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STARFIRE—A COMMERCIAL TOKAMAK FUSION POWER REACTOR CONCEPT*

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INTRODUCTION

The purpose of the STARFIRE study 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 an attractive embodiment of the tokamak as a power reactor consistent with credible engineering solutions to design problems. Another goal of the study is to give careful attention to the safety and environmental features of a commercial fusion reactor. This paper describes the major features of the reference reactor concept.

The basic design guidelines for STARFIRE assume the successful operation of a tokamak engineering test facility 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.

A survey of anticipated utility requirements in the STARFIRE time frame (e.g. ~ 2020) indicated power units of 1000 to 1500 MWe are most desirable. The fusion power was selected as 3490 MW, which results in 4000 MW thermal power and 1200 MW of net electrical output.

Recent research in plasma physics indicates 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 that component and system realiability is increased, material fatigue is eliminated as a serious concern, higher neutron wall loads are acceptable, thermal energy storage is not required, the need for an intermediate coolant loop is reduced, electrical energy storage is significantly reduced or eliminated, a full-size 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 30 percent.

The penalty for steady-state operation comes primarily from potential problems associated with a noninductive current driver, in particular, electrical power requirements, capital cost, and reliability and engineering complexity of the current driver. In STARFIRE, a lower-hybrid rf system is utilized for the dual purpose of plasma auxiliary heating and current drive. The penalty associated with the LH current drive is ~ 12 to 15 percent of the nominal cost and power requirements. Therefore, the choice of steady-state as the operating mode in STARFIRE results in a net saving in the cost of energy of ~ 15 percent. Much larger savings are potentially realizable if the performance of the current driver can be further improved or substantially better alternatives for the current driver are developed.

The maximum allowable neutron wall load is a function of the structural material, coolant, and design of the first wall. Limitations on the maximum structural

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temperature and thermal stress are often important constaints and these are sensitive to the surface heat load on the first wall. Steady-state plasma operation reduces fatigue as a constraint, thus permitting higher neutron wall load, higher thermal stress designs. On the other hand, radiating most of the alpha-particle heating power to the first wall increases the surface heat load and lowers the allowable neutron wall load. Other considerations that limit the neutron wall load are the coolant pumping power and the structure lifetime. Figure 1 shows the cost of energy as a function of the neutron wall load at two values of the integral neutron wall load, I_w, of 5 and 20 MW-yr/m² and at two different values for the total cumulative downtime, t_d, for replacement of the structural material. For I_w = 5 MW-yr/m² and t_d = 125 *d*, the neutron wall load should be kept at ~ 2-2.5-MW/m². For I_w = 20 MW-yr/m², the cost of energy (COE) decreases significantly as the neutron wall load is increased from 1 to 2 MW/m². A smaller, but significant, saving in COE is realizable by increasing P_{nw} from 2 to 3 MW/m². A slight change in COE is noticeable in the range P_{nw} ~ 3 to 4 MW/m². These results assumed water coolant and modified austenitic steel structure in the first wall. Structural materials with better thermomechanical properties and radiation damage resistance can show a more pronounced saving in COE at higher neutron wall loads. The average neutron wall load in STARFIRE is selected as 3.6 MW/m² with an average surface heat load of ~ 0.9 MW/m². Poloidal variations in the surface heat load and neutron flux result in a peak-to-average ratio of ~ 1.2.

With the fusion power and neutron wall load selected, the surface area of the first wall is determined and the plasma elongation and aspect ratio are the only two parameters required to completely describe the plasma geometry. A D-shaped plasma with a height-to-width ratio (κ) of 1.6 was selected. This was found to be nearly the upper limit on elongation if the important design goal of locating most of the EF coils external to the TF coils is to be achieved. A detailed tradeoff analysis was performed to determine the optimum aspect ratio.[1] The most dominant effects tend to be the higher stability limit for β at lower aspect ratio and the reduced electrical power requirement for the LH plasma current driver at larger aspect ratio. An aspect ratio of 3.6 yields the minimum COE. For the STARFIRE conditions, this results in a plasma major radius of 7 m and a half-width of 1.94 m.

REACTOR DESCRIPTION

The reactor is shown in Fig. 2 and the major parameters are listed in *Table I.* All superconducting EF coils are located outside the 12 TF coils and four 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. The cooling lines and rf ducts protrude through the shield doors so that high-pressure coolant disconnects can be located outside the vacuum boundary where small leaks could be tolerated.

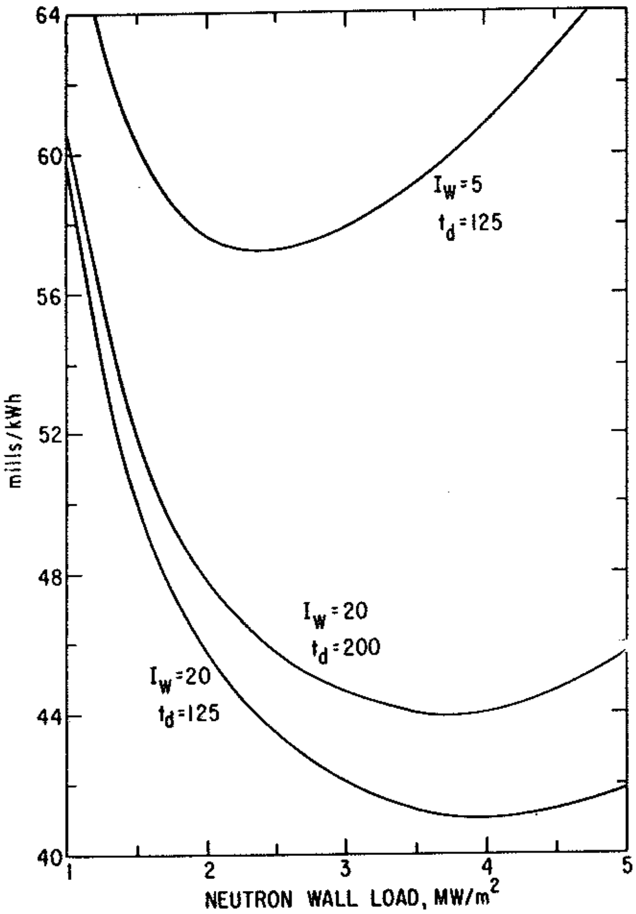

Fig. 1—Cost of energy as a function of neutron wall load. I_w is the integral neutron wall load in MW-yr/m² and t_d is the total downtime in days for replacement of the structural material. Results are based on fusion power of 3200 MW, aspect ratio of 3.6, plasma elongation of 1.6, and β_t = 0.067.

Plasma startup is accomplished by electrically breaking down the deuterium-tritium gas using a ~ 3-MW electron cyclotron resonant heating (ECRH) system, inducing 1 MA of plasma current with OH coils and building up and sustaining the 10-MA plasma current using an rf system. Plasma fueling is accomplished via gas puffing or possibly pellet or plasma gun injection.

During plasma operation the plasma impurities, including alpha particles, are removed using a limiter system and continuous vacuum pumping. The limiter consists of tantalum segments that form a continuous toroidal ring at the reactor outer midplane. The limiter is subjected to a peak heat flux of 4 MW/m² and is cooled with 150 C water, which is used for feedwater

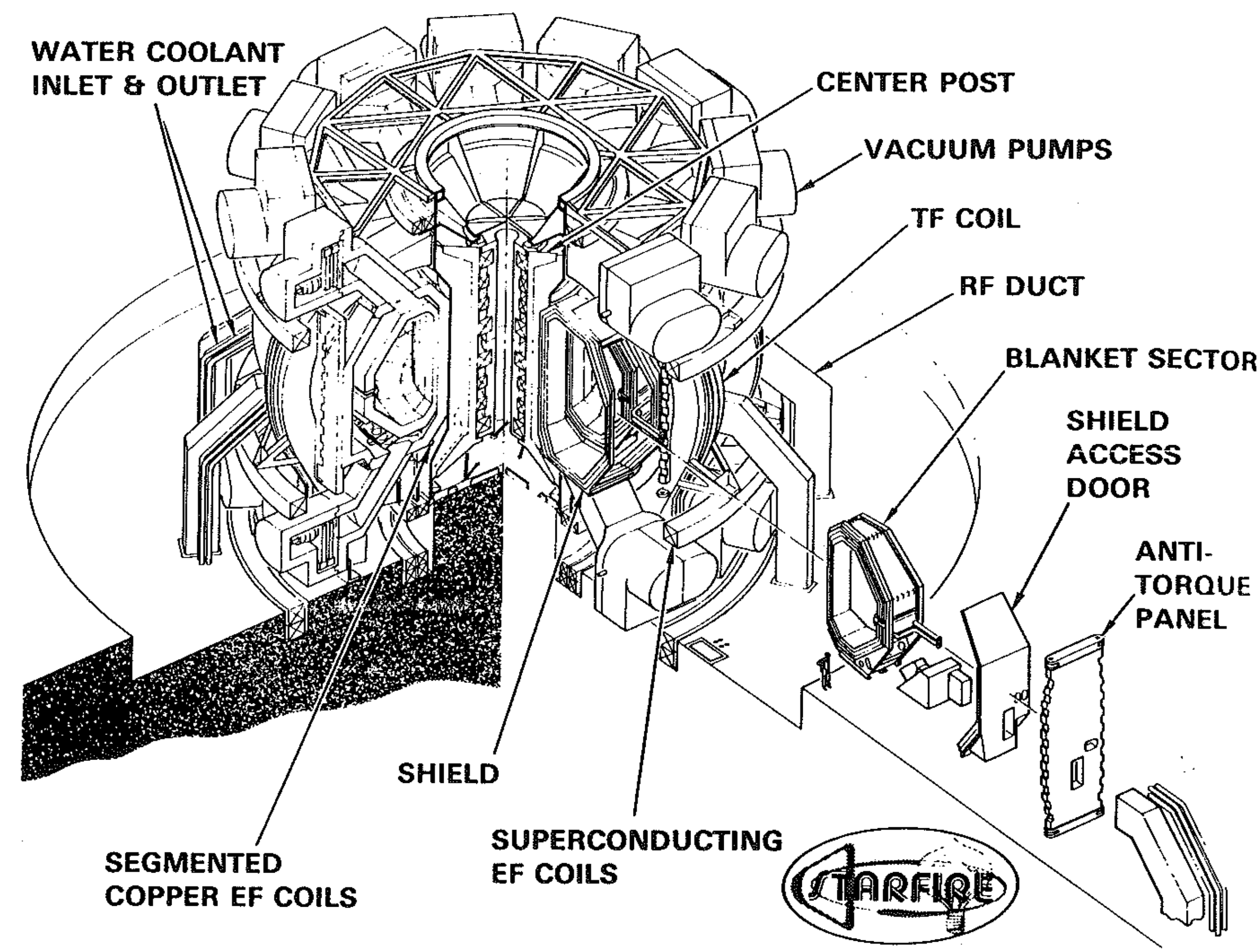

Fig. 2—STARFIRE reference design—isometric view.

heating. As particle impinge on the limiter, ~ 30 percent are directed 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. Pumps are rejuvenated hourly to minimize tritium inventory.

The limiter/vacuum system achieves a very high fuel utilization efficiency, having a tritium fractional burnup of 35 percent. As a result, tritium flow rates in the fuel cycle are very low, about 1 kg/day, or 250 g/GWth-day. By comparison, most previous reactor designs have a much lower fractional burnup, 1 to 10 percent, and the corresponding fuel cycles have to process 1 to 10 kg/GWth-day. The tritium inventory considered to be vulnerable to accidental release is < 400 grams.

The first wall is an integral part of the blanket structure. The blanket is segmented toroidally into 24 sectors to permit removal between TF coils. Two different sector sizes are used to permit location of the high-pressure coolant line disconnects outside the vacuum chamber. The first-wall and structural material is PCA stainless steel that operates at ~ 425 C maximum temperature when subjected to an average neutron wall load of 3.6 MW/m². The first-wall blanket is cooled by water with inlet and outlet temperatures at 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 20 MW-yr/

m² 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 outer shield is composed of 20 percent titanium, 20 percent lead, 30 percent B_4C and 30 percent H_2O, which offers significant environmental advantages with respect to minimum radioactive waste considerations. The shield is assembled from 12 sectors and 12 shield rings. Dielectric breaks are located in six of the shield rings near the outer surface of the shield to limit the radiation dose to 10^{10} rads. The Kapton dielectric seal is factory installed and designed for life-of-plant operation. Removable shield doors are located between TF coils to permit blanket sector removal. The shielding is ~ 1-m thick to reduce the gamma dose level to ~ 10^8 rads so that elastomer door seals can be used. Redundant seals and dielectric breaks are used to permit leak detection and isolation.

The reactor magnet system consists of 12 TF coils, 8 superconducting EF coils, 4 normal conducting EF coils, and 6 OH coils. Two of the OH coils are combined with the EF coils to simplify assembly. No intertwined supeconductors are used and the normal coils are segmented to permit maintenance. The TF coils have a common vacuum dewar at the inner coil leg and separate vacuum dewars on the outer leg. TF coil overturning moments are reacted through the 4 K center post and room temperature shear panels between adjacent TF coil outer legs. The room temperature TF coil case also supports the EF coils and the shield assembly. The vacuum pumps utilize an additional support frame.

A steam power conversion system without an intermediate heat exchanger is utilized to convert the reactor thermal energy to electrical power. Two separate heat removal circuits are utilized, one for the first wall/blanket and the other for the limiter. The power deposited in the limiter (200 MW) is used for feedwater heating, while the recoverable power (3800 MW) from the first wall and blanket is used to produce steam at 299 C and 6.3 MPa. The steam is then used in a turbine-generator unit for producing 1400 MW of electric power. The net electrical power is 1200 MW with 240 MW recirculating power for the rf system, coolant pumps, and other reactor subsystems.

CONCLUSIONS

In summary, the major features for STARFIRE include a steady-state operating mode based on a continuous rf lower-hybrid current drive and auxiliary heating, solid tritium breeder material with no liquid lithium, pressurized water cooling, limiter/vacuum for impurity control, most superconducting EF coils outside the TF superconducting coils, fully remote maintenance, and a low-activation shield. These features have resulted in a simplified tokamak reactor concept compared with previous studies while increasing the attractiveness of the reactor with respect to safety and environmental features.

Availability goals have been established as 85 percent for the reactor and 75 percent 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 five to ten years.

An important design consideration is the choice of the plasma impurity and alpha-particle removal concept. Investigations in this study indicate that modest pumping of helium with a limiter/pumping system (~ 25 percent of the alpha-particle flux) coupled with about a 1.5-T margin in the maximum toroidal field should eliminate the need for a divertor. This result is based on the provision that a significant portion of the

TABLE I
STARFIRE MAJOR DESIGN FEATURES

Net electrical power, MW	1200
Gross electrical power, MW	1440
Fusion power, MW	3490
Thermal power, MW	4000
Gross turbine cycle efficiency, %	36
Overall availability, %	75
Average neutron wall load, MW/m²	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 (lower hybrid)
TF coils material	Nb_3Sn/NbTi/Cu/SS
Blanket structural material	Austenitic steel (modified, PCA)
Tritium breeding medium	α-$LiAlO_2$
Neutron Multiplier	Zr_5Pb_3 (solid)
Wall/blanket coolant	Pressurized H_2O
Plasma impurity control	Low-Z coating + limiter and vacuum system + enhanced radiation + toroidal field margin
Primary vacuum boundary	At inner edge of shield

alpha-particle heating power can be radiated to the first wall rather than be deposited on the limiter. In general, a nondivertor option is greatly preferred from an overall reactor engineering point of view.

Another key design consideration is the location of the equilibrium field (EF) coils. The basic design approach is to locate almost all the EF coils outside of the toroidal field (TF) coils. All such EF coils would be superconducting. A limited number of segmented copper coils are located inside the TF coils, but outside of the blanket and shield.

Safety has played a major role in considering various blanket options. Solid tritium breeders instead of liquid lithium have been emphasized in this study. In addition, efforts have been made to minimize the tritium inventory in the plasma exhaust processing systems and the radioactivity induced in the materials in the magnets and shield.

ACKNOWLEDGMENT

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