STARFIRE - A COMMERCIAL TOKAMAK REACTOR

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ABSTRACT

The basic objective of the STARFIRE Project[1] is to develop a design concept for a commercial tokamak fusion electric power plant based on the deuterium/tritium/lithium fuel cycle. The key technical objective is to develop the best embodiment of the tokamak as a power reactor consistent with credible engineering solutions to design problems. Another key goal of the project is to give careful attention to the safety and environmental features of a commercial fusion reactor.

The basic design guidelines for STARFIRE assume the successful operation of a tokamak engineering test facility (ETF) and a demonstration power plant. STARFIRE is considered to be the tenth plant in a series of commercial reactors. It is, therefore, assumed that a well established vendor industry exists and that utilities have gained experience with the operation of fusion plants.

The STARFIRE Project[1] was initiated in May, 1979, with the goal of completing the design study by October, 1980. The purpose of this paper is to present an overview of the major parameters and design features that have been tentatively selected for STARFIRE.

MAJOR PARAMETERS

A primary goal of the STARFIRE study is to select the most attractive set of design parameters and concepts that make tokamaks economically competitive and environmentally acceptable. Results provided by and experience gained from previous fusion reactor design studies in the United States and worldwide have provided an excellent starting point for this project. In addition, extensive trade-off studies[2] were performed to support the design selection process for STARFIRE.

Figure 1 shows a cross section of the reactor. A summary of the major reactor parameters and design features is given in Table 1. The reactor thermal power is ∿ 3800 MW and the net electrical power is ∿ 1150 MW. This power level is expected to be in the most desirable range of power rating to the utilities in the STARFIRE time frame.

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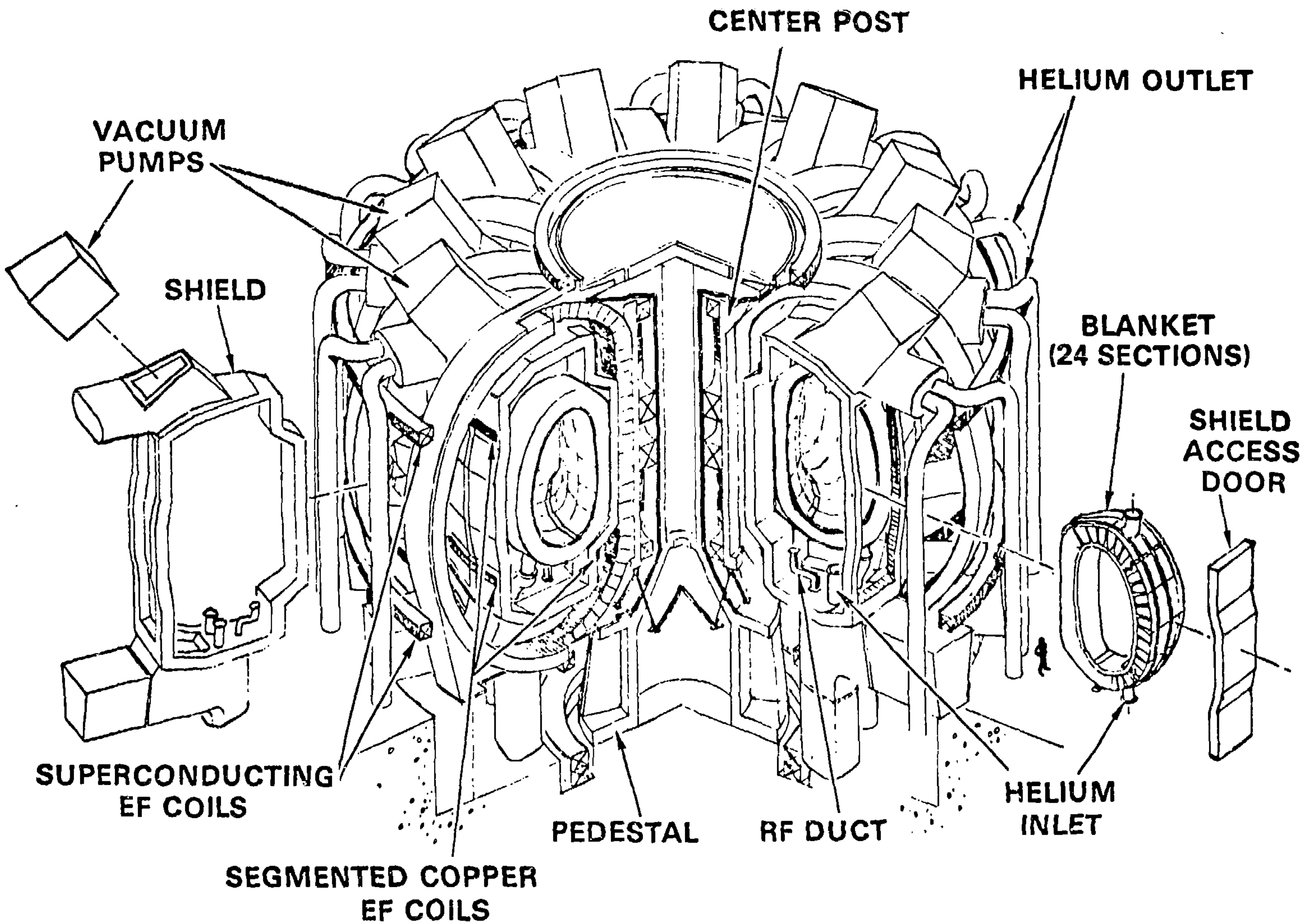

Figure 1. STARFIRE reference design.

Table 1. STARFIRE Major Design Features

Net electrical power	1150 MW
Gross electrical power	1600 MW
Fusion power	3200 MW
Thermal power (nominal)	3800 MW
Thermodynamic efficiency	41%
Overall availability	75%
Average neutron wall load	3.5 MW/m^2
Major radius	7.0 m
Plasma half-width	1.94 m
Plasma elongation (b/a)	1.6
Maximum toroidal field (nominal)	11.0 T
Number of TF coils	12
Plasma burn mode	Continuous
Current drive method	rf
Plasma heating method	rf
TF coils material	Nb_3Sn/NbTi/Cu/SS
Wall structural material	Ferritic steel
Blanket structural material	Ferritic steel
Wall coolant	D_2O
Tritium breeding medium	Li_2O
Blanket coolant	Helium
Plasma impurity control	Low-Z coating + limiter and vacuum system + enhanced radiation + field margin
Primary vacuum boundary	At inner edge of shield

The neutron wall load is 3.5 MW/m^2. Based on results from detailed systems studies, this moderately high wall load is a reasonable choice that results in a small size reactor without excessive requirements on the first-wall cooling capability, maximum toroidal magnetic field and frequency of structural material replacement.

A D-shaped plasma with a height-to-width ratio of 1.6 has been selected. This was found to be nearly the highest elongation that is consistent with the STARFIRE goal of locating almost all the equilibrium-field (EF) coils outside the toroidal-field (TF) coils.

With an aspect ratio of 3.6, the major radius is 7 m. The average plasma toroidal beta is 0.067 and the maximum toroidal magnetic field is 11 T.

OPERATING MODE

Steady-state has been selected as the mode of operation for STARFIRE. Although experimental information on plasma current drive by means other than inductive OH coils are very limited, results from recent theoretical studies justify the assumption that continuous plasma burn can be achieved in the STARFIRE time frame.

The potential advantages of steady-state reactor operation include: (a) increased component and system reliability, (b) eliminating fatigue as a serious concern for the structural material in the first wall and blanket, (c) no thermal energy storage is required and the need for an intermediate coolant loop is reduced, and (d) no electrical energy storage is necessary.

PLASMA ENGINEERING

The basic plasma parameters of the STARFIRE design are listed in Table 2. Lower hybrid waves constitute the reference plasma current driver due to the more developed theoretical understanding of this option. Other current drive candidates considered include relativistic electron beams and magnetosonic waves.

Table 2. STARFIRE Plasma Parameters

Major radius	$R = 7$ m
Plasma half width	$a = 1.94$ m
Elongation (b/a)	$\kappa = 1.6$
Plasma shape	"D"
Average toroidal beta	$\beta_t = 0.067$
Plasma current	$I = 11$ MA
Safety factor (center)	$q_o = 1.4$
Safety factor at edge	$q_a = 4.2$
Toroidal field on axis	$B_{to} = 5.62$ T
Average DT ion density	$n_{DT} = 1.0 \times 10^{20}$ m^{-3}
Average electron density	$n_e = 1.3 \times 10^{20}$ m^{-3}
Average ion temperature	$T_i = 17$ keV
Average electron temperature	$T_e = 22$ keV
Fractional impurity concentration	$n_{Be}/n_{DT} = 0.04$
Fractional alpha concentration	$n_\alpha/n_{DT} = 0.09$
Fractional burnup	$f_b = 0.11$

The ratio of fusion power to rf power (Q) required to maintain the steady-state toroidal current of a tokamak is limited by the plasma's accessibility to lower hybrid waves. In particular, low density plasmas are more easily penetrated and result in higher Q than high density plasma. This is the reason that, for a fixed β_t, relatively high temperatures and low densities are attractive for STARFIRE. However, for $\overline{T}$ > 15 keV the fusion power density decreases so quickly that relatively high magnetic fields are necessary to keep the reactor size attractively small. A related strategy to maximize Q is to drive the highest current densities where the plasma density is low, i.e., near the plasma surface. Particular attention must be given in this case to the MHD stability of such current density profiles.

Of the two candidate rf sources, crossed-field amplifiers (CFA) and klystrons, the former may promise to operate at higher efficiencies (70-90%), but, due to the large amounts of power dissipated in the rf structure, the CFA option may increase the engineering difficulties. The transmission line is envisioned to be a high power pressurized waveguide with a power splitting labyrinth to provide the grill phase shifts. The system will be designed with the ECR layer in the high pressure region; the vacuum window will be placed out of the direct line-of-sight of neutron radiation. Phase shifters will be included to provide spectral turning for the reactor startup period.

Our basic approach for impurity control is to use a system with fairly low removal efficiency, $\sim$ 30%. This is sufficient to maintain a stable operating point, while at the same time reducing the amount of pumped tritium.

The reference system chosen is a limiter/vacuum system which together serves to concentrate and pump some of the plasma particle outflux. A possible limiter system is shown later in Fig. 3; several other designs are being investigated. Major design issues include the high heat fluxes, neutron radiation damage and erosion due to sputtering by plasma particles. In addition, the TF field coil is designed with enougn field margin to contain the excess pressure of the alpha particles and electrons. There are two other basic features of the impurity control system; low-Z coatings are used on all exposed surfaces; and the plasma is operated so as to maximize the heat radiated from it and to minimize the transported heat. This approach, therefore, results in four techniques working in unison. These are: (a) low-Z coatings, (b) magnetic field margin, (c) enhanced radiation, and (d) limiter/vacuum system. A backup divertor concept will also be developed.

FIRST/WALL BLANKET

The technological and design aspects of various first-wall/blanket concepts have been considered in the selection of potentially viable designs for STARFIRE. The major emphasis has been placed on the development of a blanket design that is safe and environmentally acceptable. The primary guidelines established to meet these criteria are low tritium inventory in the blanket, minimal long-lived activation products and minimal stored chemical energy.

A comprehensive assessment of potential tritium breeding materials, coolants and structural materials was carried out. The scope of assessment included material properties, neutronics, compatibility and operating temperature limitations, generic safety aspects and tritium recovery methods. Based on the results of this assessment, the most viable combinations of breeder/coolant/structural material/neutron multiplier are given in Table 3. The

Table 3. Selected First-Wall/Blanket Materials Options

Coolant	Coolant FW	Coolant Blanket	Breeder*	Structure** FW	Structure** Blanket
Reference	D_2O	He	Li_2O	FS	FS
Alternate	D_2O	D_2O	Li_2O	FS	FS
Backup	Li	Li	Li	V	V

* Alternate options for the solid breeder include Li_2SiO_3 and $LiAlO_2$; a neutron multiplier will be necessary with these options.

** Austenitic stainless steel is an alternate selection for the first-wall structure and both austenitic stainless steel and titanium alloys are possible alternatives for the blanket structure.

reference blanket system is selected on the basis of perceived safety advantages associated with helium coolant and the solid breeding materials. Helium is selected only for the blanket coolant since water provides several advantages as the first-wall coolant. The ceramic breeding materials with their high temperature properties are most appropriate for the helium coolant. Lithium oxide is proposed as the breeding material since it is the only ceramic with potential for breeding without a neutron multiplier. The ferritic steels (FS) are compatible with both helium and water coolant and are selected for both first-wall and blanket structure. A low pressure helium purge stream over the Li_2O is used for tritium processing since direct contact of high pressure helium coolant and Li_2O is not acceptable. The feasibility of tritium extraction by a helium purge stream is being examined further. The key issue is the resulting tritium inventory in the blanket.

Since the thermal-hydraulic characteristics of pressurized water are superior to those of helium in temperature-limited systems, an alternate first-wall/blanket materials option is proposed. The alternate concept utilizes D_2O coolant in both the first wall and the blanket. Lithium oxide is retained as the breeding material and ferritic steel as the first-wall/blanket structure.

Liquid lithium, which can be used as both coolant and breeder, provides a unique blanket option. However, the inherent safety of this system has been questioned and maintenance-related problems have been identified. An intermediate coolant loop may also be desirable. However, because the thermal-hydraulics, neutronics, radiation behavior, and design simplicity of this system are generally regarded as superior to other blanket materials options, this system is suggested as a backup. The backup design will be considered if irresolvable technical or engineering problems are identified in the early phase of the project for the reference and alternate designs. Selected vanadium alloys, which are proposed for the structural material, are believed to be the most resistant to radiation damage of the candidate structural materials, produce low long-term activation products, are compatible with lithium, and possess adequate elevated temperature mechanical properties. Long blanket lifetime may be attainable since the liquid lithium is not sensitive to radiation damage and tritium-release from the liquid also does not present a problem.

Two mechanical design concepts are being considered for the reference blanket. In the first concept, the module walls are pressurized to the coolant static pressure. The solid breeder is held in sealed tubes, arranged in a staggered rod bank pattern, which are cooled by cross-flowing the helium over them. In the second concept, the helium coolant flows inside tubes each of which is surrounded by solid breeder throughout the module. In both concepts, the tritium produced in the solid breeder is removed continuously through a helium purge gas system. The two concepts will be compared in the near future after further evaluation to determine which is superior from an overall reactor design standpoint.

HEAT TRANSPORT SYSTEM

The thermal energy deposited in the blanket and first wall is delivered via the heat transport system to the power conversion system where electricity is generated. The heat transport and power cycle systems consist of the primary helium (blanket coolant) loops, primary water (first wall and non-breeding regions coolant) loops, auxiliary cooling loops, and the steam/power conversion system as shown schematically in Fig. 2.

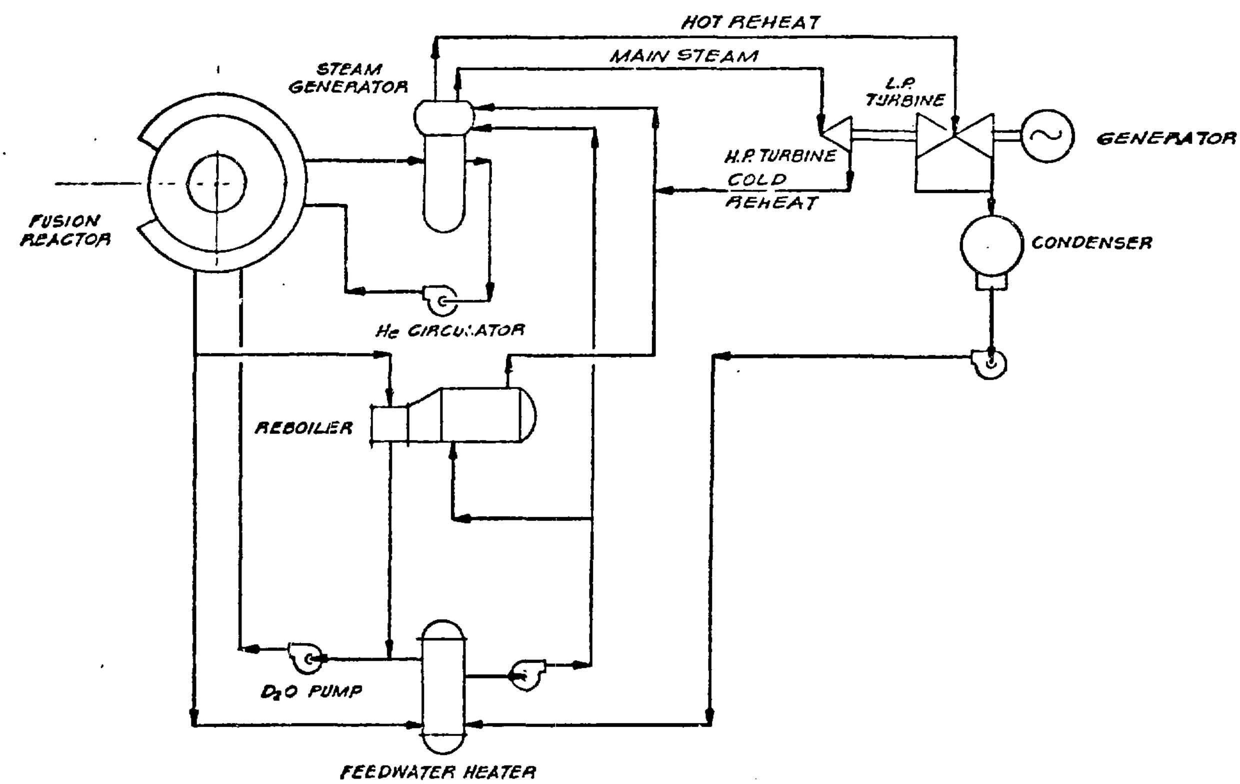

Figure 2. Power conversion system schematic.

With the water-cooled first wall and inboard blanket, approximately 40 percent of the thermal energy is in the form of 300°C heat. This energy is utilized in the power conversion system for feedwater heating and for steam generation. The remaining 60 percent of the thermal energy is transported via the helium blanket coolant to the steam generators. The primary loops consist therefore of both helium and water circuits.

Because STARFIRE operates steady-state rather than in a pulsed mode, a thermal energy storage system is not required. Furthermore, no intermediate coolant loop is provided and the helium loop interfaces directly with the steam generator. The consequences of steam ingress into the helium loop were previously addressed for the high-temperature gas cooled reactor (HTGR). It was found that adequate safeguards against steam inleakage can be provided. A calculation of permeation losses from the helium coolant through the steam generator was performed. It has been concluded that, due to the fact that the tritium is predominantly in the oxidized form, and due to the fairly low temperature, tritium permeation losses will be very small and an intermediate coolant loop is not required. The possibility of an intermediate heat exchanger in the D_2O coolant loop is being examined.

VACUUM PUMPING SYSTEM

The primary vacuum boundary of the STARFIRE vacuum system is at the inner wall of the shield. A pair of $\sim$ 60 meter long toroidal limiters pass circumferentially around the outer edge of the plasma region, and deflect ions from the scrape-off zone into the adjacent slots in the first wall. A cross-sectional view of these slots is shown in Fig. 3. The slots are 60 meters long and 20 cm wide and penetrate the first wall and blanket. The location and configuration of the limiter is optimized to maximize the probability of a molecule entering the slot after striking the limiter. Each slot contains a step to reduce neutron streaming as shown in Fig. 3. In the molecular flow region this step has minimal effects on the conductance, the only consideration being the additional slot length required to provide the step. A large volume plenum exists between the outside of the first wall blanket and the inside of the shield. Twenty-four 1.7 meter diameter, 1.5 meter long vacuum ducts penetrate the shield and provide access for liquid helium cryogenic pumps.

The liquid helium cryogenic pumps will be of the compound variety in which hydrogen and its isotopes are pumped by cryocondensation on a liquid helium cooled panel and helium is cryosorbed on a 4.2°K molecular sieve surface. The cryosorption surface on such a pump can become saturated with helium and will require regeneration after approximately eight hours of operation. Regeneration will be accomplished by stopping the liquid helium flow and allowing the sorption surface to warm whereupon the helium is released and pumped away by another pumping system.

OTHER ENGINEERING FEATURES

The reactor magnet system design includes twelve 11 Tesla TF coils. The EF coil system includes 4 segmented copper coils inside the TF coils for plasma stability control, but most of the EF coils are superconducting and located outside the TF coils for plasma equlibrium. The plasma startup, heating and current drive is accomplished by a lower hybrid rf system. Startup assistance from rf pre-ionization and/or some inductive current drive from the poloidal

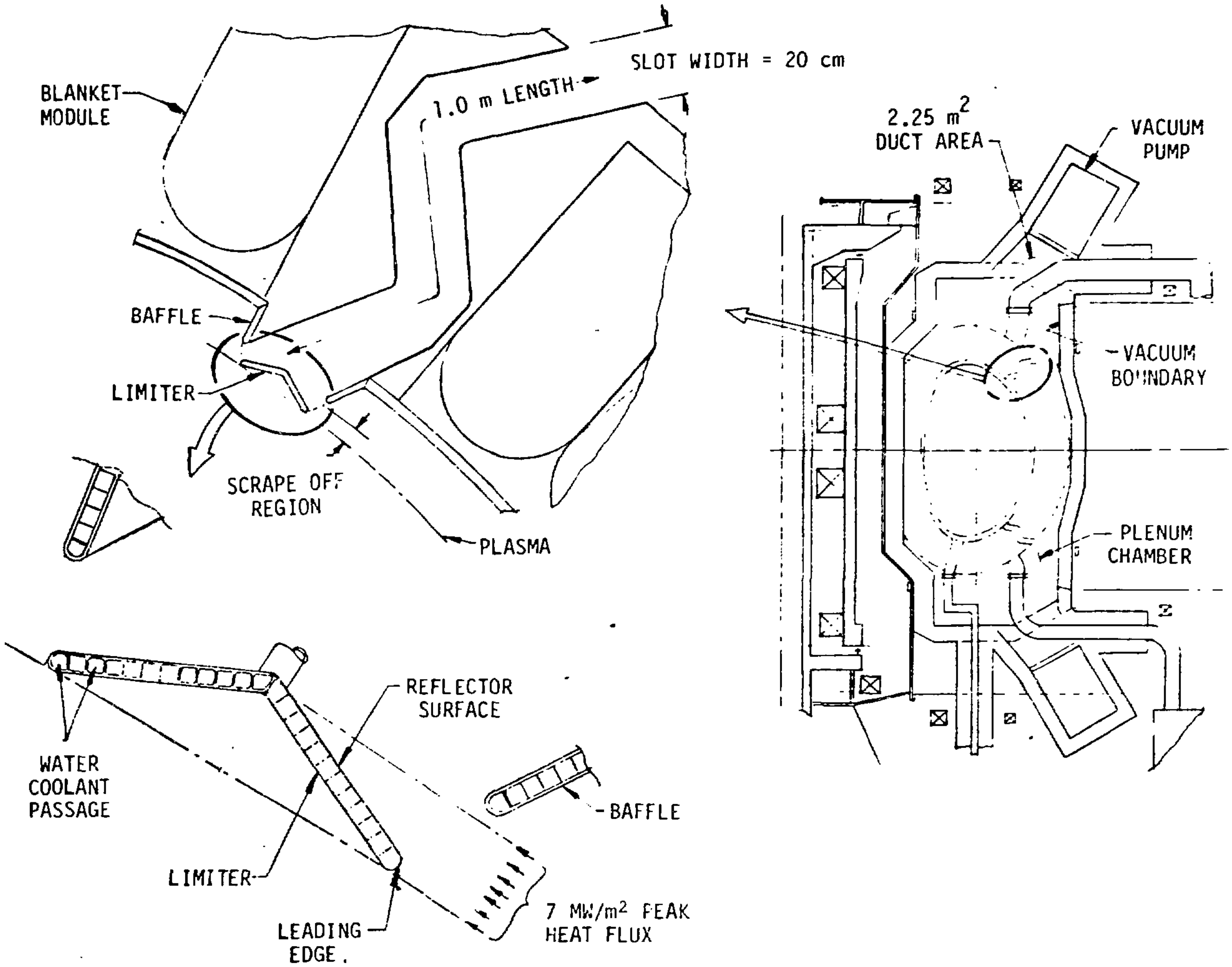

Figure 3. Limiter/vacuum duct concept.

coils is being investigated. The shield incorporates dielectric breaks to prevent toroidal current flow and permit the EF coils that are located inside the TF coil to have adequate response time for plasma stability control.

The reactor shield is designed for life-of-plant and is not removed for normal maintenance operations. The shield concept is shown in Fig. 4. The reactor shield is water cooled and serves as the primary vacuum chamber. The shield protrudes between the TF coils at the top and bottom of the reactor to provide a vacuum duct to 24 cryosorption pumps. The vacuum duct opening through the shield is oversized to permit routing of coolant lines through the opening without significantly restricting pumping capacity. Each helium coolant line is shielded at the penetration through the shield by a bed of rods in a staggered pattern that limits neutron leakage. The shield contains poloidal dielectric breaks under every third TF coil. The dielectric joint, which is at the outside of the shield and completes the vacuum barrier, is exposed to about 10^{10} rads irradiation. The dielectric break will also be redundant and incorporate intermediate pumping. The shield pieces are designed to keep all seal welds between shield pieces in a single plane without

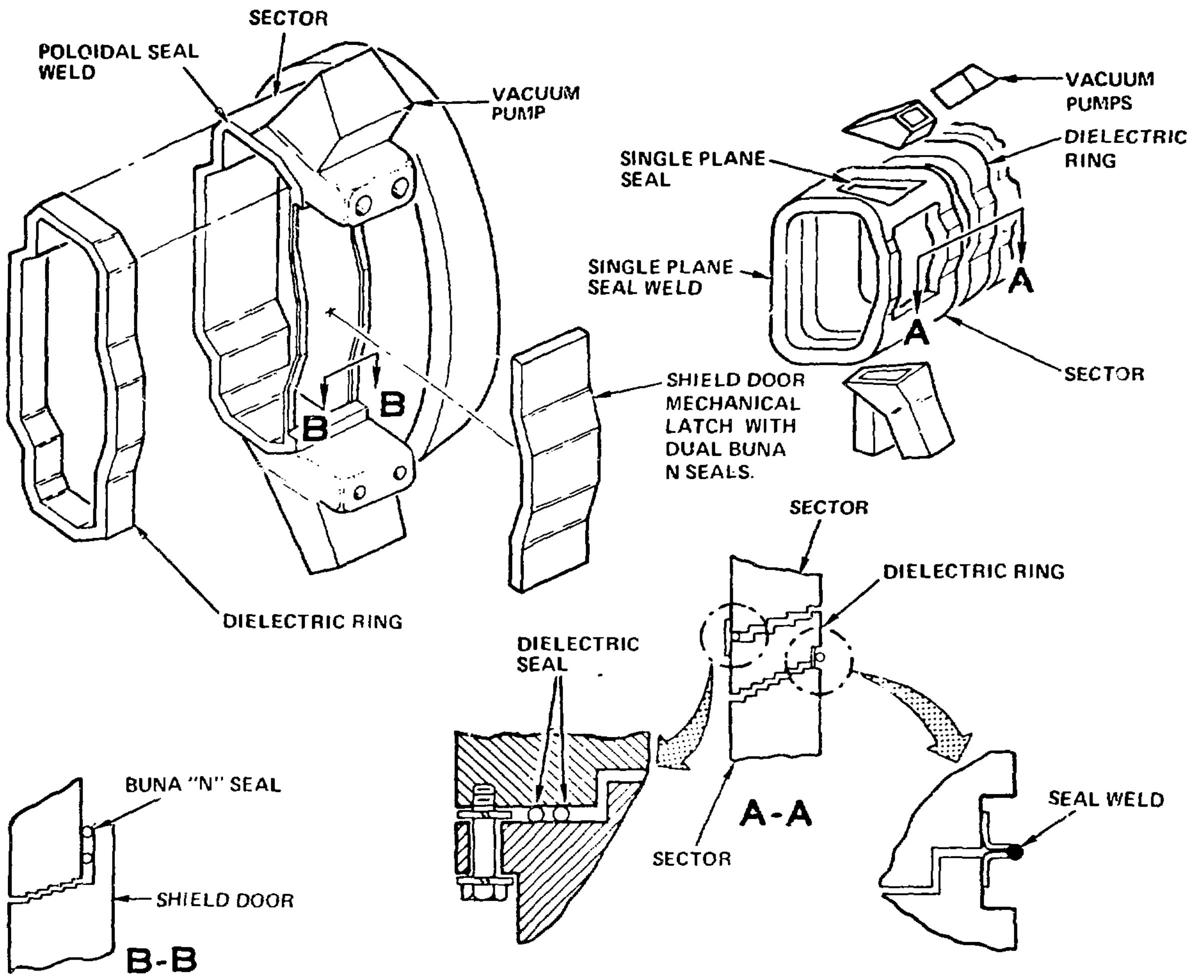

Figure 4. Shield concept for STARFIRE.

welded corners, in order to improve reliability and simplify the installation operation. The shield doors utilize redundant Buna-N seals to provide the vacuum barrier. The seals are located near the outside of the shield; however, degradation by exposure to tritium and neutrons is expected to result in the need for seal replacement each time the door is removed during maintenance.

The rf waveguides form a near continuous toroidal ring at the base of the reactor. Approximately 100 MW of rf power is required to start and drive the plasma. The power density at the plasma is 8 MW/m^2. A mechanical rf waveguide connector is required to permit blanket removal.

The blanket system is divided into 24 wedge shaped sectors to obtain a size that permits removal of sectors between TF coils. Each blanket sector incorporates an rf waveguide, two limiter segments, and two stepped toroidal slots 20 cm wide for vacuum pumping. Each sector is mounted on an air bearing pad to permit removal and replacement.

The reactor building is metal lined to provide a boundary against tritium release to the atmosphere. Access doors are provided to the interior through airlocks to permit ingress of reactor components. Continuous cleanup of the reactor building air environment is provided. The building is designed for a maximum overpressure of 3 psi which could result from an accident in which the liquid helium from one TF coil and the coolant from one first wall/blanket sector are released into the building.

MAINTENANCE APPROACH

Availability goals have been established as 85% for the reactor and 75% for the complete plant including the reactor. These goals provide a basis for design of maintenance equipment. The maintenance scenario incorporates the current utility practice of shutting down annually for one month and a four month shutdown approximately every 5-10 years.

The design philosophy being followed is to minimize the radiation levels within the reactor building; to design all components for complete remote maintenance, and to identify contact maintenance operations where personnel can safely be used with significant economic savings.

The number of different maintenance operations planned in the reactor building are minimized by using a component "remove and replace" approach. This permits each maintenance action to be preplanned and designed for use with simple push, pull, etc., operations. This approach increases the confidence in the speed of maintenance operations and simplifies maintenance equipment design requirements. Once the damaged or end-of-life components are removed from the reactor they are transported to a hot cell where more time is available for checkout, repair or disposal.

Redundancy is planned for reactor auxiliary subsystems to permit continued operation of the plant until a scheduled maintenance period or until the component can be replaced in-service.

ACKNOWLEDGEMENTS

The authors of this paper are the project managers of the various organizations involved in STARFIRE and thus represent many individual contributors. These individual contributors are listed in the various papers below as references.

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