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# FUSION ENGINEERING DEVICE (FED) FIRST WALL/SHIELD DESIGN*

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## Introduction

The torus of the Fusion Engineering Device (FED) (Fig. 1) is comprised of the bulk shield and its associated spool structure and support system, the first wall water-cooled panel and armor systems, and the pumped limiter. The bulk shielding is provided by ten shield sectors that are installed in the spool structure in such a way as to permit extraction of the sectors through the openings between adjacent toroidal field coils with a direct radial movement. The first wall armor is installed on the inboard and top interior walls of these sectors, and the water-cooled panels are installed on the outboard interior walls and the pumped limiter in the bottom of the sectors.

Fig. 1. FED torus components.

The overall design of the first wall and shield system is described in this paper. More detailed descriptions of the components of this system are presented in other papers.[1,2] A summary of the nuclear analysis used to determine the shield composition and thickness is also presented elsewhere.[3]

## Bulk Shield

The shielding in the inboard region must meet three criteria: (1) limit the heating in the toroidal field (TF) coil to 5 mW/cm$^3$ so that it can be accommodated by a refrigeration system of economic size, (2) limit the damage to the copper in the TF coil to $3 \times 10^{-4}$ dpa so that the resistivity will increase no more than 25% during the life of the reactor, and (3) limit the dose to the electrical insulation in the magnets to $10^9$ rad. These criteria are not considered to be absolute. The first two are subject to tradeoffs in establishing an optimum overall reactor design, and the absolute limits to insulation damage are not well established.

The outboard shielding has to be designed to meet a more demanding requirement: limiting the personnel exposure to 2.5 mrem/h 24 h after shutdown. This is required to permit contact maintenance on systems and components located outside the shield envelope and set up of remote maintenance equipment for planned remote operations within the shield envelope.

## Shield Sectors

A shield sector is shown in Fig. 2. The effective shield thickness on the inboard side is 60 cm; the top, bottom, and outboard shield thicknesses are 1.20 m, including 5 cm of lead on the outer surfaces. The basic shield materials are Nitronic 33 and water, the water also serving as the coolant. The stainless steel-water distribution varies from 97% stainless steel-3% water (by volume) at the plasma side to 25% stainless steel-75% water near the outer surface.

Fig. 2. Shield sector.

The shield sector is a welded box structure of Nitronic 33 enclosing Nitronic 33 plates. At the plasma side of the box, the plates are bolted to each other. These plates contain grooves through which the coolant water flows. The stiffened box structure carries the coolant pressure load. Each of the bulk shield plates is coated with $Al_2O_3$, a dielectric, in order to minimize the eddy currents generated by plasma disruptions. The maximum steady-state temperatures of the inner surface of the stainless steel shield is 200°C.

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The Nitronic 33 stainless steel was chosen because the long-life radionuclides are significantly decreased with the reduced nickel content (relative to 316 SS). Nitronic 33 is commercially available and has a high resistance to corrosion.

Three methods of adding boron to enhance the capture of thermal neutrons and thereby reduce the refrigeration costs of the TF coil cooling were investigated: (1) borating the steel shield material, (2) adding boron to the coolant water, and (3) inserting a layer of boron carbide ($B_4C$). None of these methods resulted in net cost savings in the cost of the refrigeration system over the cost of adding boron. Therefore, boron was not included in the baseline shield composition.

Replacement of each sector is accomplished by moving the sector radially out of the spool structure on built-in roller dolly assemblies. The sector load is transferred to an independent pallet during extraction. When sufficiently clear of the TF coils, the sector is transported to its destination by the overhead crane.

Bakeout of the torus interior is accomplished with hot nitrogen gas injected into the shield sector coolant passages after the water coolant has been drained. A minimum temperature of 250°C is attained in less than 24 h after initiation of the heating.

Two shield sectors are dedicated to test modules. Since each sector can accommodate six test modules, each with about 1 $m^2$ of surface area facing the plasma, a total of 12 test module locations is available.

## Spool Structure

The spool structure (Fig. 3) consists of ten panel assemblies and ten radial frame units. Each panel assembly consists of a top panel, inboard panel, and a bottom panel. The radial frames divide the spool into ten window openings or bays. A shield sector is inserted into each bay.

The spool structure not only accommodates the shield sectors but also serves as an accessible vacuum boundary for the plasma chamber. Vacuum integrity must be maintained to prevent ingress of impurities during burn and to permit pumpdown to $10^{-7}$ torr for bakeout.

A double-skin panel configuration is used to maximize the toroidal electrical resistance. Inconel was selected because it possesses the highest electrical resistance for a given structural requirement. Panel thicknesses were determined by the pressure loads and the sector dead weight. Another approach for controlling electrical resistance in the spool is the use of dielectric breaks. This option is being considered as an alternative design.

## Torus Support System

The torus support system consists of ten vertical columns, a toroidal support platform, and lateral support panels (Fig. 1). The torus shield sectors and spool structure are supported by the support platform and vertical columns, one of which is located under each shield sector. Lateral and rotational loads are taken by the lateral support panels at the base of the spool structure. The lateral support panels also provide a base for installation and removal of the shield sectors. All support system components are constructed of 304 stainless steel and operate at room temperature.

## First Wall Systems

The interior of the plasma chamber is designed to protect the basic torus from heat loads and energetic particles generated by the plasma and to protect the plasma from deleterious impurities generated by interaction of the plasma with the interior surfaces. Of special concern are the inboard and top and bottom surfaces that are subject to very high, but poorly defined, heat loads during disruption.

## Inboard, Top, and Charge Exchange Armor

The baseline design for the inboard and top armor consists of passively cooled graphite tiles attached to the torus chamber wall with graphite bolts (Fig. 4). Each tile is a square 15 cm on a side and 5 cm thick. The tiles are coated with titanium carbide to limit chemical erosion. Approximately 5300 tiles are required for the total inboard and top regions of the device. The graphite bolt attachment technique permits in-situ replacement of individual tiles.

Fig. 3. Spool structure.

Fig. 4. Wall armor tile.

The areas of significant charge exchange neutral impingement are shown in Fig. 1. Approximately 1000 tiles are required to cover the C-X impingement area. The design and attachment technique for the C-X armor is the same as that for the inboard and top armor.

For the baseline tile thickness of 5 cm, the inboard and top surfaces operate with maximum temperatures less than 1200°C during continuous operation. This is below the acetylene generation threshold (approximately 1200°C) and above the methane generation range, approximately 800°C. During startup, however, the tile surface temperature will operate in the methane generation range for approximately four consecutive burns.

Erosion caused by sublimation of the graphite tiles covering the top and majority of the inboard wall during plasma disruptions was determined. The design case was chosen as the erosion at nominal parameter values plus the root-sum-square of the maximum variation of erosion due to changing each parameter. This results in an erosion of 1.1 cm.

Erosion rates are higher where the charge exchange neutrals impinge (charge exchange neutral armor). The total erosion in these regions is 3.2 cm over the life of the device.

## Water-Cooled Outboard Wall Panels

To accommodate the surface heat loads during normal operation, water-cooled first wall panels are installed on the outboard vertical and 45° shield facets. These panels are fabricated of 316 stainless steel (Fig. 5). Each of the six smooth surface panels in each sector is approximately 2 m on a side. The panels are a multipass design, with the coolant flowing toroidally back and forth through the panels in a serpentine pattern.

Fig. 5. Outboard first wall panels.

The panels are installed from the sides of the shield sectors. Removal of the sector is required in order to replace a panel. Panel replacement should be infrequent, however, since the panel life is expected to exceed the life of the device. Tapered plates, spaced approximately 0.7 m apart, engage slotted fittings attached to the shield to provide structural attachments to react the disruption electromagnetic loads and allow for free in-plane thermal expansion. A gap of approximately 1.0 cm is provided at the sector midplane for thermal expansion. The plates also provide electrical contact between the panels and shield structure.

The design temperature for the stainless steel outboard panels is 300°C for the basic 8-T operation. The calculated elastic thermal stress is about 50 ksi for the design. A range of postulated plasma edge conditions was evaluated. The most severe surface heat flux of 30 W/cm$^2$ was adopted as the nominal design condition.

The outboard stainless steel wall panels on the vertical facet of the plasma chamber also serve as the start-up limiter and experience erosion by physical sputtering from charged particles during startup. The thickness eroded by charged particles is 0.14 cm for the full ten-year life. This erosion, added to the erosion by the ten disruptions assumed for design purposes, results in a total potential erosion of 0.34 cm for these panels. Therefore, the 0.8-cm-thick stainless steel first wall is expected to last the full life of the device.

The steady-state stainless steel panel temperature is 330°C for 10-T operation, which is 50°C higher than the expected value at 8 T and 30°C higher than the design temperature. While this will tend to increase cyclic thermal stresses, only a limited number of 10-T cycles is anticipated; and effects of irradiation creep will help reduce maximum thermal stresses.

## Mechanical Pumped Limiter

The pumped limiter forms a full toroidal belt at the bottom of the vacuum chamber (Fig. 1). The functions of the pumped limiter are to establish the plasma edge, pump helium ash and hydrogen particles, and protect first wall components from large particle and energy fluxes. Since the life of this component is likely to be very limited, it must be designed for easy replacement.

The limiter is divided into ten removable segments, one in each sector of the device. The design features independent replaceability of limiter segments without removing shield sectors. The two major components of the limiter are: a replaceable protective surface and an actively cooled stainless steel core (Fig. 6). The replaceable protective surface consists of graphite armor tiles attached with an actively cooled copper substrate at the upper and lower surfaces of the limiter blade. A brazing or bonding technique is utilized for these attachments to achieve good thermal contact between the graphite and copper. The limiter core, which is reusable, provides support for the replaceable protective surface and provides for structural attachment between the limiter and the shield.

Based on an upper temperature limit of 1200°C for the graphite surface, the maximum tile thickness was determined to be 1.25 cm. The copper substrate maximum temperature is maintained below 225°C.

The limiter leading edge is located such that 6% of the ions leaving the plasma enter the slot below the limiter. Particle trapping calculations indicate that the pumping efficiency is near unity for particles entering this slot.

Graphite tile erosion rates for four postulated plasma edge conditions are presented in Fig. 7. These plasma edge conditions are representative of the uncertainties in defining the plasma edge parameters. Erosion rates vary from 7.1 cm/year to 0.27 cm/year, which results in service lives that vary from approximately two months to four years, respectively. These results

Fig. 6. Limiter construction.

Fig. 7. Limiter erosion.

demonstrate that limiter performance is extremely sensitive to plasma edge conditions.

Table 1. Outboard Stainless Steel Panel Conditions During Normal Operation

| | 8 T | 10 T |
|---|---|---|
| Radiation from plasma - $W/cm^2$ | 18 | 22 |
| Nuclear heating in SS - $W/cm^3$ | 4.0 | 10 |
| Re-radiation from inboard armor - $W/cm^2$ | 12 | 15 |

## Conclusions

The uncertainties in loading conditions pose a serious problem in designing the plasma chamber first wall. Data on the distribution of the heat load resulting from disruptions is limited. Accordingly, the inboard, top, and bottom surfaces of the plasma chamber have been lined with 5-cm-thick graphite tiles which should be capable of withstanding a substantial number of disruptions. There is some concern, however, that chemical reactions may severely erode the graphite tile and pose problems in the ultimate disposition of the reaction products. If the disruptions are sufficiently benign and predictable, it may be possible to use a bare stainless steel wall. More data is needed on the characteristics of plasma disruptions and on the reaction of hydrogen with graphite over a wide range of thermal conditions.

The component of greatest concern with respect to service life is the pumped limiter. Depending upon the actual loading conditions, the graphite tile surfaces of the limiter modules will have to be replaced anywhere from once every two months to once every four years. This dramatically illustrates the need for better experimental data on the plasma edge conditions and on the reaction of hydrogen with graphite.

## References


[1] G. M. Fuller, B. A. Cramer, J. R. Haines, and J. P. Davisson (MDAC), Fusion Engineering Device (FED) First Wall, Armor, and Limiter Design Issues, to be published.

[2] J. Kirchner, (MDAC), The Torus Design for the Fusion Engineering Device (FED), to be published.

[3] B. A. Engholm, (GA), Status of FED Shielding Analysis and Design, to be published.

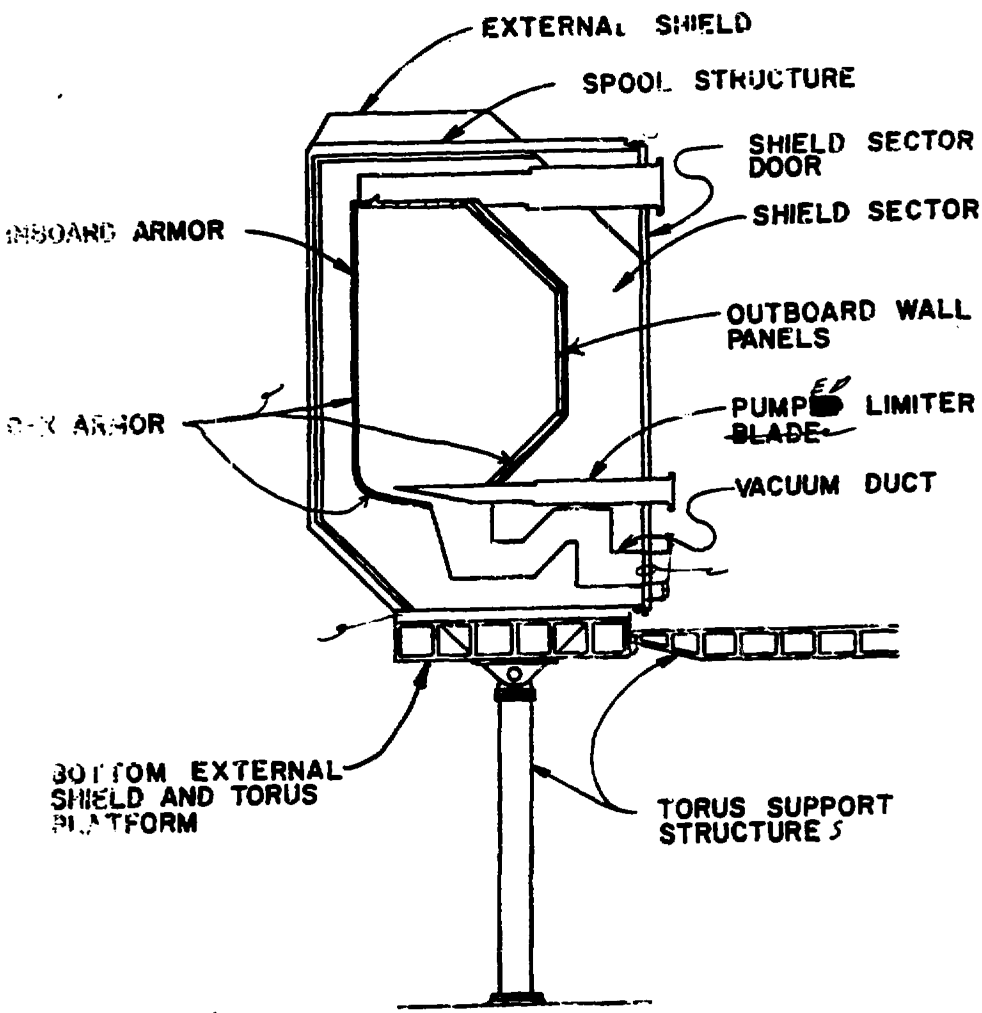



Fig. 1 FED Torus Components

~~Fig. 5-1~~

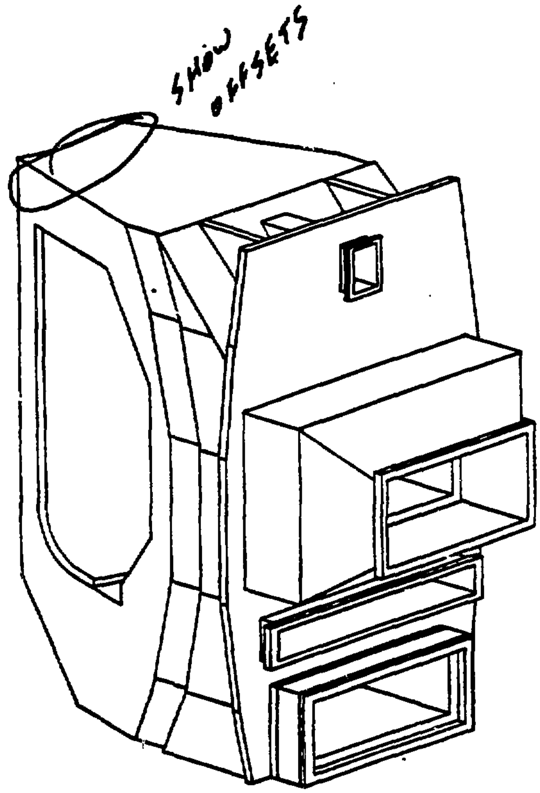
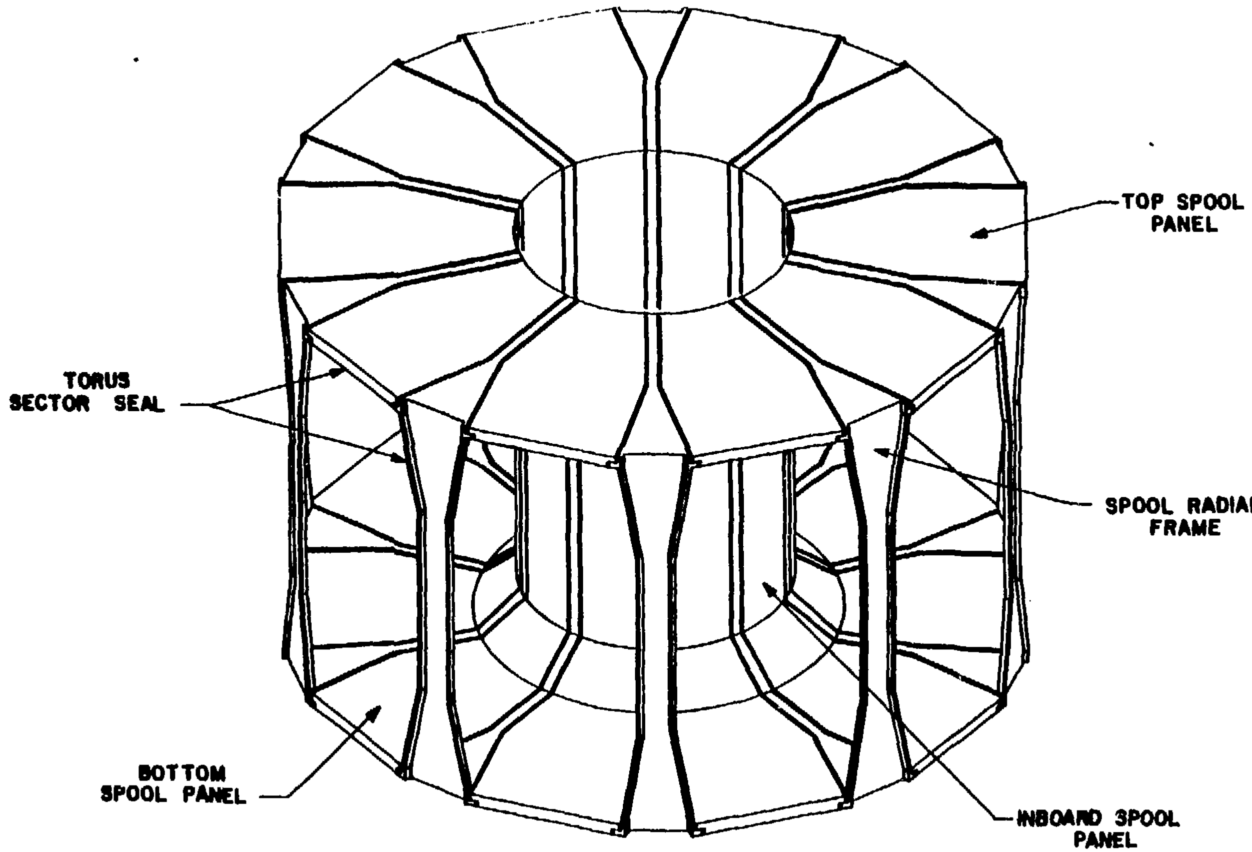
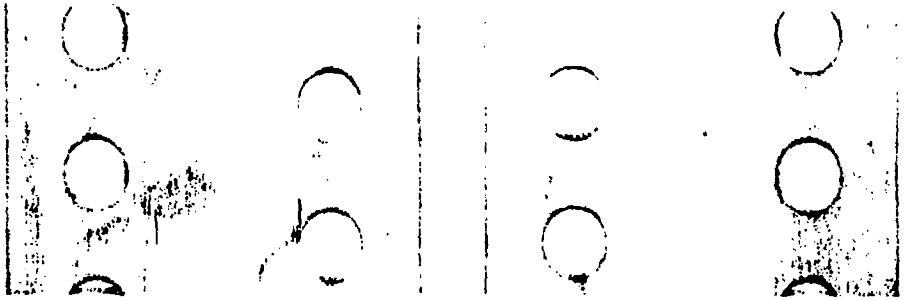



FIG 8Z Shield Sector




FIG#3 Spool Structure



GRAPHITE TILES ARE BASELINE DISRUPTION PROTECTION

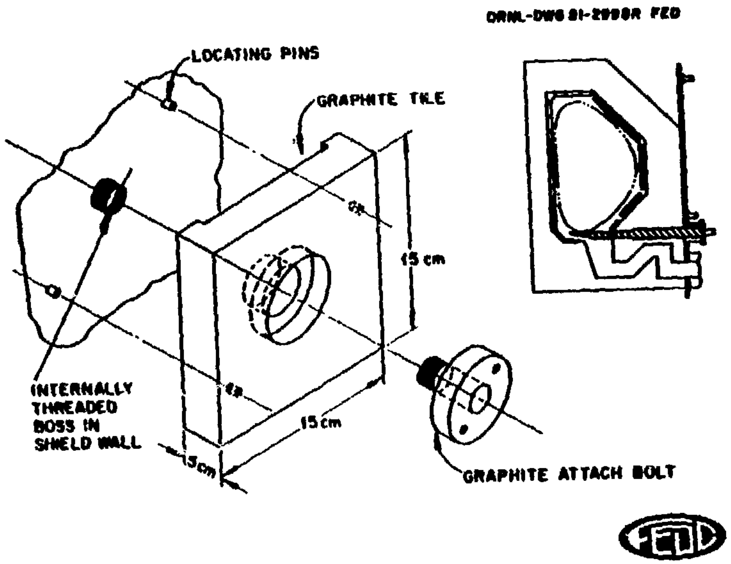
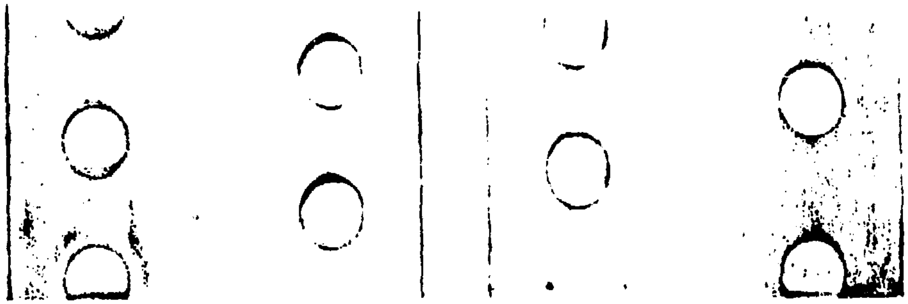



Fig. 4 Wall Armor Tile



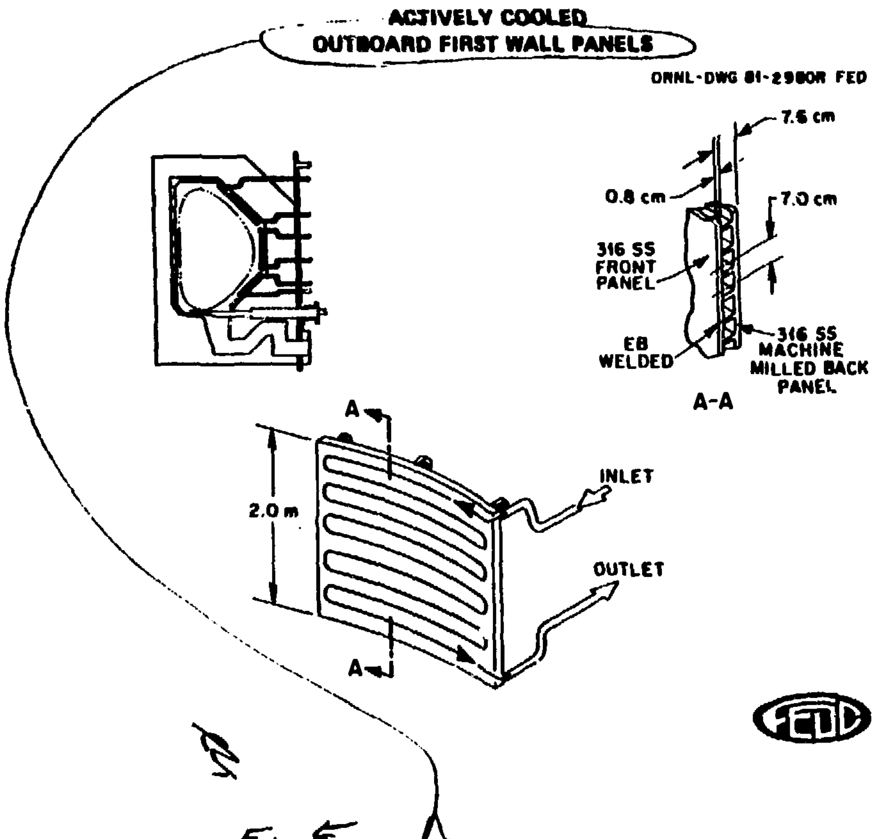

ACTIVELY COOLED
OUTBOARD FIRST WALL PANELS
ORNL-DWG 81-2980R FED
7.6 cm
0.8 cm
7.0 cm
316 SS
FRONT
PANEL
EB
WELDED
316 SS
MACHINE
MILLED BACK
PANEL
A-A
A
A
2.0 m
INLET
OUTLET
FED

Fig. 5

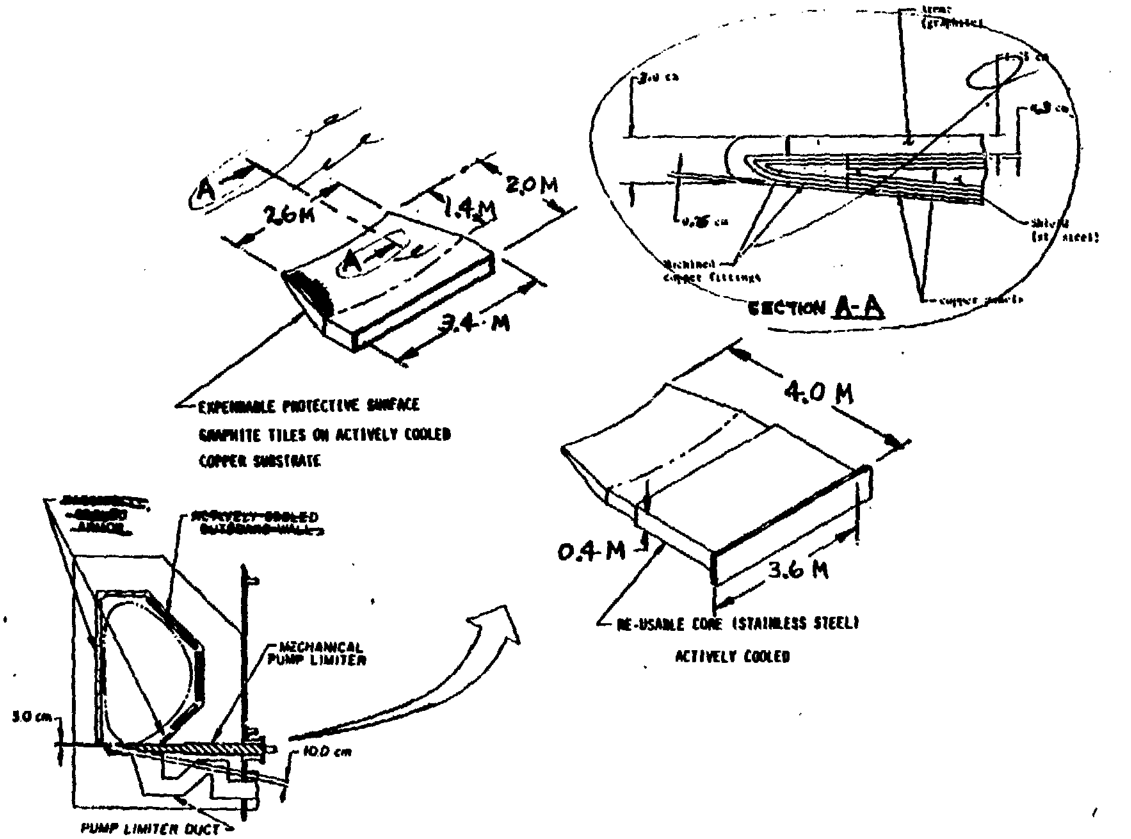
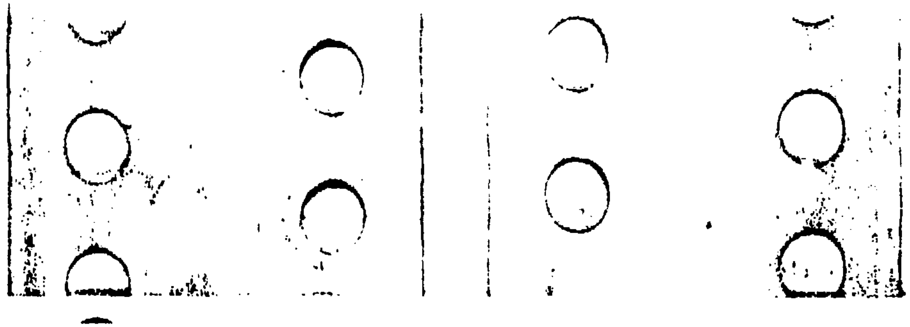



FIGURE 5.5.2-1 DIMENSIONAL CHARACTERISTICS & LIMITER CONSTRUCTION



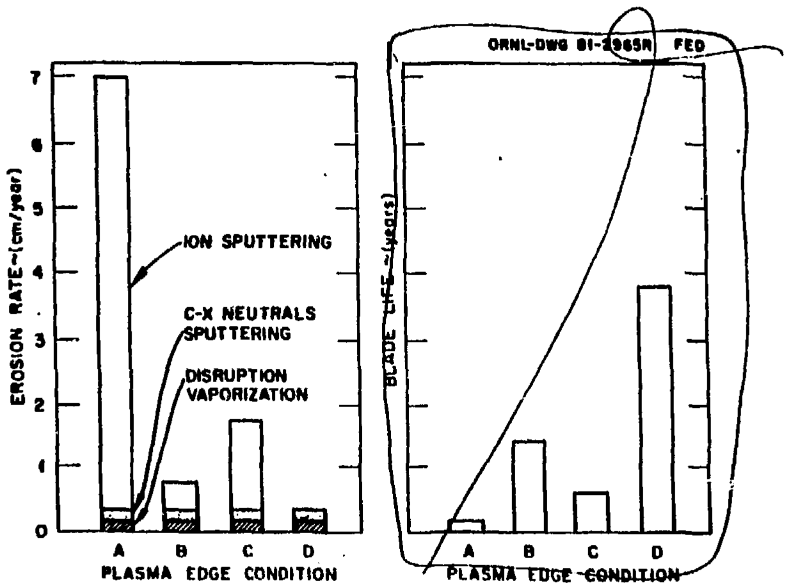



Fig. 7 Limiter Erosion