Chapter 4

Fusion Science and Technology

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Great progress has been made over the past 35 years of fusion research. Nevertheless, many scientific and technological issues have yet to be resolved before fusion reactors can be designed and built. Fundamental questions in plasma science remain, especially involving the behavior of plasmas that actually produce fusion power. Other plasma science questions involve the behavior and operation of the various *confinement concepts* that might be used to hold fusion plasmas.

To date, engineering issues have not been studied as extensively as plasma science issues. For many years, engineering studies were deferred for lack of funds; science had a higher funding priority. In addition, fusion technologies that require a source of fusion power to be tested and developed have had to await a device that could supply the power. Until recently, the fusion science database has not been sufficient to permit such a device to be designed with confidence.

This chapter discusses the various confinement concepts under study, the systems required in a fusion reactor, and the issues that must be resolved before such systems can be built. It then outlines the research plan required to resolve these issues and estimates the amount of time and money that such a research plan will take.

CONFINEMENT CONCEPTS[1]

Most of the fusion program's research has focused on different magnetic confinement concepts that can be used to create, confine, and understand the behavior of plasmas. In all of these concepts, magnetic fields are used to confine the plasma; the concepts differ in the shape of the fields and the manner in which they are generated. These differences have implications for the requirements, complexity, and cost of the engineering systems that surround the plasma.

Table 4-1 lists the principal confinement schemes presently under investigation in the United States and classifies them according to their level of development. The concepts are described in the following section.

At this stage of the research program, it is not known which confinement concept can best form the basis of a fusion reactor. The tokamak is much more developed than the others, and tokamaks are expected to demonstrate the basic scientific requirements for fusion within a few years. However, several alternate concepts are under investigation in order to gain a better understanding of the confinement process and to explore possibilities for improving reactor performance.

[1] This chapter discusses only magnetic confinement approaches. Other approaches to fusion are discussed in app. B. The concepts mentioned in this section are described in greater detail in pp. 156-204 of *Physics Through the 1990s: Plasmas and Fluids*, by the National Research Council (Washington, DC: National Academy Press, 1986).

Table 4-1.—Classification of Confinement Concepts

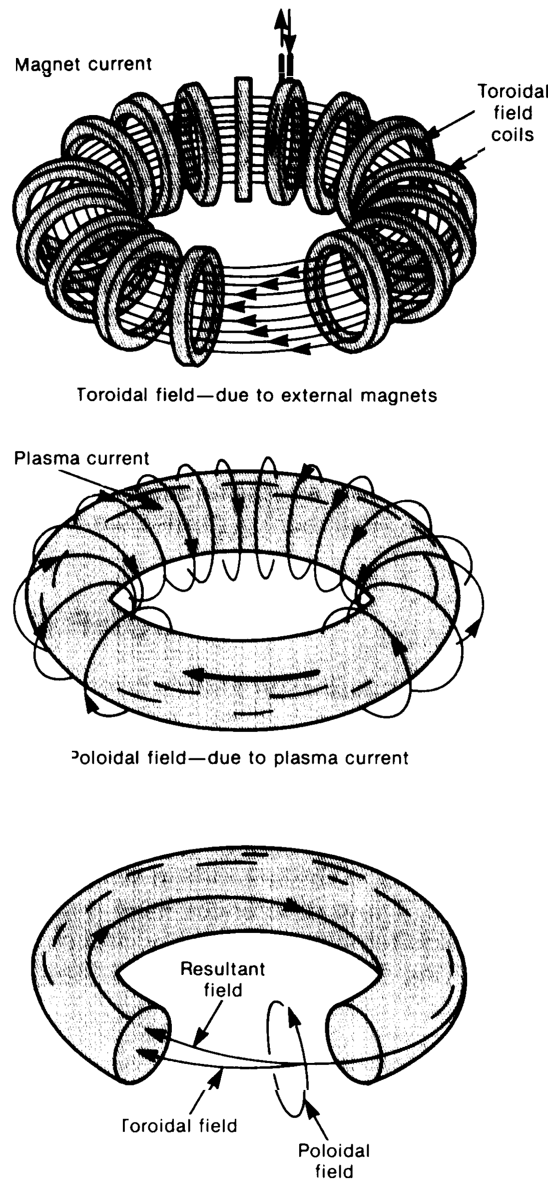
Well-developed knowledge base	Moderately developed knowledge base	Developing knowledge base
Conventional tokamak	Advanced tokamak	Spheromak
	Tandem mirror	Field-reversed configuration
	Stellarator	Dense Z-pinch
	Reversed-field pinch	

SOURCE: Adapted from Argonne National Laboratory, Fusion Power Program, *Technical Planning Activity Final Report*, commissioned by the U.S. Department of Energy, Office of Fusion Energy, ANL/FPP-87-1, January 1987, p. 15.

The major scientific questions to be answered for each confinement approach are whether and with what confidence the conditions necessary for a sustained, power-producing fusion reaction can be simultaneously satisfied in a commercial-scale reactor. Much of the experimental and theoretical work in confinement studies involves the identification and testing of scaling *relationships* that predict the performance of future devices from the results of previous experiments. ideally, such scaling models should be derivable from the basic laws of physics. However, the behavior of plasmas confined in magnetic fields is so complicated that a general theory has not yet been found. With some simplifying assumptions, limited theoretical models have been developed, but they are not broad enough to extrapolate the behavior of a concept to an unexplored range. Without a sound theoretical base, the risk of taking too large a step is great. A series of intermediate-scale experiments is needed to bridge the gap between concept development and a full-scale reactor.

Even with the tokamak—the most studied confinement concept–scaling properties are not fully understood. **Although tokamaks have attained by far the best experimental performance of any confinement concept, no proven theoretical explanation of how that performance scales with parameters such as size, magnetic field, and plasma current has yet been derived.** Without a complete theoretical basis, "empirical" scaling relationships deduced from past observations must be used. Such empirical relationships may well prove sufficient for designing a machine capable of forming reactor-scale plasmas before a fundamental theoretical understanding of tokamak behavior is reached.

Figure 4-1 .—Tokamak Magnetic Fields



SOURCE: Princeton Plasma Physics Laboratory Information Bulletin NT-1: Fusion Power, 1984, p. 4.

"Closed" Concepts

In "closed" magnetic confinement configurations, the plasma is contained by magnetic lines of force that do not lead out of the device. Closed configurations all have the basic shape of a doughnut or inner tube, which is called a *"torus."* A magnetic field can encircle a torus in two different directions (figure 4-1). A field running the long way around the torus, in the direction that the tread runs around a tire, is called a "toroidal" field. This field is generally created by external magnet coils, called toroidal field coils, through which the plasma torus passes. A magnetic field perpendicular to the toroidal field, encircling the torus the short way, is called a "poloidal" field. This field is generated by electrical currents induced to flow within the plasma itself. Together, toroidal and poloidal magnetic fields form the total magnetic field that confines the plasma.

Conventional Tokamak

In a tokamak, the principal confining magnetic field is toroidal, and it is generated by large external magnets encircling the plasma. This field alone, however, is not sufficient to confine the plasma. A secondary poloidal field, generated by plasma currents, is also required. The combination of poloidal and toroidal fields produces a total field that twists around the torus and is able to confine the plasma (figure 4-1).

The tokamak concept was developed in the Soviet Union, and, since the late 1960s, it has been the primary confinement concept in all four of the world's major fusion research programs. It has also served as the principal workhorse for developing plasma technology. The scientific progress of the tokamak is far ahead of any other concept. Major world tokamaks are listed in table 4-2.

Advanced Tokamak

Various features now under investigation may substantially improve tokamak performance. Modifying the shape of the plasma cross-section can increase the maximum plasma pressure that can be confined with a given magnetic field. The Doublet II I-D (D III-D) tokamak at GA Technologies and the Princeton Beta Experiment Modification (PBX-M) tokamak at Princeton Plasma Physics Laboratory are being used to investigate shaped plasmas according to this principle, Other variants on tokamak design would permit more compact fusion cores to be constructed, which could lead to less expensive reactors; these improvements are under study.

Still other improvements would permit tokamaks to run continuously. The technique typically used today to drive the plasma current in a tokamak can be run only in pulses. Technologies for driving continuous, or steady-state, plasma currents are being investigated at a number of different experimental facilities.

Stellarator

The stellarator is a toroidal device in which both the toroidal and poloidal confining fields are generated by external magnets and do not depend on electric currents within the plasma. The external magnets are consequently more complicated than those of a tokamak (figure 4-2). However, the absence of plasma current in a stellarator enables steady-state operation to be achieved more directly without the need for current drive.

The stellarator concept was invented in the United States. After the discovery of the tokamak in the late 1960s, however, the United States converted its stellarators into tokamaks. The stellarator concept was kept alive primarily by research

Table 4-2.—Major World Tokamaks[a]

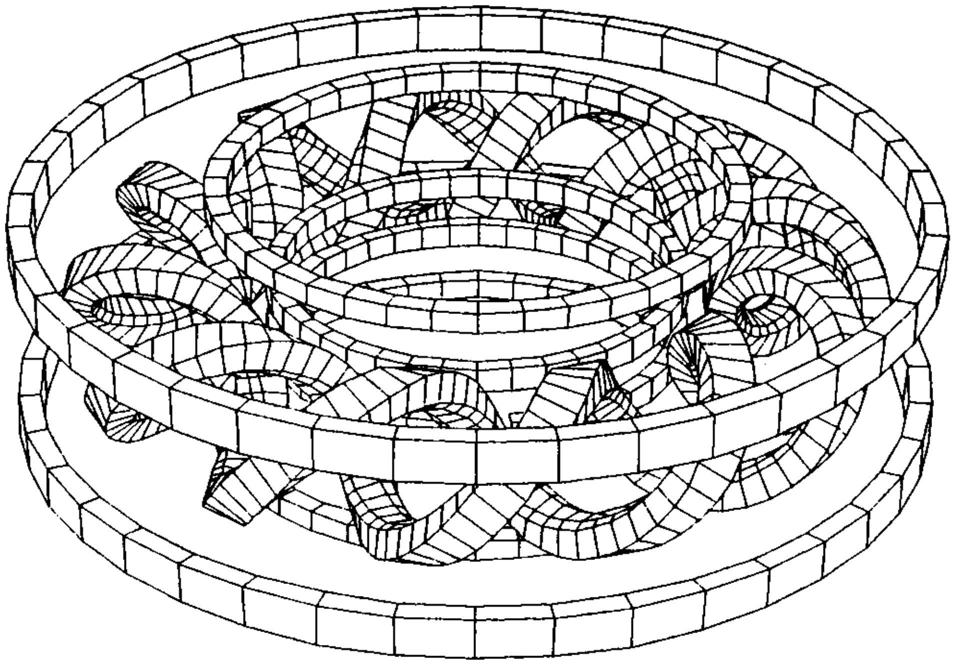
Device	Location	Status
JET	European Community (United Kingdom)	Operating
D III-D	United States (GA Technologies)	Operating
Alcator C-Mod	United States (MIT)	Under construction
T-14	U.S.S.R. (Kurchatov)	Under construction
TFTR	United States (PPPL)	Operating
JT-60	Japan (Naka-machi)	Operating
T-15	U.S.S.R. (Kurchatov)	Under construction
ASDEX-Upgrade	Federal Republic of Germany (Garching)	Under construction
Tore Supra	France (Cadarache)	Under construction
Frascati Tokamak Upgrade	Italy (Frascati)	Under construction
PBX-M	United States (PPPL)	Under construction
TEXTOR	Federal Republic of Germany (Julich)	Operating

aListed in decreaing order of plasma current, one of the many parameters that determines tokamak capability. No single factor by itself measures capability well; current is used here only to give a rough distinction between those devices at the top of the list and those at the bottom. Ranking by size, magnetic field, or other parameter would rearrange the list somewhat.

NOTE: This table includes only the largest tokamaks. The *World Survey of Activities in Corrtrolled Fusion Research, 1986 Edition* (published in *Nuclear Fusion,* Special Supplement 1966) lists a total of 77 existing and proposed tokamaks at 54 sites in 26 countries.

SOURCE Office of Technology Assessment, 1987

Figure 4-2.—Magnet Coils for the Advanced Toroidal Facility, A Stellarator



SOURCE Oak Ridge National Laboratory

in the Soviet Union, Europe, and Japan, and, due to good results, the United States has recently revived its stellarator effort. Stellarators today perform as well as comparably sized tokamaks.

Major world stellarator facilities that are operating or under construction are listed in table 4-3. Not shown on the table is the Large Helical System proposed to be built in Japan at a cost several times that of the largest stellarator machine now under construction; if built and operated, the new Japanese device would be the largest operational non-tokamak fusion experiment,

Reversed-Field Pinch

In a reversed-field pinch, the toroidal magnetic field is generated primarily by external magnets and the poloidal field primarily by plasma currents. The toroidal and poloidal fields are comparable in strength, and the toroidal field reverses direction near the outside of the plasma, giving

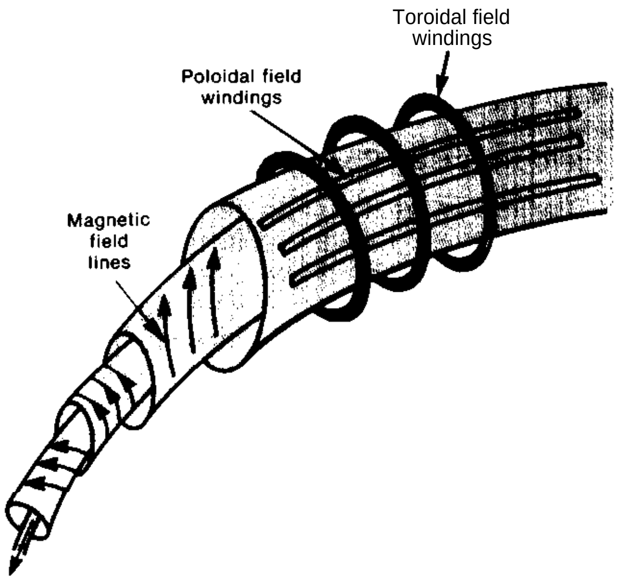
Table 4-3.—Major World Stellarators[a]

Device	Location	Status
ATF	United States (ORNL)	Under construction
Wendelstein VII-AS	Federal Republic of Germany (Garching)	Under construction
URAGAN-2M	U.S.S.R. (Kharkov)	Under construction
Heliotron-E	Japan (Kyoto University)	Operating
URAGAN-3	U.S.S.R. (Kharkov)	Operating
CHS	Japan (Nagoya University)	Under construction
L-2	U.S.S.R. (Lebedev)	Operating
H-1	Australia (Canberra)	Under construction

aListed in order of decreasing stored magnetic energy, a parameter which in turn depends both on magnetic field strength and plasma volume

SOURCE Office of Technology Assessment, 1987; from data provided by Oak Ridge National Laboratory,

Figure 4-3.— Reversed. Field Pinch



SOURCE: Adapted from National Research Council, *Physics Through the 1990s: Plasmas and Fluids* (Washington, DC: National Academy Press, 1988).

the concept its name (see figure 4-3). In a tokamak, the toroidal field dominates and points in the same direction throughout the plasma.

The reversed-field pinch generates more of its magnetic field from plasma currents and less from external magnets, permitting its external magnets to be smaller than those of a comparably performing tokamak. The nature of the magnetic fields in a reversed-field pinch may also permit steady-state plasma currents to be driven in a much simpler manner than is applicable in a tokamak. Moreover, a reversed-field pinch plasma may be able to heat itself to reactor temperatures without the complex and costly external heating systems required by tokamaks.

Los Alamos National Laboratory in New Mexico is the center of U.S. reversed-field pinch research. The Confinement Physics Research Facility (CPRF) to be built there will hold the largest reversed-field pinch device in the United States. A variant of the reversed-field pinch, the Ohmically Heated Toroidal Experiment, or OHTE, was built at **GA** Technologies in San Diego, California. Reversed-field pinch research is also conducted in both Europe and Japan. Table 4-4 lists the major world reversed-field pinches.

Spheromak

The spheromak is one of a class of less developed confinement concepts called “compact toroids,” which do not have toroidal field coils linking the plasma loop and therefore avoid the engineering problem of constructing rings locked within rings. Conceptually, if the torodial field coils and inner walls of a reversed-field pinch were removed and the central hole were shrunk to nothing, the resultant plasma would be that of the spheromak. Its overall shape is spherical; although the internal magnetic field has both toroidal and poloidal components, the device has no central hole or external field coil linking the plasma (figure 4-4). The plasma chamber lies entirely within the external magnets. If the spheromak can progress to reactor scale, its small size and simplicity may lead to considerable engineering advantages. However, the present state of knowledge of spheromak physics is rudimentary.

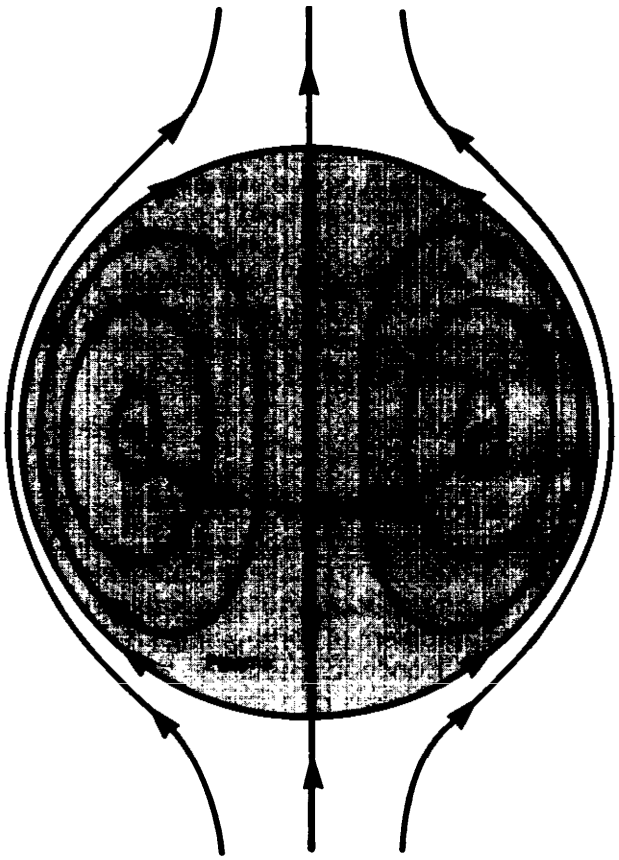
Table 4.4.—Major World Reversed-Field Pinches[a]

Device	Location	Status
CPRF	United States (LANL)	Under construction
RFX	Italy (Padua)	Under construction
OHTE	United States (GA Technologies)	Operating
HBTX 1-B	United Kingdom (Culham)	Operating
ZT-40M	United States (LANL)	Operating
MST	United States (University of Wisconsin)	Under construction
ETA BETA 11	Italy (Padua)	Operating
Repute I	Japan (Tokyo University)	Operating
TPE-1RM(15)	Japan (Tsukuba University)	Operating
STP-3M	Japan (Nagoya University)	Operating

aListed in order of decreasing plasma current, a rough measure of reversed-field pinch performance.

SOURCE: Office of Technology Assessment, 1987; from data supplied by the Los Alamos National Laboratory.

Figure 4-4.—Spheromak



B_p = Poloidal magnetic field

B_t = Toroidal magnetic field

SOURCE: M.N. Rosenbluth and M.N. Bussac, "MHD Stability of Spheromak," *Nuclear Fusion* 19(4):489-498 (Vienna, Austria: International Atomic Energy Agency, 1979).

Spheromak research at Los Alamos National Laboratory was terminated in 1987 due to fiscal constraints, and another major U.S. device at Princeton Plasma Physics Laboratory is to be terminated in fiscal year 1988. The remaining U.S. spheromak research effort takes place at the University of Maryland. Spheromaks also are being studied in Japan and the United Kingdom. Major world spheromak devices are listed in table 4-5.

Field= Reversed Configuration

The field-reversed configuration (FRC) is another form of compact toroid. Despite the similar name, it does not resemble the reversed-field pinch. It is unusual among closed magnetic confinement concepts in providing confinement with only poloidal fields; the FRC has no toroidal field. The plasma is greatly elongated in the poloidal direction and from the outside has a cylindrical shape (figure 4-5).

Like the spheromak, the FRC does not have external magnets penetrating a hole in its center; all the magnets are located outside the cylindrical plasma. The FRC also has the particular virtue of providing extremely high plasma pressure for a given amount of magnetic field strength. if its confining field is increased in strength, the FRC plasma will be compressed and heated. Such heating may be sufficient to reach reactor conditions, eliminating the need for external heating. Existing FRC plasmas are stable, but whether stability can be achieved in reactor-sized FRC plasmas is uncertain. A new facility, LSX, is under construction at Spectra Technologies in Bellevue, Washington, to investigate the stability of larger plasmas.

U.S. FRC research started at the Naval Research Laboratory in Washington, D. C., in the late 1960s. Increased effort in the United States in the late 1970s, centered at Los Alamos, was undertaken largely in response to experimental results obtained earlier in the decade from the Soviet Union

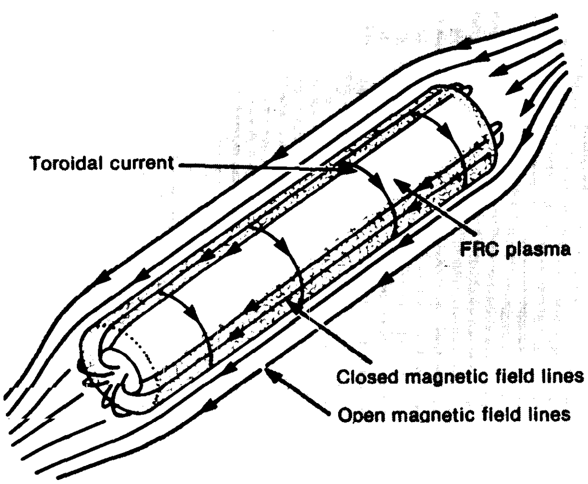
Table 4-5.—Major World Spheromaks[a]

Device	Location	Status
S-1	United States (PPPL)	To be terminated, fiscal year 1988
CTX	United States (LANL)	Terminated, fiscal year 1987
MS	United States (University of Maryland)	Under construction
CTCC	Japan	Operating
Manchester U.	United Kingdom (University of Manchester)	Operating
TS-3	Japan	Operating

[a]Listed approximately by decreasing order of the size of the spheromak research effort at each site; it is difficult to specify any single physical parameter as a rough measure of spheromak capability.

SOURCE: Office of Technology Assessment, 1987.

Figure 4-5.— Field-Reversed Configuration



SOURCE: National Research Council, *Physics Through the 1990s:* Plasmas *and Fluids* (Washington, DC National Academy Press, 1988),

and the Federal Republic of Germany. Soviet research has continued, but German and British research programs have stopped. Meanwhile, a program in Japan has begun. Major field-reversed configuration experiments around the world are listed in table 4-6.

"Open” Concepts

plasmas in open magnetic confinement devices are confined by magnetic fields that do not close back on themselves within the device but rather extend well outside the device. Since plasma particles can easily travel along magnetic field lines, some additional mechanism is required to reduce the rate at which plasma escapes out the ends of an open confinement device.

Magnetic Mirrors

Fusion plasmas can be confined in an open-ended tube by strengthening, and thereby compressing, the magnetic fields near the ends. Strengthening the magnetic field near the ends “reflects” plasma particles back into the center much as narrowing the ends of a sausage helps keep in the meat. However, the ends of a simp/e *magnetic mirror* (figure 4-6a) are not otherwise sealed. Just as the meat eventually forces its way out of an unsealed sausage when squeezed, a simple magnetic mirror cannot confine a plasma well enough to generate fusion power. In addition, simple mirrors are usually unstable, with the plasma as a whole tending to slip out sideways.

A variation of the simple mirror is the *minimum-B mirror* (figure 4-6 b), one version of which uses a coil shaped like the seam on a baseball to create a magnetic field that is lowest in strength at the center and increases in strength towards the outside. Particles leaving the center tend to be reflected back by the increasing magnetic field at the outside, just as particles leaving the simple mirror tend to be reflected back at the ends. This configuration is stable, and there is no tendency for the plasma as a whole to escape. However, despite these improvements, the minimum-B mirror cannot confine a plasma well enough to generate net fusion power.

The *tandem mirror* (figure 4-6c) improves the simple magnetic mirror by utilizing additional mirrors to improve the plugging at each end. These plugs, called end cells, are themselves magnetic mirrors. Rather than trapping the main plasma, the end cells hold particles that generate an electric field. This electric field, in turn, keeps the plasma in the central cell from escap-

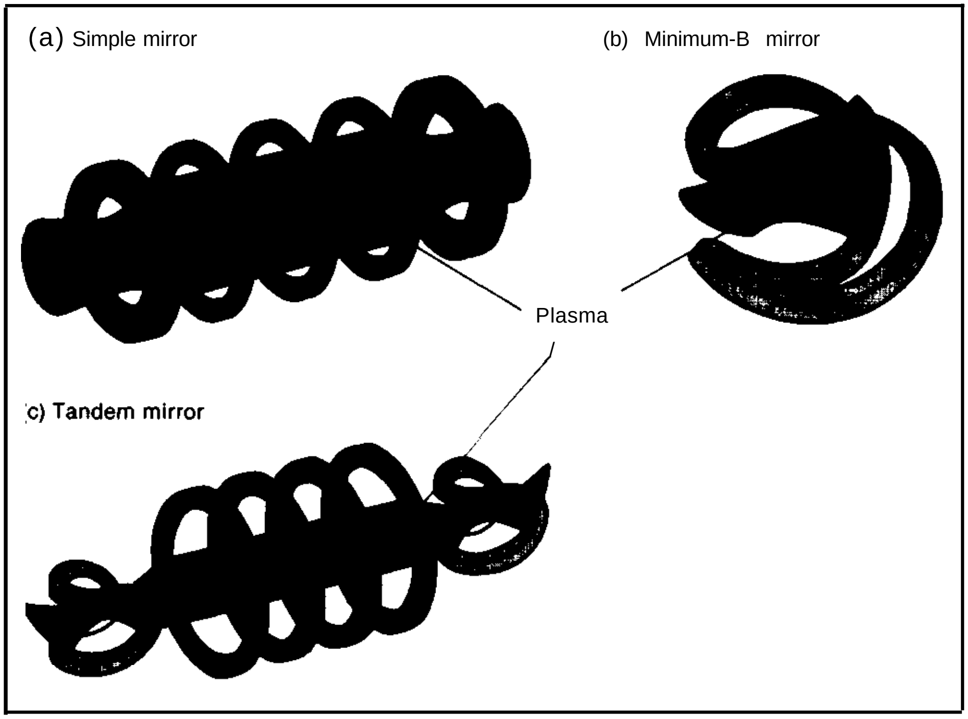
able 4.6.—Major World Field-Reversed Configurations[a]

Device	Location	Status
LSX	United States (Spectra Technologies)	Under construction
FRX-C	United States (LANL)	Operating
BN, TOR	U.S.S.R. (Kurchatov)	Operating
TRX-2	United States (Spectra Technologies)	Operating
OCT, PIACE	Japan (Osaka University)	Operating
NUCTE	Japan (Nihon University)	Operating

aListed approximately by decreasing order of size; similarly sized devices at the same institution are listed together.

SOURCE Office of Technology Assessment, 1987; from information supplied by the Los Alamos National Laboratory.

Figure 4-6.—Magnetic Mirrors



SOURCE: Lawrence Livermore National Laboratory, "Evolution of the Tandem Mirror," *Energy and Technology Review,* November 1986

ing. While particle losses from the end cells are high, these losses can be compensated by injecting new particles.

The tandem mirror concept was developed simultaneously in the United States and the Soviet Union in the late 1970s. The Mirror Fusion Test Facility B (MFTF-B), located at Lawrence Livermore National Laboratory in California, is the largest mirror device in the world and the largest non-tokamak magnetic confinement fusion experiment. Budget cuts, however, forced MFTF-B to be moth balled before it could be used experimentally. The Tandem Mirror Experiment Upgrade (TMX-U) at Livermore, a smaller version of MFTF-B, was terminated as well, and the TARA device at the Massachusetts Institute of Technology will be shut down in 1988. At that point, Phaedrus at the University of Wisconsin will be the only operational U.S. mirror machine. Mirror research is still conducted in the Soviet Union and Japan. Table 4-7 presents a list of major world tandem mirror facilities.

Dense Z= Pinch

In this concept, a fiber of frozen deuterium-tritium fuel is suddenly vaporized and turned into plasma by passing a strong electric current through it. This current heats the plasma while simultaneously generating a strong magnetic field encircling the plasma column (figure 4-7), "pinching" it long enough for fusion reactions to occur. Many devices investigated in the earliest days of fusion research in the 1950s operated in a similar manner, but they were abandoned because their plasmas had severe instabilities and were unable to approach the confinement times needed to generate fusion power.

The dense z-pinch differs from the 1950s pinches in several important aspects that, as calculations and experiments have shown, improve stability. Crucial to the modern experiments are precisely controlled, highly capable power supplies that would have been impossible to build with 1950s technology, and the use of solid, rather than gase-

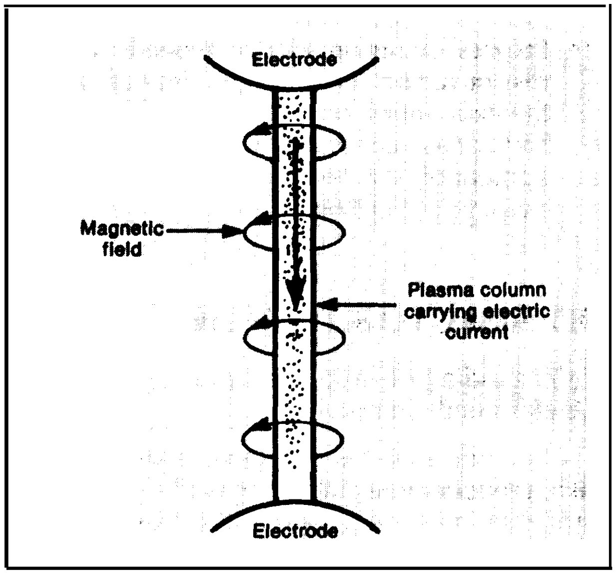
Table 4-7.—Major World Tandem Mirrors[a]

Device	Location	Status
MFTF-B	United States (LLNL)	Moth balled
TMX-U	United States (LLNL)	Mothballed
Gamma-10	Japan (Tsukuba University)	Operating
TARA	United States (MIT)	To be terminated, fiscal year 1988
Phaedrus	United States (University of Wisconsin)	Operating
Ambal M	USSR (Novosibirsk)	Under construction

aListed in decreasing order of size.

SOURCE: Office of Technology Assessment, 1987; from data supplied by the Lawrence Livermore National Laboratory.

Figure 4-7. —Dense Z-Pinch



SOURCE: Office of Technology Assessment, 1987

ous, fuel to initiate the discharge. However, it is much too early to tell whether this concept can be developed successfully. If the concept can be developed, the device has the potential to be far smaller and far less expensive than devices based on other concepts. External magnets are not needed since the plasma current supplies the entire confining field. Dense z-pinch research is taking place in the United States at two facilities: the Naval Research Laboratory and Los Alamos National Laboratory.

Conclusions Concerning Confinement Approaches

A number of general conclusions can be drawn from studies of the confinement concepts that have evolved over the past 10 to 15 years:

- **Many fusion concepts are under study because the frontrunner tokamak, while likely to be scientifically feasible, may yet be found weak in some critical area or less economically attractive than alternatives.** Features being studied in alternate concepts include eased conditions for steady-state operation, reduced external magnet complexity and cost, and improved use of the magnetic field. Searching for optimum reactor configurations and developing further understanding of the fusion process mandate that the range of concepts under investigation not be prematurely narrowed.

  The tokamak concept is by far the most developed, and it has attained plasma conditions closest to those needed in a fusion reactor. At present and for the next several years, studies of reactor-like plasmas will be done with tokamaks because no other concept has yet proven that it can reach reactor conditions. A number of other confinement concepts have features that might make them preferable to the tokamak if they are capable of progressing to an equivalent stage of performance. It remains to be seen which of these concepts will attain that performance level, what their development will cost, and to what degree the tokamak concept itself will further improve.
- **Different confinement studies complement each other. Knowledge obtained through research on a specific concept often can be generalized. Throughout the history of fusion research, plasma** science issues originally investigated because of their relevance to a particular concept have become important to studies of other concepts as well.
- **A great deal of progress in understanding fusion plasmas and confinement concepts has been made to date.** Many concepts

studied earlier, such as the simple magnetic mirror, are no longer studied today because they cannot compare attractively to improved or alternate concepts. At the same time, as in the case of the dense z-pinch, problems once considered intractable may be solved with additional scientific understanding and more advanced technology.

- **Research on all confinement concepts has benefited from international cooperation. Studies undertaken by different groups in different countries enhance each other significantly. Furthermore, advances by one program have frequently stimulated additional progress in other programs. international cooperation** in fusion research is discussed further in chapter 7.
- **Not all confinement concepts can be developed to reactor scale.** Promising concepts require study at greater levels of capability before their potential as reactor candidates can be assessed. Moreover, since this has largely been an empirical program, advanced studies will require larger and increasingly more expensive facilities. Fiscal constraints will almost certainly require that not all of the concepts be "promoted" to subsequent stages of development. Criteria such as development cost, characteristics of the end product, and likelihood of success must be developed for selecting which concepts are to be pursued further.
- **progress in fusion science depends on progress in fusion technology.** Time after time, the exploration of new ranges of plasma behavior has been made possible by the development of new heating, fueling, and plasma shaping technologies.

SCIENTIFIC PROGRESS AND REACTOR DESIGN

Different Dimensions of Progress

To form the basis of a viable fusion reactor, a confinement concept must meet two objectives. First, it must satisfy scientific performance requirements—temperature, density, and confinement time—necessary for a plasma to produce fusion power. progress towards those requirements is easy to measure.

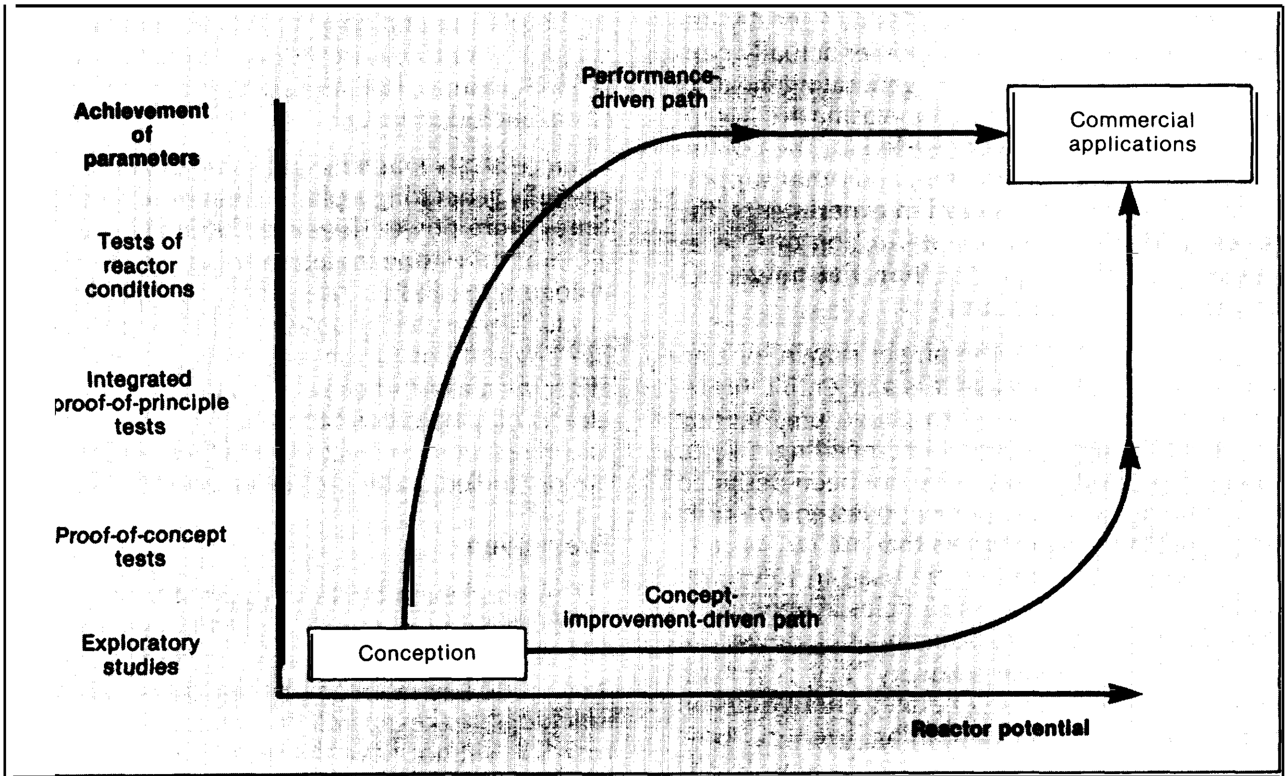
Second, a confinement concept must demonstrate "reactor potential. " Unlike scientific performance, reactor potential is difficult to measure. A viable reactor must be built, operated, and maintained reliably, and it must be economically, environmentally, and socially acceptable. While fusion's acceptability in these respects depends on factors external to fusion technology, it also depends on the choice of confinement concept.

Each concept may have different advantages, and, in the absence of quantitative measures, the process of i dentifying the concepts t hat offer t he most attractive reactors depends in large part on the innovation and technological optimism of the reactor designer. Also, attributes of an attractive reactor can be identified today, but the relative importance of these attributes may change as our understanding of fusion technology and future societal needs improves.

No matter how it is evaluated, reactor potential is a requirement that, along with scientific performance, must be satisfied by at least one confinement concept before fusion power can be realized. Figure 4-8 shows two different paths by which a concept can develop toward commercial use. Along the "performance-driven" path, a concept first demonstrates the ability to attain plasma parameters near those required to produce fusion power; subsequently, innovations or successive refinements show that the concept's scientific capabilities can be used in a viable reactor design. Alternatively, along the "concept-improvement-driven" path, features that are attractive in a fusion reactor—e.g., compact size, ease of maintenance, simple construction, and reliable operation—are apparent before the scientific performance necessary to produce fusion power is demonstrated.

The actual development of any given confinement concept will fall somewhere between these extremes. Development of the tokamak appears to be closer to the performance-driven curve. Its scientific performance, along with its use in de-

Figure 4-8.—Alternate Paths for Concept Development



SOURCE: Adapted from Argonne National Laboratory, *Technical Planning Activity: Final Report*, commissioned by the U.S. Department of Energy, Office of Energy Research, AN L/FPP-87-1, January 1987, figure 1.5, p. 56.

veloping plasma technology and diagnostics, has been the primary motivation for study to date; improvements to the basic tokamak concept are currently focused on improving reactor potential. Other concepts are more concept-improvement-driven in that their features might make them preferable to the tokamak if they can reach reactor scale. However, the ability of other concepts to attain the necessary plasma conditions is much less certain because their experimental databases are less developed.

Scientific Progress

Energy Gain

An important measure of scientific progress towards attaining reactor-relevant conditions is energy *gain,* denoted as "Q." Energy gain is the ratio of the fusion power output that a device generates to the input power injected into the plasma. Input and output power are measured at some instant after the plasma has reached its operating density and temperature. In experimental plasmas that do not contain tritium and therefore do not produce significant amounts of fusion power, an "equivalent Q" is measured. It is defined as the Q that would be produced by the plasma if it were fueled equally by both deuterium and tritium (D-T) and if it had attained the same plasma parameters.[2]

The numerator of the Q ratio includes all the fusion power produced by the plasma, even though most of the output power (80 percent)

[2]"Equivalent Q" is either calculated from the measured plasma density and temperature or derived from actual measurements of deuterium-deuterium (D-D) fusion reactions. Since the ratio between the D-T reaction rate and the D-D reaction rate under the same conditions is believed known, a measurement of D-D reactions can be used to infer what the D-T fusion yield would be under the same conditions.

immediately escapes from the plasma via energetic neutrons. The denominator of the ratio–the power used to heat the plasma—greatly underestimates the amount of power actually consumed. Losses incurred in generating heating power and delivering it to the plasma are not included, nor is the power needed for the confining magnets, vacuum system, and other support systems. **In present-generation experiments, the power excluded from the definition of Q is as much as** 35 **times greater than the power accounted for in this ratio.**[3]

Q excludes most of the power drawn by a fusion experiment because it is a scientific measure that is not intended to gauge engineering progress. Present experiments, needing only to operate for short pulses, have not been designed to minimize consumed power; to lessen construction cost, they use magnets that are far less efficient than those likely to be used in future reactors. Similarly, the inefficiencies in generating the externally applied plasma heating power are not included because auxiliary heat is not required once a plasma generates enough fusion power to become self-sustaining. Even so, some external power will be required in any steady-state plasma device except the stellarator to maintain electrical currents within the plasma.

Figure 4-9 shows the plasma temperatures and confinement parameters needed to obtain Qs of at least 1, a condition known as "breakeven." The plasma temperatures and confinement parameters that have been attained experimentally by various confinement configurations are also shown. No device has yet reached breakeven, although tokamak experiments have clearly come the closest.

Ignition

The most significant region in figure 4-9 is *ignition* in the top right corner. An ignited D-T plasma not only generates net fusion power but also retains enough heat to continue producing fusion reactions without external heat. The Q of an ignited plasma is infinite, since the plasma generates output power without auxiliary input power from external sources. (Power to drive currents in the plasma and to cool the magnets to their operating temperature will be required even for ignited plasmas, but, as stated above, this power is not included in Q.)

Successfully reaching ignition—or at least successfully generating a plasma that produces many times more power than is input into it—will be a major milestone in determining fusion's technological feasibility. The energy and the reaction products generated in a plasma producing appreciable amounts of fusion power will significantly affect the plasma's behavior. Understanding these effects may be crucial to utilizing self-sustaining fusion reactions in reactors, and these effects cannot be studied under breakeven conditions alone.

Breakeven

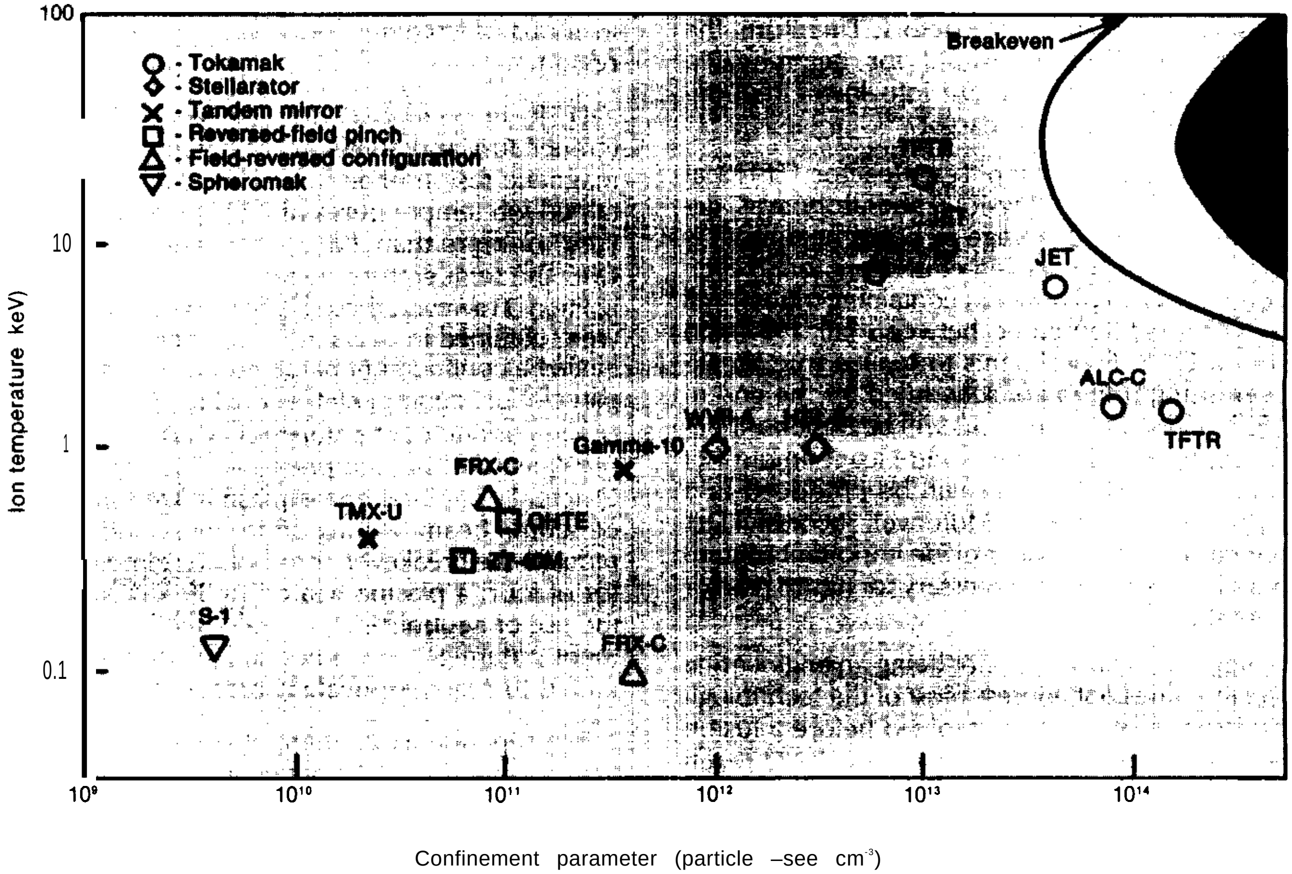
The breakeven curve in figure 4-9 shows the conditions under which a plasma generates as much power through fusion reactions as is injected into it to maintain the reactions. Although reaching breakeven will be a major accomplishment, it will not have the technical significance of reaching ignition. Due to the way that energy gain is defined, the breakeven threshold in some respects is arbitrary, and it depends significantly on the manner in which the plasma is heated. Crossing the threshold does not cause a significant change in plasma behavior and in no way indicates that the experiment is able to power itself. Problems that are not fully evident under breakeven conditions may yet be encountered on the way to ignition.

The breakeven curve in figure 4-9 is calculated for plasmas that are uniformly heated. If the plasma is heated in such a way that a small fraction of the plasma particles become much hotter than the rest, this fraction will produce a disproportionate amount of fusion power and the breakeven requirements can be substantially lowered. For this reason, plasmas heated with neutral beams can reach breakeven under conditions that would not be sufficient without the use of neutral beams.

Neutral beams are extremely hot jets of neutral atoms that can penetrate the confining mag-

[3]See specifications for the Tokamak Fusion Test Reactor at the Princeton Plasma Physics Laboratory, footnote 4 below.

Figure 4-9.— Plasma Parameters Achieved by Various Confinement Concepts



KEY: S-1: Spheromak-1; Princeton Plasma Physics Laboratory, Princeton, NJ.
TMX-U: Tandem Mirror Experiment Upgrade; Lawrence Livermore National Laboratory, Livermore, CA.
ZT-40M: Toroidal Z-pinch, -40, Modified; Los Alamos National Laboratory, Los Alamos, NM.
FRX-C: Field-Reversed Experiment C; Los Alamos National Laboratory, Los Alamos, NM.
OHTE: Ohmically Heated Toroidal Experiment; GA Technologies, Inc., San Diego, CA.
Gamma-IO: University of Tsukuba, Ibaraki, Japan.
WVII-A: Wendelstein VII-A; Institute for Plasma Physics, Garching, Federal Republic of Germany.
HEL-E: Heliotron-E; Kyoto University, Kyoto, Japan.
D III: Doublet III; GA Technologies, Inc., San Diego, CA.
JET: Joint European Torus; JET Joint Undertaking, Abingdon, United Kingdom.
TFTR: Tokamak Fusion Test Reactor; Princeton Plasma Physics Laboratory, Princeton, NJ.
ALC-C: Alcator C; Massachusetts Institute of Technology, Cambridge, MA.

SOURCE Office of Technology Assessment, 1987.

netic fields to enter the plasma. Beam atoms collide with particles inside the plasma and become electrically charged, thereby becoming trapped by the magnetic field. Through collisions, much of the energy carried by the beams is transferred to the “target” plasma, heating it up. In the process, the beam particles themselves cool down.

However, it will take many collisions for the beam particles to cool down to the temperature of the target plasma. As long as the beams are on, the most recently injected beam particles are significantly hotter than the original plasma particles. (Once the beams are turned off, the injected particles cool down to the temperature of the remaining plasma.) Since the fusion reaction rate increases very rapidly with temperature, the hotter particles from the neutral beam have a much higher probability of generating fusion reactions than other particles in the plasma. in this manner, a beam-heated plasma can achieve breakeven with plasma parameters up to a factor of 10 lower than those needed for plasmas heated by other mechanisms. However, since the beams themselves require so much power to operate, it is not expected that beam-heated plasmas will be used in reactors. Therefore, the lower breakeven threshold for beam-heated plasmas may not translate into lower requirements for a practical reactor.

The Tokamak Fusion Test Reactor (TFTR) at Princeton Plasma Physics Laboratory was designed to take advantage of beam heating. It is expected that breakeven-equivalent (breakeven conditions in a plasma not containing tritium) will be obtained sometime between fall 1987 and spring 1988. Experiments to realize true breakeven using tritium are scheduled for the end of 1990. These achievements will be important because, for the first time, a significant amount of heat from fusion power will be produced in a magnetic fusion device. Moreover, successful D-T operation of TFTR will provide important tritium-handling experience necessary for future reactor operation.

Nevertheless, TFTR–not being an engineering facility–does not address most of the technological issues that must be resolved before a fusion reactor can be built. Moreover, it will not reach ignition, and the advantage it derives from using neutral beams will probably not translate into a workable reactor. It does not incorporate advanced physics aspects that have been identified since its design in the 1970s. TFTR will not—and never was intended to—have the capability to generate electricity from the fusion power it will produce. **Even on attaining breakeven, the TFTR experiment as a whole—as opposed to the TFTR plasma alone—will produce less than 3 percent of the power it will consume.**[4]

State of the Art

Temperature and Confinement.–Figure 4-9 shows results that have been attained by each of the confinement concepts to date. Tokamak experiments have clearly made the most progress in terms of coming the closest to the ignition region.

TFTR, in particular, has reached the highest temperature and confinement parameters of any magnetic fusion experiment. In 1986, TFTR attained ion temperatures of 20 kiloelectron volts (kev) or more than 200 million degrees C, well over the temperature needed for breakeven or ignition. However, these high-temperature results were obtained in a relatively low-density plasma having a confinement parameter of 10^{13} second-particles per cubic centimeter, which is about half of the confinement parameter needed to reach breakeven at that temperature. The equivalent Q actually attained by the plasma was 0.23. Use of neutral beam heating under these conditions reduces the breakeven threshold by almost a factor of four; a plasma heated to 20 keV without the use of neutral beams would need a confinement parameter 7.5 times higher than was attained to reach equivalent breakeven.

In a separate experiment at a lower temperature of 1.5 keV, TFTR reached a confinement parameter of 1.5×10^{14} second-particles per cubic centimeter. Had this confinement been attained at a temperature of **20** keV, TFTR would have been well above equivalent breakeven, coming close to meeting the equivalent ignition condition. However, in practice, TFTR will not be able to attain temperature and confinement values this high simultaneously. Temperature can be raised at the expense of confinement, and vice versa, but the product of the two–which determines equivalent Q—is difficult to increase. With additional neutral beam power and other improvements, TFTR may well be able to raise its equivalent Q from 0.23 to 1 and reach equivalent breakeven. However, it is extremely unlikely that equivalent Qs much greater than 1 are attainable in TFTR.

Beta.–The beta parameter, also called the **“magnetic field utilization factor, ” measures the efficiency** with which the energy of the magnetic field is used to confine the energy of the plasma. Beta is defined as the ratio of the plasma pres-

[4]TFTR is being upgraded to deliver up to 27 megawatts of neutral beam power to the plasma. To reach breakeven, where the fusion power generated equals the external power injected into the plasma, 27 megawatts of fusion power would have to be generated in the plasma. If reaching breakeven were to require TFTR to draw near the maximum amount of power available from its electrical supply, it could consume close to 1,000 megawatts of electricity. This amount is 37 times greater than the fusion power to be produced at breakeven.

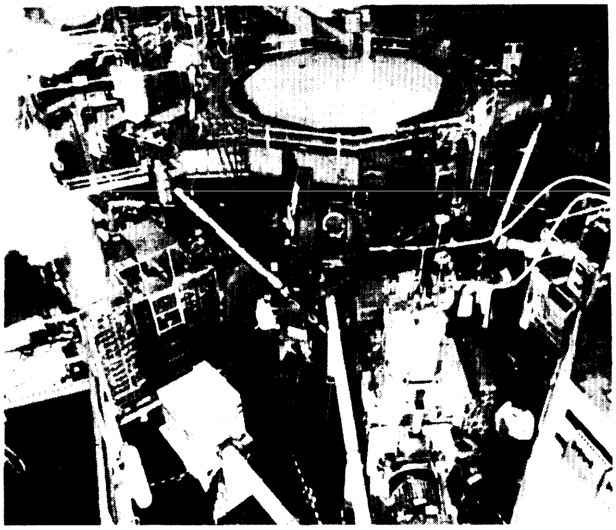


Photo credit: Princeton Plasma Physics Laboratory

The PBX tokamak at Princeton Plasma Physics Laboratory.

sure to the magnetic field pressures Record tokamak values for beta of 5 percent, in the PBX experiment at Princeton Plasma Physics Laboratory, and 6 percent, in the D II I-D experiment at GA Technologies, have been attained. These results are especially important in that they generally validate theoretical models that predict how further improvements in beta can be obtained.

In a fusion reactor, the fusion power output per unit volume of the plasma would be proportional to beta squared times the magnetic field strength to the fourth power. Since tokamaks have relatively low betas compared to many of the other confinement concepts currently studied, improving the beta of tokamaks can be useful. Betas greater than 8 percent are indicated in some system studies as being necessary for economical performance,[6] and values considerably exceeding that have been obtained by certain other confinement concepts (albeit to date at much poorer temperatures and confinement parameters). However, physical phenomena in the plasma prevent beta values from being increased indefinitely. These phenomena, which differ from concept to concept, are not completely understood; gaining additional understanding in this area is a high priority.

Low beta values can also be compensated by raising the magnetic field strength. Whereas raising beta primarily involves plasma physics issues, the issues involved in raising the magnetic field strength are primarily engineering-related: stronger magnetic fields are more difficult and expensive to generate and place greater stress on the magnet structures. At some field strength, the advantages of stronger magnetic fields will be outweighed by the additional expense of the magnets.

Scaling.–Understanding how tokamak performance can be expected to improve is crucial to evaluating the tokamak's potential for future reactors as well as to designing next-generation tokamak experiments. As mentioned earlier, the complete theoretical mechanism determining tokamak scaling has yet to be understood. Observationally, plasma confinement has been found to improve with increased plasma size. Empirical data also show that tokamak confinement improves when plasma density is increased, but that this behavior holds only for ohmically heated plasmas. Non-ohmically heated plasmas follow what has come to be known as "L (Low) -mode" scaling, in which confinement degrades as increasing amounts of external power are injected.

A few years ago, experiments on the German Axisymmetric Divertor Experiment (ASDEX) discovered a mode of tokamak behavior described by a more favorable scaling, labeled "H (High)-mode. " In this mode, performance even with auxiliary heating behaved more like the original, ohmically heated plasmas. However, H-mode scaling could be achieved only with a particular combination of device hardware and operating conditions. Subsequently, additional work at other tokamaks has broadened the range of conditions under which this more favorable behavior can be found. The challenge to tokamak researchers is to obtain H-mode scaling in configu rations and operating regimes that are also con-

[5] Plasma pressure is equal to plasma temperature times density and is proportional to the plasma energy per unit volume; magnetic field pressure, which is proportional to the square of the magnetic field strength, is a measure of the energy stored in the magnetic field per unit volume.

Typical reactor studies indicate that the plasma in an operating fusion reactor will have a pressure several times that of the earth's atmosphere at sea level. The plasma density, however, will only be about 1/1 00,000 the density of the atmosphere at sea level. That so tew par-t cles can exert such a h igh pressure IS a measure of their extreme tern perature: about 10,000 electron volts, or more than 100 million degrees C.

[6] For exam Pie, J. Sheffield, et al., *Cost Assessment of a Generic Magnet/c Fusion Reactor,* Oak Ridge National Laboratory, (3RNLKM-931 1, March 1986, p. 5.

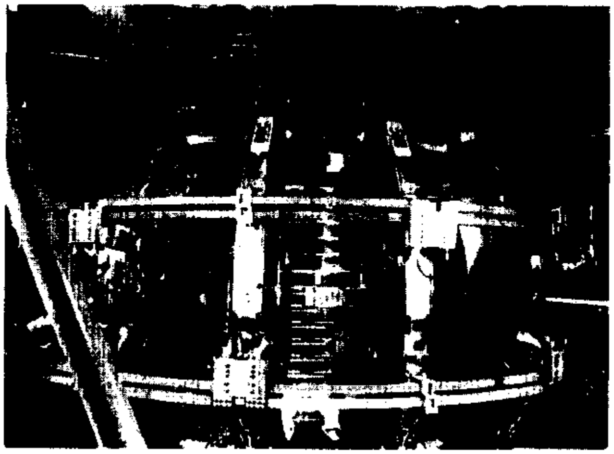


Photo credit: Commission of the European Communities

The ASDEX tokamak at the Max-Planck Institute for Plasma Physics, Garching, Federal Republic of Germany.

ducive to attaining reactor-like temperatures and densities.

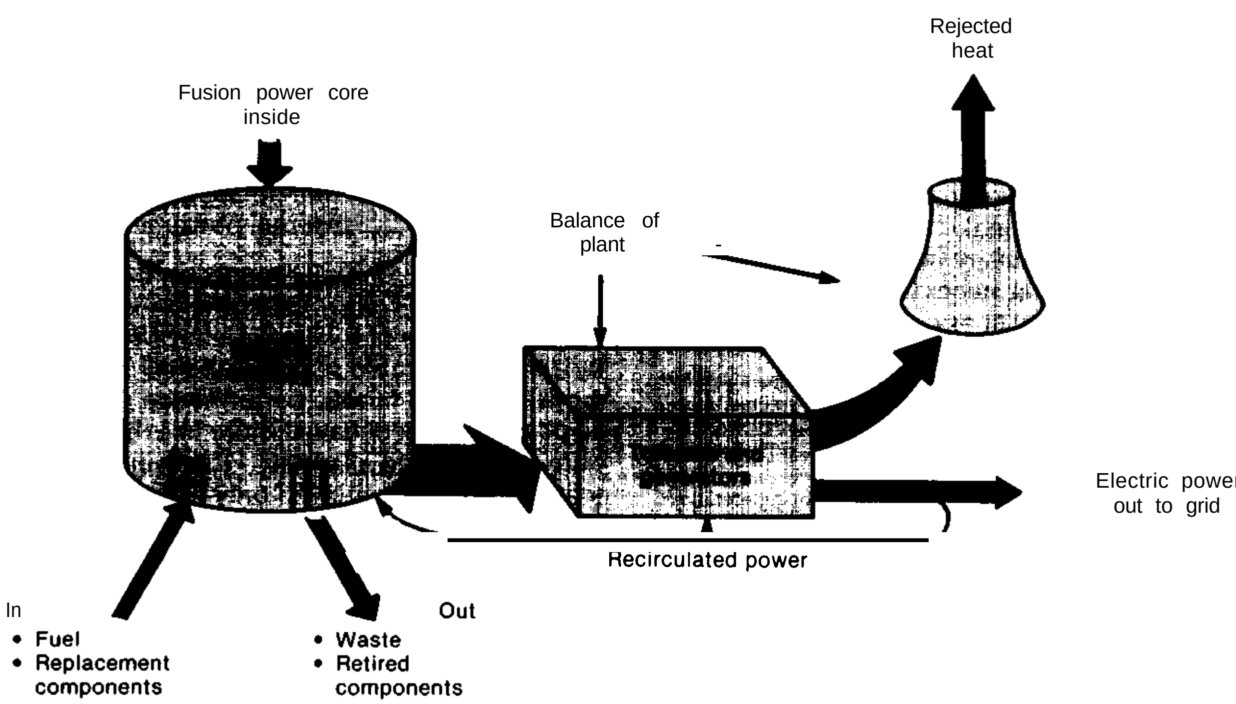
Reactor Design

Just as an automobile is much more than spark plugs and cylinders, a fusion reactor will contain many systems besides those that heat and confine the plasma. Fusion's overall engineering feasibility will depend on supporting the fusion reaction, converting the power released into a more usable form of energy, and ensuring operation in a safe and environmentally acceptable manner. **Developing and building these associated systems and integrating them into a functional whole will require a technological development effort at least as impressive as the scientific challenge of creating and understanding fusion plasmas.**

The following section describes the systems in a fusion reactor. Since the tokamak confinement concept and the D-T reaction are the most extensively studied, a tokamak-based reactor fueled with D-T is used as an example. However, most of the systems described here would be found, in some form, in reactors based on other concepts as well.

The overall fusion generating station (figure 4-10) consists of a *fusion power core,* containing the systems that support and recover energy from the fusion reaction, and the *balance of plant* that converts this energy to electricity using equip-

Figure 4-10.—Systems in a Fusion Electric Generating Station



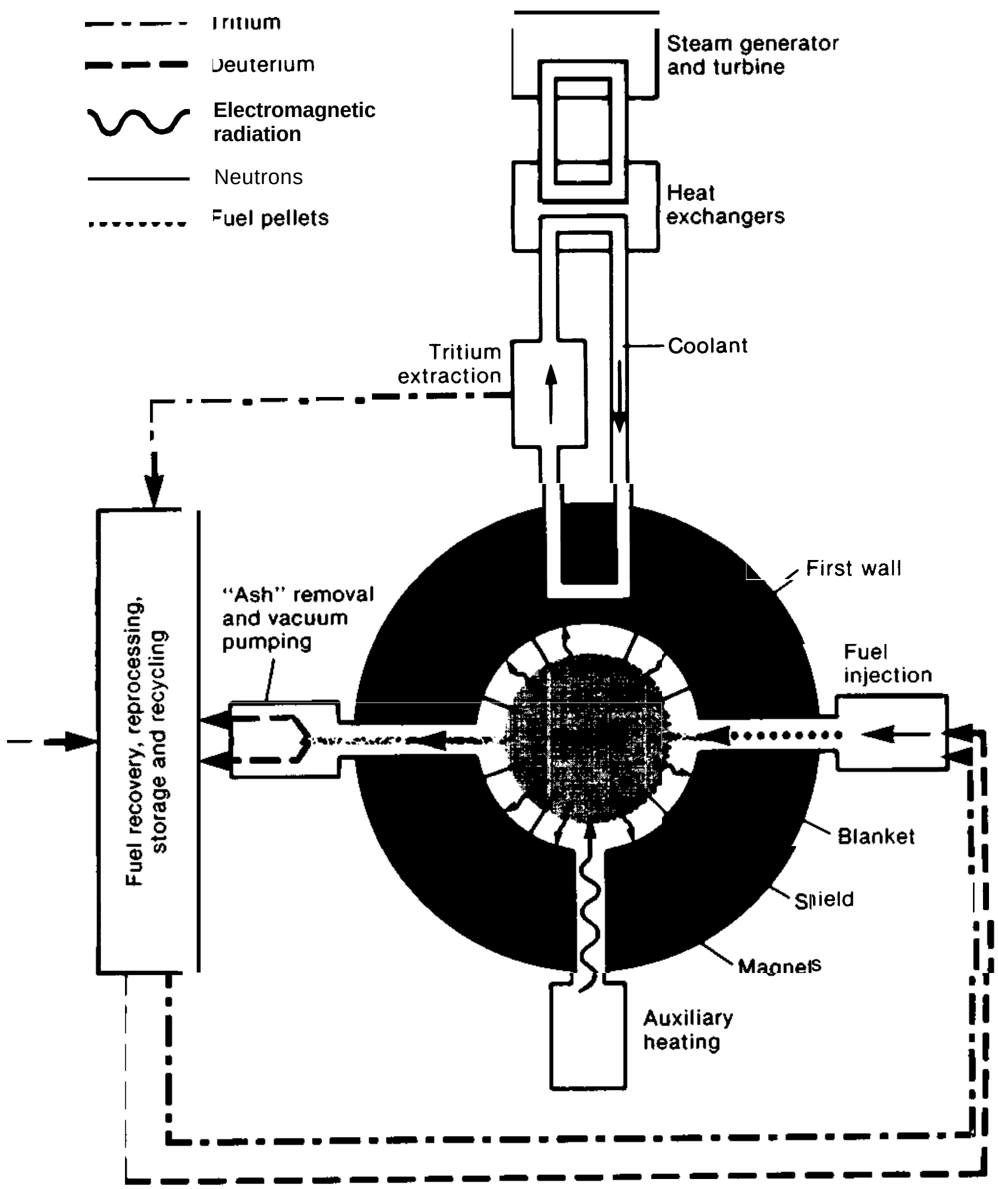
SOURCE: Office of Technology Assessment, 1987.

ment similar to that found in present electricity generating stations. Features that might convert fusion power to electricity more directly in advanced fusion reactors are described in a subsequent section.

Fusion Power Core

The fusion power core, shown schematically in figure 4-11, is the heart of a fusion generating station. It consists of the plasma chamber, the surrounding blanket and first wall systems that recover the fusion energy and breed tritium fuel, the magnet coils generating the necessary magnetic fields, shields for the magnets, and the fueling, heating, and impurity control systems. Before an acceptable design for a fusion power core can be developed, the behavior of fusion plasmas must be understood under all conditions that might be encountered. Furthermore, significant

Figure 4-11.—Systems in the Fusion Power Core



SOURCE: Modified from "The Engineering of Magnetic Fusion Reactors," by Robert W. Conn. Copyright 1983 by Scientific American, Inc. All rights reserved.

advances must be made in *plasrna technologies,* which confine and maintain the plasma, and *nuc/ear technologies,* which recover heat from the plasma, breed fuel, and ensure safe operation.

Balance of Plant

Balance of plant generally describes the systems of a fusion generating station outside of the fusion power core. In the example shown in figure 4-11, the balance-of-plant resembles systems found in other types of electric generating stations. These systems use heat provided by the fusion core to produce steam that drives turbines and generates electricity. The steam is cooled by passing through the turbines, and the remaining heat in the steam is exhausted through cooling towers or similar mechanisms.

More advanced systems that convert plasma energy directly into electricity also may be possible. Fusion reactors incorporating such systems could be made more efficient than those using steam generators and turbines.

FUSION POWER CORE SYSTEMS

The Fusion Plasma

At the center of a fusion reactor, literally and figuratively, is the fusion plasma. A number of supporting technology systems create and maintain the plasma conditions required for fusion reactions to occur. These technologies confine the plasma, heat and fuel it, remove wastes and impurities, and, in some cases, drive electric currents within the plasma. They also recover heat, breed fuel, and provide shielding.

Further development of many of these plasma technologies is required before they will be capable of producing a reactor-scale plasma. Furthermore, each of these supporting systems affects plasma behavior, and the interactions are incompletely understood. progress in both plasma technology and plasma science is therefore needed before reactor-scale fusion p lasmas can be created.

Heating

Description.–Some heat loss from a plasma is inevitable (see box 4-A), but, with good confinement, the losses can be made up by external heating and/or by fusion self-heating. Different mechanisms for heating the plasma, illustrated in figure 4-12, are listed below.

Ohmic Heating. -Like an electric heater, a plasma will heat up when an electrical current is passed through it. However, the hotter a plasma gets, the better it conducts electricity and therefore the harder it is to heat further. As a result, ohmic heating is not sufficient to reach ignition in many configurations.

Neutral Beam Heating. –Energetic charged or neutral particles can be used to heat fusion plasmas. However, the same magnetic fields that prevent the plasma from escaping also prevent charged particles on the outside from easily getting in. Therefore, beams of energetic neutral (uncharged) particles that can cross the field lines are usually preferred for heating the plasma.

Radiofrequency Heating. -Electromagnetic radiation at specific frequencies can heat a plasma like a microwave oven heats food. Radiofrequency or microwave power beamed into a plasma at the proper frequency is absorbed by particles in the plasma. These particles transfer energy to the rest of the plasma through collisions.

Compression Heating. –Increasing the confining magnetic fields can heat a plasma by compressing it. This technique has been used in tokamak devices and is one reason for studying the field-reversed configuration confinement approach. As stated earlier, there is hope that compression may be sufficient to heat an FRC plasma to ignition.

Fusion Self-Heating.–The products of a D-T fusion reaction are a helium nucleus—an alpha particle–and a neutron. The neutron, carrying most of the reaction energy, is electrically uncharged and escapes from the plasma without reacting further. The alpha particle, carrying the rest of the energy from the fusion reaction, is charged and remains trapped within the confining mag-

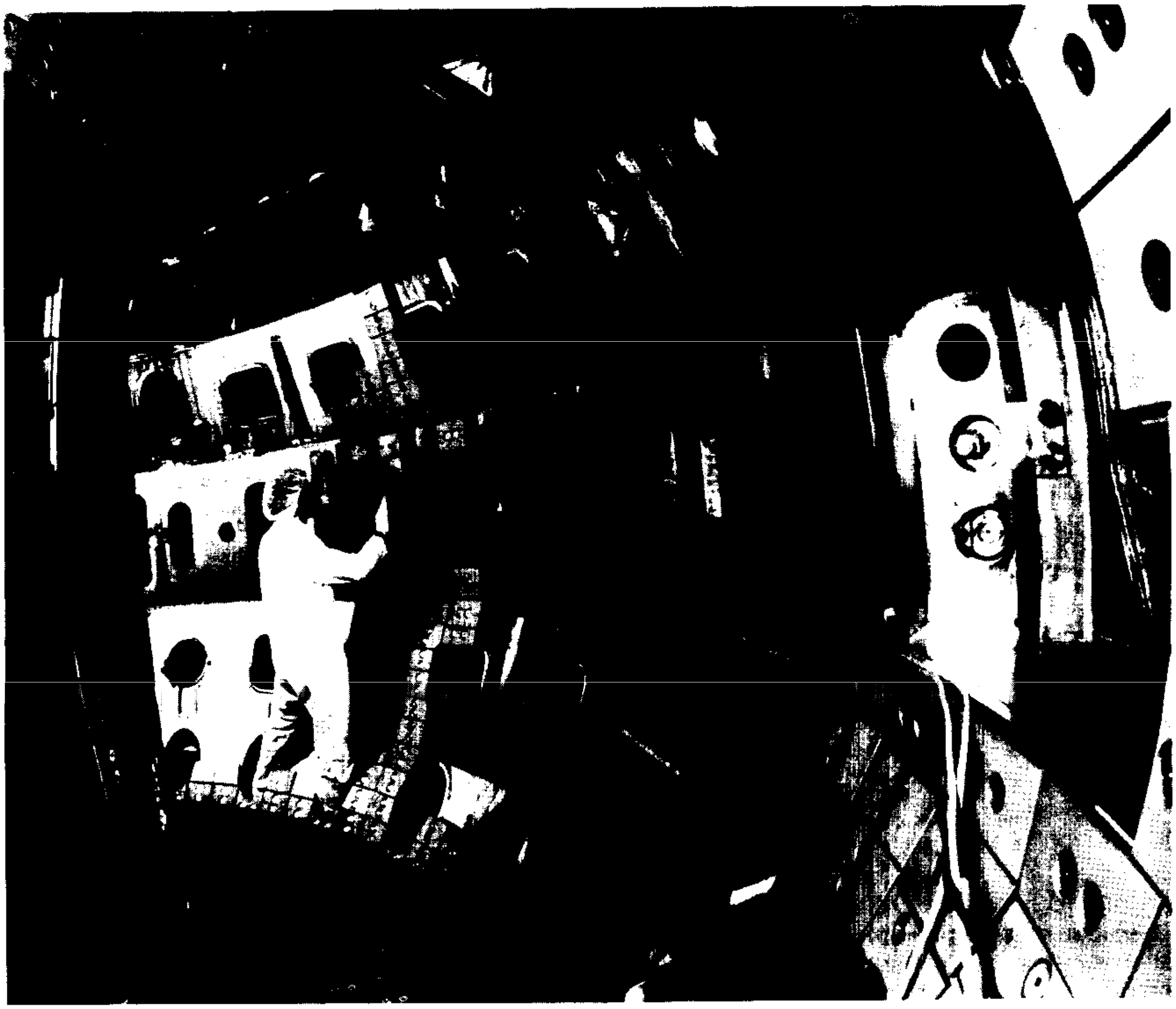


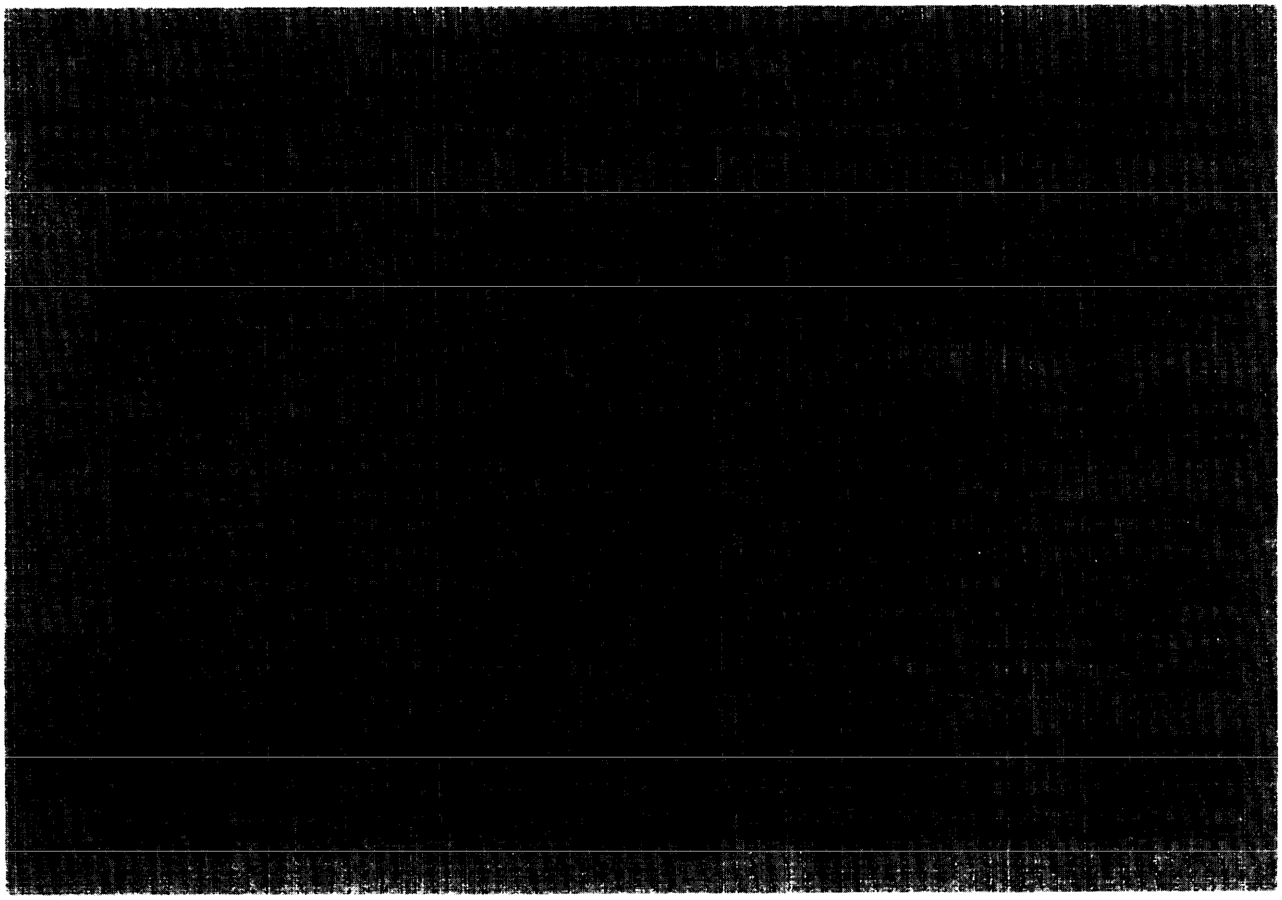
Photo credit: GA Technologies Inc

View inside vacuum vessel of the D II I-D tokamak at GA Technologies, San Diego, California. The plasma is contained within this vessel.

netic fields. Hundreds of times hotter than the surrounding plasma, the alpha particle heats other plasma particles through collisions.

Status.– Recent system studies show that radiofrequency (RF) heating offers significant advantages over neutral beam heating. Consequently, the U.S. neutral beam research program has been reduced while the RF heating program has grown. Various types of RF heating, using different frequencies of radiation from tens of megahertz (millions of cycles per second) to over a hundred gigahertz (billions of cycles per second), are under study. Each frequency range involves different technologies for generation and transmission.

Issues.—Additional research and development (R&D) in heating technologies is essential to meet the needs of future experiments and reactors. Key technical issues in RF heating are the development of sufficiently powerful sources of radiofrequency power (tens of megawatts), particularly at higher frequencies, and the development of launchers or antennas to transmit this power into



the **plasma, particularly** at lower frequencies. Resolution of these issues will require technological development as well as improved understanding of the interaction between radio waves and plasmas.

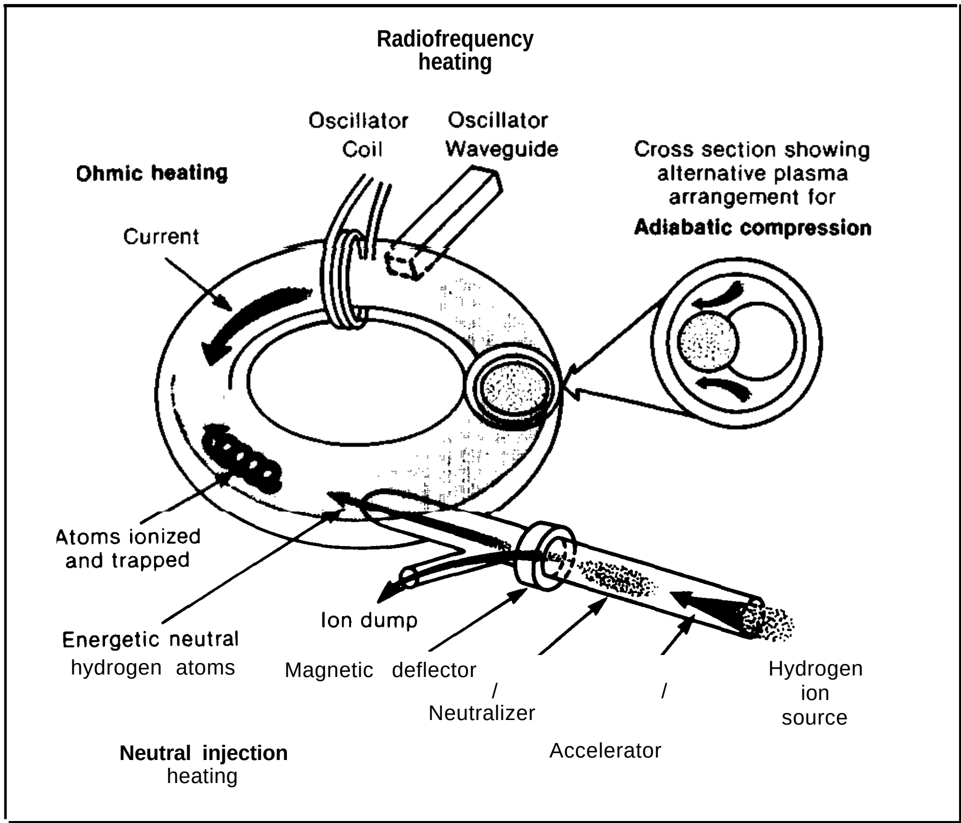
Since no ignited plasma has yet been produced, the effects of fusion self-heating on plasma confinement and other plasma properties are not experimentally known. Confinement could degrade, just as it does with other forms of auxiliary heating. Although self-heating can be simulated in some ways in non-ignited plasmas, its effects can be fully studied only upon reaching high energy gain or ignition. The ignition milestone, therefore, is crucial to the fusion program, and understanding the behavior of ignited plasmas is one of the program's highest scientific priorities.

Fueling

Description. –Any fusion reactor that operates in pulses exceeding a few seconds in length must be fueled to replace particles that escape the plasma and, to a lesser extent, those that are consumed by fusion reactions. Firing pellets of frozen deuterium and tritium into the plasma currently appears to be the best approach for fueling. Both pneumatic (compressed gas) and centrifugal (sling) injectors have been used (figure 4-13). Neutral beam fueling has been used in experiments, but fueling reactors in this way would take excessive amounts of power.

Status.–Pellets up to 4 millimeters in diameter have been fired into experimental plasmas at speeds of up to 2 kilometers per second and at

Figure 4-12.—Plasma Heating Mechanisms



SOURCE. Oak Ridge National Laboratory

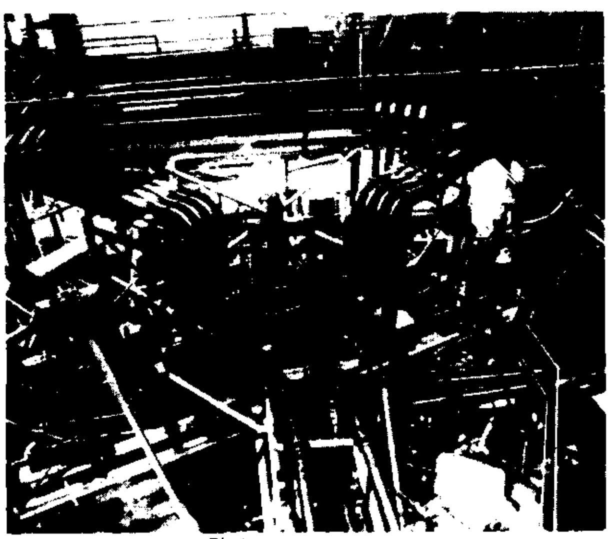


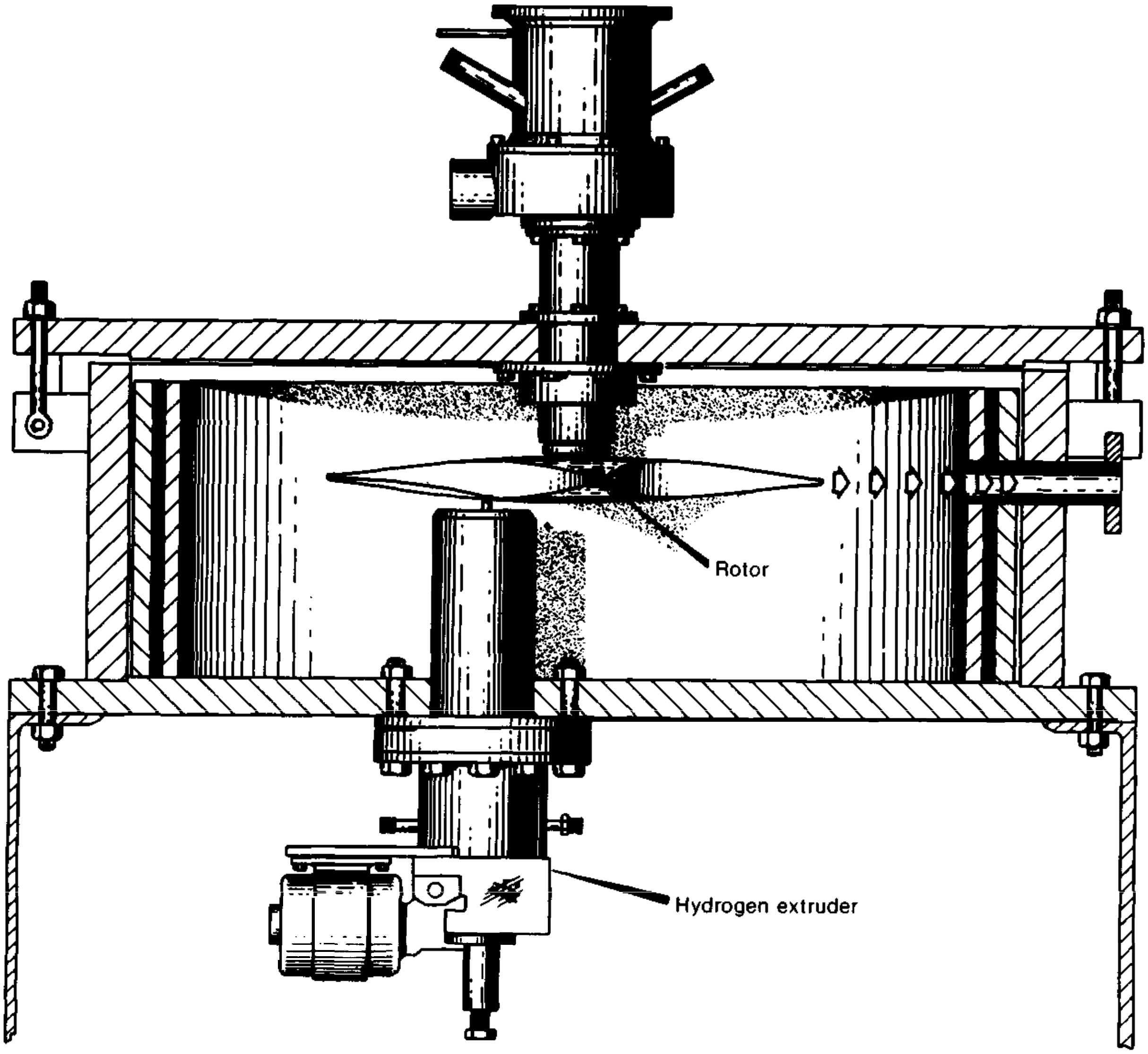
Photo credit: Princeton Plasma Physics Laboratory

Princeton Large Torus at PPPL, showing waveguides for the RF heating system.

repetition rates ofs to 40 pellets per second. U.S. development of pellet fueling technology, centered at Oak Ridge National Laboratory, is well ahead of fueling technology development elsewhere in the world. By building state-of-the-art pellet injectors for use on foreign experiments, the United States is able in return to gain access to foreign experimental facilities.

Issues.—Reactor-scale plasmas will be denser, hotter, and perhaps bigger than the plasmas made o date in fusion experiments; moreover, reactor plasmas will contain energetic alpha particles. All these factors will make it much more difficult for pellets to penetrate reactor plasmas than plasmas made in present-day facilities. Penetration to the center of the plasma, most desirable from a theoretical point of view, probably will be extremely difficult in reactor plasmas. Experiments are now underway to understand how deeply a pellet

Figure 4-13.—Centrifugal Pellet Injection



SOURCE: Oak Ridge National Laboratory

must penetrate. Tokamaks, for example, appear to have a mechanism, not yet understood, that transports fuel to the center of the plasma. Fuel might be brought into the center more effectively in some of the alternate confinement concepts with turbulent plasmas, such as the reversed-field pinch or the spheromak.

If deeper penetration is required than can now be attained, either larger pellets or higher injection speeds will be needed. Larger pellets are not difficult to produce, but they may disturb the plasma too much; additional work needs to be done to determine how fuel pellets affect plasma behavior, If larger pellets cannot be used, higher injection speed will be required, which is technologically much more difficult. Improving present techniques is unlikely to increase injection speeds by more than about a factor of 2. New techniques capable of producing much higher injection speeds are being investigated, but the pellets themselves may not survive injection at these speeds due to fundamental limitations in their mechanical properties.[7]

Current Drive

Description.–Several confinement concepts, including the tokamak, require generation of an electric current inside the plasma. In most present experiments, this current is generated by a transformer. In a transformer, varying the electric current in one coil of wire generates a magnetic field that changes with time. This field passes through a nearby second coil of wire—or in this case the conducting plasma—and generates an electric current in that coil or plasma. Varying the magnetic field is essential; a constant magnetic field cannot generate current.

in tokamak experiments, a coil located in the "doughnut hole" in the center of the plasma chamber serves as one coil of the transformer. Passing a steadily increasing current through this coil creates an increasing magnetic field, which generates current in the plasma. When the current in the first coil levels off at its maximum value, its magnetic field becomes constant, and the current in the plasma peaks and then starts to decay. If the fusion plasma requires a plasma current, its pulse length is limited by the maximum magnetic field of the first coil and the length of time taken for the plasma current to decay.[8]

Status.–Techniques are now being studied for generating continuous plasma currents, rather than pulsed ones, because steady-state reactors are preferable to ones that operate i n pulses. g Injecting radiofrequency power or neutral beams into the plasma might be able to generate such steady-state currents in tokamaks. The injected power or beams generate currents either by "pushing" directly on electrons in the plasma or by selectively heating particles traveling in one direction. Experiments have confirmed the theory of radiofrequency current drive and have succeeded in sustaining tokamak current pulses for several seconds.

Some other confinement concepts, such as the reversed-field pinch or the spheromak, can generate plasma currents with small, periodic variations in the external magnetic fields. Such current-drive technologies do not involve complex external systems.

Issues.–The principal issues involving steady-state current drive are cost and efficiency, especially under reactor conditions. I n particular, the radiofrequency technique becomes less efficient as the plasma density increases. This inefficiency could pose problems because reactors will probably operate at higher densities to maximize generated power. **At this time, it is not known whether the efficiency of continuous current drive can be increased to the point where it could replace the pulsed transformers now used in tokamaks.** However, radiofrequency current drive might also be used to augment the pulsed transformer by starting up the plasma current in a period of low-density operation. Once the plasma current was started, the density could be raised and the transformer used to sustain the current. The radiofrequency current drive together with the transformer would be able to generate longer lasting pulses than the transformer alone.

Reaction Product and Impurity Control

Description .–Alpha particles, which build up as reaction products in steady-state or very long-pulse fusion reactors, will have to be removed so that they do not lessen the output power by diluting the fuel and increasing energy loss by radiation, Devices that collect ions at the plasma edge can be used to remove alpha particles from the plasma. Alpha particles, when combined with electrons that are also collected at the plasma edge, form helium gas that can be harmlessly released. Unburned fuel ions also will be collected; these will be converted to deuterium and tritium gas, which will have to be separated from the helium and reinfected into the plasma.

[7]Argon ne National Laboratory, Fusion Power Program, *Technical Planning Activity: Final Report,* commissioned by the U.S. Department of Energy, Office of Fusion Energy, AN L/FPP-87-l, January 1987, p. 189.

[8]In tokamaks, this decay time can be thousands of seconds. For confinement concepts with plasmas that are more resistive to electric currents, the decay time is shorter, on the order of hundreds of seconds. Operating for periods of time longer than the decay time requires non-transformer current drive mechanisms.

[9]Components in pulsed systems undergo periodic stresses not experienced in steady-state systems. Pulsed reactors also require some form of energy storage to eliminate variations in their electrical output.

The same devices that collect ions at the plasma edge help prevent impurities from entering the plasma. Even small amounts of impurities can cool the plasma by greatly accelerating the rate at which energy is radiated away.

Status.–Two types of devices are being considered for these tasks: *pumped limiters* and *diverter-s.* A limiter is a block of heat-resistant material that, when placed inside the reaction chamber, defines the plasma boundary by intercepting particles at the plasma edge. A variant, the pumped limiter, combines a limiter with a vacuum pump to remove the material collected by the limiter. A divertor generates a particular magnetic field configuration in which ions diffusing out of the fusion plasma, as well as those knocked out of the vessel walls and drifting towards the plasma, are diverted away and collected by external plates.

Both limiters and diverters are in direct contact with the plasma edge. Although temperatures at the edge are far below the 100-million-degree C temperatures found in the plasma center, these components will nevertheless get very hot. All the energy injected into or produced by the plasma that is not carried away by neutrons or radiated away as electromagnetic energy is eventually deposited on the limiter or divertor plates by electrons and ions. Therefore, these devices must withstand high heat loads under energetic ion and neutral particle bombardment while being exposed to intense neutron radiation. In a fusion environment, they will become radioactive due to neutron-induced reactions and, to a much lesser extent, to permeation with tritium. Their reliability must be high since they will be located deep within the reactor, inside the vacuum vessel, where maintenance will be difficult.

issues.–A key issue for reaction product and impurity control will be the choice between pumped limiters and diverters. The devices not only have different efficiencies but have different effects on plasma confinement. Limiters are simpler, but diverters may have operational advantages. [10] More R&D is necessary to investigate issues such as the conditioning and cleaning of surfaces in contact with the plasma, the erosion of these surfaces and redeposition of their materials elsewhere in the plasma chamber, the effects of high heat loads, the development of cooling systems, and the degree and effects of tritium permeation.

Burn Control

Description. —When a fusion reactor plasma is ignited, it provides its own heat and no longer depends on external heating. Two opposing tendencies make it difficult to determine how stable, or self-regulating, an ignited plasma will be. An ignited plasma may be inherently unstable due to the strong temperature dependence of the fusion reaction rate. If, for whatever reason, a hot spot forms in the plasma, the fusion reaction rate there will go up. As a result, more fusion power will be generated in that area, heating it further and compounding the original problem.

If the plasma particles mix with sufficient speed, hot spots that form will not persist long enough to grow. If, on the other hand, the mixing is slow, this *thermal instability* might make it very difficult to maintain a steady reaction. Formation and growth of hot spots could cause output power levels to fluctuate considerably, and in the worst case these hot spots could grow until much of the fuel present in the reaction chamber was consumed. The amount of fuel would not be large—at most a few seconds' worth-making this process more of an operational problem than a safety one. The reactor would have to be designed so that it could not be damaged, and its contents not be released, by the maximum amount of energy that could be produced in this way.

Countering this possible instability is a self-regulating mechanism that limits the maximum attainable value of the beta parameter. Any instability that heated the plasma would increase the plasma beta, which is proportional to temperature. However, since the power generated by a fusion reactor is proportional to beta squared, a reactor would probably already be operating

[10] The H-mode of tokamak operation, in which confinement properties are significantly improved, is seen in tokamaks with divertors. This mode, now thought to depend on processes occurring at the plasma edge, may be difficult to reproduce with limiters.

at the highest value of beta consistent with good performance. Further increases in beta would degrade plasma confinement and increase energy loss. These increased losses would cool the plasma back down, counteracting the initial instability.

These beta-limiting processes could maintain a steady reactor power level. The plasma would tend to operate just under the limiting beta value, and, by adjusting the magnetic field or the plasma density, the power level corresponding to the limiting beta value could be controlled.

Status and Issues.– It is impossible, without creating and studying an ignited plasma, to determine how a plasma will behave in the face of the two opposing tendencies described above. Neither the causes of beta-limiting processes nor their effects on the plasma are fully understood in general. More research is necessary before the ability of these processes to stabilize a fusion plasma can be determined.

The processes that control the reaction rate and burn stability of an ignited plasma are probably the most device-dependent and least understood of any aspect of burning plasma behavior.[11] Even for tokamaks, the properties that determine stability have been studied only under conditions well short of ignition; still less is known about the properties of other confinement concepts. If these issues are indeed concept-specific, only limited information from a burning plasma experiment using one confinement concept can be used to predict the behavior of another.

The Fusion Blanket and First Wall

The region immediately surrounding the fusion plasma in a reactor is called the blanket; the part of the blanket immediately facing the plasma is called the first *wall.* In some designs, the first wall is a separate structure; most often, however, the first wall refers to the front portion of the blanket that may contain special cooling channels.

The blanket serves several functions. Cooling systems in the blanket remove the heat generated by fusion reactions and transfer it to other parts of the facility to generate electricity. Depending on plant design and materials selection, these cooling systems also might be needed to remove afterheat from the radioactive decay of materials in the blanket after a plant shutdown. In addition, the tritium fuel required by the reactor is produced, or "bred," in the blanket. Furthermore, the blanket must support itself and any other structures that are mounted on it.

The safety of the plant will be greatly influenced by the blanket breeder, coolant, and other subsystems. Since the blanket will perform multiple functions, its development will require an integrated R&D program. This program must have two primary aspects: it must develop the capability to predict the behavior of blanket components and systems under actual reactor usage, and it must develop technologies that can produce fuel and recover energy in the blanket while maintaining attractive economic, safety, and environmental features.

Intense irradiation by neutrons produced in the fusion plasma will make blanket components radioactive, with the level of induced radioactivity depending on the materials with which these components are made. Tritium bred within the blanket will add to the blanket's total radioactive inventory. As the largest repository of radioactive materials in a fusion plant, the blanket will be the focus of environmental and safety concerns.[12]

The discussion below focuses on blankets that would be used in fusion reactors that generate electricity. Reactors used for other purposes, some of which are discussed in appendix A, would have different blanket designs.

Description

Energy Conversion.–The first wall is heated by radiation from the plasma as well as by energy carried by particles leaking out of the plasma. Energetic neutrons produced in the plasma penetrate the blanket, where they slow down and convert their kinetic energy into heat. **coolant** circulating within the blanket and first wall transfers this heat to other areas of the plant, where it is

[11] *Argonne National Laboratory, Technical Planning Activity: Final Report, op. cit.,* p. 155.

[12] Plant safety characteristics are discussed further in ch. 5

used to generate electricity. The coolant also prevents blanket and first wall components from overheating during reactor operation; depending on plant design, coolant also may be needed to prevent overheating after plant shutdown, whether scheduled or emergency. Depending on the level of radioactivity within the blanket and coolant, secondary heat exchangers like those now used in nuclear fission plants may be required to isolate the coolant.

Fusion neutrons slow down by colliding with the nuclei of blanket materials, transferring energy to the blanket in the process. Additional heat is also generated in reactions that occur when the neutrons are captured by materials in the blanket. Depending on their energy, the neutrons travel up to several centimeters between collisions. Collisions change the neutrons' directions, and a blanket thickness of from one-half meter to one meter is enough to capture most of the neutron energy.

Tritium Breeding.–Through reactions with fusion neutrons, the nuclei in the blanket can be changed into other nuclei that are either stable or radioactive. In particular, if a fusion neutron is captured by a lithium nucleus, it will induce a reaction that produces tritium (see box 4-B).[13] Therefore, the presence of lithium in the blanket is necessary for tritium breeding.

The number of tritium nuclei produced in the fusion blanket per tritium nucleus consumed in the fusion plasma, called the breeding *ratio,* must be at least 1 for the reactor to be self-sufficient in tritium supply. Accounting for losses and imperfections in the blanket, as well as uncertainties in the data used to calculate tritium breeding rates, this ratio probably should be in the range of 1.1 to 1.2.

Lithium can be contained in the blanket in either solid or liquid form. Lithium metal has a low melting point (186 o C) and excellent heat transfer properties, making it attractive as a coolant in addition to its use in breeding tritium. However, in its pure form, liquid lithium can be highly reactive and may pose safety problems in fusion reactors. Liquid lithium's reactivity can be lessened by alloying it with molten lead, but the addition of lead substantially increases the production of undesired radioactive materials in the blanket.

Liquid metal coolants can be avoided by separating the function of cooling the blanket from that of breeding tritium. A non-lithium-containing fluid can be used as the coolant, and solid lithium-containing compounds can be used to produce tritium. However, solid lithium compounds contain other elements, such as oxygen and aluminum, that would capture some of the fusion neutrons and lower the breeding ratio. To com-

Box 4-B.—Tritium Breeding Reactions

Depending on which isotope of lithium (Li) is involved, either of two reactions can produce tritium in the reactor blanket:

$$n + {}^{6}Li \rightarrow T + {}^{4}He + 4.8\ MeV$$
$$n + {}^{7}Li + 2.5\ MeV \rightarrow T + {}^{4}He + n'$$

As the first equation shows, the reaction of the ⁶Li isotope with a neutron (n) produces tritium (T) and helium (⁴He, or alpha particle), and it releases 4.8 million electron volts (Mev) of energy. In the second equation, the reaction of the ⁷Li isotope with a neutron requires 2.5 MeV of energy in order to occur. Although ⁶Li constitutes only 7 percent of natural lithium, it is so much more likely than ⁷Li to react with a neutron that most of the tritium generated in the blanket results from the ⁶Li reaction. The energy released in this reaction contributes to the total reactor output.

The neutron that is regenerated in the ⁷Li reaction can be captured in the ⁶Li reaction, producing an additional tritium nucleus and making it possible to generate tritium at a higher rate than it is consumed in the plasma. The extra neutrons from ⁷Li reactions also make up for neutrons that escape from the blanket or are absorbed by other substances. Still more neutrons can be produced when neutrons react with substances called neutron multipliers. One of the best neutron multipliers is beryllium (⁹Be), which reacts with one neutron to produce two:

$$n + {}^{9}Be + 1.6\ MeV \rightarrow {}^{4}He + {}^{4}He + n' + n$$

[13] Lithium is a reactive metal that does not occur in its pure form in nature. However, chemical compounds containing lithium are found in many minerals and in the waters of many mineral springs. Fuel resources for fusion are discussed in ch. 5.

pensate for the lost neutrons, substances called *neutron multipliers* can be added to the blanket. Neutron multipliers convert one very fast fusion neutron to two or more slower neutrons by means of a nuclear reaction.

Recovering the tritium from solid breeder materials is more difficult than from liquid breeders. When tritium is bred in a liquid coolant, the coolant carries the tritium directly outside the blanket where it can be extracted. Tritium produced in solid lithium-containing compounds, on the other hand, must first diffuse out of those compounds before it can be collected and flushed out of the blanket by a circulating stream of helium gas.

Blanket Structure.—The heat loads, neutron fluxes, and radiation levels found in the blanket place stringent requirements on the materials with which the blanket is made. Conditions are most severe at the first wall, which is bombarded by neutron and electromagnetic radiation from the plasma, by neutral particles, and by plasma electrons and ions that escape confinement. First wall issues are similar to many of the issues associated with limiters and diverters (discussed previously in the section on "The Fusion Plasma, " under the heading "Reaction Products and impurity Control"), which undergo even higher heat and particle fluxes than the first wall.

Neutron irradiation introduces two major problems in the blanket materials. First, irradiation can lead to brittleness, swelling, and deformation of the reactor structural materials. During the service lives of first wall and blanket components, each atom in those components will be displaced several hundred times by collisions with fusion neutrons. The amount of radiation damage that the blanket materials can withstand determines component lifetimes and also places an upper limit on reactor power for a blanket of a given size.

Second, neutron irradiation makes the blanket radioactive. Not all the neutrons penetrating the blanket will be captured in lithium to breed tritium, Some will be absorbed by other blanket materials, making those materials radioactive.

[14]Beryllium and lead are commonly used in reactor studies as neutron multipliers.

Other fusion neutrons will penetrate the blanket to make reactor structures outside the blanket radioactive. Since the degree of radioactivity generated within the reactor structure strongly depends on the reactor's composition, development and use of *low-activation* materials that do not generate long-lived radioactive products under neutron bombardment will greatly lessen induced radioactivity,

Due to radiation damage, blanket and first wall components in a fusion reactor will require periodic replacement. After their removal, the old components will constitute a source of radioactive waste.[15]

Impact on Fusion Reactor Design .–Although the blanket and first wall components themselves may not represent a large fraction of the cost of a fusion reactor, blanket design has a substantial influence on total reactor cost. The blanket thickness (along with that of the shield, described below) determines the size and cost of the magnets, which are substantially more expensive than the blanket. The blanket coolant temperature determines the overall efficiency with which the plant converts fusion power into electricity, directly affecting the cost of electricity. The selection of materials in the blanket determines the amount of long-term radioactive waste and the amount of heat produced by radioactive decay in the blanket after plant shutdown; both the waste and the heat affect the reactor's environmental and safety aspects. Finally, the ability of the blanket materials to withstand heat loads and neutron irradiation levels determines the amount of fusion power that can be generated in a plant of a given physical size, which has a significant effect on reactor size and cost and on its behavior during accidents.

Status and Issues

A wide variety of designs have been proposed for the blanket and first wall. However, since the

[15]According to Argonne National Laboratory, *Technical Planning Activity: Final Report*, op. cit., p. **283,** the amount of long-lived (more than 5 years) radioactive waste depends primarily on the amounts of copper, molybdenum, nitrogen, niobium, and nickel used in the blanket. Radioactivity levels, radioactive wastes from fusion reactors, and low-activation materials are discussed further in ch. 5.

fusion research program has concentrated to date primarily on plasma science issues, relatively little experimental work has been done on blanket design or fusion nuclear technologies in general. As the program moves from establishing scientific feasibility to demonstrating engineering feasibility, engineering issues will become much more important.

Tritium Self-Sufficiency .-Engineering designs must be developed to produce tritium at a rate equal to the rate of consumption in the plasma plus an additional margin, The extra tritium, 10 to 20 percent of the amount consumed in the plasma, is needed to compensate for losses due to radioactive decay and to provide the initial inventory to start up new reactors. Improvements in calculating neutron flow through the reactor structure and in collecting additional basic nuclear data such as reaction rates are necessary to develop adequate engineering designs. Experimental verification of the calculation methods and data is also required to demonstrate tritium self-sufficiency.

Structural Materials.–Structural materials in the first wall and deeper in the blanket must be developed that can withstand neutron-induced effects such as swelling, brittleness, and deformation. Stainless steel alloys already have been identified that appear to show adequate performance under neutron fluxes at the low end of those expected in a reactor. However, these materials produce more radioactive products than may be desirable for commercial reactors. Developing low-activation materials that also have acceptable physical properties under irradiation remains a significant challenge. The task will require further basic research in materials science as well as progress in materials technology.

Non-Structural Blanket Materials.–The tritium-breeding properties of various lithium-containing materials must be studied and compared. The choice between solid and liquid breeder materials, in particular, will greatly affect overall blanket design. in addition to lithium, other materials may be required in the blanket such as neutron multipliers and *moderators* (which slow down neutrons to make them more easily absorbed in the blanket). Insulators, or materials that do not conduct electricity, also maybe required for high radiation areas inside the reactor. Since a typical effect of radiation damage on insulators is to increase electrical conductivity, developing materials that will remain insulators under high radiation fluxes is a challenging task.

Special Materials.–Other materials requirements for a fusion reactor may include special materials to coat plasma-facing surfaces to minimize their effect on the plasma, coatings or claddings used to form barriers to contain tritium, and advanced superconducting magnet materials (described in the section on magnets, below). Many of the specific requirements for these materials have not yet been determined.

Compatibility.–Certain combinations of materials, each suitable for a particular task, may in combination prove unacceptable in a reactor design. For example, liquid lithium reacts violently with water, so a liquid lithium-cooled blanket design would probably prohibit use of water as an additional coolant.

Tritium Permeation and Recovery.–Once produced inside the blanket, tritium must be recovered and removed. However, tritium will permeate many materials that are continuously exposed to it. Its interactions with blanket materials under the conditions inside a fusion reactor will have to be understood. In particular, tritium may be difficult to collect from solid breeder materials.

Liquid Metal Flow.—Liquid metal coolants in a fusion reactor will be subject to strong magnetic fields created both by the plasma and by external magnets. Liquid metals are conductors of electricity; when electrical conductors move through magnetic fields, voltages and currents are generated.[16] The currents induced in the coolant, in turn, are subject to forces from the magnetic field that oppose the motion of the coolant, increasing coolant pressure and adding to the power required for pumping.

The Shield

Description and Status

Since many neutrons will penetrate all the way through the blanket during fusion reactor oper-

[16] This process is the basis of electrical generators.

ation, a shield may be required between the blanket and the magnet coils.[17] The shield may be composed of materials such as steel and water, will probably contain a circulating coolant, and would have a thickness from tens of centimeters to over a meter. The shield would provide extra protection to the magnets and could reflect escaping neutrons back into the blanket to improve the efficiency of tritium breeding. Additional shielding would probably surround the entire reactor core, perhaps in the form of thick walls for the enclosing building.

Most existing fusion experiments have not been designed to use tritium and are incapable of generating significant amounts of fusion power. Consequently, shielding has generally not been an important issue for the research program. It has, however, been a factor in the design of devices such as TFTR and the Joint European Torus that are intended to use tritium. As future machines are designed that will generate appreciable amounts of fusion power, shielding will become more important.

Issues

The intensity of the incident neutron radiation; the size, shape, and effectiveness of the shield; and the permissible levels of neutron irradiation penetrating the shield must all be determined to evaluate shielding requirements. As improved magnet materials are developed that are less sensitive to neutron radiation, shielding requirements for plant components will lessen. However, protection of plant personnel alone will require substantial shielding.

The Magnets

Description

The external confining magnetic fields in a fusion reactor are generated by large electric currents flowing through magnet coils surrounding the plasma. These magnets must withstand tremendous mechanical forces,

[17] If the magnets are superconducting, the shield will be required. If they are made of copper, the shield may not be necessary. However, without a shield, the coils would require periodic replacement. See the following section on magnets.

The most important choice concerning design of the magnets is whether they will be made of superconducting materials or of conventional conductors such as copper. Copper is an excellent conductor of electricity but nevertheless has sufficient resistance to electric currents that a great deal of power is wasted as heat when the magnet is running. This heat must be removed by cooling systems. Superconducting coils lose all resistance to electricity when cooled sufficiently; below a temperature called the *critical temperature,* their magnetic fields can be sustained without any additional power. However, power is required to establish the fields initially, and a small amount of refrigeration power is required to keep superconducting magnets at their operating temperature. No heat is generated inside a superconducting magnet, but heat that leaks in from the outside must be removed.

Although recent discoveries could revolutionize the field (see "issues" section below), all superconducting materials that have so far been used in large magnets have critical temperatures within about 20° K of absolute zero. The only substance that does not freeze solid at these temperatures is helium, and the only way to cool superconducting magnets to these temperatures is to circulate liquid helium through them. Use of liquid helium makes superconducting magnets more complicated and expensive to build than copper magnets; superconducting magnets also require thicker shields. However, superconducting magnets require much less electricity to run, substantially lowering their operating costs.

Conceptual design studies typically have shown that the operational savings from using superconducting magnets in commercial fusion reactors would more than compensate for their higher initial cost. However, there may be exceptions, especially for confinement concepts with higher beta values that are able to confine fusion plasmas at lower magnetic field strengths. At lower field strengths, copper magnets, which do not require as much shielding as superconducting magnets, can be made to fit more closely around the plasma chamber. The resultant reduction in size of the magnet/shield combination might reduce its cost enough to outweigh the operational inefficiencies of copper magnets.

Whereas the magnets in future fusion reactors will operate for long pulses, if not continuously, magnets in present-day fusion experimental facilities generally operate only for several seconds at a time. For pulses this short, the cost of electricity is less of a factor in determining magnet design, making the simpler construction of copper magnets preferable in most cases. A notable exception is the MFTF-B device at Lawrence Livermore National Laboratory, which was built with superconducting magnets because copper coils would have been prohibitively expensive to operate even for 30-second pulses.

Status

The first fusion device built with superconducting magnets was the Soviet T-7 tokamak, completed 7 years before any Western fusion device using superconducting magnets. The Soviets are now building T-1 5, a much larger superconducting tokamak. Difficulties with the T-1 5 magnets have been among the reasons that the project's completion has been delayed for several years; however, these difficulties apparently have been resolved. The Tore Supra tokamak being built in France will also use superconducting magnets and will probably exceed the parameters of T-15.[18] In the United States, MFTF-B was completed in 1986; its superconducting magnets have been successfully tested at their operating conditions. Overall, DOE considers U.S. magnet development to be comparable to that in Europe and Japan and ahead of that in the Soviet Union.[19]

Generally, magnet development has been associated with individual fusion confinement experiments rather than with facilities dedicated specifically to magnet development. A major exception is the Large Coil Task, an international program to build and test superconducting magnets. Magnets developed through the Large Coil Task have worked very well and have exceeded their original design specifications.[20]

Issues

Recent discovery of new superconducting materials with critical temperatures far above those of previously known materials, and possibly with the capability to reach very high magnetic field strengths, will have a profound impact on a great many fields, including fusion. Materials have been identified with critical temperatures higher than the 77 Kelvin (–196° C) threshold that would permit use of liquid nitrogen as a refrigerant. Liquid nitrogen is cheaper and easier to handle than liquid helium, and its use could reduce the cost and complexity of superconducting magnets.

However, the discovery of these "high-temperature' superconductors does not necessarily mean that they can soon be utilized in fusion applications. Little is known about the physical processes underlying superconductivity in these materials, and they present great engineering challenges. They are difficult to fabricate into magnet coils, and they may not be able to withstand the forces exerted in fusion magnets. Although their current-carrying capability is improving, they may not be able to carry high enough currents under high magnetic fields to be useful in large-scale magnets. Moreover, their response to neutron irradiation is not known. If these materials are highly susceptible to radiation damage, their use in fusion magnets could be difficult. Conversely, if they proved more resistant to radiation effects than previous superconducting materials, thinner shields and correspondingly smaller magnets could be used.

Further research is required to see whether the new superconducting materials can be used in practical applications. The great economic advantage that they would have in numerous applications ensures that much of this research will be undertaken independently of the fusion program. However, fusion does have particular requirements for large, high-field magnets that may not otherwise be investigated.

161 nformation on Soviet tokamak development is from an oral presentation on "Assessment of Soviet Magnetic Fusion Research" by Ronald C. Davidson, Director of the Plasma Fusion Center, Massachusetts Institute of Technology, to the Magnetic Fusion Advisory Committee, Princeton, NJ, May 19, 1987.

19International comparisons are from a presentation by R.J. Dowling, Director of the Division of Development and Technology, Office of Fusion Energy, U.S. Department of Energy, to the Energy Research Advisory Board, Sub-Panel on Magnetic Fusion, Washington, DC, May 29, 1986.

20The Large Coi l Task is discussed further in ch. 7.

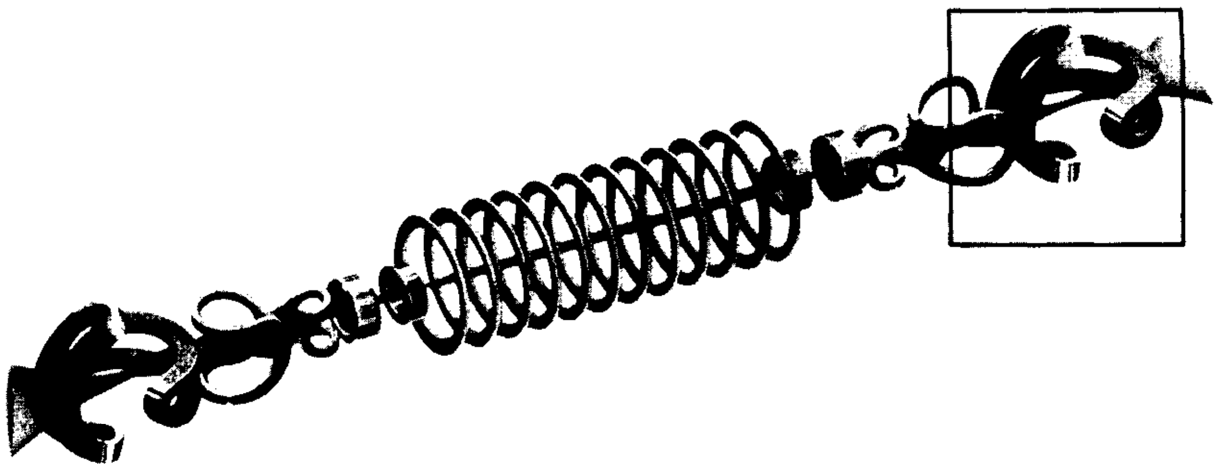


Photo credit: Lawrence Livermore National Laboratory

375-ton superconducting magnet being moved to the east end cell of MFTF-B for installation. This magnet's location within MFTF-B is shown at top.

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Issues for superconducting fusion magnet materials include further development and investigation of the new high-temperature materials. If these new materials prove unacceptable for fusion, improvements in the strength, workability, and maximum current capacity of the previously known superconductors will be important. Issues for copper magnets include fully exploiting copper's strength and developing joints (needed to assemble and maintain the magnet) that can carry large electric currents.

Fuel Processing

Description

Tritium fuel contained in the exhaust from the plasma, or generated in the reactor blanket, must be extracted, purified, and supplied back to the fueling systems for injection into the plasma. Considerable experience has been developed in handling tritium, particularly within the nuclear weapons program, making tritium technology more highly developed than many of the other nuclear technologies required for fusion. However, this experience is applicable primarily to handling the tritium in a fusion reactor once it has been produced and separated. The task of extracting tritium from a blanket under reactor conditions while at the same time generating electric power with high efficiency has yet to be done.

Status

To acquire experience with tritium handling for fusion applications, DOE has built and is operating the Tritium Systems Test Assembly (TSTA) at Los Alamos National Laboratory. A prototype of the tritium processing and handling facilities needed for a full-scale fusion reactor, TSTA includes plant safety equipment such as a room atmosphere detritiation system. TSTA operators have developed system maintenance procedures that minimize or eliminate tritium release. This system, however, does not duplicate the production or extraction of tritium from a fusion blanket.

Issues

Specific issues involved with tritium processing include monitoring, accountability, and safety. Being radioactive, tritium cannot be allowed to diffuse out of the reactor structure. if tritium collects on inaccessible surfaces within the reactor, it cannot be completely recovered, and it will make those surfaces radioactive. Developing tritium processing systems will require additional research in measuring basic tritium properties such **as** diffusivity, volubility, and oxidation chemistry. Safety needs include developing and maintaining the capability to contain and recover tritium from air and from water coolants (if any) in the event of tritium contamination.

Remote Maintenance

Due to their inventory of radioactive tritium and the activation of their structural components, the interior of all subsequent fusion experiments that burn D-T will become too radioactive for hands-on maintenance. Therefore, remote maintenance is a key issue not only for future power reactors, but also for near-term D-T experiments. Nearly all aspects of the research program, from design of experiments to operation and maintenance to decommissioning, will be affected by the need for remote maintenance.

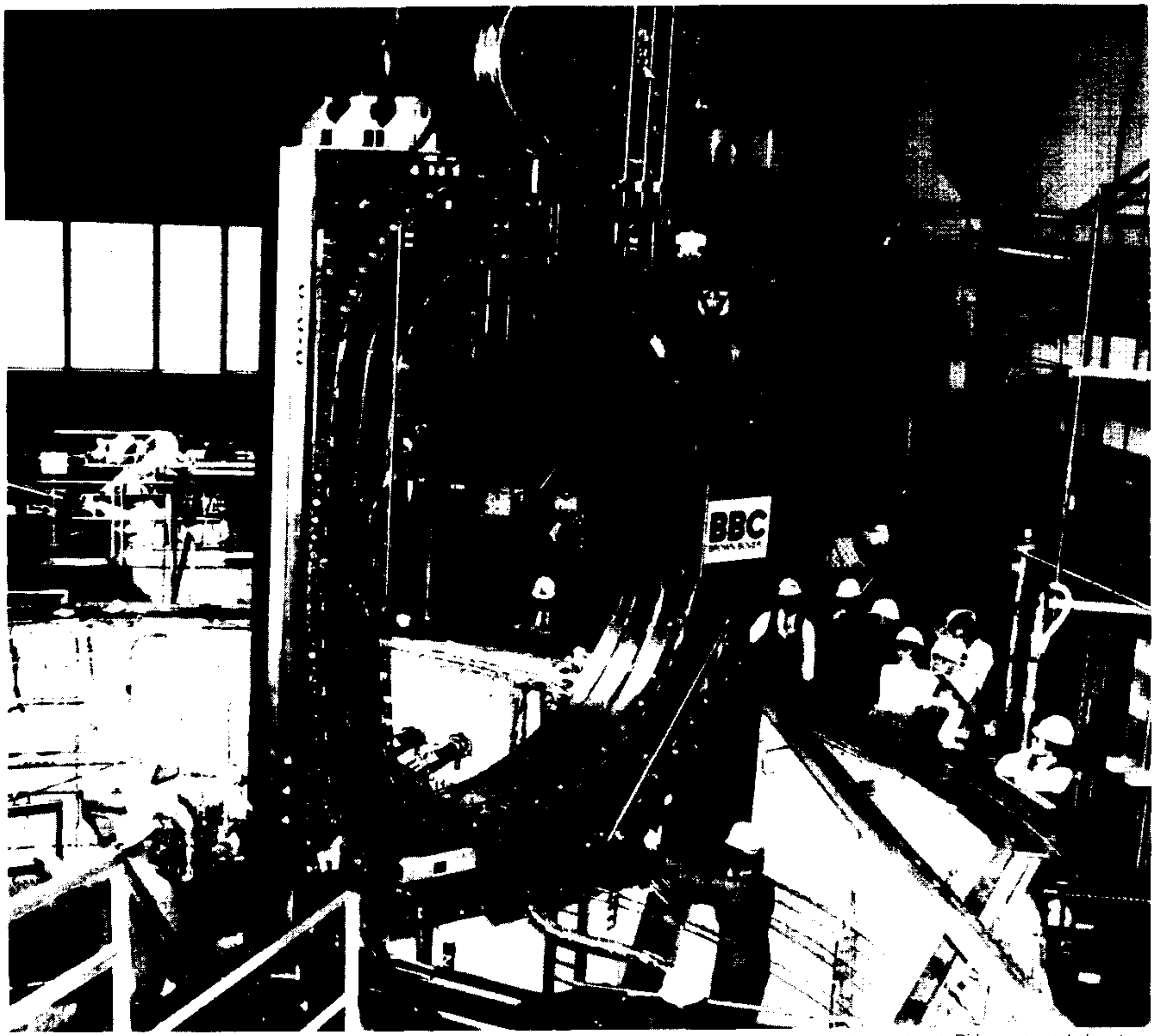


Photo credit Oak Ridge National Laboratory

The Swiss superconducting magnet coil being installed in the International Fusion Superconducting Magnet Test Facility (Large Coil Task) at Oak Ridge National Laboratory.

The fusion program at present is relying on activities outside the fusion community for general development of remote maintenance equipment. Much work in remote manipulation and remote maintenance has been done, but some applications are likely to be unique to fusion and will require special development. Remote maintenance requirements for fusion facilities will include transporters able to move heavy loads (over 100 tons) with precision alignment; manipulators made of nonmagnetic material that can operate under high vacuum conditions; and rapid and precise remote cutting, welding, and leak detection equipment. The first challenge in this field

Photo credit: Los Alamos National Laboratory

The Tritium Systems Test Assembly at Los Alamos National Laboratory

will be identification of the remote maintenance requirements for near-term facilities and the development of any necessary equipment. Subsequently, **needs** for test facilities and reactors must be identified and assessed.

ADVANCED FUEL AND ENERGY CONVERSION CONCEPTS

Even though the fusion power core systems described in the previous section have not yet been developed, researchers already are designing reactors using more advanced concepts. Improvements described below are not mere refinements of the systems already described; they are qualitatively new features that may be much more attractive. In general, these improvements involve use of either advanced fuels or advanced methods of converting fusion energy into useful forms.

Advanced Fuels

The fusion power core described in the previous section uses D-T fuel because it is by far the most reactive of all potential fusion fuels. This reactivity can be increased still further by aligning the internal spins of the deuterium and tritium nuclei, a technique known as *spin polarization.* If the spins can be aligned initially, the magnetic field of the fusion reactor will tend to keep them in alignment. Therefore, research is ongoing at Princeton Plasma Physics Laboratory to develop intense sources of spin-polarized fuel.

The principal disadvantage of D-T fuel is that the D-T reaction produces energetic neutrons that cause radiation damage and induce radioactivity in reactor structures. Moreover, reactors using D-T must breed their own tritium, substantially adding to reactor complexity and radioactivity levels. For these reasons, the possibility of using other fuels in fusion reactors is being investigated.

Fuels other than D-T require higher temperatures and Lawson confinement parameters to reach ignition and higher beta values to perform economically. Achieving these parameters will require stronger magnetic fields, higher plasma currents, and substantial improvements in other plasma technologies beyond those needed to reach ignition with D-T fuel—a task that in itself has not yet been accomplished. However, reactions that use advanced fuels would have a number of advantages:

- They would require little to no tritium, reducing or eliminating the need for the blanket to breed tritium and permitting a much wider range of blanket designs. Tritium inventories would be smaller and the consequent radioactivity levels would be lower.
- They would generate fewer and lower energy neutrons, alleviating radiation damage and minimizing radioactive wastes.
- They might permit the use of more efficient methods to generate electricity from fusion energy. In advanced fuel fusion reactions, more energy is released in the form of energetic charged particles, such as protons or alpha particles, than is the case in the D-T reaction. Therefore, these advanced fuels may be amenable to various techniques that generate electricity directly from the fusion plasma or from plasma-generated radiation without having to first convert the energy into heat. (See the following section on "Advanced Energy Conversion.")

Table 4-8 presents five fusion fuel cycles, including the "baseline" D-T cycle and four possibilities for advanced fuel cycles. Of the advanced cycles, the D-^{3}He cycle is currently drawing the most attention within the fusion community. The primary reaction produces no neutrons, and neutrons resulting from corollary D-D reactions can be minimized by using a mixture consisting mostly of 3He or by using spin-polarization.[21]

[21] Deuterium, being relatively scarce in a ^{3}He -rich mixture, would be much more likely to react with a ^{3}He nucleus than with another deuterium nucleus, making D-D reactions relatively rare. However, one consequence of this mode of operation, in addition to minimizing neutron generation, would be the lessening of output power since most of the ^{3}He nuclei would be unable to find D nuclei with which to react. Increasing the ratio of D to 'He to more nearly equal proportions, theretore, would increase both the output power and the neutron generation.

Spin polarization suppresses the D-D reaction rate because, unlike the D-T reaction, two deuterium nuclei whose spins are aligned are less likely to react with each other,

Table 4-8.—Fusion Fuel Cycles[a]

Cycle	Primary reaction	Percent of energy carried by charged particles
D-T cycle	$D + T \rightarrow {}^4He + n + 17.59$ million electron volts (MeV) [D = deuterium; T = tritium; ^{4}He = alpha particle, or helium nucleus]	20%
D-D cycle	$D + D \rightarrow p + T + 4.03$ MeV $D + D \rightarrow {}^3He + n + 3.27$ MeV [p = proton; ^{3}He = helium isotope with one less neutron than ^{4}He]	62%[b]
D-^{3}He cycle	$D + {}^3He \rightarrow {}^4He + p + 18.34$ MeV	up to 98%[c]
D-^{6}Li cycle	$D + {}^6Li \rightarrow 5$ different reactions [^{6}Li = isotope of lithium]	over 65%
p-^{11}B cycle	$p + {}^{11}B \rightarrow {}^4He + {}^4He + {}^4He + 8.66$ MeV [^{11}B = isotope of boron]	almost 100%[d]

[a] Presented in order of increasing difficulty; the last reaction is from 100 to 10,000 times harder to ignite than the first one, depending on temperature.
[b] Sixty-two percent is the fraction of the energy carried off by charged particles, assuming that the intermediate reaction products (T and ^{3}He) react further via D-T and D-^{3}He reactions. With these additional reactions, the full reaction is

$$6D \rightarrow p + p + n + n + {}^4He + {}^4He + 43.23 \text{ MeV}.$$

[c] Ninety-eight percent can be attained for mixtures lean in D and rich in ^{3}He (see footnote 21 in main text, above).
[d] A low energy (0.15 MeV) neutron is produced in the secondary reaction ${}^4He + {}^{11}B \rightarrow n + {}^{14}N + 0.158$ MeV [^{14}N = isotope of nitrogen].

SOURCE: U.S. Department of Energy, *Background Information and Technical Basis for Assessment of Environmental Implications of Magnetic Fusion Energy*, DOE/ER-0179, August 1983, p. 2-3 (table 2.1) and pp. 2-24 to 2-27, including table 2.2.

However, the D-^{3}He reaction is much more difficult to start than the D-T reaction. The minimum temperature required to ignite D-^{3}He is several times higher than that needed for D-T; the minimum confinement parameter is about 10 times higher. Given that the requirements for igniting D-T have not yet been experimentally achieved, attaining conditions sufficient to ignite D-^{3}He is considerably farther off. On top of its technological requirements, ^{3}He is scarce. It is an isotope of helium with one fewer neutron than natural helium (^{4}He), and it occurs on earth only as the end-product of tritium decay. The only way to collect ^{3}He is to make tritium and wait for it to decay or to breed ^{3}He as the product of another advanced fuel fusion reaction, the D-D reaction. Due to the scarcity of ^{3}He, the D-^{3}He reaction has been considered primarily an academic curiosity until recently.

Today, a resurgence of excitement about ^{3}He comes with the discovery that it is found in substantial amounts in the uppermost layers of soil on the moon. Analysis of moon rocks brought back by the Apollo missions shows that ^{3}He, which is constantly emitted by the sun and carried by the solar wind, is deposited and retained in the lunar surface. In principle, a rocket with the cargo volume of the space shuttle could carry back enough liquid ^{3}He to generate all the electricity now used in the United States in one year. Of course, the technology to recover ^{3}He from the moon would not be available for decades, and the energy and capital investment required to mine, refine, liquefy, and transport the ^{3}He have yet to be evaluated.[22]

Advanced Energy Conversion

Despite the very high-level technology in the fusion core, a baseline fusion reactor would generate electricity in much the same way that present-day fossil fuel and nuclear fission powerplants do. Heat produced in the reactor would be used to boil water into steam, which would pass through turbines to drive generators. Through this process, about 35 to 40 percent of the energy produced in the fusion reaction would be converted into electricity, with the remainder discharged as waste heat. This efficiency, roughly the same as that of fossil fuel and nuclear fission generating stations, is determined primarily by the process of generating electricity from the energy in the steam. Efficiency could be raised if advanced, high temperature materials in the blanket and first wall of a fusion reactor permitted higher coolant temperatures to be used.

If the intermediate step of heating steam could be bypassed, a higher percentage of the energy released in fusion reactions could be converted into electricity. Several techniques to integrate generation of electricity directly into the fusion power core have been conceived. One of these, applicable to D-T reactors as well as to advanced fuel reactors, would convert energy carried off by escaping charged particles directly to electricity by collecting the particles on plates. This technique is most applicable to open confinement concepts, in which charged particles can be allowed to escape along magnetic field lines.

Other techniques, which can work with closed confinement concepts, require plasma temperatures significantly higher than the 10- to 15-kiloelectron-volt D-T ignition temperatures. Very hot plasmas radiate more energy away in the form of microwave radiation than cooler plasmas do,[23] and it appears that this radiation could be captured at the first wall or in the blanket and converted directly into electricity. These "direct conversion" techniques would be better suited to advanced fuels, which not only burn at higher temperatures than D-T but also produce most of their energy in the form of energetic charged particles. Unlike neutrons, which escape from the plasma without heating it, charged particles are retained within the plasma. The D-T reaction, in which only 20 percent of the energy is given to charged particles, is less suitable for techniques that recover energy directly from the plasma.

Several direct conversion techniques that may convert well over 35 percent of the fusion energy to electricity have been identified. Until they can be tested experimentally under conditions similar to those in an advanced fusion reactor, they must be considered speculative. Nevertheless, they provide a tantalizing goal.

[22] Use of lunar ^{3}He is discussed in "Lunar Source of ^{3}He for Commercial Fusion Power," by L.J. Wittenberg, J.F. Santarius, and G.L. Kulcinski, *Fusion Technology* 10(2): 167, 1986.

[23] See item 2 in box 4-A, "Plasma Energy Loss Mechanisms."

RESEARCH PROGRESS AND FUTURE DIRECTIONS

In 35 years of fusion research, the technological requirements for designing a fusion reactor have become clearer, and considerable progress has been made towards meeting them. Improved understanding, based on both experiments and increased computational ability, is providing much of the predictive capability needed to design, and eventually to optimize, future plasma experiments and fusion reactors.

Major advances in plasma research have been made possible by progress in tokamak plasma technologies:

- By the 1960s, experiments demonstrated the crucial importance of attaining high vacuum and low impurity levels in the plasma to achieve high densities, temperatures, and confinement times.
- In the mid-197os, neutral beam technology was first used to heat plasmas to temperatures several times higher than those previously attained. High-performance, high-field copper magnets were used to obtain high Lawson confinement parameters in compact tokamak plasmas.
- The development in the late 1970s of pellet injectors to fuel plasma discharges led to further advances in plasma density and confinement. Development of the poloidal divertor at about the same time led to the discovery of the "H-mode," a mode of tokamak behavior that was not subject to degraded confinement when auxiliary heating was used.
- In the early 1980s, advances in high-power radiofrequency technology gave experimenters new tools to modify the temperature, current, and density distributions within the plasma. Much of this new capability has yet to be exploited.

These accomplishments have contributed to the steady progress in plasma parameters plotted in figure 4-14. Figure 4-14(a) shows the product of the temperature, density, and confinement time that has been achieved simultaneously in various experiments over the last 20 years. Since all three of these parameters must be high simultaneously for the product to be high, this product provides a rough measure of how well these three requirements have been simultaneously achieved.

The next figure, 4-1 4(b), plots the temperature alone and compares it to the minimum temperature below which neither breakeven nor ignition can occur no matter how high the density and confinement time. The TFTR point shows temperatures well into the reactor regime and far above that needed for ignition. However, the fact that the corresponding TFTR point in figure 4-14(a) is below the ignition threshold indicates that high temperature is not sufficient; the product of density and confinement time must also be high for ignition.

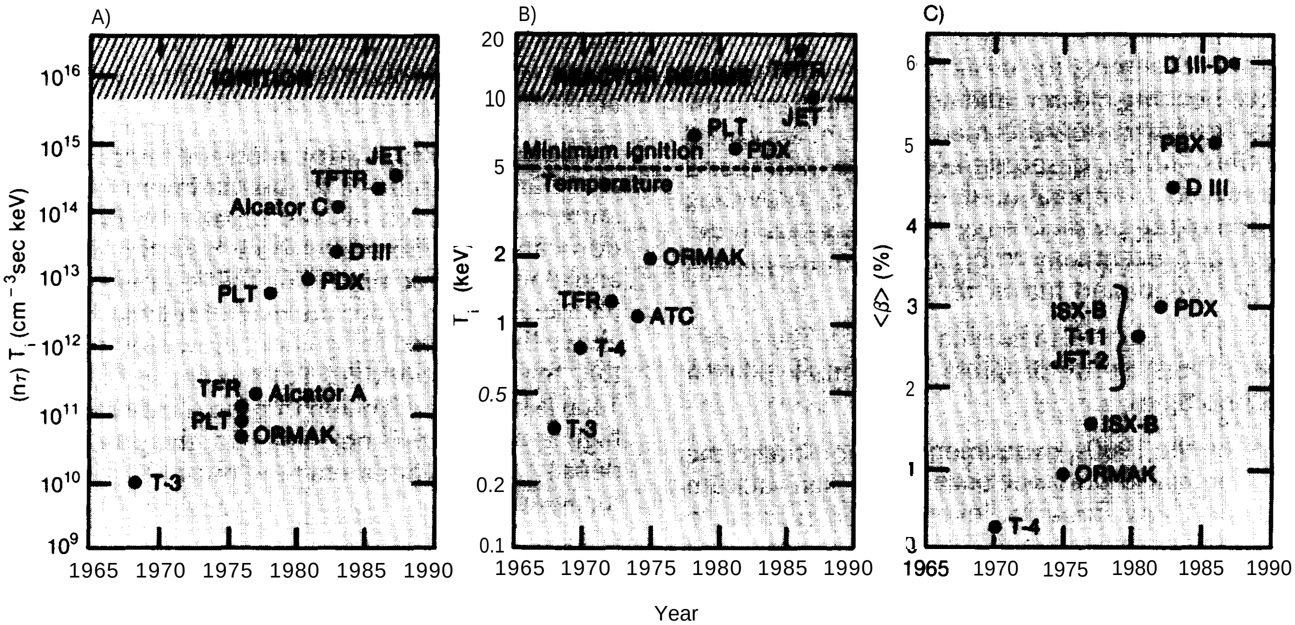
Figure 4-1 4(c) shows progress in the parameter beta, the ratio of plasma pressure to magnetic field pressure. Note that devices that have achieved high values on one of the three plots often have not been the ones that have gotten the highest values in others. Future devices will have to achieve high values in all areas simultaneously.

The Technical Planning Activity

The technological issues to be resolved betore fusion's potential as a power producer can be assessed have been examined in detail in the Technical Planning Activity (TPA), an analysis commissioned by DOE's Office of Fusion Energy and coordinated by Argonne National Laboratory.[24] Over 50 scientists and engineers from the fusion community identified and analyzed the tasks and milestones that constitute the research needed to reach the goal of the DOE fusion program: the establishment of the "scientific and technological base required to carry out an assessment of the economic and environmental aspects of fusion energy."[25] According to DOE's assignment to TPA, the assessment of fusion would be culminated by the construction and operation of "one or more integrated fusion facilities . . . in the post 2000 period."[26]

[24] Argonne National Laboratory, *Technical Planning Activity: Final Report*, op. cit.
[25] Ibid., "Technical Planning Activity Mission Statement," p. 348
[26] Ibid.

Figure 4-14.-Progress in Tokamak Parameters



(A) $n \cong T_i$, representing the simultaneous achievement of the three parameters—density, ion temperature, and confinement time—needed to produce fusion power

(B) T_i = ion temperature

(C) <β> = beta = ratio of plasma pressure to magnetic field pressure; provides a measure of the efficiency with which the magnetic fields are used

SOURCE: Updated from National Research Council, *Physics Through the 1990s: Plasmas and Fluids* (Washington, DC: National Academy Press, 1986), figure 4.6, p. 160.

The nature of an integrated fusion facility is not specified by either DOE or TPA. The TPA report describes it only as "the beginning of the commercialization phase of fusion" that could "perhaps" take the form of a demonstration power reactor.[27] The decision to proceed with an integrated facility is scheduled in the TPA report in the year 2005.

Key Technical Issues and Facilities

DOE defines four "key technical issues" that must be resolved: magnetic confinement systems, properties of burning plasmas, fusion materials, and fusion nuclear technology.[18] For each of these issues, TPA set technical goals and determined requirements for facilities that could reach these goals.

Magnetic Confinement Systems

The key issue in confinement systems is the development of confinement concepts that would be suitable for commercial fusion reactors. Progress here will require that a series of facilities be built for whichever concepts are judged worthy of further development. A preliminary experiment that investigates basic characteristics of a new concept can be done for a few million dollars or less. An experiment that looks promising can be followed up by a larger "proof-of-con-

[27]Ibid., p. 26. On p. 11 of the TPA report, table S.2 provides "representative goals" for an integrated fusion facility (IFF) and for a commercial power reactor. Comparing the two sets of goals shows that the IFF falls well short of commercial performance. It would only produce from one-sixth to one-third the heat generated by a commercial-scale reactor, and it would have considerably lower availability and shorter lifetime than a commercial plant. The IFF described by these parameters, therefore, could not be considered to be a "demonstration" power reactor.

[18]U.S. Department of Energy, Office of Energy Research, *Magnetic Fusion Program Plan, DOE/ER-0214,* February 1985, p. 6.

cept" experiment, costing up to tens of millions of dollars; such a device would explore the scaling properties of the concept and determine whether it offers the potential for extrapolation to a reactor.

If the results are promising, a "proof-of-principle" experiment would then be required to provide confidence that the concept could be scaled up to reactor-level conditions. The JET, TFTR, and JT-60 tokamaks are in this category. They are not themselves reactor-level devices, but they will enable decisions to be made about whether to proceed to the final stage of concept development: demonstration of reactor-level plasma conditions, including fueling, burn control, ash removal, and other functions necessary for reactor operation. No reactor-level devices have yet been built for any concept, including the tokamak.

Cost is very difficult to estimate for a future proof-of-principle or reactor-level device. The TPA report estimates costs ranging from 100 million to several hundred million dollars or more.[29] The costs of the existing JET, TFTR, and JT-60 devices range between $600 million and $950 million dollars,[30] but these devices perform many functions in addition to proof-of-principle for the tokamak concept. There is no reason to think that proof-of-principle devices for other confinement concepts would be as expensive. A reactor-scale device for a particular concept would have more stringent technical requirements than its proof-of-principle device and presumably would be more expensive. However, the cost of a reactor-level device depends on whether it would serve other functions such as the long-pulse burn, nuclear technology demonstration, or system integration functions discussed below.

In addition to generic device requirements for selected alternate confinement concepts, TPA also set a requirement for additional tokamaks to investigate features such as shaped plasmas (to follow up on the PBX-M and D III-D results discussed in the earlier sections "Advanced Tokamak," p. 59, and "Beta," p. 70), steady-state tokamak operation, and high-magnetic-field approaches to tokamak confinement.[31]

Properties of Burning Plasmas

Some of the most critical scientific issues yet to be resolved in the fusion program involve the behavior of ignited, or burning, plasmas. These issues include the effects of self-heating on confinement and the effects of energetic alpha particles on plasma stability, burn control, and fueling. Effects such as these, which existing experiments cannot yet address, can profoundly influence fusion's feasibility. TFTR may be able to provide some information about the effects of alpha particle generation if it attains near-breakeven conditions in D-T operation. However, TFTR does not have the capability to reach ignition and therefore will not be able to resolve burning plasma issues definitively. The European JET device should also have the capability to reach and perhaps exceed breakeven, but it too will not be able to resolve many of the burning plasma issues.

According to TPA, two different tasks are required to study burning plasma issues fully. One is a short-pulse ignition demonstration to create a self-sustaining fusion reaction. The second is a long-burn demonstration to maintain a self-sustaining fusion reaction long enough to study effects such as the evolution of the plasma under steady-state burn and the buildup of reaction products. These two tasks could be done in separate facilities or in the same facility.

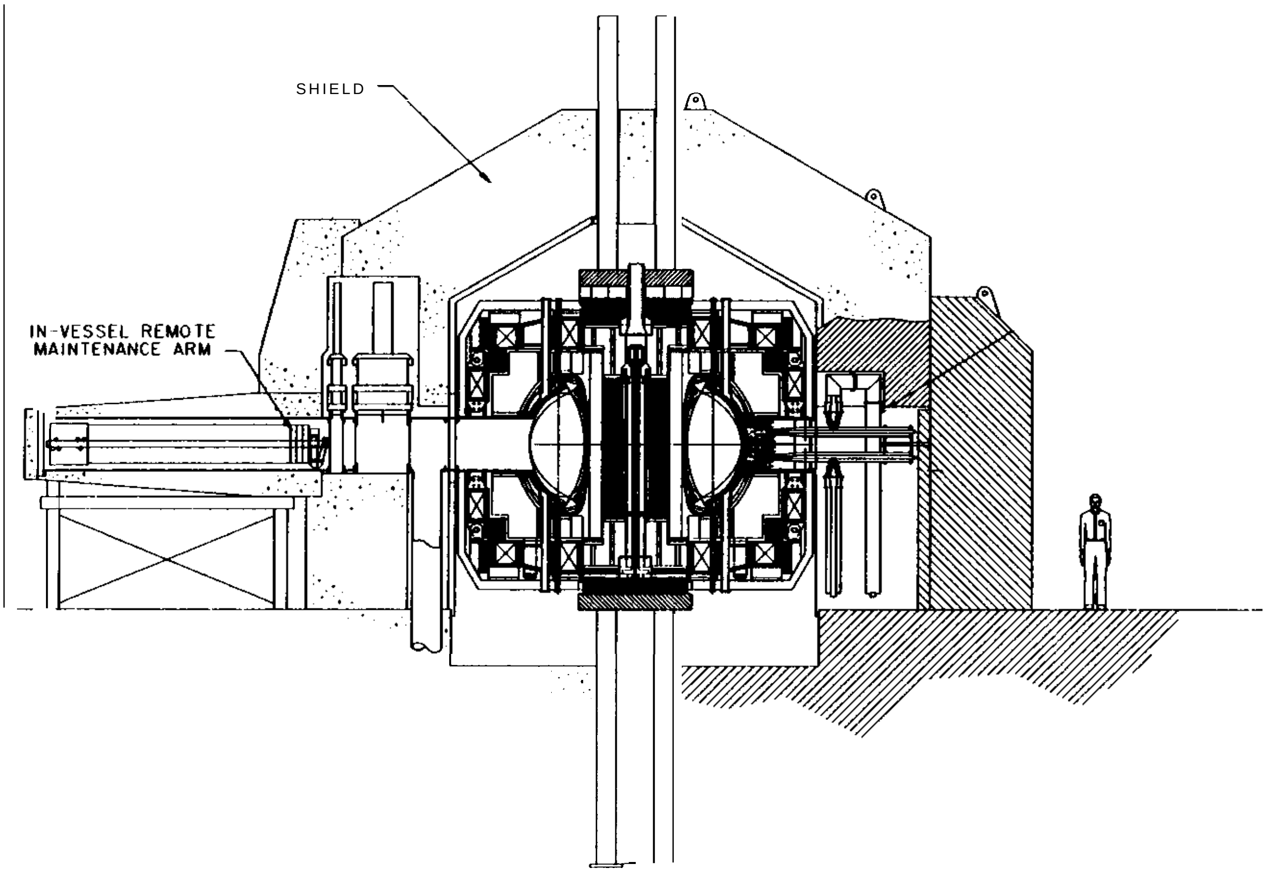
For fiscal year 1988, DOE has requested funds to start building a Compact Ignition Tokamak (CIT) to study short-pulse ignition issues (figure 4-15). This device, to be located immediately adjacent to TFTR at Princeton Plasma Physics Laboratory, is anticipated to cost about $360 million (including diagnostic equipment and associated R&D) and will take advantage of existing equipment at the site. Operation is scheduled to be-

29 Argonne National Laboratory, *Technical P/arming Activity: Final Report,* op. cit., table S. 12, p. 44 and table S. 16, p. 48.

30 U.S. Congress, *House* Committee on Science and Technology, Task Force on Science Policy, *Science Policy Study Background Report No. 4: World Inventory of "Big Science" Research Instruments and Facilities,* Serial DD, prepared by the Congressional Research Service, Library of Congress, 99th Cong., 2d sess., December 1986, pp. 111, 121, and 127.

31 Argonne National Laboratory, *Technical P/arming Activity: Final Report,* op. cit., p. 79.

Figure 4-15.—Preliminary CIT Design



SOURCE: Princeton Plasma Physics Laboratory, 1987.

gin in 1993, and annual operating cost is estimated at about $75 million. According to a review by a panel of the Magnetic Fusion Advisory Committee (M FAC)[32], CIT will be able to study most of the effects that generation of alpha particles might have in a fusion plasma.

Ever since TFTR was designed in the mid- 1970s, the design for a successor device has been a topic of active interest in the fusion community. As far back as 1977, proposals were made for compact, high-magnetic-field tokamak devices using high-performance copper magnets; this is the approach that was selected for CIT. Other proposals, which were ultimately not adopted, called for long-pulse tokamaks using superconducting magnets and costing well over over $1 billion. The CIT design is intended to focus on scientific aspects of the fusion process, and it will not necessarily form the engineering basis for future fusion reactors: CIT's copper magnets consume amounts of electricity that would be prohibitive for commercial purposes, they cannot operate for longer than a few seconds at a time without overheating, and their compact size does not provide enough room for a blanket to recover fusion energy or breed tritium. But CIT will have the ability to resolve critical physics uncertainties sooner and at lower cost than an experiment having more of

[32] The Magnetic Fusion Advisory Committee is a committee consisting of scientists and engineers from academia, national laboratories, and industry that advises DOE on technical matters concerning fusion research. An MFAC subpanel (Panel XIV, chaired by Dale Meade of Princeton Plasma Physics Laboratory) reviewed plans for CIT in the *Report on Assessment of Burning-Plasma Phenomena in a Compact Ignition Tokamak,* Feb. 10, 1986. The subpanel's report was reviewed by the full MFAC in *Magnetic Fusion Advisory Committee Report on Compact Ignition Experiments (Charge XIV),* submitted to Dr. Alvin Trivelpiece, Director, Off Ice of Energy Research, Department of Energy, Washington, DC, February 1986.

the engineering features that a reactor would require. Moreover, although specifics of the CIT design may not be applicable to future reactors, the overall approach of high-field, high-performance magnets in fact may be relevant if such magnets can be made with superconducting technology.

CIT is being designed through a national effort with wide-based technical support, and the project has been endorsed by MFAC as a "cost-effective means for resolving the technical issues of ignited tokamak plasmas."[33] MFAC determined that the "existing tokamak data base is adequate, with credible extrapolation, to proceed with the design of the CIT," and that by fiscal year 1988 "we should have acquired sufficient information from present large machines to support proceeding with the construction of the CIT."[34] MFAC also stated that CIT "should be part of a balanced overall fusion program," implying that it should not drain funds away from "other essential elements . . . in the DOE Magnetic Fusion Program Plan."[35]

Long-term burn issues, which cannot be addressed by CIT, will require another device in the future. Even if constructed solely to study physics issues, such a device would probably cost at least $1 billion. If other functions such as nuclear technology testing were incorporated, the cost would be even higher.

Fusion Materials

According to TPA, "the ultimate economics and acceptability of fusion energy, as with most other energy sources, will depend to a large extent on the limitations of materials for the various components."[36] Addressing the specific material issues identified earlier in the chapter requires facilities for testing and evaluating candidate materials. Some of this testing can be done by exposing materials to neutrons in fission reactors. Fission-generated neutrons, however, differ in energy and effects from fusion-generated neutrons; tests of materials in fission reactors have to be carefully arranged in order to provide meaningful data on fusion neutron effects.

Eventually, a high-intensity source of 14-million-electron-volt (14-MeV) neutrons will be required to evaluate the lifetime potential of most materials. To accelerate the effects of aging, the *test* facility must generate neutron irradiation levels significantly higher than those expected in a reactor. Such levels could be provided by a device such as a driven fusion reactor, which would generate fusion neutrons but consume more power than it generated. Such a device would be completely impractical as an energy source, but could bean effective method of generating high fusion neutron intensity over a small volume (1 O to 1,000 cubic centimeters).

TPA estimates that a materials irradiation test facility would cost from $150 million to $250 million and would take about 4 years to build. Materials testing would also require several additional facilities each costing $10 million or less.

Fusion Nuclear Technology

Fusion nuclear technologies are those involved with the recovery of energy from the fusion reaction and the breeding and recovery of tritium needed to replace the tritium consumed in the reaction. Most of the nuclear technology functions of a fusion reactor are incorporated in the first wall, blanket, and shield.

The first wall/blanket test program is currently at a very early stage. However, the characteristics of the required experiments have been defined in the FINESSE study .37 A number of experiments, each costing about $5 million, will be important for guiding the future of the program. Several larger experiments will be needed to follow upon the earlier ones; each of these will cost up to tens of millions of dollars to build and $1 million or more a year to operate .38

[33] *Magnetic Fusion Advisory Committee Report on Compact Ignition Experiments (Charge XIV),* letter from MFAC chair Fred L. Ribe to Dr. Alvin Trivelpiece, Director, Office of Energy Research, Department of Energy, Feb. 24, 1986.

[34] Ibid,

[35] Ibid.

[36] Argonne National Laboratory, *Technical Planning Activity: Final Report,* op. cit., p. 259,

[37] The FINESSE study is a 3-year study done to r DOE involving organizations and scientists from the United States, Europe and Japan. The study produced many publications; one example is 'A Study of the Issues and Experiments for Fusion Nuclear Technology," by M. A. Abdou, et al., *Fusion Technology 8,* November 1985.

[38] Argonne National Laboratory, *Technical Planning Activity: Final Report,* op. cit., table 4.13, pp. 234-236.

After individual facilities enable blanket concepts and components to be designed and the number of options to be reduced, a large-scale nuclear technology demonstration facility will be required to integrate these components and test them under fusion reactor conditions. Such a large-scale device must not only produce enough fusion power to provide a realistic environment for nuclear technology testing, but it must also incorporate development and construction of the individual nuclear technology systems themselves. If this device is also intended to address the physics issues associated with long-term fusion burns, it will become still more complex and expensive.

One possibility, identified by the FINESSE study and reviewed and adopted by TPA, is building a nuclear technology demonstration facility that does not simultaneously serve as the long-burn physics test facility. A device built solely to study nuclear technologies would need a source of fusion-generated 14-MeV neutrons, but this source would not need to be an ignited plasma. Such a device could test scaled-down versions of nuclear technology components, provided that these results could be applied with confidence to reactor-sized versions. TPA estimated that such a nuclear-technology-only device would cost about $1 billion and would take 5 to 6 years to build. TPA did not estimate operating costs for this device.

TPA identified as a second possibility an engineering test reactor (ETR) that would include both long-term burn physics and nuclear technology studies. Such a device would require an ignited or near-ignited plasma, making it big enough to accommodate full-size nuclear technology components. Both the more stringent physics requirements and the need for full-scale nuclear technology components would make an ETR much more expensive than a nuclear-technology-only facility. TPA estimated the cost to build an ETR at about $3 billion. It did not estimate operating costs but said that the experience with TFTR would suggest $150 million annually.[39]

If a nuclear-technology-only device is built instead of an ETR, an additional device would be required specifically to study long-term burn. This additional experiment would probably cost more than $1 billion. Further expense would come later when the tested nuclear technology systems were scaled to reactor-level and integrated with a reactor-sized plasma. Thus, the cost of an ETR cannot be compared only to the cost of a nuclear technology device plus a long-term burn device.

Although DOE recognizes the need for an ETR or equivalent, it has no plans to build one. The Japanese and European programs each have plans to design such a device independently, but neither has yet committed to its construction. The Soviet and the U.S. fusion programs, on the other hand, appear to prefer international collaboration on such a facility. The U.S. Government has proposed to the other major fusion programs that conceptual design of an international engineering test reactor, called the International Thermonuclear Experimental Reactor (ITER), be jointly undertaken. The U.S. proposal does not extend to multinational construction of such a device. However, at the conclusion of the conceptual design effort, the parties could decide whether they wanted to proceed with construction, either independently or jointly. Possible avenues of future international collaboration in fusion research are discussed in chapter 7.

Resource Requirements

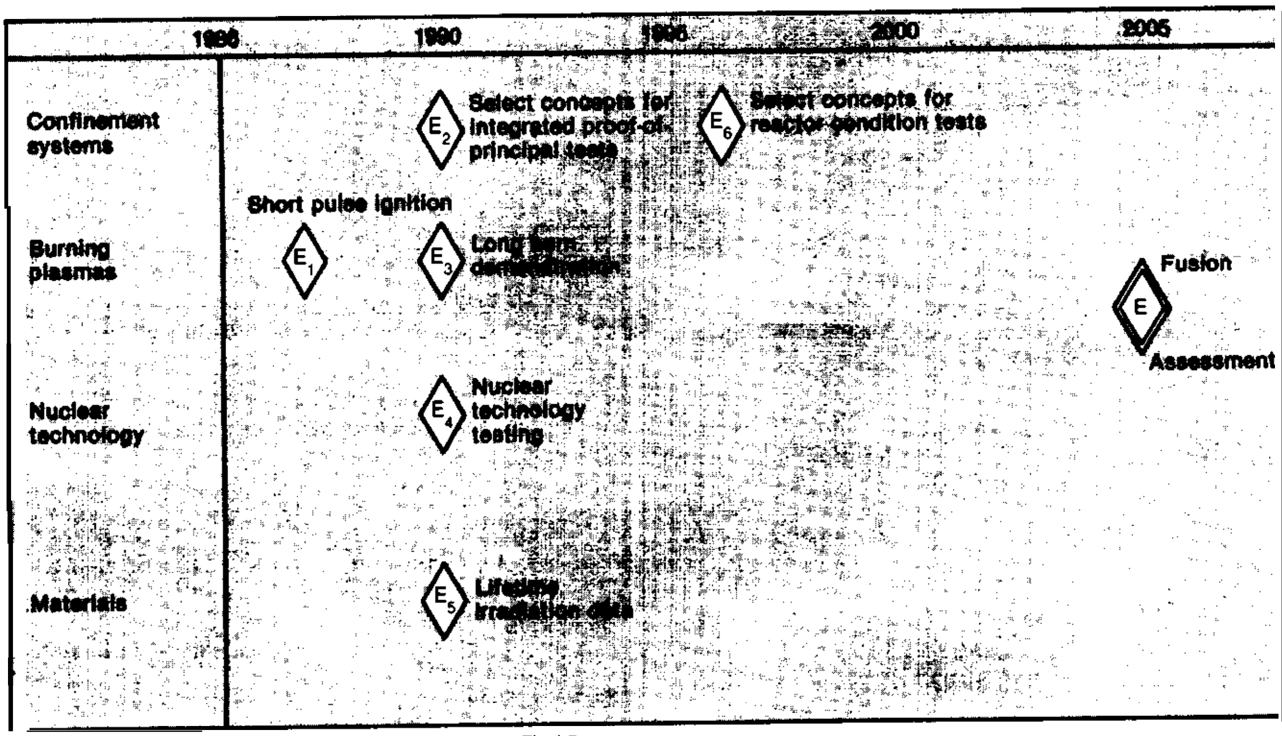
Schedule

TPA identified six major decision points that determine the course and schedule of future fusion research, leading up to the overall assessment of fusion's potential. In figure 4-16, each decision is pegged to a key technical issue and to a year.

[39]It is not clear how this estimate of $150 million is obtained; it is not computed merely by assuming the annual operating expense of a device to be a specific fraction of its capital cost. Were the estimate made in this manner, it would be considerably higher. The ratio of annual operating expense to capital expense for TFTR is about 15 percent, and that projected for CIT is about the same if the value of existing facilities at Princeton Plasma Physics Laboratory is included in CIT's capital cost. Applying this 15 percent ratio to an engineering test reactor would predict annual operating expenses of close to $500 million.

However, fusion scientists argue that there is no reason to believe the ratio of operating expense to capital expense to be the same for an ETR as it is for significantly smaller devices. They agree that the more a device costs to build, the more it will cost to operate. However, they maintain that there is no reason to expect capital and operating costs to increase at the same rate,

Figure 4-16.-Top Level Decision Points in the Magnetic Fusion Program



SOURCE: Argonne National Laboratory, Technical Planning Activity: *Final Report,* commissioned by the U.S. Department of Energy, Office of Fusion Energy, ANL/FPP-87-1, January 1987, figure S.8, p. 23.

TPA did not examine all the possible scenarios resulting from different timings and outcomes for these decisions but instead adopted a “reference scenario” believed to reflect current DOE planning. The years in figure 4-16 are taken from the reference scenario, which is shown in greater detail in figure 4-17.

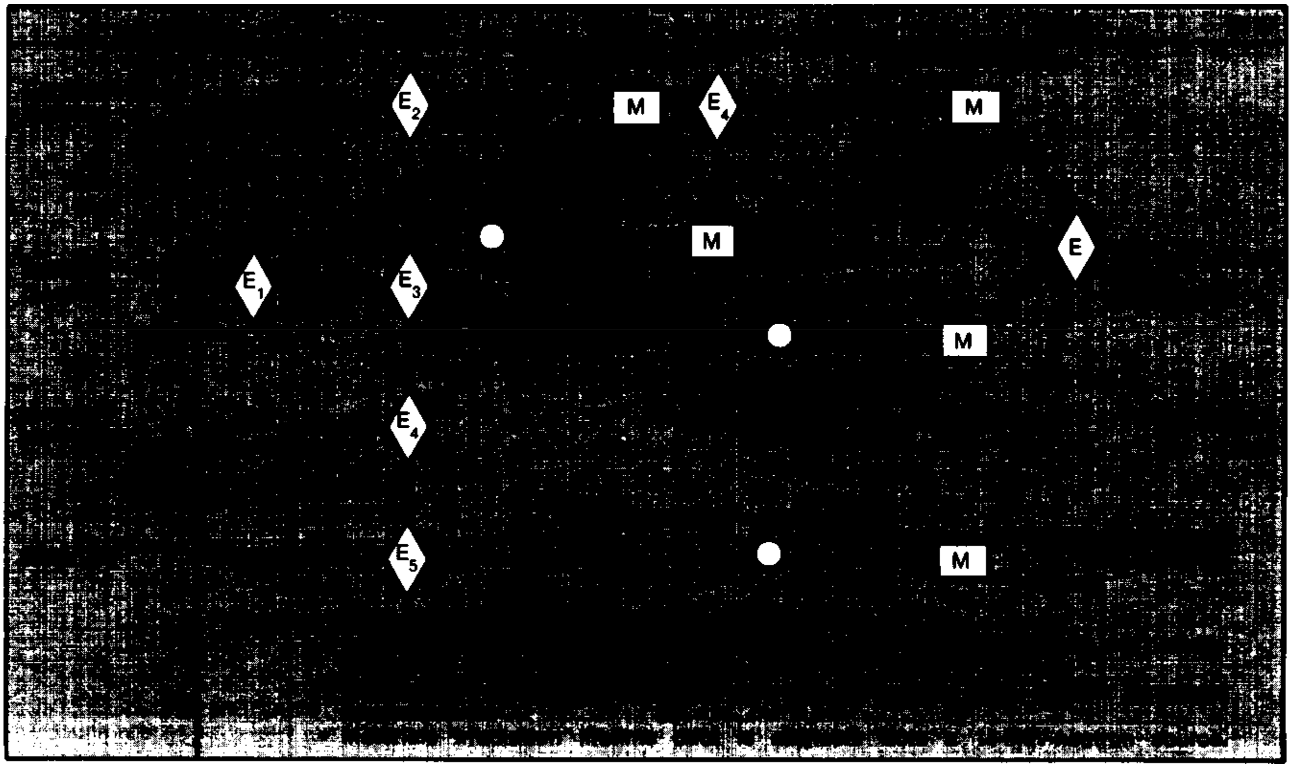
In the reference scenario, the decision to proceed with CIT as the short-term burn experiment is made in 1987. By about 1990, the decision is made to combine nuclear technology studies and long-term burn physics issues in a single engineering test reactor, possibly ITER. At about the same time, decisions are made concerning the nature of a materials testing facility and the selection of confinement concepts to be tested at the proof-of-principle scale. By 1997, certain alternate confinement concepts successfully showing proof-of-principle are selected for reactor-scale demonstration, and the overall assessment of fusion is targeted for 2005. TPA does not conclude that the reference scenario is fastest, cheapest, or most assured of success. Rather, it shows that an exhaustive process of technical review has not uncovered any inconsistencies.

cost

TPA estimates that the worldwide research required between 1987 and 2005 under the reference scenario will cost in the range of $20 billion. This cost does not include an integrated fusion facility, demonstration reactor, or any other facility constructed after the assessment of fusion in 2005. Total operating cost worldwide is judged to be relatively constant at about $800 million annually, and total yearly funding for fusion research increases to about $1.5 billion in the mid-1990s when construction cost is added. The total construction budget is estimated at about $6 billion, half of which is required for an engineering test reactor.

TPA acknowledges that the ground rules used for cost projection could have been applied more

F gure 4 7 Reference Scena o o he Magne c Fus on P ogram



abo Repo Depa
January 1987, figure S.1O, p. 27.

uniformly, that no iterative review of the cost estimates was undertaken, and that no effort was made to estimate the resources required for alternate research scenarios. Nevertheless, the report states that the information gathered is sufficient to present "a broad view of the resources required for a full assessment of the commercial potential of fusion."[40]

Some critics of TPA's cost estimates argue that, since TPA was not charged with designing, managing, and executing an actual research strategy—indeed, TPA was forbidden from performing such programmatic planning, which is strictly DOE's domain–the total cost represents a sum of "wish lists" rather than a realistic budget. According to these critics, without the requirement to conduct the politically difficult task of eliminating research that, although useful, may not be necessary, the estimated total cost is higher than an actual manager would spend when faced with fiscal constraints. Similarly, these critics complain that any study done by experts from a single field has an inherent bias towards overestimating that field's research requirements. Researchers in a field are presumed to have an interest in maintaining or increasing their field's funding, these critics argue, and the researchers would not have any incentive to prepare a study underestimating the cost of research if such a study might be used as justification for cutting the field's research budget.

Other critics, however, feel that any bias in TPA's estimated total is likely to be in the direction of underestimating the total rather than inflating it. Since each technical aspect of the study was analyzed by experts in the field, some degree of technical optimism is probably inherent at each stage; unanticipated difficulties would drive the cost of the research program above TPA's estimates. Moreover, these critics suggest,

[40] Argonne National *Laboratory, Technical Planning Activity: Final Report,* op. cit., p. 28.

it would *not* be in the collective interest of the fusion community to estimate a higher total cost than necessary, since continued support of the fusion program depends on perceptions that its benefits are worth its cost. Overestimating the total cost could threaten the program's support.

The process by which TPA estimated the total cost of future fusion research involved a wide degree of fusion community participation, and OTA can find no evidence that the estimate is flawed. However, OTA also recognizes that while the researchers in any technical field are the most qualified to estimate costs of experiments in that field, they are also the beneficiaries of support given to the field. Therefore, their estimates may be influenced by non-technical factors, although it is not clear whether the estimate would be too high or too low as a result.

Summary

Probability of Success

It seems likely that at the conclusion of the research program, fusion's technological feasibility—the ability to use fusion power to generate electricity—can be shown. The fusion program has made steady progress over the last 35 years on the key technical issues. It is still possible that fusion's scientific feasibility will be impossible to demonstrate, due to surprises in the behavior of a plasma that generates substantial amounts of fusion power. However, successfully attaining ignition in CIT will resolve most of the scientific uncertainties.

Most of the subsequent scientific and engineering challenges in designing and building a reactor have been identified. **Once scientific feasibility is established, a concerted and well-funded research effort should be able to develop a reactor that produces fusion power. However, it cannot yet be determined whether or not such a fusion reactor will be commercially attractive.** Characteristics and prospects of fusion as a commercial energy source are discussed in chapter 5.

Findings

Estimates of the annual worldwide funding required to evaluate fusion's potential early in the next century are several times today's annual U.S. fusion research budget. The estimated annual worldwide funding is, however, on the order of the amount now spent each year by all the world's major fusion programs put together. The funding estimates suggest three possibilities to U.S. policy makers for continuing the U.S. fusion research program:

1. With funding levels several times their present level and with a significant measure of technical success, the U.S. fusion program can decide on its own whether or not to begin the demonstration and commercialization of fusion power in the early part of the next century.
2. If the major world fusion programs can collaborate and plan their research efforts to complement each other and eliminate duplication, and if the effort has a significant measure of technical success, a collective assessment to proceed with fusion's development could be made early in the 21st century. In this case, only modest increases in funding would be required for each of the world's fusion programs, with the exact amount of the increases depending on how well the programs were able to avoid duplication of research.
3. If major international collaboration is not attained, and if the U.S. fusion budget is not increased to the point where the necessary research can be carried out domestically, the United States cannot assess fusion's potential until later in the next century.

These possibilities form the basis of the policy options discussed in greater detail in chapter 8,