AN OVERVIEW OF THE STARFIRE REFERENCE COMMERCIAL TOKAMAK FUSION POWER REACTOR DESIGN*

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Paper presented at and published in the Proceedings of the
4th ANS Topical Meeting on the Technology of Controlled Nuclear Fusion
14-17 October 1980, King of Prussia, PA

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*Work supported by the U.S. Department of Energy

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Summary

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 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, pressurized water cooling, limiter/vacuum system for impurity control and exhaust, high tritium burnup, superconducting EF coils outside the TF superconducting coils, fully remote maintenance, and a low-activation shield.

Introduction

A comprehensive conceptual design of the STARFIRE reactor and balance-of-plant has been developed.[1] The primary criteria for commercial attractiveness emphasized in the STARFIRE study are economics, safety, and environmental impact. The approach to meeting these criteria involved building upon experience from previous studies, developing new design concepts, and selecting features that simplify the engineering design and enhance reactor maintainability. Table 1 shows the key features of STARFIRE. The reactor is operated steady state with the plasma current maintained by lower-hybrid waves. This mode of operation results in a reduction in the plant capital cost and an increase in the plant availability. The capital cost savings are due to the elimination of electrical and thermal energy storage, derating of power supplies and the reduction in the reactor size, which is made possible by the increase in the permissible wall loading. The improvement in reactor availability is brought about by the increase in component reliability, elimination of material fatigue as a life-limiting effect in the first wall, and the reduction in the probability of plasma disruption occurrence. The reactor design is simplified by utilizing the lower-hybrid rf system, with its attractive engineering features, for the dual purpose of plasma heating and current drive. The problems associated

Table 1. Key Features of STARFIRE

- Steady-state plasma operation
- Lower hybrid rf for plasma heating and current drive
- ECRH-assisted startup
- Limiter/vacuum system for plasma purity control and exhaust
- All superconducting EF coils outside TF coils
- Vacuum boundary at the shield, mechanical seals
- Total remove maintenance with modular design
- Water-cooled, solid tritium breeder blanket with stainless steel structure
- All materials outside the blanket are recyclable within 30 yr
- Less than 0.5 kg of vulnerable tritium inventory
- Minimum radiation exposure to personnel
- Conventional water/steam power cycle with no intermediate coolant loop and no thermal energy storage

* Work supported by the U.S. Department of Energy.

with plasma initiation and startup have been eased by the use of electron cyclotron resonance heating to reduce the OH voltage.

The limiter/vacuum system concept has been selected for the plasma impurity control and exhaust system.[2] Compared to divertors, the limiter/vacuum system greatly simplifies the reactor design and improves its reliability and accessibility. Detailed analysis showed that the system can be designed to credible engineering standards.

The characteristics of the plasma operating point and the plasma support systems in STARFIRE are different from those in previous conceptual designs. The major differences are due to the choice of the steady state operation and the limiter/vacuum system. These choices were motivated by the desire to simplify the engineering design. It was assumed in the early stages of the design that these options could be developed in the STARFIRE time frame. Fortunately, results from recent plasma physics experiments on noninductive current drive and on limiters are very encouraging and they suggest that these options can be developed in the next few years.

A major effort has been devoted in STARFIRE to enhancing reactor maintainability and improving plant availability. The approach was to select design features and develop a design configuration that reduced the frequency of failure and shortened the replacement time. Relevant examples are: (1) steady state operation with lower hybrid current drive; (2) limiter/vacuum system for impurity control and exhaust; (3) vacuum boundary located at the shield with all mechanical seals (no welds); (4) all service connections (e.g., for high pressure coolant) are located outside the vacuum boundary (shield); (5) optimized modular design; (6) all superconducting EF coils are outside the TF coils; (7) conservative TF coil design; (8) fully remote maintenance permitting some repairs during reactor operation; (9) "remove and replace" maintenance approach (failed parts are replaced with spare parts and the reactor is operated while repairs are made in the hot cell) that minimizes downtime; (10) combining components for simplicity (e.g., TF coil room-temperature dewar provides support for the EF coils and shield); and (11) providing redundancy where it is justified (e.g., for the EF coils trapped below the reactor). These features as well as potential future improvements in component reliability provide optimism that the plant availability goal of 75% can be achieved.

The safety and environmental considerations have played a major role in the STARFIRE design effort. A solid tritium breeder was selected in preference to liquid lithium in order to minimize the stored chemical energy. The impurity control and exhaust system was selected and designed so that the tritium fractional burnup is maximized and the vulnerable tritium inventory in the fueling and vacuum pumping systems is minimized. Furthermore, the reactor design was developed to contain the tritium[3] with multiple barriers and minimize the size of potential tritium releases. The shield[4] was designed and all reactor materials selected to permit recycling of all materials outside the blanket in less than 30 yr. Radiation exposure of personnel has been minimized by the use of extensive remote maintenance operations and by providing adequate shielding. The use of resource-limited materials was minimized. Mechanisms for rapid reactor shutdown and auxiliary cooling systems have been incorporated into the design. The beryllium coating on the first wall and limiter provide an inherent safety feature that terminates the plasma burn if the metal temperature reaches ∿ 900°C. Calculations show that the reactor will be automatically shut down in less than one second, if a hot spot forms on 10% of the first wall, without the need for any active control system. No major damage, other than some first wall coating ablation, will occur.

The use of water coolant,[5] steam cycle and conventional materials in STARFIRE makes the heat transport and energy conversion system a state-of-the-art technology. The balance of plant[6] has been designed to maximize the utilization of current power plant features. However, the reactor hall, hot cell and tritium facility are unique to fusion reactors. The tritium facility utilizes current day design practices of the Tritium Systems Test Assembly (TSTA). The reactor building houses the reactor and modules for auxiliary systems that may become contaminated.

Reactor Overview

Reactor Configuration

The major reactor parameters for STARFIRE are listed in Table 2. The reactor design has a major radius of 7.0 m and operates at a first wall average neutron loading of 3.6 MW/m^2. The reactor delivers 1200 MWe to the grid in addition to providing 240 MWe for recirculating power requirements. The reactor operates with a continuous plasma burn and develops 4000 MW of useful thermal power. Approximately 3800 MW is provided to the main heat transport system and 200 MW is collected from the active limiter for use in feed water heating. An isometric view of the reactor is shown in Fig. 1 and a reactor cross-section is shown in Fig. 2.

The reactor configuration utilizes 12 toroidal field (TF) coils and 12 superconducting poloidal coils (EF and OH) located external to the TF coils. Additionally, four small normal conducting control coils (CF) are located inside the TF coils and outside the bulk shield to provide the necessary response time for plasma control while permitting good access for reactor maintenance. The magnet systems and shield are expected to last the full 40 year design life under normal operating conditions; however, provisions are incorporated for their replacement. Blanket sectors, including the limiters and rf ducts, will require replacement

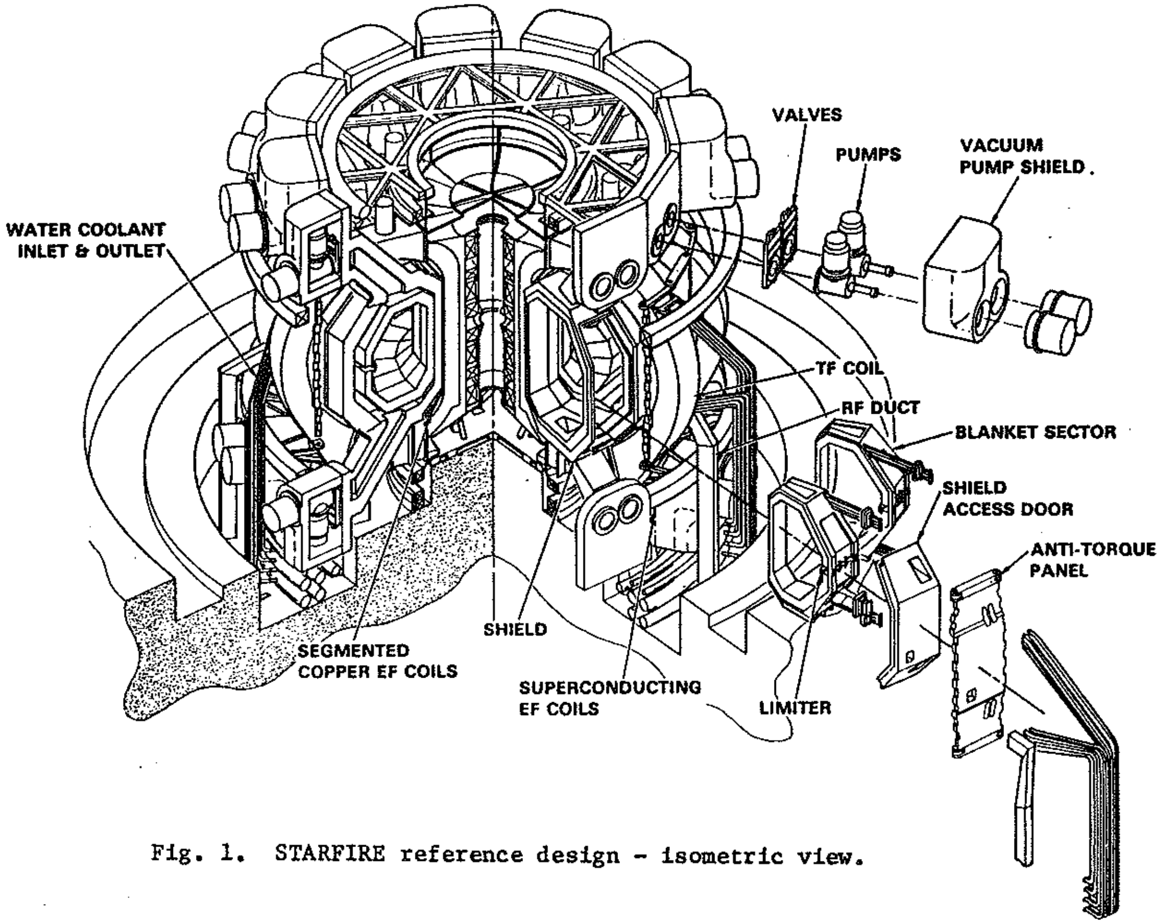

Fig. 1. STARFIRE reference design - isometric view.

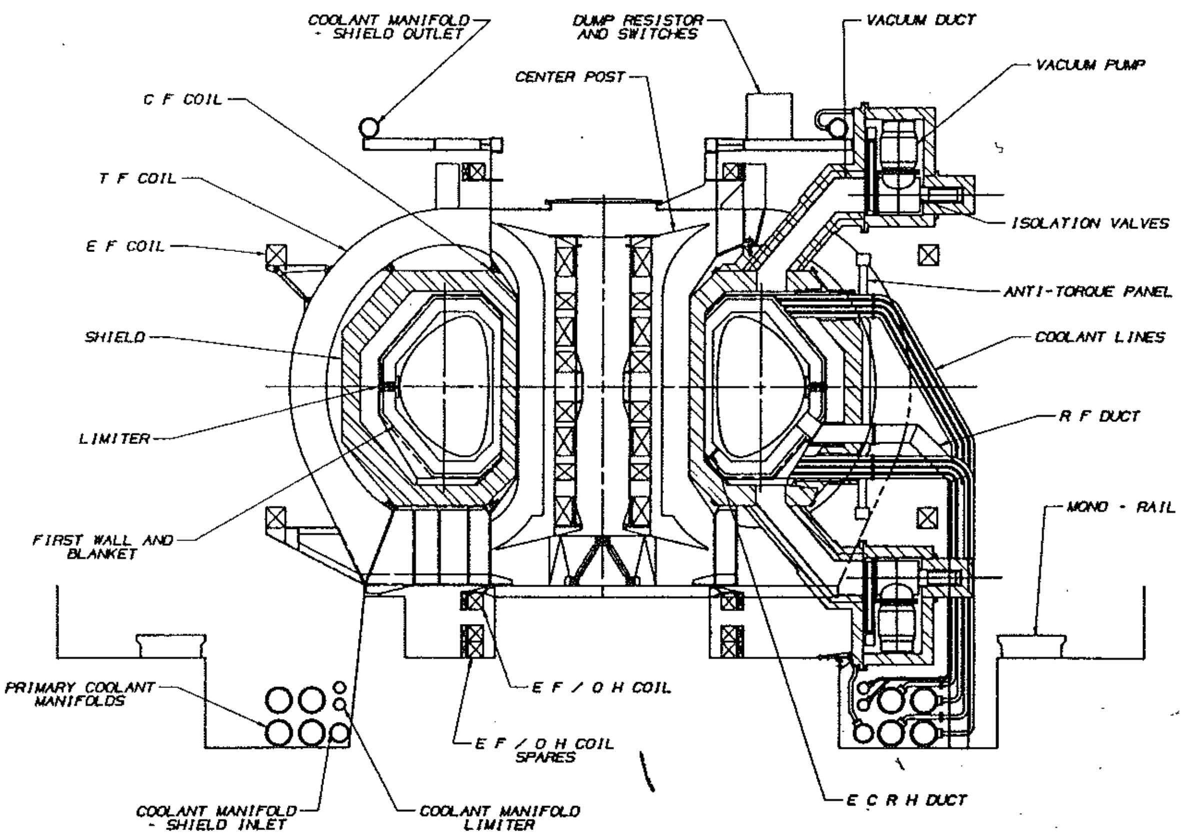

Fig. 2. STARFIRE commercial tokamak - cross section.

Table 2. STARFIRE Major Design Parameters

Net electrical power, MW	1200
Gross electrical power, MW	1440
Fusion power, MW	3510
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 (lower hybrid)
Plasma heating method	rf (lower hybrid)
Plasma startup	ECRH-assist
TF coils material	Nb_3Sn/NbTi/Cu/SS
Blanket structural material	PCA[a]
Tritium breeding medium	Solid breeder (α-LiAlO)
Wall/blanket coolant	Pressurized water (H_2O)
Plasma impurity control	Limiter and vacuum system supplemented by low-Z coating, enhanced radiation and field margin
Primary vacuum boundary	Inner edge of shield

[a] Primary Candidate Alloy (PCA), an advanced austenitic stainless steel.

every six years. Vacuum pumps and the isolation vacuum valves will require replacement every two years.

The TF coil, and hence the reactor, configuration was developed primarily by the desire to keep the superconducting EF coils external to the superconducting TF coils so that their replacement is possible without fabrication of a new coil on the reactor. External placement of the EF coils increases the incentive to reduce the TF coil size to minimize the stored energy of the EF system. The TF coil outer radius is constrained to 13 meters by the clearance required for shield installation.

The EF/OH coils inside the center post are grouped in two modules to simplify their removal from the top of the reactor without significantly increasing the overhead crane or building height. The outer EF coils and upper EF/OH coil can be removed vertically. Spares have been provided for the lower EF/OH coils that are trapped under the reactor because the inherent complications of replacing a failed coil, even if only once in every few plant lifetimes, make it cost effective.

The shield is assembled as twenty-four sectors to permit its installation between TF coils. The 12 sectors that fit under the TF coils also incorporate dielectric breaks in every other sector. The other sectors incorporate an access door and two vacuum ducts. The sectors are joined together by a welded vacuum seal and are not expected to require frequent replacement (i.e., they are life-of-plant components).

The vacuum boundary location was selected at the shield interior with access door seals located at the outer surface in order to (1) provide a convenient way of providing pumping for the limiter slot system, (2) minimize the complexities of providing a vacuum boundary at the blanket/first wall and (3) permit the inboard vacuum seals which have limited access to remain intact during maintenance. The vacuum seals that must be opened for maintenance were located at the outer shield surface to provide access for maintenance and to reduce the damage to seal materials by radiation exposure. The shielding is effective enough to permit use of elastomer seals which can be sealed repeatably and easily. The vacuum pumps were located at the top and bottom of the reactor to minimize the neutron heating on the cryopanels and to permit the pumps to remain in-place during blanket replacement.

The blanket was divided into large sectors to permit replacement with a minimum number of in-reactor maintenance actions. Twenty-four toroidal sectors of two different sector sizes are used to permit installation in the space between adjacent TF coils. The overall blanket installation was simplified by mounting the limiter, rf duct and ECRH duct to the sector for removal as a unit. Coolant connections to the blanket sector were located outside of the vacuum boundary to minimize the effects of irradiation on the joint and to permit use of less than high integrity "leak-tight" mechanical seals. The penetration through the vacuum boundary is sealed with elastomer seals located at the external shield surface.

The limiter consists of 96 elements that form a near continuous toroidal ring at the outer midplane of the blanket. Four limiters are mounted on each blanket sector in front of a slot through the blanket that provides a path for particles to a plenum. Particles are then pumped by 24 vacuum pumps at the top and bottom of the reactor. An additional 24 vacuum pumps are provided to permit pump rejuvenation every two hours.

Twelve rf ducts provide for heating and current drive of the plasma. These ducts are mounted in the blanket sector located between TF coils. An rf window and phase monitor are located in the duct near the shield while phase shifters, circulators and crossed field amplifiers are located in the reactor building basement where personnel access during operation is possible. Twenty-four ECRH ducts are provided for initial plasma breakdown and wall cleaning. Two ECRH ducts are located on each blanket sector between TF coils.

Fuel is provided to the reactor by extracting bred tritium from a solid breeding blanket and injecting it into the plasma via gas puffing. Two gas ports are provided. Gas enters the plasma through the limiter which incorporates a drilled passage to the innermost protrusion of the plasma at the outer blanket midplane.

Plasma Engineering

STARFIRE employs a DT burning, D-shape plasma to produce 3510 MW of fusion power. The plasma is operated at a moderate β of 6.7% and is moderately elongated, with a height to width ratio of 1.6. The major plasma parameters and plasma engineering features of STARFIRE are listed in Tables 3 and 4, respectively. The plasma current is driven in steady state with 90 MW of lower hybrid rf power. The first wall and all other components in the vacuum chamber are coated with Be. The impurity control system maintains a steady-state concentration of 14% helium and 4% Be in the plasma. The fairly low DT removal efficiency (15%) of the impurity control system permits a high fractional burnup of tritium. For the same reason, most of the plasma fueling is done automatically by DT neutrals recycling from the first wall and limiter. Additional fueling is done by gas puffing.

In order to minimize the heat transport load on the limiter, as well as to establish a thermal equilibrium, the plasma is operated in an "enhanced radiation" mode, whereby a small amount of high-Z material, nominally iodine, is added along with the fuel stream. This serves to radiate most of the heating energy and stabilizes the thermal operating point.

The plasma MHD equilibrium is of the low current, hollow profile type. The plasma position is controlled with two sets of coils,[7] a main equilibrium field (EF) coil set and a control field (CF) coil set. The main EF coils are superconducting and are located outside of the TF coils. They are used to provide the basic positional equilibria. The CF coils consist of small copper coils inside the TF coils and are used to control position and to stabilize against plasma

Table 3. STARFIRE Plasma Parameters

Parameter	Unit	Value
Major radius, R	m	7.0
Aspect ratio, A	—	3.6
Elongation, κ	—	1.6
Triangularity, d	—	0.5
Safety factor at limiter,	—	5.1
Average beta, β		0.067
Maximum toroidal field at coil, B_M	T	11.1
Toroidal field at plasma center, B_o	T	5.8
Plasma current, I_p	MA	10.1
Plasma volume, V_p	m^3	781
Average electron temperature, T_e	keV	17.3
Centerline electron temperature, T_{eo}	keV	22.5
Average ion temperature, T_i	keV	24.1
Centerline electron temperature, T_{io}	keV	31.3
Average fuel density, N_{DT}	m^{-3}	0.806×10^{20}
Center fuel density, N_{DT_o}	m^{-3}	1.69×10^{20}
Electron energy confinement time, τ_E	s	3.6
Ion energy confinement time, τ_I	s	10
Particle confinement time, τ_p	s	1.8
Fractional helium concentration, N_α/N_{DT}	—	0.14
Fractional beryllium contration, N_{Be}/N_{DT}	—	0.04
Fractional iodine concentration, N_I/N_{DT}	—	0.001
Fusion power, P_F	MW	3510 MW
Lower hybrid rf power to plasma, P_{rf}	MW	90
Average neutron wall load, P_{WN}	MW/m^2	3.6

disruptions. To further aid in the latter task, the first wall is designed with a time constant of 300 ms to stabilize against rapid vertical instabilities.

Because of the need to minimize the lower hybrid rf current drive power, the plasma density is lower and the plasma ion temperature is higher

Table 4. Plasma Engineering Features of STARFIRE

Operating Point	
Equilibrium type:	Elongated, D-shape, moderate β, hollow current profile
Equilibrium generation method:	Outside superconducting equilibrium field coil system
Position stabilization method:	Inside control field coils and conducting first wall with 300 ms time constant
Burn Cycle	
Startup time:	∿ 24 minutes
Method:	Tritium lean startup; vary rf power, DT density, T fraction; 5% per minute fusion power ramp
Normal shutdown time:	∿ 24 minutes
Emergency shutdown:	Induced disruption method, time < 3 s
Plasma initiation method:	5 MW electron cyclotron resonance heating
Burn method:	Steady state, lower hybrid current drive
Thermal stabilization:	Enhanced radiation mode operation by iodine injection
Fueling	
Fueling method:	Recycling DT plus gas puffing

than most previous tokamak reactor designs. The plasma is operated with $T_i > T_e$, which makes better use of the available β.

Most of the STARFIRE burn cycle[8] is substantially different from pulsed reactor burn cycles. Plasma breakdown is done with 5 MW of electron cyclotron resonance heating (ECRH) and does not require a high voltage OH coil. The startup period takes 24 minutes and conforms to the desire that the fusion power should be ramped at a 5% per minute rate, to minimize thermal problems in the energy recovery and conversion systems. The OH coil as well as the OH and EF power supplies have modest requirements compared to pulsed reactor requirements. The steady state burn phase of the burn cycle has a thermal equilibrium maintained by the addition of iodine.

Several types of shutdown scenarios have been developed for STARFIRE. The normal shutdown is basically the reverse of the startup period, whereby the fusion power is reduced at a 5% per minute rate by reducing the tritium fraction in the plasma. There are three types of emergency shutdowns. The fastest is an "abrupt" shutdown whereby a plasma disruption is induced by injecting excess high-Z material. There is a more orderly "rapid" shutdown which also uses a disruption, but where most of the plasma energy is radiated away prior to the disruption. Finally, a naturally occurring "ablative induced shutdown" has been identified which occurs as a result of a hot spot formation on the first wall or limiter.

Various fueling options for STARFIRE were studied. The high fractional burnup rate of 42% in STARFIRE permits a fairly low fueling rate from an external source. Gas puffing is the most desirable engineering option and has been adopted as the STARFIRE fueling method.

Plasma Heating and Current Drive

The design of a tokamak reactor which can run in a steady state mode is basically different from the design of a pulsed tokamak, because the circulating power required to sustain the toroidal current against collisional dissipation may be a substantial fraction of the power plant's electric output. Consequently, the STARFIRE design focused on efforts to minimize the circulating electric power for steady state operation, and the resulting lower-hybrid rf system was optimized with this goal.[9] In addition, the same system appears adequate to provide auxiliary heating during the startup phase to bring the plasma to ignition temperatures.

One obvious means of reducing rf power to the reactor is the selection of operating regimes with the lowest plasma currents. Generally, the higher the plasma current, the higher is the stable β_t which can be achieved. The increasing rf power required at higher β_t motivated the selection of a comparatively modest design value - $\beta_t = 6.7\%$. It was shown that hollow current density profiles can have favorable stability while requiring less total current than more conventional centrally peaked profiles.

Lower hybrid current drive theory shows that the rf driving power is proportional to the local electron density where the current is generated, which makes the hollow current profiles especially attractive. In addition, for a fixed β, the average electron density ($\overline{n}_e$) may be reduced by operating the plasma at a higher temperature. Above 20 keV, the decreasing fusion reactivity of DT tends to offset these reductions in the rf power at low n_e. (Maximum Q occurs in the range 20-30 keV.) Despite low Z operation, the net electric output peaks for $T_e \simeq T_i \approx 11$ keV. However, the capital outlay for rf power supplies at 11 keV far exceeds that needed for auxiliary heating to ignition. The minimum cost of electricity appears when $T_e = 17$ keV, $T_i = 24$ keV, and $n_e = 1.2 \times 10^{20}$ m^{-3}, which results in a fusion power of 3510 MW.

Plasma Impurity Control and Exhaust

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

These goals are found to be best satisfied by a limiter/vacuum system together with a beryllium coating on the first wall, limiter, and all other surfaces exposed to the plasma. 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 high-Z material, e.g., iodine, along with the DT fuel stream. The iodine atoms enhance the line-and-recombination radiation over most of the plasma volume. The helium removal efficiency of the limiter/vacuum system is intentionally kept low for three reasons: (1) to reduce the heat load on the limiter; (2) to simplify the vacuum system and reduce radiation streaming; and (3) to minimize the tritium inventory tied up in the vacuum and tritium processing systems. The major features of the STARFIRE impurity control and exhaust system are presented in Ref. 2.

The basic principles of how the limiter works are rather simple. Ions that hit the front face of the limiter will be neutralized and reflected back into the plasma. Ions that enter into the limiter slot hit the back surface and are neutralized. Some of the scattered neutrals will directly reach the limiter duct and follow a multiple-scattering path into the plenum region and into the vacuum ducts where they are pumped out by the vacuum pumps. Other particles neutralized at the back surface of the limiter will scatter back in the direction of the plasma. These neutrals have a high probability of being ionized and returned back to the limiter surface. Calculations show that this trapping or "inversion" effect is so large for helium that $\sim 90\%$ of the helium entering the limiter slot will be pumped. Hydrogen can charge-exchange as well as be ionized. These charge-exchange events significantly reduce the inversion probability for hydrogen because the resulting neutral will tend to make its way out of the slot region into the plasma. Therefore, the beneficial effect of higher helium pumping probability and enhanced hydrogen recycling into the plasma is obtainable in the limiter/vacuum system.

Magnets

The superconducting toroidal field (TF) coils and poloidal field (PF) coils have been designed[10] with a cabled conductor consisting of a copper stabilizer and NbTi superconductor, except for the inner turns of the TF coils, where field requirements in excess of 9 T have led to the choice of Nb_3Sn superconductor. In both the TF and PF coils, each cable conductor is contained in its own structure, which bears against the structure of neighboring conductors to transmit radial and axial forces. All coils are bath-cooled by pool boiling liquid helium at 4.2°K. The structure around the conductor contains transverse and longitudinal channels, to carry liquid helium to where cooling is needed and to carry helium vapor away.

The TF coils bear radially inward against the G-10 fiberglass-epoxy centerpost support cylinder, within whose bore is located the inner ohmic heating (OH) and equilibrium field (EF) coils. All of these elements share a common vacuum volume. The centerpost region is surrounded by a common vacuum tank section with individual vacuum tanks surrounding each TF coil outer leg.

The EF and OH coils must be superconducting; normal conducting coils would consume an unacceptable amount of power. Being superconducting, these coils must be outside the TF coil system to facilitate maintenance and possible coil replacement. External location of the EF coils exposes the outer TF coil region to large fields, which interact with the TF coil current to generate large out-of-plane (overturning) loads. The magnitude of the overturning moment on each coil is about 1.5×10^9 N-m. The centerpost region of the TF coil reacts a small portion of this load. The major portion of the load is reacted in the outer curved coil region, where the distributed out-of-plane load is transmitted from the helium vessel to the surrounding vacuum tank by closely packed pairs of cold-to-warm tiebars. The individual coil vacuum tanks are in-turn supported by substantial intercoil shear panels.

First Wall/Blanket/Shield System

The development of the reference STARFIRE first-wall/blanket design[11-13] involved numerous tradeoffs in the materials selection process for the breeding material, coolant, structure, low-Z coating, neutron multiplier and reflector. The coolant and structural material selections were greatly influenced by the choice of the solid breeder concept which was used as a basis for the STARFIRE design. The most important criteria considered in the selection of potentially viable solid breeding materials include breeding performance, chemical stability, compatibility and tritium release characteristics. Of the two types of solid breeding materials considered as primary candidates, viz., intermetallic compounds and oxide ceramics, only selected ceramics appear to have satisfactory tritium release characteristics. The α-$LiAlO_2$ is selected for the reference design on the basis of the best combination of these critical materials requirements. It is one of the most stable compounds considered and compatibility should not be a major problem; however, adequate tritium breeding is attainable only with the aid of a neutron multiplier. The high tritium solubility and greater reactivity with the structural materials were primary factors in the elimination of Li_2O as the reference breeding material.

Pressurized water, both H_2O and D_2O, and helium were considered for the coolant. Major concerns regarding the use of helium relate to difficult neutron shielding problems, large manifold requirements, leakage into plasma chamber, lower tritium breeding because of the large structure requirements and the high temperatures required for the energy conversion system. An acceptable structural material for use with high temperature helium in a radiation environment has not been identified. Also, design constraints associated with the use of helium as a first-wall coolant appear to be prohibitive. Major advantages of the water coolant are its characteristically low operating temperature and its excellent heat transfer characteristics. However, the use of water with the intermetallic compound breeder materials is probably not acceptable because of the high reactivity, and hence, safety concern. Although D_2O has several neutronic advantages compared to H_2O, the cost is considered prohibitive.

The choices of breeding material and coolant limit the number of viable candidate structural materials. Key factors in the selection of the advanced austenitic stainless steel relate to the steady state reactor operation and the low operating temperatures characteristic of a water-cooled system. Because of the high thermal stress factor associated with austenitic stainless steel, acceptable first wall lifetimes could not be attained with a cyclic burn. Also, radiation damage effects are less severe at the proposed operating temperatures than at temperatures above 500°C.

The low-Z coating concept for the first-wall is incorporated as part of the plasma impurity control system. The low-Z coating concept provides flexibility in that the structural material can be selected primarily on the basis of structural requirements and the coating can be selected primarily on the basis of surface-related properties. Favorable properties such as high thermal conductivity, high heat capacity and compatibility with hydrogen were important considerations in the selection of beryllium as the first-wall coating/cladding material. A primary consideration in the selection of the candidate coating/cladding is that it can be used on all components exposed to the plasma. This is important because considerable redistribution of the material throughout the chamber is expected as a result of sputtering and ablation.

An effective neutron multiplier is required to obtain adequate tritium breeding with the $LiAlO_2$. Two candidate materials are proposed. Beryllium provides good neutronics performance and can be easily incorporated into the blanket design since it has low density, high thermal conductivity and high heat capacity. Because of the concern regarding limited resources of beryllium, an alternate neutron multiplier Zr_5Pb_3, is also proposed. This compound retains some of the beneficial characteristics of lead but remains solid at the operating temperatures.

A schematic diagram of the reference STARFIRE blanket concept is given in Fig. 3. The water-cooled blanket module, with a thickness of 68 cm, consists of 1-cm thick first wall, a 5-cm thick neutron multiplier, a 1-cm thick second wall, a 46-cm thick breeding zone, and a 15-cm thick reflector zone that contains the blanket support structure and the manifolding. The modules are 2-3 m wide by ∿ 3 m high depending on the location within the reactor. The module walls and all support structures in the high-radiation zone are fabricated from an advanced low-swelling austenitic stainless steel. The internal structure is integrally cooled to remove the nuclear heating and maintain the structure below 400°C.

The first wall, which is a water-cooled austenitic stainless steel panel coil, is an integral part of the blanket module. The corrugated plasma side of the first-wall panel is constructed of a 1.5-mm thick advanced austenitic stainless steel. The 3.5-mm thick back plate is formed from the same material. The pressurized water coolant is maintained between 280 and 320°C throughout the first wall and blanket. For the average neutron wall loading of 3.6 MW/m^2, the average surface heat flux on the first wall is 0.92 MW/m^2 with a peak-to-average value of ∿ 1.2. The maximum structural temperature in the stainless steel wall is ∿ 450°C for the reference conditions. For steady state operation at these relatively low temperatures, an estimated wall design life of six years is considered reasonable for the advanced austenitic stainless steel. The proposed panel-type construction provides integral cooling of the blanket wall and avoids the necessity for a large number of pressure boundary tube welds in the high radiation zone. Also, the panel-type structure is perceived to have less vibration problems than an unsupported tube bank.

A ∿ 1-mm thick beryllium coating or cladding on the first wall serves to protect the plasma from the high-Z wall material. This thickness will provide sufficient material to withstand the predicted surface erosion for the reference blanket lifetime of six years. The beryllium coating/cladding on the inboard wall will also accommodate the projected number (∿ 10 per wall lifetime) of plasma disruptions for the assumed conditions.

The 46-cm tritium-breeding zone consists of a packed bed of α-$LiAlO_2$ with 1.25-cm diameter stainless steel coolant tubes spaced appropriately throughout the zone to maintain a maximum breeder temperature of 850°C. The spacing of the horizontal tubes increased from ∿ 2 cm at the front of the breeder zone to ∿ 10 cm at the back. The nominal coolant pressure is 15.2 MPa (2200 psi) with a coolant inlet temperature of 280°C and an outlet temperature of 320°C. The relatively low

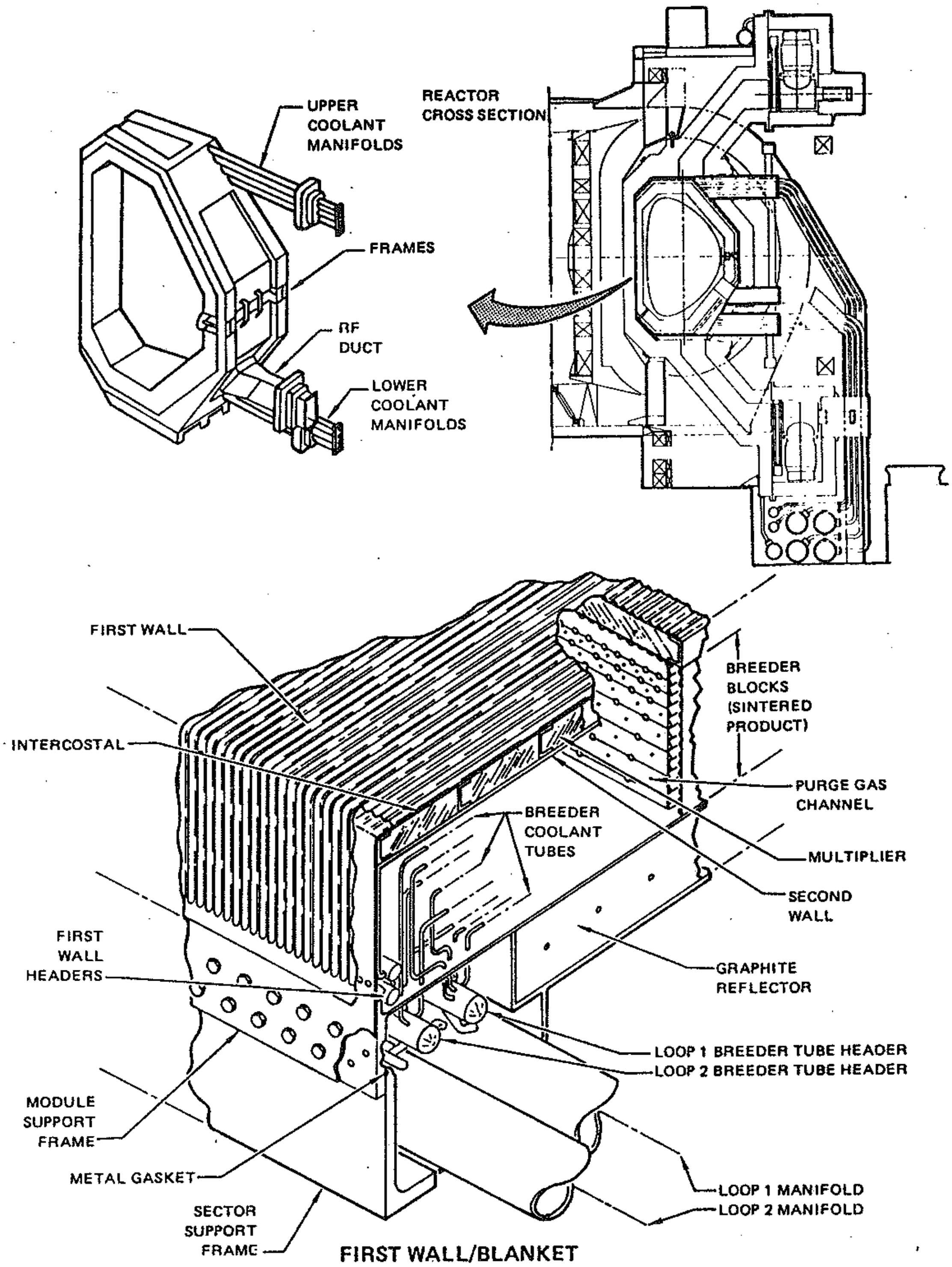

Fig. 3. STARFIRE blanket concept.

temperature of the austenitic stainless steel tubes (< 400°C) and the oxide film on the water side of the tubes provide an adequate tritium barrier for inleakage into the coolant. Natural lithium is used for the beryllium neutron multiplier option; however, 60% enriched ^6Li is required to achieve adequate tritium breeding with the Zr_5Pb_3 neutron multiplier option. The $LiAlO_2$ is in the form of low density (60%) sintered product with a tailored bimodal pore distribution, i.e., a small grain size (< 1 μm) and a fine porosity within particles that are fairly coarse (∿ 1 mm) with a much coarser porosity between particles. The sintered $LiAlO_2$ is perforated with ∿ 2-mm diameter holes through which low-pressure (0.5 atm) helium passes to recover the tritium from the breeder. The low density ceramic with a tailored microstructure is proposed to facilitate percolation of tritium (as T_2O) to the helium purge channels.[12] A breeder lifetime of six years before lithium burnup becomes excessive is considered feasible.

A two-loop coolant system is provided in the blanket to reduce the consequences in the event of a loss-of-flow or loss-of-coolant accident. One loop provides coolant for the first-wall and alternate tube banks in the breeder region beginning with the first row of tubes. The second loop provides coolant for the second wall and the remaining coolant tubes in the blanket. Under the reference plasma shutdown conditions, cooling provided by either loop is sufficient to prevent excessive temperatures in all regions of the blanket. The two-loop concept will also reduce the pressure release and activation release in the event of a coolant-tube failure.

For plasma stability, an electrical conducting path equivalent to 2 cm of stainless steel is required near the first wall. The conductivity of the first wall and the neutron multiplier meets this requirement in the modules. Bimetallic contacts between the modules are provided adjacent to the multiplier region to complete the current path. Upon cooling, these contacts recess into the module wall to allow for sector removal.

The design criteria for the shield included: (a) protection of all reactor components from excessive nuclear heating and radiation damage; (b) the biological dose rate outside the shield at 24 h after shutdown should be sufficiently low, ∿ 1-2 mrem/h, to facilitate personnel access into the reactor building; and (c) material composition and dimensions of the shields were selected so that all reactor components, including the shields, outside the blanket are recyclable within about 30 years.

The inboard blanket/shield thickness is 1.2 m. This includes space for 9-cm vacuum gaps between the blanket and shield, shield and TF coils and thermal insulation inside the TF vacuum tank; 3-cm vacuum tank (alloy Fe14Mn2Ni2Cr, referred to as Fe-1422) and 7-cm helium vessel (stainless steel). The inner blanket is 37-cm thick and must breed tritium as the breeding margin with the solid breeder is small. The inboard shield is 54 cm thick and consists of alternating layers of tungsten and boron carbide with water for cooling and Fe-1422 for structure.

The maximum radiation-induced resistivity in the copper stabilizer after 40 yr operation, is 2.2×10^{-10} Ω·m, assuming a magnet anneal every 10 yr with 83% recovery. The maximum radiation dose in the shield dielectric break is 7.4×10^7 Gy after 40 yr operation.

The outboard bulk shield is 1.1 m-thick. It includes 2-cm shield jacket at the plenum region with the rest divided into three regions. The first region, in the high flux zone, is 0.5-m-thick and has a material composition of 5% Ti alloy + 65% TiH_2 + 15% B_4C + 15% H_2O. The second region, middle zone, is 0.40-m-thick with the material composition as 70% Fe-1422 + 15% B_4C + 15% H_2O. The third region, outer zone, is 0.18-m-thick of Fe-1422. The biological dose rate in the reactor building decays very rapidly and reaches ∿ 1.5 mrem/h at 24 h after shutdown. Although the STARFIRE plans call for fully remote maintenance, the dose rate of 1.5 mrem/h shows that personnel access into the reactor building with all shielding in place is permissible within one day after shutdown. This provides a degree of confidence in improving the plant availability factor, if desired, by allowing some maintenance tasks to be carried out in a contact or semi-remote mode.

One of the important shield considerations is radiation streaming through void regions that penetrate the blanket and bulk shield regions. In general, the direct radiation flow in neutral beam ports, divertors, etc., has been one of the primary sources of design complexity and shielding difficulties in previous tokamak designs. In the STARFIRE design, a serious effort has been devoted to minimizing possible design difficulties associated with these penetrations. The STARFIRE design features the selection of a lower-hybrid rf system in preference to a neutral beam heating system and a limiter impurity control concept rather than divertors. A great advantage of the rf and limiter/vacuum systems is the elimination of any direct radiation streaming path from the plasma to the reactor exterior. These design features have helped reduce the shielding problems to a manageable level and brought about overall simplicity in the shield design.

The importance of the major radioisotopes has been examined in terms of radioactivity and radioactivity-related parameters such as biological hazard potential (BHP) in air and BHP in water. With regard to material recycling, an effort has been devoted to establishing a criterion for potential material recycling categorization. In addition to the conventional waste level classification by specific radioactivity concentration (Ci per unit volume), a criterion based on the contact biological dose has been suggested and used for the recycling analysis.

It is found that the magnitude of high-level, long-term radwaste from STARFIRE is dominated by the PCA first wall/blanket structural material. It is shown, however, that the average annual discharge rate of PCA from STARFIRE is only about 9.5 m^3 ($\sim$ 75 metric tons in weight) which is considerably less than a typical annual discharge of high-level waste from an LMFBR of the same power. The activation level of most reactor components external to the blanket, including the major penetration subsystems, decays to a category of low-level waste in 30 yr at most. Thus, the potential for recycling of materials from most reactor components is excellent.

Heat Transport and Energy Conversion

The thermal energy deposited in the blanket, first wall and limiter is delivered via the heat transport system to the steam power conversion system where it is converted to electricity. Two separate heat removal systems are utilized, a dual loop circuit for the blanket/first wall and a single loop for the limiter which is used for feedwater heating. The power flow diagram for STARFIRE is shown in Fig. 4.

Tritium Systems

The high fractional burnup (0.42) for STARFIRE results in a minimized tritium inventory in all fuel processing systems. This reduces the magnitude of a possible tritium release in the reactor building to approximately 10 g (as T_2O). In the tritium facility, as much as 50 g of T_2 could be released if multiple failures occur in an isotope separation unit.

The inventory in the STARFIRE plant is designated "vulnerable" or "nonvulnerable" depending upon the degree of control which can be enforced on a system and also the physical state of the tritium in that system. The tritium within the blanket (10 kg) is considered "nonvulnerable" since it is tenaciously retained by the solid breeding material and thus is relatively immobile. The tritium in the pump and fuelers (in the reactor building) is considered "mobile". The total "vulnerable" inventory for STARFIRE is less than 400 g.

Plant Construction

The schedule for plant construction shows that six years will be required between the first concrete is poured until initial power delivery. The pacing time in the construction

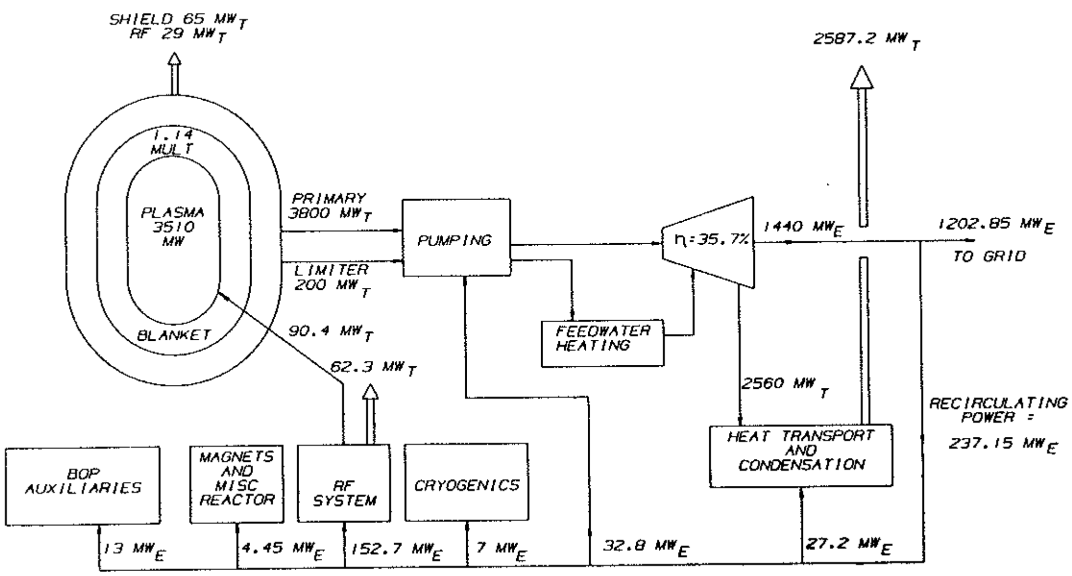

Fig. 4. STARFIRE power flow diagram.

schedule is the reactor building and reactor. Approximately three years are required to repair the reactor building for initial installation of the reactor components. Two years are required for reactor erection and one year is allowed for reactor and plant tests. Reactor building construction proceeds until the roof over the reactor area is complete and the 600 tonne overhead crane is complete. One end of the reactor building is left open and temporary crane rails are extended to permit direct entry of major reactor components. Modularization, with factory assembly and checkout, of reactor components is used where possible. Off-site winding of magnets and fabrication of blanket and shield sectors is planned with water shipment to the site.

Operation and Maintenance

A goal of the STARFIRE design has been to maximize utility compatibility not only in current practice but also with anticipated trends for future utility operations. Startup power is drawn from the grid and after plasma initiation the reactor power is brought up slowly to minimize the thermal stress effects on the blanket and steam generators. Once operating, the plant has the ability to load-follow at a rate of 5% of rated power increase or decrease per minute, although the plant is designed as a base load unit. The plant will normally operate continuously with one scheduled shutdown per year for maintenance. Once the reactor is shut down, it can be restarted to full power in approximately 0.5 h; however, approximately 12 h will be required for a restart if the TF coils are discharged for maintenance or if the vacuum chamber has been breached briefly. After major vacuum chamber breaches, 36 hours are required for restart.

The plant availability goal is 75% and the system reliability requirements have been established and accordingly based on the projected time for replacement of system components. The reactor maintenance schedule was developed to fit within the typical utility balance-of-plant maintenance scenario. It includes an annual shutdown for four weeks (28 days) to perform maintenance and inspection, and a four-month (120 days) shutdown every ten years for overhaul of the turbine-generator. During this period, the TF coil system is also annealed. This results in an average annual scheduled outage of 37 days. In addition, current utility experience indicates that approximately 20 days downtime per year is caused by failures in the balance-of-plant (BOP).

The 37 days of reactor scheduled maintenance is derived from the assumption that the reactor will be maintained simultaneously with the BOP on a noninterference basis. The total of 57 days outage required for BOP maintenance leaves 34 days for reactor unscheduled maintenance if the 75% availability (91 days outage/year) is to be achieved.

Economics

The total direct and indirect costs are \$2000/kWe in 1980 constant year dollars and \$2665/kWe in 1986 then-current dollars. The total busbar energy cost is 35 mills/kWh in 1980 constant year dollars and 67 mills/kWh in 1986 than-current dollars. These costs are higher than are currently being projected for new fission plants, but fusion power plants, similar to STARFIRE, are expected to become competitive as the cost of fissile fuel continues to escalate compared to the negligible cost of the fusion fuel.

Conclusions

The results of the STARFIRE study have increased our confidence in the potential of tokamaks as power reactors. The study has identified new and important directions for the development of fusion reactors, in general, and tokamaks, in particular, to further enhance their potential for commercial applications.

The results of the STARFIRE study show important incentives for developing the steady state option for tokamaks. Steady-state operation offers many engineering, technological and economical benefits in commercial reactors. Among these are: reactor reliability is increased, serious concerns about material fatigue are eliminated, electrical and thermal energy storage systems are not required; higher neutron wall load, and hence smaller size reactors, are acceptable; and the frequency of plasma disruption occurrence is greatly reduced. It has been estimated that the benefits of steady state can result in a saving in the cost of energy of as much as 30%.

Extensive efforts were made in STARFIRE to minimize the electrical power requirements for the LH current driver. Nevertheless, the reference design calls for 150 MW of electrical power to drive the relatively low plasma current of 10.1 MA. This represents $\sim$ 10% of the plant gross electrical output, which is relatively large. Therefore, strong incentives exist for additional efforts to further improve the performance of the LH current driver. Other potential current drivers must also be seriously explored.

The impurity control and exhaust system is one of the key components in a fusion reactor. It has a substantial impact on the engineering simplicity, reliability, maintainability, economics and safety of the power plant. Divertors and divertorless options were surveyed. It was concluded that the limiter/vacuum (also called "pumped" or "active" limiter) concept is a very attractive option for power reactors. It is relatively simple and inexpensive and deserves serious experimental verification.

The STARFIRE study finds it an important design approach to radiate most of the alpha-power from the plasma to the large surface area of the first wall. This reduces the heat load on the particle collection medium (limiter or divertor target plate) to a manageable level and it deposits more energy in the primary coolant of the first wall. One means of enhancing plasma radiation is by injecting small amounts of high-Z material along with the DT fuel stream. The large ignition margin in commercial reactor-size plasmas makes operation in such an enhanced radiation mode feasible.

A low-Z coating on all surfaces exposed to the plasma will probably be required in future tokamak reactors unless very low plasma edge temperatures can be established and maintained. Beryllium appears to be one of the best choices for the low-Z coating. Erosion of the limiter coating is predicted to be large but redeposition seems to extend the coating life to an acceptable level. However, there is a need for experimental results and theoretical work on the physics of the scrape-off region and the performance of low-Z coatings. There is also a need to develop in-situ low-Z coatings techniques for fusion reactor applications.

The results of STARFIRE indicate that a high efficiency exhaust system is not necessarily desirable. It is very beneficial to keep the removal efficiency low so that the tritium fractional burnup is high. This reduces the gas load in the exhaust system and simplifies the vacuum system design in addition to lowering the vulnerable tritium inventory in the fueling and vacuum systems.

Safety considerations provide major incentives for the development of solid breeders. Serious efforts have been devoted in STARFIRE to evaluation of solid tritium breeders and to the development of a design that optimizes their performance. The results are encouraging and show that the solid breeder option should continue to be pursued. However, the results of the detailed analyses in STARFIRE indicate potentially serious problem areas that must be further investigated before the viability of solid breeder blanket concepts can be accurately assessed. The most critical of these problems concerns the tritium release characteristics of solid breeders. Radiation effects, such as radiation-induced trapping of tritium within the grains and pore closure, may increase the tritium inventory in the solid breeders to unacceptably high levels.

Trade-off studies comparing helium and water coolants were performed. The results show clear advantages for the use of pressurized water for the STARFIRE conditions. The study also identified the key technology development requirements that are necessary for effective utilization of the helium cooling option.

Results of the economics analysis for the STARFIRE tokamak power plant indicate that fusion reactors can be developed to be economically competitive. The cost of energy estimated for STARFIRE is comparable to that of future light-water fission reactors and lower than for coal power plants. There are, of course, uncertainties in predicting now the cost of energy for future fusion reactors. However, there appears to be no fundamental reason that fusion will not be economically competitive.

Simplifying the reactor design has been a key approach in STARFIRE to enhancing component reliability and maintainability. The choices of the lower-hybrid current driver and the limiter/vacuum system concept have contributed significantly to simplifying STARFIRE. Other features found important in enhancing reactor maintainability include: modularity; locating the vacuum boundary at the shield with all mechanical seals; placing all service connections outside the vacuum boundary; and locating all superconducting EF coils outside the TF coils. A low number, 12, of TF coils was used to increase accessibility. There remains a great incentive for further reducing the number of TF coils. Therefore, more accurate information on the allowable field ripple in reactor-size plasmas is needed. The STARFIRE maintenance plan calls for a "remove and replace" approach; i.e., the failed components are replaced with stand-by units and the reactor is operated while the failed parts are repaired in the hot cell. This approach seems necessary in order to achieve reasonable availability goals.

The safety and environmental considerations have been emphasized in STARFIRE. The choice of a solid breeder in preference to liquid lithium was motivated by the desire to minimize the stored chemical energy. Significant effort was devoted to minimizing the vulnerable tritium inventory. This was achieved by selecting the limiter/vacuum system and designing it for a low particle removal efficiency in order to maximize the fractional tritium burnup. The reactor was designed to contain the tritium with multiple barriers and to minimize the size of tritium release. The maximum tritium release in any single accident was estimated to be 10 g.

No runaway accident that could pose a major risk to the public can be identified for STARFIRE. Furthermore, no plausible scenario could be formulated for the release of radioactive materials from the blanket (excluding corrosion products in the primary coolant loop) to the outside of the reactor building. In addition, mechanisms for rapid reactor shutdown have been incorporated into the design and auxiliary cooling systems are provided to serve as a backup in cases of off-normal conditions involving the primary coolant. A dual primary coolant loop system was designed to avoid complete loss of coolant. The beryllium coating on the first wall and limiter provides an inherent safety feature that terminates the plasma burn if the metal temperature exceeds $\sim 900°C$. Calculations show that the reactor will be automatically shut down in less than one second if a hot spot forms on a small area (< 10%) of the first wall without the need for an active control system. No major damage, other than ablation of some of the coating on the first wall, will occur.

References

1. C. C. Baker, et al., "STARFIRE - Commercial Tokamak Fusion Power Plant Study," Argonne National Laboratory, ANL/FPP-80-1 (1980).

2. M. Abdou, et al., "A Limiter/Vacuum System for Plasma Impurity Control and Exhaust in Tokamaks," Fourth ANS Topical Meeting on the Technology of Controlled Nuclear Fusion, King of Prussia, Pennsylvania (October 14-17, 1980).

3. P. A. Finn, et al., "Tritium Handling Considerations for ETF and STARFIRE," ibid.

4. J. Jung and M. Abdou, "Importance of Shield Design in Minimizing Radioactive Material Inventory in Tokamaks," ibid.

5. M. A. Abdou and D. Graumann, "The Choice of Coolant in Commercial Tokamak Power Plants," ibid.

6. J. Kokoszenski and D. Graumann, "Power Conversion and Balance of Plant Considerations by the STARFIRE Commercial Tokamak Reactor," ibid.

7. K. Evans, Jr., et al., "STARFIRE Poloidal Coil Systems," ibid.

8. J. N. Brooks, et al., "The STARFIRE Burn Cycle," ibid.

9. D. A. Ehst, et al., "Lower-Hybrid Heating and Current Drive System for the STARFIRE Tokamak," ibid.

10. J. S. Alcorn, L. R. Turner and S. T. Wang, "Superconducting Toroidal and Poloidal Field Coil Systems for STARFIRE," ibid.

11. D. L. Smith, et al., "First-Wall/Blanket Materials Selection for STARFIRE Tokamak Reactor," ibid.

12. Y. Gohar, et al., "Neutronic Optimization of Solid Breeder Blankets for STARFIRE Design," ibid.

13. D. L. Smith, et al., "Analysis of In-Situ Tritium Recovery from Solid Fusion-Reactor Blankets," ibid.