A PRIMARY ENERGY CONVERSION SYSTEM FOR THE TOKAMAK EXPERIMENTAL POWER REACTOR

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Abstract

Major efforts are underway at ANL[1] and elsewhere[2,3] to define the objectives of a Tokamak Experimental Power Reactor (TEPR). The ultimate goal of these studies is to establish the scientific and engineering basis for a detailed reactor design. This paper will concentrate on the TEPR primary energy conversion system (PECS) as developed in the ANL study. The PECS includes all components that lie between the plasma and the toroidal-field coil, vis, (1) the first-wall assembly; (2) the blanket region, which comprises the 30 to 40 cm zone immediately backing the first wall (99 percent of the thermal energy is generated in the first wall and the blanket); (3) the primary coolant that removes all of the sensible heat from the first wall and blanket, together with its associated circulation and processing equipment; (4) the magnet shield, which lies between the blanket region and the toroidal-field coils and provides the attenuation of neutron and gamma radiation required to protect the magnet systems; and (5) the penetrations that provide access for the neutral beams, vacuum system, diagnostics, fuel injector, etc.

The efforts of the primary energy conversion system studies centered around concepts that would:

1. Provide for removal of sensible heat at temperatures that would allow the generation of 5 to 50 MW of electrical power.
2. Provide adequate protection of the superconducting toroidal field coil from radiation damage, and excessive nuclear energy deposition.
3. Allow for continuous operation of reactor-level plasmas at a plant factor of $\lesssim 50$ percent and an instantaneous wall loading of ~ 0.2 MW/m^2.
4. Demonstration of the principal features of a reactor PECS in the initial stages of TEPR operation. (Experimentation with tritium breeding and breeder blanket performance is provided for during the second stage operation.)

This paper will summarize the results of the studies which led to the current ANL PECS design concepts. In particular, aspects of the neutronics, materials, thermal-hydraulics and mechanical design studies will be presented.

Introduction

A major purpose of the EPR study is to focus and refine the objectives of a scientifically and technologically feasible device that satisfies the requirements of the U.S. Tokamak Fusion Power Reactor Development Program.[4] Such a device should represent the logical next step beyond the TFTR[5] in the achievement of D-T plasma confinement adequate for power reactors. The TEPR should be a major step in the demonstration of engineering feasibility of Tokamak power reactors. It should provide experience with and demonstrate the technologies needed to synthesize a power reactor, thereby serving as a focal point for the technology development program. It is intended to demonstrate electrical power production, if possible, to the reactor electrical power breakeven point. The TEPR should serve as a major test facility for the technology development program. These general objectives must be reconciled among themselves, with the state of the art and required developmental programs for the various technologies, with the DCTR requirement for initial operation in 1985-87, and with a yet unspecified fiscal constraint.

The TEPR should be the logical step beyond the TFTR (initial operation 1980) along the path to achievement of plasma confinement that is adequate for a power reactor. Achievement of sufficient pulse length and pulse repetition rate so that the cycle-averaged thermonuclear power, if converted to electrical power, would be in the range 5 to 50 MWe is a major goal. Conversion of this energy into sensible heat in an energy-conversion blanket is a second major goal. If these two goals could be achieved, conversion of the thermal energy in the primary coolant into electrical energy in a secondary system could be accomplished straightforwardly. These two objectives are sufficiently important in themselves to provide strong motivation to minimize any other factors which might complicate or compromise their attainment. This consideration argues for a "minimum-risk" approach that utilizes, to the maximum extent possible, technology that is either "state of the art" or is extrapolated therefrom with reasonable confidence.

The minimum-risk approach is contrary to many of the general objectives envisioned for the TEPR. The specific technological choices (e.g. materials) might not be extrapolatable to a commercial power reactor, or at best might not represent optimal choices for the latter. Certain functions of a commercial reactor (e.g. tritium breeding) are not essential to the plasma performance and sensible heat production objectives for the TEPR; therefore such "extraneous" complications would certainly be eliminated in a minimum-risk design. A minimum-risk design does not provide a focus for a broad-base reactor technology development program, nor does it lead to a demonstration that the technology can be sufficiently developed so that a full-scale fusion power reactor is technologically feasible. Thus, there are strong arguments against the minimum-risk, or minimum technology development, approach and in favor of utilizing the TEPR as a vehicle for advancing the technologies which will be needed for fusion power reactors. The pros and cons of the minimum-risk approach must be carefully weighed in the initial design iterations.

Staged operation is proposed as one means of reconciling these seemingly conflicting requirements. A minimum-risk approach would be followed in designing a device to achieve the plasma physics objectives, with an energy conversion blanket (no tritium breeding) to achieve the sensible heat-power objectives. The device would be operated during Stage I (approximately 1986-89) to obtain these objectives and, in addition, to serve as a materials irradiation facility. Provision would be made in the design to replace portions of the blanket with experimental test modules. During Stage II (approximately 1990-95), the device would function primarily as a test facility in which various blanket, tritium breeding and extraction, materials, and coolant concepts are studied under reactor conditions. Staged operation has a number of advantages: (1) the fundamental plasma-performance objectives can be focused upon first, with minimum complications; (2) decisions to allocate funds for the blanket test modules can, to some extent, be made after the plasma physics objectives have been demonstrated; and (3) additional time is allowed for the development of the technologies required for a tritium breeding blanket, without delaying the overall TEPR schedule. The disadvantages of staged operation are associated with the added complications to the basic design and with the need for remote removal and reassembly of several components.

Stage I PECS Design Summary

The Stage I PECS (see Fig.1) is designed to (1) generate and remove 120 MW of sensible heat, (2) adequately protect the magnet system from radiation damage and activation and excessive nuclear-energy deposition, (3) allow for continuous operation of reactor-level plasmas at a plant duty factor of 30-50 percent and (4) demonstrate all operational aspects of a Tokamak fusion power reactor except tritium breeding and breeder-blanket performance. Materials and design approaches for the Stage I PECS were selected on the basis that they: (1) satisfy the nuclear requirements for energy deposition and radiation attenuation, (2) permit PECS construction, operation, and maintenance with minimum advancement in existing technology, and (3) provide a reasonable point for extrapolation to a demonstration scale Tokamak fusion power reactor.

First Wall Design Consideration

The first wall (treated as a subsystem of the PECS) includes the vacuum vessel that surrounds the plasma region and other associated components, i.e., a low-Z liner, a plasma-aperture limiter, a flux breaker, and the vacuum and neutral beam wall penetrations. The principal requirements of the TEPR first wall are: (1) to protect the plasma region from excessive atmospheric contamination, (2) to prevent excessive plasma contamination by products of plasma-wall interactions, and (3) to maintain its structural integrity for sufficient times under the severe irradiation, thermal, and stress conditions imposed by an operating fusion reactor environment.

The first-wall system design options that have been considered for the TEPR include: (1) a bare-metal first wall, (2) a first wall fabricated from sintered metal-metal oxide product of the SAP type, (3) a composite consisting of a metal vacuum wall with a protective low-Z coating, and (4) a metal vacuum wall protected from the plasma by a separate low-Z liner. The bare-metal wall is the most attractive of the four options on the basis of fabricability; but recent results[1] show that the relatively high atomic number (high-Z) of the atoms sputtered from typical structural metal surfaces (e.g., stainless steel, vanadium-base alloys) prevents the attainment of satisfactory plasma performance. The latter three first-wall options are aimed at use of a low-Z material as the first surface facing the plasma. The SAP-type materials are more resistant to surface radiation damage, primarily blistering; however, fabrication problems are more difficult. For the present design effort emphasis was placed on the use of graphite, silicon-carbide, or beryllium either as a coating on the vacuum vessel inside wall or, in the case of graphite and silicon carbide, as a separate, radiatively-cooled liner. The coating option has several attractive features, the major one being simpler fabrication; the coating could be put on after the vacuum wall is assembled and replaced remotely after extended reactor operation. Drawbacks of a radiatively-cooled liner stem from the high operating temperature of the liner (1500 C), the possibility of establishing a discharge in the annulus between the liner and the vacuum vessel, and the additional vacuum pumping requirements.

Blanket/Shield Design Considerations

The blanket region for the TEPR has been designed to convert the kinetic energies of the neutrons and associated gamma-rays into sensible heat. Other important functions of the combined blanket/shield region are: (1) to reduce radiation damage in the toroidal-field coils to acceptable levels from the standpoint of induced electrical resistivity in the copper stabilizer, decreased critical current density in the NbTi, and super-insulation deterioration; (2) to reduce nuclear heating in the toroidal-field coil system to tolerable levels; and (3) to minimize the induced activation and biological dose in the magnet structure such that magnet maintenance can be carried out in place with a minimal degree of radiation protection after reasonably short cooldown periods (i.e., a few weeks). All of these functions have to be performed with materials that: (1) are mutually compatible; (2) can withstand radiation damage for reasonable operating lifetimes; and (3) can be fabricated and/or implemented with existing or near-term technology.

This study showed that an optimum design value for the radiation-induced resistivity in the copper stabilizer of the toroidal-field coil system is 3×10^{-8} Ω-cm (which corresponds to $\sim 2 \times 10^{-4}$ dpa) from the standpoint of magnet and reactor economics and magnet reliability. This resistivity results from atomic displacement damage caused by the neutrons and can be annealed out by warming the magnet to near room temperature. Since the superconducting magnet cooldown could take as much as two months, the blanket/shield region should be designed to permit reactor operation at reasonable plant-duty factors (up to 50 percent) and wall loadings (≥ 0.2 MW/m^2) for sufficiently long time spans (≥ 2 years) between anneals. Nuclear heating of the toroidal-field coils must be reduced to the point at which the refrigeration power requirements are on the order of 1 percent of the reactor power output. In addition, there is a very strong incentive to minimize the blanket plus shield thickness on the inside of the torus so as to maximize the magnetic field in the plasma and reduce reactor size and cost.

Nuclear performance characteristics of a variety of plausible material compositions, which might meet the design requirements for the TEPR blanket/shield region, have been investigated[6]. Some materials and compositions that have been considered are summarized in Table I. Options employing stainless steel (SS) were chosen because of the excellent radiation-attenuation characteristics of SS and because it is a construction material for which a substantial technology base exists. Options employing tungsten (W) and tantalum (Ta) were considered on the basis of their superior to SS in attenuating neutron and gamma radiation. Vanadium was considered because of the favorable compatibility of vanadium-base alloys with liquid lithium and because of its reasonably good nuclear performance characteristics. Options containing graphite (C) and aluminum (Al) were investigated because they lend themselves to the development of a minimum activation blanket/shield assembly for a TEPR. Figure 2 shows the relationship between displacement damage in the magnet stabilizer and overall blanket/shield thickness for the various materials compositions investigated. Results of a similar sensitivity study, wherein the refrigeration power (as a percentage of the plant electrical power output) needed to overcome the nuclear heat load generated in the toroidal-field coil is plotted against overall blanket/shield thickness as shown in Figure 3.

Implications of the results in Figs. 2 and 3 and of other related parameter sensitivity analyses performed on the compositions given in Table I may be summarized as follows: (1) Mixtures of stainless steel (SS) and boron carbide (B_4C) are superior to all material compositions investigated, except for mixtures of tungsten (or tantalum) and B_4C, from the standpoint of reduced stabilizer displacement damage and nuclear energy deposition in the magnet. (The composition 50 percent SS - 50 percent B_4C was found to be optimal for reducing displacement damage to the stabilizer, while the composition 75 percent SS - 25 percent B_4C was optimal for minimizing nuclear energy deposition in the magnet.) (2) Studies of the induced radioactivity generated in the blanket/shield, after representative operating times, indicate that remote handling and maintenance would be required for the options employing stainless steel, tungsten (and tantalum), and to a lesser degree vanadium, even after a year of cooldown. whereas graphite and/or aluminum-containing compositions appear to be accessible (after only a few weeks cooldown) with a minimal degree of radiation protection. (3) Designs employing W-(or Ta)-B_4C and SS-B_4C mixtures offer the best prospects for achieving magnet-protection objectives with the smallest blanket/shield thickness (see Fig. 2 and 3). (4) Compositions that offer the best prospects for minimal remote maintenance (e.g., compositions employing only C and/or Al together with B_4C) appear to be incapable of leading to a blanket-shield design with overall thickness less than 1 meter that would protect the magnets.

In light of the above results, the materials option employing optimized compositions of SS and B_4C was selected for the preliminary blanket/shield design for the TEPR. An alternative design for the interior blanket/shield region, which consisted of an optimized composition of tungsten and B_4C, was carried along in parallel with the SS-B_4C design in the event that (1) additional radiation attenuation, over and above that provided by the SS-B_4C mixture, would eventually be required or (2) a sufficiently strong incentive arose to reduce the overall blanket/shield thickness to an absolute minimum value in order to increase the thermal power output. Further details of the design selection process are given in Ref. 1.

Thermal Fluid Analysis for the Preliminary PECS Design

Both pressurized helium and pressurized water were considered for the primary blanket coolant in the reference PECS design. Preliminary thermal fluid analyses for the blanket composition of alternate layers of SS and B_4C are summarized in Table II. For the same cooling configuration, blanket temperature profile, total thermal power and thermal power deposition profile (determined from the nuclear heating calculations), pressurized water is found to require significantly less pumping power and coolant-channel volume fraction. (Details of the thermal fluid calculations are given in Ref. 1.) Figure 4 shows the dependence of coolant channel void fraction on coolant tube diameter and radial distance from the first wall for selected cases that have been investigated.

It should be noted that (1) the two alternative coolant schemes (water and helium) were not optimized with respect to electrical power output per unit of thermonuclear power, (2) the coolant-channel geometry was not optimized, and (3) the stresses resulting from the indicated temperature profile remain to be analyzed. While helium appears to have greater long-range potential as a fusion reactor coolant, the advantages of reduced coolant-channel volume fraction and lower pumping power afforded by water could help to ameliorate several of the design complications for the Stage I PECS.

The shield region, which backs up the blanket, is simply an extension of the blanket insofar as materials composition is concerned, but it is operated at or near ambient temperature and is cooled with a low-temperature coolant circuit(probably borated water) that is separate and distinct from the primary coolant circuit. The substitution of tantalum or tungsten alloys for the stainless steel zones in the inner blanket and shield will require that the respective coolants be channeled through stainless steel ductwork or that the tantalum (or tungsten) be canned in stainless steel since both helium and water are incompatible with the refractory metals.

Stage II PECS Design Summary

The criteria of the Stage II PECS design are largely the same as those for the Stage I except that appropriate modules will be modified as required to include a tritium-breeding medium. These modifications consist of introducing a relatively thin zone (~10 cm) of liquid lithium or a solid lithium compound. In the liquid lithium design, the blanket structure would be altered to include the use of a vanadium-base alloy for lithium containment. The solid lithium compound would be selected to be compatible with stainless steel and would be operated in a packed-bed mode. If magnetohydrodynamic effects on heat transfer and pressure drop, resulting from the circulation of a liquid metal in Tokamak-type magnetic-field environments, prove to be tolerable, consideration will be given to the use of liquid lithium as primary coolant in a Stage II blanket module. Otherwise, both the liquid lithium and solid lithium compound blankets could be cooled with pressurized helium, although thermal energy removal and material compatibility problems are expected to be exacerbated in this case.

Consideration was given to the demonstration of net breeding gain for the Stage II modules. (The breeding gain with a Stage II module would be sufficient to achieve net breeding if the entire blanket was so constructed.) The overall thickness of the Stage II PECS modules must not exceed a value that is compatible with the basic Stage I design. The relationship between displacement damage to the toroidal-field coil stabilizer and overall blanket/shield thickness for three plausible breeding configurations is given in Fig. 5, together with the curve for an all stainless steel blanket for comparison purposes. The results in Fig. 5 show that a net breeding gain is possible with a relatively thin zone (<10 cm) of enriched lithium (~90 percent ^6Li) and beryllium. A demonstrated breeding gain in a zone of thickness $\leq$10 cm does not appear to be possible for designs that employ enriched lithium without any beryllium or natural lithium (~92 percent ^7Li).

Mechanical Design Considerations

Remote Maintenance & Repair

All aspects of maintenance, repair, or modification will necessarily be done with remote handling equipment due to the residual radioactivity from fusion neutrons. Thus all components of the PECS and the reactor need be designed for remote handling, a very impressive task considering the size, weight, and geometric complexities encountered.

The proposed TEPR design has provisions for assembly and disassembly of the blanket and shield in separate or separately joined pairs of block sections (see Fig. 6). The blanket and shield pieces are removed from between the TF coils using an overhead crane for the top members and special lift vehicles for the lower units. Remote removal of the vacuum apparatus is done in the same manner with all components and sub-components designed specifically for remote coupling. All coolant connections will also be done with remotely operated equipment.

Maintenance, repair and inspection of the vacuum wall will be done with specially designed machines which will be inserted into the torus through either the vacuum ports or the experimental access ports. A full-size section of the PECS will be maintained as a mockup housing duplicate servo machines. Using the mockup and servo machines in planographic coupling with the reactor units, pretested repairs or assembly activities may be duplicated within the reactor with technicians guiding the manipulation utilizing TV cameras.

1st Wall Vacuum Chamber

The first wall is an independently supported structure which will be cooled using a separate circuit from the blanket. Both helium and water are under consideration. Water cooling has more apparent advantages, including smaller coolant channels, simple manifolding and routing, lower temperature operation (620 F) away from the creep range, low pumping power, and ease of locating and repairing leaks. Disadvantages are thermal gradient problems associated with placement and attachment of coolant tubing, high pressure requirements to 2000 psi, tritium buildup in the water system, and radioactivity buildup in the water system.

Present design effort favors a 2-cm thick wall with ring and spar reinforcement cooled on the interior surface with separately stacked panel sections. The design includes a standoff liner which may have an initial low Z coating. The liner has provisions for cooling if necessary. Coolant lines and manifolding are nested between the vessel wall and the standoff liner with the piping exiting through an annulus around the vacuum ports.

Blanket

The present blanket design effort for the TEPR segments the blanket annulus into contoured blocks; sixteen blocks make up an annular wedge section, and a total of 32 wedge sections make up the complete blanket, 512 pieces in all. Each of these contoured blocks is an average of 1 meter, weighs ~2 tons, and contains coolant piping and lead connections, standoff insulation and support to the companion shield; torque limiters, and handling fittings. Cooling systems designs using all helium and all water are being carried.

Shield

The geometrical arrangement of the shield is similar to that of the blanket (Fig. 6). There are 512 pieces of shielding, the largest ~13 tons, the smallest ~2 tons. The major shield design problems are found in trying to minimize fabrication costs of this large volume of materials.

References

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2. C. C. Baker et al., Experimental Power Reactor Conceptual Design Study, General Atomic Co., Report GA-A13534, July 1975.
3. M. Roberts et al., Oak Ridge Tokamak Experimental Power Reactor Study-Reference Design, Oak Ridge National Laboratory Report, August 1975.
4. S. Dean, The Tokamak Fusion Reactor Research and Development Plan, 1975 Annual Meeting of the American Nuclear Society, New Orleans, La., June 8-13, 1975.
5. Two Component Torus--Joint Conceptual Design Study, Princeton Plasma Physics Laboratory, 1974.
6. M. Abdou, Nuclear Design of the Blanket/Shield System for a Tokamak Experimental Power Reactor, Nuclear Technology, April 1976.

Table I. Material-Composition Options Considered for the TEPR Blanket/Shield Region

Design No.	Blanket (30cm thick)	Magnet Shield (Variable thickness)
102	30cm stainless steel(SS)	50%SS+50% B_4C
108	30cm 50% W+50% B_4C	50%W+50% B_4C
113	30cm aluminum(Al)	50%SS+50% B_4C
114	30cm aluminum (Al)	50%Al+50% B_4C
116	30cm stainless steel(SS)	75%SS+20%H_2O+5%B
117	30cm graphite	50%SS+50% B_4C
118	30cm graphite	50%Al+50% B_4C
119	30cm vanadium	50%SS+50% B_4C

Table II. Thermal Fluid Analysis Parameters for the TEPR Preliminary Blanket Design

Parameter	Coolant Water	Coolant Helium
Thermal power removed, MW	186	186
Coolant inlet pressure, atm	136	50
Pressure drop in blanket, atm	< 1	~1
Coolant channel diameter, cm	1.0	2.5
Coolant channel volume fraction in blanket, %	1→0.1	10→2
Pumping power for blanket, MW	< 1	~7
Coolant inlet temperature, °C	38	357
Coolant exit temperature, °C	302	527
Maximum blanket temperature, °C	600	600
Tube wall temperature, °C	327	550
Number of radial zones	6	6

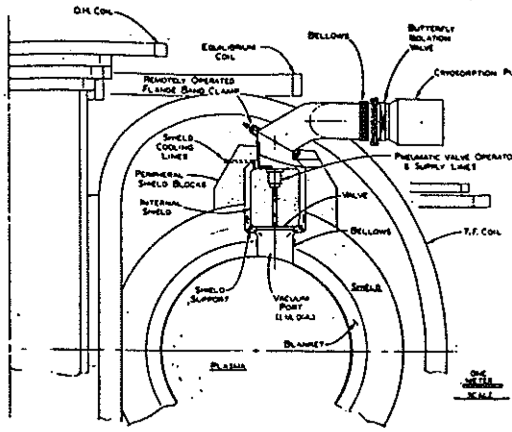

Fig. 1 ANL-TEPR Verticle Section Showing the McDonnell Douglas Vacuum Port Shield Plug Design (Courtesy C. Trachsel, McDonnell Douglas Astronautics Co.)

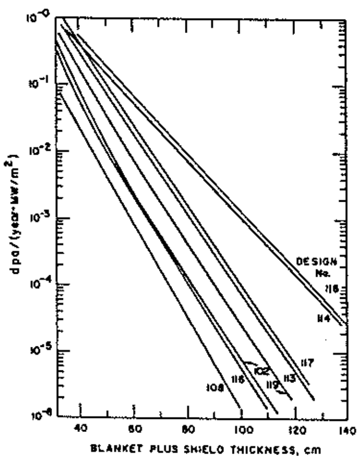

Fig. 2. Displacement Damage to the Superconductor Stabilizer (copper) versus Overall Blanket/Shield Thickness for the Materials Compositions Given in Table I

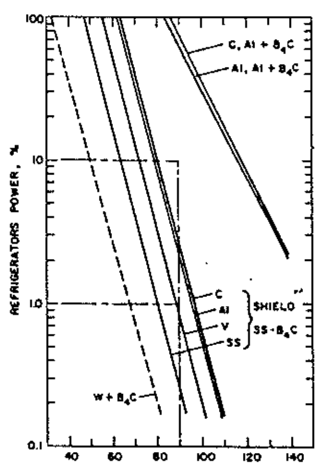

Fig. 3. Refrigeration Power Requirements (in % of Plant Electric Power Output) versus Overall Blanket/Shield Thickness for the Materials Compositions given in Table I (Assumed Thermal-to-Electrical Conversion Efficiency = 30%)

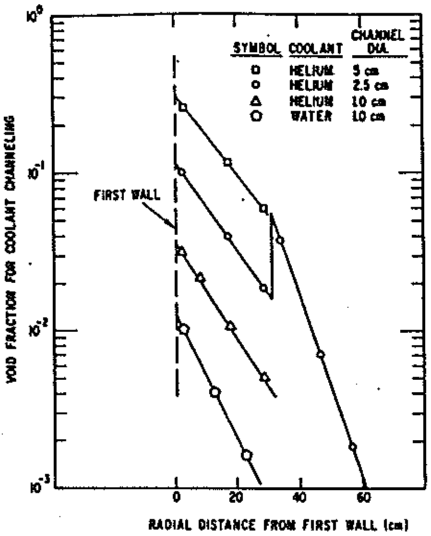

Fig. 4. Void Fraction versus Radial Distance from First wall for the TEPR Preliminary Blanket Design

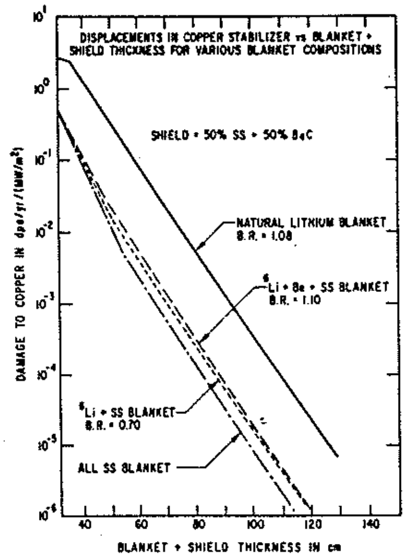

Fig. 5. Displacement Damage to Superconductor Stabilizer (copper) versus Overall Blanket/Shield Thickness for Selected Tritium-Breeding/Blanket-Materials Configurations

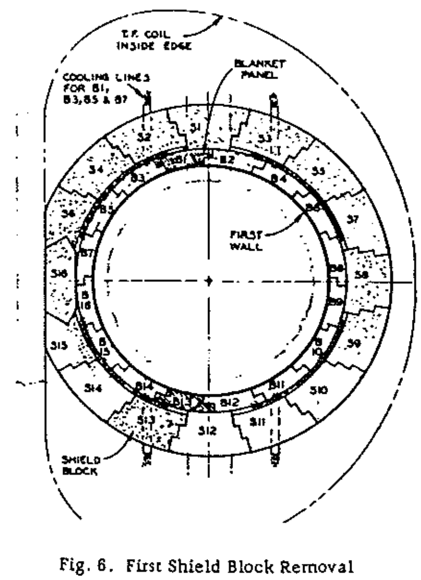

Fig. 6. First Shield Block Removal