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A STEADY-STATE TOKAMAK REACTOR WITH NON-DIVERTOR IMPURITY CONTROL - STARFIRE

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by

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ABSTRACT

STARFIRE is a conceptual design study of a commercial tokamak fusion electric power plant. Particular emphasis has been placed on simplifying the reactor concept by developing design concepts to produce a steady-state tokamak with non-divertor impurity control and helium ash removal. The concepts of plasma current drive using lower hybrid rf waves and a limiter/vacuum system for reactor applications are described.

1. OVERVIEW OF THE STARFIRE REACTOR

The purpose of the STARFIRE study [1-3] is to develop a design concept for a commercial tokamak fusion electric power plant. The major features of STARFIRE include a steady-state operating mode based on continuous rf lower-hybrid current drive and auxiliary plasma heating, solid tritium breeder material, pressurized water cooling, limiter/vacuum system for impurity control, all superconducting equilibrium field (EF) coils outside the superconducting toroidal field (TF) coils, fully remote maintenance, and a low-activation shield. The major parameters of STARFIRE are listed in Table I. An isometric view of the reference design is shown in Fig. 1.

The first wall/blanket (FW/B) is segmented into 24 sectors to permit removal between the 12 TF coils. The material of the FW/B is PCA (prime candidate alloy) stainless steel that operates at a maximum temperature of 425°C. The FW/B, limiter and rf waveguide are designed for a 16 $MW\text{-}yr/m^2$ integral neutron wall loading which results in a lifetime of about six years between changeouts. All components in contact with the plasma are coated with beryllium. The FW/B is cooled by pressurized water (15.4 MPa) with inlet and outlet temperatures of 280°C and 320°C, respectively. This permits the $LiAlO_2$ to operate within an appropriate temperature range to permit tritium release to a low-pressure helium purge stream without significant sintering of the solid tritium breeder material [4].

Table I. Major Parameters for STARFIRE

Net electrical power, MW	1200
Gross electrical power, MW	1440
Fusion power, MW	3510
Thermal power, MW	4000
Overall availability, %	75
Average neutron wall load, MW/m^2	3.7
Major radius, m	7.0
Plasma half-width, m	1.94
Plasma beta	0.067
Toroidal field on axis, T	5.8
Maximum toroidal field, T	11.1
Blanket structural material	Austenitic stainless steel
Tritium breeding medium	$\alpha\text{-}LiAlO_2$
Neutron multiplier	Zr_5Pb_3 (solid) or Be
First wall/blanket coolant	Pressurized H_2O
Primary vacuum boundary	At inner edge of shield

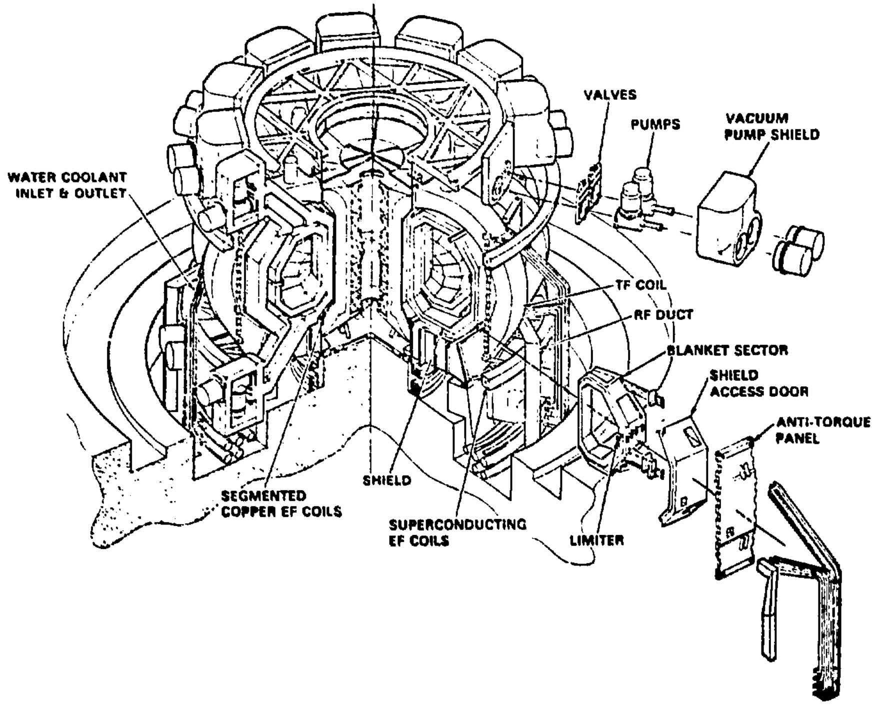

Fig. 1. STARFIRE reference design.

The shield serves as the vacuum boundary for the system. Twelve shield access doors with mechanical seals are provided between each pair of TF coils. Dielectric breaks are located in the outer regions of the shield where the maximum radiation dose is < 10^{10} rads. The shield will last the entire life of the reactor. The shield is 1.1 m thick and is composed of TiH_2, B_4C and a low-nickel steel. This composition and thickness will limit the radiation dose in the reactor building so that manned access is possible within 24 hours after shutdown. In addition, the activation of the shield and other components outside the blanket is low enough to permit recycling of the materials within 30-50 years after reactor decommissioning.

The maintenance concept for STARFIRE is similar to the approach developed by the Culham group [5]. The basic approach is to use modularized components with simplified "push/pull" procedures in the reactor building and to carry out almost all repairs in a hot cell away from the reactor. Fully remote maintenance is planned for all operations, although some manned access is possible as a backup. This will significantly reduce the radiation dose for plant workers. Availability goals of 75% for the overall plant result in permissible downtimes of 37 days/year for scheduled maintenance for the reactor and balance of plant (BOP), 34 days/year for unscheduled maintenance for the reactor, and 20 days/year for BOP unscheduled maintenance.

2. CURRENT DRIVE AND BURN CYCLE

There are several advantages that result from steady-state operation of a tokamak by non-inductive plasma current drive. These advantages include increased component and system reliability, higher neutron wall loadings, elimination of material fatigue of the first wall as a major life-limiting concern, no need for thermal and electrical energy storage, and elimination or significant reduction of the size of the ohmic heating (OH) coils.

Several current drive concepts were considered for STARFIRE [6]. Lower hybrid rf current drive was selected because its technological features of simple waveguides and existing power sources were compatible with reactor engineering considerations, and because the theory of rf current drive by lower hybrid waves is reasonably well established [7].

A cross-section of STARFIRE showing the rf system is illustrated in Fig. 2, and the rf system parameters are listed in Table II. The rf power is minimized by operating at low electron densities, radiating in a narrow band spectrum and minimizing the total plasma current. For STARFIRE, these considerations result in a plasma current of 10 MA (limiter q value of 5) with the current density peaked near the plasma edge where the density is low. Such a "hollow" current profile is consistent with a penetration depth by lower hybrid waves of about 15% of the plasma radius in the STARFIRE plasma. The wave frequency corresponds to twice the local lower hybrid frequency to avoid parametric instabilities. For an aspect ratio of 3.6 and a plasma elongation of 1.6, the plasma was found stable to betas of at least 6.7%, provided that the first wall provides some wall stabilization. The first wall has been designed with a current decay (L/R) time of 300 ms for this purpose.

The Brambilla theory [8] has been adapted to the STARFIRE conditions for design of the waveguides. Traveling waves are launched by phasing adjacent guides by $2\pi/3 = 120°$. The narrow opening in the toroidal direction (3 cm) and the spacing between openings (septum) define the toroidal wavelength. The spectral width is established by limiting the number of guides in the toroidal direction in one array to 18. The vertical dimension is 17 cm; four banks are stacked in the poloidal direction forming one module between each TF coil of size 0.66 m x 0.78 m. The average rf power density at the waveguide is 1.6 kW/cm^2 which should not

POWER BALANCE	MW
R F POWER TO PLASMA	90.4
(CURRENT DRIVE)	(88.5)
LOSSES	62.3
TOTAL R F ELECTRICAL POWER	152.7

POWER LOSSES	MW
A GRILL	0.5
B WAVEGUIDE	1.9
C WINDOW (3)	0.2
D WAVEGUIDE	18.2
E CIRCULATOR	10.8
F PHASE SHIFTER	1.5
G C F A	21.8
ELECT. BLDG POWER SUPPLIES	7.8
TOTAL LOSSES	62.3

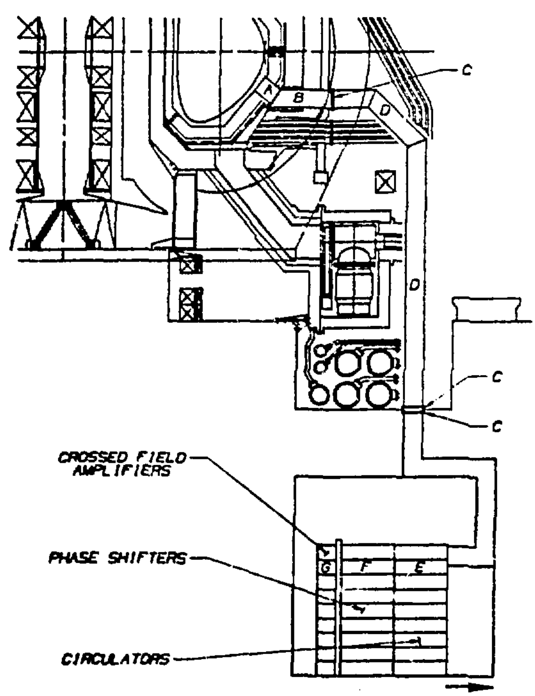

Fig. 2. STARFIRE rf system.

Table II. Lower Hybrid RF System Parameters for STARFIRE

1.67 GHz	wave frequency
1.40-1.86	spectrum required
10.9 cm	toroidal wave length
1.6 kW/cm^2	wave intensity at antenna
$2\pi/3$	phase difference
3 cm	guide opening in toroidal direction
0.70	septum
17.0 [illegible]	vertical (poloidal) guide opening
0.736	spectral fraction driving current
0.443	antenna reflection coefficient

result in any nonlinear plasma responses. The system will deliver ∿ 90 MW to the plasma of which ∿ 67 MW will drive the plasma current. The remaining power is in various side bands which is not useful for current drive, but is useful for plasma heating.

The rf duct assembly is mounted to the blanket and is constructed of PCA stainless steel. The interior of the duct is coated with copper to minimize power losses. Near the plasma, the waveguides are also coated with beryllium. A window of alumina is located near the TF coils to prevent electron cyclotron breakdown in the slots and to prevent backflow of tritium.

Figure 2 lists the power losses in the various components of the rf system. The rf system components are located in the basement of the reactor building and the power supplies are located in a separate building. A total electrical power of 153 MW is required for the rf system. Because substantial power is reflected by the array, it is important to recover this power. Crossed-field amplifiers appear to be well suited for this system.

For steady state operation, there is a broad range of startup, burn and shutdown scenarios. The STARFIRE burn sequence consists of the following phases:

- Electron cyclotron resonance heating power of 5 MW for about one second to break down and ionize the initial fill of DT gas (96% D) at about 10^{19} particles/m^3.
- Use of a small OH coil to produce an initial plasma current of ∿ 2 MA in 14 seconds.
- RF current drive at 45 MW for 250 seconds at low density which raises the plasma temperature to ∿ 6 keV, and increases the plasma current to the final value of 10 MA.
- Main heating phase with rf power at 90 MW for 200 seconds while the mainly deuterium plasma density is raised to its final value (0.8×10^{20} m^{-3}).
- A fusion ramp phase of 17 minutes when the fusion power is increased at a 5% per minute rate by slowly changing the tritium fraction from 4% to 50%. A small amount of impurities (0.1% of iodine) is added to maintain a stable plasma temperature.

The total startup time lasts about 24 minutes; the plasma will burn until a shutdown is required for maintenance or some other reason. Normal shutdown is essentially the reverse of the startup case described above. An abrupt shutdown capability is provided by injecting impurities into the plasma and creating a plasma disruption which will terminate the plasma in ∿ 100 ms. STARFIRE is designed to take several disruptions in a year. Rapid, but less severe, shutdowns of ∿ 2.5 s are also possible. It has also been shown that any hot spot formation on the first wall (∿ 1% of first wall area) will cause sufficient beryllium to ablate and extinguish the plasma in < 1 s, before any substantial damage can be done to the first wall. This illustrates one of the inherent safety features of fusion.

3. LIMITER/VACUUM SYSTEM

A limiter/vacuum system concept has been developed for STARFIRE for the purpose of impurity control and removal of the helium ash. The design utilizes a toroidal limiter, located at the outer side of the torus as shown in Figs. 1 and 3. For the STARFIRE design,

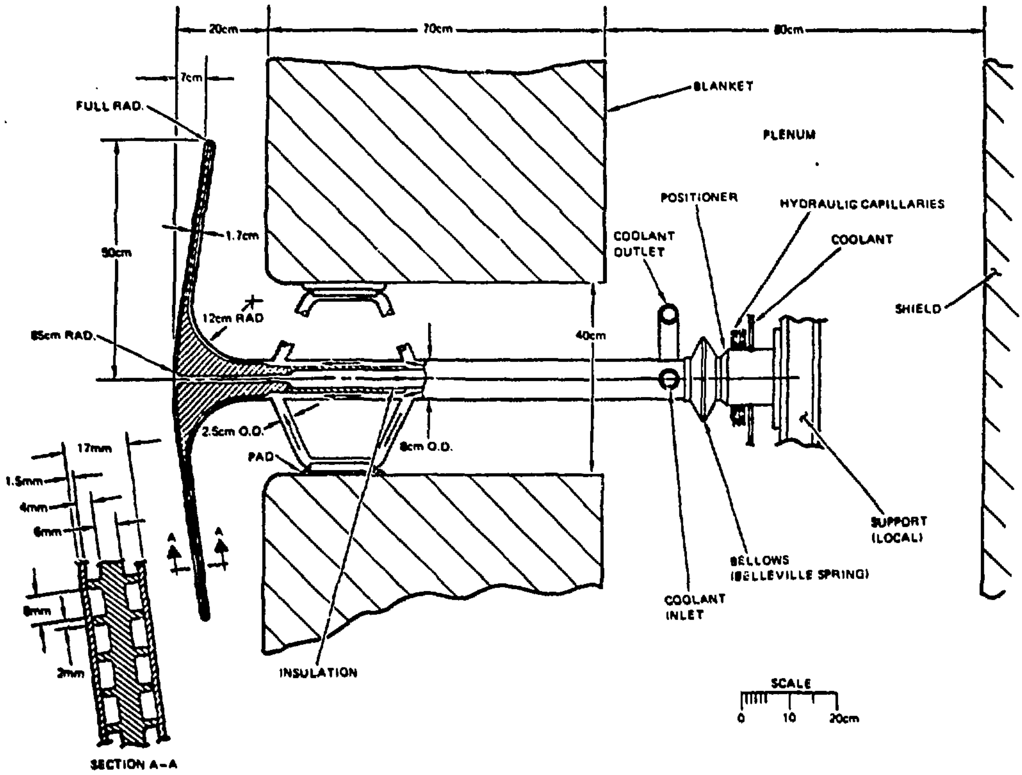

Fig. 3. STARFIRE limiter concept.

about 28% of the charged particle flux will be directed into the slot behind the limiter, where the charged particles will be neutralized. Some of the helium atoms will be directed into the 0.4 m high vacuum duct behind the limiter, which leads to a large vacuum plenum between the blanket and shield. There are 12 vacuum ducts at the top and bottom of the reactor with 1 m diameter penetrations through the shield (see Fig. 1). This arrangement significantly reduces neutron streaming. There are 48 compound cryopumps with 24 on-line at any given time, each with a helium pumping speed of 120 m^3/s. The pumps are rejuvenated every two hours.

Helium atoms streaming back out of the limiter slot have a very high probability of being ionized by the incoming plasma. The helium ions will be directed back to the interior surfaces of the limiter by electric fields set up by sheath effects at the limiter. Thus, there is a high probability that all of the α-particle flux entering the limiter slot will be removed by the limiter/vacuum system. This will result in about a 14% equilibrium α-particle concentration in the plasma, which is accommodated by providing an additional margin in the on-axis toroidal field of 0.85 T.

Because of significant charge-exchange reactions for the DT atoms in the limiter slot, only ∿ 10% of the DT flux will be pumped. This results in a high recycling of the DT particles and a high tritium burnup fraction (42%), with a resulting low inventory of tritium in the pumping and fueling system. This is one of the key features of the limiter/vacuum system.

Table III presents a summary of the major parameters of the limiter/vacuum system. In order to minimize the heat load to the limiter, most of the alpha-heating power to the plasma is radiated to the first wall by injecting a small amount of iodine along with the DT fuel stream. The total heat deposited on the limiter is 200 MW with a maximum heat flux of

Table III. Major Features of the Limiter/Vacuum System

Helium production rate, s^{-1}	1.24×10^{21}
Helium reflection coefficient, R_{α}	0.75
Hydrogen reflection coefficient, R_{DT}	0.90
Alpha particle concentration (n_{α}/n_{DT})	0.14
Beryllium (low-Z coating) concentration (n_{BE}/n_{DT})	0.04
Toroidal-field margin at plasma center, T	0.85
Fractional burnup, tritium	0.42
Tritium inventory in vacuum pumps and fueling system, g	200
Scrape-off region thickness, m	0.2
Limiter (one toroidal limiter centered at midplane)	
Structural material	Ta-5W, V-20Ti or FS-85
Low-Z coating material	Beryllium
Coolant	Water
Coolant inlet temperature, °C	115
Coolant outlet temperature (2 pass), °C	145
Maximum coolant pressure, MPa (psia)	4.2 (600)
Total heat removed from limiter, MW (90 MW transport, 56 MW radiation plus neutrals and 54 MW nuclear)	200
Maximum heat load (at leading edge), MW/m^2	4

4 MW/m^2. This heat is used for feedwater heating in the power conversion cycle. A detailed assessment of material candidates that included radiation effects, thermal-hydraulics, and stress analyses was performed. Tantalum, niobium, vanadium, and copper alloys appear to have the greatest potential as limiter materials. The limiter is designed to accommodate the electromagnetic forces induced under plasma transient conditions.

The limiter and first wall are coated with beryllium. The first wall will be eroded at a rate of 0.14 mm/year for STARFIRE's plasma conditions of a total first wall flux of 6.5×10^{22} particles/s. An initial coating of 1.2 mm on the first wall will have a lifetime of $\sim$ 7 yr. While the beryllium coating is also being sputtered off of the limiter, much of the beryllium from the limiter and first wall is being redeposited back on the limiter. Sufficient beryllium density ($\sim$ 4%) is maintained in the plasma so that there is no net erosion of beryllium from the leading edge of the limiter.

4. SUMMARY

The STARFIRE design has incorporated a number of concepts which has resulted in a more simplified reactor with increased safety and environmental features. In particular, the limiter/vacuum system has the advantages that it does not require magnets, can have manageable heat loads, has a reasonable vacuum system which minimizes neutron streaming, has a high tritium burnup and minimizes tritium inventory in the fuel cycle, and it significantly reduces the complexity of the system leading to easier maintenance and assembly.

STARFIRE is designed to operate in a steady-state mode with the plasma current maintained by momentum transfer from lower-hybrid waves launched from end-fire waveguide arrays. The steady-state mode of operation significantly enhances the reactor potential of the tokamak with the specific advantages listed in Sec. 2. The lower-hybrid rf system appears adequate but the major disadvantage is the relatively large recirculating power requirements ($\sim$ 10% of the plant gross electrical power output). Larger savings are potentially realizable with steady-state operation if the performance of the lower-hybrid current driver can be further improved or substantially better alternatives for the current driver are developed.

The results of the STARFIRE study indicate important directions for the development of tokamaks to significantly enhance their potential as power reactors. Simplifying the reactor design and improving component reliability must be important goals in developing fusion as a practical and economical energy source.

5. ACKNOWLEDGEMENT

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