INTRODUCTION

Fusion is the primary source of energy in the universe—it is the phenomenon that powers the sun and the stars. In a fusion reaction, two light nuclei combine to produce a heavier nuclide, thus liberating a large amount of energy. Because the reacting nuclei have positive charges, their relative speed must be high enough to overcome the electrostatic repulsive forces. Relative kinetic energies of 100-200 keV are required to fuse hydrogen-isotope nuclei, which have the smallest Coulomb barrier because they are singly charged. For a thermalized mixture of the fusing nuclei (i.e., with Maxwellian velocity distributions), production of an appreciable amount of fusion energy requires a temperature of more than 5 keV (6×10^7 K) and peaks at tens of keV. At these high temperatures, the fusing fuel is in the plasma state, i.e., a completely ionized gas in which electrons are not bound to the nuclei. Obviously, no material wall can contain the fusion plasma, nor is the gravitational confinement of the fusion plasma (as in the sun) practical. The two principal methods for isolating the fusion plasma from the surroundings are magnetic- and inertial-confinement schemes (see Fusion, Magnetic Confinement; Fusion, Inertial Confinement).

The fusion reaction that involves two isotopes of hydrogen, deuterium (D) and tritium (T),

\[ \text{D} + \text{T} \rightarrow ^{4}\text{He}(3.52 \text{ MeV}) + n(14.06 \text{ MeV}), \]

is the least difficult one to achieve in terrestrial conditions. This is because it has the largest cross section and occurs at the lowest plasma temperature. Other fusion reactions of hydrogen isotopes, which are usually referred to as advanced-fuel reactions, have also been considered for terrestrial application and are discussed in Sec. 4.

Examination of the above-mentioned D-T fusion reaction has underlined the potential desirable features of fusion and caused a worldwide quest for a practical fusion reactor. It has also highlighted the challenges in achieving this goal. First, fusion reactions produce a large amount of energy from a small amount of fuel. For example, a 1000-MWe power station requires about 1 kg/day
of D-T fuel, while a similar coal-fired plant would require 2000 tonne/day of coal. The fuel for fusion reactions is readily available. Since for every 6700 hydrogen atoms there is a deuterium nucleus, the water in the Earth's oceans contains a virtually inexhaustible supply of fusion fuel. Because tritium is radioactive with a half-life of 12.3 years, it is not found in significant quantities in nature. It can, however, be produced from lithium, which is an abundant natural resource. If the fusion plasma chamber is surrounded with a blanket containing lithium or lithium compounds, the neutrons produced in a D-T fusion reaction interact with the lithium to produce tritium, which is subsequently burned in the reactor.

Second, fusion reactors have the potential to be environmentally benign. The amount of fusion fuel in the reactor is limited. Therefore, even if an uncontrolled fusion burn occurs, it quickly consumes the available fuel and is extinguished. The “ashes” from fusion reactions are α particles, the nuclei of helium, which are not radioactive. There are two sources of radioactivity in a fusion reactor. The first is the tritium fuel itself. Because the tritium is generated in the reactor and is burned in situ, the amount of tritium in the system is much smaller than the amount of nuclear fuel in a fission reactor. The second source is from the neutrons produced by the D-T reactions, which can transmute the materials in the reactor and make them radioactive. Fortunately, research has shown that by the proper choice of materials and by careful design, the level of induced radioactivity can be kept low (many orders of magnitude smaller than a comparable fission reactor).

Achieving and maintaining the high temperatures, $T$, required for the fusion burn are the most fundamental challenges for fusion energy research. Basically, the fusion fuel should be heated from ambient temperature to about $10^8$ K for fusion reactions to start. At these temperatures, the plasma loses its energy through radiation (bremsstrahlung, synchrotron, and line radiation), heat conduction, turbulent convection of plasma particles, etc. The average time it takes before the energy escapes from the plasma is referred to as the energy-confinement time $\tau_E$. To maintain the fusion temperature, energy must be supplied. Although one of the by-products of fusion reactions (the energetic neutron) leaves the plasma, the other (charged α particle) slows down by colliding with the plasma particles and deposits its energy into the plasma, which sustains the plasma temperature. The necessary condition for a sustained D-T thermonuclear burn is that the "triple product" $n \tau_E$ exceed $2 \times 10^{21}$ s keV/m$^3$ ($n$ is the plasma density). If the triple product is smaller than this value, the α-particle power is not sufficient to sustain the burn and the temperature decreases.

Most of the fusion research in the past 40 years has focused on achieving the necessary plasma temperature and triple product for fusion burn. The required fusion temperature has been attained and several experiments are approaching the required triple product. To date, almost all of the experiments have been performed in hydrogen or deuterium plasmas in order to avoid difficulties associated with handling radioactive tritium. Thus fusion-power production has been small and from D-D reactions (see Sec. 4). Operation with D-T plasmas is planned for two major experiments in the world. The Joint European Torus (JET) at Culham Laboratories has already produced 2 MW of D-T fusion power for 2 s with a small amount of T fuel, in preparation for extensive and powerful (several tens of MW) tritium operation planned for 1995. The Tokamak Fusion Test Reactor (TFTR) at the Princeton Plasma Physics Laboratory is expected to start D-T operation in 1993 with a fusion power of about 10 MW. The fusion ignition and long-pulse operation remain to be demonstrated in a larger experiment, the International Thermonuclear Experimental Reactor (ITER), proposed as a joint venture by the European Community, Japan, the Russian Federation, and the United States. The ITER project is now in the engineering design activity phase with construction planned to begin before the year 2000.

Conceptual fusion-reactor design studies have been carried out since the early 1970s. In addition to assessing whether a confinement system is attractive as a fusion-power reactor, these studies have uncovered potential problems, proposed solutions that are consistent with other reactor subsystems, and identified critical areas that should receive priority in experimental programs. It is a characteristic of this field that many of the research papers appear as laboratory reports or proceedings of conferences. The most up-to-date assess-
Fusion Technologies

1. FUSION-REACTOR OVERVIEW

Although several experiments are approaching the triple product that is required for fusion ignition, many other technological issues must be resolved before an economical fusion-power reactor can be realized. These issues can be loosely divided into two sets, the first of which relates to the plasma itself (Sec. 2). It is necessary to be able to create a plasma reliably, provide auxiliary heating to achieve fusion temperature and burn, and provide control of the major plasma parameters (e.g., \( n \) and \( T \)) for the duration of the burn. For steady-state or long-pulse reactors (e.g., the majority of the magnetic-confinement schemes), the plasma must be replenished with new fuel and the fusion ash (\( \alpha \) particles produced by the D-T reaction) removed from the plasma. Obviously, the plasma technologies are confinement-method specific (see FUSION, MAGNETIC CONFINEMENT; FUSION, INERTIAL CONFINEMENT).

The second set of fusion-reactor technology issues relates to the systems surrounding the plasma (Sec. 3). The functions of these systems are mainly to convert the energy produced by the plasma into heat and then into electricity, to attenuate the neutron flux produced by the D-T plasma, and to produce sufficient tritium to fuel the reactor. Not only must these systems operate reliably in a harsh radiation environment, but they should also convert the plasma energy into electricity efficiently so that the reactor is economically viable. Furthermore, these systems must be built of special materials and designed in such a way that the attractive safety and environmental features of fusion are realized.

The technologies of a fusion reactor encompass many fields of science and engineering such as plasma physics, plasma–material interaction physics, materials science and engineering, heat transfer and fluid dynamics, chemical engineering, electrical engineering, electronics, power-systems engineering, vacuum-system engineering, all aspects of nuclear engineering, and (for magnetic-confinement concepts) superconductivity or (for inertial-confinement concepts) lasers, optics, accelerators, and pulse-power technologies. In some areas, new fields of science (e.g., plasma physics) had to be established, and in others, new frontiers had to be sought.

2. FUSION PLASMA TECHNOLOGIES

2.1 Magnetic Fusion-Energy Reactors

The most advanced magnetic-confinement approach is that of the tokamak (see FUSION, MAGNETIC CONFINEMENT). Therefore the plasma technologies for a tokamak reactor are emphasized here. A conceptual tokamak reactor design is shown in Fig. 1. (Alternative magnetic-confinement concepts are briefly described in Sec. 2.1.5.)

In the tokamak, which is similar in shape to a doughnut, the confining magnetic field is the result of combining a “toroidal field” (the long way around the doughnut), produced by coils outside of the plasma, and a “poloidal field,” which is produced by the toroidal current flowing in the plasma. This combination causes the formation of a magnetic configuration that consists of nested surfaces. Each field line wraps helically and endlessly around one of these nested surfaces. The plasma heat and particles that flow quickly along the magnetic field lines are, therefore, well confined in this magnetic configuration. The plasma then diffuses slowly across the flux surface through cross-field transport processes.

The most efficient way to produce the toroidal current flow in the plasma is by magnetic induction (i.e., transformer action). A solenoid coil is placed in the center of the tokamak doughnut. Then, as the current flows in the solenoid coil, it induces a toroidal current in the plasma. Unfortunately, the toroidal current is induced only when the field produced by the solenoid is increasing (i.e., its current is rising)—a state that is practically impossible to maintain. Therefore, in most of the present-day experiments, when the cur-
rent in the solenoid has reached its maximum, the plasma is shut down, the solenoid current is reset, and a new discharge is initiated. Theoretical and experimental research, however, has shown that the toroidal current can be produced by noninductive means (Sec. 2.1.2).

2.1.1 Plasma Generation and Heating. The plasma is usually generated by breaking down the deuterium gas that fills the plasma chamber. This is accomplished by initiating a current ramp in the solenoid coil set. The resulting induced voltage in the chamber then breaks down the gas. To facilitate plasma generation in most tokamaks, microwaves are also employed to ionize the initial gas partially.

As the plasma current is raised to its final value, the plasma heats Ohmically. However, since plasma electrical resistivity is reduced drastically as plasma temperature increases, an Ohmically heated tokamak plasma can only reach temperatures of about 2–3 keV. Auxiliary plasma heating is, therefore, required to achieve fusion ignition temperatures (~10 keV).

One of the most common methods of auxiliary heating employs an intense beam of high-energy neutral particles (neutral beam), since neutral particles can move through the magnetic field unaffected and penetrate the plasma. The beam particles are then ionized and trapped in the plasma as a result of collisions with plasma ions and electrons. The resulting high-energy ions then “slow down” by Coulomb collisions and deposit their energy in ions and electrons. The mean distance that a neutral ion travels in a given plasma before it is ionized scales directly with its energy. Since it is most efficient to heat the center of the plasma, the beam energy is usually determined by the plasma density, temperature, and size. For a fusion-reactor plasma, beam energies of 100–200 keV are
required for perpendicular injection (perpendicular to the toroidal circumference of the plasma) and 500–3000 keV for tangential injection (e.g., for current-drive purposes, Sec. 2.1.2).

The schematic of a neutral-beam injector is shown in Fig. 2. The neutral beam begins as a hydrogen plasma produced by a radio-frequency (rf) plasma generator or by a high-current discharge. Then the ions leaving the source are accelerated, in one or several stages by a series of electrostatic grids, to their final energy. The resulting ion beam is then neutralized by passing through a neutralizer chamber containing cold hydrogen gas. The beam ions pick up electrons from the neutral gas via the charge-exchange process and travel out of the neutralizer chamber as high-energy neutral particles. The neutral gas, which is ionized by the ion beam, drifts slowly to the walls of the chamber. Meanwhile, the neutralized beam passes through a drift tube before entering the reactor plasma. The small fraction of the original ion beam that is not neutralized in the neutralizer chamber is deflected by a small magnet in the drift tube. At present, the neutral-beam injectors on the TFTR experiments are capable of operating at 120 keV and delivering ~30 MW of heating power to the plasma for a pulse length of ~2 s (easily upgradable to steady-state operation).

Because the efficiency of the gas neutralizer chamber to neutralize the positive-ion beams degrades rapidly for a beam energies larger than ~150 keV, negative-ion beams should be utilized for higher energies. A completely different technology is required for this scheme. To produce a beam of negative ions, either the plasma in the ion source is seeded with cesium atoms, or the ions extracted from the source are reacted with specially coated surfaces. The negative ions are then accelerated by either electrostatic or rf quadrupoles. The high-energy negative-ion beam is then neutralized either through the charge-exchange process in a xenon plasma or by using the laser photodetachment process. Negative-ion beam accelerators are under development. At present, negative-ion sources are capable of 50-keV $\text{H}^-$-ion beams with a current of ~0.5 A at steady state. Stable $\text{H}^-$ beams at 300 keV for 0.5 s have also been achieved. The first heating experiments with negative-ion beams, which are planned for JT-60U (a large toka-

FIG. 2. A schematic of a positive-ion neutral-beam injector system.
mak experiment at the Japan Atomic Energy Research Institute) around 1995, will employ 500-keV deuterium beams (22 A for 10 s).

Another common method of auxiliary heating is by rf waves. In principle, the electromagnetic wave energy can be converted into plasma-particle kinetic energy through resonant wave-particle interaction. Because the plasma consists of charged particles, a variety of wave modes exist in a magnetized plasma and several rf heating schemes are possible and have been successfully demonstrated on present experiments (e.g., Alfvén, lower-hybrid, and electron-cyclotron resonance heating, etc.). One of the most advanced rf-heating techniques is ion-cyclotron range-of-frequencies (ICRF) heating. Depending on the wave frequency and the launcher design, ICRF waves can heat electrons, ions, or a minority ion species. In the latter scheme, a deuterium plasma is doped with a small amount of hydrogen or helium, and the ICRF wave frequency is matched to the cyclotron frequency of this minority species (∼100 MHz, similar to television broadcast frequencies). Then the rf wave heats these minority species to the MeV range of energies, and their energy is transferred to plasma electrons and ions by Coulomb collisions.

All rf heating systems are, in essence, similar to a radio transmitter. They include an oscillator, an amplifier, a transmission line, and a launching structure inside or at the plasma chamber wall. The oscillator and amplifier can be a tetrode, a klystron (a linear beam device having features of both klystron and tetrode), or a solid-state device based on field-effect transistors. The transmission line is a coaxial cable or waveguide, and the launching structure can be either an antenna or a waveguide. The major fusion-specific component of the rf heating system is the launcher.

In order for the launched rf power to couple well to the plasma, the wave number and polarization of the launched wave must be strictly controlled. Then the wave is able to penetrate the plasma and be absorbed at the desired location. To protect the launcher from electromagnetic radiation from the plasma, a "Faraday shield" is usually placed in front of it. The launcher should also be hardened against damage from the neutrons and charged particles from the plasma (Sec. 3.1).

2.1.2 Noninductive Current Drive. As mentioned before, plasma current driven by inductive means cannot be sustained indefinitely. Even though a pulsed tokamak reactor with a burn time of a few hours is a possibility, extensive research has also been directed towards noninductive means of driving the plasma current.

Theoretical research in the early 1970s predicted that in a sufficiently hot plasma, the radial pressure gradient and the dynamics of plasma flow on the flux surfaces combine to produce a self-driven current, usually referred to as the bootstrap current. Existence of the bootstrap current was experimentally confirmed only recently. Major tokamak experiments have operated with up to 80% of the plasma current driven by the bootstrap effect. Unfortunately, no tokamak plasma equilibrium has yet been found (either theoretically or experimentally) in which all of the plasma current is driven by the bootstrap effect. The difference between the plasma and bootstrap currents must still be driven by some other means.

Research has shown that both neutral-beam and rf-wave heating can be used to drive the plasma current. For example, by tuning the ICRF launching structure, most of the energy of rf waves can be used to accelerate the electrons parallel to the field line, which generates toroidal currents. The efficiency of these noninductive current-drive schemes, however, is low, and most steady-state tokamak reactor designs operate in regimes that maximize the bootstrap current, reducing the necessary current-drive power to a manageable level (∼100 MW of current-drive power in a 1000-MWe power station).

2.1.3 Magnets. In addition to the toroidal-field magnets and the "Ohmic-heating" solenoid discussed above, other magnets are required for a tokamak. In principle, since a current-carrying loop can move along its axis freely, a tokamak plasma can move up or down and hit the plasma chamber walls. To keep the plasma at its location, a pair of "equilibrium-field" coils with currents flowing in the toroidal direction (but opposite to that of the tokamak plasma) are required. These coils are usually located symmetrically at about the toroidal midplane of the tokamak.
torus (one above and one below) and have radii that are usually larger than that of the plasma.

An important measure of the efficiency of a magnetic-confinement concept is the plasma $\beta$, the ratio of the plasma pressure to the pressure exerted by the external magnetic field that confines the plasma. The higher the $\beta$ value, the smaller and more economical the external magnets can be. A major accomplishment of plasma physics research during the last decade has been achieving an understanding of the relationship of the plasma $\beta$ to plasma shape, current, toroidal field, etc. It has been shown, both theoretically and experimentally, that elliptical or D-shaped plasma cross sections (with an elongation of about 2) can lead to improved plasma $\beta$. Plasma shaping, however, requires additional coils, all having toroidal currents, strategically placed around the plasma.

Most of the magnetic fusion-reactor concepts use superconducting magnets because the power consumed by normal-conducting magnets would be very large. Basically, superconductivity occurs in certain materials when their temperatures are reduced below some critical temperature. Most of the superconducting magnets, at present, use niobium–titanium (NbTi) or niobium–tin (Nb$_3$Sn) alloys as the superconducting material cooled by liquid helium to temperatures of about 4 K (some designs use either of these superconductors at lower temperatures). The commercial superconducting magnets available today mainly use NbTi and are of medium size (e.g., those used for magnetic-resonance imaging). Nearly all experience with large-size superconducting magnets to date is in high-energy physics (e.g., accelerators, bubble chambers) or in fusion. Compared to other superconducting magnets available today, fusion magnets are usually larger (many meters in diameter), may have noncircular cross sections (e.g., D-shaped toroidal-field coils), and have a high magnetic stored energy (tens of GJ). As a result, the design of these magnets is dominated by the structure, which is embedded in the coil to support the massive electromagnetic forces. Furthermore, the design of fusion magnets is also severely constrained by fusion-specific issues, such as allowable location for the coils and the structure, ability to operate in a radiation environment, and capability to produce and withstand high rates of field change (i.e., several T/s for poloidal-field coils).

Because of the engineering constraints, the peak field produced by a practical superconducting magnet is usually well below the critical field limit of the superconducting material. The maximum field produced by a Nb$_3$Sn superconducting magnet (a small-bore solenoid) is 21 T. The maximum field produced by a fusion magnet with a noncircular cross section is $\sim$ 9 T. It should be noted that the magnetic field strength on the plasma axis in a tokamak is considerably lower than the peak field on the toroidal-field magnets because of geometrical effects. For the ITER design, research and development are directed towards superconducting magnets with peak fields of 11–14 T, while conceptual tokamak-reactor designs have envisioned magnets with a peak field ranging from 14 to 21 T (depending on the value of plasma $\beta$).

A cross section of a toroidal-field magnet is shown in Fig. 3. In this type of design, filaments of niobium and tin are twisted together in groups and embedded in a copper sleeve to form a cable. Several cables are then placed together in a steel conduit, which also has passages for liquid-helium coolant. This conduit is heat treated so that the niobium and tin filaments interact, forming the superconducting alloy. A layer of insulating material is placed on the outside of the conduit, and layers of the superconductor conduits are interleaved with steel supports. The coil is then placed in a stainless Dewar or cooling vessel.

The superconducting magnets must be kept insulated from the nearby plasma because neutrons produced by fusion reactions can damage the superconducting material. Furthermore, so much heat would be generated by the interaction of neutrons with the magnet material that refrigeration of the magnets would be impractical. In order to attenuate the neutron flux on the magnets, they are usually placed behind the blanket and shield components (Sec. 3.2).

2.1.4 Particle Exhaust and Fueling. In most modern tokamaks, the plasma-shaping coils are energized such that a magnetic separatrix is formed inside the plasma chamber (Fig. 4). In the region inside this separatrix,
called core plasma, the nested flux-surface structure of the tokamak is preserved. The field lines outside the separatrix, however, leave the vicinity of the plasma within a single poloidal transit (but many toroidal transits) and intersect material walls (divertor plates) in certain locations. The heat and particles from the core plasma diffuse outward through the separatrix into the scrapeoff-layer region and then flow rapidly along the field lines to the divertor plates. Clearly, very high heat and particle fluxes appear at the intersection of the separatrix with the divertor plate (strike point). Note that there is a large body of experimental evidence that shows that controlling the scrapeoff-layer plasma properties has profound influence on the quality of core-plasma confinement. In fact, the best tokamak discharges have been obtained in a divertor configuration.

The plasma that intersects the divertor plates is neutralized through surface recombination processes. Some of the neutral atoms are ionized, follow the field lines, and flow back to the divertor plate and, in effect, are recycled in the divertor region. If the separatrix plasma density is high enough, a high-recycling divertor regime is formed in which more than 90% of the neutral particles from the divertor plate are re-ionized in the plasma and are recycled. This mode of operation spreads the heat and particle load on the divertor plate and reduces the peak fluxes at the strike point. Further reduction in heat and particle fluxes can be achieved by slanting the divertor plate with respect to the field lines (Fig. 4).

At present, the design of the divertor represents one of the most fundamental challenges in designing fusion reactors, and it is not yet clear whether the high-recycling divertors will be able to reduce the particle and heat fluxes to manageable values. Many other concepts are, therefore, being pursued. In one, the plasma near the divertor is doped with impurities that cool the divertor plasma through line radiation. Another involves terminating the divertor plasma in a dense, but much cooler, neutral gas. Both of these concepts would spread the heat and particle load over a much larger area in the divertor.

Most of the neutral atoms that are produced on the divertor plate are ionized and flow back to the divertor plate. The rest, which are a mixture of fusion fuel (D and T) and \( \alpha \) particles, is pumped, which removes the fusion ash from the system. Cryosorption pumps (see CRYOGENICS) are a leading technology for fusion reactors. Cryopumps include an array of sieves that are cooled to liquid-helium
temperature. The sieves are surrounded by chevrons that are cooled to liquid-nitrogen temperatures. Impurity gases such as oxygen and nitrogen are adsorbed by the chevrons while hydrogen isotopes condense on the sieves. Helium can also be trapped by the sieves if liquid argon is sprayed onto the surfaces to overlay the helium atoms. Cryopumps are simple and attain very high pumping speeds at low pressures ($10^{-7}$ atm). Their drawbacks are that they require periodic warming up to remove the gases trapped on the surface, and that the trapping of tritium onto the cryopump surfaces will result in a high inventory of tritium in the reactor. An alternative approach is to design the divertor system such that the neutral pressure at the front of the pumping duct is relatively high ($\sim 10^{-5}$ atm). In this case, nonadsorbing pumps, such as turbomolecular pumps, can be used. The exhaust gas from the reactor is processed to remove the helium ash and impurities.

For the fusion burn to continue, the reactor must be refueled with the D and T atoms that have been exhausted from the reactor, together with a supply of fuel to replenish that which was burned. In most current experiments, fueling is performed primarily by puffing the fuel gas at the edge of the plasma. This fueling technique is not very effective because most of the fuel does not reach the plasma center. An alternative method, also currently used in major tokamak experiments, is to inject high-speed pellets of frozen fuel. A typical pellet is several millimeters in diameter and is accelerated to speeds of $\sim 2 \text{ km/s}$ by high-pressure gas propellant in a gas gun (Fig. 5).

2.1.5 Alternative Magnetic-Confinement Concepts. Of the many magnetic-confinement concepts that have been investigated in the past 30 years, the tokamak is the most advanced. Other promising concepts include the stellarator, reversed-field pinch, and field-reversed configurations (see FUSION, MAGNETIC CONFINEMENT). Each has desirable features that simplify or eliminate plasma support technologies. These features are highlighted below.

The confining magnetic field structure in a stellarator is similar to that of a tokamak, although it is produced by complicated coils external to the plasma. The major advantage of the stellarator is that because it does not require a toroidal current in the plasma, it can operate in steady-state mode without a current-drive system.

The reversed-field pinches also have nested flux surfaces similar to those in the tokamak. However, most of the toroidal field is produced by current flowing in the plasma. The direction of the toroidal field is reversed near the plasma edge and only a weak external toroidal field is required to maintain the plasma configuration. Ohmic heating to ignition may also be possible. Elimination of the auxiliary heating system and major simplification of the external magnets are major features of this concept. Steady-state operation, however, requires a noninductive current-drive technique.

The field structure in a field-reversed configuration is very different from the above-noted concepts; it resembles that of a current loop (plasma) in an external solenoidal field. The experimentally achieved plasma $\beta$ is very high, the external magnets are simple, and it is more amenable to direct conversion of charged-particle energy to electricity. At this
time, however, the experimental data base for this concept is limited.

2.2 Inertial Fusion-Energy Reactors

Because of its inertia, material at fusion temperature stays together for a finite, albeit short, period of time (nanosecond time scale). Since fusion ignition and burn require attainment of the triple product $nT_{E}$, inertial-confinement fusion can be achieved if the fusion fuel is compressed to a very high density (1000 times liquid density). Such a compression can also adiabatically heat the fuel to fusion temperatures.

In the inertial-confinement scheme (see Fusion, inertial confinement), a small pellet containing the fusion fuel is bombarded by powerful lasers or particle beams (typically 100–1000 TW of power is needed). Ablation of the material on the pellet surface, therefore, causes an implosion that compresses the center of the pellet to high density and high temperature. A small region in the center of
the pellet can become hot enough to initiate a fusion reaction. The energy released from the fusion reactions can heat the surrounding colder fuel to fusion temperature, resulting in a fusion-burn front that rapidly propagates outward and consumes the rest of the fusion fuel before the pellet flies apart.

Candidate drivers include lasers and particle-beam accelerators. Solid-state neodymium (Nd) glass lasers are presently used in inertial-confinement experiments. The efficiency of this laser (~1%) is too low for reactor applications, however. Krypton fluoride (KrF) gas excimer lasers are predicted to have a higher efficiency (~10%) and are under development as a potential driver. Particle beams are more attractive since they have higher intrinsic efficiency (~25%). Low-energy light-ion beams with a high current are used in the present experiments, but high-energy heavy-ion beams with a low current, which are under development, are thought to be the best driver for a power reactor. The technologies of the above drivers are covered in detail in Fusion, Inertial Confinement, Sec. 5.

The ratio of the energy released by pellet explosion to the driver energy is usually referred to as the pellet gain $G$. A high pellet gain (typically 100 for a D-T reaction) is required to compensate for the inefficiencies in the driver and thermal-power-conversion system. Achieving ignition and high pellet gain imposes severe constraints on the driver and pellet designs. First, the driver pulse should be shaped properly in order to maximize the fraction of the driver energy that is used in compressing the pellet. Second, and more importantly, the pellet implosion should be highly uniform (pressure nonuniformity should be less than 1%). This requires that the pellet surface should be very smooth (less than 100-Å imperfections). Also, the pellet should be heated uniformly, either by ensuring that the driver beam strikes the pellet uniformly (direct drive) or by converting the driver energy into x rays in a Hohlraum (indirect drive).

In an inertial-fusion reactor, the pellet microexplosion occurs in the reactor chamber. Practical constraints on the size and lifetime of the reactor chamber limit the yield of pellet explosions to a maximum of 500–1000 MJ (a pound of high explosive releases about 2 MJ of energy). Therefore, a typical power reactor (1000 MWe) requires a pulse rate of a few Hz.

During each pulse, a pellet is injected several meters into the reactor chamber and should be tracked until it is in the precise location and alignment for the driver. The driver should deliver its energy with the precise pulse shape and focus on the target. After the implosion, the debris of the pellet (and the Hohlraum, if used) should be evacuated from the reactor chamber so that the beam can propagate through the reactor chamber with minimum loss while maintaining adequate focus. These pulses should be repeated reliably millions of times per year, and the wall of the reactor chamber should be able to withstand the many microexplosions (see Sec. 3.6). Mass production of millions of highly smooth pellets at an acceptable cost is also a challenging technological issue. Research is ongoing on most of these issues.

3. FUSION-REACTION TECHNOLOGIES

3.1 First Wall, Blanket, and Shield

3.1.1 Function. The plasma reaction chamber is the first material boundary that faces the plasma. Usually referred to as the first wall, this region is subjected to bombardment by electromagnetic radiation from the plasma (mainly x rays), the flux of charged particles diffusing from the plasma, fast neutral atoms produced by charge-exchange processes in the plasma, and energetic neutrons produced by the D-T reaction. Each component of this harsh environment imposes severe constraints on the first-wall design (Sec. 3.1.2). The energy deposited in the plasma by the auxiliary heating system and by the fusion $\alpha$ particles must eventually leave the plasma. This energy appears mostly as surface heat flux on the first wall (radiation and charged-particle flux) and should be removed as useful heat. Sputtering of the first wall by plasma and neutral particles would cause erosion of the first-wall material, and the first wall may have to be coated with a low-Z material so that eroded material will not contaminate the plasma dramatically.

The fusion-produced neutrons have a long mean free path between collisions with solid and liquid materials (typically ~10 cm for 14-MeV neutrons). Most of these neutrons pass through the first wall and are absorbed in the blanket. The main function of the blanket
is to moderate and capture these neutrons, extract their energy, and simultaneously breed enough tritium to replace that which is burned in the fusion process.

Typically, blankets are 0.8 to 1 m thick and attenuate the neutron flux by about 2 orders of magnitude. In MFE reactors, a shield is usually placed behind the blanket in order to attenuate the neutron flux by about 4 orders of magnitude so that a superconducting magnet can be located behind the shield. Such shields are typically 0.5 to 1 m thick.

### 3.1.2 Tritium Breeding

The primary reactions leading to tritium production are

\[ ^{6}\text{Li} + n \rightarrow ^{4}\text{He} + ^{3}\text{He} \]

and

\[ ^{7}\text{Li} + n \rightarrow ^{4}\text{He} + n \]

The first reaction leads to energy production (4.8 MeV released per reaction). The second consumes energy (2.87 MeV) but it gives rise to a second neutron that can subsequently react with \(^{6}\text{Li}\). The natural abundances of these lithium isotopes are 7.4% \(^{6}\text{Li}\) and 92.6% \(^{7}\text{Li}\). The technology for enriching the natural lithium in the \(^{6}\text{Li}\) isotope is available and is not very expensive. In some conceptual blanket designs, therefore, enrichment of up to 20% in \(^{6}\text{Li}\) is utilized in order to optimize the nuclear performance of the blanket.

The number of tritium atoms produced in the blanket for each D-T neutron is referred to as the tritium breeding ratio (TBR). Since each fusion reaction uses one tritium atom to produce one neutron, a TBR of 1 or greater is required for tritium self-sufficiency. In a fusion reactor, not all of the fusion neutrons are absorbed in Li; some are absorbed in the structural material or coolant of the blanket. Therefore, in certain designs, a neutron-multiplying material is added to the blanket. Of particular interest is beryllium, which interacts with one neutron and produces two neutrons. These extra neutrons then compensate for neutron absorption in the blanket structural material.

The tritium-breeding materials can be lithium or lithium compounds. The choice is usually based on high-temperature capability; chemical compatibility with the coolant and the structural material; the desire for a high tritium-release rate (the tritium produced leaves the material rapidly, leading to a low tritium inventory in the blanket); the availability of, or the need for, a neutron-multiplier material; the tritium-recovery method; simplicity of design; etc. The first and most obvious choice is lithium itself in liquid form, to be used as both the breeder and coolant. Lithium eutectic compounds such as lithium-lead (Li\(_{62}\)Pb\(_{38}\)), aqueous solutions of lithium salts, and fused lithium salts such as FLiBe can also be used as the coolant and breeder. Lastly, solid lithium compounds, such as lithium oxide (Li\(_2\)O) and lithium orthosilicate (Li\(_4\)SiO\(_4\)), have also been considered.

### 3.1.3 Structural Material

Material choices are critical in designing the fusion first wall and blanket. Structural material choices are usually based on their ability to operate at high temperatures, to withstand radiation damage by neutrons, and to have only a minimum amount of induced radioactivity caused by the fusion neutrons. Radiation damage is the most important criterion because it has the greatest influence on material performance and lifetime. During neutron irradiation, the absorption of a neutron by the material through scattering and absorption reactions can induce two major nuclear effects: atomic displacement and production of impurities and gas (hydrogen and helium).

During the scattering events, the energy transferred from the neutron to the nuclei can cause an atom in the solid structure to be displaced from its equilibrium lattice site, leaving behind a vacancy in the lattice. A scattered neutron, as well as the displaced atom, can interact with other atoms in the lattice and cause displacement cascades. This process continues until the energy transferred to the lattice atoms becomes low enough so that it can be absorbed as vibrational energy by the lattice. The displaced atoms themselves usually come to rest at locations that are not regular lattice sites and, therefore, are referred to as interstitials. Even though each neutron produces many vacancy-interstitial pairs, most recombine rapidly and leave quite small net levels of point defects in the lattice structure. Voids in the material form if the vacancies are mobile enough (if the material temperature is high enough) to combine and if some agent, such as a small amount of gas,
is present to stabilize the embryonic void. The void will then subsequently grow and cause dimensional changes of the structural material, referred to as "swelling," which can easily reach several tens of percent. Because the swelling rate depends on the material temperature and neutron flux, which are not uniform in a fusion blanket, the swelling rate will not be uniform, which can lead to large internal stresses in the structure.

The strength (or hardness) of annealed metals usually increases under irradiation and typically both the yield strength and the ultimate tensile strength increase. Unfortunately, this increase in hardness is generally accompanied by a loss of ductility, or embrittlement (material fails under small levels of strain). Embrittlement can have several causes; the most important is related to the large amount of helium gas that is produced in the structural material through \((n,\alpha)\) reactions. At high temperatures, the helium atoms become mobile and diffuse to the boundaries between grains in the metal. Subsequently, helium bubbles form and intergranular failure occurs.

In addition to swelling and embrittlement, physical, mechanical, and thermal properties of the material change by neutron irradiation (e.g., through the production of point defects or by the change in the composition of the alloying elements). As mentioned before, the irradiation effects are very sensitive to both neutron flux and fluence and also to the operating temperature of the material. Chemical interaction among the structural and breeding materials and the coolant, as well as gradients in the temperature, flux, and stress, can also have a dramatic impact on the response of the material to radiation and on its lifetime.

During the 1970s, most of the research effort on structural materials was focused on stainless steels because of the experience in the fission industry. During the 1980s, research focused on low-activation structural material (nuclear waste produced in a reactor built with these materials does not require geological disposal as is needed for fission plants). Primary materials in this category are low-activation ferritic steels (in which induced activation is reduced by substituting for certain alloying elements) and vanadium alloys. Recently, ceramic composites such as silicon carbide (SiC) and fiber-reinforced SiC-matrix composites have received considerable attention. Ceramic composites are of particular interest because the level of induced radioactivity is very low, as shown in Fig. 6 (6 orders of magnitude lower than a metallic structure one day after shutdown).

Because of the radiation damage, the lifetime of the first wall and blanket of a D-T reactor is limited and they should be replaced periodically (about every three to six years would be acceptable). Unfortunately, experimental radiation-damage data are limited to a small number of materials and to a low value of neutron fluence. Furthermore, experiments are usually carried out in fission reactors (14-MeV fusion neutrons cause a much higher production of gases, which drastically impacts the material behavior). Fusion-reactor-material test facilities, therefore, have been proposed that incorporate an intense 14-MeV-neutron source.

**3.1.4 Conceptual Designs.** In addition to liquid lithium or lithium eutectic compounds, water and helium gas have been considered as the coolant. In principle, engineering issues such as chemical compatibility and operating temperature determine the coolant once the structural and breeder material are chosen. Three design types are favored at present. The first is a water-cooled system with steel struc-
ture and a solid breeder material such as lithium oxide (Fig. 7). This combination requires a neutron multiplier (e.g., Be) in order to achieve a TBR > 1. This design builds upon pressurized-water fission-reactor experience. The main drawbacks to this design are that the outlet temperature of the water coolant is low, which results in a low thermal-to-electrical conversion efficiency, and that the radioactivity induced in the steel structure, even though lower than a fission reactor, is still large by fusion-reactor standards. Tritium leakage into the water coolant also requires expensive tritium-separation systems.

The second type of design (Ehst et al., 1987) uses vanadium alloy as the structural material and liquid lithium as the coolant (no multiplier). Actually, because most of the energy is deposited in the lithium itself, the lithium is a heat-transport fluid rather than a coolant—a major design simplification. Because lithium is electrically conductive, the magnetohydrodynamic (MHD) pressure associated with a conductive fluid moving perpendicularly to a magnetic field can lead to high pressure and high pumping power. Reducing the pressure and pumping power to a manageable level will require either a low-velocity, large-area flow channel or nonconducting coolant-channel walls. The thermal-to-electrical conversion efficiency is high. From safety and environmental viewpoints, it ranks better than the water-steel design discussed above; however, lithium fires, which may be caused by an accident, are safety concerns. Lastly, vanadium manufacturing is not as mature as that of steel.

The third class of designs uses a ceramic composite such as SiC composite as structural material, inert helium gas as the coolant, and a solid breeder material such as lithium oxide (Fig. 8). This type of design can achieve the full safety and environmental potential of fusion because the radioactivity inventory is several orders of magnitude lower than fission reactors, and the level of decay afterheat is similarly smaller. Although ceramic composite manufacturing is still in its infancy, it is being pursued vigorously by the aerospace industry.

3.1.5 Tritium-Recovery Systems. Tritium that is bred in the blanket should be recovered rapidly to minimize the T inventory.
in the reactor. Designs with solid breeder material rely on diffusion of T out of the solid. As such, the breeder is usually fabricated in small pellets to increase the surface area of the solid. The temperature of the breeder material is also chosen to maximize T diffusion. The T diffusing from the breeder is transported out of the blanket by a purge stream of helium gas. In designs with liquid-lithium breeder, T is in the form of lithium tritide. A small side stream of liquid lithium is brought into contact with an equal volume of a molten lithium salt. Lithium tritide tends to pass to the salt stream and can be readily separated because it is several times lighter. Tritium is then recovered by electrolysis. The molten-salt process is similar to that developed to reprocess fission-reactor fuel.

3.2 Power-Conversion Systems

Most conceptual fusion-reactor designs utilize thermal-to-electrical power-conversion systems similar to those in existing fission or coal-burning power plants. The energy produced in the reactor is used to heat a gas (usually steam) to high temperatures. The hot gas, in turn, expands and cools down in a turbine, producing mechanical energy that is converted to electricity. Of the three types of blanket designs discussed above, the water-cooled system probably needs an intermediate coolant loop and has a low conversion efficiency. Liquid-metal-cooled designs also require an intermediate liquid-metal loop because tritium is present in the primary coolant. Gas-cooled designs can either use a Brayton cycle (coolant sent directly to the turbine) or advanced Rankine steam cycles similar to those proposed for near-term coal-fired power plants. Intermediate loops can be eliminated from these designs if care is taken to keep the tritium separated from the primary coolant.

The power produced in the fusion plasma is carried away by radiation (microwaves and x rays) and by charged particles with energies from several tens of eV (scrapeoff-layer
plasma) to several keV (bulk plasma) to the MeV range (fusion by-products) depending on the magnetic-confinement concept. Because the "effective" temperatures of the charged particles or radiation photons are very high, the Carnot bound for the efficiency of converting their energy into electricity is virtually 100%. Furthermore, such direct-conversion schemes can avoid expensive and complicated "conventional" thermal-power-conversion systems.

Several schemes have been proposed for converting charged-particle energy directly into electricity. Such a system would operate as an inverse of a particle accelerator (i.e., the charged particles are slowed down and absorbed by a series of electrodes at different voltages). Since the plasma particles move readily along the magnetic field lines, direct conversion of charged-particle energy is more amenable to confinement schemes in which some field lines leave the main plasma (a field-reversed configuration as opposed to a tokamak). Furthermore, each electrode only absorbs particles with energies close to its biasing voltage. Therefore if the plasma exhaust has a Maxwellian energy distribution, the conversion efficiency for a practical system (limited number of electrodes) is not very high. In certain magnetic-confinement concepts (such as mirrors), the plasma has a positive voltage with respect to the ground and, therefore, the energy distribution of exhausted plasma particles is narrow. In this case, conceptual designs predict a conversion efficiency of ~80% (Logan et al., 1984).

Several schemes have also been proposed for converting plasma radiation into electricity. In one proposal, the plasma radiation is utilized to superheat the coolant for use in an advanced thermal-conversion scheme such as MHD generators. In another, solid-state rectennas (a rectenna is combination of an antenna and a rectifier) have been used to convert synchrotron radiation from the plasma into electricity. These devices have been studied for conversion of microwave energy into electricity (for use with solar-power satellites, which transmit their energy to Earth stations through microwaves). One of the major difficulties, which is similar to that of the charged-particle conversion scheme, is that each rectenna is tuned to a certain frequency range while the synchrotron radiation from the plasma has a broadband spectrum.

3.3 Inertial Fusion-Energy Reactors

The major difference between IFE and MFE reactors is the first-wall design. While the first-wall design is a challenge in an MFE reactor, the design constraints are even more severe for an IFE reactor. This is because of the thermal and structural fatigue caused by millions of microexplosions as well as erosion caused by debris. Any beam-steering or focusing element that is in direct sight of the explosion should also be protected.

Several innovative designs have been suggested for an IFE reactor. They include

1. inclusion of sacrificial coating on the first wall that would be reapplied in situ occasionally,
2. a thick flowing layer of small lithium ceramic granulates,
3. a wetted first wall in which liquid metals such as lithium or lead diffuse through the wall onto its surface, and
4. a thick flowing waterfall (or jet) of lithium.

These concepts are reviewed in detail in Fusion, Inertial Confinement, Sec. 6 and Fig. 12.

In an MFE reactor, the external magnets are usually placed as close as possible to the plasma in order to minimize their costs. The first wall, blanket, and shield are, therefore, squeezed into the space between the plasma and the magnet, making maintenance operations difficult. In an IFE reactor, there are no magnets. The first wall and blanket have a more regular geometry (cylindrical or spherical), which would facilitate maintenance.

4. ADVANCED FUELS

In addition to the D-T fusion reaction, the following have cross sections that are large enough to be of interest in terrestrial applications (as expected, all involve hydrogen isotopes):

\[ \text{D} + \text{D} \rightarrow ^3\text{He}(0.92 \text{ MeV}) + n(2.45 \text{ MeV}), \]
\[ \text{D} + \text{D} \rightarrow ^1\text{T}(1.01 \text{ MeV}) + p(3.03 \text{ MeV}), \]
\[ \text{D} + ^3\text{He} \rightarrow ^4\text{He}(3.67 \text{ MeV}) + p(14.67 \text{ MeV}). \]

The two D-D reactions occur with equal probability. The resulting T and \(^3\text{He}\) produced by these reactions can subsequently undergo D-T
and D-\(^3\)He reactions. Roughly 40% of the fusion energy is released in neutron power in a D-D reactor. Even though the D-\(^3\)He reaction does not produce neutrons, D-D reactions occur in a D-\(^3\)He plasma, which lead to neutron production.

These advanced-fuel reactions require a triple product that is \(\sim 10\) times larger than that required for D-T fusion. They also require a higher plasma temperature (40–60 keV) and, obviously, require more efficient confinement concepts. It also appears that the required pellet gain for an IFE reactor is so large that an advanced-fuel IFE reactor may be impractical.

The major advantage of advanced fuels is that no T breeding is required, making blanket design less difficult. Tritium-handling systems, however, are still needed because T is produced in the plasma, albeit in smaller quantities. Because the neutron flux in a D-\(^3\)He reactor is 10 times less than in a D-T reactor, the first-wall and blanket component may last the reactor lifetime and need not be replaced. In should be noted that the choice of structural material has a more drastic impact than advanced fuels on achieving the safety and environmental features of fusion (the amount of induced radioactivity in a D-\(^3\)He reactor is only a factor of \(\sim 5\) lower than that in a D-T reactor with similar structural material, while the induced radioactivity can be reduced by up to 6 orders of magnitude by choosing the appropriate structural material). Lastly, \(^3\)He is not available in appreciable quantities on Earth. The proposed eventual source of \(^3\)He is the regolith of the lunar surface, which contains about 10 parts per billion of \(^3\)He, deposited by the solar wind.

5. CONCLUSIONS

Long-term energy needs require energy sources that are inexhaustible and environmentally benign. Fusion is one of the very few long-term energy options.

Achieving and maintaining the high temperatures required for the fusion burn are the most fundamental challenges for fusion-energy research. Technical progress towards achieving a fusion reactor has been dramatic in the past decade. The required fusion temperature has been attained, and several experiments are approaching the required triple product for maintaining fusion burn. These results, together with the D-T experiments planned for JET and TFTR, serve to demonstrate the scientific feasibility of fusion power.

These accomplishments have initiated an internationally coordinated strategy to address the remaining technological issues that will demonstrate the economic feasibility of fusion power. The first step in this direction is the International Thermonuclear Experimental Reactor (ITER), proposed as a joint venture by the European Community, Japan, the Russian Federation, and the United States.

Conceptual fusion-reactor design studies have been carried out since the early 1970s in order to uncover potential problems, to propose solutions that are consistent with other reactor subsystems, and to identify critical areas that should receive priority in experimental programs. These studies have also shown that potential safety and environmental attributes of fusion can be realized at least at the level of a self-consistent conceptual design study.

At present, the national energy strategy of the United States calls for a fusion demonstration power reactor to begin operation, following the ITER program, in 2025. For this goal to be realized, several other major facilities are required in order to develop the necessary technologies. Steady-state tokamak experiments, a high-fluence 14-MeV-neutron source for material testing, and a technology device for testing blanket concepts (usually referred to as the volumetric neutron source) are necessary and are in various stages of planning.

GLOSSARY

**Auxiliary Heating System:** The system which heats the plasma to fusion temperatures (\(\sim 10^8\) K).

**Blanket:** The components that surround the fusion chamber and convert the fusion energy into heat. In a D-T fusion reactor, the blanket contains lithium or lithium-containing compounds to breed the tritium fuel.

**Breakeven:** Refers to a fusion plasma condition in which the fusion energy released
is equal to auxiliary-heating energy needed to sustain the plasma temperature.

**Cryopump:** A high-vacuum pump that uses plates cooled to cryogenic temperatures to adsorb gas atoms and molecules.

**Current Drive:** A system to drive an electric current in a magnetically confined plasma.

**Direct Drive:** The inertial-confinement fusion method in which the driver directly irradiates the pellet surface.

**Driver:** A very high-energy laser or a particle beam used for implosion of the pellets (inertial-confinement fusion).

**Embrittlement:** A process caused by neutron irradiation which leads to a decrease in the ductility and fracture toughness of the material.

**Energy-Confinement Time:** The average time it takes before the energy escapes from the plasma.

**First Wall:** First material wall facing the plasma in a fusion device.

**Gain:** The ratio of fusion energy released during pellet implosion to the driver energy.

**Hohlraum:** An enclosure in which the energy of the driver is converted into x rays. It is used in the indirect-drive approach to the inertial-confinement fusion.

**ICRF Heating:** An efficient method of auxiliary heating of the magnetically confined plasmas by using electromagnetic waves. ICRF is an acronym for ion cyclotron range of frequency.

**ITER:** The acronym for International Thermonuclear Experimental Reactor, a large tokamak experiment proposed as a joint venture by the European Community, Japan, the Russian Federation, and the United States.

**Ignition:** A fusion plasma condition in which the fusion energy released supplies enough energy to the fusion plasma to sustain the fusion reactions.

**Indirect Drive:** The inertial-confinement fusion method in which the driver energy is first converted into x rays in a hohlraum. The x rays then irradiate the pellet surface.

**Inertial-Confinement Fusion:** A method for isolating the fusion plasma from the material wall. In this scheme, bursts of fusion energy are produced by using lasers or particle beams to implode small pellets containing deuterium and tritium. The fusion reaction takes place in the short time before the pellet flies apart.

**JET:** An acronym for Joint European Torus. It is the largest tokamak experiment in the world today and is located at Culham Laboratories in the United Kingdom.

**JT60:** A large tokamak experiment located at Japan Atomic Energy Research Institute at Naka, Japan.

**Low-Activation Material:** Certain material for which the radioactivity induced by neutron irradiation is low. In particular, disposal of such material does not require geological storage as is required for nuclear-fission waste. It is proposed that fusion-power reactors will be built of this sort of material.

**Magnetic Confinement Fusion:** A method for isolating the fusion plasma from the material wall. In this scheme, the plasma is confined in a magnetic bottle.

**Neutral Beam Injector:** An efficient method of auxiliary heating of magnetically confined plasmas. It employs an intense beam of high-energy electrically neutral particles, since neutral particles can move through the magnetic field unaffected.

**Ohmic Heating:** Heating of the plasma by the electric current flowing in the plasma.

**Pellet Injector:** A method for fueling magnetically confined plasmas. A frozen pellet of deuterium and tritium is injected at high velocity (∼2 km/s) into the plasma.

**Plasma:** A state of matter; an ionized gas in which electrons are not bound to the nuclei.

**Radiation Damage:** Processes leading to degradation in the properties of materials, caused by neutron irradiation. Examples includes atomic displacement, swelling, and embrittlement.

**Scrapeoff Layer:** The region of the plasma in a tokamak in which the field lines intersect a material wall (i.e., limiter or divertor plates).

**Shield:** A component located behind the blanket of a magnetic-confinement fusion reactor to protect the superconducting magnets.

**Superconductivity:** A state achieved by certain alloys and compounds in which electrical current can flow without any resistance. It usually occurs at cryogenic temperatures.

**Swelling:** A process caused by neutron irradiation that leads to an increase in the size of the material.

**TFTR:** An acronym for Tokamak Fusion Test Reactor. It is a large tokamak experiment located at the Princeton Plasma Physics Laboratory.
**Tokamak:** The most advanced magnetic-confinement concept.

**Triple Product:** The product of plasma density, \( n \), temperature, \( T \), and energy-confinement time, \( \tau_E \). The necessary condition for a sustained D-T fusion burn is that the triple product exceed \( 2 \times 10^{21} \) s keV/m\(^3\).

**Tritium Breeding Material:** Lithium or a lithium-containing compound included in the blanket of a D-T fusion reactor to produce tritium to fuel the reactor.

**Tritium Breeding Ratio:** The number of tritium atoms produced in the blanket for each D-T neutron. Since each D-T fusion reaction uses one tritium atom to produce a neutron, a TBR of 1 or greater is required for tritium self-sufficiency.

**List of Works Cited**


**Further Reading**


