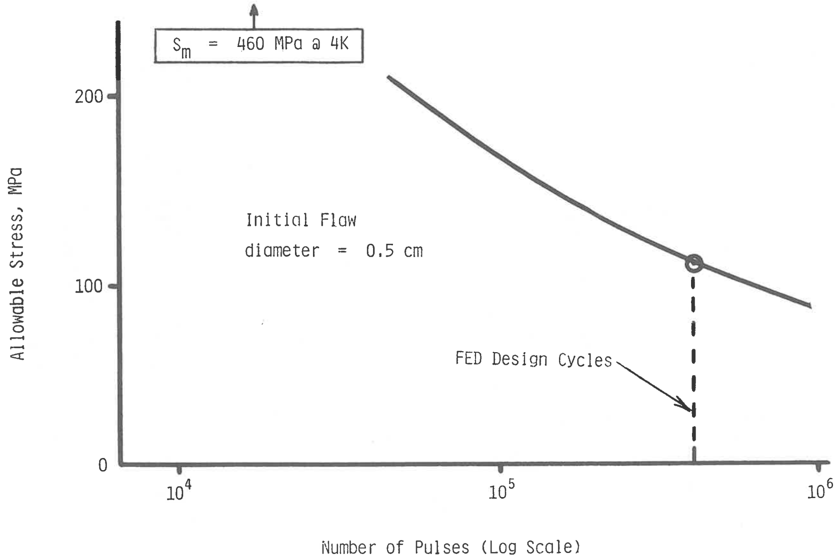

Figure 3. Allowable stress in magnet support structures based on fracture mechanics considerations

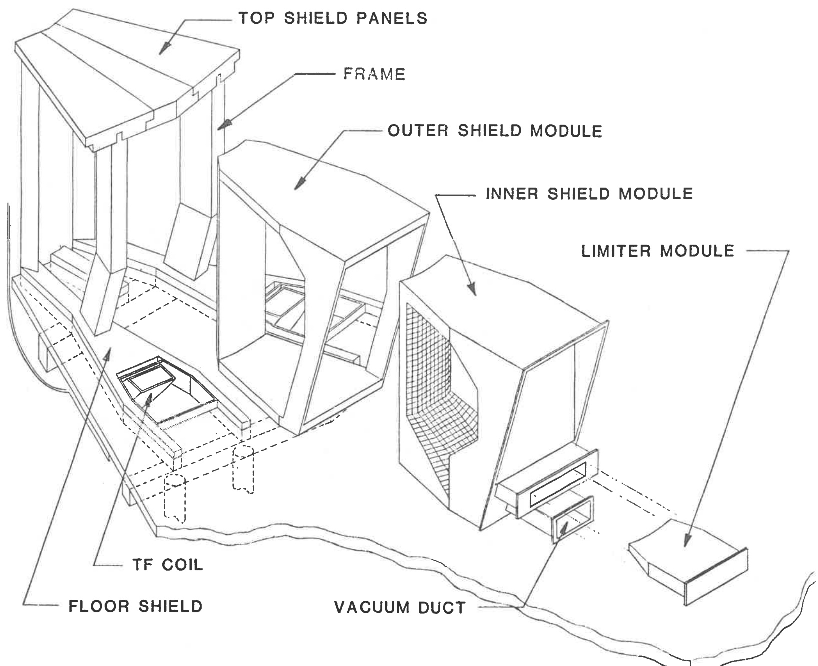

Figure 4. Shield segmentation, installation, and removal approach

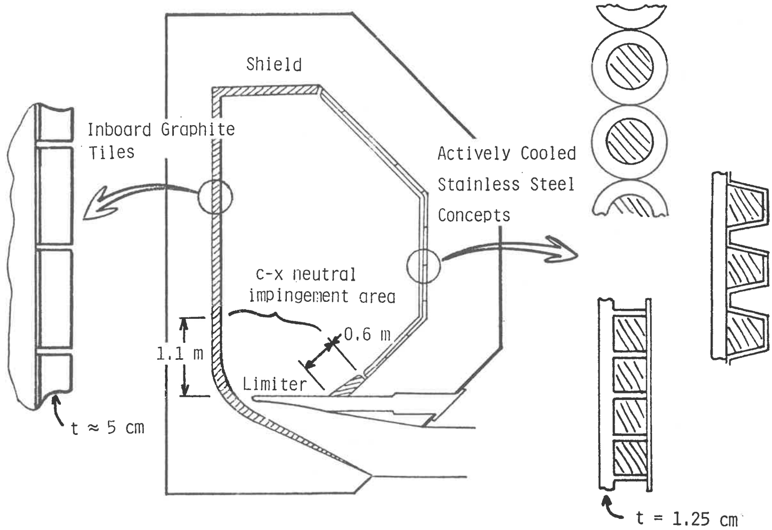

Figure 5. First wall protection for the shield

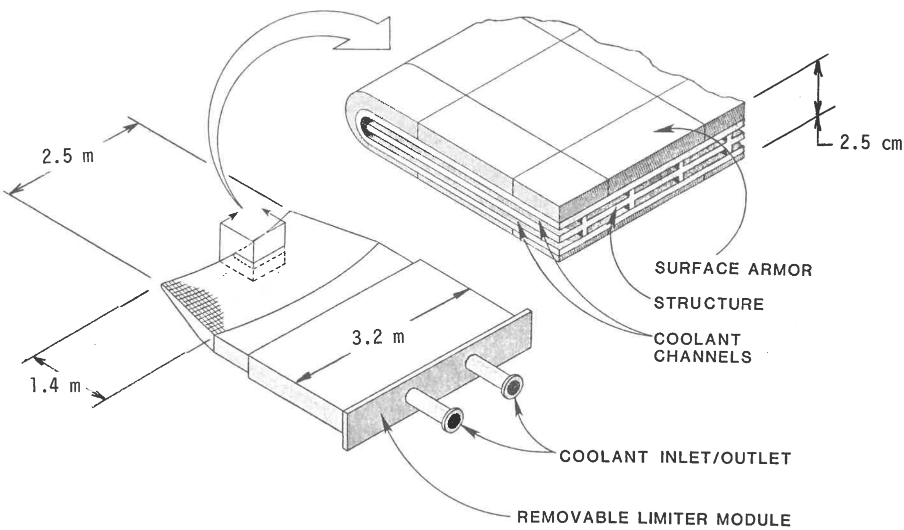

Figure 6. Pumped limiter configuration