# STARFIRE – A CONCEPTUAL DESIGN OF A COMMERCIAL TOKAMAK POWER PLANT*

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## Abstract

STARFIRE – A CONCEPTUAL DESIGN OF A COMMERCIAL TOKAMAK POWER PLANT.

Starfire is a conceptual design for a commercial tokamak power plant based on the deuterium/tritium/lithium fuel cycle. The emphasis of the study is on the simplicity of the engineering design, maintainability, lower electricity cost, and improved safety and environmental features. The reactor has a 7-m major radius and produces 1200 MW of electric power. Starfire operates in a steady-state mode with the plasma current driven by a lower-hybrid RF system. The plasma purity control and exhaust system is based on the limiter/vacuum concept, which offers unique advantages for commercial power reactors. The blanket utilizes a solid lithium compound for tritium breeding and pressurized water as the coolant.

## 1. INTRODUCTION

A conceptual design (STARFIRE) for a commercial tokamak power plant has been developed. The key technical objective of the STARFIRE study is to develop an attractive embodiment of the tokamak as a commercial power reactor consistent with credible engineering solutions to design problems. STARFIRE is based on the deuterium/tritium/lithium fuel cycle and is considered to be the tenth plant in a series of commercial reactors. This paper

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describes the major features of the reference reactor concept and presents a summary of the key conclusions derived from the study. The details of the study are documented in Ref. 1.

The primary criteria for commercial attractiveness emphasized in the STARFIRE study are economics, safety, and environmental impact. In addition to experience gained from previous design and systems studies in the United States and worldwide, critical assessments and extensive tradeoff analyses were carried out to guide the selection process for STARFIRE.

## 2. OVERVIEW OF REACTOR CONCEPT

The major reactor parameters for STARFIRE are listed in Table I. These parameters were derived based on results from system analyses [2] to minimize the cost of energy subject to constraints of physics, engineering and technology.

TABLE I. STARFIRE MAJOR DESIGN FEATURES

| | |
|---|---|
| Net electrical power, MW | 1200 |
| Gross electrical power, MW | 1400 |
| Fusion power, MW | 3490 |
| Thermal power, MW | 4000 |
| Gross turbine cycle efficiency, % | 36 |
| Overall availability, % | 75 |
| Average neutron wall load, $MW/m^2$ | 3.6 |
| Major radius, m | 7.0 |
| Plasma half-width, m | 1.94 |
| Plasma elongation (b/a) | 1.6 |
| Plasma current, MA | 10.1 |
| Average toroidal beta | 0.067 |
| Toroidal field on axis, T | 5.8 |
| Maximum toroidal field, T | 11.1 |
| No. of TF coils | 12 |
| Plasma burn mode | Continuous |
| Current drive method | rf |
| Plasma heating method | rf |
| TF coils material | $Nb_3Sn$/NbTi/Cu/SS |
| Blanket structural material | PCA[a] |
| Tritium breeding medium | Solid breeder |
| Wall/blanket coolant | Pressurized water |
| Plasma impurity control | Low-Z coating + limiter and vacuum system + enhanced radiation + field margin |
| Primary vacuum boundary | Inner edge of shield |

[a]Advanced austenitic stainless steel.

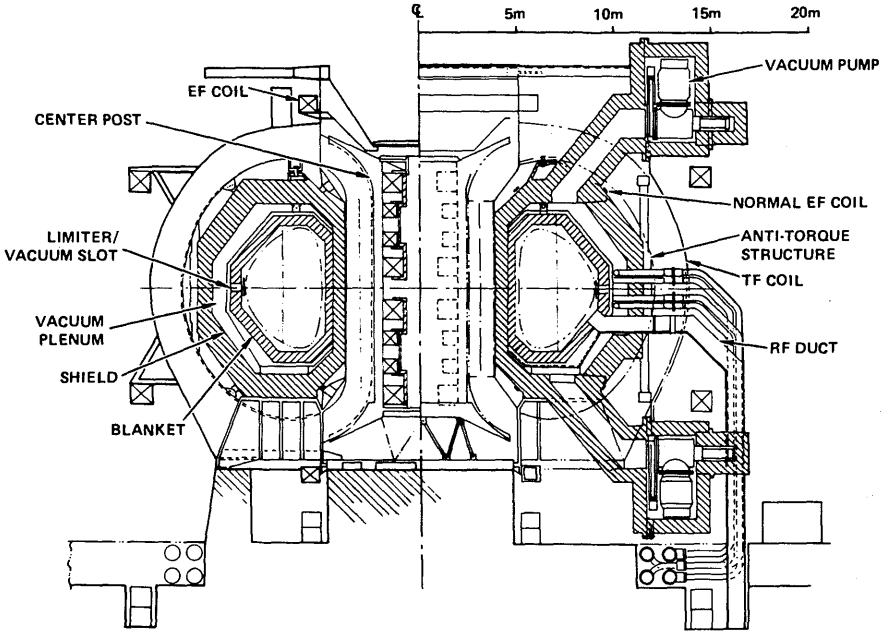


*FIG.1. Starfire reference design: cross-section.*

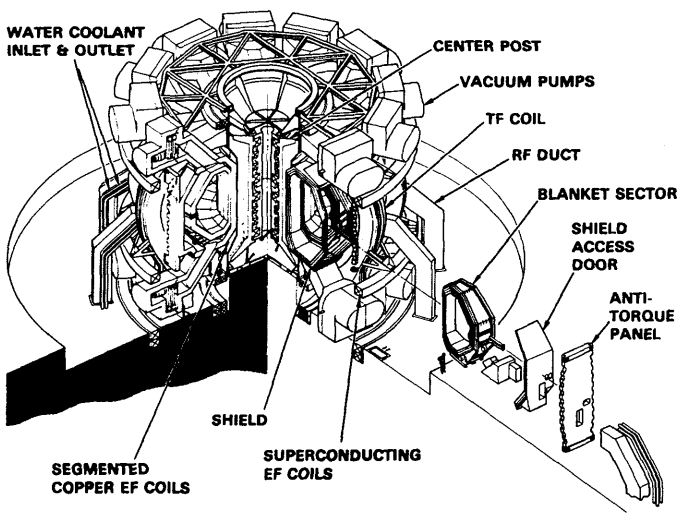


*FIG.2. Starfire reference design: isometric view.*

The reactor cross-section is shown in Fig. 1 and the major features are shown in the isometric drawing of Fig. 2. All superconducting equilibrium-field (EF) coils are located outside the 12 toroidal-field (TF) coils and 4 small segmented copper coils are located inside for plasma stability control. The shield provides the primary vacuum boundary. Twelve shield access doors are provided to permit removal of 24 toroidal blanket sectors.

A plasma burn cycle with slow startup and shutdown, consistent with steady-state operation, was developed. During the breakdown phase, ∿5 MW of electron cyclotron resonant heating (ECRH) is applied. The limited OH coil system induces ∿1 MA of plasma current. The lower-hybrid rf drive is then applied to gradually heat the plasma and bring the current up to the full value of 10 MA. The length of the burn period is limited only by the shutdown needs for reactor maintenance. The required electrical power for startup and shutdown is low enough to be taken off the grid with no need for electrical energy storage.

The plasma purity control and exhaust system is based on the limiter/vacuum concept. The limiter consists of segments which form a continuous toroidal ring at the reactor outer midplane. The limiter concentrates the plasma impurities, including alpha particles, and directs a fraction of the neutralized particles into a slot behind the limiter. These particles are then pumped through a vacuum plenum region between the blanket and shield into 24 vacuum ducts at the top and bottom of the reactor. Forty-eight cryosorption/cryocondensation pumps are used. Twenty-four of the pumps are operated while the remaining 24 are rejuvenated.

The first wall/blanket is segmented toroidally into 24 sectors to permit removal between TF coils. The first wall and structural material is PCA stainless steel that operates at ∿425°C maximum temperature. The first wall/blanket is cooled by pressurized water with inlet and outlet temperatures of 280°C and 320°C, respectively. This permits operation of the $LiAlO_2$ solid breeder material within a broad temperature range to enhance tritium release without sintering. A helium purge stream is used to extract the tritium.

The first wall/blanket sectors also provide mounting for the 12 ECRH and 12 lower-hybrid waveguides, the fueling ports and the limiter system. The waveguides and fueling ports are located on the sector between TF coils. The first wall, limiter, and waveguides are coated with beryllium to minimize the effects of sputtered impurities on the plasma. The first wall/blanket, limiter and waveguide assembly are designed for a 16 MW-yr/$m^2$ life. Blanket sectors are manifolded separately to permit leak detection and isolation.

The shield provides neutron and gamma-ray attenuation and serves as the primary vacuum boundary for the plasma. The shield is assembled from 12 sectors and 12 shield rings. Dielectric breaks are located in 6 of the shield rings near the outer surface of the shield to limit the radiation dose to $10^{10}$ rads.

## 3. STEADY-STATE OPERATION

Theory and experiments indicate the possibility that toroidal plasma currents may be maintained in tokamaks with noninductive external momentum sources to the electrons. This suggests that steady state may be an achievable mode of operation for tokamaks. Steady-state operation offers many technological and engineering benefits in commercial reactors. Among these are: (1) component and system reliability is increased; (2) material fatigue is eliminated as a serious concern; (3) higher neutron wall load is acceptable; (4) thermal energy storage is not required; (5) the need for an intermediate coolant loop is reduced; (6) electrical energy storage is significantly reduced or eliminated; and (7) an ohmic heating solenoid is not needed, and external placement of the EF coils is simplified. It has been estimated that the combined benefits of steady state can result in a saving in the cost of energy as large as $\sim$25-30%.

The penalty for steady-state operation comes primarily from potential problems associated with a noninductive current driver; in particular: (1) the electrical power requirements; (2) the capital cost; and (3) reliability and engineering complexity of the current driver.

In STARFIRE, a lower-hybrid rf system is utilized for the dual purpose of plasma heating and current drive [3]. The rf system has been designed for consistency with plasma physics constraints, and its components have been selected with the goal of minimizing the electric power required to maintain the plasma current. The system has been analysed in the thermal, electrical, magnetic and radiative environments of reactor operation and appears to provide reliable performance.

The theory of lower-hybrid driven currents suggests a threefold strategy for reducing the power required for current generation: (1) minimization of the total plasma toroidal current I; (2) generation of the current density j primarily in regions of low electron density; and (3) transmission of a narrow wave spectrum with a low toroidal index of refraction. With these ends in mind, we have surveyed the Grad-Shafranov equilibria appropriate to STARFIRE. The absence of a large ohmic-heating transformer permits the strategic location of equilibrium field coils and the creation of an elongated ($\kappa$ = 1.6), highly triangular plasma.

The most suitable equilibrium at a volume-averaged beta of 6.7% has I = 10.1 MA with j peaked near the plasma surface. This profile is found stable to local interchange and ballooning modes, but conducting blanket segments may be desirable in order to stabilize the n = 1 kink mode. STARFIRE is designed to operate at a high electron temperature ($\bar{T}_e$ = 17 keV) and a low plasma density ($\bar{n}_e = 1.2 \times 10^{20}$ m$^{-3}$), to further assure minimum rf power requirements. For these parameters, and with the spectrum peaked in the range $n_{\parallel}$ = 1.20-1.82, it was found that 63 MW of power at 1.4 GHz is dissipated in maintaining the equilibrium current.

The Brambilla theory of lower-hybrid wave launching from a phased waveguide array has been employed to design the waveguides. Under the assumption that the efficiencies of the tubes and rf components can be modestly increased by a development program, it is found that steady-state reactor operation could be sustained with 150 MW of electrical power, as compared to the gross electric plant output of 1440 MW.

The penalty associated with the lower-hybrid current drive is ∿12-15% of the cost of power. Therefore, the choice of steady state as the operating mode in STARFIRE results in a net saving in the cost of energy of ∿15%. Much larger savings are potentially realizable if the performance of the lower-hybrid current driver can be further improved or substantially better alternatives for the current driver are developed.

## 4. LIMITER/VACUUM SYSTEM

A serious effort has been made in STARFIRE to develop a plasma impurity control and exhaust system that satisfies the following goals: (1) have manageable heat loads in the medium where the alpha and impurity particles are collected; (2) have a reasonable and reliable vacuum system that minimizes the number and size of vacuum ducts; (3) have a high tritium burnup to minimize the tritium inventory in the fuel cycle; and (4) have engineering simplicity compatible with ease of assembly/disassembly and maintenance.

These goals are found to be best satisfied by a limiter/vacuum system [4] together with a beryllium coating on the first wall and limiter. The design, shown in Figs 1 and 3, utilizes a toroidal limiter, centered at the midplane at the outer scrape-off region, for concentrating the alpha particles diffusing out of the plasma. The slot region formed between the limiter and the first wall leads into a 0.4-m-high limiter duct that penetrates through the blanket and opens into a plenum region. The conductance of the plenum region is large enough to permit locating the vacuum ducts in the bulk shield sufficiently removed from

TABLE II. MAJOR FEATURES OF THE LIMITER/VACUUM SYSTEM

| | |
|---|---|
| Helium production rate, $s^{-1}$ | $1.24 \times 10^{21}$ |
| Helium reflection coefficient, $R_{\alpha}$ | 0.75 |
| Hydrogen reflection coefficient, $R_{D,T}$ | 0.90 |
| Alpha-particle concentration ($n_{\alpha}/n_{DT}$) | 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 | 0.35 |
| 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 (Nb or V) alloy |
| 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 |
| Coolant channel size | 8 mm × 4 mm |
| Wall thickness, mm | 1.5 |
| Maximum material temperature (coolant side), °C | 215 |
| Maximum material temperature (coating side), °C | 330 |

the midplane that radiation streaming from the limiter duct in the blanket to the vacuum pumps is acceptable. Monte Carlo calculations show that the main constraint on the vacuum system performance is the conductance of the vacuum ducts located in the bulk shield (Fig. 1). Table II presents a summary of the major features 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 deuterium-tritium fuel stream [5]. The helium removal efficiency of the limiter/vacuum system is intentionally kept low (25%) to ease the limiter design and to minimize the tritium inventory tied up in the vacuum and fueling systems.

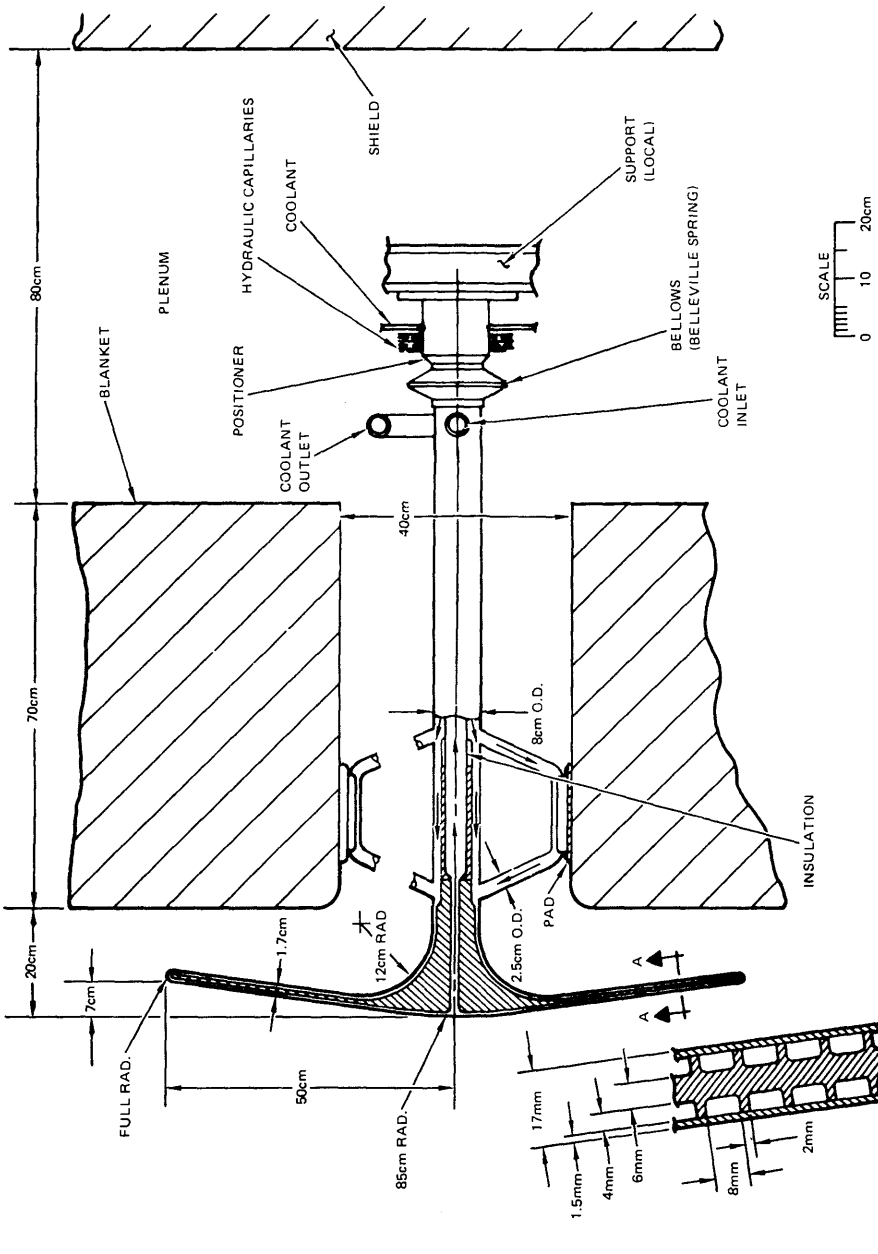


*FIG.3. Schematic of limiter design showing dimensions.*

The total heat deposited on the limiter is 200 MW with a maximum heat flux of 4 $MW/m^2$. Water is used as the coolant with a maximum pressure of 4.2 MPa and an exit temperature of 145°C. The heat from the limiter 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 and vanadium alloys appear to have the greatest potential as limiter materials. The limiter is designed to accommodate the electromagnetic forces induced under plasma transient conditions.

This study shows that the limiter/vacuum concept is an attractive option for the impurity control and exhaust system in tokamak power reactors. Its main advantages are: (1) it does not require magnets; (2) it has minimal impact on access and breeding blanket space; (3) it dramatically reduces radiation streaming problems; (4) the surface area available for particle collection is relatively large and the heat load is manageable; and (5) it permits higher tritium fraction burnup and lower tritium inventory in the vacuum pumps and fueling system. The limiter/vacuum system is relatively simple and inexpensive and deserves serious experimental verification.

## 5. BLANKET AND ENERGY CONVERSION

The choice of coolant and the physical form of the tritium breeding medium has a substantial impact on the design, operation, maintenance, safety and economics of fusion power plants. The promising coolant types are liquid lithium, molten salts, helium and water. Liquid lithium offers unique advantages. It can simultaneously perform the functions of tritium production, heat deposition and heat transport resulting in a simple low-pressure system. However, the potential safety problems associated with the relatively large stored chemical energy in liquid lithium systems provide an incentive for seriously examining other options.

A major effort of STARFIRE has focused on the use of solid compounds for breeding tritium. One of the difficult problems with solid breeders is the development of an efficient tritium recovery scheme to keep the tritium inventory in the blanket to a low level. Periodic removal of the breeder appears to be an unacceptable option because it entails an intolerably high tritium inventory that could reach ∿40 kg/GW of fusion power for annual replacement. Another approach for tritium recovery is continuous circulation of the solid breeder. This approach presents very difficult engineering problems in tokamak geometries. A non-mobile [6] solid tritium breeder blanket with in-situ tritium recovery appears to be the preferred approach. A low-pressure ($\sim$0.1 MPa) helium is circulated through formed channels in the highly porous solid breeding material.

A blanket utilizing continuous in-situ tritium recovery from a solid breeder imposes significant design constraints [7]. The temperature must be high enough to permit the bred tritium to diffuse out and yet not be so high that pore closure and sintering occurs. Only $Li_2O$ and $LiAlO_2$ are predicted to have acceptable (>200°C) operating temperature ranges for diffusion in $\sim$1 μm grain size. The calculated solubilty of tritium in $Li_2O$ at a $T_2O$ partial pressure of $10^{-1}$ Pa in the helium is substantially in excess of 100 wppm at temperatures of 600-1000 K. This solubility translates to >35 kg of tritium in the STARFIRE blanket. The calculated solubility of tritium in $LiAlO_2$ is $\sim$10 wppm at the same $T_2O$ pressure ($10^{-1}$ Pa). Therefore, $LiAlO_2$ is selected for the STARFIRE reference design.

Another difficult problem is that the most promising solid breeders (the ternary oxides) require a neutron multiplier. Beryllium is the best candidate but it has the problems of limited material resources and toxicity. Lead is a good neutron multiplier but it has a low melting point. The problem appears to be resolvable by using $Zr_5Pb_3$ which has a high melting point ($\sim$1400°C) and its neutron multiplication is adequate.

Tradeoff studies [8] comparing helium and water as coolants were performed. The results show clear advantages for the use of pressurized water for the STARFIRE conditions.

## 6. ECONOMICS

Preliminary analysis of the economics of tokamak reactors as exemplified by STARFIRE was performed. A comparison of the cost of electricity generated by fusion (STARFIRE), fission (light-water reactors with no reprocessing) and coal was carried out.

The busbar electricity cost consists of three components: (1) the return on capital (fixed cost); (2) the operating and maintenance (O&M); and (c) the cost of fuel. The percentage contribution of these three components varies substantially for the three energy options as follows: (a) return on capital: 90%, 70% and 50%; (b) O&M: 10%, 5% and 10%; and (c) fuel: 0.1%, 25% and 40% for fusion, fission, and coal, in respective order. Trends of the past decade indicate that the cost of fuel experiences a much higher escalation rate than the cost of labor, materials and construction. Table III shows the busbar electricity cost for fusion, fission and coal as a function of the initial

TABLE III. BUSBAR ELECTRICITY COST AS A FUNCTION OF INITIAL YEAR OF OPERATION

(Fuel Escalation = 4% above inflation)

| Initial Year of Operation | Fusion | Fission[a] | Coal[a] |
|---|---|---|---|
| 1990 | 74 | 62 | 69 |
| 2000 | 74 | 69 | 82 |
| 2010 | 75 | 80 | 101 |
| 2020 | 75 | 96 | 129 |

[a] Ref. 9.

year of operation for a 1200-MW(e) power plant assuming that fuel escalation is 4% above inflation rate. It can be seen from the table that new fusion plants will be economically competitive at their first year of operation by the year 1995 compared to coal and by the year 2005 compared to new fission plants.

An important finding is worth pointing out. The electricity cost for a given power plant continues to rise from year to year during the lifetime of a coal or fission power plant, while it remains approximately constant for a fusion plant. This effect is due to the larger increase in the annual cost of fuel for coal and fission. Detailed economic analysis to estimate the levelized energy cost, which accounts for the energy cost over the lifetime of the plant, shows that fusion is competitive to fission (LWR) and cheaper than coal for new plants constructed for 1990 initial year of operation.

## 7. CONCLUSIONS

The conceptual design of the STARFIRE tokamak power plant demonstrates that fusion reactors can be developed to be economically competitive with attractive safety and environmental features. The cost of energy estimated for STARFIRE is comparable to that of future fission light-water reactors and lower than for coal power plants. No runaway accident that could pose a major risk to the public can be identified for STARFIRE. The results of the STARFIRE study clearly indicate new and important directions for the development of fusion reactors in general and tokamaks in particular to significantly enhance their potential as power reactors.

The greatest uncertainties in the economics of future tokamak power plants are in the areas of plant availability and construction time. These are crucially dependent on ease of assembly, ease of maintenance, and component reliability. Therefore, simplifying the reactor design and improving component reliability must be realized as important goals in developing tokamaks as economical power reactors.

STARFIRE is designed to operate in a steady-state mode with the plasma current maintained by lower-hybrid waves. Steady-state operation significantly enhances the reactor potential of tokamak with the specific advantages listed in Section 3. The lower-hybrid rf system appears adequate but its major disadvantage is the relatively large recirculating power requirements (~10% of the plant gross electrical power output). Much larger savings in the cost of energy are potentially realizable with the 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 impurity control and exhaust system is one of the most difficult components in tokamak reactors. Divertors and divertless options were evaluated. It is concluded that the limiter/vacuum system is an attractive option for power reactors, as discussed in Section 4.

The potential safety problems with liquid lithium led the STARFIRE study to focus on blankets with solid tritium breeders. In-situ tritium recovery schemes are the most appropriate for tokamak geometries. Experimental information on tritium release characteristics of solid lithium compounds in prototypical reactor conditions are needed. Pressurized water is found superior to helium cooling for the STARFIRE conditions.

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## DISCUSSION

F.L. RIBE: How do you arrive at the estimate of 10% recirculating power for the RF current drive?

M.A. ABDOU: The details of the calculations can be found in Ref. [1] of the paper. The minimum RF power required to achieve 10 MA of plasma current is determined from a calculation of the lower hybrid wave propagation and damping. For the Starfire density profile, the antenna must deliver 63 MW at 1.4 GHz with an index of refraction of 1.2 and a width of 0.62. Taking into account the power lost to the sidebands (as a result of wave interference), we can calculate that a total of 90 MW must be absorbed in the plasma to give the 63 MW needed to drive the current. We use a Brambilla waveguide array and crossed-field amplifiers. Allowing for all the losses in the RF hardware system, we estimate that we need ~150 MW of electrical power. This is about 10% of the plant gross electrical output.

T. HIRAOKA: How did you estimate the life of the beryllium coating and what was your result?

M.A. ABDOU: The primary mechanism for erosion of this coating is sputtering. This is estimated to be ~0.1 mm per annum for the first wall. A large fraction of the sputtered beryllium will end up at the limiter. Because of the variation in temperature and density in the scrape-off region, the erosion of the beryllium coating on the limiter is not uniform. There is also significant redeposition. We predict that the lifetime of the first-wall coating will be roughly the same as that of the structural material, which is about six years in Starfire. However, we have also developed the capability of in-situ recoating. This can be done each year during the 28 days of shutdown for normal maintenance of the reactor and the power plant.