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# **ENGINEERING FEATURES OF AN UPGRADABLE IGNITION TEST REACTOR**

**by**

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**ENGINEERING FEATURES OF AN UPGRADABLE IGNITION TEST REACTOR\***  
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A conceptual design has been developed for an upgradable ignition test reactor (UITR) as part of the ERDA-sponsored TNS program to identify the next major tokamak device beyond Doublet III and TFTR. This reactor, which is based on the doublet plasma concept, has a major radius of 4.2 m, a minor radius of 1.2 m, and a plasma height-to-width ratio of 2.7. The plasma is contained in a water-cooled vacuum vessel fabricated of Inconel 625. A 25 cm thick helium-cooled blanket made of 316 stainless steel is installed outboard of the plasma chamber. The field-shaping coil system, shielding system, toroidal field coils, and induction coil system are installed outside the blanket. In its ITR mode, the reactor operates in moderate length pulses (30 sec) with a duty factor of 0.1 (5 min cycles). For net power experimental power reactor (EPR) operating mode, the machine operates with 90 sec pulses and a duty factor of 0.78. While the UITR initially costs about 30% more than an ITR without upgrade capability, and additional costs are incurred when implementing the upgrade, the total program costs far less than the sum of separate ITR and EPR facilities.

Introduction

Since the objectives of the next major magnetic confinement device beyond Doublet III and TFTR have not been established, it has been necessary to consider a range of possible devices. This range of devices accounts not only for different degrees of ambition and risk as implicit in defining the objectives, but also for differences in plasma physics results in forthcoming experiments and state of the art developments in the various technologies being pursued in the fusion program.

Previous studies<sup>1-3</sup> have indicated that the construction of an experimental power reactor designed to produce net electrical power involves a very high funding commitment and a high risk in terms of current knowledge of plasma behavior and state of the art for the technologies involved. Accordingly, less ambitious alternatives are being explored, including various concepts which are being designed to achieve ignition and provide sufficiently long burns to study plasma burn dynamics. One such possible design is the 3.8 m Ignition Test Reactor (ITR) currently under intensive study at General Atomic in the TNS Program sponsored by ERDA.<sup>4</sup>

Another option which has received considerable attention at General Atomic during the past year is an upgradable ignition test reactor (UITR). This machine is designed for initial operation as an ignition test reactor and ultimate operation as an experimental power reactor. The plasma chamber is sized so as to produce sufficient thermal power to achieve net power in the EPR mode of operation. Likewise, a blanket is included and sufficient shielding is provided to permit ultimate operation with a high temperature coolant at high duty factor. Initially, however, no power conversion system would be provided and only the power supplies and energy storage for operation in an ITR mode would be installed. All the heat produced would be rejected through cooling towers. After several years operation as a low duty factor ignition test

reactor, the power conversion system would be installed and the power supplies and energy storage systems would be upgraded for EPR operation at high duty factor.

Overall System Description

The reference UITR design is illustrated in Fig. 1. The reactor has a major radius of 4.2 meters and an aspect ratio of 3.5, resulting in a plasma half-width of 1.2 meters. The selection of the major radius was made on the basis of achieving a significant level of net power with realistic scaling assumptions. System analysis was performed in order to evaluate component geometry and space allocation variations within the given plasma physics and engineering constraints.

Table 1 gives the major design parameters for the UITR. Comparisons are also made in this table for the two different modes of operation (e.g. ITR and EPR).

Table 1  
UITR Major Design Parameters

Reactor major radius	4.2 m
Aspect ratio	3.5
Plasma height-to-width ratio	2.7
Toroidal maximum magnetic field	8 tesla
Toroidal magnetic field on axis	3.8 tesla
Plasma current	11.7 MA
Plasma average beta	0.10
Plasma thermal power	808 MW(t)
Neutron wall loading peak	1.1 MW/m <sup>2</sup>
Burn time	
ITR	30 seconds
EPR	90 seconds
Dwell period	
ITR	270 seconds
EPR	26 seconds
Duty factor	
ITR	0.10
EPR	0.78
Blanket average power	250 MW(t)
Gross electric power	103 MW(e)
Plant auxiliary power	88 MW(e)
Plant net electrical power	15 MW(e)
Blanket thickness	25 cm
Inboard shield thickness	57 cm

The plasma chamber for the UITR is a continuous toroidal shell structure of Inconel 625. Pressurized water (2.5 MPa) is used as coolant to maintain the material temperatures in the chamber wall at low temperature. The coolant water enters at 150°C and exits the chamber at 200°C absorbing approximately 215 MW(t). A larger portion (65%) of this low grade heat is utilized by the power conversion cycle in the form of feedwater heating, thereby enhancing the power cycle performance.

The reference design has a 25 cm stainless steel blanket cooled by helium. The blanket surrounds a major portion of the plasma chamber. The blanket has the capability of generating high grade heat and sustained operation at a high duty factor. The hot helium from the blanket is routed to a steam generator which drives a Rankine steam power conversion cycle system with turbine inlet steam conditions of

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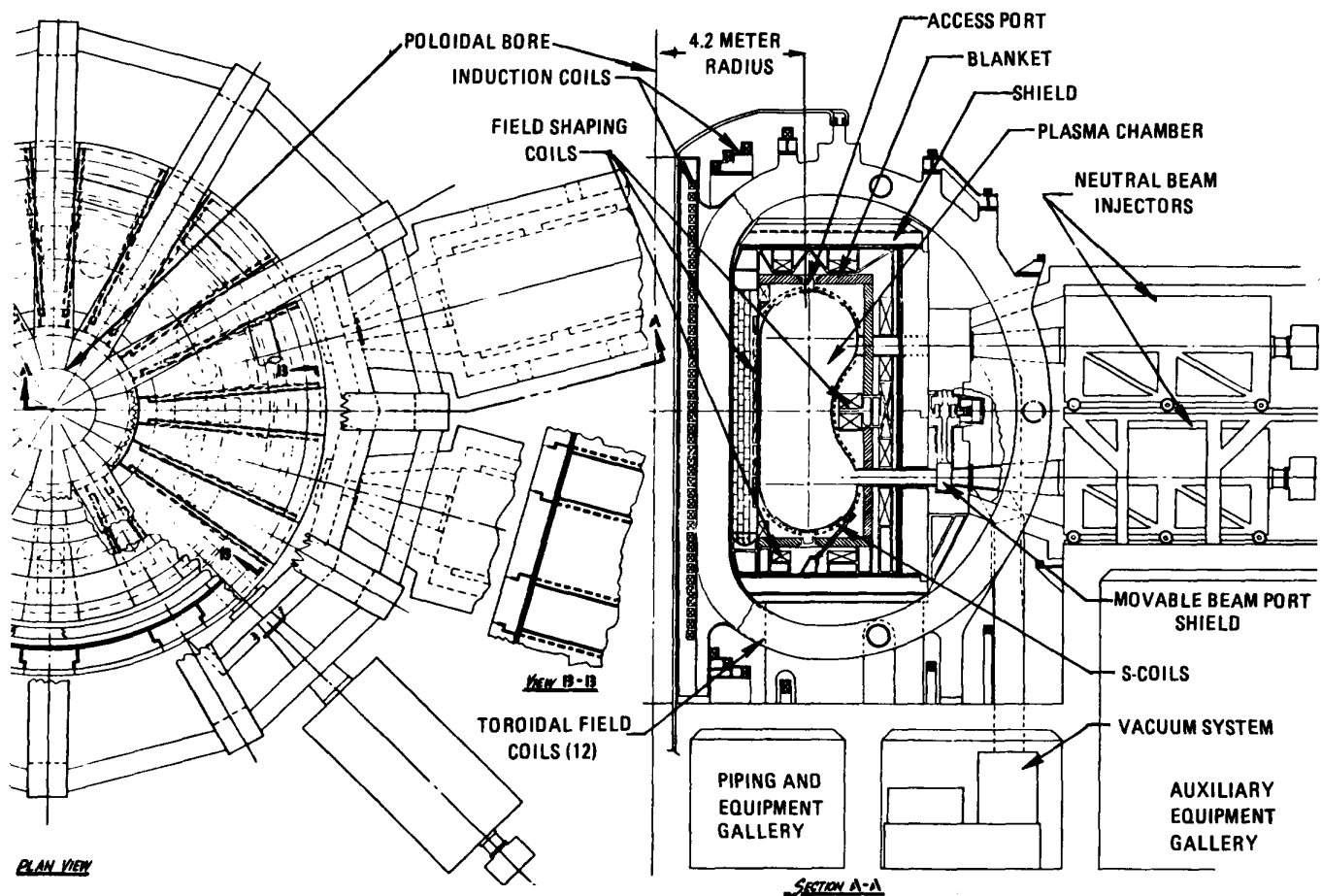


Fig. 1. Upgradable Ignition Test Reactor (UITR)

450°C/12.4 MPa. Operation at a duty factor of 0.78 results in a blanket power level of 250 MW(t) and a gross electrical power output of 103 MW(e). Allowing for a total parasitic power of 88 MW(e), the levelized net plant output is 15 MW(e).

Pressurized water could also be used as the cooling medium, but the choice of helium results in a substantially higher plant efficiency and therefore a smaller and lower cost reactor.

Located outside the plasma chamber and blanket are the coolant manifolds, the field-shaping coils, the shielding and the toroidal field coils (B-coils). Inside the poloidal bore is the induction coil (E-coil) system. There are 23 normal conducting copper F-coils each with 20 turns.

Part of the field-shaping coil system are the S-coils which are located in close proximity to the plasma chamber. There are 16 three-turn coils attached around the outside portion of the vacuum vessel.

The inboard shield (located in the region closest to the poloidal axis) is made up of tungsten bricks encased in a large stainless steel tank. Cooling is accomplished by gross water flow up through the brick coolant holes in the tank. This shield limits the nuclear heat deposition in the toroidal field coils to approximately 1000 KJ. The toroidal field coil system consists of 12 NbTi superconducting constant tension coils cooled to 4.2 K by liquid helium.

The toroidal field coil is sized for 10 tesla, in order to provide design margin for ignition. For normal operation at 8 tesla, an on-axis toroidal field of 3.8 tesla is achieved. A plasma average  $\beta$  of 0.10

and a plasma current of 11.7 MA results in a plasma power of 808 MW(t). A nominal burn time of 90 seconds and a dwell of 26 seconds between burns results in a duty factor of 0.78 for the EPR mode of operation. During initial operation as an ITR the nominal burn time of 30 seconds and a dwell of 270 seconds results in a duty factor of 0.10.

Design concepts have been developed for access to the regions within the shield envelope using locally controlled remote handling machines. Sufficient shielding is provided on the reactor to permit contact maintenance outside the shield envelope.

Twelve neutral beam injectors with a rating of 150 keV are located around the periphery of the reactor. The injectors have a total output power of 56 MW in beam power during the beginning 2 seconds of each burn cycle to raise the temperature of the plasma by injecting high energy deuterium atoms into it. During the dwell time between burns the neutralized gas in the plasma chamber is removed by cryosorption pumps installed below the reactor. The vacuum pump ducts from the plasma chamber are branched off the neutral beam ports. Tritium carried away by the vacuum chamber pumps is processed by a tritium handling system.

Power supply systems are required by the neutral beam injectors as well as the B-coils, E-coils, F-coils, and S-coils. Major energy storage units are also required for the E-coils and neutral beam injectors.

#### Plasma Chamber

The plasma chamber for the UITR is a continuous toroidal shell of Inconel 625 with a major radius of

4.2 m, a minor radius of 1.25 m and a height of 6.58 m. The basic membrane wall thickness varies from 0.75 cm at the inboard wall to 1.40 cm at the outboard wall. The cross section of the chamber showing the characteristic doublet shape is shown in Fig. 1. The convoluted contour was chosen over truncated sections in order to minimize the distance between the F-coils and the plasma surface. At each 30° around the top is an access port located on the doublet centerline. This port is covered with a removable section that permits access to the inside of the chamber.

The chamber wall, which is water cooled, is made up of two layers of metal, a thick section on the inside containing grooves that form coolant channels and a thin sheet on the outside to seal the open side of these grooves. A typical cross section is shown in Fig. 2. All coolant channels formed by the grooves are oriented in the poloidal direction around the vessel, terminating in supply and outlet manifolds located at the top and bottom. Each 30° section has separate manifolds with the coolant lines from them taken through penetrations in the lower shield.

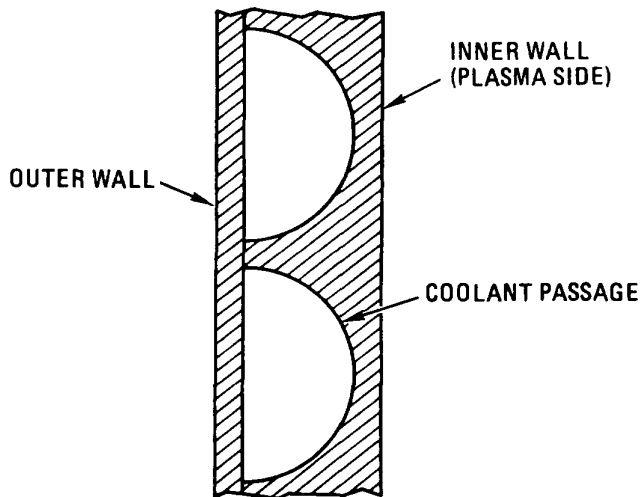


Fig. 2. Plasma chamber wall section

The vessel wall thickness varies around the poloidal direction with the thickest section in the outboard midplane region in order to withstand the plasma disruption loads imposed.<sup>5</sup> Around each penetration through the vessel wall additional material thickness must be added. This additional material must be cooled since it is subjected to the same high wall loading as the remainder of the vessel.

The chamber is supported by 12 sets of posts located every 30°. These posts are pinned at each end to allow for unrestrained thermal expansion of the chamber. The unusual angle they assume permits the thermal expansion to be downward as well as upward, keeping the midplane in the same relative position. Coolant is circulated up through the post to the closed end and returned. Brackets that carry the support loads into the vessel are mounted on raised pads which distribute the loads on the vessel.

Inside the chamber, the plasma surface is defined by 24 limiters, 12 located at the top and 12 at the bottom, in line with the doublet centerline. The contoured shape of the limiters covers about 90° of the

poloidal span at the top and bottom of the chamber. Perpendicular to this plane along the toroidal direction, each limiter is 70 cm in length, which makes it possible to remove them through the chamber access port. A drive mechanism located outside the chamber activates a drive shaft which also carries coolant to the highly cooled limiter. The limiter is constructed of beryllium, the atomic number of which is sufficiently low to make limited sputtering acceptable to the plasma.

Tiles made from graphite are attached to the inside surface of the vessel in the inboard region. Plasma disruptions which are expected to be directed primarily toward the inboard surface will damage the tiles instead of the wall. Beneath the tiles a carbon coating is applied to the entire chamber inner surface to prevent sputtering from the particles that get through the tiles and strike the wall.

The chamber is assembled by welding it in place around the center cluster of toroidal field coils and inboard shield from twelve individual sectors of 30° each. Extensive bracing and fixturing is used to ensure alignment.

### Blanket

A 25-cm thick stainless steel helium coolant blanket is provided to extract the plasma thermal power transported by the high energy neutrons. This blanket is placed outside the plasma chamber and inside the shield (Fig. 1).

The blanket was omitted in the inboard region. The blanket in this region would increase the radial thickness of components inboard of the plasma chamber and thereby increase the major radius and reduce toroidal field on axis. This would result in an increase in the size of the reactor and a decrease in thermal power. Also, in order to permit placement of the outboard mid-plane field-shaping coils as close as possible to the plasma, the blanket was eliminated at this location. Even so, the blanket intercepts about 73% of the neutron flux from the plasma.

The blanket assembly is comprised of a total of 96 segments, twenty-four 15-degree flat segments on each the top and bottom and twenty-four 15-degree cylindrical segments each above and below the midplane in the outboard region.

The modules are made up of stacks of stainless steel plates into which grooves are machined and coolant tubes are placed (Fig. 3). Coolant is supplied to a manifold fabricated onto the outside surface of the module along one edge. The flow is distributed to the coolant tubes which conduct the coolant to the inner face of the blanket module and then back through the module in serpentine fashion to an outlet manifold on the opposite side of the module.

The helium coolant, which is pressurized at 50 atmospheres, enters the module at 300°C and exits at 507°C. The maximum blanket temperature at these conditions is 530°C. In order to minimize thermal stresses, the first four rows of coolant tubes in the inside region are placed at a 2.5 cm series pitch. Further out, the pitch is increased to 5 cm.

### Shielding

The shield design involves the consideration of two separate regions: the inboard shield, i.e. that shield between the plasma and the poloidal bore, and the outboard shield, i.e. that shield above, below

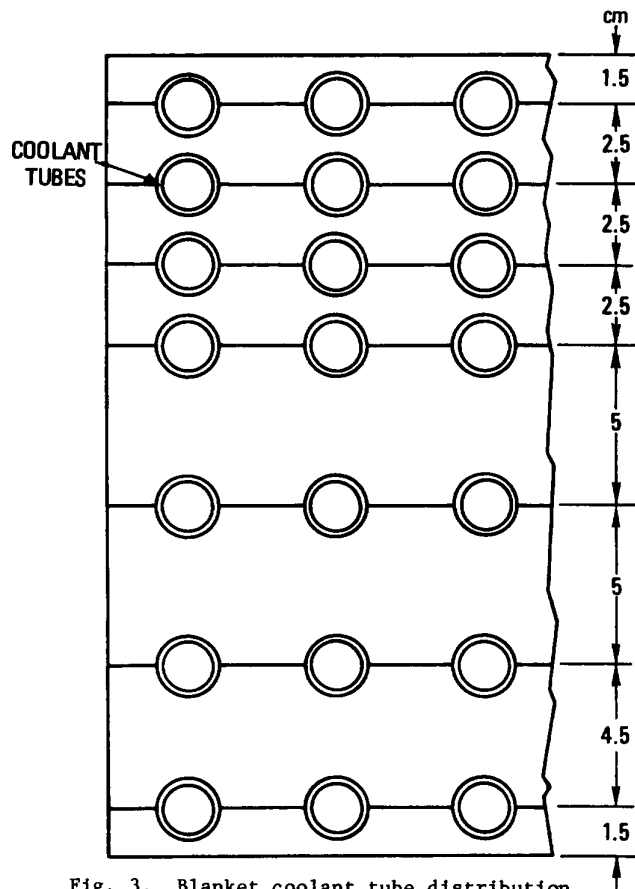


Fig. 3. Blanket coolant tube distribution

and radially outward of the plasma chamber. These two shielding regions are distinctly different in terms of attenuation criterion, spatial limitations, activation requirements, structural responsibility and mechanical design requirements pertaining to disassembly.

#### Inboard Shield

The inboard shield is designed to limit the nuclear energy deposited to the superconducting toroidal field coils (B-coils) to 1 MJ over the burn.

The inboard shield consists of water-cooled tungsten bricks stacked in a tank which also accommodates the F-coils (Fig. 1). The tungsten Heavymet bricks ( $\sim 95\%$  W) have coolant channels occupying approximately 40% of the volume of the brick. The bricks are stacked, with the coolant channels roughly aligned and held rigid by simple aligning rods. Demineralized water flowing through the bricks and across the face of the F-coils in a bath flow mode provides adequate cooling and combines with the tungsten and copper to form an extremely efficient shield medium. The tank walls are 2 cm thick and are made of stainless steel. These walls form two concentric steel cylinders that easily accommodate the loads applied by the upper section of the outboard shield. Access to the F-coils is obtained by draining the water and pulling the bricks out of the tank.

The inboard shield concept proves to be cost effective despite the high cost of tungsten (\$44/Kg as a brick). This is possible because the reduction of 15 cm in the thickness of the inboard region, gained by the tungsten tank concept, accrues a savings in overall reactor costs that far exceeds the additional cost of employing tungsten (i.e. as compared to 316SS, etc.).

#### Outboard Shield

The outboard shield's basic nuclear function is to limit the shutdown dose rate at the reactor periphery

to levels commensurate with hands-on maintenance at one day after shutdown. In satisfying this criterion, the shield far exceeds the attenuation levels required for magnet protection criteria.

The outboard shield is designed with large top and vertical segments (Fig. 1). The top and vertical segments consist of two distinct zones. A thin, 10 cm, thermal shield utilizing a bath flow of water between steel plates is positioned nearest the plasma. This is followed by a 70 cm zone consisting of steel support structure, coolant lines and tanks of boronated water embedded within a lead casting. The top and vertical shields are segmented into 24 sections. The  $20^\circ$  segments between adjacent toroidal field coils are designed for ready disassembly to provide a route for access to the reactor interior. The bottom shield unit is constructed as a steel monolith uniformly cored with coolant passages forming a composition of 70% steel + 30%  $H_2O$  and is assumed to be a permanent part of the structure.

In addition to the bulk shielding of the outboard region, special shielding units are required for the neutral beam ports and the injectors themselves. As depicted in Fig. 1, the beam ports are surrounded by approximately 50 cm of shielding. This shield is made up of a 35 cm, water cooled, steel section jacketed by a 15 cm zone of lead. The total nuclear heating and induced activation of the injector is inhibited by the use of the movable shield plugs affixed to the exterior of the bulk outboard shield. These 50 cm thick steel plugs are rotated into the beam port once ignition of the plasma is accomplished. They reduce the integrated load to the injector by roughly an order of magnitude. The dose to the injector housing, however, is still sufficient to create an activity problem. Therefore, the injectors are encased in a housing of steel and lead on the order of 30 cm thick.

#### Magnet Systems

##### Toroidal Field Coils

The twelve constant tension shaped toroidal field coils employ NbTi/Cu superconductor and while the normal maximum operating field is 8 tesla, they are designed to generate a peak field as high as 10 tesla (5 tesla on the plasma axis). The cryogenically stable coils are bath cooled with liquid helium at 1 atm. and 4.2 K.

A half cross section of the coil/helium vessel in the centerpost region is shown in Fig. 4. The coils are pancake (spiral) wound directly onto the weldment formed by the minimum perimeter wall, and central radial flange of the stainless steel helium vessel. The conductor of each pancake is graded on the basis of magnetic field and radial bearing load. Radial (hoop load) support is provided by pretensioned stainless steel strips interwound with the conductor. The average conductor current density is  $2300 \text{ amp/cm}^2$ , and the overall helium vessel packing factor is 0.5. Conductor current is 10 KA; total stored energy is 13.4 GJ.

##### Ohmic Heating (E-) Coils

The E-coil system supplies the poloidal magnetic flux change (volt-seconds) through the tokamak bore which induces and sustains the toroidal plasma current. The general arrangement of the E-coils is shown in Fig. 1. The E-coils form a part of the poloidal coil circuit, shown schematically in Fig. 5.

The six key features of the E-coil system are:

1. Superconducting coils
2. Parallel coil connection
3. Energy storage using a controlled homopolar generator
4. Half biased flux swing

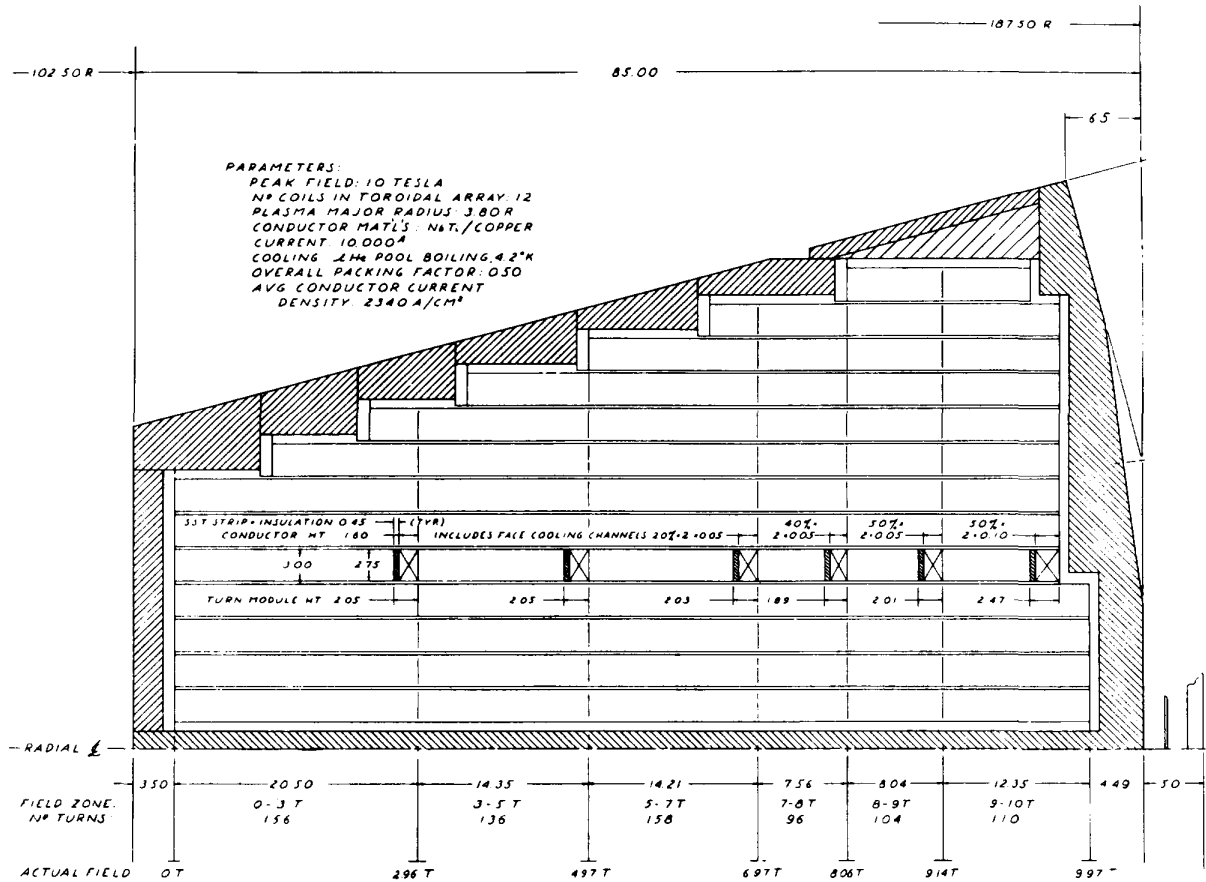
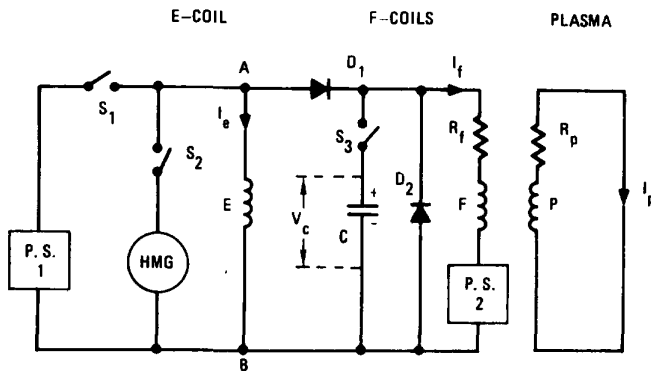


Fig. 4. Toroidal field coil



- P. S. 1 = E-COIL POWER SUPPLY
- P. S. 2 = F-COILS POWER SUPPLIES
- HMG = CONTROLLABLE HOMOPOLAR MOTOR GENERATOR
- S<sub>1</sub>, S<sub>2</sub>, S<sub>3</sub> = SWITCHES
- E = E-COIL
- F = F-COILS
- P = PLASMA
- C = STARTUP CAPACITOR
- R<sub>f</sub> = F-COILS EQUIVALENT RESISTANCE
- R<sub>p</sub> = PLASMA RESISTANCE
- D<sub>1</sub> = BLOCKING DIODE
- D<sub>2</sub> = STARTUP DIODE

Fig. 5. Poloidal coils circuit

5. E-coils entirely outside of toroidal field coil envelope
6. Inner, solenoidal E-coils imbedded in the toroidal field coil centerpost support cylinder.

The E-coils, presently under design by Argonne National Laboratory, utilize cables NbTi/Cu conductor, cryogenically stabilized by bath cooling. Peak current is 50 KA in the solenoid region.

#### Field Shaping (F-) Coils

Figure 1 shows the disposition of the inner and outer F-coils about the plasma chamber.

The primary role of the F-coils is to shape the plasma into the doublet configuration. Partial shaping is achieved passively, by reflecting current induced in the F-coils by the plasma. However, the F-coils are also driven by power supplies in order to overcome resistive losses, and to provide additional shaping, particularly in the center plane (equatorial) region.

For effective economical operation, the F-coils must be close to the plasma boundary. This precludes the use of shielded superconducting coils for ignition machines.

The six inboard F-coils are located within an annular water jacket which also contains the inner shielding. Bulk water flow through the tank eliminates the requirement for forced flow internal cooling of the inboard F-coils, thus affecting a net saving in coolant complexity. Demineralized water has a dielectric

strength of about 10 kV/cm, and will not be seriously degraded by the neutron flux. Since during startup each 20 turn F-coil will experience a voltage of about 8 KV, a nominal coil to ground standoff distance of 2 cm is maintained. The conductor is OFHC copper, and all insulation is alumina.

The massive outboard F-coils are paired in series, symmetrically above and below the midplane, in order to half the number of turns required. In this way they act inductively as a single 20 turn coil, thereby allowing sufficient turn thickness (4 cm) to permit radially bolted lap joints. Cooling is provided by internal water channels, jumpered across each demountable joint.

For net power devices, the heat absorbing blanket must be placed between the plasma chamber and the massive outer F-coils, which would otherwise intercept much of the neutron flux. Because of this, F-coils are supplemented by low power and relatively thin "S"-coils, disposed closely around the chamber (inside the blanket). In addition to augmenting the plasma shaping of the basic outer F-coils, the S-coils provide the requisite plasma stabilization (hence "S-" = stabilizing coils). These coils are not parallel connected into the basic poloidal circuit, but are connected in parallel together, each with its own trimming supply.

#### Reactor Auxiliary Systems

##### Plasma Heating

Neutral beams based on positive deuterium ion sources are used for plasma heating during the start of each cycle. Three sets of injectors, each set with one injector for the upper lobe of the doublet and one for the lower lobe, are located on either side of the reactor providing a total of 12 injectors. The injectors provide 56 MW of monoenergetic 150 keV D<sup>0</sup> beam for about 2 seconds.

Each beam line has a positive ion source and accelerator, a double focusing bending magnet, a neutralizer for the atomic ions, and a second bending magnet to remove atomic ions from the beam emerging from the neutralizer. Direct energy converters are used to recover energy from the molecular and atomic ions removed from the first and second separator magnets.

Zirconium-aluminum panels are used for neutral beam injector vacuum pumping. In order to maintain a high pumping speed for hydrogen, the panels must be operated at temperatures of 300°C to 400°C. This requires external power requirements, but electrical power demand is somewhat alleviated by the thermal loading from the reactor neutron flux.

##### Plasma Chamber Vacuum Pumping

The plasma chamber vacuum pumping system parameters are presented in Table 2. Cryocondensation pumps are used for pumping hydrogen isotopes and gaseous impurities, while gas bearing turbomolecular pumps are installed downstream of the cryocondensation pumps to remove helium.

A total of 24 cryocondensation pumps are used — two connected to each of 12 pumping ducts — so that one set of 12 can be regenerated while the other set is on line. The cryopumps are connected to the plasma chamber by means of 70 cm ducts which branch off the neutral beam injector ducts just outside the shield. It probably will be necessary to install

Table 2  
Plasma Chamber Vacuum Pumping System

Number of ducts	12
Duct diameter	70 cm
Duct length	8 m
Number of pumps, total	24
Required pump speed, each	31 m <sup>3</sup> /sec
Nominal pump speed, each	50 m <sup>3</sup> /sec
Post-burn gas pressure	9.3 × 10 <sup>-1</sup> Pa
Pre-burn pressure	4 × 10 <sup>3</sup> Pa
Regeneration time	4 hr

debris separators in these pumping ducts to prevent the transport of radioactive particulates from the plasma chamber into the pumping system. Isolation valves, capable of maintaining a hard vacuum are also required to permit regeneration of individual vacuum pumps during reactor operation.

##### Tritium Handling System

The tritium handling system is similar to the system used for the 3.8 m ITR.<sup>6</sup> A turbomolecular transition pump, a series of bellows pumps, and a diaphragm compressor compress the vaporized gas components from the cryocondensation pumps being regenerated. The gas stream is then passed to a hydrocarbon cracking cascade where the hydrocarbons are broken down. From the cascade, the stream passes to a hot getter bed where all chemically active elements except hydrogen are absorbed. Thereafter, argon and helium are removed by a pair of cryogenic stripping columns.

The deuterium, tritium, protium stream is then fed to a liquefier and thence to a cryogenic distillation cascade comprised of three to four columns with about 100 stages per column. This cascade produces deuterium for neutral beam recycle, tritium-deuterium for cold fuel supply, and protium-deuterium waste.

##### Maintenance

Basically the maintenance philosophy for this reactor design is to perform all possible operations exterior to the reactor shield in a contact or hands-on mode and perform those operations on the interior side of the shield (including shield removal) in a remote manner.

The shield design does not allow operating personnel to be in the reactor building while the reactor is in operation, but it does allow access to the reactor exterior 24 hours after reactor shutdown. In addition, the shield is sufficient to limit the activation of the B-coils to a safe operating limit and the shield itself also employs low activity materials at its exterior to prevent high level exposure.

##### Contact Maintenance

Typical of those components to be maintained using contact maintenance are the overhead E-coils which are wholly outside of the shield. Hands-on inspection and maintenance of the associated cryogenic, electrical, and vacuum connections and structural supports are readily conducted.

The outboard portions of the B-coils that lie outside of the shield also can be maintained manually, similar to the E-coils. The poloidal portions of these coils, however, are not available for contact maintenance.

A major concern is the maintenance of diagnostics and controls. At least major elements of this equipment can be located outside the shield envelope, and they can be repaired and adjusted in a contact mode.

The reactor shield consists of numerous massive interlocking lead/steel modules. Each module is cooled from a manifolded water coolant system installed on the exterior face of the shield. Contact maintenance will consist of manual adjustment/replacement of valves and flow controls as well as disconnection operations of the coolant lines prior to shield module removal. Lift fixtures may be manually attached to the first module but once the shield is removed and the highly radioactive interior face of the shield is exposed, the operation shifts to a remote operation including storage of the modules and disengagement of the lift fixture.

Since the shield is the final radiation barrier that protects personnel and components within the reactor building, all services to the reactor components within the shield must by necessity pass through the shield modules. These service lines include the neutral beam injector ducts, diagnostics, electrical power to the F and S-coils, coolant water to the vessel and helium to remove heat from the blanket modules. Many mechanical connections for these services must be provided on the exterior side of the shield and can be manually operated and maintained. The portions of these lines inside of the shield that have mechanical connectors must be remotely disconnected to allow removal of other associated reactor components.

#### Remote Maintenance

All reactor components located within the reactor shield will require remote handling techniques and equipment to inspect and/or maintain these items after the limited operation has activated these components. Components inside the shield such as the vessel, diagnostics, blanket, F-coils, S-coils, piping, structural supports and neutral beam injector ducts will progressively become very radioactive and will be subject to radiation damage.

Specialized remote handling equipment to perform these remote operations will be required and will include facility systems such as a remotely operated 150-ton bridge crane, a shielded manned overhead electro-mechanical manipulator and a shielded manned vehicular manipulator unit. These all-purpose type systems working in conjunction with each other and using special remote handling equipment will be able to remotely maintain all of the reactor components as required.

Some of the remote specialized equipment that will be required are:

- In-chamber remote maintenance machine
- Optical viewing equipment including TV and periscopes
- Welding and cutting equipment
- Bolting equipment
- Lift fixtures
- Leak detection equipment
- Manipulators and hoists
- Portable shielding
- Transfer casks

In addition to the reactor components located within the reactor shield, the neutral beam injectors (located outside of the shield) also will require remote maintenance. The interior of the injectors and connecting ducts to the vessel will quickly become activated and will require their own neutron and gamma shielding to protect personnel within the reactor building. Replacement of the injector ion source may be accomplished with the injector in-place using portable shielding, transfer casks and remote tools to

move the source to a facility hot cell. Further repair to the neutral beam injector assemblies can be accomplished by disconnecting all utility services to these shielded injectors and using air bearings transfer the 600 ton units to a disassembly area for final repair.

Failed reactor components such as blanket modules, F- and S-coils or vessel limiters will be designed to be removed using remote equipment and the facility manipulator systems. As an example, the F-coils will be designed to disassemble easily into 60° sectors before removal from the reactor through the B-coils. Once removed, these sectors would be transferred to the facility hot cells for examination and repair.

#### Conclusions

It is concluded that the design and construction of an upgradable ignition test reactor is a viable option. In order to ultimately produce net power, it is necessary to include a high temperature blanket and make the reactor somewhat larger than a reactor designed for ignition only (4.2 m versus 3.8 m major radius). As a result, the facility has to be somewhat larger as must be the auxiliary systems and balance-of-plant.

Due to the larger size, the UITR would cost somewhat more (~ 30%) than a non-upgradable ITR. In addition, another increment of funding would be required to implement the upgrading. These extra costs, however, would be more than offset by the program savings in the elimination of the high costs of a separate second device (EPR). Also, the overall program could be substantially accelerated by upgrading the UITR as compared to constructing an ITR and an EPR in series.

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