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TOKAMAK REACTOR STUDIES

by

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ABSTRACT

This paper presents an overview of tokamak reactor studies with particular attention to commercial reactor concepts developed within the last three years. Emphasis is placed on DT fueled reactors for electricity production. A brief history of tokamak reactor studies is presented. The STARFIRE, NUWMAK, and HFCTR studies are highlighted. Recent developments that have increased the commercial attractiveness of tokamak reactor designs are discussed. These developments include smaller plant sizes, higher first wall loadings, improved maintenance concepts, steady-state operation, non-divertor particle control, and improved reactor safety features.

1. INTRODUCTION

1.1 Background and Purpose

The purpose of this paper is to provide an overview of fusion tokamak reactor concepts which is largely taken from a portion of a recent review paper [1] on the trends and developments in magnetic fusion reactor concepts. The emphasis of this paper is on the engineering and technology considerations of commercial tokamak reactors. A companion paper in this Special Issue by J. Sheffield describes in detail the basic tokamak concept and the status of tokamak research. Some discussion is also included in this paper on the design of more near-term engineering test reactors. For general background, the reader may consult several excellent past reviews on fusion reactor designs and technology requirements (see, for example, Refs. 2-8).

A variety of elements can undergo fusion reactions, as illustrated in Fig. 1. The deuterium/tritium (DT) fuel cycle has the highest reaction rate together with a high energy release per reaction (17.6 MeV), and, therefore, the largest power density for a given plasma density. In addition, the DT reaction has the lowest plasma temperature requirement of any of the fusion fuels. It is generally thought that the first commercial fusion reactors will operate on the DT fuel cycle, thus DT tokamak reactors will be the emphasis in this paper.

The fusion reaction of interest is

$$D + T \rightarrow {}^4He\ (3.5\ MeV) + n\ (14.1\ MeV). \qquad (1)$$

Note that about 80% of the fusion energy is released in the form of a high-energy neutron. Also, since tritium does not occur naturally, it must be

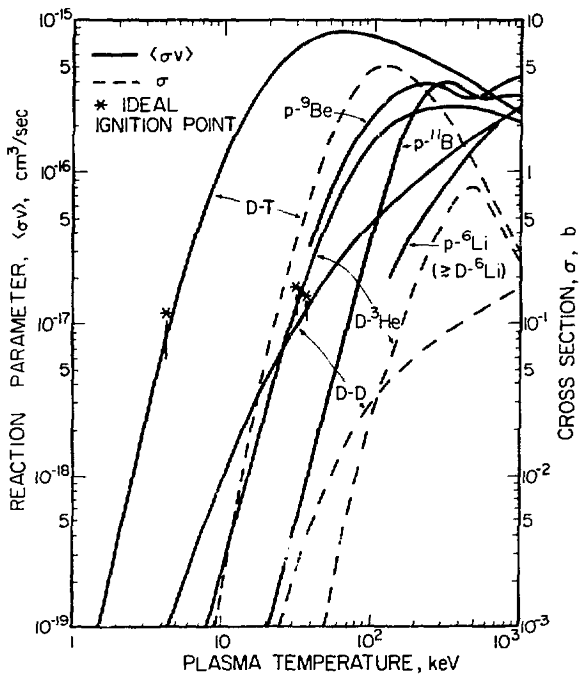


Figure 1. Reaction parameters and cross sections for various fusion reactions. The reaction parameter is averaged over a Maxwellian ion distribution. The curves shown for p-^{6}Li, p-^{9}Be, and p-^{11}B contain large uncertainties. Five strong D-^{6}Li reactions occur with different <σv>, but all lie near or below p-^{6}Li.

bred from neutron-induced reactions with lithium in the blanket of a fusion reactor; the reactions are

$$^{6}\mathrm{Li} + n \rightarrow T + {}^{4}\mathrm{He} + 4.8\ \mathrm{MeV}$$

and

$$^{7}\mathrm{Li} + n \rightarrow T + {}^{4}\mathrm{He} + n - 2.5\ \mathrm{MeV}. \qquad (2)$$

Natural lithium is composed of 92.6% ^{7}Li and 7.4% ^{6}Li. The fact that the breeding reaction in ^{7}Li releases a neutron that is available to induce a tritium producing reaction in ^{6}Li is the basic reason why a breeding ratio substantially greater than unity can be achieved with natural lithium.

The non-DT fusion fuels, often called "advanced" or "alternate" fuels, are of interest because they offer the potential features of no tritium breeding, reduced tritium handling requirements, reduced neutron activation of structural materials, and a larger fraction of the energy released by charged particles which may permit the use of direct energy conversion into electricity. On the other hand, the smaller cross sections and higher temperature requirements imply more difficult plasma physics confinement problems and the need for larger magnetic fields. The physics and technology issues of alternate fuels are receiving increasing attention [9,10].

It is important to note that there are several potential end uses of fusion energy. The focus of this paper is on electricity production, usually in the form of a base-loaded central station power plant. However, there is continuing interest in the application of fusion energy to synthetic fuel production [11-13] (e.g., hydrogen) and the production of fissile fuel and electricity in fusion-fission hybrid reactors [14-16].

1.2 Basic Fusion Reactor Parameters

There are several basic concepts and parameters regarding magnetic confinement fusion reactors which are briefly described in this section. In order to derive useful energy production from fusion, it is necessary to heat a DT plasma to temperatures of the order of 10^8 K and contain the fuel at a density (n) for a time (τ) such that $n\tau \gtrsim 10^{14}$ s·cm^{-3} [17]. The temperature of a fusion plasma is often given in units of kinetic energy, i.e., 1 keV is equivalent to 1.2×10^7 K. Most tokamak fusion reactors are designed to operate in the range of $n \sim 10^{14}$–10^{15} cm^{-3}.

There are three important parameters that characterize a tokamak fusion reactor concept and which will be frequently mentioned in the following sections of this paper. These parameters are the plasma β, power amplification (Q), and the duty cycle (d).

Plasma Beta (β)

The parameter β is defined as the ratio of the plasma kinetic pressure to the pressure of the confining magnetic field, i.e.,

$$\beta = \frac{\sum_i n_i T_i}{B^2/2\mu_o} , \qquad (3)$$

where i is summed over all plasma species, T is the particle temperature and B is the magnetic field strength (mks units). The plasma beta is related to the fusion power density (p) by $p \propto \beta^2$.

Power (Energy) Amplification (Q)

Fusion reactors are inherently power (energy, if pulsed) amplification devices where Q is defined as

$$Q = \frac{\text{power (energy) out}}{\text{power (energy) in}} , \qquad (4)$$

where the power (energy) balance is on the fusion plasma. $Q = 1$ is generally referred to as "energy breakeven", and $Q = \infty$ is referred to as "ignition", i.e., the point at which the plasma self-energy (created by the 3.5 MeV α-particles in DT fusion) is sufficient to compensate for other energy loss mechanisms, e.g., transport and radiation losses from the plasma.

Duty Cycle (d)

The plasma duty cycle is defined as

$$d = \frac{\text{time of production of fusion energy (burn time)}}{\text{total cycle time}} \quad . \qquad (5)$$

2. SURVEY OF TOKAMAK REACTOR STUDIES

The tokamak concept represents today the mainline experimental device in the field of magnetic confinement fusion research throughout the world. It is the approach which has achieved the most promising results to date, and is expected to be the vehicle by which energy breakeven will first be demonstrated for fusion energy in the next few years. Research on the tokamak was initiated in the Soviet Union in the mid-1950s. For comprehensive reviews of tokamak research see Refs. 18-20 and J. Sheffield's article in this issue.

Tokamak confinement research has progressed to the point where there is high confidence that energy breakeven will be achieved in a DT plasma in the next few years in the Tokamak Fusion Test Reactor (TFTR) [21] currently under construction at the Princeton Plasma Physics Laboratory (PPPL). A schematic drawing of TFTR is shown in Fig. 2. TFTR will employ approximately 30 MW of neutral beam heating in a device with $R = 2.48$ m, $a = 0.85$ m, and an on-axis toroidal field (B_{TO}) of 5.2 T. The device is expected to produce a plasma with densities of 10^{14} particles/cm^3, plasma temperatures of $5 \rightarrow 10$ keV, and fusion power densities of at least 1 MW/m^3. The device will begin initial operations in 1982.

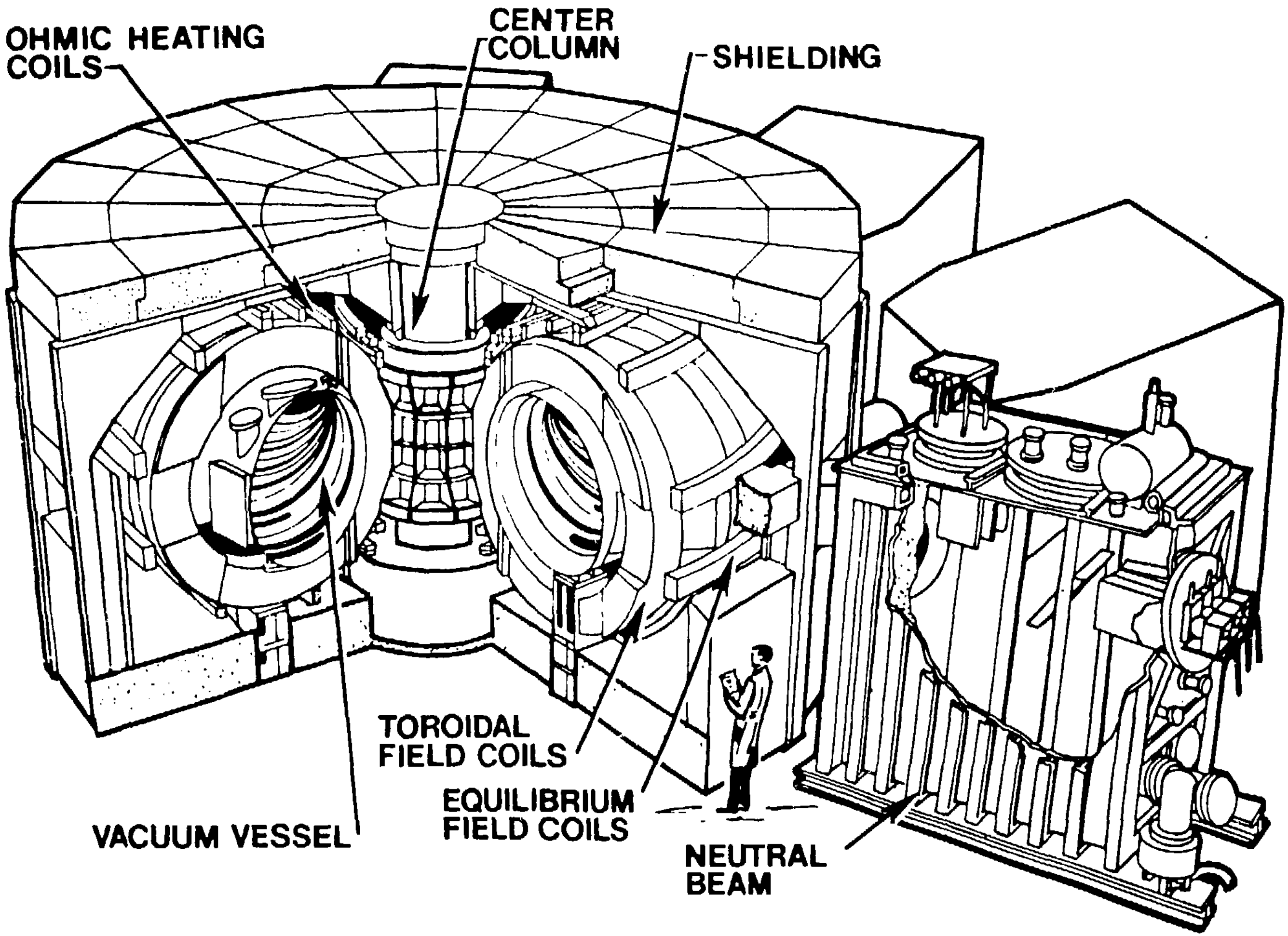


Figure 2. Schematic drawing of TFTR

TFTR is representative of a class of large tokamak devices [22] which will begin operation within the next few years. Examples include the Joint European Torus (JET) which is being constructed at the Culham Laboratory in the United Kingdom under the auspices of the European Economic Community and the JT-60 device which is being constructed at the Japan Atomic Energy Research Institute. The Soviet Union is also considering building a tokamak device of similar scale.

During the past two years, two design efforts, one national and one international in scope, have addressed the next step in the tokamak development program after TFTR. Within the U.S., an activity has been underway to establish a design concept of a tokamak Engineering Test Facility [23] which is being carried out under the direction of the ETF Design Center located at Oak Ridge National Laboratory. A schematic diagram of one of the concepts under consideration for the ETF is shown in Fig. 3. Major parameters for the ETF are $R = 5.4$, $a = 1.3$ m, and $B_{To} = 5.5$ T. With a $\beta \lesssim 5\%$, it is expected that the ETF would produce an ignited DT plasma with burn times of > 100 seconds. It would employ superconducting magnets and be capable of demonstrating the capability to breed tritium and remove the heat generated by the fusion neutrons in the blanket.

More recently, the ETF design effort has evolved into a design study of a tokamak Fusion Engineering Device (FED). These developments, as well as other programmatic aspects of the national fusion program, are described in a paper in this issue by J. Clarke. Typical parameters for the FED currently under consideration include $R = 4.8$ m, $a = 1.3$ m, $Q = 5$ and a fusion power level of 180 MW.

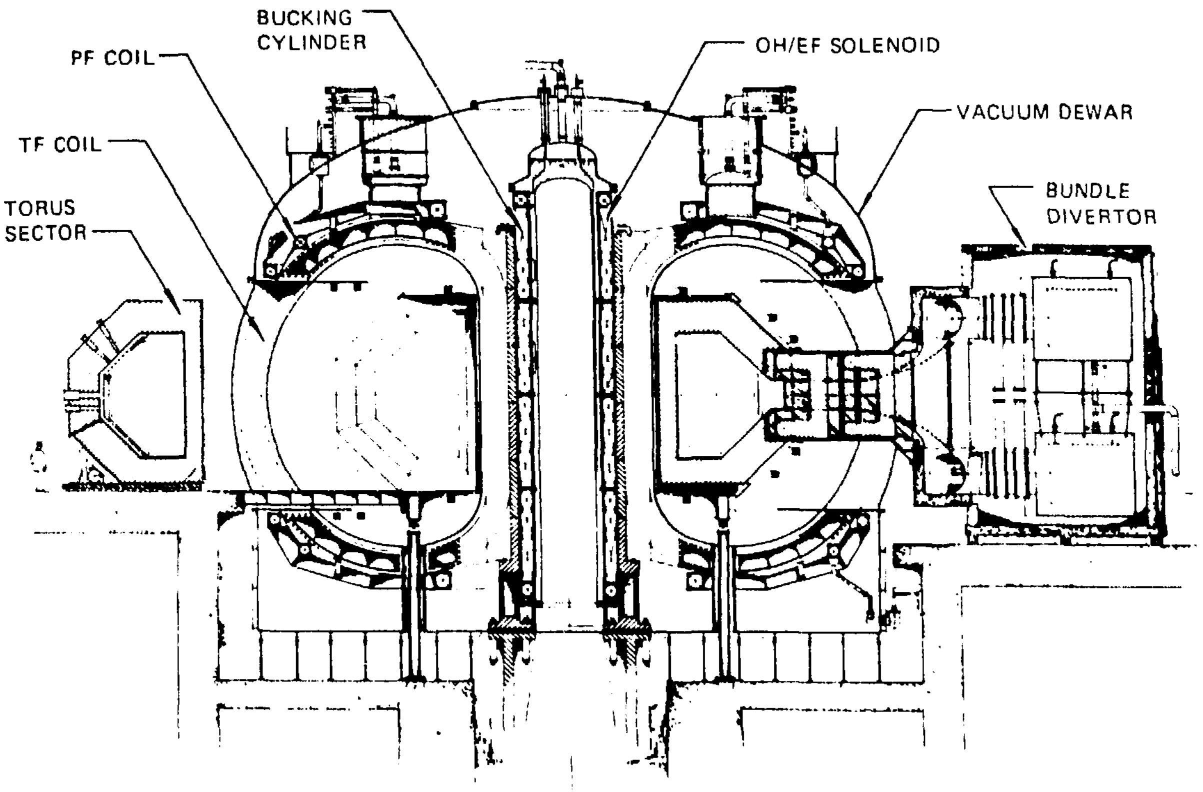
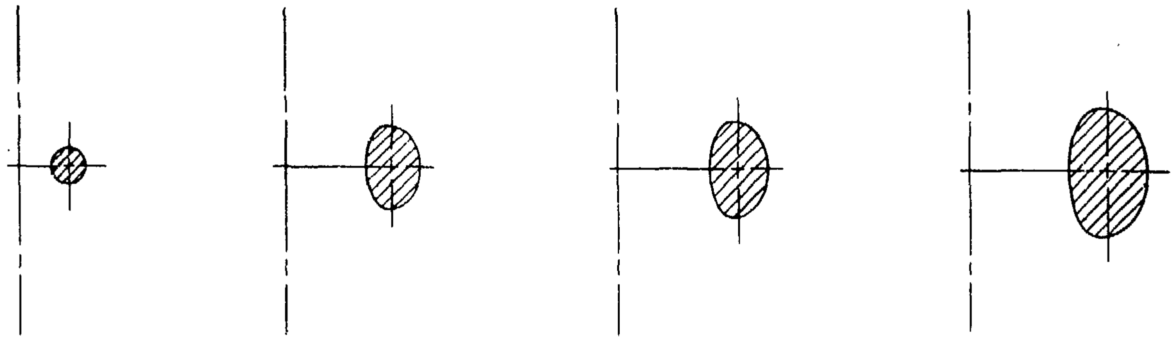


Figure 3. Cross-section view of an ETF design option with a bundle divertor

On the international front, the International Atomic Energy Agency organized a world-wide effort to assess the feasibility of undertaking a major fusion reactor development project. The project was termed the International Tokamak Reactor (INTOR) Workshop [24] and included representatives from Euratom, Japan, the U.S.A. and the U.S.S.R. The results of this assessment, which developed some initial conceptual designs and reviewed, in depth, the technical issues of such a project, indicated that such a project was feasible and that the tokamak should be selected because of its advanced stage of development. The workshop is continuing with emphasis on developing a conceptual design as a basis for a possible international project.

The various steps required to take the tokamak from scientific feasibility to a commercial reactor concept is illustrated schematically in Fig. 4. These steps include TFTR, an ETF/INTOR or FED type device to demonstrate long DT burn physics and the integration of major reactor technologies, a demonstration reactor (DEMO) represented by a design developed by ORNL [25], and a commercial-size tokamak reactor represented by the STARFIRE design [26]. One notes that there is about a factor of two scale-up in physical dimensions from TFTR to FED or ETF, but that there is only a small increase in dimensions from ETF to a commercial reactor. There are, of course, substantial differences between the FED/ETF and DEMO with respect to reliability and lifetime of components.

Studies of tokamak reactor concepts have been underway for more than ten years. More than 30 studies have been carried out within the United States as well as several studies in the Soviet Union, United Kingdom, Japan and Italy. These efforts have included studies of physics and ignition test reactors [21,27-30], engineering and materials test reactors [31,32],



	TFTR	INTOR/ETF	ORNL / DEMO	STARFIRE
R (m)	2.5	5.2 - 5.4	6.0	7.0
a (m)	0.9	1.3	1.5	2.0
THERMAL POWER (MW)	20	620-1130	1700	4000

Figure 4. Tokamak development steps to commercial reactors

experimental power reactors [33] demonstration power reactors [25,34] and commercial power reactors. The focus of this review is on commercial tokamak reactor concepts.

The early commercial reactor studies [35-41] covering the period up to 1974-75 were directed at developing an initial understanding of the general features of conceptual tokamak reactors. They were not intended to represent optimal designs (which clearly could not be done then and cannot be done at this time), and were based on the general physics knowledge available at that time. Generally, these early designs were large reactors, both in terms of their physical size and power output, and were not considered to be very attractive as commercial reactors.

A second series of tokamak commercial reactor studies [42-44] were carried out during the period 1975 to 1977. These studies explored new features of tokamak reactor concepts, such as non-circular plasma cross-sections and higher β values, rf heating, higher efficiency power conversion systems and improved maintenance concepts.

A third series of reactor studies were completed in 1978-1979 which included the University of Wisconsin's NUWMAK study [45] and MIT's High Field Compact Tokamak Reactor (HFCTR) study [46]. These studies further focused on the topic of reducing the physical size and plant power output of commercial tokamak reactor concepts.

NUWMAK is a medium field tokamak reactor which shows that tokamaks can have high power density, a high degree of modularity, and moderate size. The power density (10 MW/m^3) and electrical power output (660 MW) are chosen as typical of a full scale reactor operating in a base-loaded mode. The plasma has a noncircular D shape and a toroidal β of 6%. Plasma heating is by rf at $\omega = 2\omega_{CD}$ (92 MHz) based on the fast magnetosonic mode, where ω_{CD} is the angular cyclotron frequency of deuterons. The TF coil set is unique in that just eight superconducting coils are used. A set of 16 small water cooled copper trim coils that do not encircle the vacuum chamber correct the field ripple to below 2%. The blanket is constructed of the titanium alloy, Ti-6Al-4V, and is designed to minimize thermal cycling, to provide internal energy storage, and to eliminate the need for an intermediate heat exchanger. A lithium-lead eutectic, $Li_{62}Pb_{38}$, with a melting point of 464°C is used as the tritium breeding and thermal energy storage material. The latent heat of fusion for $Li_{62}Pb_{38}$ provides the required energy between plasma burns. Boiling water at 300°C, 1250 psi is the coolant and this further reduces thermal fatigue problems (see Sec. 4.7).

HFCTR is a compact (R_o = 6.0 m) high field (B_{To} = 7.4 T) tokamak power reactor which can produce fusion power densities as high as 10 MW/m^3 with a spatially averaged value of toroidal beta of less than 5%. The HFCTR design is based upon minimal extrapolation from experimentally established plasma confinement and MHD stability in tokamak devices. A unique design for the Nb_3Sn toroidal-field magnet system reduces the stress in the high-field trunk region and allows the achievement of high fields with a small radial build. An integrated system of automated actuators, vacuum and current-carrying mechanical joints and flexible cryostats allow total modularization of the reactor, including the coil systems. The modest value of toroidal beta permits a simple plasma-shaping coil system, located inside the TF coil trunk. Heating of the central plasma is attained by the use of ripple-assisted injection of 120-keV D^o beams, which are also used for dynamic control of the plasma temperature during the burn period. A FLIBE-lithium blanket is designed especially for high-power-density operation in a high-field environment, and gives an overall tritium breeding ratio of 1.05.

The latest study, STARFIRE [26], represents an attempt to incorporate several features considered in past reactor studies. These features will be discussed in more detail in Sec. 3.0 and 4.0. Schematic diagrams of the reactor of each of these three recent studies are shown in Figs. 5, 6, and 7. A summary of major reactor parameters for these three studies, along with the parameters for UWMAK-I, UWMAK-III, and the ORNL/DEMO for comparison, is given in Table I. A detailed listing of parameters for most of the earlier studies can be found in Ref. 47.

The progression of tokamak reactor concepts is schematically illustrated in Fig. 8, which illustrates the various stages of reactor design studies using representative reactor concepts (UWMAK-I [39], UWMAK-III [43], NUWMAK [45] and STARFIRE [26]). Clearly, there has been substantial improvements in the tokamak concept in terms of reducing its size and power output. This improvement [48] has resulted because of better estimates of the plasma size required for ignition (minor radii of 1 → 2 m), expected stable values of β in the range of 5 → 10%, and first wall/blanket lifetimes of 10-20 MW-yr/m^2 which permit higher wall loadings (see Fig. 8). These developments in physics and technology have resulted in tokamak reactor plant outputs in the range of 600 → 1200 MWe which are compatible with current utility requirements.

The next section provides a more detailed description of the STARFIRE reactor, which is taken to be representative of the latest series of tokamak reactor studies. Section 4.0 then describes in more detail several recent developments in tokamak reactor studies.

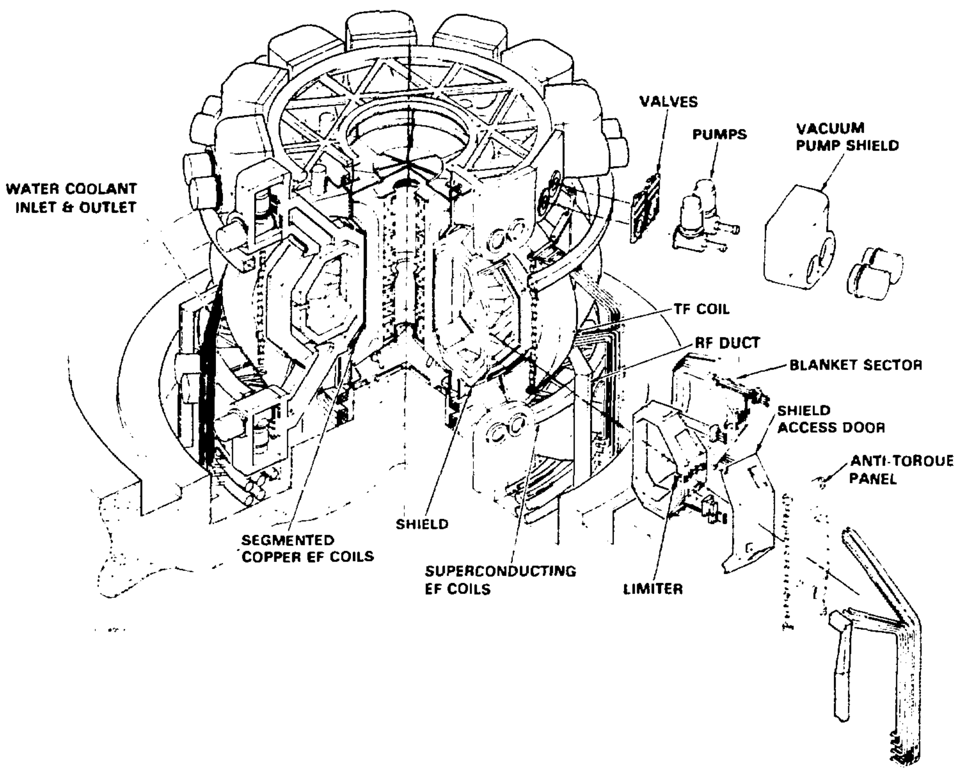


Figure 5. STARFIRE reference design - isometric view

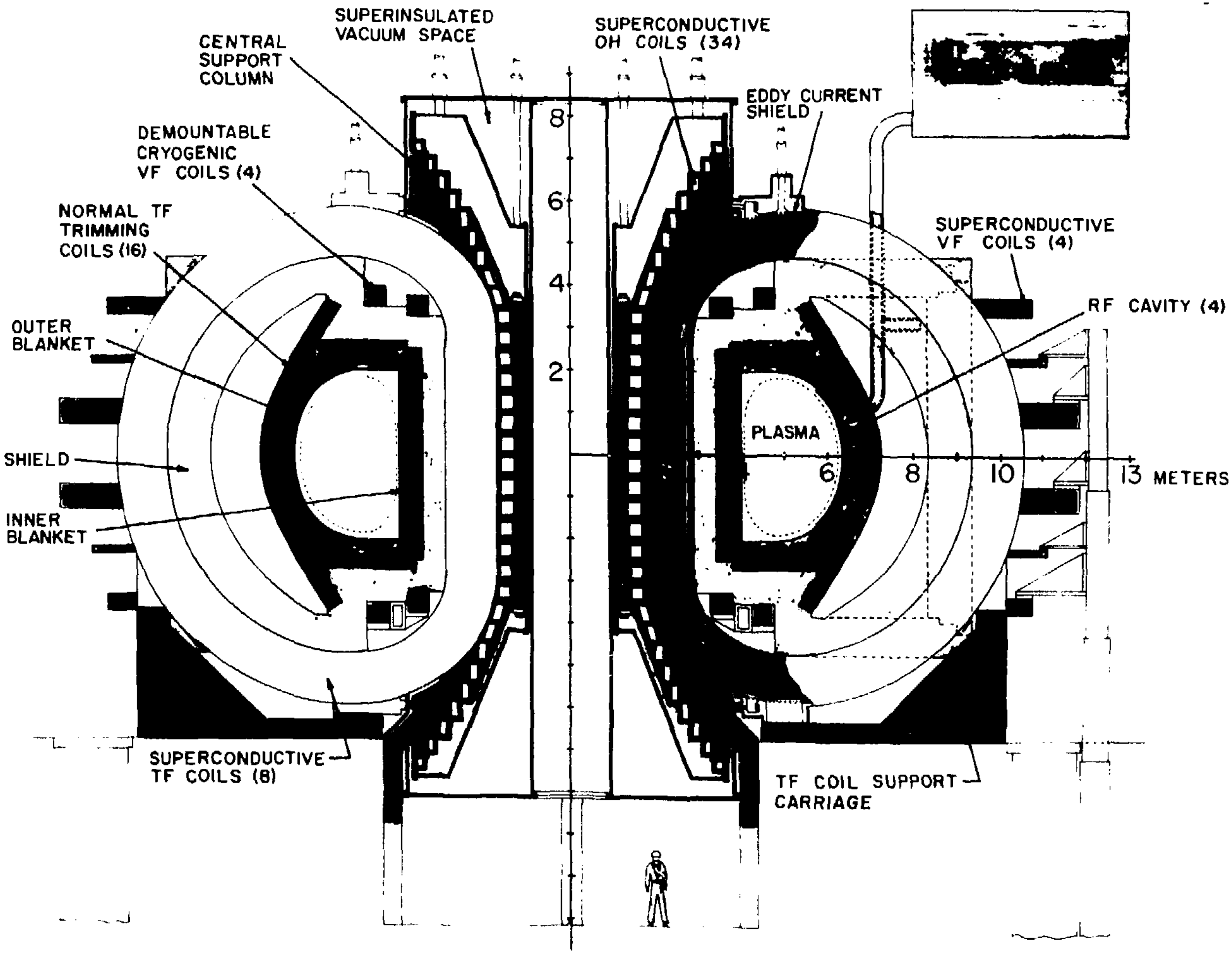


Figure 6. Cross-sectional view of NUWMAK

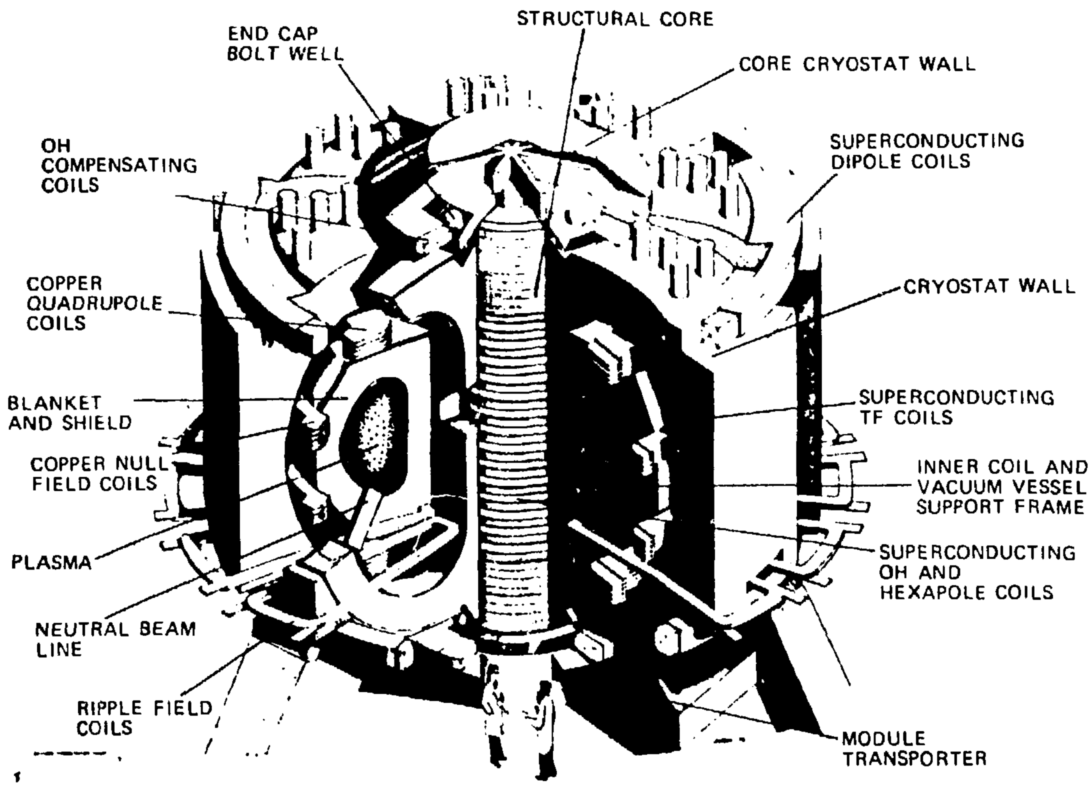
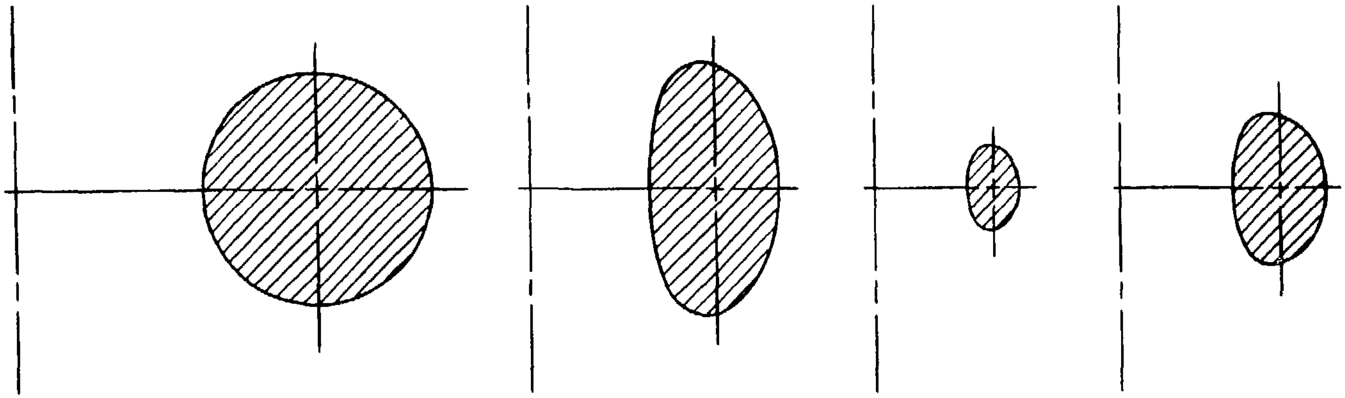


Figure 7. Schematic diagram of the HFCTR

TABLE I

Summary of Major Parameters for Some Selected Tokamak Reactor Studies

Parameter	UWMAK-I	UWMAK-III	ORNL[15] DEMO	HFCTR[16]	NUWMAK	STARFIRE
R (m)	13	8	6.2	6	5.2	7.0
a (m)	5	2.5	1.5	1.2	1.1	1.94
P_{th} (MW)	5000	5000	2150	2470	2300	4000
P_e (MW)	1450	2000	825	775	620	1200
Material	316 SS	TZM	Mod. 316 SS	TZM	Ti	Mod. 316 SS
Coolant	Li	He/Li	He	Flibe	Boiling H_2O	Press. H_2O
Breeder	Li	Li	Li	Li	$Pb_{38}Li_{62}$	$LiAlO_2$
P_{MW}(MW/m^2)	1.25	2.0	2.7	3.4	4.0	3.6
$B_{T,max}$ (T)	390	8.7	7.1	13.1	11.9	11.1
Burn Time	5400	1800	1260	500	224	Continuous
Off Time	390	100	60	90	21	--
Duty Cycle	93%	95%	95%	85%	91%	100%
$\langle\beta\rangle$ (%)	5.2	5.8	10.0	4.0	6.0q	6.7
I_p (MA)	21	16	3.9	6.7	7.2	10.1
b/a	1.0	2.0	1.6	1.5	1.6	1.6
T Burnup	7.2%	0.83%	∿ 3%	Not Avail.	1.5% (∿ 20%)	42%
Power Cycle	Li/Na/ Steam	Direct Cycle/ Li/Na/He	He/ Steam	Flibe/ Steam	Steam	H_2O/ Steam
Impurity	Double Null Divertor	Double Null Divertor	--	Limiter/ Vac. Halo	Gas Puffing	Limiter/ Vacuum
Heating	NB	RF	NB	NB	RF	RF



	UWMAK-I 1973	UWMAK-III 1976	NUWMAK 1978	STARFIRE 1980
R (m)	13	8.1	5.1	7.0
a (m)	5.0	2.7	1.1	2.0
P_{NW} (MW/m^2)	1.25	2.5	4.0	3.6
THERMAL POWER (MW)	5000	5000	2100	4000

Figure 8. Evolution of commercial tokamak reactors

3. OVERVIEW DESCRIPTION OF THE STARFIRE DESIGN

The major reactor parameters for STARFIRE are listed in Table I and the reactor cross section is shown in Fig. 9. The major features are shown in the isometric view in Fig. 5. 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 major feature for STARFIRE is a steady-state operating mode based on a continuous plasma current drive. An rf lower-hydrid current drive option has received the most attention in the study. The potential advantages of steady-state reactor operation are numerous and are further discussed in Sec. 4.2.

A plasma burn cycle with slow startup and shutdown, consistent with steady-state operation, was developed. During the breakdown phase, $\sim$ 5 MW of electron cyclotron resonant heating (ECRH) is applied. The limited ohmic heating or induction coil system induces $\sim$ 2 MA of plasma current. The rf current 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 very little need for electrical energy storage (see Sec. 4.5).

Availability goals have been established as 75% for the reactor and for the overall plant. This goal provides 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.

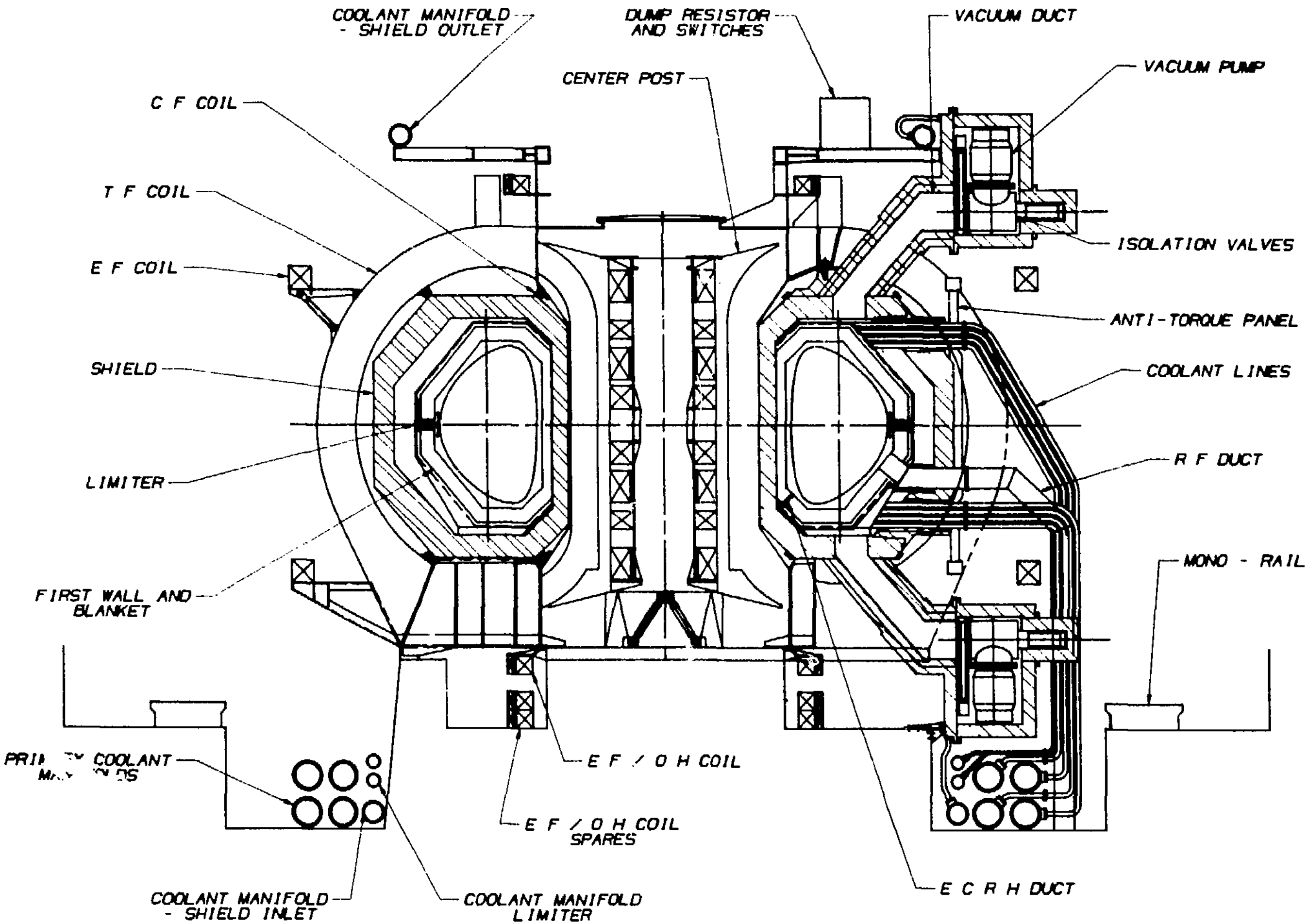


Figure 9. STARFIRE reference design - cross section

An important design consideration is the choice of the plasma impurity and alpha-particle removal concept. Initial investigations indicate that modest pumping of helium with a limiter/vacuum system ($\sim$ 28% of the alpha-particle flux) coupled with about a 1.5 T margin in the maximum toroidal field (TF), is an attractive alternative to divertors. This result is based on the provision that a significant portion of the alpha-particle heating power can be radiated to the first wall rather than be deposited on the limiter. In general, a simple non-divertor option is preferred from an overall reactor engineering point of view. This is discussed further in Sec. 4.3.

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 compound cryopumps are used. Twenty-four of the pumps are operated while the remaining 24 are rejuvenated.

Another key design consideration is the location of the poloidal equilibrium field (EF) coils. The basic design approach is to locate almost all the EF coils outside of the TF coils. All such outside EF coils would be superconducting. Four segmented copper coils are located inside the 12 TF coils, but outside of the blanket and shield.

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 $\sim$ 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 suitable temperature range to enhance tritium release without sintering. A helium purge stream is used to extract the tritium. Further details are presented in Sec. 4.7.

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 six of the shield rings near the outer surface of the shield to limit the radiation dose to 10^{10} rads.

4. TOKAMAK REACTOR DESIGN ADVANCES

The primary purpose of reactor design studies is to explore new ideas and examine concepts being developed in research programs from the point-of-view of their implications for reactor development. This serves the important function of providing insight and guidance for ongoing physics and technology research efforts. The reactor concepts summarized in the preceding sections represent the most current ideas regarding the tokamak as a power reactor, yet it is likely that considerable further development and improvement will occur as research continues. This section contains a description of some recent developments in tokamak reactor studies. Further information on fusion technology development in such areas as magnets, safety, plasma heating, materials, and tritium handling can be found in other papers in this special issue.

4.1 Physical Size and Power Output

As discussed in Sec. 2.0, recent tokamak reactor designs have shown a significant reduction in size compared to earlier studies. NUWMAK illustrates that tokamak power plants outputs of $\sim$ 600 MWe are feasible and that tokamak power plant outputs are not necessarily very large compared to a current-day power plant. NUWMAK probably represents the lower range of physical size that is feasible for a tokamak power reactor.

4.2 Steady-State Operation

Most tokamak reactor designs in the past were pulsed, albeit with very long pulses (typically 200-1000 seconds) and high duty cycles (> 95%). Nevertheless, pulsed operation implies certain penalties, such as the need for thermal and electrical energy storage, large OH coils, and first wall thermal fatigue concerns which dictate lower first wall loadings and result in reduced reliability. Fortunately, theory and experiments indicate the possibility that toroidal plasma currents may be maintained in tokamaks with noninductive external momentum sources to the electrons [49]. 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 loads are acceptable; (4) thermal energy storage is not required; (5) the need for an intermediate coolant loop is reduced; (6) a significantly reduced probability of plasma disruptions; (7) electrical energy storage is significantly reduced or eliminated; and (8) a full-size ohmic heating solenoid is not needed, and external placement of the EF coils is simplified. 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. Lower hybrid waves constitute the reference driver for STARFIRE due to the amount of previous effort invested into understanding this option [50-53]. An alternative would be intense relativistic electron beams (REB). Calculations [54] indicate that small amounts of REB power (a few MW) are required to drive $\sim$ 10 MA in a tokamak reactor. However, the beam trapping and plasma equilibrium theory have not been as well developed for this option as the rf option. Other wave drive candidates include low frequency transit time magnetic pumping [55,56] and high frequency electron cyclotron resonance detrapping of electrons to enhance the bootstrap current [56]. The bootstrap current formed the basis for the steady-state reactor design (Mark I) studied by the Culham group [57]. Neutral beam driven currents [58] suffer mainly from low efficiency neutralizers for positive beams and from the large size and shielding difficulty associated with linear beam lines. The "continuous" tokamak [59] shifts the engineering difficulties to the poloidal coil system, requiring periodic transfer of large plasma volumes into various chambers of the torus.

The theory of lower-hybrid driven currents suggests a three-fold 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. The absence of a large ohmic heating transformer permits the strategic location of EF coils and the creation of an elongated ($\kappa = 1.6$), highly triangular plasma. The most suitable equilibrium in STARFIRE 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 necessary 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 somewhat 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 $n_{\parallel}$ = 1.40 - 1.86, it was found that 66 MW of power at 1.7 GHz is dissipated in maintaining the plasma 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. Further details on the rf system are described in Sec. 4.4.

It has been estimated that steady-state operation can result in a cost savings of at least 30%. This assumes the same availability for steady-state and pulsed reactors. It is likely that a steady-state system will result in a more reliable reactor with a higher availability. 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 at least ∿ 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.

4.3 Impurity Control and Ash Removal

One of the most difficult problems confronting fusion research is to develop workable, attractive concepts for controlling the influx of impurities into the plasma and for the removal of the fusion reaction products (helium in the case of the DT fuel cycle) from the reactor. This is essential in order to achieve long burn times or steady-state operation.

Most earlier reactor studies examined the use of magnetic divertors which are systems to divert a portion of the magnetic flux, either poloidal flux in the case of a poloidal divertor (see e.g., Fig. 10 which shows a cross-section of UWMAK-III) or toroidal flux in the case of a bundle divertor. An example of a bundle divertor design is shown in Fig. 3 for one of the ETF options currently under consideration.

The idea of using non-divertor options were explored in NUWMAK, STARFIRE, and, to some extent, in HFCTR. NUWMAK considered the concept of periodic gas puffing and a trapped ring of impurities acting as a halo around the plasma which would radiate the plasma alpha energy (transported to the halo from the plasma interior) to the first wall. This would result in depositing the

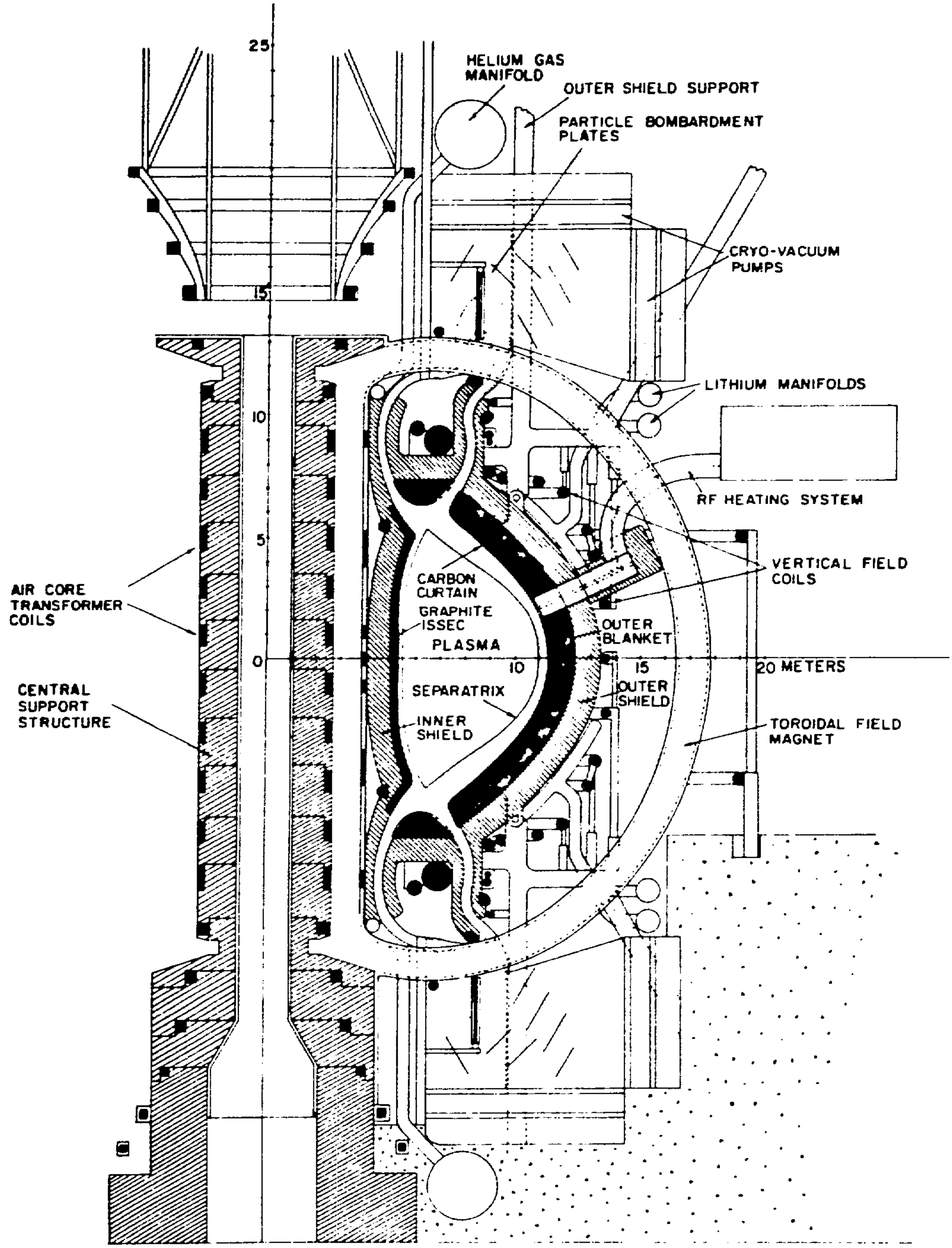


Figure 10. Cross-sectional view of UWMAK-II showing poloidal divertor concept

alpha power (390 MW) uniformly on the first wall with a surface loading of $\sim$ 1 MW/m^2. NUWMAK also explored the concept of fueling the center of the plasma with injected tritium pellets while puffing in deuterium gas at the plasma edge. This concept would significantly reduce tritium pumping at the plasma edge and, hence, increase the fractional burnup of tritium.

The STARFIRE study has developed in some detail the concept of using mechanical limiters and vacuum pumping ports for impurity control and ash removal [60]. In order to develop a viable limiter/vacuum system concept, one must consider four key problems.

The first problem is that of a high heat load on the particle collection medium. In steady-state, the alpha power plus any auxiliary heating power must be removed from the plasma region. In STARFIRE, the alpha power is 700 MW and the rf power is 90 MW, giving a total of 790 MW. In past designs, about half of this energy is radiated to the first wall leaving more than 400 MW to be transported to the particle collection medium. Previous designs for divertors showed that the surface area of the particle collection medium is limited to $\sim$ 20 m^2. For these designs the average heat loqd would be > 20 MW/m^2 and, given the fact that the particle heat load drops exponentially across the scrape-off region, the peak load would be > 50 MW/m^2. Such an extremely high heat load is beyond the capability of any suitable structural material. This problem is solved by employing two techniques. First, one enhances the plasma radiation to reduce the transport power to the particle collection medium. This is accomplished by injecting small amounts of high-Z material (iodine) along with the DT fuel. Most of the alpha energy is thus radiated to the first wall which has a large surface area. Second, one increases the surface area of the collection medium by minimizing the angle between the direction of incidence of the charged particles and the surface of the collection medium.

The second problem is the complexity of the vacuum system and associated neutron and gamma-ray streaming. An important part of the solution is to design for only a modest helium removal efficiency. This removal efficiency is defined as the probability that a helium particle diffusing out of the plasma will be pumped rather than reflected into the plasma. At steady-state, the rate of helium particle removal must equal the production rate. By requiring a removal efficiency considerably less than unity, particles will have to recycle until they are eventually pumped. The penalty of low removal efficiency is a higher alpha particle equilibrium concentration in the plasma. This is compensated for by a modest increment in the toroidal field for a fixed β to keep the fusion power the same. By designing for only a low removal efficiency for helium (i.e., 20-30%), the requirements on the vacuum pumping system are considerably relaxed. For example, the size of the vacuum ducts can be significantly reduced and a number of sharp bends can be tolerated. This approach makes it possible to reduce neutron and gamma-ray streaming with less shielding requirements and lower nuclear heat load at the cryopanels inside the vacuum pumps.

The third problem is that of a high tritium inventory in the exhaust and fueling systems. The solution to this problem follows automatically from requiring that the helium removal efficiency be low. The hydrogen removal efficiency will be even lower because the hydrogen species can charge exchange in the limiter vacuum pumping slots and thus re-enter the plasma, thereby increasing the tritium recycling and the fractional burnup. This reduces the tritium inventory in both the vacuum pumps and the fueling system.

A fourth problem is the overall complexity of the topology of the design concept. The limiter/vacuum system is a concept with many inherent features that can simplify a commercial power reactor:

- It is a mechanical system that does not require magnets.
- It has minimal requirements on space; the limiter fits naturally into the scrape-off region.
- Because of its location inside the first wall, the surface area available for the limiter is relatively large, thus permitting operation at reasonable heat fluxes.
- The limiter can be replaced simultaneously with the first wall with no special maintenance requirements.
- The limiter/vacuum system can be designed to dramatically reduce radiation streaming.
- The system is simple and inexpensive.

The limiter/vacuum system can be employed in any toroidal system (e.g., EBT, RFP, stellarators) as well as possibly the tandem mirror concept.

Figure 11 shows the STARFIRE limiter concept. The limiter consists of 96 segments that form one toroidal ring centered at the midplane and positioned at the outer side of the plasma chamber. This location was selected because it is the least likely place for a thermal energy dump from a plasma disruption, and it assures symmetry in particle and heat load on the upper and lower branches of the limiter. Each of the limiter segments is 1 m high and $\sim$ 0.6 m wide. The limiter slot, which is the region between the limiter and first

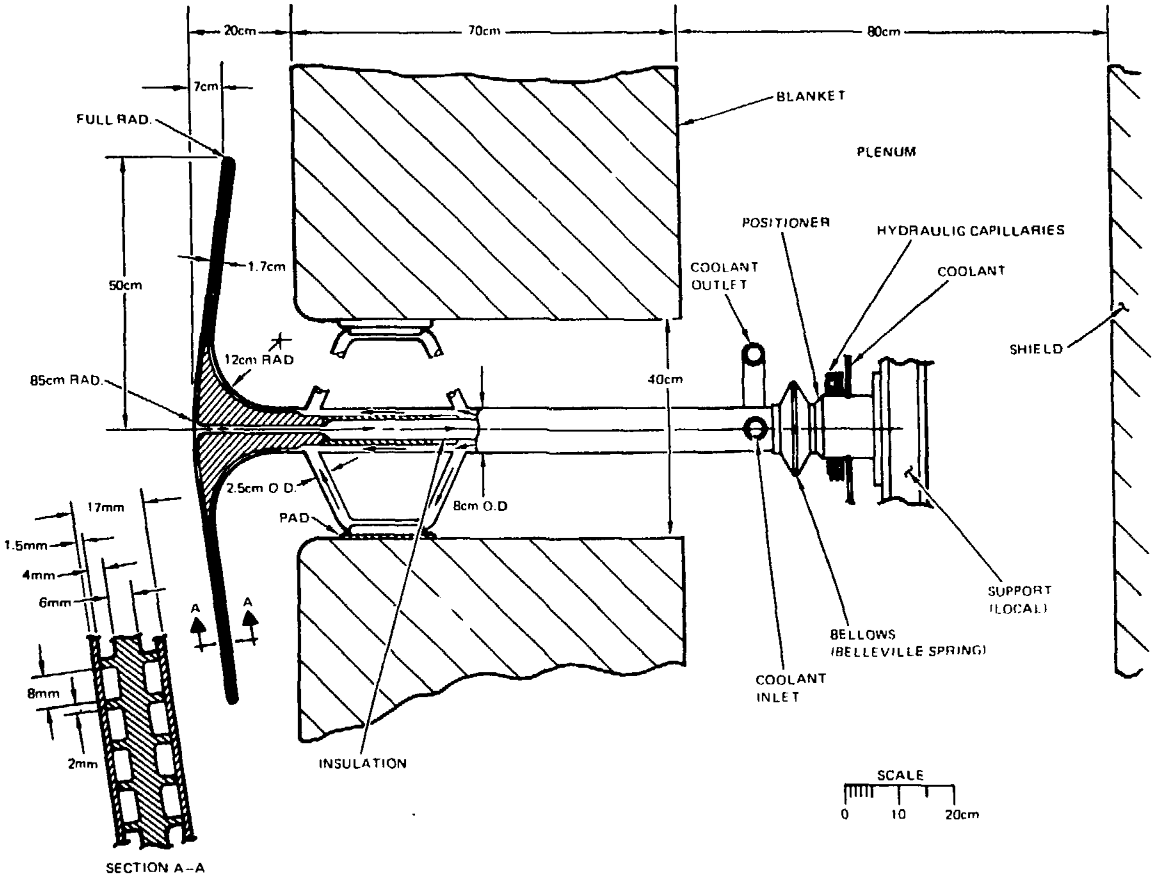


Figure 11. STARFIRE limiter design

wall, leads to a 0.4-m high limiter duct that penetrates the 0.7-m thick blanket. The limiter duct opens into a plenum region that is located between the blanket and shield and extends all the way around the torus (see Fig. 9). This plenum region is large enough so that it spreads the radiation leakage from the limiter duct into a larger surface area of the bulk shield. The conductance of the plenum region is large enough to permit locating the vacuum ducts in the bulk shield sufficiently removed from the midplane so that radiation streaming from the limiter duct in the blanket to the vacuum pumps is acceptable.

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. Some charged particles flowing along the magnetic field lines will be directed into the limiter slot and hit the back surface where they will be 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 captured 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 probability for hydrogen of being removed by the limiter system 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. The major parameters of the STARFIRE limiter/vacuum system are summarized in Table II.

One of the major improvements that resulted from the limiter/vacuum system is a substantially higher tritium burnup in STARFIRE (42%) compared to previous tokamak reactor designs (see Table I). This is a direct result of the higher reflection coefficient for tritons compared to α-particles. This has resulted in a significant decrease in the tritium inventory in the vacuum pumping and fueling system. This illustrates the important point that the tritium burnup fraction is more a function of the type of impurity/ash removal system than the type of magnetic confinement concept.

4.4 Plasma Heating Technology

The two principal methods considered for heating tokamak plasmas to ignition temperatures are neutral beam and radiofrequency (rf) heating. General reviews of this topic can be found in Refs. [61] and [62].

If neutral beams are used to heat ignition size plasmas at full densities ($\sim 10^{14}$ particles/cm^3), then neutral beam energies of about 300 keV will be required to penetrate the plasma [48]. This will require the development of negative ion beams in order to achieve acceptable beam efficiencies. Fortunately, there are some recent concepts which may reduce the beam energy requirements. Plasma modeling studies indicate that alpha particle heating in the center of the plasma will reduce the need for full beam penetration. If low density startup scenarios are feasible, then beam energies of $\sim$ 150 keV may be adequate [63], thus permitting the use of the more developed positive ion beam technology. It has also been proposed to create a top-bottom asymmetry

Table II. Major Features of the STARFIRE Limiter/Vacuum System

Helium production rate, s^{-1}	1.24×10^{21}
Helium reflection coefficient, R_α	0.75
Hydrogen reflection coefficient, $R_{D,T}$	0.90
Alpha particle concentration (n_α/n_{DT})	0.14
Beryllium (low-Z coating) concentration (n_{BE}/n_{DT})	0.04
Toroidal-field margin at plasma center, T	0.85
Fractional burnup, tritium	0.42
Tritium inventory per pump, g	2.60
Scrape-off region thickness, m	0.20
Limiter (one toroidal limiter centered at midplane)	
Structural material	Ta-5W
Alternate structural materials	AMAX-MZC, FS-85, V-20Ti
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 x 4 mm
Wall thickness, mm	1.5

in the toroidal field ripple of a tokamak using suitable auxiliary coils [64] which will assist the penetration of the beam. In the HFCTR, this concept results in a required neutral beam energy of 120 keV.

RF heating offers an attractive alternative to neutral beam heating with important reactor advantages. These include reduced neutron streaming effects, easier penetration of the blanket and shield, much freer access to the reactor to facilitate maintenance, reduced penetration shielding costs, removal of much of the system's components from the reactor building, and (except for electron cyclotron resonance heating) a largely existing technology at the power levels and efficiencies required for reactors.

UWMAK-III employed an rf heating system delivering 100 MW at 60 MHz for 15 s. NUWMAK's plasma heating system is based on fast magnetosonic waves (92 MHz). Approximately 80 MW of power will ignite NUWMAK in about one second of heating. The overall system efficiency is about 60%. A wedge shaped coaxial cavity (with a total of four in the torus) is mounted flush to the first wall and is fed by coaxial transmission lines.

STARFIRE's plasma heating system employs lower hybrid wave power of 90 MW at 1.7 GHz which is also used for current drive to achieve steady-state operation (see Sec. 4.2). No launching structure internal to the first wall is required. There are 12 waveguide modules between each TF coil, each module has a cross section of 0.66 m x 0.78 m. Averaged over the total antenna area, the wave intensity at the plasma is $\sim$ 1.5 kW/cm^2. Higher intensity could possibly result in nonlinear plasma responses.

The power supplies and rf system elements are located outside the reactor building. These elements are connected to the reactor by 12 rf duct assemblies that have 72 slots in each duct. The duct assembly will be mounted to the blanket sector as shown in Fig. 12. The duct will be constructed from PCA stainless steel. PCA stainless steel was chosen because it is being used in a similar environment in the first wall. Welds near the plasma will be minimized by using a machined part within ∿ 5 cm of the first wall. The interior of the steel ducts is coated with copper to minimize the power losses through the guide. Near the first wall, an additional coating of beryllium is added to the copper to minimize the effects of sputtered impurities.

The grill assembly will be cooled by 40°C water. The maximum structural temperature is ∿ 310°C. The grill assembly protrudes through the shield door where a mechanical disconnect is located that permits removal of the directional coupler and rf window as part of an elbow. Removal of the elbow permits access to the blanket/shield for maintenance. A window is included in the duct elbow to prevent electron cyclotron breakdown in the slots. The TF coil field profile requires the window to be within 3 m of the TF coil leg. The window material is sapphire (alumina), which has been shown to withstand 10^{21} n/cm^2 before serious degradation occurs. Neutronics analysis has indicated this fluence will be reached in > 10 yr of reactor operation. This indicates that window replacement at the 6-yr blanket replacement interval is appropriate. A second window is located at the reactor building wall to provide an additional barrier.

4.5 Startup, Burn and Shutdown Requirements

The power supply, electrical energy storage, and OH/EF magnet design considerations related to the startup and shutdown of large tokamak reactors has been recognized as an area of key concern [65]. One of the major motivating factors in examining steady-state operation is the desire to

POWER BALANCE	MW
R F POWER TO PLASMA	90.4
(CURRENT DRIVE)	(66.5)
LOSSES	62.3
TOTAL R F ELECTRICAL POWER	152.7

POWER LOSSES		MW
A	GRILL	0.5
B	WAVEGUIDE	1.9
C	WINDOW (3)	0.2
D	WAVEGUIDE	18.2
E	CIRCULATOR	10.6
F	PHASE SHIFTER	1.5
G	C F A	21.8
ELECT. BLDG	POWER SUPPLIES	7.6
	TOTAL LOSSES	62.3

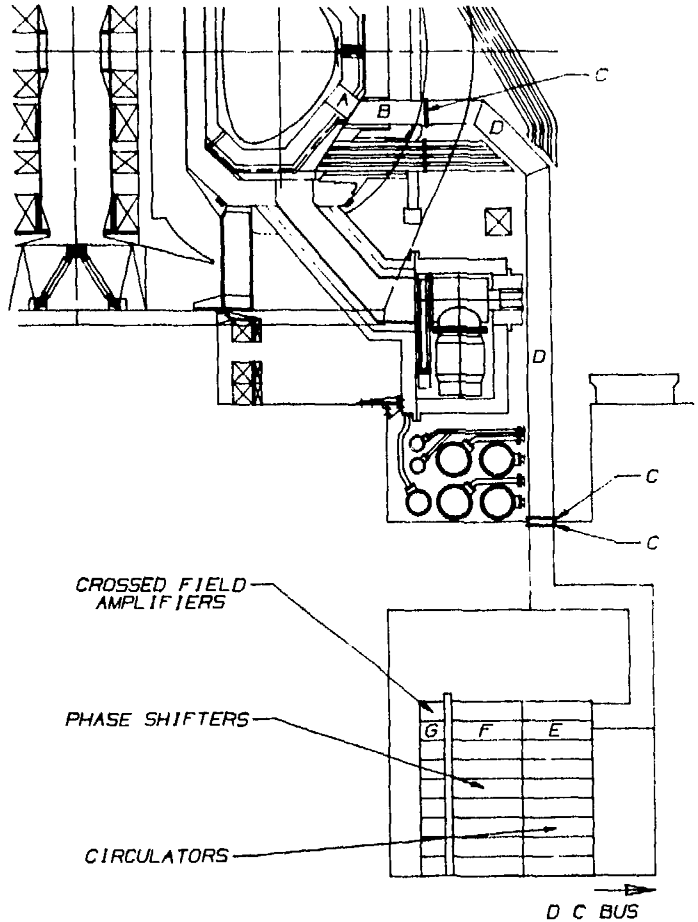


Figure 12. STARFIRE rf system

significantly reduce, if not eliminate, the need for large OH coils and associated energy storage systems. If the reactor is capable of truly steady-state operation, then one can consider a much broader range of startup and shutdown scenarios than with pulsed reactors.

A good example of a scenario for startup for steady-state operation is provided by STARFIRE. The plasma burn cycle starts when the previously evacuated torus is filled with fresh DT gas. The initial tritium fraction of this gas is 4%, i.e., the gas is 96% D. Five MWs of electron cyclotron resonance heating (ECRH) power is then applied to the plasma through a series of waveguides built into the first wall. The ECRH breaks down the fill gas to an ionized plasma at several hundred eV temperature in 10 ms. Next, the previously reverse-biased OH coil is discharged through a dump resistor circuit. The OH coil is completely discharged in about 14 seconds; it is then disconnected from the dump resistor circuit and has no further role in the burn cycle. The OH coil supplies about 25 V-s to the plasma. Most of the total volt-seconds required for startup (180 V-s) are supplied by the EF coils. The OH coil consists of a few turns located in the center of the torus (see Fig. 9) and is used only to achieve an initial plasma current of 1 → 2 MA. If it proves possible to induce current immediately, i.e., at low temperatures with the rf system, then even this small OH coil could be eliminated.

During this "ohmic heating" period, a variable amount of rf power is applied to the plasma, first with the ECRH system, for about the first three seconds, and thereafter with the lower hybrid system. For about the first half of the OH period (14 s), the rf power is varied so as to gradually heat the plasma to a temperature ($\sim$ 1.4 keV) where the lower hybrid waves can begin inducing current. The rf power is then linearly increased up to about 45 MW and kept constant for ∿ 250 seconds. During this 250 s period, which can be described as the main current inducement period, the plasma temperature is held constant at ∿ 6 keV, and the rf drive raises the plasma current to its final value of 10 MA. During this period, the plasma density is held constant to a low value of about 10% of the burn value. This combination of low density and moderate temperature minimizes the current inducement time and the supplied rf energy.

The next phase of the startup is the main plasma heating phase. At the start of this phase, the rf power is brought up to the full 90 MW level. This is done in 15 seconds to provide for a reasonable power load change from the grid. Only a very small electrical energy storage capability (i.e., 10 MJ for the OH coil) is required for STARFIRE. All other energy is supplied from the utility grid over the relatively long startup times. The plasma DT density is also brought up to full value, in ∿ 200 s, but a 4% tritium fraction is still maintained. The reason for using this level of tritium is to have some fusion power, to aid in heating the plasma, but to not have enough to thermally stress the first wall, blanket, steam generators and turbines. The combination of rf power, and the α-heating power from fusion, serve to heat the plasma to near full temperatures. At the end of this phase, the plasma is at full density, current and approximately at full temperature.

The last and longest phase of the startup period is the fusion power ramp phase. During this phase, the tritium fraction is increased from 4% to 50% in ∿ 17 minutes, in such a way as to linearly increase the fusion power to the 3500 MW level. Control of the tritium fraction is done by increasing the tritium content of the fueling stream. The duration of this phase is set at 17 minutes to achieve a 5% per minute increase in fusion power. If needed, this rate could easily be made faster or slower by changing the rate of increase of the tritium fraction.

In order to provide for a thermal equilibrium during the fusion ramp phase, and the subsequent burn period, iodine atoms are slowly added to the plasma. The iodine serves to increase the X-ray power loss of the plasma, and hence, to radiate most of the α-heating power. The iodine is added according to a simple control algorithm intended to maintain a constant value of $\beta = 0.067$. About 0.1% of iodine must eventually be added.

During the startup period, the EF current is monotonically increased in order to keep the plasma in MHD equilibrium. The maximum reactive power of the EF power supply is set during the startup and is 290 MVA. This is only about 20% of the estimated value that would be needed if STARFIRE were operated in a pulsed mode. The maximum draw of power from the grid is determined by the sum of rf and EF instantaneous powers. For a value of 150 MW of rf input power (for 90 MW output) and assuming a 95% conversion efficiency for the EF supply, the maximum power from the grid is about 250 MW. The utility grid would supply this power during a few times each year.

Two basic types of shutdown are envisioned for STARFIRE, a "normal" shutdown used once or twice a year to routinely shut the plant down, and an "emergency" shutdown used in accident or other non-routine situations. Like the startup period, the normal shutdown phase of STARFIRE is not restricted by time limitations. The reference shutdown scenario is divided into two parts, fusion ramp down and current ramp down. The fusion ramp down phase is essentially the reverse of the last phase of the startup scenario. For shutdown, the fusion power is reduced, at a 5% per minute rate, by reducing the plasma tritium fraction from 50% to 4%. At the same time, the iodine concentration is reduced to zero to keep the plasma in thermal equilibrium. During this phase, the DT density is held constant at its full value, but with a 4% tritium fraction. The lower hybrid rf power is ramped down linearly from 90 MW to 10 MW. This causes the plasma to cool and the remaining fusion power to fall off. The plasma current also decreases because of the reduction in rf drive and because of the transformer action of the EF coil as the EF current is reduced. Finally, the rf power is terminated and the plasma is extinguished.

During the shutdown period, energy is extracted from the EF coil and fed back to the grid. The maximum value of this power is (-) 70 MW. The EF power supply requirements for the shutdown phase are less than for the startup phase; the same supply is therefore used for both periods.

As discussed above, the normal shutdown period for STARFIRE takes about 25 minutes and would be done typically once a year for a scheduled maintenance period. A faster shutdown capability is needed for emergency conditions, such as loss of coolant, turbine trips, etc. Several types of emergency shutdown scenarios have been developed for STARFIRE; these are summarized in Table III.

One way of shutting down the plasma is to induce a plasma disruption. During a disruption, the plasma extinguishes completely in a fraction of a second, so the fusion power is essentially stopped immediately. A disruption can be initiated by injecting excess high-Z material (iodine, etc.) into the plasma. This can be done with the normal high-Z gas injection system. The addition of as little as 1 mg of excess iodine would more than double the

Table III. STARFIRE Shutdown Scenarios

Type of Shutdown	Purpose	Time for Complete Shutdown	Method
Normal Shutdown	Reduce fusion power slowly for normal shutdown	1450 s	Reduce tritium fraction
Abrupt Shutdown	Emergency shutdown - stop power immediately	$\lesssim$ 100 ms	Induce a plasma disruption by injection of excess high-Z material
Rapid Shutdown	Emergency shutdown - reduce power quickly	2.5 s	Terminate refueling and rf power
Ablative Induced Shutdown	Occurs automatically if small hot spot develops	$\sim$ 0.5	Ablating beryllium causes plasma to cool and disrupt

iodine content of the plasma and should cause an immediate disruption. The time needed for this type of shutdown should, therefore, be limited by the time needed to detect an emergency condition and initiate a shutdown command.

STARFIRE is designed to take several plasma disruptions in a year with no damage except for the ablation of a small amount of beryllium coating on the first wall. However, frequent occurrence or induction of plasma disruptions is undesirable. Therefore, exercising the option of emergency shutdown by inducing a plasma disruption is limited only to those critical failures in the system that require very abrupt shutdowns.

An alternate emergency shutdown, labeled as a "rapid" shutdown in Table III, was also developed. This type of shutdown also involves a disruption except that most of the plasma energy is dissipated in a controlled manner prior to the disruption. This type of shutdown might be used for a less critical situation where it was still desired to shutdown in 2 $\rightarrow$ 3 seconds.

The third type of shutdown has been called an "ablative induced shutdown". As its name implies, this shutdown is not induced by external control but occurs naturally as a result of the ablation of the Be coating from the first wall or limiter. This type of shutdown was examined by inputting different rates of Be ablation into a plasma dynamic code. The ablation rates correspond to various degrees of hot spot formation on the first wall. It was found that if as little as 10^{22} Be atoms per second ablate into the plasma, this will cause a disruption in a maximum of 0.7 seconds. An ablation rate of 10^{22} s^{-1} corresponds to as little as 1% of the wall area increasing in temperature to about 1000 K. These results tend to confirm the general safety feature of fusion; any accident serious enough to affect the first wall will almost immediately shut the plasma down before there is any major damage to the first wall. This would include water leaks or virtually any introduction of foreign material into the plasma.

There has also been a significant reduction in the magnet, power supply and energy storage technology requirements for pulsed tokamak power reactors. This has resulted because of two developments. The first is the use of rf-induced breakdown [24,66] during the startup phase. Such techniques would reduce the startup voltage requirements from typically $\sim$ 300 V to $\sim$ 50 V as well as reduce the rate of change of the magnetic field of the OH coil to 2 $\rightarrow$ 3 T/s, which is well within the state-of-the-art for pulsed superconducting coils [67].

The other key development is the recognition that startup periods for pulsed reactors of $\sim$ 10 s, instead of 1 $\rightarrow$ 2 seconds, will significantly reduce technology requirements [65]. This will make it possible to use conventional technology, such as motor-generator-flywheel energy storage and SCR-type power supplies, in place of more advanced technologies such as homopolar generators and inductive energy storage.

4.6 Accessibility and Maintenance Considerations

The early tokamak reactor studies were not primarily concerned with the specific questions of maintenance. Rather, the focus of these early studies was on basic features and feasibility of the major components of a reactor. These early studies have evolved into more detailed, sophisticated studies which have begun to seriously address the issues of maintenance and its relationship to availability and costs of tokamak reactors [68-70]. This has been facilitated by the increased involvement of industrial companies

(systems designers, component vendors and architect engineers). This growing involvement of industry in fusion reactor design is, in itself, a major improvement in the last few years.

A number of ideas and concepts have been suggested in various reactor studies, which taken together, have resulted in designs with much improved accessibility and maintenance features. These ideas include the following developments:

Improved modularization of the first wall/blanket shield -- Several ideas have been developed [71-73] which involve the removal of self-supporting blanket subassemblies which are replaced with pre-tested units. A great deal of the pioneering work of such concepts has been carried out by the Culham group [71]. These concepts emphasize a minimum of in-reactor operations which involve relatively simple in-out linear motions. Blanket components are repaired in a hot cell removed from the reactor building. The shield is fixed in place and should last the life of the reactor. This concept has been adopted for STARFIRE and is illustrated in Fig. 5.

Placement of EF coils outside TF coils -- This idea has been utilized in almost all recent tokamak reactor studies. While placing the EF coils outside the TF coils increases the stored energy of the EF coils and places larger torques on the TF coils, the penalty is more than compensated for by eliminating the problem of linked superconducting TF and EF coils and improving the access to the blanket. In STARFIRE, for example, (see Fig. 9), the blanket sectors can be removed without moving any of the poloidal coils. Spare coils are provided for the lower EF coils to facilitate replacement if this should ever be required.

Reduced number of TF coils -- Most recent reactor designs have reduced the number of TF coils from 16-20 to typically 8 to 12. This has significantly improved the access between TF coils so that large, modular blanket components could be easily removed. The toroidal magnetic field ripple has been kept to tolerable values by increasing the radius of outer TF coil leg, or by providing TF trimming coils [72] or pull-back coils [56], which permit one to reduce the size of the TF coils outer leg.

Location of the vacuum boundary -- It is now generally agreed that the vacuum boundary should not be the reactor's first wall. A first wall vacuum boundary places very difficult requirements on the design of the first wall and makes access for maintenance very difficult. It appears that the most attractive location for the vacuum boundary is outside the reactor shield, yet inside the TF coils. This places the vacuum boundary outside the high radiation environment where one can use mechanical joints with O-ring type seals rather than depending on welded joints. Access to the shield access door is also relatively easy (see e.g., Fig. 5).

It has also been suggested that the vacuum boundary could be placed at the walls of the reactor building [73]. However, evaluation [24] of this concept points out the disadvantages of significantly diminished personnel access and the need for much improved reliability of systems inside the reactor building.

Use of rf heating -- As noted above, rf heating greatly improves the access to the reactor by leaving the entire perimeter of the torus free for access for maintenance. Effects of neutron streaming are also reduced, which makes it easier to reduce the dose levels in the reactor building so that personnel access is feasible soon after shutdown.

Use of non-divertor impurity control -- The use of such concepts as the limiter/vacuum system discussed in Sec. 4.3 also provides for much better access to the reactor and diminished neutron streaming effects.

The ideas outlined above have been developed in several recent studies and have provided the basis for establishing the maintenance approach for STARFIRE. The basic approach is to use a "remove and replace" approach that minimizes the number of replaceable assemblies and the number of required different maintenance operations. This approach increases confidence in the speed of maintenance operations and simplifies maintenance equipment design requirements.

Because the reactor hall will be exposed to some tritium and decay radiation during maintenance operations, remote maintenance is planned for all reactor maintenance operations to minimize radiation exposure to maintenance personnel. Use of remote equipment can also permit some repairs while the reactor is operating.

All components within the reactor building are replaceable. Some are replaced on a scheduled maintenance basis while others are designed for the life of the plant and are replaced only in the event of failure. Items designed for the life of the plant include the overhead crane, TF coils, EF coils, coolant piping, reactor support structure, and shield. The blanket sector assemblies including limiters, rf ducts and fueling assemblies, shield door seals, vacuum pump isolation valves, etc., are replaced on a scheduled basis.

Spares are provided for all components with potentially high failure rates, so that as one part is removed, a pretested replacement is installed so that reactor operation can continue while damaged or end-of-life assemblies are moved from the reactor to a hot cell where more time and equipment are available for checkout, repair, or disposal.

Availability goals have been established as 75% for the reactor and the complete plant. The maintenance scenario incorporates the current utility practice of shutting down annually for one month of maintenance and a four-to-five month shutdown every five to ten years for turbine repair. The resultant permissible downtime goal per calendar year has been allocated as 37 days for scheduled maintenance of the entire plant and reactor, 34 days/year of unscheduled reactor maintenance, and 20 days for unscheduled maintenance of the balance of the plant. Convenient preventive maintenance and repair of redundant components is included as part of the maintenance scenario during unscheduled outages. A normal scheduled maintenace interval would include four manipulators working on the reactor at 90-deg intervals. Blanket sectors would be replaced at two locations while vacuum pump isolation valves are replaced at the other two locations.

4.7 Energy Conversion and Tritium Breeding

Most previous reactor studies have emphasized liquid lithium blanket concepts for tritium breeding with either liquid lithium or helium as the coolant (see Table I). In two of the most recent studies, STARFIRE and NUWMAK, attention has been focused on water as a coolant and on non-liquid lithium blankets.

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. Possible coolant types are liquid lithium, molten salts, helium, and water [74]. 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 $\sim$ 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 nonmobile [75] solid tritium breeder blanket with in-situ tritium recovery appears to be the preferred approach. 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 [76]. 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 sizes. The calculated solubility 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, $LiAlO_2$ was selected for the STARFIRE reference design. $LiAlO_2$ was also employed in UWMAK-II.

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 may be resolvable by using Zr_5Pb_3 which has a high melting point ($\sim$ 1400°C) and its neutron multiplication is adequate.

The $\sim$ 40-cm tritium-breeding zone consists of a packed bed of α-$LiAlO_2$ with 1-cm diameter stainless steel coolant tubes spaced appropriately throughout the zone (see Fig. 13) to maintain a maximum breeder temperature of 850°C. The tube spacing increases from $\sim$ 2 cm at the front of the breeder zone to 5 $\rightarrow$ 10 cm at the back. The coolant inlet temperature is 280°C with an outlet temperature of 320°C. The relatively low 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. The $LiAlO_2$ is perforated with $\sim$ 2 mm diameter holes through which low-pressure helium passes to recover the tritium from the breeder. The $LiAlO_2$ is $\sim$ 60% dense to facilitate percolation of tritium (as T_2O) to the helium purge channels.

Maximum lithium burnup in the blanket region will be $\sim$ 25% for a first-wall lifetime of 20 MW-yr/m^2. In addition to changing the stoichiometry of the breeder material, breeding characteristics and the energy generation profile in the blanket will be affected. When a neutron multiplier is used, the tritium breeding comes almost exclusively from ^6Li. Therefore, highly enriched lithium (> 50% ^6Li) is used to minimize the lithium and tritium inventories. Although some tradeoffs are possible, a limit of $\sim$ 20 MW-yr/m^2

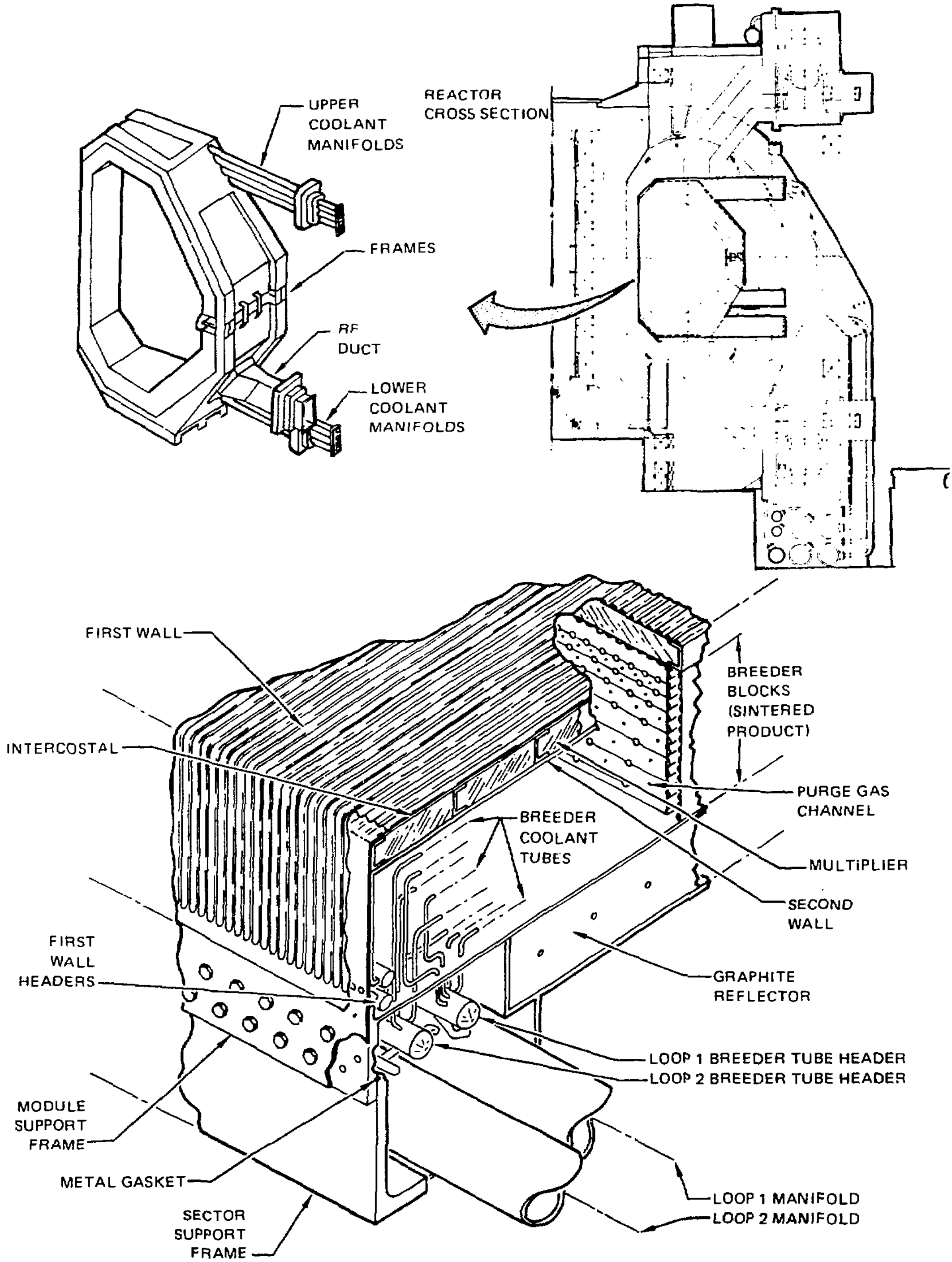


Figure 13. STARFIRE blanket concept

is reasonable for the breeder. This restriction tends to limit the value of a longer lifetime structure for a solid breeder blanket.

The power conversion system, shown schematically in Fig. 14, 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 feed-water 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 1440 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. This system corresponds to a rather conventional pressurized water reactor (PWR) balance of plant system.

The NUWMAK concept is based on the idea of using a breeding material that operates at its melting point. The energy stored in the latent heat of fusion of the material eliminates the need for an external thermal energy storage system which would otherwise be required due to NUWMAK's pulsed operating mode. The blanket is cooled by boiling water which also helps to minimize temperature cycling. The breeding material is the eutectic $Li_{62}Pb_{38}$ which undergoes a solid-liquid phase change at 464°C. The use of the boiling water coolant and internal blanket energy storage permits the use of a simple direct boiling water reactor (BWR) cycle.

In both of the above examples, the use of relatively conventional power conversion systems results in net plant efficiencies of typically ∿ 30%. This is lower than some previous studies, e.g., UWMAK-III which was 40%. However, this disadvantage is expected to be more than offset by significant reductions in the cost of the balance of plant.

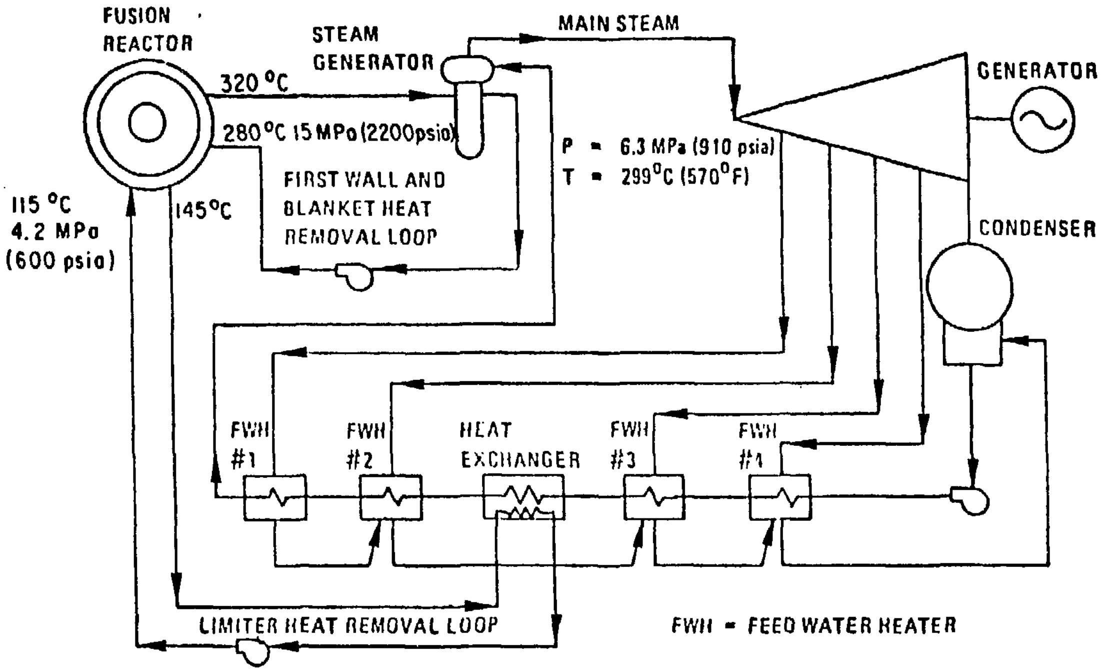


Figure 14. Power conversion system schematic diagram for STARFIRE

4.8 Economics

In discussing cost estimates of fusion energy, one must obviously exercise considerable caution because fusion is in the early phases of addressing the full scope of developing a reactor physics and technology capability. For tokamaks, the level of detail available in some of the conceptual designs does permit a reasonable estimate of the capital costs, recognizing that our present perception of a commercial power plant should be taken only as a guide for future research and development. Estimates of the cost of energy (COE) are more difficult because there is a limited data base upon which to estimate the reliability, and, hence, availability of commercial plants. Nevertheless, estimates of capital costs and COE are useful, particularly when considered in the context of the trends of recent reactor studies.

For example, capital cost estimates for NUWMAK and STARFIRE are about \$1500/kWe in 1980 dollars, compared to UWMAK-III which was about \$2100/kWe. One of the major reasons is the much simplified balance of plant in NUWMAK and STARFIRE as compared to UWMAK-III. The COE for NUWMAK and STARFIRE in 1980 constant dollars is in the range of 35-40 mills/kWhr compared to $\sim$ 90 mills/kWhr for UWMAK-III. These cost estimates illustrate the beneficial impact of the trend of tokamak reactors to smaller, more simplified design concepts.

5. SUMMARY

A summary of the design developments for tokamaks discussed in the preceding sections is given in Table IV. In general, these designs developments have emphasized less complex, more reliable components and systems. Several of these developments are applicable to other reactor concepts; e.g., nondivertor impurity control, rf heating, maintenance concepts and simplified energy conversion systems.

Table VI. Summary of Tokamak Design Advances

Design Features	Impacts
Reduced physical size and reactor output	More compatible with current plant sizes and lower unit costs
Steady-state operation	More reliable operation, lower energy costs
	Higher wall loadings--smaller reactors, lower capital costs
	No energy storage–lower capital costs
	Simplified energy conversion system–lower capital costs
	Reduce size of, or eliminate, OH coil
Non-diverter concepts for impurity control and ash removal	Lower capital costs
	Less complexity--more reliability
	Improved access for maintenance
	Higher tritium burnup, lower tritium inventories
	Reduced neutron streaming
Plasma heating technology	
Reduce neutral beam energy	Use existing positive ion beam technology
rf heating	Reduce neutron streaming
	Easier interface with blanket/shield
	Better access for maintenance
	Reduced shielding costs
Startup and shutdown	
rf assisted startup	Reduced OH voltage requirements
	Reduce *dB/dt* for OH coil
	Reduce volt-second requirement
Steady-state operation	Permits long startup times with very small or no electrical energy storage
Longer startup times (~10 s) for pulsed operation	Can use conventional power supply and energy storage technology
Maintenance	
Modularized first-wall/blanket concept	Simplified maintenance operations with minimum in-reactor operations
EF coils outside TF coils	Better access without moving or disconnecting coils
Reduce number of TF coils	Better access
Location of vacuum boundary at shield	Reduced requirements on first wall
	Simpler mechanical vacuum seals
	Easier access
Energy conversion	
Solid tritium and liquid lead/lithium breeders	Improved safety
H_2O coolants and PWR and BWR power conversion	Simplified BOP systems
	Reduce BOP costs

Some of these ideas are reasonably well supported by current experiments and theory (e.g., projected size of tokamaks for ignition) while some of the other concepts (e.g., non-divertor impurity control, steady-state operation, solid tritium breeders) require much more research and development. As these concepts are developed, as well as what is sure to be further improvements in the tokamak which are not envisioned today, then the tokamak will continue to improve as an attractive commercial reactor concept. The improvements in the more recent tokamak reactor studies compared to the earlier designs has been substantial.

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REFERENCES

1. C. C. Baker, G. A. Carlson and R. A. Krakowski, "Trends and Developments in Magnetic Confinement Fusion Reactor Concepts", *Nucl. Technol. - Fusion,* 1 (1981).
2. D. J. Rose, *Nuclear Fusion,* 9, 183 (1969).
3. *Proceedings of a Symposium on the Technology of Controlled Thermonuclear Fusion Experiments and the Engineering Aspects of Fusion Reactors,* Austin, Texas, CONF-721111 (April, 1974).
4. *Proceedings of the International Atomic Energy Agency Workshop on Fusion Reactor Problems,* Culham Laboratory, Abingdon, England (1974).
5. F. L. Ribe, *Reviews of Modern Physics,* 47, 7 (1975).
6. D. Steiner, *Nuclear Science and Engineering,* 58, 107 (1975).
7. F. L. Ribe, "Recent Developments in the Design of Conceptual Fusion Reactors," *Nucl. Technol.*, 34, 179 (1977).
8. Summary of the Second IAEA Technical Committee Meeting and Workshop on Fusion Reactor Design, Madison, Wisconsin, October, 1977, IAEA-TC-145 (1978).
9. R. W. Conn, et al., *Proceedings of the 8th International Conference on Plasma Physics and Controlled Nuclear Fusion Research,* Brussels, July, 1980, IAEA-CN-38/V-5.
10. C. C. Baker, et al., "The Impact of Alternate Fusion Fuels on Fusion Reactor Technology," Argonne National Laboratory, ANL/FPP/TM-128, (November, 1979), also, "D-D Tokamak Reactor Studies", Argonne National Laboratory, ANL/FPP/TM-138 (1981).

11. J. Fillo, *Proceedings of the 3rd Topical Meeting on the Technology of Controlled Nuclear Fusion*, Santa Fe, New Mexico, May, 1978, p. 193.

12. E. E. Kintner and C. M. Stickley, *Proceedings of the 3rd Topical Meeting on the Technology of Controlled Nuclear Fusion*, Santa Fe, New Mexico, May, 1978, p. 143.

13. J. F. de Paz, Y. A. Gohar and H. L. Schreyer, "Development of a Falling-Bed Fusion Blanket System for Synthetic Fuel Production," *15th Intersociety Energy Conversion Engineering Conference*, Seattle, Washington (August, 1980).

14. B. R. Leonard, Jr., *Nucl. Technol.*, 20, 161 (1973).

15. L. M. Lidsky, *Nuclear Fusion*, 15, 151 (1975).

16. W. C. Wolkenhauer, *Proceedings of the 3rd Topical Meeting on the Technology of Controlled Nuclear Fusion*, Santa Fe, New Mexico, May, 1978, p. 132.

17. J. D. Lawson, AERE-GP/R-1807, Atomic Energy Research Establishment, Harwell, England (1955).

18. L. A. Artsimovich, *Nuclear Fusion*, 2, 215 (1972).

19. H. P. Furth, *Nuclear Fusion*, 15, 487 (1975).

20. "Status of Tokamak Research," ER-0034, U.S. Department of Energy, edited by J. M. Rawls (October, 1979).

21. P. J. Reardon, *Proceedings of the 3rd Topical Meeting on the Technology of Controlled Nuclear Fusion*, Vol. I, 612 (1977).

22. D. L. Jassby, *Nuclear Fusion*, 17, 373 (1977).

23. W. R. Becraft and P. J. Reardon, *Proceedings of the 8th International Conference on Plasma Physics and Controlled Nuclear Fusion Research*, Brussels, July, 1980, IAEA-CN-38/V-1.

24. "International Tokamak Reactor - Phase 0", International Atomic Energy Agency, Vienna (May, 1980).

25. D. Steiner, et al., "ORNL Fusion Power Demonstration Study: Interim Report," Oak Ridge National Laboratory, ORNL/TM-5813 (1977).

26. C. C. Baker, et al., "STARFIRE - Commercial Tokamak Fusion Power Plant Study," Argonne National Laboratory, ANL/FPP-80-1, (1980); M. A. Abdou, et al., "STARFIRE - A Conceptual Design of a Commercial Tokamak Power Plant," *Proceedings of the 8th International Conference on Plasma Physics and Controlled Nuclear Fusion Research*, Brussels, July, 1980, IAEA-CN-38/E-1; C. C. Baker, et al., "STARFIRE - A Commercial Tokamak Power Plant Design," *Nuclear Engineering and Design* (to be published).

27. E. Bertolini, *Proceedings of the 3rd Topical Meeting on the Technology of Controlled Nuclear Fusion*, Santa Fe, New Mexico, May, 1978, p. 38.

28. D. L. Jassby, et al., "SLPX - Superconducting Long-Pulse Experiment," Princeton Plasma Physics Laboratory, PPPL-1481 (October, 1978).

29. "PITR - Princeton Ignition Test Reactor," Princeton Plasma Physics Laboratory, PPPL-1525, (December, 1978).

30. "TNS Scoping Studies," General Atomic Company and Argonne National Laboratory, GA-A14614 (1978); "General Atomic TNS Project," General Atomic Company, GA-A15100, (1978); M. Roberts, et al., Oak Ridge National Laboratory, ORNL/TM-5982 (1977); D. Steiner, et al., Oak Ridge National Laboratory, ORNL/TM-6201, (1978).

31. W. M. Stacey, Jr., et al., "Prototype Experimental Power Reactor Preliminary Considerations," Argonne National Laboratory, ANL/CTR/TM-64, (1976).

32. B. Badger, et al., "A Tokamak Engineering Test Reactor," University of Wisconsin, UWFDM-191, (1977).

33. W. M. Stacey, Jr., et al., *Proceedings of the 6th International Conference on Plasma Physics and Controlled Nuclear Fusion Research*, Berchtesgaden, Federal Republic of Germany, October, 1976 (also ERDA-76/139); W. M. Stacey, Jr., et al., "Tokamak Experimental Power Reactor," Argonne National Laboratory, ANL/FPP/TM-93 (1977).

34. C. C. Baker, et al., General Atomic Company, GA-A13992, (1976); D. Kearney, et al., General Atomic Company, GA-A14741, (1978).

35. I. N. Golovin, et al., *Proceedings of the British Nuclear Energy Society Nuclear Fusion Reactor Conference*, Culham Laboratory, England (1969).

36. J. T. D. Mitchell and R. Hancox, Culham Laboratory, CLM-P319, (1972).

37. A. P. Fraas, Oak Ridge National Laboratory, ORNL-TM-3096, (1973).

38. K. Sake, et al., Japan Atomic Energy Research Institute, JAERI-M5502, (1973).

39. B. Badger, et al., "UWMAK-I," University of Wisconsin, UWFDM-68, Vols. I and II, (1973).

40. R. G. Mills, Princeton University, MATT-1050, (1974).

41. R. W. Conn and G. L. Kulcinski, "UWMAK-II," University of Wisconsin, UWFDM-112, (1975).

42. C. C. Baker, C. R. Harder, P. H. Sager, Jr., "Characteristics of a First General Commercial Fusion Power Plant," General Atomic Company, GA-A13661, (December, 1975).

43. B. Badger, et al., "UWMAK-III," University of Wisconsin, UWFDM-150, (1976).

44. J. T. D. Mitchell and A. Hollis, *Proceedings of the 9th Symposium on Fusion Technology*, Garmisch-Partenkinchen, (1976).

45. R. W. Conn, et al., *Proceedings of the 3rd Topical Meeting on the Technology of Controlled Nuclear Fusion*, Santa Fe, New Mexico, May, 1978, p. 1049; B. Badger, et al., "NUWMAK," University of Wisconsin, UWFDM-330, (1979).

46. D. R. Cohn, et al., "High Field Compact Tokamak Reactor (HFCTR)," MIT Plasma Center, RR-78-2, (March, 1978).

47. "Major Features of D-T Tokamak Fusion Reactor Systems," Electric Power Research Institute (February, 1976).

48. D. Steiner, *Proceedings of the 3rd Topical Meeting on the Technology of Controlled Nuclear Fusion,* Santa Fe, New Mexico, May, 1978, pg. 440.

49. "Steady-State Current Drive in Tokamaks Workshop Summary," U.S. Department of Energy, ET-0077, (February, 1979).

50. D. Ehst, *Nuclear Fusion,* 19, 1369 (1979).

51. N. J. Fisch, *Phys. Review Letters,* 41, 873 (1978).

52. C. F. F. Karney, et al., Princeton Plasma Physics Laboratory, PPPL-1506, (1979); *Phys. Fluids,* (to be published).

53. S. Y. Yuen, et al., "Characteristics of RF-Driven Steady-State Tokamak Power Reactors," MIT Plasma Fusion Center, RR-79-7, (1979).

54. B. R. Kusse, "Plasma Heating and Confinement with High Current Relativistic Electron Beams in a Torus," Electric Power Research Institute, EPRI-108, (1976); also, "The Injection of Intense Relativistic Electron Beams into Tokamaks to Demonstrate Current-Drive Capability," Physics International Company, PIP-2101, (1978).

55. D. J. H. Wort, *Plasma Physics,* 13, 258 (1971); P. C. Thonemann, et al., *Nature*, 169, 34 (1952).

56. R. Prater, et al., "Conceptual Design of RST, an RF-Driven, Steady-State Tokamak," General Atomic Company, GA-A15229 (1978).

57. J. T. D. Mitchell, et al., *Proceedings of the 7th Intersociety Energy Conversion Engineering Conference,* San Diego, California, p. 1275 (1972).

58. T. Ohkawa, *Nuclear Fusion,* 10, 185 (1970); R. A. Bolton, et al., "New Techniques for Inducing Plasma Current in Large Tokamak Devices," Princeton Plasma Physics Laboratory, PPPL-1415, (1978).

59. Y.-K. M. Peng, "Continuous Tokamaks," Oak Ridge National Laboratory, ORNL/TM-8319, (1978).

60. J. F. Schivel, Princeton Plasmas Physics Laboratory, PPPL-1342, (1977).

61. "Proceedings of the Plasma Heating Requirements Workshop," U.S. Department of Energy, CONF-771241, Edited by H. S. Staten (December, 1977).

62. "Plasma Heating, Fueling and Maintenance - A Technical Assessment," U.S. Department of Energy, HCP/T 1401-01, Edited by H. S. Cullingsford (March, 1978).

63. Y.-K. M. Peng, et al., Oak Ridge National Laboratory, ORNL/TM-6150 (1977).

64. D. L. Jassby and R. J. Goldstone, *Nuclear Fusion*, 16, 613 (1976).

65. J. N. Brooks and R. L. Kustom, *Nucl. Technol.*, 46, 61 (1979).

66. Y.-K. M. Peng and S. K. Borowski, Oak Ridge National Laboratory, ORNL/TM-6112, (1977).

67. S. H. Kim, et al., "Operating Characteristics of the ANL 1.5 MJ Pulsed Superconducting Coil," *Proceedings of the 1979 Cryogenic Engineering Conference* (August, 1979).

68. G. H. Fuller, et al., "Developing Maintainability for Tokamak Fusion Power Systems," McDonnell Douglas Astronautics Company, COO-418426 (November, 1978).

69. M. A. Abdou, et al., *Proceedings of the 8th Symposium on Engineering Problems of Fusion Research*, San Francisco, California, p. 645 (November, 1979).

70. "Fusion Reactor Remote Maintenance Study," Electric Power Research Institute, ER-1046 (April, 1979).

71. J. T. D. Mitchell, *Proceedings of the Third Topical Meeting on the Technology of Controlled Nuclear Fusion*, Santa Fe, New Mexico, p. 954 (May, 1978).

72. I. N. Sviatoslavsky, *Proceedings of the Third Topical Meeting on the Technology of Controlled Nuclear Fusion,* Santa Fe, New Mexico, p. 984 (May, 1978).

73. R. W. Werner, *Proceedings of the 3rd Topical Meeting on the Technology of Controlled Nuclear Fusion,* Santa Fe, New Mexico, p. 994 (May, 1978).

74. D. L. Smith, et al., "Fusion Reactor Blanket/Shield Design Study," Argonne National Laboratory, ANL/FPP-79-1 (July, 1979).

75. M. Abdou, C. W. Maynard, L. Wittenberg, *Nucl. Technol.,* 26, 400 (1975).

76. D. L. Smith, et al., "Analysis of In-Situ Tritium Recovery from Solid Fusion-Reactor Blankets," *Proceedings of the 4th ANS Topical Meeting*, King of Prussia, Pennsylvania, (October, 1980).