Progress in Fusion Research, 1978-1982

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FUSION TECHNOLOGY INSTITUTE

UNIVERSITY OF WISCONSIN

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Section 1

INTRODUCTION

1.1 PURPOSE OF THIS STUDY

The fields of plasma physics and fusion technology have been moving at an extremely rapid pace over the past few years. Scientists and engineers have continuously pushed back the frontiers of science on a broad front and they now are within sight of the first breakeven experiments in both the magnetic and inertial confinement approaches. For those who have labored in this field, it is often difficult to "see the forest for the trees" and therefore it is necessary to periodically take stock of where the field has been and where it is going. The latter question is very difficult to answer even in more stable times, but considering the present (1983) uncertainty in financial resources for all governmental supported research it would be very risky indeed to speculate on the future of fusion research. However, the past efforts can be documented with some degree of accuracy and it is the purpose of this report to give a perspective on the last 5 years (1978, 1979, 1980, 1981 and 1982) in fusion research.

The past 5 years were selected for three reasons. First, because it allows one to focus on specific recent events and achievements rather than to cover all aspects of the history of fusion which have been so aptly documented in previous books (1-3). Second, it is a period which is still quite vivid in the minds of most of the readers of this report and therefore may illustrate how far the field has come in such a short time. Third, the total world level of effort in fusion has doubled in the last 5 years and this period can be characterized as a transition between the purely scientific experiments of the mid 1970's to the large scale physics and technology work of the early 1980's.

In order to more firmly establish the time frame of our subsequent discussions, it is worthwhile to restate some of the major news stories which appeared in 1977, the year before our review period begins.

* *Jimmy Carter is Sworn in as the 39th President of the United States.*
* *Trans-Alaskan Pipeline Opens.*
* *Menachem Begin Becomes the 6th Prime Minister of Israel.*
* *United States Department of Energy is Created.*

* *Panama Canal Treaties Signed by U.S. and Panamanian Heads of State.*

The transition from President Ford to President Carter and the shift from ERDA to DOE are certainly events that can act as convenient time markers for the beginning of our period of study.

The outline of the rest of this report is as follows. First a brief overview of fusion physics is included for those not intimately familiar with the field. Next the report tries to put into perspective the level of effort in fusion research (as indicated by identifiable budgets) for both the U.S. and the rest of the countries around the world. Unfortunately, detailed numbers for the U.S.S.R. are unavailable at this time. Next the progress in magnetic fusion is addressed by first presenting an up-to-date listing of major facilities around the world followed by a description of the progress made on the major issues over the past 5 years in both physics and technology. This is followed by a listing, and brief description, of the major reactor studies (for both the near-term and the commercial class) that have been conducted in the reference period. Appropriate observations on the trends in the field are also given. Because of the desire to limit the total length of this document to approximately 200 pages or less only the major parameters of such devices are included and the reader is referred to extensive review articles and detailed reports for more information.

The Inertial Confinement Fusion (ICF) program is then examined mainly from the viewpoint of the U.S. because of a general lack of information from other major countries doing research in ICF. The smaller, but significant ICF programs in Japan and Europe are briefly described within the limits of information that is available. An updated list of laser and ion beam facilities is presented and followed by a brief discussion of the currently available information on the interaction of these forms of radiation with targets. The last section in this chapter includes a brief look at the ICF fusion reactor studies completed in the last 5 years.

An attempt was made to condense the voluminous literature which has appeared in the Fission-Fusion Hybrid design field over the past 5 years. This has proved to be a difficult task but it does reveal some definite trends toward the suppressed fission concept. The work in synthetic fuels and advanced fuels in the reference time period has also shown that a significant narrowing of the possible options has been achieved.

There have been two major efforts in the U.S. with respect to Safety (coordinated by EG&G Idaho) and Environmental Impact Assessments (ORNL) in the recent past. Both of these programs are reviewed with respect to clarifying our view of fusion as a "safe and clean" power source.

The past 5 years have seen intense political activity in the area of fusion and this has also been the case for scientific review panels. There have been at least 5 major reviews in the U.S. and one in Europe. The results of these review panels are compared in the last section and a brief description of the Magnetic Fusion Act of 1980 is also presented.

This report is meant to be an overview and as such it does not contain a detailed list of references such as that which would be present in a normal journal article. An attempt was made to reference general reports and review articles which might be helpful to the reader and these are listed at the back of each subchapter. Finally, since the fusion community tends to make liberal use of acronyms, a Glossary is included at the back of the report for those who might be confused by the numerous permutations of the alphabet.

REFERENCES FOR SECTION 1.1

1. A.S. Bishop, "Project Sherwood: The U.S. Program in Controlled Fusion," Addison-Wesley, Reading, MA, 1958.
2. D. Willson, "A European Experiment," Adam Hilger Ltd., Bristol England, 1981.
3. J.L. Bromberg, "Fusion, Science, Politics, and the Invention of the New Energy Source," MIT Press, Cambridge, MA, 1982.

1.2 INTRODUCTION TO FUSION

1.2.1 Fusion Reactions

The basic goal of the fusion program is to utilize nuclear fusion reactions to produce useful energy, especially electricity. Some of the basic nuclear reactions being considered are listed in Table 1.2-1 and their cross-sections (a measure of the reaction probability) are shown in Fig. 1.2-1. Because the fusion cross-sections are significant only at high energies (tens to hundreds of keV corresponding to 100's to 1000's of millions of degrees), the fusion reactions will occur in the plasma state of matter, which consists of ionized atoms (ions) and free electrons. This state of matter is relatively uncommon on earth, but is the dominant state of matter in the universe. The stars and much of interstellar space are composed of plasma.

The power released by fusion reactions is given by

$$\frac{\text{power released}}{\text{volume}} = n_1 n_2 \langle \sigma v \rangle \, Q_f \ ,$$

where n_1 is the density of species number 1 which is reacting with species number 2 (density n_2), $\langle \sigma v \rangle$ is the reaction cross-section (see Fig. 1.2-2) averaged over the velocity distribution of the reacting particles, and Q_f is the energy released per fusion event. Consequently, a high particle density, n_1 or n_2, is desired to have a high fusion power density. Because of the high

TABLE 1.2-1

Selected Fusion Reactions

Reaction	Energy Release, MeV
$D + T \rightarrow {}^{4}He + n$	17.6
$D + D \rightarrow T + p$	4.0
$D + D \rightarrow {}^{3}He + n$	3.3
$D + {}^{3}He \rightarrow {}^{4}He + p$	18.3
$T + T \rightarrow {}^{4}He + 2n$	11.3
$D + {}^{6}Li \rightarrow {}^{4}He + {}^{4}He$	22.4
$D + {}^{6}Li \rightarrow {}^{7}Be + p$	3.4
$D + {}^{6}Li \rightarrow {}^{7}Li + p$	5.0

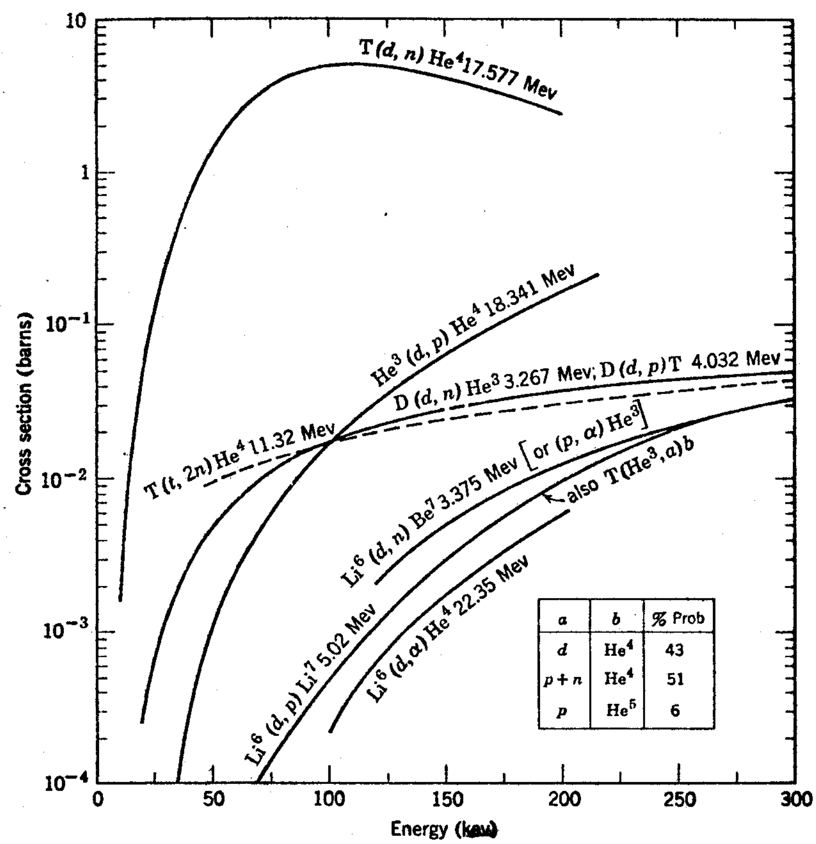


Fig. 1.2-1. Fusion cross sections.
Note that 1 keV equals roughly 10,000,000 °K.

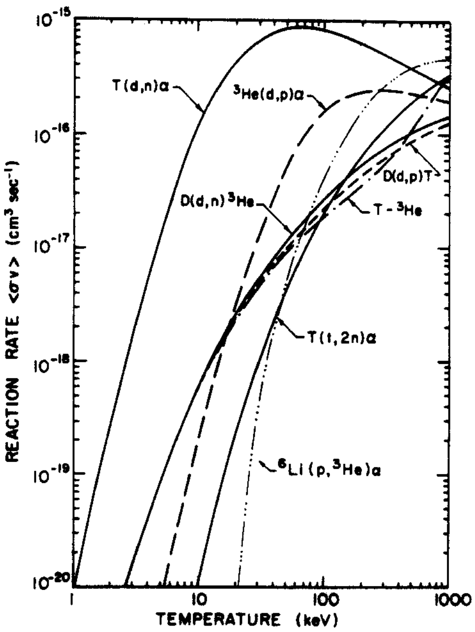


Fig.1.2-2. The Maxwellian-averaged reaction rate $\langle\sigma v\rangle$ as a function of ion temperature for the main fusion-reaction cycles.

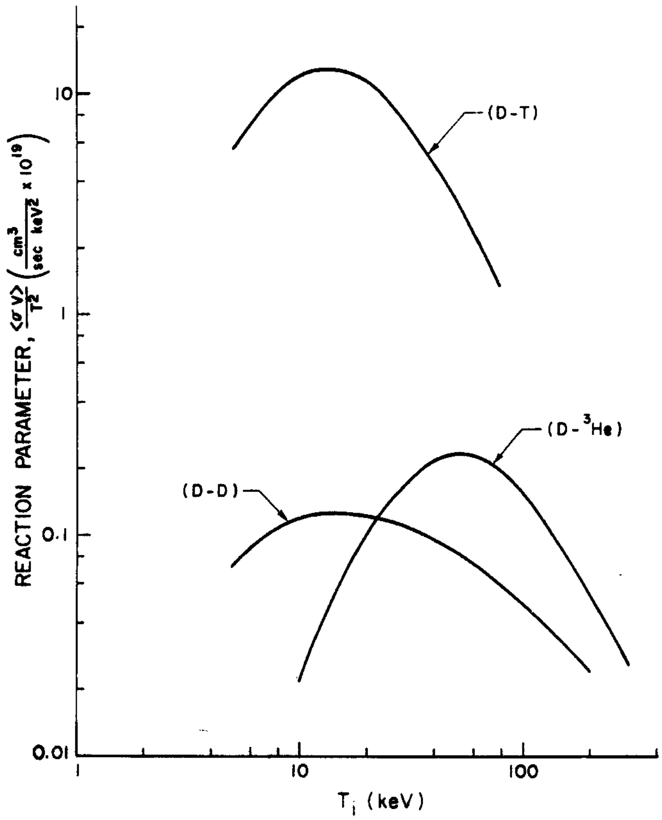


Fig. 1.2-3. The reaction parameter $\langle\sigma v\rangle/T_i^2$ as a function of ion temperature for several fusion fuel cycles.

particle energy, or equivalently a high temperature, this implies a high pressure of the plasma. In a pressure limited system, such as in magnetic confinement, the optimum temperature, T, is that at which $\langle\sigma v\rangle/T^2$, and not $\langle\sigma v\rangle$, is maximum. For the DT (deuterium-tritium) reaction this occurs at 13 keV (1 keV = 1.16 x 10^7 °C), as shown in Fig. 1.2-3. For other reasons, it is often desirable to have the temperature somewhat higher. DT fusion reactors are often designed for a plasma temperature of ~ 10-30 keV.

Since the DT reaction has the highest reaction rate and occurs at the lowest plasma temperature, it is considered to be the most likely fuel for early fusion reactors. Its primary disadvantage is that it releases an energetic neutron (14.1 MeV) which escapes the plasma, causes damage to the reactor structure and induces radioactivity by nuclear transmutations. Since 80% of the energy released is carried by the neutrons, it is important to capture the neutrons in a blanket surrounding the plasma. This blanket converts the kinetic energy of the neutrons to thermal energy which is then available for driving a turbine as in any other thermal power cycle. Nuclear reactions in the blanket also produce energy, so that the total thermal output can be considerably greater than the yield from the fusion reactions. The other particle produced by the DT fusion reaction is a 3.5 MeV alpha particle which reacts strongly with the plasma and keeps it hot.

A disadvantage of the DT reaction is that tritium is not found in nature. It is also a weak beta emitter with a half-life of 12.3 years. Consequently, it must be bred in the fusion reactor. This can be done using neutron capture in lithium; the two relevant reactions are

$$n + {}^{6}Li \rightarrow {}^{4}He + T + 4.86 \text{ MeV}$$

$$n + {}^{7}Li \rightarrow {}^{4}He + T + n' - 2.87 \text{ MeV} \quad .$$

The first reaction yields additional energy; the second is endothermic and therefore occurs only with energetic neutrons. It yields a lower energy neutron which can then react with ^{6}Li to produce another tritium nucleus. Consequently one can get tritium breeding ratios above unity using natural lithium. The natural abundances are 7.4% for ^{6}Li and 92.6% for ^{7}Li. A schematic of the total DT fusion fuel cycle is shown in Fig. 1.2-4. It should be noted that deuterium is found in all water sources at a concentration of one part in every 6700 parts of hydrogen. Therefore there is an enormous amount of D in the world, $\sim 10^{13}$ tonnes.

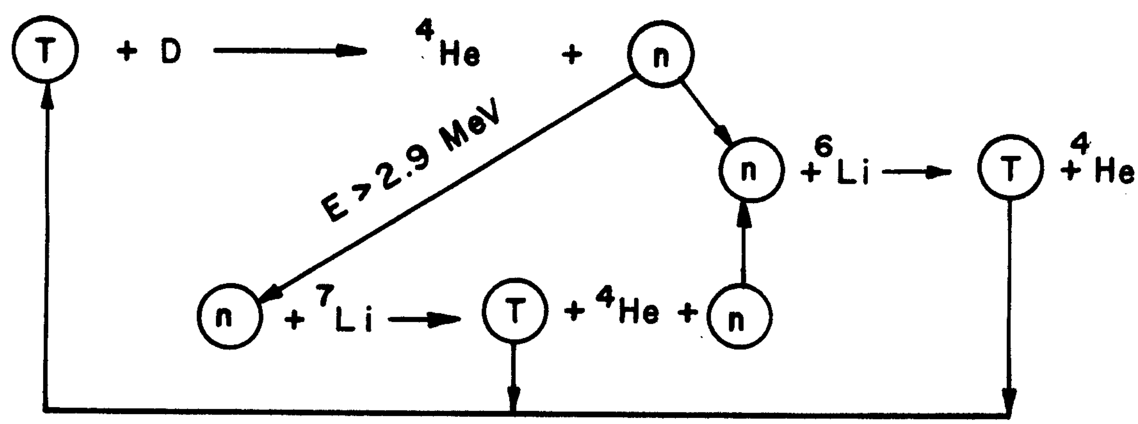


Fig. 1.2-4. The D-T Fuel Cycle.

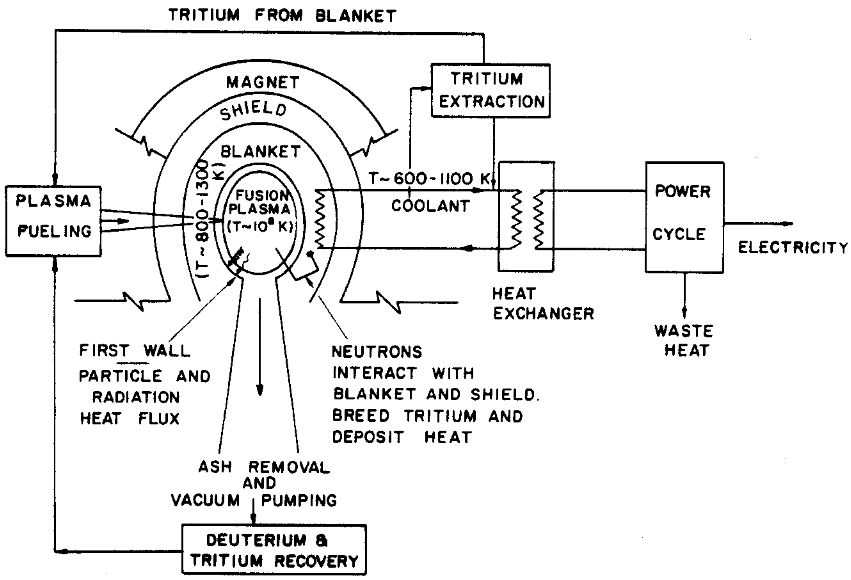


Fig. 1.2-5. A schematic diagram of components and material flows in a magnetic fusion reactor operating in the D-T fusion cycle.

1.2.2 Elements of a DT Fusion Reactor

Figure 1.2-5 illustrates the basic elements of a DT fusion reactor. It consists of a region containing the fusion plasma surrounded by a blanket. The neutrons produced in the fusion reactions in the plasma enter the blanket where they react with lithium to produce tritium. The lithium need not be elemental, but could be a lithium containing compound or an alloy, e.g. Li_2O or a LiPb alloy. The blanket is heated by both the kinetic energy of the neutron and also by nuclear reactions in the blanket. The blanket coolant transfers the heat to a thermal cycle through a heat exchanger. The lithium-containing breeding compound is processed to remove the tritium, which is then reinjected into the plasma. A heating system is also required to ignite the plasma and initiate the fusion process. Some conceptual designs of fusion reactors are discussed in more detail in Sections 2.3 and 3.5. Alternative applications and the possibility of other, more advanced fuels are discussed in Chapter 4 while safety and environmental issues are discussed in Chapter 5.

1.2.3 Plasma Confinement

The region containing the reacting ions (the plasma) is very hot and has to be kept away from material walls or else the plasma would be cooled by heat transfer to the walls. Two approaches of confining the plasma have been developed: magnetic and inertial. In magnetic confinement, a magnetic field is used to hold the plasma. This takes advantage of the electrical charge of the ions and electrons comprising the plasma. In inertial confinement, a small pellet containing deuterium and tritium is heated to fusion conditions in a very short time. Because of the finite inertia of the plasma, it holds together for a long enough time to yield net fusion energy before it expands and cools.

A figure-of-merit for the required plasma confinement time is the Lawson criterion. If the fusion energy produced in a confinement time τ is converted to electricity at some thermal efficiency η, then the resulting energy should be greater than that required to heat the plasma to the temperature required for net power production. The criterion resulting from this is that

$$n\tau \gtrsim 2 \times 10^{14} \text{ s/cm}^3$$

at a temperature of about 20 keV for a thermal efficiency of 30%. Here n is the total ion density and τ is the confinement time. This figure-of-merit, originally due to J.D. Lawson, gives a rough indication of the required density, temperature, and confinement time in order to have net power from fusion.

In the magnetic confinement approach, the density is limited by the available magnetic field strength to about 10^{14} cm^{-3}. Hence a confinement time of a few seconds is required. This confinement time is the mean residence time of the particles in the plasma; because of refueling, the plasma itself is assumed to last a much longer time. In the inertial confinement approach, one attempts to achieve a density on the order of 10^{25}-10^{26} cm^{-3}; hence τ can be as short as 10^{-11} s. In this case, τ is the time for the superdense plasma to blow itself apart.

1.2.3.1 Magnetic Confinement.

The magnetic field causes the charged particles to gyrate in helices about the magnetic field lines; this is the principle of the cyclotron. This constrains plasma motion across a magnetic field, but not along it. There are two basic approaches to dealing with the motion along the magnetic field. The toroidal approach is to avoid having the magnetic fields lines intersect walls, but instead use a magnetic field configuration shaped like a doughnut (i.e., a torus). Consequently particle motion along the field lines stays within the plasma and does not strike a material surface. The tokamak, stellarator, and EBT confinement schemes are based upon this concept. The second major approach is the magnetic mirror, where field lines strike the end walls, but plasma motion along the field is inhibited by a modulation of the field strength.

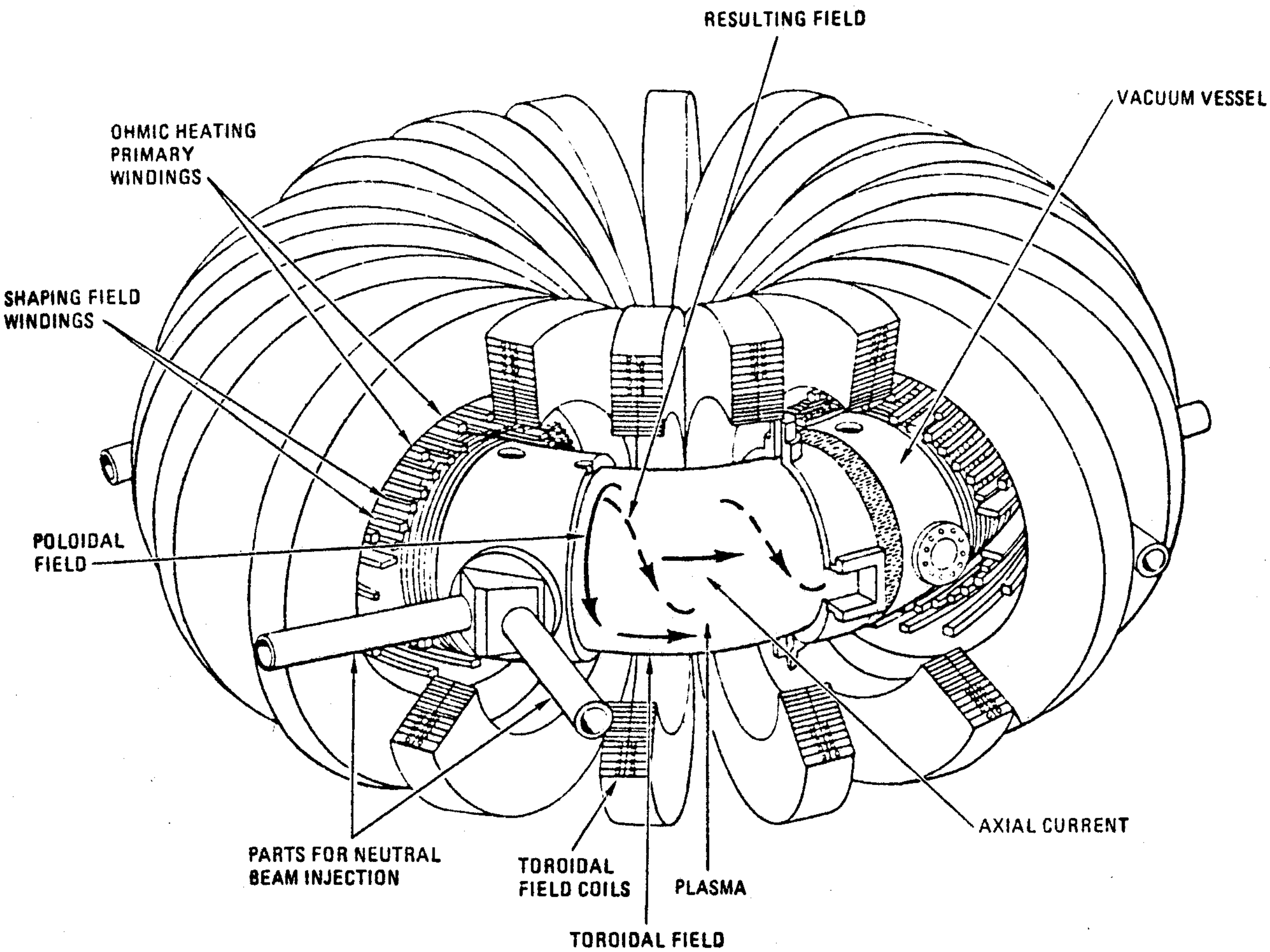


Fig. 1.2-6. Basic components of the tokamak.

The tokamak utilizes a strong toroidal magnetic field; an electrical current flowing in the plasma produces a second magnetic field (called the poloidal magnetic field), as shown in Fig. 1.2-6. This second magnetic field is necessary for plasma equilibrium and stability; otherwise, the plasma would escape and strike the container walls in a very short time, typically microseconds. The plasma current also heats the plasma through ohmic heating, but this is insufficient to achieve fusion ignition.

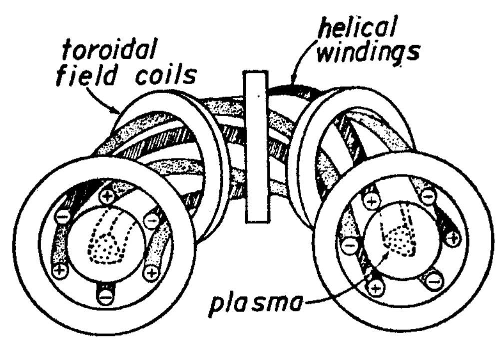


Fig. 1.2-7. Shape of coils and plasma for an ℓ = 3 stellarator.

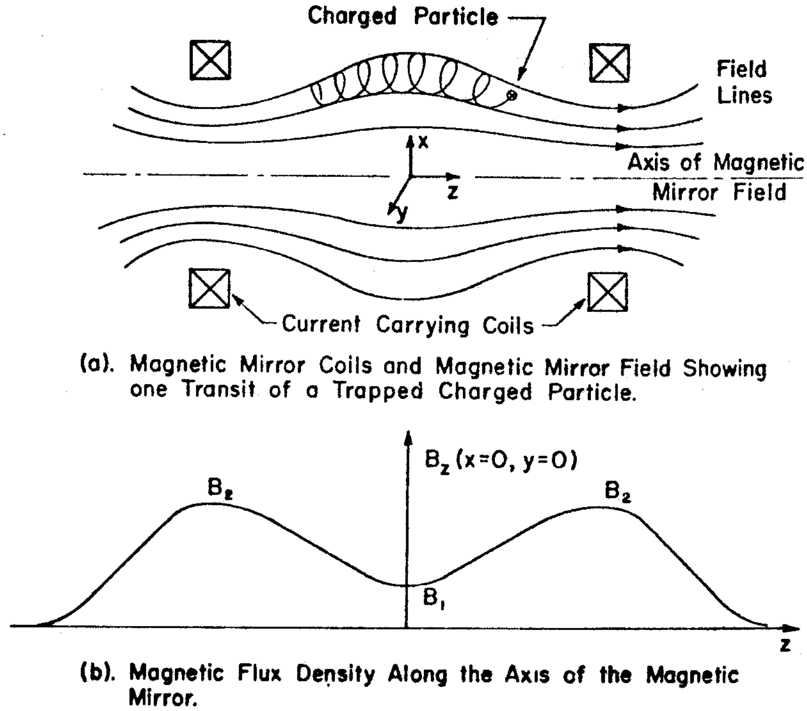


Fig. 1.2-8. Schematic of a magnetic-mirror configuration used for the confinement of plasma

Hence, supplemental heating, usually in the form of neutral beams or radio frequency waves, is required.

The stellarator (see Fig. 1.2-7) is similar to the tokamak, but the poloidal magnetic field is provided by helical coils outside the plasma, rather than by a current in the plasma.

A disadvantage of the toroidal approach is that the bending of the magnetic field into a torus introduces curvature of the magnetic field. This curvature can cause plasma instabilities if it is directed towards the plasma; in this case it is called bad curvature. One can arrange the average curvature to be good in torodial confinement schemes, but there will always be regions of locally bad curvature.

The mirror approach (see Fig. 1.2-8) is based on the magnetic mirror effect. Particles whose velocity vector at the mirror midplane lie outside a "loss-cone" are confined while those whose velocity vector is within the "loss cone" are lost. The angle of the loss-cone depends on the ratio of magnetic field strength at the peak to that at the midplane. Collisions can cause the

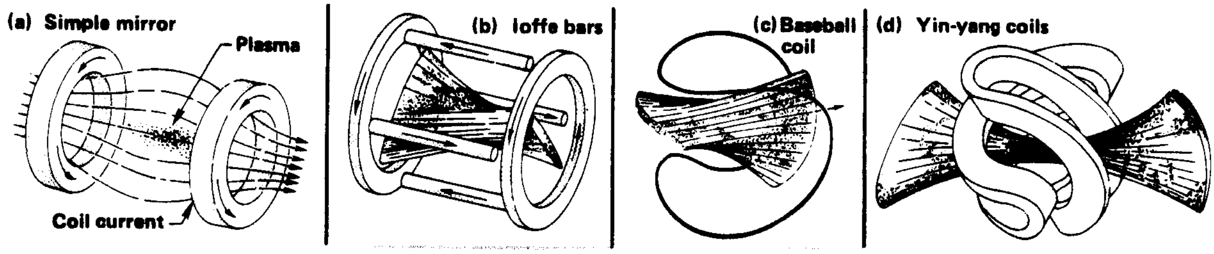


Fig. 1.2-9. (a) Simple magnetic-mirror cell with axisymmetric field concentrated at the ends (the mirrors) to reflect ions back toward the center.
(b) Current in Ioffe bars imposes a transverse multipole field on the simple-mirror field resulting in a magnetic pocket (minimum-B) at the center.
(c) The single baseball coil produces the same mimimum-B field configuration more efficiently than the Ioffe-bar system. (d) The two nested yin-yang coils produce the same minimum-B field but provide greater flexibility by permitting different currents in the two coils, thus different strengths of magnetic mirrors.

velocity vector of a particle to scatter into the loss-cone. This leads to the immediate loss of the particle which can be sufficiently rapid that a simple mirror probably cannot produce enough net fusion power to be a viable reactor. A simple mirror, such as shown in Fig. 1.2-8, is unstable because of the bad curvature of the magnetic field between the coils. It can be made stable, however, by the addition of quadrupole currents parallel to the axis (Fig. 1.2-9b). This makes a minimum-B configuration with good curvature between the mirror peaks. The quadrupole currents can be combined with the circular currents in a variety of ways as shown in Fig. 1.2-9. One configuration which produces a good magnetic geometry for mirrors is the so-called yin-yang magnet.

A magnetic mirror plasma develops a positive electrostatic potential because otherwise the electrons would be lost faster than the ions. A recent invention is the tandem mirror concept which utilizes this potential to electrostatically confine the plasma in a solenoid. In this concept there is a magnetic mirror, called the "end plug", at each end of a uniform central cell as shown in Fig. 1.2-10. The plasma in the end plugs is sustained by ionization of an injected neutral beam. The fusion power comes primarily from the central cell which can be made as long as necessary to achieve the desired fusion power output. The end plugs can also be in a minimum-B configuration and thereby provide stability for the entire plasma. In the late 1970's the tandem mirror replaced the single mirror as the main mirror confinement scheme.

1.2.3.2 Inertial. We have already seen that the basic requirements of fusion involve heating a plasma fuel (e.g., DT) to thermonuclear temperatures (~ 10 keV) and then confining this high temperature fuel for a sufficiently long time that it produces more fusion energy than the energy invested in its heating and confinement. As we have also noted, the traditional approach to fusion has been to attempt to confine a very low density plasma fuel (at $n \sim 10^{14}$ cm^{-3}) for a relatively long time ($\tau \sim 1$ sec) in a suitably shaped magnetic field.

The inertial confinement fusion scheme takes the opposite approach. The aim is to heat a dense fuel to thermonuclear temperatures extremely rapidly so that a significant amount of thermonuclear energy will be produced before the fuel blows itself apart. To see how this works consider a small solid pellet of DT fuel of radius 1 mm. The "disassembly" time t_d required for the heated pellet to blow apart is roughly the time it takes a shock wave to traverse the pellet. Since the speed of sound in a 10 keV DT plasma is roughly 10^8 cm/sec, the disassembly time is $t_d \sim 0.1/10^8 = 10^{-9} = 1$ ns. Hence to satisfy the Lawson criterion in $\tau \sim 10^{14}$ cm^{-3} sec, we require $n = 10^{23}$ cm^{-3} which is roughly solid state density.

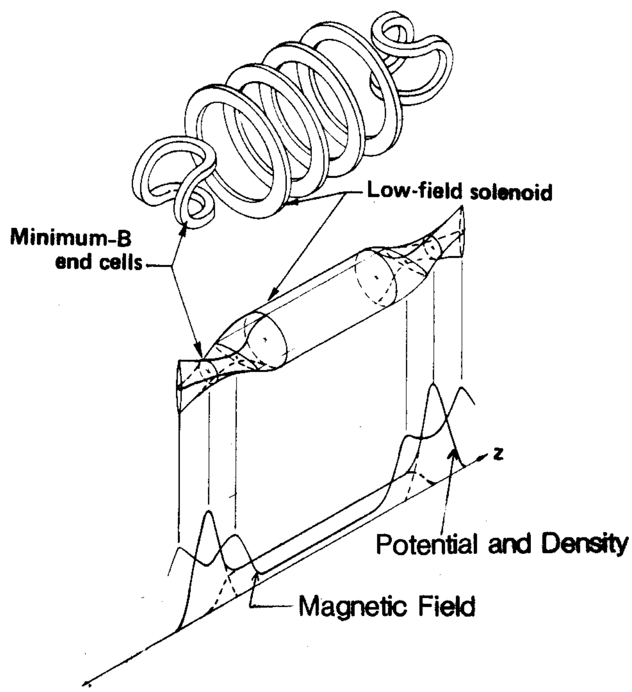


Fig. 1.2-10. The basic plasma shape magnetic fields, and electrostatic potential of a tandem mirror are shown above.

A more careful analysis shows that the disassembly time is ~ 10^{-11} seconds and the fuel density must be 10^{25}-10^{26} cm^{-3}. This is about 1000 times greater than liquid density.

Therefore we must heat and compress a small, liquid density DT pellet to thermonuclear temperature and high density to achieve net energy gain. Actually the energy required is not too great (roughly 1 MJ or about 0.28 kWh), about the energy consumption in one evening's operation of a television set. But when this energy is delivered in 10^{-9} second, it corresponds to a power level of $10^6/10^{-9} = 10^{15}$ W! This is roughly 1000 times greater than the current generating capacity of all the power plants in the United States.

Hence we are faced with the task of generating enormous powers and focusing these down onto a tiny pellet. Lasers are very good at doing this but considerable improvement in their efficiency of operation is needed. A schematic of the most powerful laser ever built, SHIVA, located at the Lawrence Livermore National Laboratory is shown in Fig. 1.2-11.

In order to examine this pellet heating process in a little more detail one can follow through the sequence of physical effects that are expected to occur as shown in Fig. 1.2-12.

1. The process is initiated by irradiating a 1-mm sphere of liquid or solid D-T fuel uniformly about its surface with intense laser light (which will reach a peak power intensity of 10^{17} W/cm^2).

2. The outer surface of the pellet heats, ionizes, and ablates off to surround the pellet in a cloud or "corona" of low density plasma, characterized by electron densities $n_e \sim 10^{19}$ to 10^{22} cm^{-3}.

3. The electrons in the corona continue to absorb more energy from the incident laser beams, but now the beam can only penetrate into the critical density where the plasma frequency

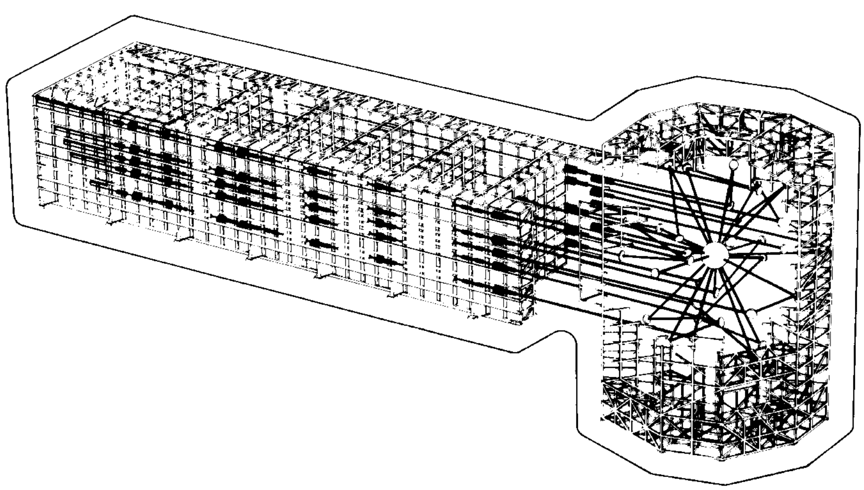


Fig. 1.2-11. LLNL SHIVA Laser Facility

equals the light frequency. This critical density surface occurs at 10^{21} cm^{-3} for Nd laser light at 1.06 μm and 10^{19} cm^{-3} for CO_2 laser light at 10.6 μm.

4. The energy deposited by the laser at the critical surface is then transported into the surface of the pellet by processes such as electron thermal conduction. This energy continues to heat the pellet surface, driving the ablation process and producing high pressures.

5. As the ablation of the surface continues, a shock front is formed that converges (implodes) inward, pushing cold D-T fuel ahead of it to higher and higher densities along the "Fermi degenerate adiabat". Here it is important to compress the pellet fuel isentropically (without appreciable heating) in an effort to bring it to very high density while still leaving it relatively cold.

6. When the shock fronts converge at the center of the highly compressed pellet core, they shock heat a small region at the center of the compressed core to thermonuclear ignition temperatures (2 to 5 keV). If $\rho R > 0.5$ g/cm^2 (where ρ is the density of the compressed core and R is the radius of the same region) alpha particle self-heating will occur (that is, the alpha particles will deposit most of their energy in the dense fuel before they escape), and the intense spark at the center of the com pressed core will rapidly heat to optimum burn temperatures of 20 to 100 keV.

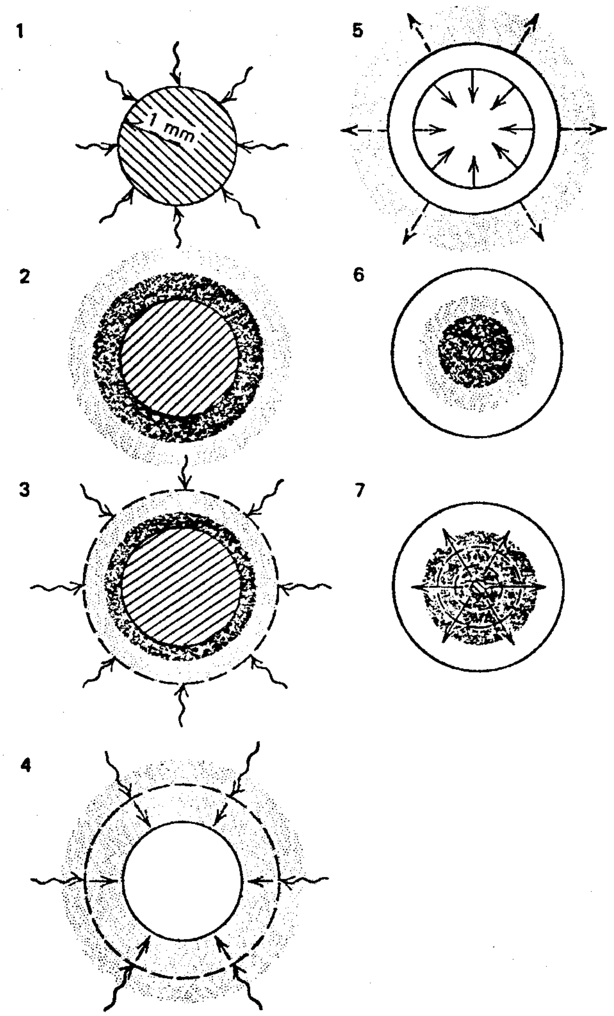


Fig. 1.2-12. The stages in the implosion of an inertial confinement fusion target: (1) irradiation by driver beams, (2) formation of plasma atmosphere, (3) driver beam absorption in atmosphere, (4) ablation driven imploding shocks, (5) compression of fuel core, (6) ignition, (7) burn propagation.

7. As the central spark burns, some alpha particles are deposited in a adjacent cold fuel, bringing it to ignition temperatures. The tendency of the burning fuel spark to become more transparent to alphas as it heats up enhances this process. The adjacent fuel material layer burns, producing further self-heating in cold fuel material and producing a thermonuclear burn wave that propagates outward consuming the dense pellet core.

To achieve the effect described above, a high power energy source must be used. In recent years, electron and ion beams as well as lasers have been investigated for this purpose.

1.3 HISTORY OF FUSION RESEARCH

1.3.1 Chronology

Scientists have been aware that thermonuclear reactions can release tremendous amounts of energy since 1928 (1). However, it took more than twenty years before fusion energy was released here on earth, initially in an uncontrolled manner during the first hydrogen bomb tests in 1952 (2), and later in a controlled fashion at several laboratories around the world. A brief account of the introduction of various fusion concepts by the fusion community is listed below and depicted in Fig. 1.3-1.

Date	Event
1946	Magnetic Pinch Reactor Concept Proposed
1951	Stellarator Concept Proposed
1952	Mirror Concept Proposed Project Sherwood Initiated (Classified)
1953	Tokamak Concept Proposed
1956	Declassification of Selected Parts of Fusion Research
1958	First Results Reported from Mirror Devices Major Release of Information on Fusion at Second Conf. on Peaceful Uses of Atomic Energy - Geneva
1961	Minimum "B" Mirror Configuration Proposed First Classified Proposal for Laser Fusion
1965	First Results from Tokamak Openly Reported
1968	Significant Results from Tokamak Reported First Neutrons Detected in Laser Fusion Experiments

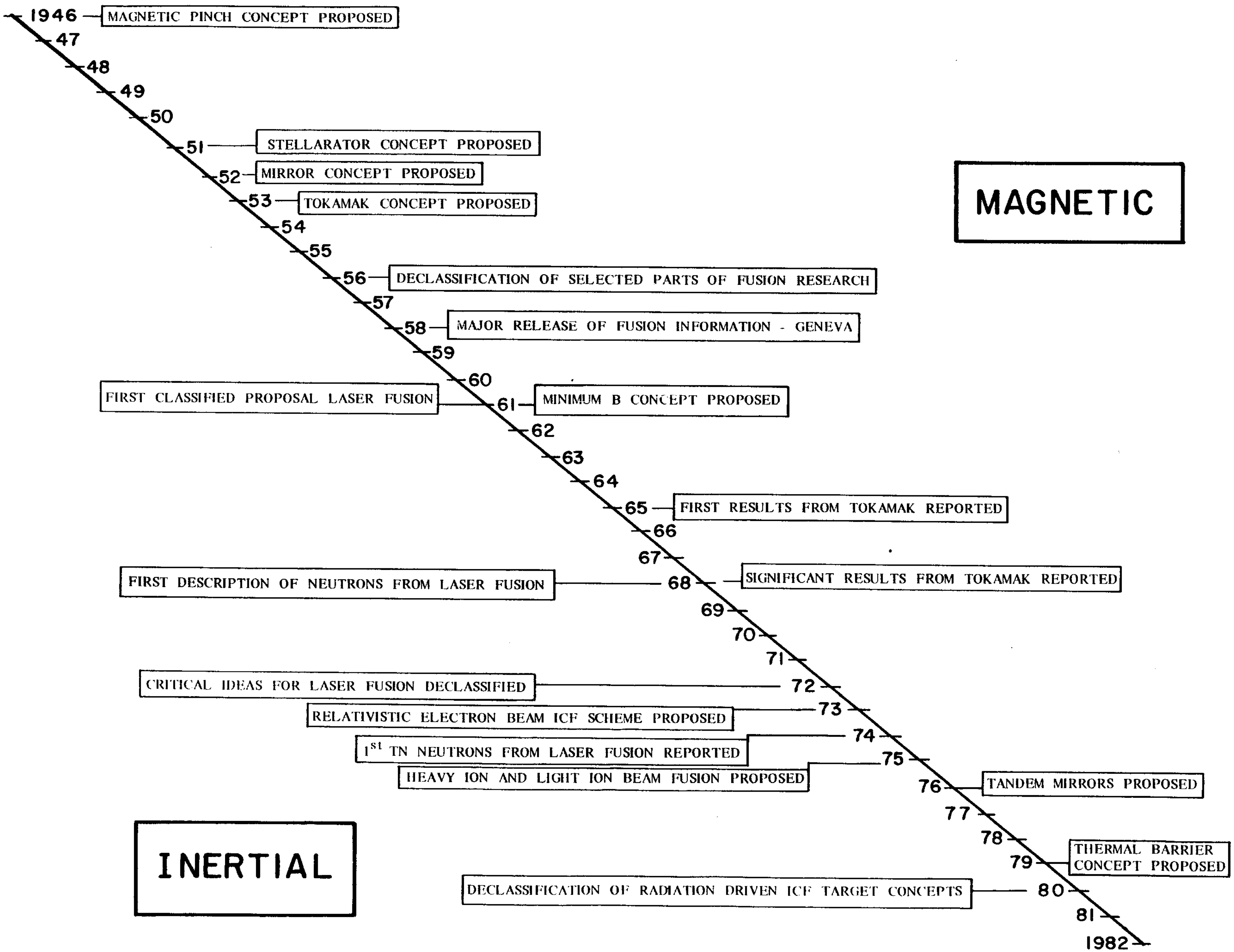


Fig. 1.3-1. Chronology of concept introduction in fusion research.

1972	Critical Ideas for Laser Fusion Declassified
1973	Relativistic Electron Beam ICF Scheme Proposed
1974	First Thermonuclear Neutrons from Laser Induced Fusion Reported
1975	Heavy Ion Beam Fusion Proposed Light Ion Beam Fusion Proposed
1976	Tandem Mirror Concept Proposed
1978	Tandem Mirror Concept Verified
1979	Thermal Barrier Concept Proposed for Tandem Mirrors
1980	Declassification of Radiation Driven ICF Target Concepts

It is of interest to note that roughly 30 years of research have been devoted to the magnetic confinement concept of fusion and that over 20 years of research have been devoted to research in inertial confinement. There have been major magnetic confinement approaches proposed (Mirrors, Stellarators, Pinches, Tokamaks, Reversed Field Configurations, Compact Tori, and Elmo Bumpy Torus) and a multitude of other magnetic systems have been considered around the world (e.g., multipoles, ion ring compressor, TORMAC, beam heated solenoids, etc.). We also note that there have been 4 major inertial confinement schemes (i.e., lasers, electron beam, light ion beam, and heavy ion beam driven systems) proposed while a smaller number of alternate ICF concepts have also been considered (e.g., exploding liners, impact fusion).

Commercial fusion reactor studies began at a very early stage in the development of fusion in order to determine what kind of an impact fusion could have on the overall energy picture. In fact, L. Spitzer and his colleagues performed the first major power reactor study in 1954 (3). The work in this area then dropped off during the 1960's as the fusion community concentrated on solving some very severe plasma physics problems but by 1969 interest was picking up as evidenced by the papers in the Nuclear Fusion Reactor Conference at Culham (4). The early seventies saw a great deal of activity in reactor design area, mainly in the U.S. and U.K., but there were smaller, but significant efforts in Germany, Japan and the Soviet Union. Early tokamak reactor studies included the UWMAK series at the University of Wisconsin, the MARK-I and II series at Culham in England, along with power plant studies at ORNL and PPPL. Early mirror reactor designs were performed at LLNL and the LANL performed detailed reactor analyses of a theta pinch device. Laser fusion reactor studies were also initiated at LANL and at Jülich in West Germany. By the mid-1970's the number of reactor design groups had expanded significantly in the U.S. (e.g., ANL, GA, LLNL) as well as in Japan, Italy, and West Germany. In the early 1980 period, the number of commercial or near term reactor design efforts was greatly reduced in the United States and a large scale international study (INTOR) was initiated. At the time of this writing, the number of long range reactor studies in the U.S. has been reduced drastically and practically all designs now are focused on near term devices. At the same time, other major fusion programs around the world are expanding their reactor design capabilities.

With this brief overview of fusion history in mind it is now appropriate to examine the extent of the present fusion research program around the world.

1.3.2 Major Fusion Programs Around the World

It is convenient at this point to divide the world fusion programs into three categories; major, intermediate and selective. The 4 major programs (each employing more than 500 scientists) are conducted in the U.S., U.S.S.R., European community (UK, France, Federal Republic of Germany, Italy, Netherlands, Belgium, Denmark, Sweden, and Switzerland), and Japan. There are 4 intermediate programs (each having between 50 and 500 scientists) currently conducted in Canada, the People's Republic of China, Australia, and Poland. Finally there are at least 18 selective programs (< 50 scientists each) around the world.

1.3.2.1 United States. The United States' Magnetic Fusion Program, initially classified and called Project Sherwood, was launched in 1951. By 1952 work had started in five locations around the country; Princeton University, Los Alamos, University of California at Berkeley, Oak Ridge National Laboratory, and the newly created Lawrence Livermore Laboratory in California. Shortly after Project Sherwood was initiated, scientists at MIT in Cambridge, NYU in New York, and the Naval Research Laboratory in Washington, DC joined the group. The fusion program in the U.S. has grown rapidly in the past 30 years and at the present time over 2000 professionals are thought to be engaged in research around the country. A listing of the major research groups is given in Table 1.3-1 along with their main interests. Obviously we cannot include all the laboratories, companies, and universities that have fusion related research, but the ones that are listed in Table 1.3-1 account for most of the funded research in the U.S. The location of the major experimental facilities in the country is given in Fig. 1.3-2.

Approximately 2/3 of the U.S. program is devoted to magnetic confinement and 1/3 is devoted to inertial confinement. However, the goals of the inertial confinement program now are entirely related to military activities and there are no stated civilian related goals. Since the ICF program has in the past, been perceived to have civilian related activities, we will continue to include it in our report. Within the current magnetic fusion program, approximately 50% of the effort is directed towards tokamaks, 20% towards mirrors, 15% towards alternatives and general physics support, and 15% to generic technology. About 80% of the inertial confinement program is devoted to lasers, ~ 10% to light ion beams and ~ 10% to general support including a very small (~ 1%) program on heavy ion beam fusion.

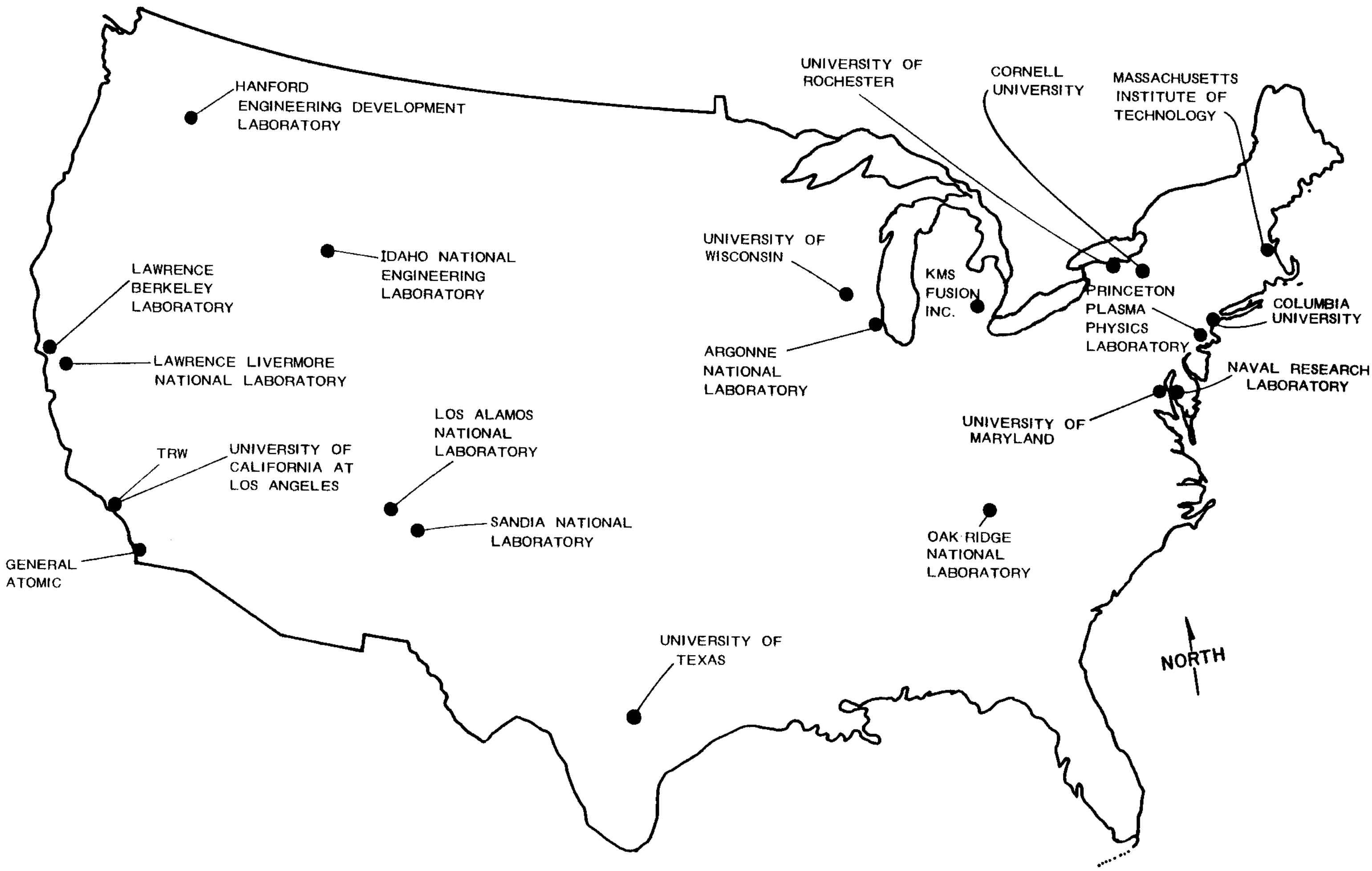
TABLE 1.3-1

Summary of Major U.S. Organizations Currently Participating in Fusion Research

Location	Major Concepts or Activities Currently Pursued
National Laboratories*	
Princeton Plasma Physics Laboratory	Tokamak, Stellarator, Compact Tori
Lawrence Livermore National Laboratory	Mirror, Laser
Oak Ridge National Laboratory	Tokamak, EBT, Stellarator
Los Alamos National Laboratory	Reversed Field Pinch, Laser, Compact Tori
Sandia National Laboratory	Light Ion
Lawrence Berkeley 'aboratory	Heavy Ion
Naval Research Laboratory	Inertial Confinement
Argonne National Laboratory	Technology
Hanford Engineering Development Laboratory	Materials, Technology
Idaho National Engineering Laboratory	Safety
Private Industry*	
GA Technologies	Tokamak, RFP
KMS	Laser
TRW	Mirror, Technology
McDonnell Douglas	EBT, Technology
INESCO	Compact Tori
Ebasco	Tokamak
Grumman	Tokamak
Westinghouse	Technology
Science Applications, Inc.	Theory, Technology
Jaycor	Theory, Technology
Universities**	
Massachusetts Institute of Technology	Tokamak, Mirror, Technology
Wisconsin	Mirror, Stellarator, Tokamak, Technology
Texas	Tokamak, Theory
UCLA	Mirror, Tokamak
Rochester	Laser
Cornell	Ion Beam
Maryland	Pulsed Systems, Diagnostics
NYU	Theory

* Those programs with more than 25 scientists.

** There were 35 different University Programs in FY-83, but 8 of those programs accounted for over 90% of the financial support to universities.



MAJOR U.S. FUSION EXPERIMENTAL FACILITIES

Fig. 1.3-2.

<u>1.3.2.2 European Euratom Program.</u> The European program in magnetic fusion research began in the U.K. at Harwell about the same time as the research programs in the U.S. and U.S.S.R. were started. It has now grown to include ~ 1000 professionals in 9 countries (see Fig. 1.3-3 for more details on the location of major European fusion laboratories). The current trend is to concentrate on the tokamak line (~ 75%) with ~ 15% devoted to alternate concepts (principally stellarators and RFP's) and ~ 10% for general technology. Less than 2% of the present budget is devoted to ICF (only lasers) and experimental mirror work has essentially ceased.

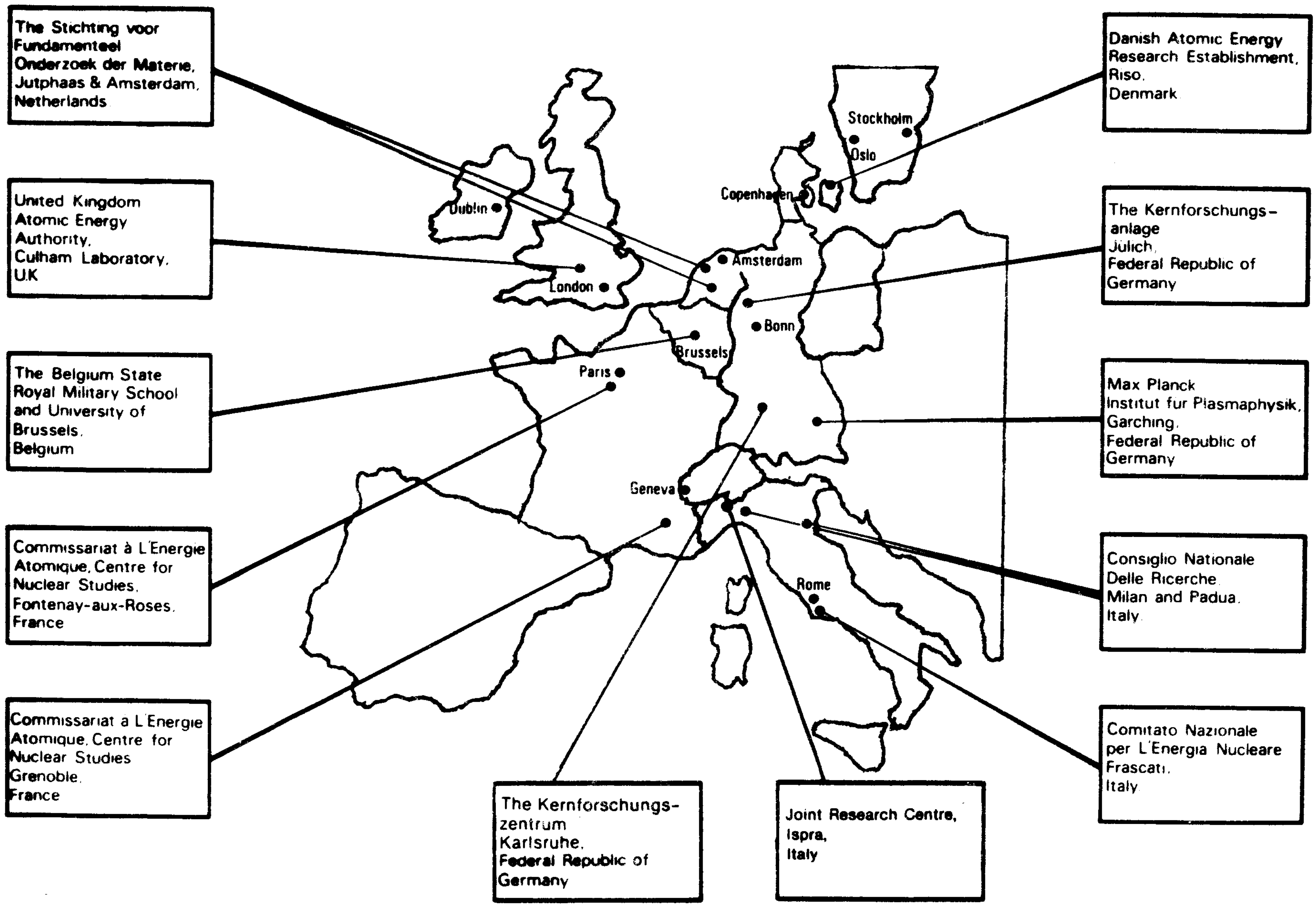


Fig. 1.3-3. Fusion Research in the European Community.

1.3.2.3 U.S.S.R. Program. The fusion program in the Soviet Union also began in the early 1950's and has now grown to a large, broadly based effort employing an estimated 2000 professionals at the seven major facilities listed in Table 1.3-2 and and at other institutions around the country. See Fig. 1.3-4 for the location of the laboratories.

It is impossible to determine the relative weights of the U.S.S.R. magnetic and ICF programs but it may be 2:1 or even 3:1 in favor of the magnetic concept. Within the magnetic program, the tokamaks are by far the most heavily pursued but programs at the ~ 10% level are devoted to mirrors and stellarators. The ICF program was initially heavily weighted toward the laser but in the past few years it appears that the electron beam program may command a substantial amount of the resources devoted to ICF. There is no known light or heavy ion beam program at the present time. The Soviet Union is heavily committed, regardless of the confinement approach, to the fission-fusion hybrid. This commitment tends to influence their reactor designs and their estimates as to when useful energy can be produced by early plasma devices.

TABLE 1.3-2

Summary of Major Fusion Laboratories in the U.S.S.R.

Laboratory	Professional Staff - 1977 (est.)*	Main Programs
Kurchatov	720	Tokamaks, Electron beam, Mirrors, Technology
Lebedev	100	Lasers, Stellarators
Kharkov	150	Stellarators
Novosibirsk	35	Mirrors
Efremov	350	Technology
Ioffe	50	Theory
Sukhumi	200	Tokamaks, Stellarators
TOTAL	1605	

*J.F. Clarke, Presentation to International Studies Association on US-USSR Cooperation in Science and Technology, Washington, DC, Feb. 25, 1978.

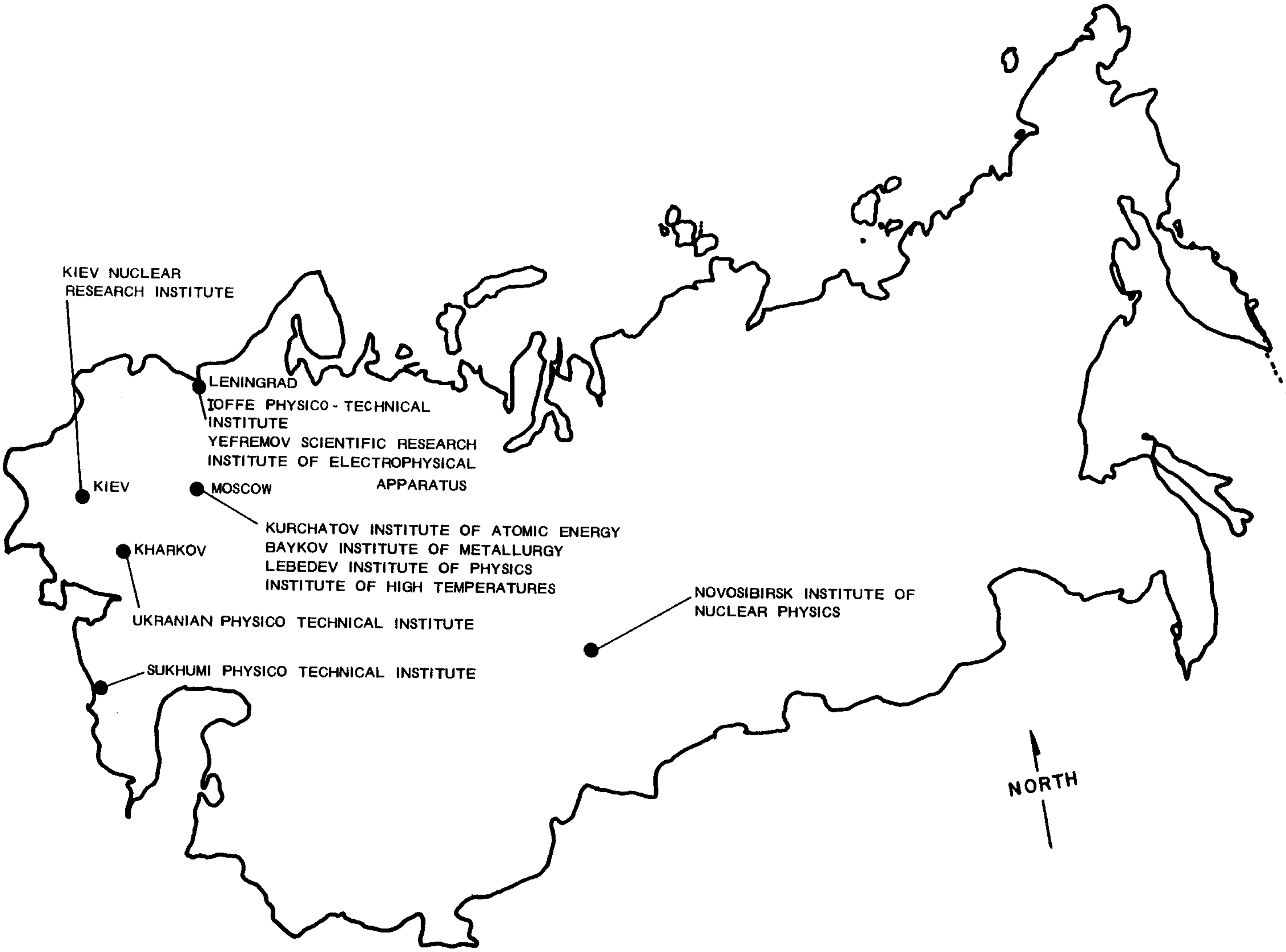


Fig. 1.3-4. Major fusion facilities in the U.S.S.R.

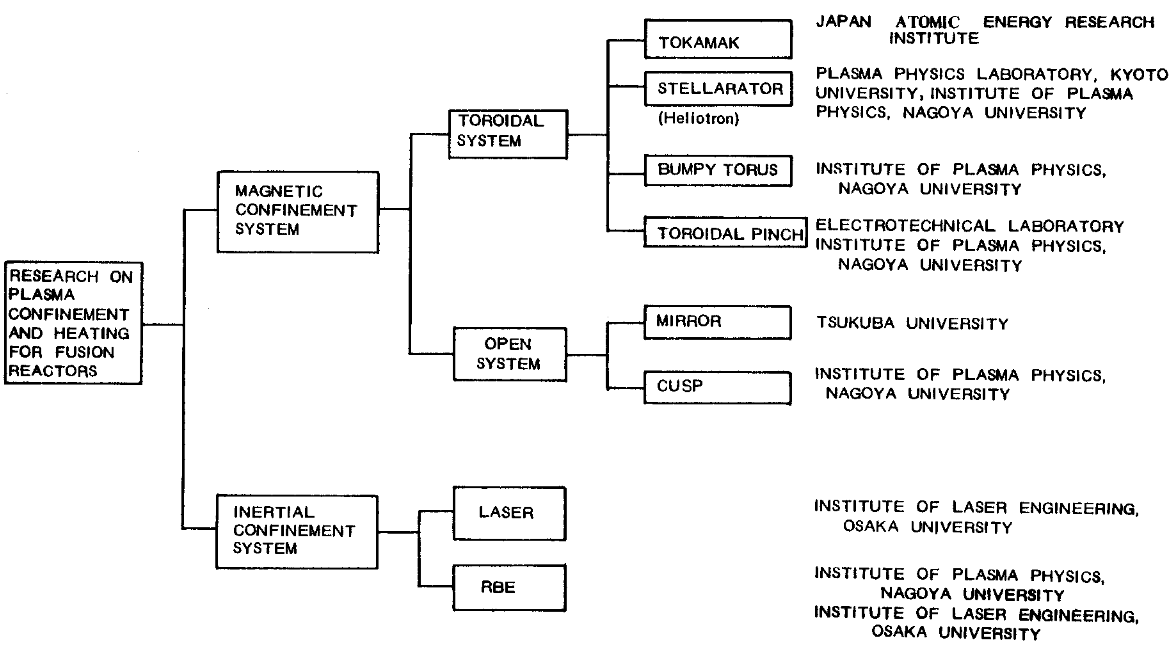


Fig. 1.3-5. Classification of fusion research in Japan.

1.3.2.4 Japanese Program. The fusion program in Japan is being given high priority as a national project. It was started at a very low level in the mid-1950's by the Japanese Atomic Energy Research Institute and began to receive significant funding in the 1960's. A major increase in the program began in the mid-1970's and the program has increased by more than a factor of 60 over the past 10 years! There are now between 700 and 800 scientists in the program. Industry has a very strong involvement in fusion as do the universities and at the present, the Japanese program is within a factor of two of the larger U.S. and U.S.S.R. efforts.

A schematic of the organization of the Japanese fusion program is given in Fig. 1.3-5 and the location of the research sites is given in Fig. 1.3-6. It is evident from these figures that universities have a major role in the program, especially in inertial confinement.

From the beginning, the Japanese program has been heavily weighted toward the tokamak concept as indi cated by Table 1.3-3 below.

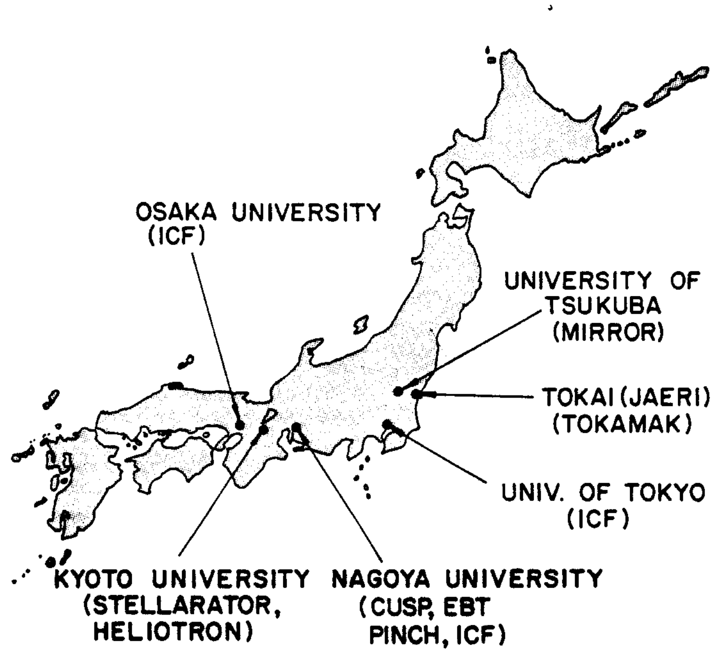


Fig. 1.3-6. Location of major Japanese fusion research facilities.

The program has become more heavily biased towards tokamaks in the early 80's although the program in mirror research has become significant since 1979. It is estimated that the present program balance is as follows: 80% tokamaks, ~ 3% stellarators, 5% mirrors, 6% ICF, and about 6% for pinches and EBT concepts. The Japanese government has recently supported, both with money and scientists, the Doublet-III project in the U.S. and currently supports the RTNS-II neutron source at LLNL with the same degree of enthusiasm.

1.3.2.5 China. The magnetic fusion program in China began in 1958 at the Institute for Atomic Energy in Peking. This program has expanded to 3 other institutes in China (Southwest Institute of Physics, Leshan; Hefei Institute of Physics, Hefei; and the Institute of Physics in Beijing). The lead responsibility now lies with the Southwest Institute of Physics (see Fig. 1.3-7). The inertial confinement program began at the Shanghai Institute of Optics and Fine Mechanics in 1966 with a glass laser system although a small ion beam program has been recently initiated at the Institute of Atomic Energy in

Beijing. An estimate of the staffing, funding, and technological goals of these institutes, as of 1980, is given in Table 1.3-4 below.

TABLE 1.3-3

Estimated Fusion Program Balance in Japan 1975-1982

% of Program Effort

Fiscal Year	Mirror	Laser	Stellarator + Heliotron	EBT + Pinch	Tokamak
1975	---	5	5	40	50
1976	---	11	13	30	46
1977	---	8	11	25	56
1978	---	10	15	26	49
1979	0.3	7	12	9	72
1980	5	6	2 5	82	
1981	5	7	1 6	81	
1982	5	6	3 6	80	

TABLE 1.3-4

Summary of Current Fusion Program in the People's Republic of China

Laboratory	Number of Scientists and Technicians	Est. Budget M $US (1980)	Main Areas of Interest
Southwest Inst. of Physics, Leshan	400	13	Tokamaks, Mirrors, Pinches, Technology
Hefei Inst. of Plasma Physics	200	5	Tokamaks
Shanghai	40	2	Lasers
Institute of Physics, Peking	70	?	Tokamak, Pinch
Institute of Atomic Energy	10	?	Electron Beam
Total	~ 720	~ 20	

1.3.2.6 Canada. The National Research Council (N.R.C) is establishing, in collaboration with provincial partners, a coordinated National Program of

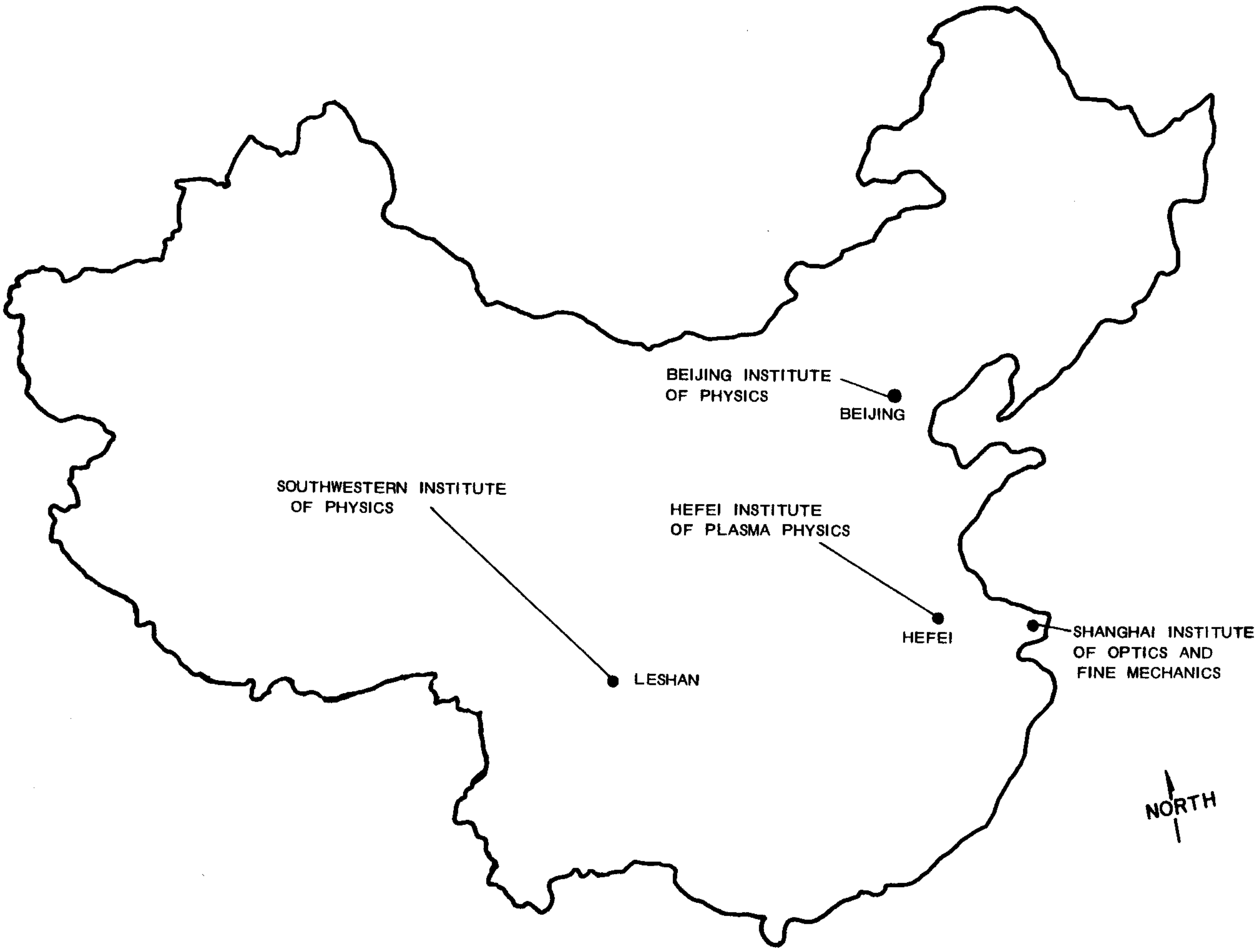


Fig. 1.3-7. Major Chinese fusion facilities.

Fusion Research and Development which is planned to grow from an insignificant level in 1980 to a total annual operating level of about $20 million in 1985. The long term (20 year) objective of the program is to put Canadian industry in a position to manufacture subsystems and components of fusion power reactors. In the more immediate future, the program is designed to establish a minimum base of scientific and technical expertise in fusion technology sufficient to make recognized contributions, and thereby gain access, to the international effort. Three broad areas of fusion technology have been chosen and they are: magnetic confinement technology, inertial confinement technology and special materials or engineering problems.

The chosen specialization in magnetic confinement is a tokamak with a unique capability of quasi-steady state operation. This tokamak is being constructed on an equal, cost-shared basis with Hydro Quebec at Varennes and will be operational as a national facility in 1984 at a total cost of about $37.4 million. It will provide data on high-duty-cycle operations of tokamaks which is of international interest for the design of the next generation of engineering test reactors planned for operation in the early 1990's.

The chosen specialization in materials and engineering is a Fusion Fuels Technology Project centered on the management of the fusion fuel, tritium. This is a 5 year, $20.6 million, R/D program with the cost shared 2:1:1 by N.R.C., Ontario Hydro and the Ontario government. Foreign programs have shown particular interest in this project because of Ontario Hydro's expected unique position as the world's dominant, non-military producer of tritium during the next two decades.

The specialization in inertial confinement is a laser fusion project focused on high-power, gas laser technology and sophisticated diagnostic instrumentation. Details of the scientific program and of a national facility and its organization are being developed.

1.3.2.7 Australia

The fusion program in Australia was actually started in 1959 by Oliphant (who produced the first D-T reaction in Cambridge, England, in the 1930's). The first tokamak outside the USSR was operated at the Australian National University (ANU) in Canberra, Australia in 1964. The current program involves ~ 68 scientists and graduate students mainly working at ANU and at the University of Sydney, University of New South Wales and Flinders University. The LT-4 tokamak at ANU and the TORTUS tokamak in Sydney command most of the resources of the Australian program estimated to be ~ 4 million dollars per year (1983$). Future emphasis may be on tokamaks, stellarators and lasers.

1.3.2.8 Other Fusion Programs Around the World. In addition to the major and intermediate programs around the world, there are many smaller, but significant programs that should be mentioned. Since it would be impossible to do an indepth survey for this report only the scientific staff levels reported in the 1982 report of the IAEA (5) will be quoted (see Table 1.3-5). The fusion programs are divided into 4 groups (outside of those already reported): (1) Europe (outside of the European community nations discussed in Section 1.3.2.2), (2) Mid-East, India, and Africa, (3) Far East, and (4) South America. Since one cannot obtain specific information on Poland, they were included in this section even though Poland does have more than 50 scientists. We realize that some countries are not included in the IAEA survey (e.g., Egypt, GDR, etc.) but until such numbers are available for these countries, the value of ~ 350 additional scientists outside the major and intermediate programs discussed previously, should be considered only as a lower limit.

TABLE 1.3-5

Summary of Smaller Fusion Programs Around the World - 1982 (5)

Country	Scientific Staff
Europe (outside EC)	
Austria	23
Czechoslovakia	29
Finland	2
Hungary	30
Poland	92
Portugal	13
Romania	17
Spain	11
Subtotal	217
Mid-East, India and Africa	
India	8
Iran	5
Israel	18
Sudan	1
South Africa	16
Turkey	8
Subtotal	56
Far East	
Korea	12
Malaysia	6
New Zealand	6
Subtotal	24
South America	
Argentina	23
Brazil	23
Subtotal	46
Total	343

TABLE 1.3-6

Summary of the U.S. Department of Energy Funding Levels for Fusion*

Fiscal Year	Magnetic	Inertial	Total
1951-3	1.1	---	1.1
1954	1.8	---	1.8
1955	6.1	---	6.1
1956	7.4	---	7.4
1957	11.6	---	11.6
1958	29.2	---	29.2
1959	28.9	---	28.9
1960	33.7	---	33.7
1961	30.0	---	30.0
1962	24.8	---	24.8
1963	25.5	0.3	25.8
1964	22.6	1.4	24.0
1965	23.1	1.4	24.5
1966	23.1	1.2	24.3
1967	23.9	1.5	25.4
1968	26.6	1.3	27.9
1969	29.7	2.4	32.1
1970	34.3	3.6	37.9
1971	32.2	11.3	43.5
1972	36.3	20.1	56.4
1973	43.4	36.1	79.5
1974	62.8	49.5	112.3
1975	118.2	65.1	183.3
1976(**)	219.1	103.0	322.1
1977	316.3	111.9	428.2
1978	322.4	130.6	453.0
1979	355.9	144.1	500.0
1980	364.0	194.9	558.9
1981	393.6	209.6	603.2
1982	453.8	209.1	662.9
Total (1951-82)	3101.4	1298.4	4399.8
1983 (est.)	447.1	189.7	636.8
1984 (est.)	467.	169.7	636.7

*) Some of the data in this table was provided by M. Roberts and T. Godlove, U.S. DOE

**) Includes Transition Year.

1.3.3 Funding Levels of World Fusion Program

There has been a dramatic increase in the level of funding for the world fusion research program over the past 5 years. While the exact funding levels of the various fusion programs are impossible to obtain on a self-consistent basis, it is possible to make reasonable estimates, conversions, and adjustments for different accounting systems to come up with an overall world annual spending level of ~ 2 billion dollars in 1982. The analysis which has been used to arrive at this number is given in the next several subsections.

1.3.3.1 U.S. Department of Energy Program. The historical levels of funding for the U.S. DOE magnetic and inertial confinement fusion programs are given in Table 1.3-6. This number represents budget authorizations (BA) in the year indicated. Sometimes the money is not spent in that same year so that these numbers may differ from the actual budget outlays (BO) in that year. However, the integral of the BA and BO over a long period of time should be quite comparable. The funding information is displayed graphically in Figs. 1.3-8, and 1.3-9. A few of the more interesting observations on this data are given below.

1. More than 5 billion dollars will have been spent by the U.S. government on fusion research from FY-1951 through FY-1983.

2. Approximately 2/3 of the funds have been spent on magnetic fusion vs. 1/3 on inertial confinement.

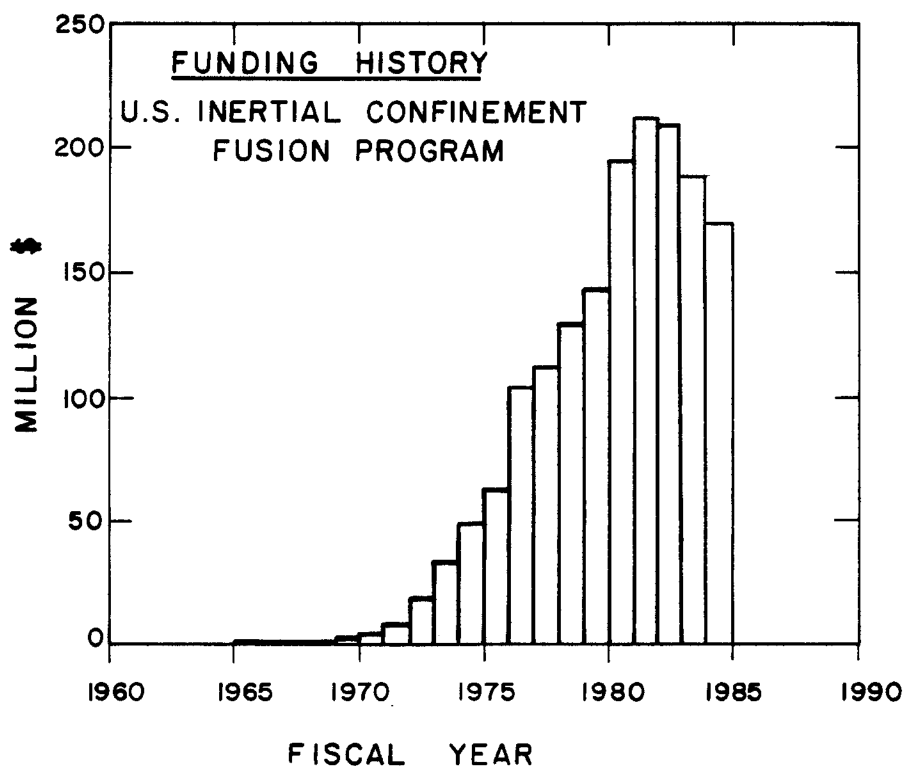


Fig. 1.3-8. Funding history of U.S. inertial confinement fusion program. FY 83 and 84 are estimates.

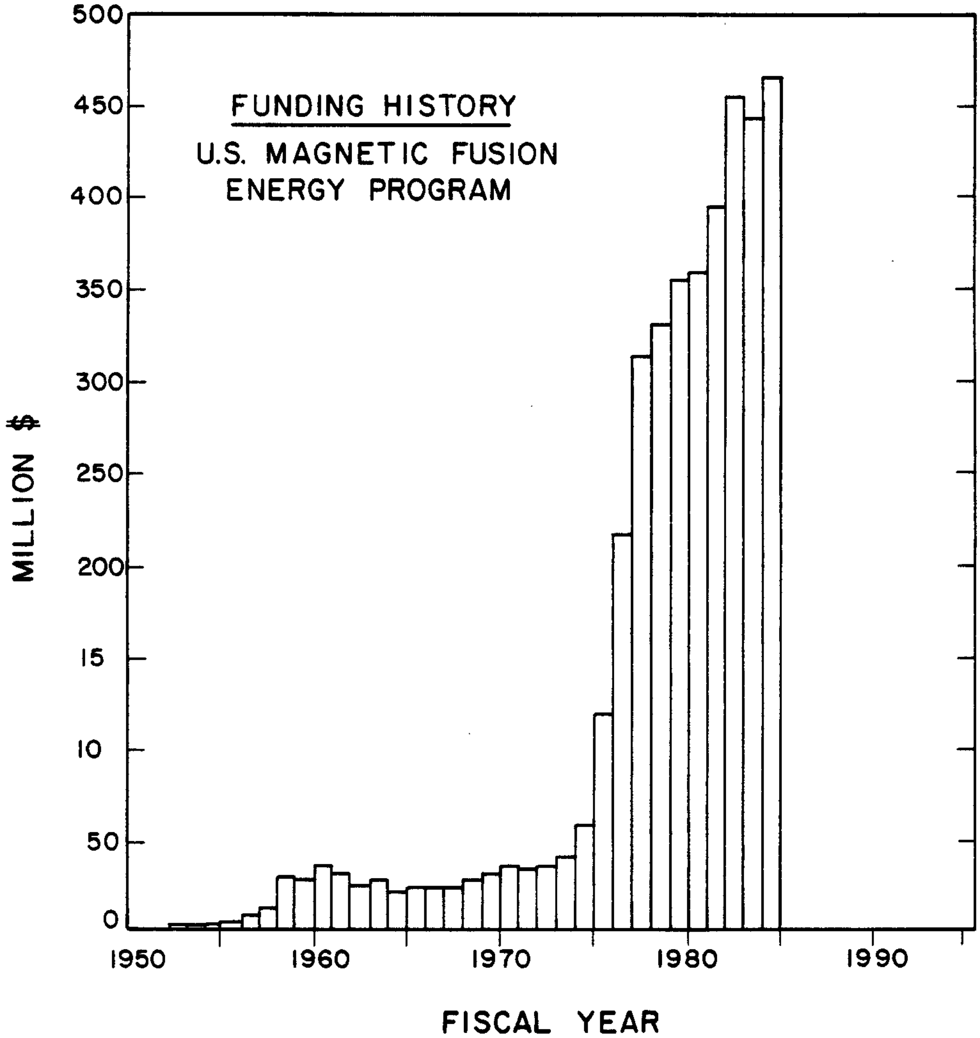


Fig. 1.3-9. Funding history of U.S. magnetic fusion program. FY 83 and 84 are estimated.

3. Over the past 5 complete fiscal years (1978-1982) the annual funding has increased by 41% and 60% for the Magnetic and Inertial programs, respectively, while the Consumer Price Index has increased by 48% over the same period.

4. The estimated total funding level in FY-83 is ~ 640 million. This represents the first time in 16 years that the total fusion budget has declined compared to a previous year.

It should be noted that electric utility, NSF, and private company funded research is not included in Table 1.3-6. However (see below), the conclusions would not be changed qualitatively if they were considered.

The U.S. government support for fusion through the National Science Foundation (NSF) can only be estimated at this time. Much of the research has been in the area of plasma physics which has a broad application over many fields. However, a preliminary analysis of the NSF budgets reveal that from FY-65 to FY-83, the funding level has been between 1 to 2 million dollars per year. Taking an average of 1.5 million dollars, this amounts to ~ 30 million to date.

1.3.3.2 European Community Program. The EC program began in 1959 and has been funded at a level of ~ 1/2 of the total U.S. program for the past 5 years. The exact amounts in millions of units of account are listed in Table 1.3-7. We have also converted the funding level to U.S. dollars using the official exchange rate which varies from year to year (e.g., 0.99$/ECU in 1982 to 1.39$/ECU in 1980). Some observations on the EC program are:

TABLE 1.3-7

Summary of Recent Funding Levels for Fusion Research in Europe

	Millions of Units of Account (ECU)		Millions Equivalent U.S. $[a]
Year	Base	JET	Total
1959-62	30	---	30
1963-68	158	---	158
1969-70	50	---	50
1971-75	290	---	290
1976	95	---	95
1977	103	---	103
1978	130	13	143
1979	171	48	219
1980	209	70	279
1981	186	90	276
1982	178	109	287
TOTAL	1600.0	330	$1930
1983 (est.)			317

a) Does not include early expenditures from U.K. (1952-9) which are estimated to be ~ 30 million dollars.

1. The total amount spent on fusion from 1959-83 is over 2.2 billion dollars.

2. The JET activity has accounted for ~ 1/4 of the total expenditures in Europe in the past 5 years.

3. The magnitude of the current EC magnetic fusion program is ~ 2/3 of the U.S. magnetic fusion program.

4. Since there is essentially no inertial confinement program in Europe, the total EC fusion budget is only ~ 1/2 of the total U.S. program.

The EC program has increased much more in the past 5 years (1978-82) than the U.S. program. The increase in European funding is ~ 160% vs. ~ 50% in the U.S. program. However, it is expected that the rate of increase in the EC program over the next 5 years will be limited to ~ 10% per year (see Section 6.7).

1.3.3.3 Japanese Fusion Program. Significant funding in the Japanese fusion program began in 1963. As was stated earlier, the total program is the sum of that spent by the Science and Technology Agency and the Ministry of Education. A summary of both programs is given in Table 1.3-8 along with a conversion to U.S. dollars at the average rate in force during the year of expenditure. It is important to note that these numbers do not include salaries, administrative fees and building construction costs, and therefore are not directly comparable to U.S. and EC numbers. With these restrictions in mind, one can make some interesting observations.

1. The total amount of money invested (exclusive of salaries) by the Japanese through FY-82 is ~ 0.9 billion dollars.

2. The funding level for the fusion program has increased by a factor of 210% in the last 5 years, ahead of the EC (160%) and U.S. (50%) increases in that same time period.

3. Estimates of the current number of professionals in the Japanese program are 400 at JAERI, 50 at other science and technology facilities, and 300 supported by the Ministry of Education. Using a value of 750 and a figure of 100,000 dollars per professional per year, we estimate that the FY-83 total expenditure rate is ~ 300 million \$/year (~ 1/2 of the U.S. level and equal to the EC program).

TABLE 1.3-8

Summary of Japanese Fusion Program Funding[a]

		Science & Tech. Agency		Ministry of of Education		Total	
FY	Ave. Yen/$	Million Yen	Million $ U.S. (Eq.)	Million Yen	Million $ U.S. (Eq.)	Million Yen	Million $ U.S. (Eq.)
1963-8	360.9	238	0.66	1,288	3.57	1,526	4.23
69	358.7	116	0.32	297	0.83	413	1.15
70	358.5	345	1.02	288	0.85	633	1.87
71	338.	524	1.55	230	0.68	754	2.23
72	312.5	654	2.09	228	0.73	882	2.82
73	277.8	624	2.25	305	1.10	929	3.34
74	294.1	992	3.37	914	3.11	1,906	6.48
75	295.3	2,801	7.49	1,553	5.26	4,354	14.74
76	295.6	4,326	14.63	3,920	13.26	8,246	27.90
77	263.9	8,069	30.60	4,265	16.17	12,334	46.77
78	204.7	11,999	58.62	5,377	26.27	17,376	84.89
79	219.2	23,959	109.30	7,852	35.82	31,811	145.12
80	227.3	29,708	130.70	8,240	36.25	37,948	166.95
81	219.5	36,320	165.44	10,126	46.12	46,446	211.56
82	248.1	41,795	168.43	11,903	47.97	53,698	216.40
Total 1963-82		162,470	698.47	56,786	234.96	219,256	933.43

a) Numbers do not include salaries, administration fees and building construction costs.

4. If the number of professionals is ratioed from 750 in FY-82 to the previous years, and assuming an average salary inflation rate of 6%, we estimate an additional 375 million dollars for salaries should be added to the total documented expenditures, bringing the total expenditures to ~ 1.3 billion dollars from 1963 through 1982.

5. Approximately 3/4 of the funds have been spent at Tokai (or subcontracted to industry) and ~ 1/4 at universities.

Regardless of how the Japanese program is analyzed, it is clearly the most rapidly expanding program in the world.

1.3.3.4 Soviet Union Program. Unfortunately, there are no detailed analyses of the Soviet program which can be compared to the U.S., Japan, and EC programs. At various times in the past, Western scientists have referred to the U.S.S.R. program as being as large (money-wise) as the U.S., even though the total number of people involved (at all levels) has been placed above the U.S. program. For example, it was estimated in 1978 that the ratio of professional staff for the U.S.S.R. to U.S. was 1600/900 and that the ratio of the total staff was 4000/1800. It has also been said that the Soviets have a bigger magnetic fusion program than the U.S. but a smaller ICF program. Lacking any reliable documentation it will be assumed, for the purposes of this report, that the total U.S.S.R. financial program has been equal to the U.S. program since the early 1950's. When more substantive information is available, the numbers in this report will be adjusted appropriately.

1.3.3.5 Other Governmental Supported Fusion Research. The current estimates of funding outside the programs in the U.S., Europe and Japan suffer from a general lack of historical data. It is known, for example, that in 1980 the People's Republic of China was supporting fusion at a rate of ~ 20 million dollars per year. Similarly, it is known that the program in Canada was supported at the rate of ~ 15 million dollars in the 1982-83 funding period. However, there is little information beyond that. Therefore an estimate of a country's expenditures will be obtained by multiplying the number of scientists by $100,000 each to get an idea of the level of program funding outside the major programs. Referring to Table 1.3-5, one finds that in 1982 a total of ~ 350 scientists were employed outside the previous programs analyzed. This would indicate that another 35 million dollars per year is being spent, worldwide, in these research programs.

1.3.3.6 Utility Support in the U.S. There are two sources of funding in this category; that from the Electric Power Research Institute (EPRI) and that from the U.S. utilities directly to research groups. While the former amounts are well documented, the accuracy of the latter numbers is relatively poor because there is no central "clearinghouse" for the funds. The two sources will be treated separately for the time being.

The fusion funding history for EPRI is given in Table 1.3-9. The actual numbers are also compared to the total spending of EPRI for all research and also as a fraction of the U.S. national program. Some interesting observations can be made.

TABLE 1.3-9

Summary of EPRI Fusion Funding

Calendar Year	EPRI Fusion $K	Total EPRI $K	Fusion % of EPRI	EPRI Fusion % of U.S. National Program*
1973	1,920	26,200	7.3	2.2
1974	1,222	34,100	3.6	1.3
1975	4,636	63,100	7.3	3.2
1976	4,297	99,500	4.3	2.0
1977	3,602	129,500	2.8	1.1
1978	3,380	165,000	2.0	0.78
1979	3,305	206,000	1.6	0.70
1980	2,200	223,300	1.0	0.41
1981	2,222	219,600	1.0	0.38
1982	2,000	260,000	0.77	0.32
Total	28,784	1,426,300	2.0	
1983 (est.)	2,800	268,000		

* Use value for N^{th} calendar year for EPRI divided by ave. of FY for N-1 and N^{th} year for U.S. program.

1. Nearly 29 million dollars has been spent by EPRI to support fusion research over the past 10 years.

2. The ratio of EPRI's expenditures for fusion to its total budget has dropped by a factor of 10 over the past 10 years.

3. EPRI has spent ~ 2% of its total research budget on fusion over the past 10 years.

4. The ratio of the EPRI fusion budget to the U.S. fusion budget has also dropped by a factor of ~ 7 over the past 10 years to a level of ~ 0.3% of the national effort.

In contrast to the expanding governmental budgets for fusion over the past 5 years, the EPRI fusion budget has dropped significantly. It is now only 60% of its 1978 level and adjustments for inflation would reduce this fraction to 40% of its 1978 level.

Non-EPRI utility support for fusion began at a very early stage. In 1957, eleven investor-owned power companies of Texas formed the Texas Atomic Energy Research Foundation (TAERF). Over a period of 10 years TAERF supported the General Atomic Company to perform research in fusion to a total level of 11.5 million dollars. The TAERF group has also supported the Univers ty of Texas at Austin through 1983 for a total of 6.9 million dollars.

The Wisconsin Electric Utilities Research Foundation (WEURF) has provided support for the University of Wisconsin over the past 10 years. To date, the WEURF group has provided over 1.5 million dollars of research funds. It is also known that several eastern utilities have supported work at the University of Rochester and Princeton University but the level of funding is unknown. Similarly, several western utilities have supported work at LLNL but again, the support level is unknown.

1.3.3.7 U.S. Private Research Support. Again, we can only estimate the level of effort for private companies. It has been inferred that INESCO is spending ~ 12 million dollars a year to develop small fusion reactors. General Atomic also has a substantial private funding program but the exact financial level in unknown. It is known that KMS Inc. has invested on the order of 10 million dollars into its own fusion research program over the past decade. Other companies such as McDonnell Douglas, TRW, Grumman, Ebasco, General Electric, Westinghouse, Exxon, etc., very often "match" funds obtained from the U.S. Federal Government, but there is no way to know whether those are "new" funds or part of the profit made on previous contracts. In lieu of any firm reports by the companies, it is the author's estimate that the total amount of money currently put into fusion research is ~ 20 million dollars per year heavily weighted by the INESCO project.

1.3.3.8 Total Worldwide Fusion Program. Table 1.3-10 summarizes the estimates of the FY-82 spending rate, the total amount spent in the last 5 years, and the total amount spent by all the fusion programs since 1951. In addition, the FY-82 expenditures of $ per capita have been calculated for some of the major programs. The key features of these comparisons are given below.

1. The world fusion program is now being funded at ~ 2 billion dollars per year with the U.S. and U.S.S.R. each contributing an estimated 1/3, the EC and Japanese programs contributing 15% each, and ~ 4% from other countries.

2. The U.S. is spending the most per capita on fusion at ~ 3 $/person, Japan and the U.S.S.R. are at ~ 2.5$/person and the EC countries are supporting fusion at ~ 1$/person.

3. In the last 5 years, approximately 8 billion dollars has been spent supporting fusion research.

TABLE 1.3-10

Estimated Financial Support of the Worldwide Fusion Effort

Millions of U.S. $ Equivalent

Government	FY-82	FY-82 $/Capita	Total FY-78 through 82	Total Through FY-82 Since Program Began
U.S.				
Magnetic	456		1,898	3,131
Inertial	209		888	1,298
Utilities	3		15	50
Private	20		50	100
	688	3.1	2,851	4,579
EC	287	1.1	1,204	1,960[a]
Japan	229[b]	2.0[b]	823[b]	935[b]
	~ 300[c]	~ 2.6[c]	~ 1,085[c]	~ 1,300[c]
U.S.S.R.	~ 660[d]	~ 2.5[d]	~ 2,800[d]	~ 4,500[d]
Other Countries (including Canada & China)	~ 70[d]		~ 150[d]	~ 200[d]
TOTAL	~ 1,930[b] ~ 2,000[c]		~ 7,830[b] ~ 8,090[c]	~ 12,200[b] ~ 12,500[c]

[a] Including estimated UK funding, 1952-59.
[b] Without salaries.
[c] Estimated with salaries for Japan.
[d] Estimated.

4. From the start of the first fusion program in the U.S., U.K., and U.S.S.R., between 12 and 13 billion dollars has been invested in fusion research.

5. At the current rate of spending the world fusion programs will have invested ~ 30 billion dollars (neglecting inflation) in fusion by the year 2000, the time at which only the most ambitious programs anticipate the operation of the first fusion power plant.

It seems obvious that anything which can save a few years of research time can have tremendous financial pay off. That is, if the first fusion Demo could be brought on 5 years earlier, 5 to 10 billion dollars (1982 $) might be saved worldwide!

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Section 2

MAGNETIC CONFINEMENT FUSION

2.1 CONFINEMENT SYSTEMS

Magnetic confinement fusion utilizes a magnetic field to confine the plasma in two basic configurations: toroidal and open systems. In toroidal systems, the magnetic field lines form a torus, and hence plasma flowing along the field does not strike material surfaces. The tokamak, stellarator, Elmo Bumpy Torus, and toroidal pinches are members of this category of confinement systems. The other major category is the open system, which includes magnetic mirrors and cusps. In this category, the magnetic field lines strike material surfaces, but plasma flow to these surfaces is inhibited by the magnetic mirror effect which requires a strong magnetic field at the ends of the plasma. In the following sections, the current status of the tokamak, which is the leading toroidal system, and the tandem mirror, which is the leading open system (at least in the U.S. program) are described. The status of stellarators, bumpy tori, and compact tori, which are alternate concepts is then reviewed.

2.1.1 Tokamaks

2.1.1.1 Present Facilities. The tokamak concept (see Section 1.2.3.1) was invented in the U.S.S.R. and tokamak experiments began in Europe, Japan, and the U.S.A. after encouraging results were reported at international conferences in 1968 and 1969. The first tokamak experiments in the U.S. were the ST, which was converted from the Model C stellarator at Princeton, ORMAK at Oak Ridge National Laboratory, and Alcator at MIT. Table 2.1-1 lists the present major tokamak facilities (1, 2). The "typical" values listed are the maximum values obtained in normal operation. These devices are a second generation of tokamak facilities and have considerably larger dimensions, higher magnetic fields, and greater auxiliary heating power than the earlier devices. The HL-1 device in China is under construction and scheduled for operation in 1984. China also has several smaller tokamaks which are in operation. Construction of a larger tokamak (CT-8) was begun, but cancelled for financial reasons.

2.1.1.2 Key Parameters and Issues. The best plasma parameters achieved in present tokamak facilities (through the end of 1982) are shown in Table 2.1-2. It is important to note, however, that these parameters are not achieved under

TABLE 2.1-1

Present Major Tokamak Facilities

Device	Country	Date Initial Operation	R (m)	a (m)	Magnetic Field Design (T)	Magnetic Field Typical (T)	Plasma Current Design (MA)	Plasma Current Typical (MA)
PLT	USA	1975	1.32	.45	5.0	3.2	1.6	.5
ISX-B	USA	1977	.93	.27	1.8	1.4	.2	.23
Alcator-C	USA	1978	.64	.17	14.	10.	1.5	.5
D-III	USA	1978	1.43	.45	2.5	2.5	.1	.1
PDX	USA	1978	1.45	.57	2.4	2.3	.5	.5
JFT-2	Japan	1972	.90	.25	1.8	1.6	NR	.16
TFR	France	1973	.98	.20	6.0	4.5	.6	$\leq$ .6
DITE	UK	1976	1.17	.26	2.7	2.7	.25	.25
T-10	USSR	1975	1.50	.37	5.0	3.5	.8	.65
FT	Italy	1977	.83	.21	10.	8.0	1.0	.6
ASDEX	W. Germany	1980	1.65	.40	2.8	2.8	.5	.5
Textor	W. Germany	1981	1.75	.50	2.6	NR	.48	NR
HL-1	China	Const.	1.0	.2	5.0	---	.40	---

R = major radius
a = minor radius
NR = not reported

the same discharge conditions. For example, the highest ion and electron temperatures are achieved at low plasma density, whereas the highest values of $n\tau_e$ (product of density and energy confinement time) are generally achieved at high density. The record for the highest electron and ion temperature was obtained in PLT (3) using neutral beam heating at low density ($n = 2 \times 10^{13}$ cm^{-3}). This result was significant because it produced a very collisionless plasma with scaling parameters chosen to what one expects for a reactor plasma. It was found in Alcator and later seen in other devices that the energy confinement time increases linearly with density. Consequently, the highest values of $n\tau_e$ are obtained with high density plasmas. More recently, Alcator C has begun operation with a much higher magnetic field and plasma density. They have found that the improvement of τ_e with density is beginning to saturate, so that prospects for improved plasma confinement by operation at even higher density do not appear favorable.

Figure 2.1-1 shows $n\tau_e$ and corresponding ion temperatures achieved in various devices. The present record for $n\tau_e$ is 4×10^{13} s/cm^3; reactors require $n\tau_e \gtrsim 2 \times 10^{14}$ s/cm^3 at an ion temperature ~ 10-20 keV. The present major tokamaks also have a significant amount of auxiliary heating power. This is usually in the form of neutral beam heating, although various forms of wave

TABLE 2.1-2

Best Plasma Parameters Achieved (Not Necessarily Obtained Simultaneously)

Device	Density (cm^{-3})	T_i (keV)	T_e (keV)	$n\tau_e$ (s/cm^3)	β (%)
PLT	1.5×10^{14}	7.3	4	1.5×10^{13}	2
ISX-B	1.3×10^{14}	1.8	1.9	NR	2.5
Alcator-C	8×10^{14}	1.5	2.5	3×10^{13}	.8
D-III	1.2×10^{14}	2	1.7	1×10^{13}	4.6
PDX	7×10^{13}	6	3	4×10^{12}	3.
ASDEX	1.2×10^{14}	4.8	2.0	NR	1.0
FT	7.5×10^{14}	1.4	1.7	4×10^{13}	NR

T_i = ion temperature
T_e = electron temperature
β = ratio of plasma pressure to magnetic pressure
NR = not reported

heating are receiving increasing interest. Table 2.1-3 lists the amount of heating power installed and "injected" into the plasma in various facilities to date. (The technology for neutral beam and wave heating is discussed in Section 2.2.) Increased heating capability has been important in the study of plasma energy confinement at elevated temperatures as well as allowing investigation and optimization of heating techniques. A reactor plasma requires substantial heating in order to bring it to ignition. Although the highest heating power has been achieved with neutral beam heating, wave heating is also being investigated in order to find a suitable alternative to neutral beams. Wave heating is potentially cheaper and more compatible with reactor requirements than neutral beam injection. Some forms of wave heating also allow the possibility of better control over the plasma profile; this may allow one to avoid plasma disruptions or optimize the plasma performance in other ways.

An important parameter in the performance of the plasma is the value of β (the ratio of the plasma pressure to the magnetic pressure) that can be achieved. Beta values in the range 5-8% are considered necessary for an economic tokamak reactor. The advent of substantial neutral beam power coupled with operating the experiments at low magnetic field has allowed one to investigate plasma operation at higher β values. Beta values in the range of 2.5% to 3% have been achieved in several experiments; recently a beta value of 4.6% in Doublet III (4) using a vertically elongated plasma shape has been reported. Generally, the energy confinement time has degraded with increasing neutral beam power so that higher β values have been difficult to obtain. The reason for this is not yet understood.

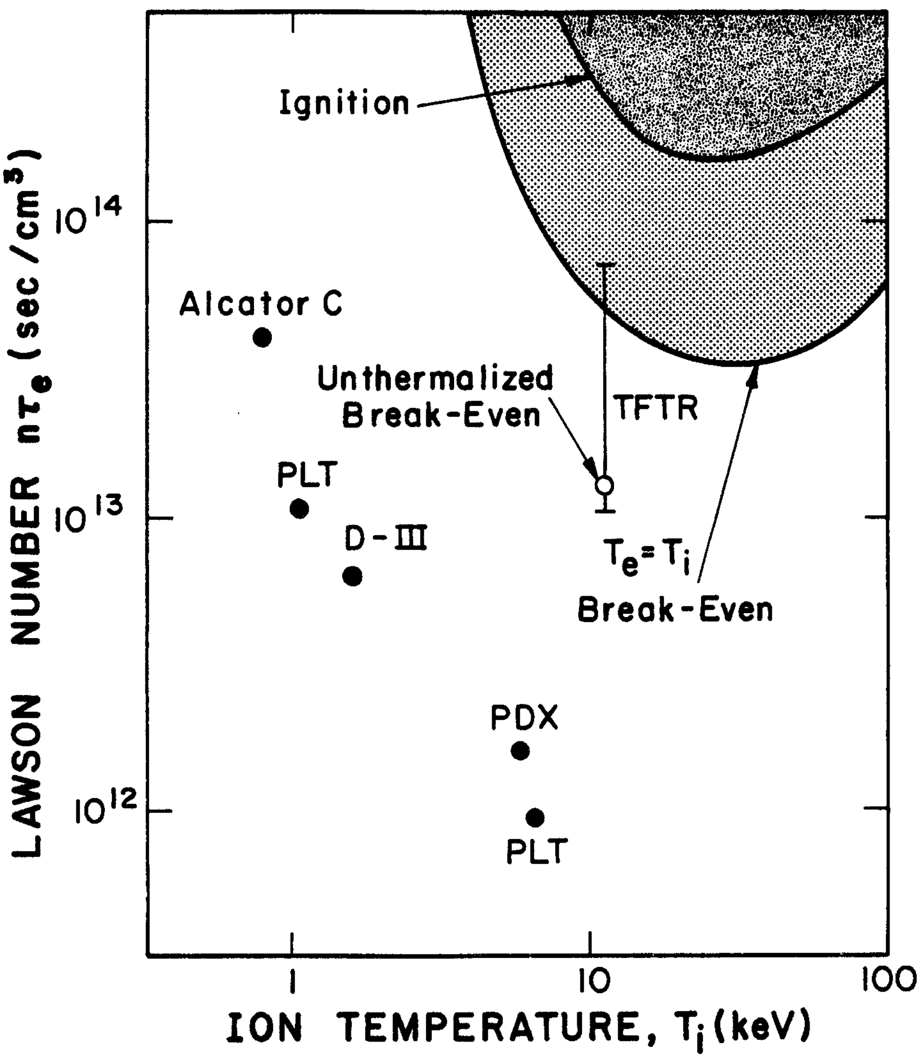

Fig. 2.1-1. $n\tau_e$ vs. ion temperature for some recent tokamak experiments and the projected values for TFTR.

The reactor prospects of the tokamak concept are significantly enhanced if the plasma current can be maintained steady-state. This would mitigate fatigue problems and increase plant availability, in addition to other possible benefits. In present devices, the current is driven inductively by an iron or air-core transformer which is unfortunately suitable only for pulsed operation. There is the possibility of achieving steady-state operation using wave or beam injection techniques to drive the plasma current. So far experiments in large tokamaks have only been performed using high frequency waves for current drive. In low density discharges in PLT, the plasma current has been maintained up to 3.5 s at the 165 kA level and up to 0.3 s at the 429 kA level (5). Similar experiments in Alcator-C at moderate plasma density have maintained a 160 kA current for 0.15 s. A figure of merit for current drive is the product of the current and the average density divided by the power required to drive that current. Values of 8 x 10^{13} amp/watt cm^3 in PLT and

TABLE 2.1-3

Heating Power

Device	Type of Heating* (Frequency)	Installed Power (MW)	Injected Power (MW)
PLT	NBI	3	2.5
	ICRF (25 MHz)	1.5	NR
	ICRF (42 MHz)	NR	3
	LH (800 MHz)	.8	NR
PDX	NBI	6	5.5
Alcator-C	LH	4	.6
ISX-B	NBI	3	2.5
Doublet-III	NBI	7	2.8
	ECRH	2	NR
ASDEX	NBI	NR	3.1
TFR	ICRF (60 MHz)	2	1.2
T-10	ECRH (86 GHz)	.8	NR
	(100 GHz)	.35	NR
JFT-2	NBI	2	1.7
	ICRH	1	.8

*NBI = Neutral Beam Injection, ICRF = Ion Cyclotron Range of Frequencies, LH = Lower Hybrid, ECRH = Electron Cyclotron Resonance Heating, NR = Not Reported

1.9×10^{13} amp/watt cm^3 in Alcator-C have been achieved at low density so far. Unfortunately, in PLT at least, the value of this figure of merit drops drastically for densities above 7×10^{12} cm^{-3}. The STARFIRE reactor design called for a value of 1.2×10^{13} amp/watt cm^3 at $\overline{n} = 8 \times 10^{13}$ cm^{-3}. Consequently, considerable improvement in the effectiveness of current drive is required at high plasma densities.

Impurity accumulation in plasmas has been a major concern for many years. The radiation loss from impurities can quench a thermonuclear burn or make it impossible to ignite the plasma. Sources of impurities are desorption from the walls and sputtering by ion impact. Considerable progress has been made in cleaning and conditioning the walls so that the quantity of impurities attached to the walls has been reduced. This has resulted in cleaner initial plasmas with reduced radiation loss. Sputtering of wall material by energetic ions is still potentially a problem in higher power density and hotter plasmas. The magnetic divertor is a concept for removing impurities from the plasma. Poloidal magnetic divertors have been tested on PDX and ASDEX and a bundle divertor has been tested on the DITE tokamak. The results have been favorable, especially for the poloidal divertor. Clean plasmas with very little impurity radiation loss have been produced.

The cost and technological complexity of divertors have led to the invention of a new concept, the pumped limiter. The pumped limiter is shaped so that most of the plasma strikes the front face and is recycled directly into the main discharge. Some of the plasma flows through a restricted channel and strikes a neutralizer plate located in front of vacuum pumping ports. The neutral gas produced has a substantial pressure and is removed by the vacuum pumps. The portion of the plasma that is pumped removes with it impurities and, in a reactor, the helium ash produced by the fusion reactions. This concept is just beginning to be tested. Small scale tests have shown the buildup of neutral pressure but a full scale test in a major tokamak has yet to be performed. A fundamental question is whether the edge plasma contacting the pumped limiter can be kept cold enough so that sputtering of the limiter material is not excessive. If there is too much sputtering, then the resulting impurity level in the core of the plasma could be too large or the ablation of the pumped limiter itself could be excessive. Tests of large scale pumped limiters are scheduled for PDX and Textor in 1983-84.

2.1.1.3 Next Generation Facilities. Based upon the results of the present generation of tokamaks, a series of larger and more powerful tokamaks are being constructed; these are listed in Table 2.1-4. Of particular significance to the U.S. program is Tokamak Fusion Test Reactor (TFTR); a schematic of it is shown in Fig. 2.1-2. This device employs neutral deuterium injection into a tritium plasma. Collisions between the injected fast deuterium ions and the colder tritium ions will produce fusion reactions; the goal is to have energy breakeven, i.e. the power output greater than or equal to the injected power. In addition to being larger and hotter than present tokamaks, TFTR will have the added effect of fusion born energetic alpha particles being produced and affecting the plasma behavior. This represents a new step for magnetic fusion research. TFTR achieved its first plasma on December 24, 1982 and tritium tests will be performed in 1985 after extensive tests with non-tritium plasmas have been conducted.

The Joint European Torus (JET) is currently under construction at Culham Laboratory in England; it is scheduled to have the first plasma in 1983. It will also undertake deuterium - tritium operation in 1988. JT-60 is a similar experiment under construction in Japan to be operated in 1984. It is planned to reach conditions in a hydrogen plasma sufficient for breakeven if a DT plasma were used. The other devices in Table 2.1-4 are also of this category; they are designed to provide necessary physics information under thermonuclear plasma conditions without the technological problems associated with tritium and neutron production.

TABLE 2.1-4

Authorized Future Facilities

			Device		
	TFTR	JT-60	JET	Tore-Supra	T-15
Country	USA	Japan	Euratom	France	USSR
Date begin operation	1982	1985	1983	1985/86	1985
Major radius (m)	2.5	3	3	2.15	2.40
Minor radius (m)	.85	.95	1.25	.70	.70
Magnetic field (T)	5.2	4.5	3.5	4.5	5
Plasma current (MA)	2.5	2.7	4.8	1.7	1.7
Heating type	NBI	NB/LH	NB/LH	NB/ICH	NB/ECH
Heating power (MW)	33	30	32	15	14
Density (cm^{-3})	$\sim 10^{14}$	$\sim 10^{14}$	$\sim 10^{14}$	$\sim 10^{14}$	2×10^{14}
Ion Temp. (keV)	10	5-10	5-10	1-4	5-10
Electron Temp. (keV)	5-10	5-10	5-10	1-4	5-7
$n\tau_e$	$\gtrsim 3 \times 10^{13}$	$2\text{-}6 \times 10^{13}$	10^{14}	NR	10^{14}
β (anticipated)	2%	4%	4%	NR	NR

NR = Not Reported

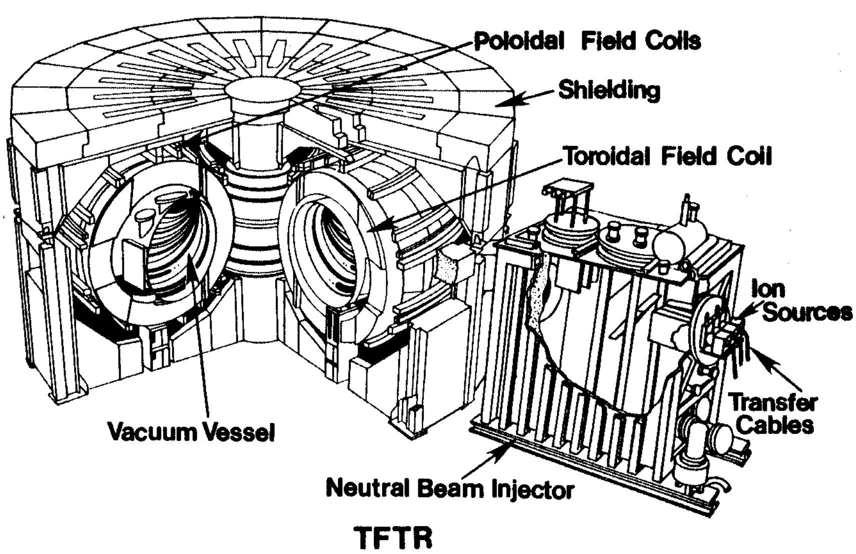

Fig. 2.1-2. A schematic of the Tokamak Fusion Test Reactor at Princeton Plasma Physics Laboratory.

2.1.1.4 Summary. There has been considerable progress in the tokamak confinement approach over the past 5 years (1978-82). The present generation of devices, which began operation in the middle to late 1970's, have yielded valuable information on high β operation, transport losses, and impurity control. There have been considerable advances in the amount of heating power applied to the plasma and in the plasma temperatures and densities achieved. Current drive for steady-state operation has been demonstrated, although improvements in the efficiency are required.

There still remain, however, important questions which need to be answered before the reactor suitability of the tokamak can be established. These include plasma transport at higher temperatures and under high power neutral beam injection, plasma stability at higher β, and impurity control in longer pulse, higher power density plasmas. Furthermore, the problem of plasma disruptions, where the plasma suddenly goes to the walls and the discharge terminates, must be solved in larger devices where the potential for disruptions which damage the walls and limiter is greater. The next generation of tokamak facilities (TFTR, JET, etc.) will contribute greatly to answering those questions.

2.1.2 Tandem Mirrors

The tandem mirror is a configuration invented independently in 1976 by Dimov et al. (6) in the U.S.S.R. and by Fowler and Logan (7) in the U.S. This concept utilizes the positive ambipolar potential of a mirror confined plasma at each end of a long central cell to provide axial confinement of the central cell plasma. A number of devices have been built to test this concept; they are listed in Table 2.1-5 and their main parameters are given in Table 2.1-6. References (8) and (9) provide a general overview of tandem mirror facilities.

TABLE 2.1-5

Major Tandem Mirror Facilities Completed in the 1978-82 Period

Experiment	County	Date of Completion	Main Mission
GAMMA-6	Japan	1978	Test tandem mirror concept
TMX	US	1979	Test tandem mirror concept
Phaedrus	US	1981	Study RF heating in tandem mirrors
AMBAL	USSR	under construction	Study NB heating and tandem mirror confinement

TABLE 2.1-6

Standard Tandem Mirror Parameters

Best Plasma Parameters Achieved (Not Simultaneously)

Parameter	GAMMA-6	Phaedrus	TMX	AMBAL* (design values)
Central Cell				
Density, cm^{-3}	1×10^{13}	6×10^{12}	3×10^{13}	1×10^{13}
Ion temp., eV	40	35	250	500
Electron temp., eV	15	40	200	1000
Plasma radius, cm	2.4 x 42	12	30	27
On-axis magnetic field, T	1.5	.6	2	2.25
Plug				
Density, cm^{-3}	1.5×10^{13}	1×10^{13}	4×10^{13}	3×10^{13}
Confining potential, eV	35	40	300	1100
Midplane magnetic field, T	.4	.26	1.0	1.5
Power				
ICRF power, kW	8	400	---	---
ECRF power, kW	---	50	---	---
Neutral beam power, MW	.1	1.8	4	2.5
Neutral beam energy, keV	13-14	4.5	13	25

*Since AMBAL is still under construction, the parameters given are design values.

Although the concept came originally from the USSR and the US, the first operating tandem mirror was GAMMA 6 at Tsukuba University in Japan (10). This small facility was the first to demonstrate electrostatic potential plugging of a tandem mirror central cell, and hence provided the initial concept verification.

The Tandem Mirror Experiment (TMX) at LLNL is the largest tandem mirror operated through 1982 (11). It ran during 1979 and 1980, then shut down to be upgraded (see next section). Figure 2.1-3 shows the axial magnetic field, plasma density and potential profiles, plus the magnet set. Important results

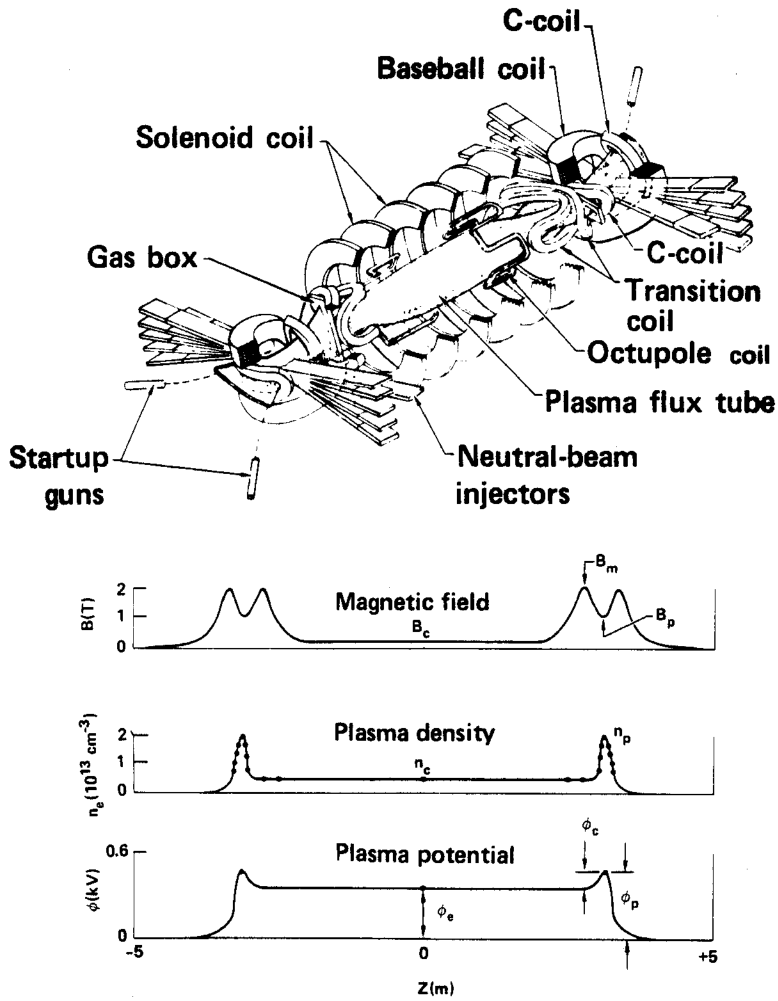

Fig. 2.1-3. TMX configuration and axial profiles of magnetic field, density, and potential.

include the demonstration of electrostatic confinement of the central cell plasma at relatively high temperatures and of high beta (≡ plasma pressure/ magnetic field pressure) operation. Note that the central cell neutral beams were used to achieve a peak beta of 0.4, but that this was not the normal mode of operation. A comprehensive summary of results from TMX is given in Ref. (11).

Phaedrus, a medium-sized device at the University of Wisconsin, aims primarily at testing the role of RF heating and the interaction of RF and neutral beams in tandem mirrors. To date, the main mode of operation has been the RF sustained mode, where the plasma is sustained for > 20 ms with no neutral beams. The mode may have important implications for future devices, since neutral beam systems generally create more machine design problems than do RF systems.

AMBAL, a TMX-size tandem mirror in the USSR, will soon finish construction. Its goals are similar to those of TMX: to investigate microstability, MHD stability at high beta, and radial transport. Complementing AMBAL is a strong, well-developed tandem mirror theory program in the U.S.S.R.

The Symmetric Tandem Mirror (STM) at TRW is an axisymmetric tandem mirror. Should the concept prove feasible, reactor magnet design problems would be greatly eased. The chief purpose of STM is to test the physics of using ECRF heating to create hot electron rings or disks, and thereby achieve MHD stability. This is also relevant to EBT (see Section 2.1.3.2). The experiment is presently at an early stage.

Besides the tandem mirrors, a multitude of other mirror confinement experiments exist around the world. These include:

- RFC-XX, a double-cusp device at Nagoya University of Japan.
- LAMEX, the Large Axisymmetric Mirror Experiment at UCLA, which consists of multiple, close-packed cusps.
- MMX, the Multiple-Mirror Experiment at UC-Berkeley.

2.1.2.1 Thermal Barrier Tandem Mirrors. Reactor systems studies of tandem mirrors indicated that the standard mirror is marginal as a reactor concept and requires very high technology in the end plugs. This is discussed further in Section 2.3.3. As a response to these problems, the thermal barrier was proposed in 1979 (12). The function of the thermal barrier is to provide thermal isolation of the electrons in the end plug from those in the central cell; this reduces the power needed to heat the plug electrons and increases the confining potential. Consequently, a variety of newer facilities incorporating thermal barriers are under construction and one is beginning operation. The essential differences among the various designs occur in the placement of plug, thermal barrier and anchor. Figure 2.1-4 shows axial magnetic field and potential profiles for four different facilities. Parameters are given in Table 2.1-7. The primary goal of all of these devices is to investigate relatively high temperature and density tandem mirror plasmas in a thermal barrier mode of operation.

TMX-U, an upgrade of TMX at LLNL, has the most compact configuration, with plug, barrier, and anchor all in the same cell (13). Unfortunately, this does not scale well to a reactor since the very high field required on the central cell side of the barrier cannot be made in minimum-B geometry. However, it should be the first machine to verify thermal barrier operation. It has just begun operation (late 1982) and the chief results to date are that the plug neutral beams give a sloshing ion distribution and that ECRF heating has occurred. TMX-U will also be the first machine to explore questions relating to the microstability and MHD stability of a thermal barrier tandem mirror.

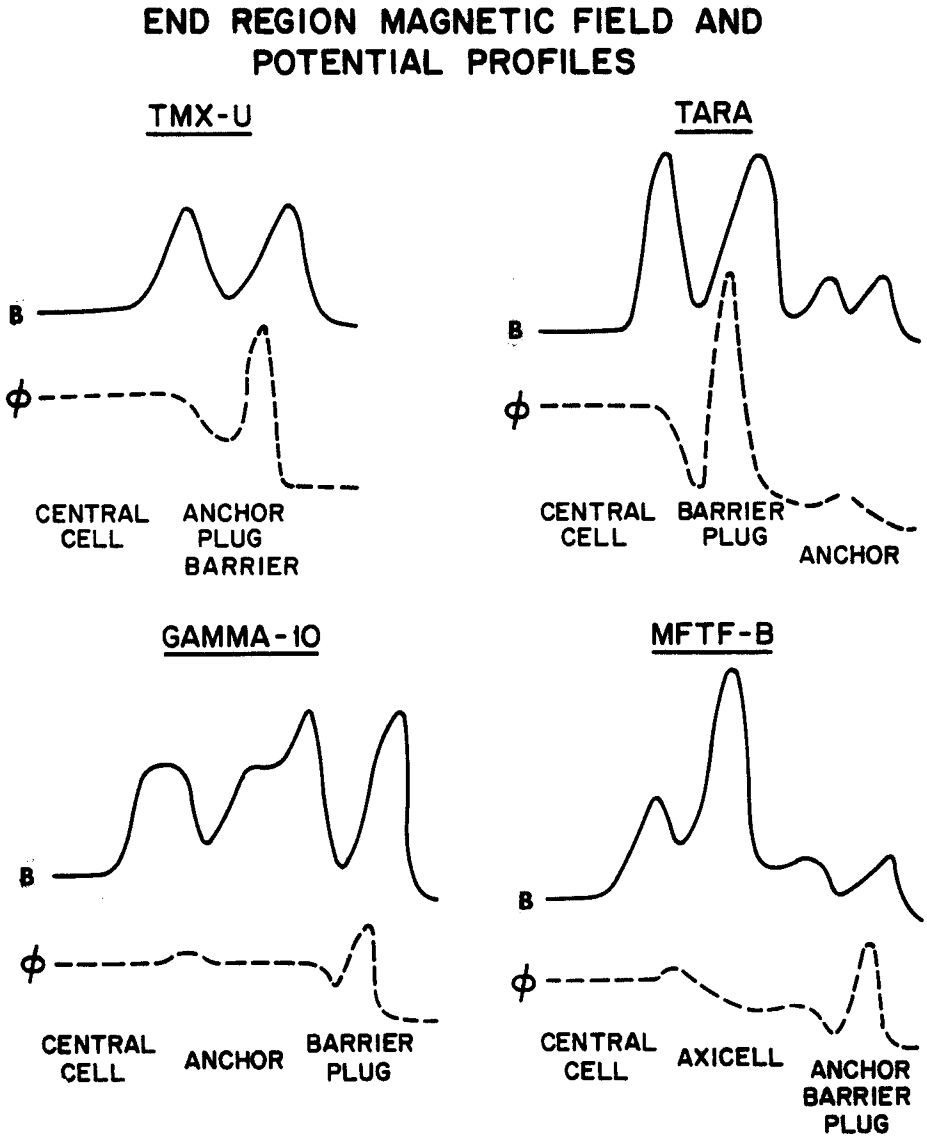

Fig. 2.1-4. Axial magnetic field (B) and potential (ϕ) profiles for TMX-U, TARA, GAMMA-10 and MFTF-B.

GAMMA-10, in the final stages of construction at Tsukuba University in Japan, will be a machine with the anchor between the central cell and an axicell which serves as the barrier and plug (14). This arrangement is excellent for testing radial transport and should be immune to trapped particle modes. It may also test the possibility of putting the barrier and plug in separate cells. However, it will be difficult to scale up to a reactor since the yin-yang magnets are then exposed to high neutron fluxes.

TABLE 2.1-7

Thermal Barrier Tandem Mirror Experiment Design Parameters

Parameter	TMX-U	GAMMA-10	TARA	MFTF-B
Year of Operation	1982	1983	1984	1986
Central Cell Plasma				
Density, cm^{-3}	2×10^{13}	1×10^{13}	4×10^{12}	4.8×10^{13}
Ion temp., keV	0.9	1	0.4	15
Plasma radius, cm	~ 15	13	12	30
Length, cm	800	660	1000	1650
Beta	0.25	~ 0.1	0.03	0.5
Magnetic Fields				
Central cell, T	0.3	0.4	0.2	1
Barrier solenoid peak, T	---	3	3	12
Yin-yang peak, T	2	2.1	1	6
Power				
ICRF power, kW	---	1000	---	---
ECRF power, kW	800	400/400	9/2	800/800
ECRF frequency, GHz	28	35/28	11/28	56/(35,28)
NB power, kW	3315/1980	1800/1500	2500	3850
NB energy, keV	15/11	60/25	20	80

TARA, under construction at MIT, will run in a mode where the anchor is outside of an axicell in which the barrier and plug are formed (15). This allows the use of relatively low-field yin-yang coils for the anchor and should also reduce radial transport to a minimum since almost all ions are confined within an axisymmetric region. Unfortunately, the configuration may be susceptible to trapped-particle instabilities and testing that theory is a prime TARA objective. Like GAMMA-10, TARA has the versatility to run in a variety of modes, including a hot-electron anchor mode. Parameters in Table 2.1-7 are given for the neutral beam anchor reference mode.

Presently, the Mirror Fusion Test Facility, MFTF-B (see Fig. 2.1-5), at LLNL is the largest authorized thermal barrier tandem mirror (8). It is scheduled to begin operation in 1986 and one of the superconducting yin-yang magnets has already been successfully tested. MFTF-B parameters are also given in Table 2.1-7 for the MARS mode (see Section 2.3.3) of operation, but the device can test a number of schemes. MFTF-B will test the ability of a thermal barrier tandem mirror to approach Q (fusion power/injected power) = 1, if DT fuel were to be used. Presently, only deuterium fuel operation is authorized.

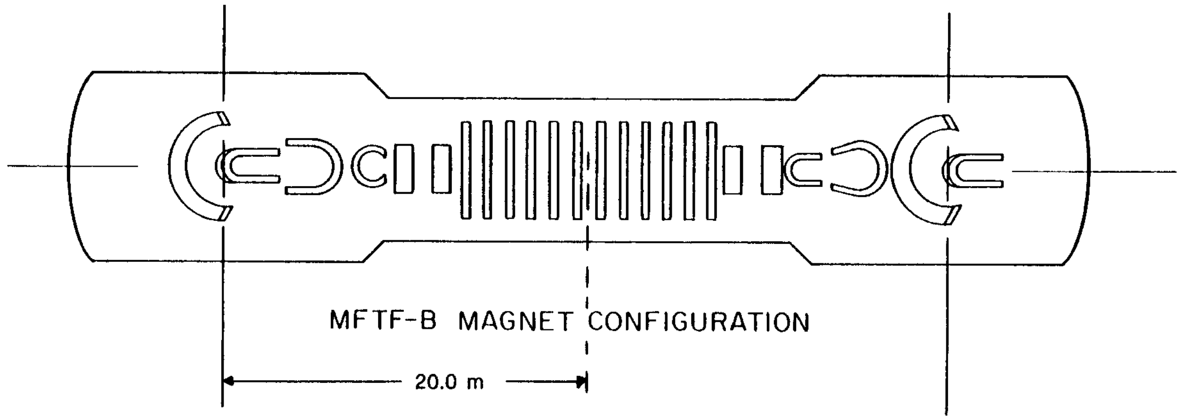

Fig. 2.1-5. The MFTF-B Tandem Mirror Experiment.

2.1.2.2 Key Issues. The basic concept of electrostatic "plugging" of the central cell axial plasma loss has been demonstrated in GAMMA-6, TMX and Phaedrus in reasonable agreement with theoretical analysis. The next goal is to establish a thermal barrier between the central cell and plug. This concept has received considerable theoretical attention, but experimental demonstration of a thermal barrier has not been attempted as of the writing of this report. Current program plans call for the creation and measurement of a thermal barrier in TMX-U during 1983.

As in the tokamak, beta (the ratio of plasma pressure to magnetic pressure) plays an important role in tandem mirror reactor economics--high central cell beta implies efficient use of the magnetic field. For tandem mirrors, present reactor concepts call for volume-averaged beta of ≅ 30% or more in the central cell. The TMX experiment has achieved an on-axis beta value of about 40% using neutral beam heating in the central cell.

Microstability of the mirror-confined plasma in the end plugs has been the fundamental key issue of mirror plasma confinement for a long time. The ions necessarily have a velocity distribution somewhat analogous to the inverted energy levels in lasers. This produces a free energy source for plasma instabilities which can dump ions into the loss-cone and seriously degrade plasma confinement. Some, but not all, of the possible instabilities have been observed in experiments. Theoretical indications are that thermal barrier tandem mirrors can be made stable with appropriate design and early indications from the TMX-U experiment support this conclusion, but a full scale demonstration of a microstable mirror plasma has yet to be accomplished. In advanced thermal barrier concepts, hot electrons are used to enhance the potential dip of the thermal barrier. This opens up the possibility of microinstabilities driven by the electrons.

Recently, concern has arisen that an instability will occur in designs where most of the ions are trapped in regions of bad average curvature, and don't enter the good curvature of the anchors. In fact, this concern caused a change in the MFTF-B design from the TARA configuration to the present configuration. Current theory predicts that this instability can be avoided by proper design.

For end-plugging of the central cell to be effective, radial plasma loss must be minimized. Thus, a large amount of theoretical effort has gone into the investigation of radial transport. Basically, nonaxisymmetric magnetic fields can cause enhanced radial diffusion. Careful design of the non-axisymmetric field regions can minimize this effect, and a great deal of effort has gone into optimizing the magnetic field geometry on MFTF-B and other devices. Most experiments to date have had large axial losses which obscured radial loss, but TMX saw an axial electron loss current--implying some radial ion loss.

2.1.2.3 Summary. The tandem mirror concept was invented about seven years ago. During the last five years there has been experimental confirmation of the basic principles of the tandem mirror, and the initiation of next generation facilities to push the plasma density and temperature into a more reactor-relevant regime. The thermal barrier was invented to enhance the reactor potential of the tandem mirror. It is still awaiting experimental confirmation, but significant progress has been made on a number of fronts: obtaining a microstable plug, achieving high β, significant electron heating. This is remarkable progress for such a new concept. The next generation facilities should go a long way towards establishing the reactor suitability of the tandem mirror.

2.1.3 Alternate Concepts

2.1.3.1 Stellarators. The stellarator is a toroidal magnetic confinement concept similar in principle to the tokamak, but with additional magnetic coils outside the plasma which eliminate the need for a current in the plasma. These coils provide the poloidal magnetic field required for plasma confinement which are usually helical and are interlocked with the main toroidal magnets. This causes some unique construction and maintenance problems. Because a plasma current is not required, the stellarator can operate steady-state and does not have the problem of plasma disruptions which plagues tokamaks. The stellarator was invented in the U.S. in the 1950's, but with the shift in emphasis to tokamaks, only a small U.S. program remains. There are substantial stellarator programs in Europe, Japan, and the U.S.S.R.

A closely related concept is the torsatron, which also has helical coils outside the plasma. For the purposes of this report, it is not necessary to distinguish between them. Torsatrons are included here as a part of the stellarator category. Table 2.1-8 lists all the stellarator type devices which are presently operating or under construction in the world and gives their major parameters. It should be noted that only two devices are in the U.S. The major facilities are CLEO, L-2, JIPP T-2, Wendelstein VII-A, Heliotron E and URAGAN-III. These devices are all quite different with W VII-A and JIPP T-II having low magnetic shear and moderate rotational transform, CLEO and L-2 with more shear and transform, Heliotron E with high shear and transform and Uragan-III with high shear and moderate a transform.

There has been a recent surge of interest in stellarators in the past 5 years. The two main facilities responsible for this are Wendelstein VII-A and

TABLE 2.1-8

Present Stellarator Devices

NAME		LOCATION	R (cm)	r (cm)	B (kG)
Wendelstein VII-A		W. Germany	200	10	~ 40
Heliotron DM		Japan	50	5	~ 10
Heliotron D		Japan	105	10	~ 5
Heliotron E		Japan	220	≅ 21-40	~ 20
JIPP-I		Japan	50	7	~ 4
JIPP T-II		Japan	91	17	~ 30
Cleo		UK	90	13.5	~ 20
IMS	*	USA	40	5	6
Proto-Cleo		USA	40	5	~ 5
Chrystall-2	*	USSR	36	8.7	25
Uragan-II		USSR	1035	10	20
Uragan-III	*	USSR	100	17	30-45
Sirius		USSR	600	10	20
Saturn-1		USSR	36	8.7	10
VINT-20		USSR	31.5	7.2	20
M-8		USSR	8		
L-2		USSR	100	11	
R-0		USSR	50	5	~ 8
RT-2		USSR	65	4	~ 20

R = major radius
r = minor radius
B = toroidal magnetic field
* These devices under construction

Heliotron E. Both of these devices have been operated in a currentless plasma phase and have displayed no plasma disruptions or serious instability activity. An extremely low level of small-scale turbulence and transport losses which are lower than in ohmically heated discharges of similar plasma conditions has been observed. In the latest Wendelstein VII-A (16) experiments, operating with a high density deuterium plasma at a magnetic field of 3.2 T and with neutral beam injected power of 1 MW, scientists have obtained conditions of $\beta \leq 1\%$, a density $\geq 10^{14}$ cm^{-3}, an ion temperature of 1 keV, and an electron temperature of 0.7 keV at the center of the plasma, and a confinement time ≈ 100 msec. The high density was achieved with deuterium pellet injection. These are truly remarkable results for such a relatively small device.

Currentless plasma operation was achieved in Heliotron-E (17) with 200 kW of ECRH at 28 GHz for a duration of 40 ms. The field was reduced to 1.0 T so that the electron cyclotron resonance would occur on the magnetic axis. The achieved conditions are an electron temperature of 1.1 keV and an ion temperature of 120 eV on axis, and an average density of 5 x 10^{12} cm^{-3}. It is observed that the heating rate is higher by a factor of 2 than in tokamaks. This machine can also be operated with neutral beam injection of up to 1.6 MW. Under those conditions, the temperature is ~ 0.66 keV at the center and a density of 3 x 10^{13} cm^{-3} with the central beta reaching 1%. At this power level, no plasma energy saturation is observed.

A new concept that has been proposed is the modular stellarator which has no helical coils, but instead uses deformed toroidal field magnets to generate both the toroidal and poloidal magnetic fields. The IMS (Interchangeable Modular Stellarator) at the University of Wisconsin, is the first device to test modular coils. It is designed to duplicate the field characteristics of Proto-Cleo with a set of modular coils, and utilizes the same vacuum chamber and power supplies. It began operation in 1983.

The W VII-A facility will be upgraded with another device, W VII-A-S (standing for advanced stellarator). This system will have a normal copper modular coil set of 45 coils with a major radius of 2.1 m, a minor radius of 20 cm and a B_0 = 3 T. As planned, this device will have 0.6 MW of ECRH heating between 68-86 GHz, 3 MW of ICRH at 35-70 MHz and 1.2 MW of 40 keV neutral beams. The anticipated plasma parameters are an ion temperature of 2 keV at density of 10^{14} cm^{-3} with average β values exceeding 2%.

2.1.3.2 Elmo Bumpy Torus. The Elmo Bumpy Torus (EBT) is a confinement concept utilizing a series of toroidally linked magnetic mirror cells. Normally, a pure toroidal field is unstable for plasma confinement. The key ingredient in EBT which produces a stable plasma equilibrium is a series of hot electron rings generated by microwaves and confined in the individual mirror cells. If this ring has sufficient energy density then it can "dig" a magnetic well, which can confine a warm toroidal plasma. There are two major facilities of the EBT type. The first is the EBT-S (18) is an upgrade of the original EBT-1 at Oak Ridge National Laboratory. The second facility is the Nagoya Bumpy Torus (19) (NBT) at Nagoya University in Japan. The parameters of these two devices and currently planned next generation devices are listed in Table 2.1-9. Another facility, Symmetric Tandem Mirror (STM) at TRW can also produce hot electron rings in mirror cells and provide physics data for both the EBT and tandem mirror programs. It is described in Section 2.1.2.

The plasma parameters given in Table 2.1-9 are modest in comparison to those achieved in other confinement concepts. This is partly because the EBT facilities are modest in comparison to other facilities (e.g., tokamaks). Present microwave sources are limited to 28 GHz and this limits the magnetic field strength to about 9 kG through the cyclotron resonance condition. Further developments in gyrotrons or free electron lasers to produce high power high frequency microwaves will allow higher magnetic fields and, therefore, better plasma confinement in EBT type devices.

TABLE 2.1-9

Machine and Plasma Parameters of Various Bumpy Torus Devices

	In Operation		Planned*	
	EBT-S	NBT-1	EBT-P	NBT-2
Major radius (m)	1.5	1.6	4.5	5.0
Minor radius (m)	0.1	0.1	0.18	0.3
Number of cells	24	24	36	36
Resonant magnetic field (T)	1.0	0.3	2.1	2.5-3.5
Plasma density (cm^{-3})	$0.5\text{-}2\times10^{12}$	$\sim 10^{12}$	1.7×10^{13}	3×10^{13}
Electron temperature (keV)	0.2-1.0	0.1-0.2	2	1-2
Ion temperature (keV)	~ 0.02	$\lesssim 0.1$	0.4	1-2
$n\tau$ (sec/cm^3)	$\sim 10^9$	10^9	5×10^{11}	$\sim 10^{12}$
Microwave frequency (GHz)	28	8.5	60-110	60-110

*The plasma parameters for the planned devices are anticipated, but as of this writing, the EBT-P has been indefinitely postponed.

The EBT concept provides an alternative to other toroidal concepts. It is truly steady-state, does not have the problem of plasma disruptions which plagues tokamaks, and has a modular configuration desirable for maintenance purposes in a reactor.

2.1.3.3 Compact Tori, Reversed Field Pinches and Related Concepts. The common characteristic of schemes discussed in this section is that conceptual reactor versions of such devices tend to have very high fusion power densities (~ 100 watts/cm^3). This is economically attractive and leads to relatively small devices. The chief technological difficulties lie in the resulting high neutron wall loads (~ 10 MW/m^2) and surface heat loads (~ 5 MW/m^2) which produce associated materials damage problems. On the other hand, since a major problem with advanced fuel reactor designs is low power density, these devices hold interesting possibilities for low neutron yield fuels (see Section 4.3).

Compact tori differ from tokamaks, stellarators, and torsatrons in that no magnetic field coils intersect the central region of the torus. Generic magnetic field geometry is shown in Fig. 2.1-6, which shows a field-reversed mirror (FRM) configuration. Compact tori are of two general types: (1) the field-reversed configuration (FRC), which has negligible toroidal magnetic field, and (2) the Spheromak, which has comparable toroidal and poloidal fields. Reversed Field Pinches (RFP) are toroidal devices in which the toroidal magnetic field reverses direction on the outside of the plasma.

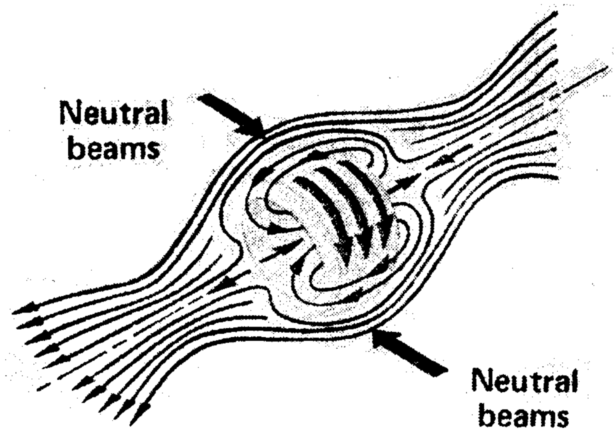

Fig. 2.1-6. Field reversed mirror configuration.

A fairly extensive compact torus and RFP experimental program exists, including devices in England, Italy, Japan, and the U.S. A major U.S. RFP experiment is ZT-40M at LANL (20), which has achieved ion temperatures, T_i, of 300 eV at $\beta \sim 0.1$ for ~ 8 ms. A proposed upgrade called ZT-40U would reach ~ 5 keV. The field-reversed theta-pinch FRX-C at LANL (21) has reached an ion temperature of 150 eV for ~ 100 μsec. The CTX spheromak experiment at LANL (22) has achieved densities ~ 10^{14} cm^{-3} with temperatures ~ 10 eV for ~ 200 μsec. A larger spheromak, S-1, is presently under construction and it aims for an electron temperature of about 100 eV.

Work is underway on high field, high density devices with toroidal plasmas and chambers similar to tokamaks. The Riggatron, at INESCO in San Diego, is a very high field, copper coil tokamak. OHTE (23), funded by General Atomic, is an RFP surrounded by helical magnetic field coils. Both possibly could lead to small, economic reactors.

A number of compact torus reactor studies have been done, including a reversed-field pinch reactor (RFPR) and compact reversed-field pinch reactor (CRFPR) by LANL (24), an FRM reactor by LLNL, General Atomic, and Pacific Gas and Electric Co. (25), and the TRACT reversed-field theta pinch reactor by Mathematical Sciences Northwest, Inc. (26) and moving ring reactors by Nagoya and PG&E (27, 28). Parameters for these devices look encouraging (see Table 4.3-2), but the existing physics data for compact torus plasmas are very far from the reactor regime.

2.1.3.4 Conclusions - Alternate Concepts. In summary, the past 5 years have seen considerable competition between the alternate concepts for a relatively small piece of the fusion research budget. At times, stellarators, EBT's, or RFP's seem to be most favored, only to be replaced by one of the other competitors. The compact torus designs such as Riggatron or OHTE, also have been considered as possible backups for the mainline tokamak or tandem mirror approaches. At the end of the 1978-82 period, the picture is still not clear as to which of the alternates will eventually go on to proof of principle stage.

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2.2 TECHNOLOGY FOR MAGNETIC FUSION

The realization of magnetic fusion requires advances in technology, as well as in plasma physics. Among the relevant technological areas are neutral beams, RF, and microwave power sources for plasma heating, superconducting magnets, tritium containment and handling, and properties of materials under neutron irradiation. In addition, toroidal systems generally require either a divertor or limiter for edge plasma control, and direct convertors are envisioned for open systems. These components are generally subjected to locally high heat and particle bombardment fluxes. We review the current status and trends of these areas in the following sections.

2.2.1 Neutral Beams

2.2.1.1 Introduction. At present, the best understood technique of elevating the temperature of a magnetically confined plasma is the injection of intense beams of neutral hydrogen or deuterium atoms. On entering the plasma, a portion of these energetic neutrals are ionized and the resulting trapped ions transfer their energy to the plasma particles. Neutral beams have additional important roles in mirror fusion machines such as providing sources of energetic particles with high transverse velocities (to the magnetic field). These ions contribute to the maintenance of magnetically-trapped ion populations while, in a complementary role, injection at precise angles (to the field) causes the "pumping" of deleterious trapped ions by charge exchange. Neutral beams also supply important plasma heating and confinement functions during startup and provide one method of refueling steady-state fusion plasmas.

2.2.1.2 Positive Ions. Positive-ion hydrogen and deuterium beams can be converted to neutral beams with reasonable efficiency by passing them through vapor or gas neutralizers. However, the conversion efficiency decreases with increasing ion energy and becomes very inefficient above ~ 75 or 150 keV for H or D, respectively. A second disadvantage of positive-ion-based neutral beams is that the source-extracted ions consist of a mixture of atomic ions and undesired diatomic and triatomic ions. The latter, when accelerated and neutralized, form undesired neutral atoms of one-half and one-third the required energy. Table 2.2-1 illustrates the neutral beam injection parameters and requirements for several current and near term fusion devices. Note that, with the exception of INTOR, all these employ positive-ion-based technology which, in the near term, provides the only credible method of achieving the very high power densities required, especially for those machines operating in quasi-steady-state mode. Figure 2.2-1 illustrates a typical neutral beam line.

The near term objectives of the tandem mirror program in the U.S. will be met by the successful development of the 80 keV, 30-50 A, 30 s deuterium neutral beam sources required for operation of the tandem Mirror Fusion Test Facility (MFTF-B) at Lawrence Livermore National Laboratory (1). In 1980, a fractional-area (7 x 10 cm) accelerator module for this source at Lawrence Berkeley Laboratory (LBL) operated satisfactorily at 80 keV for 30 s and at 120 keV for 5 s, the pulse length being limited by the test stand capabilities. Construction of the full-scale module (10 x 40 cm) was completed in 1981 and initial testing at 80 keV for 2 s has been performed. Testing of the prototype ion source and beam line for the Tokamak Fusion Test Reactor (TFTR) at Princeton Plasma Physics Laboratory (PPPL) was completed at LBL in 1981 (2).

TABLE 2.2-1

Neutral Beam Parameters for Some Present and Near Term Fusion Devices

Device	Country	Projected Operating Date	Beam Energy (keV)	Extracted Current (A)	Pulse Length (s)	Monatomic Fraction	Sources Per Injector	Neutral Power Per Injector (MW)
PLT[a]/ ISX-B	U.S.	Operating	40	60	0.3/0.1	0.8-0.85	1	0.7-0.9
PDX[a]/ ISX-B(U)	U.S.	"	50/40	100	0.5/0.2	0.8-0.85	1	1.5-1.8
DITE[a]	U.K.	"	30	40	0.12	~ 0.8	1	0.6
D-III[a]	U.S.	"	80	85	0.5	0.8	2	3.6
TMX[b]	U.S.	"	17/20/ 40	60/250[c]	0.025	NR	NR	NR
TMX-U[b]	U.S.	"	17	240/360[c]	0.075	NR	NR	NR
JFT-2[a]	Japan	"	40	30	0.05	NR	NR	NR
TFTR[a]	U.S.	1983	120	65	1.5	0.8	3	7
JET[a]	Euratom	1984	80/160	60/30	10/20	0.8	8	5
T-15[a]	U.S.S.R.	1984	80	35	1.5	0.7	2	2.5
JT-60[a]	Japan	1984	75	35	10	0.75	2	1.5
AMBAL[b]	U.S.S.R.	1982	13	300/100[c]	100/500	NR	NR	NR
GAMMA-10[b]	Japan	1983	50	25[c]	100	NR	NR	NR
MFTF-B[b]	U.S.	1985	30/80	10/25/40[c]	30	NR	NR	NR
INTOR[a]	International	~ 1990-1995	175	100	10	0.85-0.9		15

a. Tokamak b. Tandem Mirror c. Each Plug NR = Not Reported

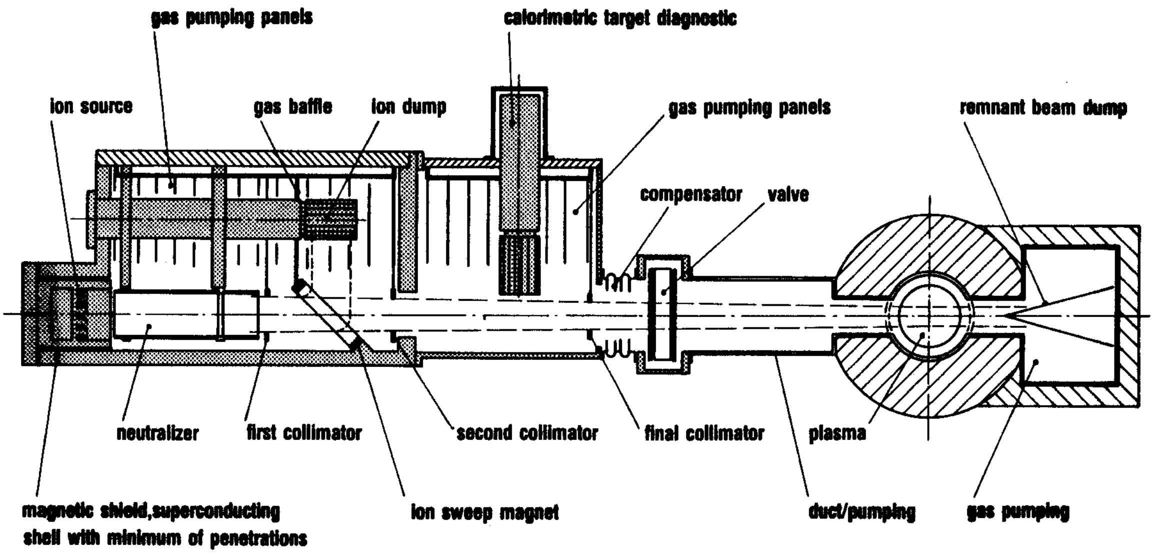

Fig. 2.2-1. Schematic of a Typical Neutral Beam Line.

In these tests, automatic conditioning of the ion source was achieved, 80% D^+ was produced from a self-conditioned ion-source and long pulse testing resulted in 1.5 s pulses at 110 keV and 1.0 s pulses at 120 keV.

Neutral beam injectors developed at Oak Ridge National Laboratory (ORNL) have been successfully utilized in various tokamak experiments. In 1978, four injectors operating simultaneously on the Princeton Large Torus (PLT) at PPPL delivered 2.4 MW of deuterium neutrals to the plasma raising the ion temperature to 6.5 keV. In 1979/80, two similar injectors were employed to heat the Impurity Study Experiment (ISX-B) plasma at ORNL to 1 keV and were each capable of operating at 40 keV, 6 A, for ~ 100 ms. Considerable improvement in performance has also been realized in the recently developed 50 keV, 100 A, 500 ms injectors for the Poloidal Divertor Experiment (PDX) at PPPL. A neutral beam power of 2 MW of deuterium has been measured at a 30 x 34 cm target. All of these ORNL sources have measured atomic (full energy) fractions of ~ 80-85% (3).

Advanced positive ion systems are now under development at ORNL to meet the multisecond, multi-MW neutral beam requirements of several near-term devices and are addressing the technology problems of ion source development, direct energy recovery and beam line design. Indirectly-heated LaMo cathodes capable of emitting 5-6 A cm^{-2} for several seconds have been employed in 120 V, 1200 A, 35 s plasma generators. An ion source incorporating these cathodes is being developed and will be incorporated in the 40 A, 80 keV alternative pre-prototype pump beams system for MFTF-B (4).

The neutral beam heating requirements for the joint U.S./Japan (JAERI) Doublet III tokamak experiment at General Atomic pose a significant technical challenge because of the simultaneous need for high power (~ 10 MW), high voltage (80 kV), low beam divergence (0.5° x 1.5°) and long pulse length (0.5 s) (5). These requirements led to the design, in collaboration with LBL, of a unique dual-source beam line containing two 80 kV, 80 A, 0.5 s H° sources arranged to inject through a single injection port. Commissioning of the first beam line was successfully completed in 1981 (6). Two additional beam lines are now under development and ultimately all four will be available for use on the Doublet III to Big Dee conversion, scheduled for operation in 1985.

The major effort of the neutral beam technology program in Europe has been applied to the JET tokamak program, where projected neutral beam requirements for the operational phases are as follows: hydrogen checkout (1984/5), ~ 7 MW of H° at 80 keV; deuterium phase (1986/88), ~ 7-10 MW of D° at 160 keV; DT phase (1988/90), 10 MW of D° at 160 keV. Culham Laboratory in England has responsibility for the development of the Plug-In Neutral Injectors (PINI) and operational tests with beam extraction were started in 1981; ion currents of 46 A at 82 keV were obtained (7). Problems with the manufacture of cooled extraction grids limited these tests to ~ 0.5 s pulse lengths. Manufacture of two prototype plasma sources for the PINIs--the bucket plasma source at Culham and the periplasmatron at Fontenay-aux-Roses Laboratory in France--is now approaching completion. A pre-prototype bucket source has been operated at Culham with current density of ~ 0.2 A cm^{-2} for 5 s and is due to incorporate a magnetic mass filter designed to enhance the fraction of atomic ions in the extraction region.

Neutral beam development in Japan has been marked by considerable progress since 1977. In 1978, successful tests of positive ion sources were conducted at the Universities of Nagoya (30 keV, 10 A, 0.1 s) and Tsukuba (20 keV, 35 A, 0.003 s) and have led to the development and incorporation of 40 keV, 33 A, 0.05 s beams on the Japanese Atomic Energy Research Institute's (JAERI) present experimental tokamak JFT-2. Programs are also established at the Universities of Kyoto and Tokyo in 30 keV neutral beam development. The major near term fusion device now under construction in Japan is the JT-60 tokamak. The NBI system will comprise fourteen 75 keV injectors affording a total heating power of ~ 20 MW for 10 s. A prototype 75 keV injector unit has been constructed and was tested at 35 A for ~ 0.1 s in 1981.

2.2.1.3 Negative Ion Sources. The two major disadvantages of positive-ion-based neutral beams (low neutralization efficiency at higher beam energies and multi-atomic-species beams) have promoted research into negative-ion-based sources. Negative ions can be efficiently neutralized almost independently of energy while providing mono-energetic beams. They present the only viable method of supplying the high energy beams (~ 200-500 keV) required for future advanced magnetic mirror devices.

At present, however, yields from negative ion sources are severely limited by the fact that negative ions produced in volume or surface-type discharge processes are readily destroyed by the discharge itself. In addition, those which survive must be separated from free electrons before reaching the acceleration system, a process which significantly perturbs the ion optics. These formidable limitations would imply that applications of negative ion technology are at least 5 years away.

Two separate types of negative sources are under development at ORNL utilizing electron addition to positive ions incident on cesiated surfaces (3). One of these employs a Penning discharge in a modified Calutron positive-ion source. The positive ions are accelerated onto a cesium surface from which a 50 mA cm^{-2}, 20 keV, 5 s negative ion beam has been extracted. The other concept utilizes a modified duo-PIGatron plasma generator in conjuction with a cesium convertor.

Up to 1979, the LBL/LLNL negative-ion-based neutral beam program concentrated on a system in which negative ions were produced by double charge-exchange on sodium. A beam of 10 keV D^+ ions was converted to 2 A of D^- ions in a supersonic sodium vapor jet. At present, the emphasis is on a self-extraction source in which negative ions are produced on a biased Cs-loaded convertor plate imbedded in the plasma; electrons are trapped by a confining magnetic field. This source is operating in quasi-steady-state (~ 10 minutes) and is producing ~ 0.5 A of 200 eV negative ions (3).

In addition, Brookhaven National Laboratory in the U.S. is investigating alternative accelerator designs based around a hollow cathode ion source (see Table 2.2-2). At Culham Laboratory in England, work is progressing on the extraction of negative ions directly from the hydrogen discharge rather than from a cesiated surface. Negative ion research is also underway in Japan and at Novosibirsk in the U.S.S.R.

Table 2.2-2, taken from Ref. (1), outlines the recent progress and current status of the U.S. negative ion technology program. Although its intermediate goal is a 1 MW demonstration about 1986, it can be seen that a negative-ion-based neutral beam source and its accelerator (for $V > 200$ kV) must undergo significant development before such a high power density system can be demonstrated.

2.2.2 Wave Heating Technology

2.2.2.1 Introduction. Probably one of the most interesting features of magnetic confinement fusion in the last five years has been the rapidly growing interest in the utilization of radio frequency (RF) waves. The major potential use of RF energy for plasma heating has been expanded by its capacity for sustaining current drive in tokamaks and for potential enhancement and thermal barrier maintenance in tandem mirror machines. Wave heating can be employed in various modes, including electron-cyclotron resonance heating (ECRH), ion-cyclotron resonance heating (ICRH) and lower hybrid heating (LHH). Since the two resonance zones, ECRH and ICRH, allow selective heating of electrons and ions, their frequency regimes differ by about three orders of magnitude and, therefore, their respective technological requirements are significantly different. As a pertinent example of the current emphasis on RF and its potential displacement of neutral beams as the primary heating mecha-

TABLE 2.2-2

Current Status of Negative-Ion-Source Development in the U.S. - July 1982 (1)

	LBL	ORNL	BNL
Source	self-extract	Calutron	hollow cathode
Plasma Pulse Length	10 min.	10 s	15 min
Beam Pulse Length (s)	7	5	1
Voltage (kV)	34	18	12
Current (A)	1	0.65	0.5
Total Beam Area (cm^2)	138	12.1	25
Average Current Density (mA cm^{-2})	7.3	> 50	20
Electron % in Beam	3.8	< 1	*
Power Efficiency (kW/A)	10	> 5	7.5
HV Beam Optics	Unknown	Unknown	Unknown
Major Problems	Cs coverage	Cs coverage; cost	Cs coverage; electron suppression
Possible Solutions	HV acceleration; geometry change	Hollow cathode for low gas flow	New Cs feed; double discharge current

* Not available.

nism in magnetic fusion devices, consider the recent policy statement on heating in FED/INTOR (8):

> *"The decision between ICRH and neutral beam injection (NBI) rests on a trade-off between the engineering and technological advantages of ICRH and the greater confidence in the physics basis for NBI. The recent advances in ICRH physics (indicate) that the balance has shifted in favor of ICRH. Therefore, ICRH should be adopted as the prime heating option for long range tokamak applications. The principal backup for near term and long term (device commitments) should be positve and negative ion beams, respectively."*

2.2.2.2 ECRH Technology. In magnetic fields of several teslas typical of current and projected fusion devices, the required range of ECRH frequencies spans ~ 28 GHz to over 100 GHz (i.e., wavelengths of ~ 10 mm to less than 3 mm). The recent availability of high power gyrotron tubes enables the approach to these frequencies at MW power levels with acceptable efficiencies.

The gyrotron is a type of microwave tube which uses the cyclotron resonance condition to achieve coupling between an electron beam and the microwave electromagnetic field. This interaction can be described by either a classical analysis in terms of energy modulation and spatial bunching of the electrons or by a quantum-mechanical analysis leading to the treatment of the gyrotron as a cyclotron resonance maser. It was as late as 1974 that truly significant practical gyrotrons were demonstrated in the U.S.S.R. (9) where outputs of 12 kW CW at 108 GHz were reported with an efficiency of ~ 31%. Current state of the art for developed gyrotrons is the 28 GHz, 200 kW CW tube completed in 1980 by Varian Associates Inc. in the U.S. with an operating efficiency of ~ 40-50%. A tube of this design was operated at 340 kW with ~ 37% efficiency in 1981.

The DOE National Gyrotron Development program in the U.S. is managed by Oak Ridge National Laboratory (ORNL) (10). The program coordinates the activities of two industrial contractors, Varian Associates Inc. and Hughes Aricraft Co., and cooperates with gyrotron R&D activities at other laboratories, including basic research at the Massachusetts Institute of Technology (MIT) and the Naval Research Laboratory (NRL).

Gyrotrons operating at 28 GHz are currently in use on the Elmo Bumpy Torus (EBT-S) and Impurity Study Experiment (ISX-B) at ORNL with output powers of 200 kW (CW) and ~ 100 kW (100 ms pulsed) respectively. Four 28 GHz, 200 kW gyrotrons are currently installed on the Tandem Mirror Upgrade Experiment (TMX-U) at Lawrence Livermore National Laboratory (LLNL) for the purposes of producing magnetically trapped electrons at the sloshing-ion-density minimum in the thermal barrier and high temperature electrons near the sloshing-ion-density peak. Gyrotrons operating at 28 GHz are also currently employed on experiments in England, Italy and Japan, the latter including the JFT-2 tokamak at Tokai, the Heliotron at Kyoto and the Bumpy Torus at Nagoya.

Gyrotrons operating at 56-60 GHz will be required for EBT-P at ORNL, the Mirror Fusion Test Facility (MFTF-B) at LLNL, the Poloidal Divertor Experiment (PDX) at Princeton Plasma Physics Laboratory (PPPL) and the Doublet III tokamak at General Atomic (GA). In MFTF-B, for example, each of eight gyrotrons will generate 200 kW of RF power at 56 GHz for a maximum pulse duration of 30 s repeated at two minute intervals. High voltage DC power is provided by a modified sustaining neutral beam power supply and eight independent pulse-regulator isolation networks supply the highly regulated DC at -85 kV to the cathode of their attendant gyrotron. Each gyrotron interfaces with a waveguide system which will transport microwave power to an antenna located inside the vacuum vessel. The resulting ECRH is directed into the resonant heating zones of the plasma and is essential to establish temperature and potential profiles for thermal barrier operation.

A three year construction program to add 2 MW of 60 GHz ECRH is underway on the Doublet III tokamak at GA. Preliminary experiments at 28 GHz conducted in the Japanese JFT-2 tokamak during 1980 confirmed the ability to heat such high density plasmas and the availability of the first 400 kW of power on Doublet III is anticipated in late 1982. Various waveguide sections and mode convertors required on this device are not available commercially and have been custom-designed at GA (6); similar devices are being supplied by GA to LLNL and PPPL for ECRH experiments there.

Efforts to develop pulsed and CW 200 kW 60 GHz gyrotrons are now underway at Varian and Hughes. During 1981, 60 GHz pulsed tubes were operated by both companies. At Varian, the first experimental pulsed tube operated at a peak power of 200 kW for 100 ms and, at lower peak powers, exhibited an average power of 10 kW. At Hughes, the first experimental pulsed tube operated at a peak power of 160 kW with a 15 μs pulse length (equipment limited). Varian's first 60 GHz tube produced 70 kW for a few minutes in early 1982. The 60 GHz, 200 kW CW gyrotron development program is expected to be completed in 1983.

Possibly the most advanced gyrotron development program in the world today is being conducted in the U.S.S.R. Tubes operating at 84 GHz, 200 kW (pulsed) are in operation on the T-10 tokamak. In addition, a 100 GHz quasi-optical gyrotron program is now underway for integration on the future T-15 tokamak. Prototype designs for these 100 GHz tubes have been tested on T-10 with power outputs of 350 kW for ~ 16 ms, pulse lengths being limited by heating in ceramic transmission windows.

Near term requirements in ECRH technology include the 60 GHz gyrotron systems discussed above. Other hardware requirements--including waveguides, couplers, and antennas for handling large amounts of mm-wave power and coupling it to a plasma--are also critical. Longer term requirements for reactor-type fusion devices include the development of CW gyrotrons producing ~ 100 GHz with a unit size of at least 1 MW. Note, however, that as the size of conventional microwave-cavity gyrotrons increases, the collector cavity must be enlarged to dissipate the beam power, making it more difficult to couple the desired cavity-mode efficiently. One of the multi-MW gyrotron concepts, the quasi-optical gyrotron, offers a potential solution to this problem. The possibility of initiating an industrial program to develop this and other multi-MW tube designs is now under consideration in the U.S. (10).

One other important aspect of ECRH technology should be mentioned here, namely the launching system required for coupling the RF wave from the gyrotron source(s) and injecting it into the plasma. Current launching systems comprise conventional metallic waveguides. However, because of power density limitations, efficiency considerations, and the requirement for vacuum windows for containment of pressurized insulating gas, such systems are expensive and impractical when scaled to the requirements of full scale power reactors. Accordingly, increasing interest is being focused on beam waveguide systems which operate under quasi-optical principles (11). In these quasi-optical launching systems, the ECRH beam is characterized by a Gaussian radial dependence on the electric field, power density constraints are alleviated, and the system can operate in a vacuum thus obviating the need for ceramic windows.

2.2.2.3 ICRH Technology. Ion-cyclotron resonance frequencies are in the tens of MHz regime, where efficient generation of large amounts of RF power is relatively straightforward. With the exception of the launching antennas which are somewhat device-specific, ICRH hardware is already commercially available. The principal engineering advantage of ICRH is the ability to locate the bulk of the equipment in an area remote from the reactor core, thus promoting reliability and simplified maintenance. In addition, bends can be introduced into the ICRH transmission system to minimize neutron streaming and shielding requirements. In comparison to neutral beam heating systems, high power ICRH systems have the potential advantages of higher efficiency, increased component life and reduced complexity of required support equipment.

In a progression of experiments on the Princeton Large Torus (PLT) tokamak at PPPL over the past four years, record amounts of ICRH power have been coupled to the plasma. RF heating in the ion-cyclotron range of frequencies began in 1979 with absorbed powers of up to 0.75 MW. The ICRF generators on PLT consist of a 2 MW, 25 MHz unit and two 2 MW, 42 MHz units and there are multiple coil arrangements for both frequencies thus permitting variability in the relative phasing. In 1981, 1.5 MW of RF power was coupled to the plasma via four half-turn antennas and 0.4 MW at 42 MHz was delivered through two half-turn antennas, power limits being constrained by breakdown at vacuum interface feedthroughs. Recent operation with six coils at 42 MHz has resulted in the coupling of 3.2 MW of ICRH power for ~ 300 ms and was utilized for second harmonic H resonance heating at 1.4 T and for minority H heating at 2.8 T in a D-H plasma. It is interesting to note that in FY-81 and in contrast to previous operation, neutral beam heating in PLT was used mainly in support of the RF experiments.

In 1980, ICRH was selected as the main heating system for the Joint European Torus (JET). This system comprises four main subsystems as follow: (1) 22 kV high voltage power supplies; (2) ten 25-55 MHz ICRF generator amplifiers each with output powers of 3 MW for ~ 10-20 s; (3) twenty 22 cm diameter coaxial transmission lines of 72 m length for generator-antenna coupling; and (4) ten 3 MW center-fed antennas with "antenna limiters" to prevent surface damage under antenna-plasma close-coupled operation. In the final DT phase of operation in ~ 1988/90, JET is envisaged to operate with 25 MW of (absorbed) auxiliary heating, 15 MW of which will be supplied by the ICRH system. Figure 2.2-2 taken from Ref. (7) illustrates the JET ICRF antenna. Note that the electrostatic screen and antenna conductor are recessed below the level of the protection limiter surface.

In the French Tokamak Fusion Reactor (TFR), ICRH heating was limited to 2 MW above which a strong metallic impurity influx caused the plasma temperature to fall and the discharge to disrupt. It is presently considered that the antenna-limiter has a strong responsibility for this parasitic effect and currently JET-type limiters are being evaluated on the antennas of this device.

From the technological standpoint, the most critical component of the ICRH system is the RF launching structure. Present ICRH systems employ loop antennas. However, due to radiation damage, lifetime and maintenance problems, several reactor studies including NUWMAK, FED and STARFIRE have chosen waveguide launchers (12). The ICRH experiments on PLT in 1981 (see above) employed end-fed half-loop antennas with average surface power densities of

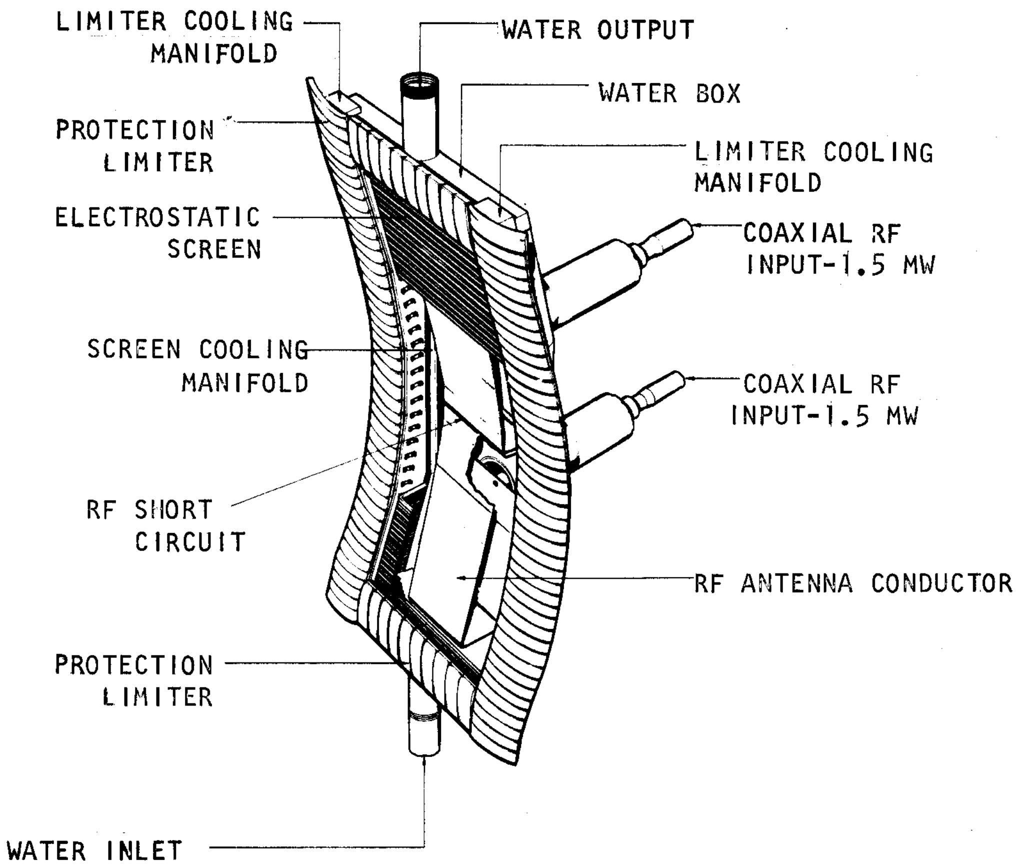

Fig. 2.2-2. JET ICRF Heating Antenna.

~ 0.5 kW cm^{-2}; power densities were limited by breakdown at feedthroughs, not antenna surface effects, and were increased four-fold with improved designs. Power densities in this range have been achieved on the Japanese tokamak JFT-2, i.e. 1.4 kW cm^{-2} on the antenna (13).

2.2.2.4 Conclusions. The current status of RF plasma heating in the ion-cyclotron range of frequencies appears very promising indeed as of late 1982. Three critical R&D areas which need to be addressed in the near future are: (1) the demonstration of ICRF heating at reactor (high β) conditions; (2) accumulation of general systems experience at high ($\geq$ 10 MW) power levels including the operation of multi-MW, multi-second tubes and transmission line components at high reactive powers; and (3) the design and operation of a high-power, reactor-relevant launcher including Faraday shields, voltage standoffs, vacuum feedthroughs and antenna surface-limiters compatible with reactor-level neutron, gamma and charged particle fluxes. It is expected that JET will address issues (1) and (2) and that JT-60 will address issue (3); however, further near term confirmation is needed on all three issues simultaneously (14).

2.2.3 Superconducting Magnets

2.2.3.1 Introduction. Recent years have seen major progress in the development of superconducting magnets for fusion. Several coils have now been constructed and tested which are among the largest superconducting magnets in existence today. The general philosophy in most projects has been a demonstration of subreactor size coils. Significant developments have occurred in conductor design, superconducting materials, structural configurations and cooling schemes in the 1978-82 period.

2.2.3.2 Fusion Magnet Projects (15). Numerous projects exist worldwide for the development of superconducting magnets for fusion. These can be categorized in terms of which concept they support.

Tokamak. The present tokamak fusion reactor concept calls for principally two kinds of superconducting magnets: large steady-state toroidal field (TF) coils and smaller pulsed poloidal field (PF) coils (although current drive might greatly reduce the pulsing requirements). The major effort in superconducting magnet development has been directed toward the TF coils.

Presently, there exists only one operating superconducting tokamak, the T-7 experiment at Kurchatov Institute in Moscow. This device is rather small having 48 TF coils each with inner bore of 0.85 m. The device has operated successfully since 1978 achieving a maximum central field of 2.4 T with 4.0 T at the windings. The coils employ a somewhat conventional NbTi embedded in copper technology with forced liquid helium flow cooling.

The Kurchatov Institute is also in the process of building a larger tokamak experiment based on their experience with T-7. This project, called T-15, is designed with oval shaped coils (2.1 m by 3.3 m bore) having maximum central field of 3.5 T with 6.1 T on the conductor. The major change apart from size is that these coils are to be constructed using Nb_3Sn superconductor and thus obtain higher stability because of higher critical temperature. Completion of the superconducting coils is expected sometime in late 1984.

The largest tokamak supported magnet project presently underway is the large coil task (LCT), which is an international effort involving Switzerland, Japan, the U.S., and Euratom. Euratom is represented by Kernforschungszentrum Karlsruhe, West Germany. Six coils are being constructed, one by each of the foreign participants and three by U.S. manufacturers. The completed coil assembly is to be operated in a facility at ORNL, Fig. 2.2-3.

All coils for the LCT are to have the D-shape bore 2.5 m by 3.5 m and produce a peak field of 8 T. This field was selected to allow either NbTi or Nb_3Sn

technology. Five of the six coils are constructed using the former, with the remaining coil employing Nb_3Sn internally cooled cabled superconductor (ICCS) design.

A comprehensive magnet test program is planned when all the coils are installed in the facility by 1984. Although the device will not be a plasma experiment, the effect of plasma current will be simulated by normal conducting pulsed coils.

The other major tokamak which is presently under design is the Tore-Supra project in France. Here the intent is not to have the largest device, but rather to have a high field, 9 T, long pulse experiment which is stable against plasma disruptions. The system consists of 18 TF coils of circular cross section, 2.23 m bore. The peak field of 9 T is achieved by use of NbTi operating at 1.8 K in superfluid helium (He-II). Thus the project has the dual purpose of developing the technology of He-II magnet cooling. It is expected to be completed in 1985.

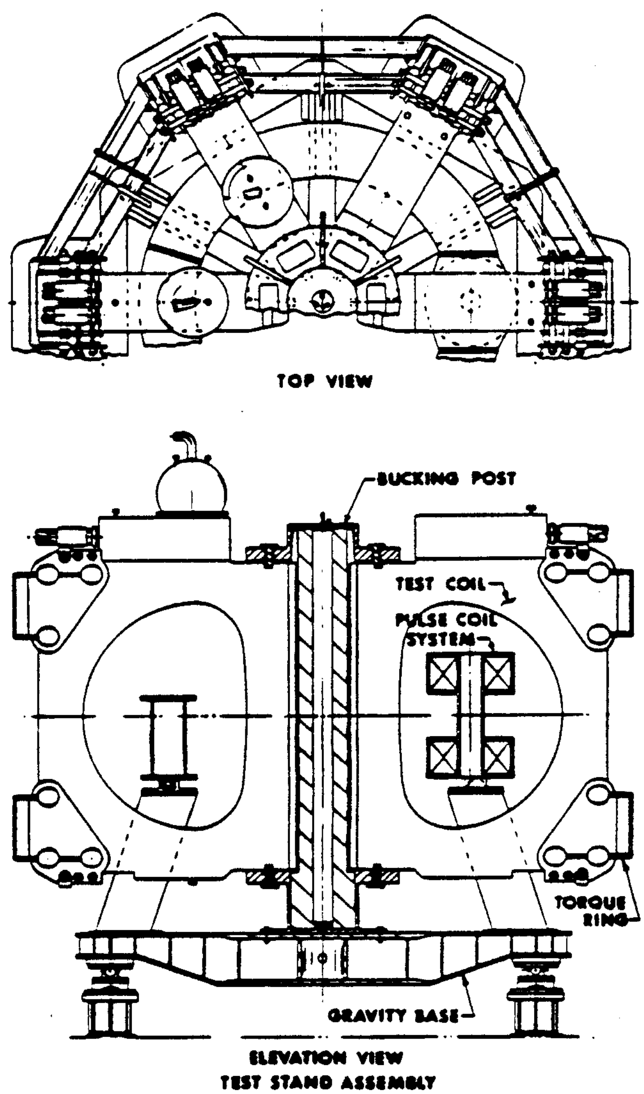

Fig. 2.2-3. LCT Test Stand Configuration.

Apart from the TF coil requirements, tokamaks must have pulsed coils for plasma startup and control. The poloidal field coil development program has not progressed to the same level as have TF coils. Programs are underway in the U.S. at LANL and Argonne to develop fast pulse coils. To date the largest U.S. coil is the 1.5 MJ, 10 kA magnet at Argonne. There are also pulse coil development projects in Japan, Europe and U.S.S.R. The Japanese have recently established a record by discharging a 1.6 MJ, 30 kA coil in excess of $\dot{B}$ = 200 T/s. In the U.S. future developments are part of the U.S. TPFS (Tokamak Poloidal Field System) program at LANL which is to operate a 20 MJ, 50 kA coil with a 7.5 T to -7.5 T field swing in 1 to 2 seconds. Unfortunately these programs have received less financial support recently owing to uncertainty in pulsed field requirements for tokamaks.

Mirrors. The mirror fusion concept has long seen the need for superconducting magnet technology to go hand-in-hand with the plasma physics developments. Mirrors are inherently steady-state devices, so the coils, some which are complex in geometry, are without the difficulties of pulsed field coils. The first superconducting plasma physics experiment was Baseball I at Livermore in 1965. Baseball II was a larger device built in 1970. Both devices have the characteristic yin-yang configuration. Baseball II operated at a maximum field of 5.5 T until its decommissioning in 1977.

Recently, Livermore has undertaken the ambitious project, MFTF-B, which is a large superconducting tandem mirror plasma physics experiment. The device contains two yin-yang end cells each with maximum field of 7.7 T and stored energy of 409 MJ, (Fig. 2.2-4). The first of the two yin-yang magnets has been constructed and tested successfully at full current. The second yin-yang is under construction. These end cells employ NbTi in copper conductor technology with pool boiling normal helium cooling. The central cell coils, with 3 T axial field and 5 m bore, are being fabricated in industry. When completed, this device will have a total stored energy in excess of 1.6 GJ.

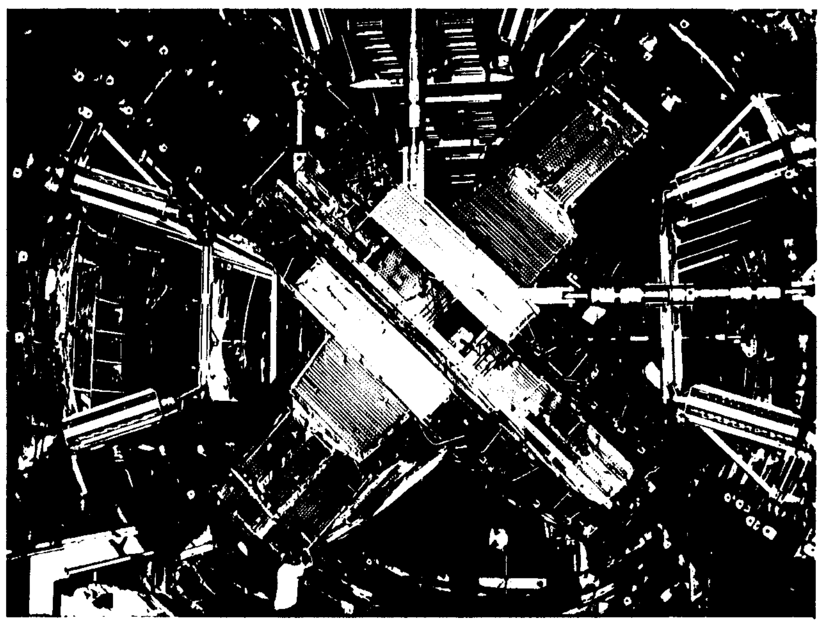

Fig. 2.2-4. MFTF End Cell Test Facility.

Alternate concepts. In the U.S., the principal contender as an alternate concept to the two main line fusion devices is the Elmo Bumpy Torus (EBT). EBT-P is planned to be built in collaboration between ORNL and McDonnell-Douglas. It consists of 36 standard coils of 0.4 m bore operated in a toroidal configuration. The baseline experiments call for a 3.2 T central field, while a 4.8 T field is an upgrade. These coils also utilize NbTi in copper technology with pool boiling helium. Because of high current density demands, the coils are not fully stable. Unfortunately, budget restrictions have forced the indefinite postponement of EBT-P.

2.2.3.3 Recent Technological Developments. Numerous advances in the technology of superconductivity and cryogenics have occurred in recent years which strongly impact the field of magnet systems. In discussing these advances, one should consider the general sub-areas of conductor design and fabrication, superconductive materials, structural materials, and cryogenic systems.

The most significant development in superconductivity which is specifically applicable to fusion technology is in the area of large, high current, stable conductor design and fabrication. Here, substantial advances have been required over the early high energy physics bubble chamber coils. The major drivers in these developments have been high magnetic fields, the need to reduce AC losses, while increasing stability and safety. All these factors have led to three main options in conductor design: conventional monolythic conductors, cable conductors consisting of numerous strands, and internally cooled cabled superconductors (ICCS). The first two are designed to transfer their load to an external structural case while the ICCS carries a substantial fraction of its structure in a conduit surrounding the cable.

Superconducting materials have experienced two outstanding technical developments in the past 5 years. The technology of multifilament Nb_3Sn has now progressed to the point where reliable, high current density material is readily available. Several solenoidal coils of considerable size and peak fields above 12 T have been fabricated using this new technology. The other improvements have centered around modifications to the more ductile NbTi. The most notable of these has been the effect of the third alloying element Ta. Conductors fabricated with NbTiTa have shown higher critical fields than the binary alloy which translates to higher critical currents at temperatures below 4.2 K, and fields greater than 10 T (16).

Besides superconducting material, composite conductors contain high conductivity normal metals which provide electrical and thermal stability. To date, high conductivity copper has been almost exclusively employed in superconducting magnet construction. However, there has been a growing interest in utilizing high purity aluminum to replace the copper. Aluminum's principal advantages are relative ease of purification and lower magnetoresistance.

Structural materials for superconducting magnets are now fairly well characterized. For metallic structures, numerous choices exist among the stainless steel and aluminum alloys. Most magnets are fabricated using 304 LN (low nitrogen) stainless steel because it is structurally stable at low temperatures and non-magnetic. Insulating materials have undergone a considerable developmental effort. Candidate materials are the fiberglass composites of epoxy and more recently polyimide (17). The latter has the advantage of being more radiation resistant.

Finally, as large systems develop, the scale and sophistication of their cryogenics has also progressed. Large fusion reactor sized cryogenic plants are now available owing mostly to the needs of high energy physics accelerators. The cooling mode employed in fusion scale magnet systems has changed because of the needs for high field and safety. The ICCS has encouraged the development of forced flow cooling at supercritical pressures. Also, mostly resulting from the Tore-Supra project (18), there is a considerable effort in the technical aspects of superfluid helium cooling. As a result, large magnet systems now have available three cryogenic cooling options each of which have advantages for different applications.

2.2.3.4 Conclusions. In summary, the past 5 years have been marked with a great deal of progress in the superconducting magnet area. In particular, magnets have been built which are reactor relevant for both tokamaks and mirrors. Operating experience has shown that such magnet systems can be relatively maintenance free and it is anticipated that the experience with LCP, Tore-Supra, T-15 and MFTF-B will add to that confidence. The use of Nb_3Sn has become more widely accepted and He-II is now being seriously considered for reactor applications. Superconducting magnets seem to have passed from being the number one technological problem to one considerably less critical in the view of fusion engineers.

2.2.4 Materials

It was recognized very early in the design of fusion power plants that materials would probably be one of the most limiting features of any confinement concept. Over the past decade, major problems (and some solutions) have been identified in the area of the first wall, solid tritium breeders, and in-vessel components such as limiters, divertors, and beam dumps. The last two items have been covered elsewhere in this report and we will concentrate on structural materials here.

Historically (in the late 60's and early 70's), the first materials proposed for first wall and blanket structures were refractory metals such as alloys of Nb, V or Mo. The reason for their choice had to do mainly with their high temperature potential. However in the early 1970's an austenitic steel was proposed for the UWMAK-I reactor and this ushered in a decade of emphasis on such alloy systems. Even though the maximum temperature was kept to ~ 500°C (both from corrosion and radiation damage considerations) the overall efficiencies were still respectable and the large amount of radiation damage information from the fission breeder program allowed more confident estimates of useful lifetimes to be made. On the basis of available data in the early 1970's, it appeared that we could reasonably extrapolate useful lifetimes of ~ 2 MW-y/m^2, limited mainly by swelling as long as the operating temperature was kept below that which induces helium embrittlement.

The major advances in the last 5 years for the structural materials have been in three areas:

1. More detailed stress and lifetime analyses of first wall materials.
2. Increases in the swelling resistance of the austenitic alloys.
3. Demonstration of more favorable radiation damage resistance in the ferritic alloys.

One area in which there has been little progress in the past 5 years is the construction of a high volume, high fluence 14 MeV neutron source. There is very little data today (1983) typical of a fusion reactor environment, except for Ni containing alloys in thermal fission reactors. Such information is sorely needed and perhaps the Fusion Materials Irradiation Test Facility (FMIT) at HEDL can provide some information. However, at this writing, the FMIT project is being delayed because of financial considerations.

On the positive side the ability to account for cyclical, thermal, and swelling induced stresses along with steady state creep has allowed materials scientists to make more precise and defensible lifetime analyses of first wall components. Such analyses have revealed that at low temperatures (i.e., ~ 300°C) austenitic stainless steels can probably reach lifetimes of ~ 5 MW-y/m^2 before reaching some lifetime limiting criterion.

However, power reactors operate at higher temperatures (i.e., 500°C) and in this temperature range void swelling becomes important. Over the past 5 years considerable information on 316 stainless steel and various modifications thereof have been obtained with respect to how cold-working, alloying additions, and helium can alter the swelling behavior. The details are not appropriate here, but the final results can be simply stated. Void swelling for austenitic steels seems to be characterized by a relatively long incubation period during which the microchemical characteristics of the steel are being altered to allow void nucleation to occur. Once nucleation takes place, the swelling in the material is linear with fluence and a value of ~ 1% per dpa is reached regardless of the initial dislocation content, chemistry, or amount of helium in the specimen. The major effect of alloy modification was to increase the incubation dose but it seemed to have very little effect on the ultimate swelling rate. Once swelling starts at a rate of ~ 1% per dpa, the useful lifetime is quickly reached in usually less than 1 MW-y/m^2 (remember 1 MW-y/m^2 ≅ 10 dpa). The ultimate conclusion of the recent work is that it may be able to push the incubation dose out to 3, 4 or even 5 MW-y/m^2, but the useful lifetime of the 316 austenitic steel will then be limited to that value plus 1 MW-y/m^2. In general, a safe value is perceived to be not more than ~ 5 MW-y/m^2.

While the new value is a great improvement from the initial value of 2 MW-y/m^2, it is far from the program's desire to get 40 MW-y/m^2. In fact, when it appeared that austenitic steels would not be able to attain much more than ~ 5 MW-y/m^2, scientists began to investigate other alloy systems for a potential solution. Fortunately, it appears that the ferritic class of steels may meet the more ambitious requirements. That is, extensive fission reactor neutron and ion irradiation studies reveal that while the maximum incubation dose may also be as low as ~ 10 MW-y/m^2, the equilibrium swelling rate is only 0.1% per dpa (or ~ 1% per MW-y/m^2). This means that if the design limit on swelling is ~ 10%, one should be able to attain ~ 20 MW-y/m^2, a far more acceptable situation. The current swelling data on both steel systems is shown in Fig. 2.2-5.

The introduction of ferritic steels into magnetic fusion reactors is, however, not without some problems. These steels are magnetic and as such, can alter

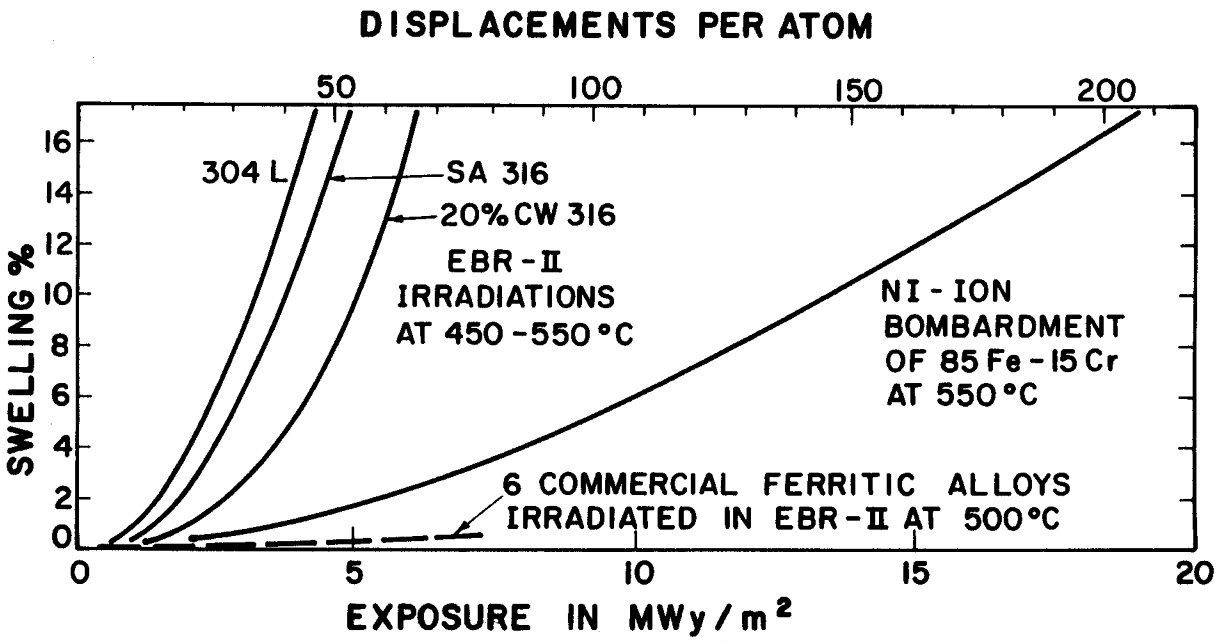

Fig. 2.2-5. Comparison of swelling for austenitic and ferritic alloys based on the following sources: 304 L--swelling in annealed fuel capsules, Garner and Porter (19); SA 316 and 20% CW 316--DO-heat irradiated in EBR-II and HFIR, Maziasz and Grossbeck (20), Brager and Garner (21); 85 Fe-15 Cr--Johnston, Lauritzen, Rosolowski and Turkalo (22); ferritic alloys--Powell, Peterson, Zimmerschied and Bates (23).

the magnetic fields and the magnetic fields can cause large forces to act on the steel. Both of these effects are currently under examination by reactor designers in the MARS (Section 2.3.3.2) project but it appears that they can be handled with reasonable design modifications.

Finally, there has been continued activity over the past five years on graphite and SiC structures, especially for the ICF area. Interest in low levels of long-lived radioactive materials has continued but at a relatively subdued level. There also seems to be a general conclusion that refractory metals are not necessary for fusion reactor design and the level of interest in them has dropped dramatically.

2.2.5 Tritium

The successful operation of D-T fusion reactors will require safe and reliable tritium handling and containment systems. Major programs in fusion tritium technology development exist primarily at Los Alamos, Argonne and Mound Laboratories with several other laboratories and industries addressing critical tritium issues (24). The goal of the U.S. tritium program, funded by the Department of Energy, is to provide an extensive data base in tritium handling for the first large scale D-T fusion machine (such as an ETR device) which optimistically could be constructed in the early 1990's. The Tokamak Fusion Test Reactor (TFTR) at Princeton Plasma Physics Laboratory will burn D-T in the latter stages of the reactor lifetime (~ 1985) but the tritium inventory will be 5 g or less and there will be no on-site fuel reprocessing.

The tritium handling systems in a fusion device depend largely on the physics concept and the choice of breeding blanket and coolant. Many options have been investigated as shown in Table 2.2-3 which illustrates some of the major tritium parameters for several conceptual fusion designs. There are three main tritium systems in a fusion reactor: the fueling and reprocessing of reactor exhaust, tritium breeding and extraction from blankets, and tritium containment and safety. Some important aspects of these three main areas and the major programs studying them are discussed briefly below.

The quantity of fuel that must be supplied to the reactor is much larger than the amount consumed by nuclear fusion. A low burn fraction results in large amounts of tritium which must be pumped and recycled. The effect of burn fraction on the amount of tritium pumped is shown in Fig. 2.2-6. The burn fraction may vary from < 1% to ~ 40% in fusion designs (Table 2.2-3). The amount of tritium recycling can be minimized by using efficient fueling methods such as pellets instead of neutral beams, and by maximizing physics parameters such as the particle confinement time and the reflection coefficient. Unburned fuel can be sent to a fuel cleanup unit (FCU) to remove impurities accumulated from the harsh fusion environment in the reactor and then to an isotopic separation system (ISS) to remove hydrogen and to adjust the D-T ratio. Finally, the tritium stream is sent to either a storage unit or a fuel preparation apparatus before reinjection into the reactor. The Tritium Systems Test Assembly

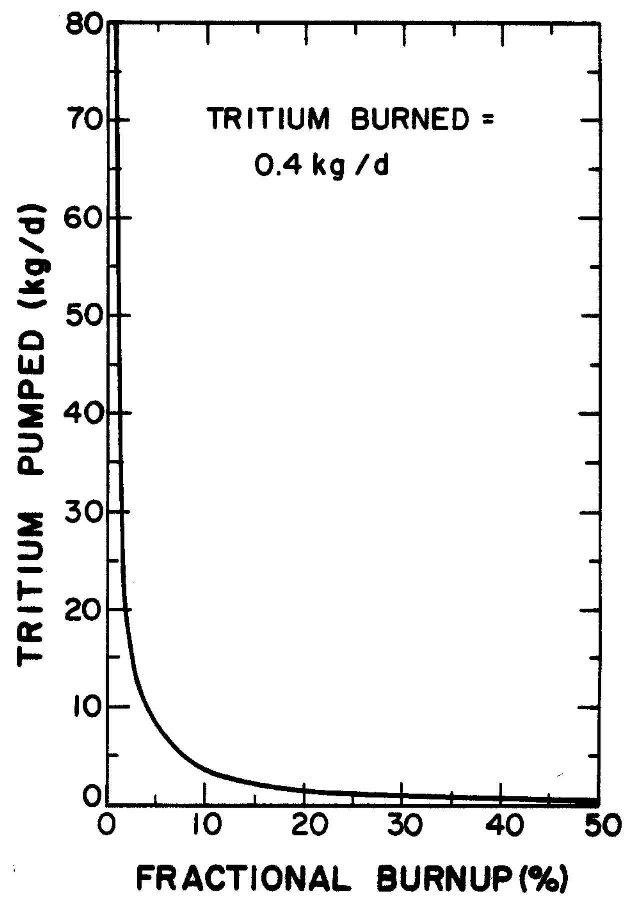

Fig. 2.2-6. Variation of the tritium pumping rate with fractional burnup for 1000 MW of fusion power.

TABLE 2.2-3

Tritium Parameters in Conceptual Fusion Designs

CONCEPT	Design Study	Fusion Power MW_t	Fuel Cycle: Tritium[a] Exhausted (kg/d)	Fuel Cycle: Burn Fraction (%)	Fuel Cycle: Inventory[b] (kg)	Breeding: Breeder	Breeding: Breeding Ratio	Breeding: Extraction Process	Breeding: Inventory[c] (kg)	Storage: Inventory (kg)	Total: Inventory (kg)
TOKAMAK	FED (8T)[d]	180	3.5	0.53	0.325					0.50	0.825
	(10T)	450	4.2	0.84	0.370					1.10	1.470
	INTOR	620	1.43	5.0	0.320	Li_2SiO_3	0.6	Continuous He purge	0.52-7.12[e]	2.20	3.04-9.64
	NUWMAK	1990	19.4	1.48	1.94	$Li_{62}Pb_{38}$	1.54	Molten salt	0.098	19.4	21.4
	STARFIRE	4000	0.76	42	0.271	$LiAlO_2$	1.05	Continuous He purge	7.7-380[e]	1.071	9.0-381
MIRROR	TASKA	86	1.50	0.26[f]	0.460	$Li_{17}Pb_{83}$	1.04	In-situ permeation	0.033	5.00	5.493
	WITAMIR-I	3000	1.67	21.6	0.425	$Li_{17}Pb_{83}$	1.07	Gas sparging	0.205	2.135	2.765
	MARS	2600	2.46	14	0.264	$Li_{17}Pb_{83}$		Vacuum degassing	0.264	3.00	3.264
ICF	SOLASE	3160	0.67	41	0.80	Li_2O	1.33	Diffusion from solid	1.0	9.4/[h,i] 23.9	11.2/[h] 25.7
	HIBALL	8000	2.90	29	0.494	$Li_{17}Pb_{83}$	1.25	Transport into reactor vacuum	0.025	12.3[i]	12.82

[a]Steady state tritium flux.
[b]Includes pumps, fueling and fuel purification systems.
[c]Includes breeding material and breeder reprocessing.
[d]Two cases with 8 and 10 Tesla magnetic fields.
[e]Radiation effects included.
[f]Recycle of tritium in neutral beam lines included in burn fraction.
[g]Includes organic intermediate loop.
[h]Plastic coated (PVA)/glass targets.
[i]Includes manufacture and storage of targets.

(TSTA) (25) at Los Alamos National Laboratory will test the various technologies required for fuel re-processing. The TSTA will simulate a fusion facility by circulating a large DT gas loop containing a variety of impurities through the reprocessing steps described above. The facility will also demonstrate tritium containment systems, monitoring, data acquisition and control and tritiated waste treatment. The gas loop is designed to handle 360 mol/day (1800 g/d) of DT, which is approximately the flow rate anticipated for a near term reactor. The on-site tritium will provide essential experience and data for fusion reactors.

There have been many studies on the choice of different breeder/coolant/ structure options for fusion reactors. The primary breeding material candidates in current reactor designs (Table 2.2-3) include the liquid metals: lithium and lithium-lead alloys (in particular the alloy of atomic composition $Li_{17}Pb_{83}$) and the solid breeders (lithium oxide, lithium aluminate and lithium silicate). Other breeding materials such as the solid Li_7Pb_2 and lithium zirconate have also been proposed and studied in the past 5 years. Liquid lithium processing has been investigated in the Lithium Processing Test Loop at Argonne National Laboratory (ANL) and molten salt extraction has been demonstrated as a method for recovering tritium and controlling impurities. This method is applicable for tritium removal from lithium-lead alloys although vacuum degassing techniques have also been proposed for tritium recovery from $Li_{17}Pb_{83}$. Critical issues for liquid breeders include safety (Li-water reactions), corrosion (LiPb- and Li-steel), embrittlement, and tritium containment.

Solid breeders have been studied in the International Tokamak Reactor study (INTOR) (26) and the STARFIRE reactor design at ANL. Recovery of tritium from solid breeders in-situ by flowing a stream of helium gas over the solid appears to be feasible; however, a carefully controlled temperature distribution is required in the breeding material to prevent tritium buildup in low temperature regions, and to prevent sintering and gas phase transport of lithium species in high temperature regions. Some critical issues facing solid breeders include difficulties in designing for and maintaining the specified temperature range, restructuring of particle morphology under reactor conditions, buildup of tritium inventory and formation of corrosive LiOH in the presence of water. The Solid Breeder Task Force, headed by ANL, is expected to increase the material data base for solid breeders and to obtain data on in-situ tritium extraction through several in-reactor experiments which have been or are currently being conducted, e.g., the TULIP, TRIO or FUBR experiments.

Progress has been made in the last five years in several other areas: the recovery of tritium from water by Combined Electrolysis and Catalytic Exchange (CECE) has been developed at Mound Laboratories, the permeability of tritium through structural materials and the effect of surface coatings on permeation rates has been investigated at Oak Ridge National Laboratory and the modeling of tritium implantation and trapping in first wall and divertor structures has been developed at Sandia Laboratories. Considerable experience exists in the handling of tritiated water from the Canadian heavy water reactors (CANDU). A heavy water detritiation unit is being constructed at the Pickering reactor site for operation in 1985.

Tritium safety, monitoring, containment and waste disposal are also areas of current study in connection with the programs mentioned previously as well as others. The environmental effects of tritium releases from future fusion devices have also been addressed. Goals for tritium losses in fusion power reactor designs have been in the modest range of 10-20 Ci/d. These losses are between current tritium releases from light water and heavy water fission reactors and are considered to be environmentally acceptable. The demonstration of safe tritium handling and effective containment is critical for TSTA and other tritium facilities for the future of fusion.

2.2.6 Divertor and Limiter

The primary functions of the exhaust system are to remove helium ash and to control impurity levels. Helium is a low Z impurity and, if not removed, will dilute the reaction of D and T. A fusion power level of 500 MW in a 250 m^2 INTOR (27) scale device implies a helium production rate of 2 x 10^{20} He/s, or a buildup rate of about 0.5% per second. A 20 s pulse is permissible before exhaust and must be considered. For a pulse much longer than 20 seconds, helium must be removed at the same rate it is produced. Divertors and limiters are the devices to accomplish these functions.

Design studies have been undertaken for both poloidal and bundle divertors in connection with the INTOR (27) and ETF (28) studies. The current concepts are shown in Figs. 2.2-7 and 2.2-8. The principal advantage of the two systems is that the poloidal divertor, being axisymmetric, does not upset confinement, and the bundle divertor, being localized, can be treated as an auxiliary appendix to the tokamak. Both systems add varying degrees of complexity but do permit neutralization, with the attendant heat load and sputtering problems to be handled in a divertor chamber somewhat isolated from the plasma chamber. This allows a wider choice of material and concepts to be considered.

Divertor designs always add complications to the tokamak system by the additional coils required to divert the exhaust. Also, diverting, and inevitably concentrating, the exhaust from the entire chamber wall to a divertor chamber causes severe heat flux and particle flux to the divertor plate. This has given rise to the concept of a pumped limiter. Pumped limiter concepts are illustrated in Fig. 2.2-9 where ions moving along field lines in the plasma edge region are neutralized against the limiter. Those which are neutralized at the rear surface of the limiter have a reasonable probability of being pumped away from the return plasma. Due to the inefficiency of the pumping system, both the ash (He) and fuel (D and T) will be recirculated back to the plasma. Therefore, a pumped limiter system will have a high impurity level and high plasma burn fraction.

The engineering problems associated with divertors and pumped limiters are mainly caused by heat flux and particle flux. The heat flux has been greatly reduced from the earlier value of ~ 5-10 kW/cm^2. The current designs have a

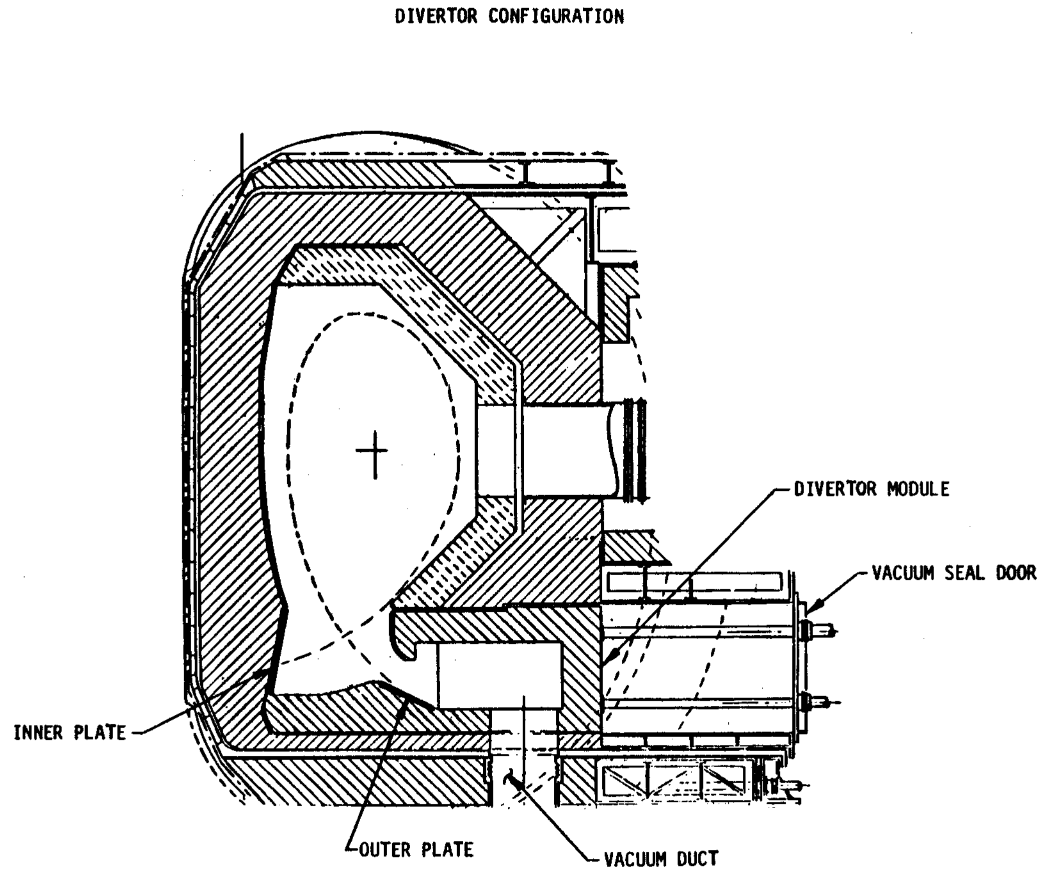

Fig. 2.2-7. A Poloidal Divertor Configuration for INTOR (27).

maximum heat flux of ~ 200-500 W/cm^2, which is manageable by forced convection cooling. The particle flux causes sputtering problems on the target which will also limit its lifetime. A coating is a possible method for protecting the structure. The most often suggested protective materials are beryllium for its low Z, and tantalum for its high sputtering resistance.

Regardless of the impurity control mechanism, a vacuum system is needed to remove the helium ash and the unburned fuel fraction. Typically, power reactors require pumping speeds on the order of 10^6-10^7 ℓ/s. Whereas cryopumps are ideally suited for pumping hydrogen species and He due to their high pumping speeds, they cannot be used in a radiation environment because of damage to the molecular sieves and nuclear heating. Certain getter materials such as Ti and ZrAl can be used at high temperatures and have reasonably high pumping speeds for hydrogen isotopes. Both cryopumps and getters are being investigated in conjunction with TSTA at Los Alamos and at Lawrence Livermore National Laboratory.

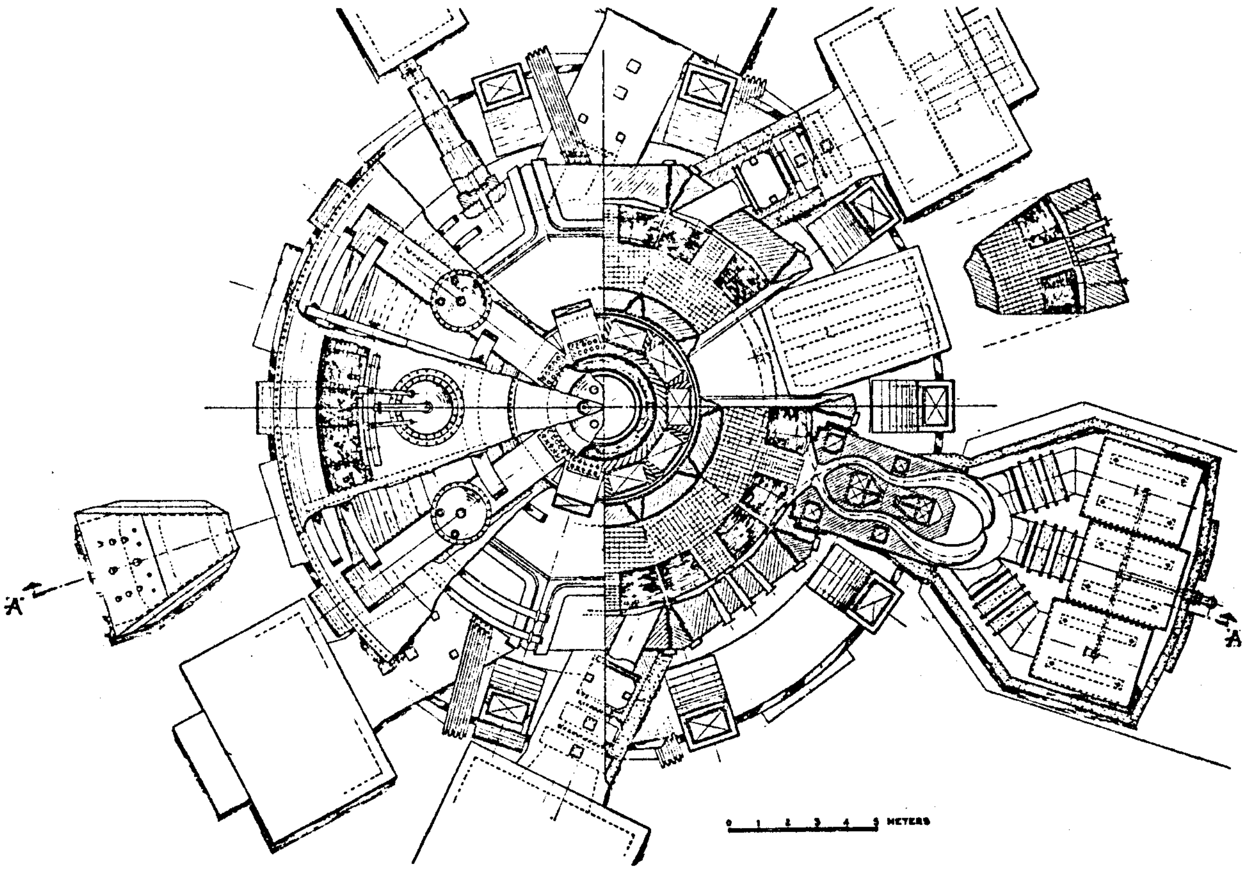

Fig. 2.2-8. ETF Design I Baseline Design with a Bundle Divertor (28).

In summary, the major advance in the past 5 years of research into impurity control is the introduction of the pumped limiter concept. Major analyses of divertors have also been performed such that the problems (and some solutions) are more clearly defined.

2.2.7 Direct Convertors

Plasma which escapes through the loss-cone fan at each end of open-ended fusion devices (e.g., the tandem mirror), contains both energetic plasma ions (D^+, T^+, He^{2+}) and electrons and, therefore, provides a means of direct conversion of plasma energy to electrical energy. The importance of recent and near term research in this area is evident from the fact that, unlike conventional thermal plants, direct conversion can theoretically approach an efficiency of unity and is expected to comprise the main power conversion system in future fusion devices employing advanced fuel cycles (e.g., D-^{3}He).

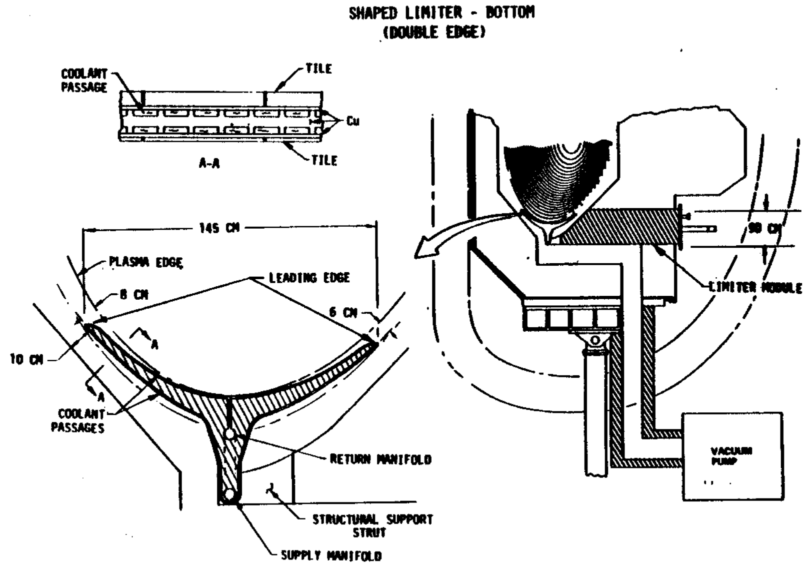

Fig. 2.2-9. A Pumped Limiter Configuration for INTOR (27).

Because the direct conversion (D.C.) process represents an important element in mirror fusion research, experimental and theoretical investigations of this process for fusion power applications in the past five years have been almost exclusively carried out in conjunction with the mirror programs in the USA and Japan. No research programs currently exist in Europe. Similarly, there is no evidence of such programs in the Soviet Union, a fact which is rather surprising in view of the otherwise strong commitment to tandem mirror R&D in that country (see, for example, Section 2.1.2).

The results from three recent D.C. experiments at Lawrence Livermore National Laboratory (29) support theoretical design concepts currently employed in mirror fusion reactor studies. Two of these tests were performed under reactor-like conditions utilizing a 100 keV, 6 kW ion beam test-stand yielding steady-state power densities of up to 70 W cm^{-2}. A single-stage unit and a two-stage unit of the venetian-blind type were tested for a total time of ~ 80 h. In the process of upscaling ion energies over previous experiments, two new effects were evident, namely the ionization of background gas and the release of secondary electrons at electrode surfaces. In the third D.C. test, a single-stage unit was mounted to intercept a portion of the end-loss plasma in the Tandem Mirror Experiment (TMX). For 138 W of incident plasma energy, 79 W was recovered and 12 W was used to power the suppressor grid, yielding a net efficiency of 48%. Larger scale tests of direct convertors are projected on both the TMX-U and MFTF-B devices at LLNL.

An ambitious Japanese research program in direct convertors has been initiated at the Institute of Atomic Energy at Kyoto Universty. This program is under-taking experimental investigations of D.C. systems and will likely contribute to future design studies of direct convertors for the projected GAMMA 20 experimental device and GAMMA R tandem mirror reactor.

The conceptual design of full scale direct convertors for tandem mirror reactors has also seen recent progress. D.C. systems developed by the University of Wisconsin in the WITAMIR (30) tandem mirror design and by the design team of the Mirror Advanced Reactor Study (MARS) (31) propose to expand the end-loss plasma from the reactor until the power is low enough to allow the insertion of grids directly into the plasma stream. The grids are then biased to repel the electron flux and the ions are decelerated and deposited on a cooled collector surface. These systems occupy large volumes (> 10^3 m^3 for WITAMIR and > 5 x 10^3 m^3 for MARS) and have projected efficiencies of ~ 50-65%. Inefficiencies can be attributed to energy spreading of ions and their direct interception by the grids.

A novel direct convertor design has been employed in the conceptual design study of the SATYR D-D tandem mirror reactor (32). With the aid of a magnetic field, successive layers of the end-loss plasma stream are peeled off and allowed to expand radially as well as axially thereby forming a conical shape. Electron suppression rings are placed between plasma layers while the ion collector surface area is much larger than that in other D.C. designs, thus allowing for a more compact system. Single stage electrostatic deceleration of ions without direct interception by grids leads to energy recovery estimates of 65% to 75%.

In summary, recent experiments and conceptual studies have generated confidence that direct convertors can be successfully integrated into fusion reactors of the open-ended type. Remaining design issues which will require attention in the near future include: maximization of net efficiency by means of multistage systems, volume reduction of the direct convertor (D.C.) end cells through efficient collector design, continuing improvement of high-heat-flux surfaces for direct plasma interception, reduction of tritium diffusion losses to D.C. coolant, and assessment of high voltage breakdown effects in D.C. systems with large values of capacitative stored energy.

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2.3 EXPERIMENTAL AND POWER REACTOR STUDIES

Reactor studies are directed twoards the conceptual design of fusion reactors based upon the present experimental and theoretical understanding of both the plasma physics and technological aspects of fusion. They provide a framework to:

1. Discover and characterize problems that occur at the interfaces between physics, engineering, economics, and the environment.

2. Investigate and develop approaches to solving specific design and engineering problems.

3. Develop new and innovative solutions to specific problems.

4. Perform tradeoff studies to define the most promising technology choices.

5. Provide direction for physics and technology R&D.

6. Provide perspective on the commercial potential of fusion reactor concepts.

Design studies must be viewed as continuous activities which lose value if not constantly updated to incorporate the latest ideas. Initial design studies normally explore only a limited number of options; follow-on studies explore other, usually more innovative, options with the aim of providing a more attractive and realistic system.

In the past 10 to 15 years design studies have been performed for most reactor concepts. In the following sections the status of reactor studies for tokamaks, stellarators and tandem mirrors performed in the 1978-82 period is reviewed. These concepts have received the major attention for reactor studies, since they represent the major approaches to magnetic confinement fusion.

2.3.1 Tokamaks

The largest fraction of the world effort in fusion reactor design has certainly been devoted to the tokamak concept. In the late 60's and early 70's, this effort was devoted almost exclusively to commercial reactor systems. However in the mid-70's the effort (at least in the U.S.) turned toward Experimental Power Reactors (EPR's). Then, in the late 70's and early 80's, the emphasis shifted again to more near term devices. Therefore, the following analysis is divided along those lines, briefly reviewing first the EPR studies, then concentrating on the near term reactor studies performed over the past 5 years. This will be followed by a discussion of the commercial designs produced in the same time period.

2.3.1.1 Experimental Power Reactor Designs. There was a great deal of activity on the design of tokamak Experimental Power Reactors (EPRs) in the U.S. between 1974 and 1976. Separate designs were proposed by Argonne National Laboratory, General Atomic, and Oak Ridge National Laboratory. These were summarized in a paper by Stacey in late 1976 (1). The three separate studies filled the void between the operating tokamaks of that time (e.g., ORMAK, PLT, D-II) and commercial reactor designs (UWMAK-I, II, III, the

Princeton design, or the ORNL design). These studies indicated that the data base available was insufficient to accurately predict either the thermal power or the dissipative losses to be incurred in a reactor grade plasma. As a consequence of this uncertainty, the requirement of net power production translated into a design with unacceptably high cost and unrealistic technological risks.

In 1976 the ERDA sponsored another set of studies to define the next step (TNS) in the tokamak program. There were two main groups performing the study. The first was an ORNL/Westinghouse team and the second was a General Atomic/ANL team. The work was completed in late 1977 and an extensive series of documents was published (2). The major accomplishment of the TNS studies was to more sharply define the R and D, schedule and costing of tokamaks which produce large amounts of power. No positive action was taken by DOE on the TNS design.

2.3.1.2 Near Term Reactor Designs. In late 1978, the start of our review period, two major events occurred. The Office of Fusion Energy in DOE established (December 1978) the Engineering Test Facility Design Center at ORNL. The ETF Design Center was directed to prepare an engineering design of a tokamak reactor which could be submitted as a line item to support a 1984 Title I start date. The operation date was to be the early 1990's. Such a design was to be considerably scaled down in its physics and technology ambitions from the EPR and TNS concepts. The ETF was to be the result of extensive industrial and university collaboration with the national laboratories. The project was terminated in mid-1980 when it was determined that the total cost of the ETF would be higher than the goal set by DOE and that the ETF technological goals were too ambitious.

It was also in 1978 that the INTOR project (3) was proposed by Academician Velikhov of the U.S.S.R. and sponsored by IAEA in Vienna. The initial meeting was held in November 1978 between representatives of the U.S., U.S.S.R., Japan, and EC. Initially, the purpose of INTOR was to:

Phase 0 (1979)	Identify the programmatic and technical basis for the next reasonable step beyond the next generation of large tokamaks (TFTR, JET, JT-60, T-15) in the world fusion program.
Phase 1 (1980-1)	Provide a conceptual design of the device that would demonstrate the physics and technological requirements for a large scale Demonstration Reactor (DEMO).
Phase 2A (1981-3)	Examine and propose solutions to the critical issues for reactor operation identified in Phase 1.
Phase 2B (1983-?)	Provide a detailed design of the INTOR device that would satisfy the requirements established in Phases 0, 1, and 2A.

Phase 3 (?) Construction of the INTOR device on an international scale.

The first effort involved about 40-80 man-years of effort in the four countries. Subsequent phases (1 and 2A) required an excess of 100 man-years of effort each. The program never got to phase 2B due to the withdrawal of the U.S. from IAEA activities after the IAEA stripped Israel of its voting rights. When relations were normalized in 1983, a new plan of operation was proposed and is presently being considered by all the countries that were involved.

The initiation of the next major tokamak design effort in the U.S. (after the ETF), came from the conclusions of the Buchsbaum Committee (4) (see Section 6.4):

> *"Some of the objectives of the recently proposed Engineering Test Facility (ETF) in particular, the level of neutron flux and duty cycle, as well as the role envisioned for the ETF on the road to commercialization of fusion--are inappropriate at this stage of fusion development. Rather, the program we advocate should center around a more modest; tokamak-based Fusion Engineering Device (FED) which should have the following goals:*
>
> * *Provide a burning, perhaps an ignited, plasma.*
> * *Provide a focus for developing and testing reactor relevant technologies and components.*
> * *Explore and firmly delineate problems of operator and public safety.*
> * *The device should be in operation within 10 years and cost not more than one billion of current dollars."*

As a result of this recommendation, language in the Magnetic Fusion Energy Act of 1980 directed DOE to proceed with the design of FED. For two years, 1980 and 1981, an extensive design effort on FED was mounted by the FEDC in Oak Ridge and the result is documented in a 6 volume report released in October 1981 (4). During the rest of 1981 and part of 1982, the FED design was continued before it was realized that even the FED design was not possible within the budget and technological constraints set up by DOE.

The U.S. DOE rescoped its thinking and presented it to the community in the Comprehensive Program Management Plan in May of 1982. After some spirited discussion of the plan it was rewritten and released in 1983. At about the same time, the Magnetic Fusion Advisory Committee was formed (Section 6.8) to provide community input into the fusion program. One of the subgroups of this committee, the Forsen panel, studied and endorsed a new device, called DCT-8, to follow on the TFTR device at PPPL. While the DCT-8 has not gone through any detailed design analysis by the community, and has not been formally reviewed by DOE, it does represent the best thinking of the tokamak community as of the time of this document.

A summary of the key features of ETF, FED, INTOR, and DCT-8 is given in Table 2.3-1 and schematics of the four devices are shown in Fig. 2.3-1a-d.

2.3.1.3 Demonstration and Commercial Power Reactors. Design studies of commercial tokamak power reactors began about 1971 and since 1977, five major reactor studies have been reported. These rely upon the experience of the earlier studies, but have a number of new features. In addition, there have recently been efforts directed towards the design of demonstration power reactors, which are nearer term devices, and have the objective of establishing commercial feasibility. The commercial power reactors are generally viewed as the tenth reactor of a standardized design; consequently they are considered to have benefitted from substantial industrial development to avoid "one of a kind" costs.

The major parameters of the five recent power reactor and two demonstration (DEMO) designs are listed in Table 2.3-2 and a cross section view of the STARFIRE and NUWMAK designs are shown in Figs. 2.3-2 and 2.3-3, respectively. All of these designs have a number of features in common, but also a number of differences which we will discuss below.

The NUWMAK (5), Culham Mark II C (6), and HFCTR (7) designs utilize pulsed operation; the plasma current is driven inductively by a transformer, as is the case in present experimental facilities. A long pulse and high duty factor is achieved in the NUWMAK and HFCTR designs; this requires plasma refueling during the burn phase. The Mark II C design explores the implications of a non-refuelled short-burn system. The STARFIRE (8) and SPTR-P (9) designs postulate a steady-state burn using RF waves to drive the plasma current. The advantages of steady-state operation, if it can be obtained, are considerable. Fatigue problems due to stress--and thermal--cycling are reduced. This may allow for a higher power density due to eased materials requirements. One also expects a higher availability with steady-state operation.

Earlier tokamak reactor designs utilized the magnetic divertor for impurity control. The more recent designs assume some alternative approach such as a cool plasma blanket or pumped limiter. This simplifies the design of some of the magnetic field coils and reduces the amount of tritium tied up in the vacuum pumping and plasma refuelling systems. These alternative approaches to impurity control have not yet received the same degree of experimental tests as magnetic divertors, but the concept is promising.

There is a trend in current tokamak reactor designs towards using radiofrequency (RF) waves for plasma heating. This is partly due to the use of RF for steady-state current drive in some designs, but also represents an attempt to find a better alternative to neutral beam heating. RF heating provides the possibility of reduced neutron streaming problems; the RF sources are also in a more advanced stage of development compared with the negative ion neutral

TABLE 2.3-1

Summary of Main Parameters of Near Term Tokamak Test Facilities

	TFTR	JET	ETF	INTOR	FED 8T/10T	DCT-8
Year Conceptual Design Begun	1973	1973	1978	1979	1981	1982
Year Conceptual Design Finished	1975	1977	1980	1981	1982	?
Projected Operation Date	1982	1983	early 90's	early 90's	early 90's	1992
DT Power - MW	20	120	750	620	180/450	360
Pulse Length-s	1.5	15	100	100/200	100/50	10
Ave. Neutron Wall Loading MW/m^2	0.25	0.5	1.5	1.3	0.5/1.2	1.9
Availability Target/y, %	< 1	< 1	25	25/50	10-20	?
Number of Pulses/Lifetime	$4x\ 10^3$	4×10^3	5×10^5	7×10^5	2×10^5/ 2×10^4	2×10^4
Total DT Lifetime-yr	4	?	10	15	6	?
MW-y/m^2-Life	5×10^{-5}	$\sim 10^{-3}$	2.4	6.6	0.37	0.01
Plasma Parameters						
Plasma Q	1-2	3.4	∞	∞	5	∞
Major Radius-m	2.5	2.96	5.4	5.3	5.0	3.6
Minor Radius-m	0.85-1.0	1.25	1.3	1.2	1.3	0.8
Current MA	2.5-3.0	4.8	4.9	6.4	5.4	6.0
Beta limit at full field-%	3.5	5	6	5.6	5.5	4.0
Field on axis-T	5.2	3.5	5.5	5.5	3.6	7.0
Toroidal Field Coil						
Type	Cu	Cu	NbTi or Nb_3Sn	Nb_3Sn or NbTi	NbTi or Nb_3Sn	SC/Cu Hybrid
Number	20	32	10	12	10	16
Plasma Heating						
NB Power-MW	33	10	60	0	50	---
NB Energy-keV	120	80	150	---	150	---
ICRH - MW	---	30	---	50	50	20
ICRH - MHz	---	50	---	85	54	55-110
ECRH - MW	---	---	5-10	---	---	10 (LHRF)

TABLE 2.3-2

Long-Term Tokamak Reactors - Design Parameters

	Commercial					Demonstration	
	HFCTR	NUWMAK	Culham Mark II C	STARFIRE	SPTR-P	FINTOR-D	STARFIRE-DEMO
Net electric power, MW	775	660	600	1200	1000	600	290
Gross thermal power, MW	3,300	2100	1825	3000	3700	1900	1050
Major plasma radius, m	6.0	5.1	7.8	7.0	6.8	8.1	5.2
Minor plasma radius, m	1.2	1.13	2.0	1.94	2.0	2.2	1.3
Plasma current, MA	6.7	7.2	11.0	10.1	16.4	8.9	9.0
Av. toroidal beta, %	4.0	6.5	7.7	6.7	7.0	7.0	8.0
Toroidal field on axis, T	7.4	6.1	3.9	5.8	5.3	4.4	
Plasma heating	NBI 100 MW	RF, 80 MW, 92 MHz	NBI 86 MW	RF, 90 MW, 1.7 GHz	RF	NBI, 290 MW, 160 keV	REB
Plasma current drive	None	None ---	None ---	RF, 90 MW, 1.7 GHz	RF, 80 MW	None ---	REB
Burn pulse length, s	500	225	27	Continuous	Continuous	Quasi-steady	Continuous
Duty factor	.96	0.91	0.67	---	---	---	
Impurity control method		Gas blanket	None	Pumped lim.	Pumped lim.	Plasma mantle	Pumped lim.
Neutron wall loading, MW/m^2	3.4	4.0	1.5	3.6	3.3	1.3	1.8
Blanket breeding material	Lithium	$Li_{62}Pb_{38}$	Unspec.	$LiAlO_2$	Li_2O	Li	(Solid)
Blanket struct. matl./coolant	TZM/FLIBE	Ti-Al-V/H_2O	Unspec.	St. steel/H_2O	St. steel/H_2O	St. steel/He	St. steel/H_2O
Max. structure/coolant temp., °C	760/750	500/300	Unspec.	450/320	425/330	550/500	450/320
Max. magnetic field, T	13.1	11.9	8.3	11.1	12.0	8.0	10.0

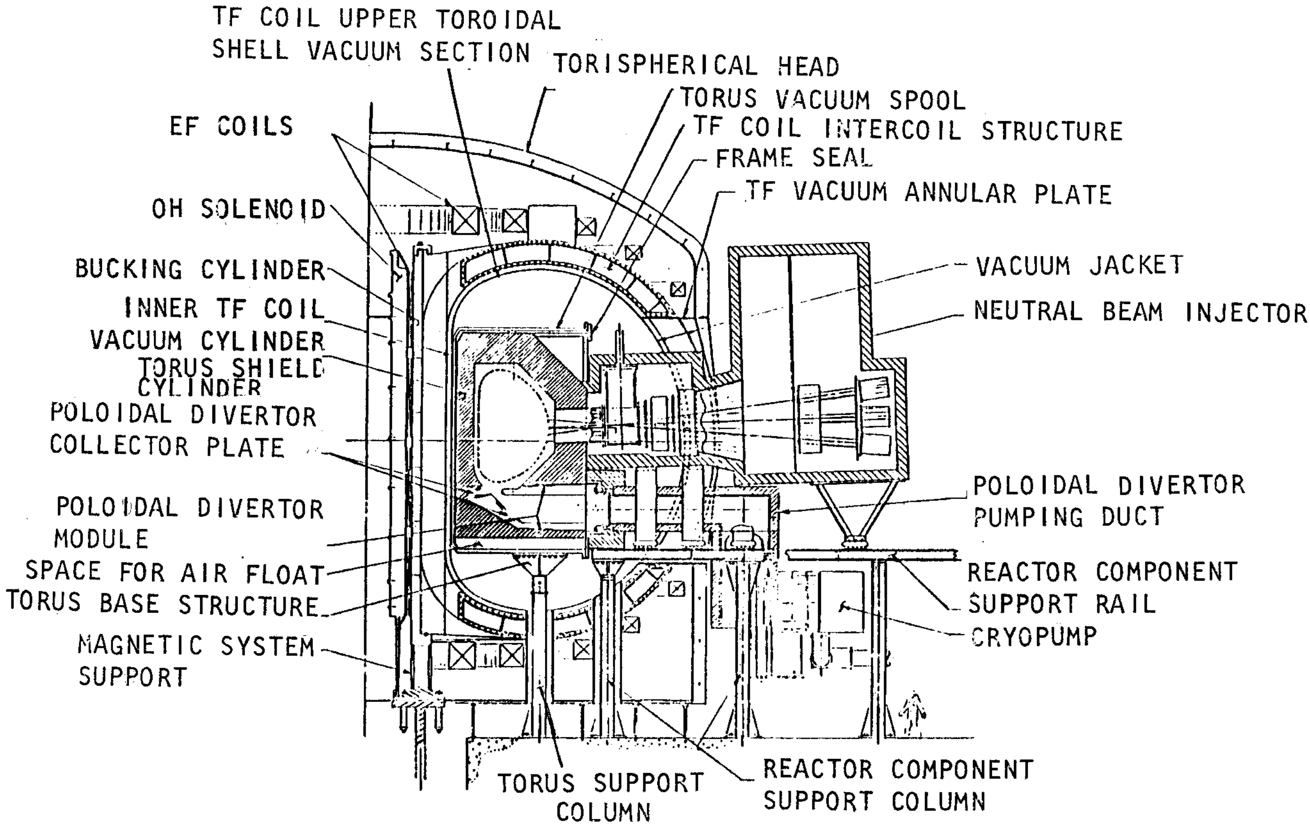

Fig. 2.3-1a Schematic of ETF Design.

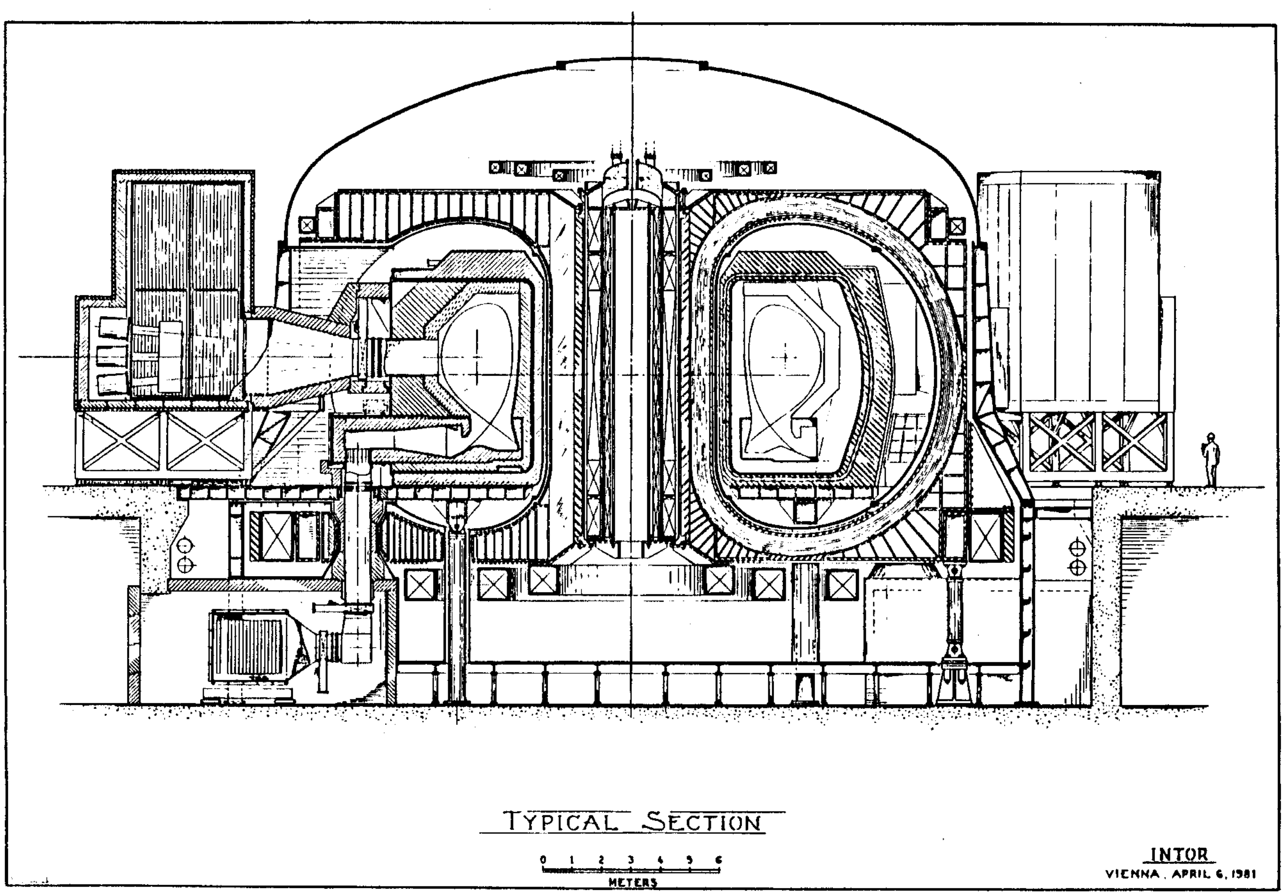

Fig. 2.3-1b INTOR reference design.

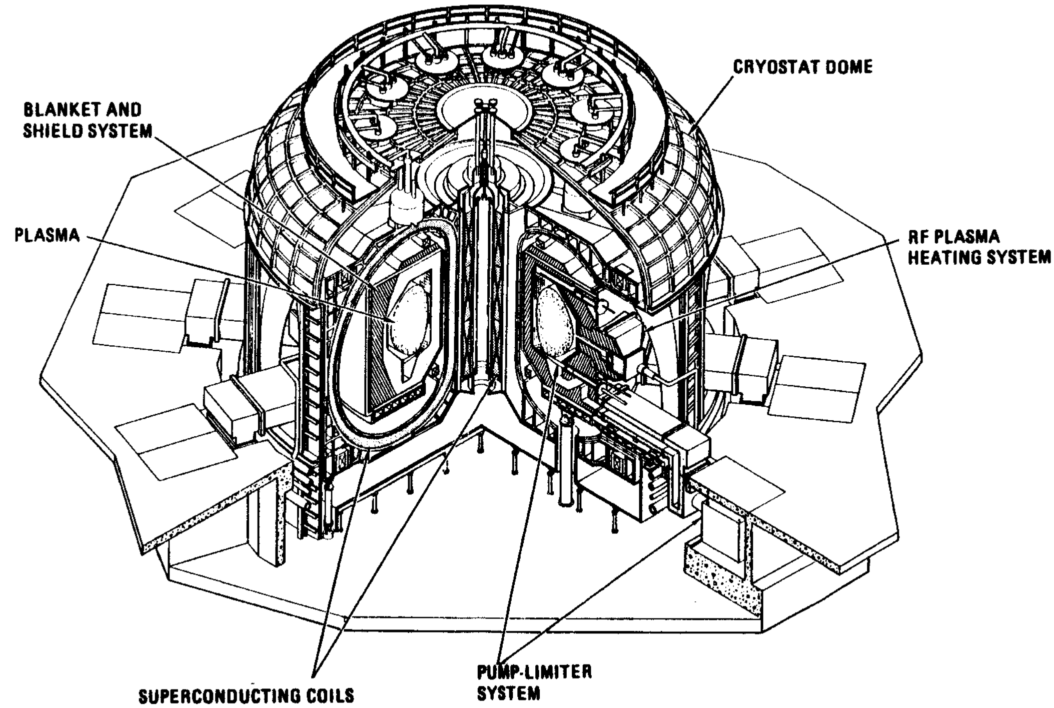

Fig. 2.3-1c Cutaway view of FED, Fusion Engineering Device.

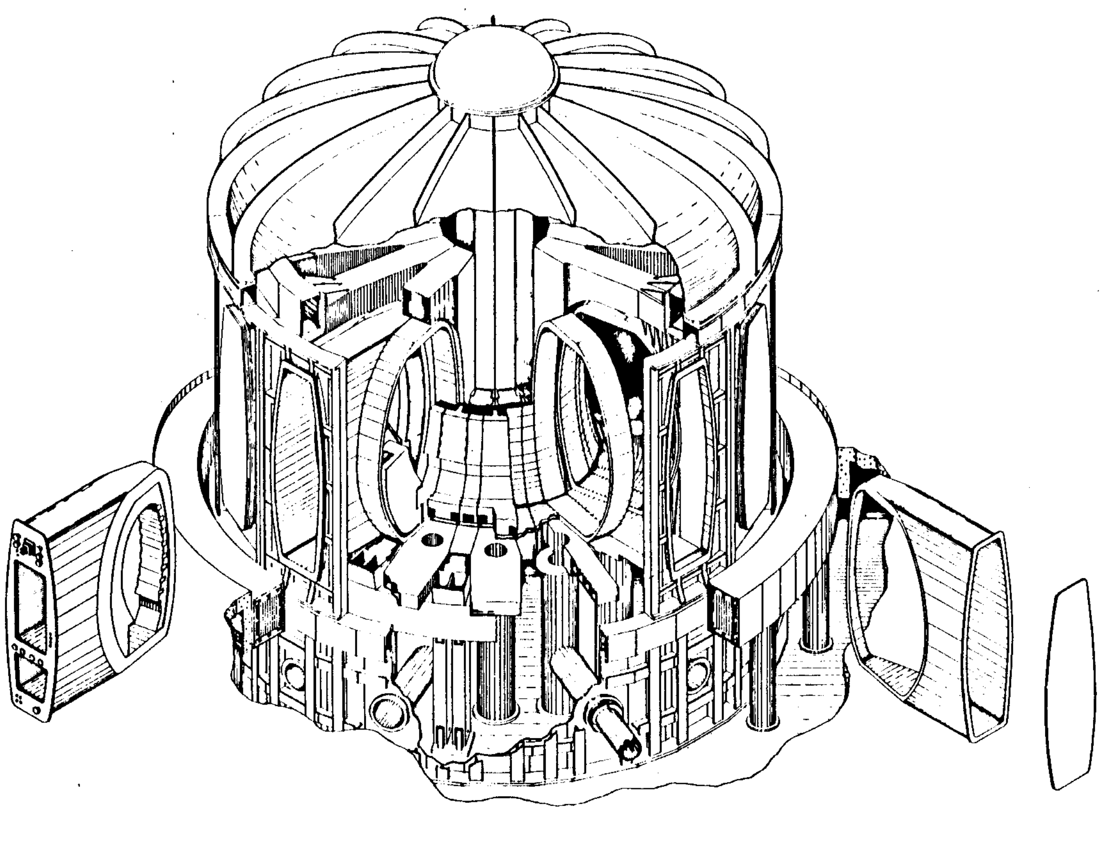

Fig. 2.3-1d Isometric view of DCT-8, showing the vacuum module removal.

STARFIRE REFERENCE DESIGN

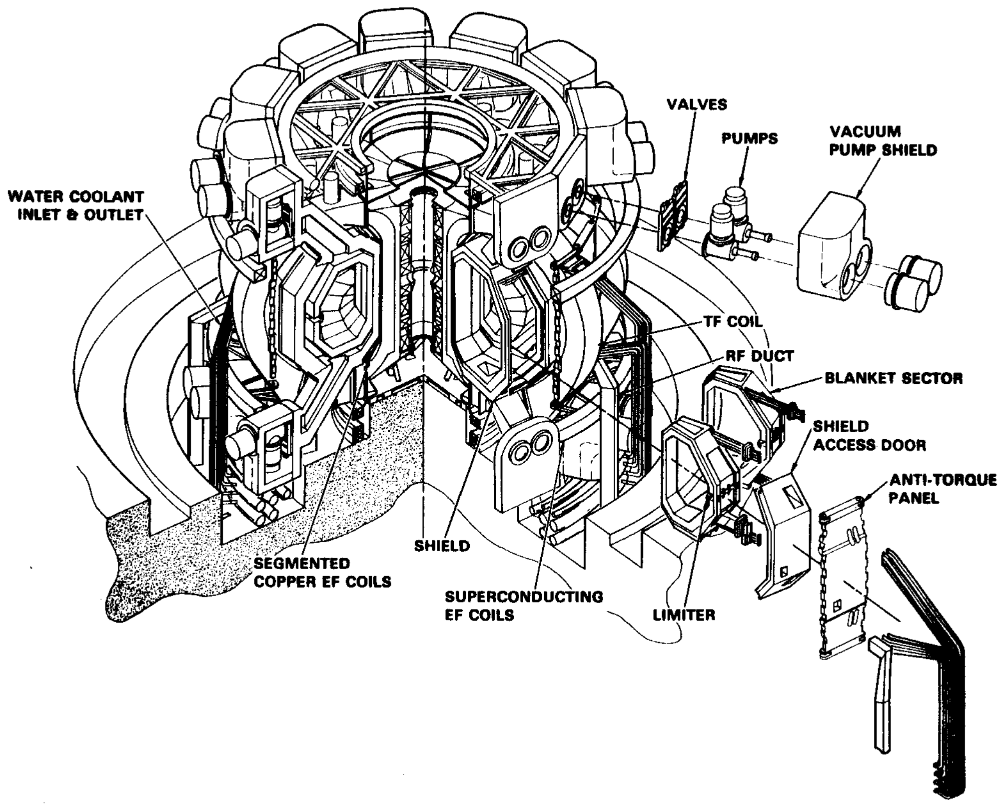

Fig. 2.3-2 Cross sectional view of STARFIRE commercial tokamak.

CROSS-SECTIONAL VIEW OF NUWMAK

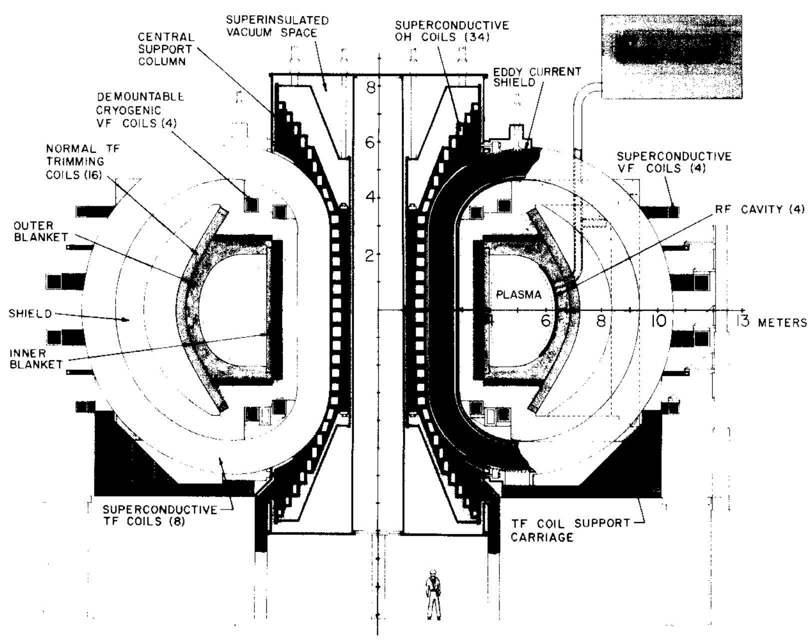

Fig. 2.3-3 Cross sectional view of NUWMAK tokamak reactor.

beam sources required for reactors (see Section 2.2.1 and 2.2.2). The problems of antenna compatibility with the fusion environment and coupling of the RF power to the plasma needs more attention. RF heating experiments on existing facilities have been encouraging, however.

The use of tritium breeder materials other than liquid lithium has been another trend. The focus has been on lithium-lead and lithium in the liquid form and solid breeders in the lithium-aluminate, silicate and oxide forms. These forms offer improved safety through the decrease in the fire potential. However, there is some concern that the use of the solid ceramics could lead to a higher tritium inventory in the blanket because of more difficult tritium extraction problems.

A recent development in tokamak reactor design studies has been the design of demonstration (DEMO) reactors. The DEMO is supposed to bridge the gap between an experimental power reactor such as INTOR or FED and a first commercial power reactor. The objectives of the DEMO are generally assumed to be the demonstration of a level of performance of all components, and the system as a whole, in an integrated power plant system which can be safely extrapolated to a first commercial power reactor. This includes system availability, tritium breeding, and safety aspects of plant operation. Two DEMO studies (10, 11) have been done to date; the parameters of the DEMO reactors are given in Table 2.3-2. The STARFIRE DEMO study is in the completion stage so the parameters are somewhat tentative. It is interesting to note that the STARFIRE DEMO is about the same size and magnetic field level as INTOR or FED, but uses physics assumptions (current drive, pumped limiter) similar to those for the STARFIRE power reactor.

2.3.2 Stellarators

Recent encouraging experimental results on stellarators have revived interest in these concepts as possible fusion power reactors. The use of modular coils to generate the field topology has added impetus to this interest. Two studies of the modular coil approach to stellarators have been conducted, UWTOR-M at the University of Wisconsin and MSR at LANL in cooperation with Westinghouse. Table 2.3-3 gives the main parameters for each of these designs.

UWTOR-M (12) is a 4820 MWt, $\ell = 3$ modular stellarator reactor with 18 discrete twisted coils, a major radius of 24 m and a plasma aspect ratio of 14. This magnetic topology leads to a rotational transform of 1.1 on the plasma edge producing high shear, and gives optimism that the assumed β of 6% can be achieved. The natural divertor with externally located divertor targets is used for impurity control. This study, which has primarily emphasized engineering aspects, has concentrated on two areas in its later phase, a credible coil design and overall system integration.

The initial constraints on the study were coil modularity and a magnetic divertor topology. Modularity is deemed essential for a power reactor to be maintainable. This is because of the complexity of helical magnets interfering with blanket replacement. A magnetic divertor, if it could be accom-

TABLE 2.3-3

Parameters for Several Stellarator Reactor Designs

	UWTOR-M	MSR-IIA	MSR-IIB
Major Radius (m)	24.1	27.9	23.0
Average Plasma Radius (m)	1.73	2.25	0.81
Total Thermal Power (MW)	4820	5100	4000
Net Electric Power (MW)	1836	1640	1290
Average Neutron Wall Loading (MW/m^2)	1.4	1.0	1.9
Peak Magnetic Field Coil (T)	11.7	10.9	~ 12
Average Beta (%)	6	4	8
Average Plasma Density (cm^{-3})	1.5×10^{14}	1.4×10^{14}	3.6×10^{14}
Average Ion Temperature (keV)	9.8	8	8
Edge Rotational Transform	1.1	0.43	0.92
Multipolarity	3	2	2

modated, would be a definite advantage. Although the assumed 6% beta was not quantitatively coupled to any stability/equilibrium model, the design goal was to achieve a high rotational transform with shear, avoid island formation and provide an adequate magnetic volume within a practical coil system.

A top view of the UWTOR-M magnet set is shown in Fig. 2.3-4 and a schematic of the reactor is shown in Fig. 2.3-5. Note that there are only two different types of magnetic coils. Detailed analysis of the magnets and support structure (not shown) with a finite element stress code has shown reasonable stresses for central fields of 4.5-5 T.

The stellarator flux surface requires twisting in the toroidal direction and so has no planes of

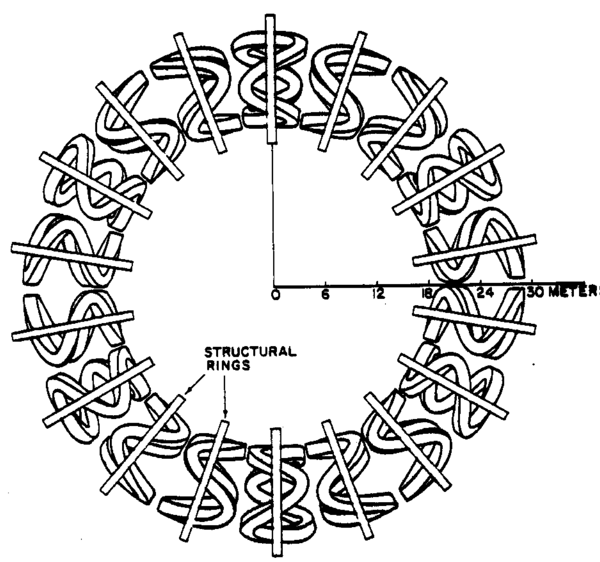

Fig. 2.3-4. Top view of UWTOR-M modular stellarator reactor coil set.

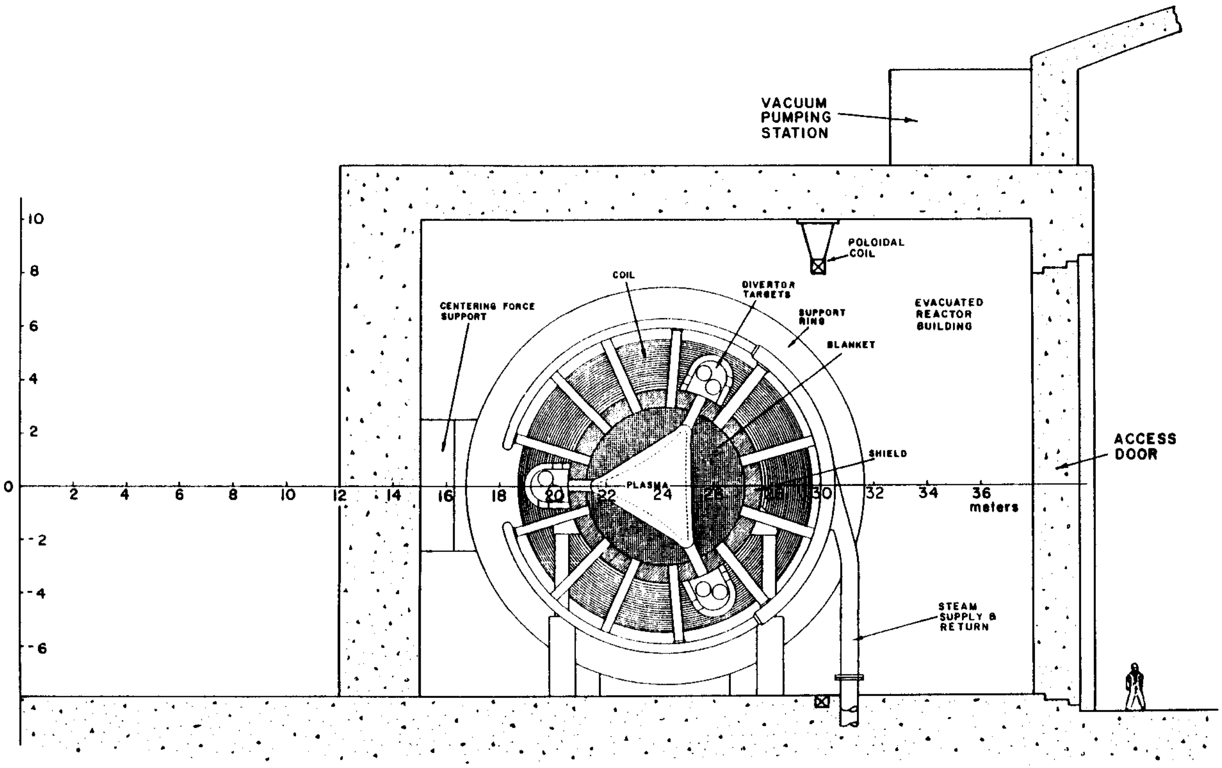

Fig. 2.3-5. Schematic of UWTOR-M Stellarator Reactor.

symmetry. This tends to make the reaction chamber shape more complicated. The blanket, however, can be divided into toroidal segments of constant shape reaction chamber with no twisting within each segment. Each segment has the reaction chamber rotated to accommodate the helicity of the flux surface. Divertor targets are located outside the shield and consist of a pair of cylinders for each divertor slot within a cooled housing. The core of the cylinders is an actively cooled stationary shield while the surface, consisting of a graphite covered shell, rotates slowly and radiates energy both to the core and the housing. This high grade energy is converted at a high efficiency in the power cycle. Neutron streaming from the divertor slot is prevented by the core shield.

Maintainability is provided by radial extraction of every other coil module, thus providing access to all the blanket segments and the divertor targets. The drained blanket segments are then removed, one from each side of the coil.

Earlier phases of the Los Alamos study of the Modular Stellarator Reactor (MSR) (13) have characterized parametrically the critical relationships between the plasma, blanket, shield, coil set, and overall reactor plant performance. Although "self-consistent", this approach used a yet-to-be proven, pessimistic model for stability/equilibrium beta limits based on diffusion-driven plasma currents. A "beta-decoupled" model has been adopted for more recent MSR studies in which beta is treated parametrically in order to better understand the crucial coupling and tradeoffs between the key elements of the design.

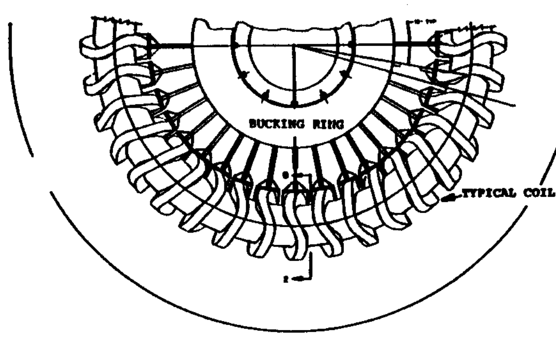

Fig. 2.3-6. Top view of one-half of the MSR coil set.

Two representative MSR design points, emphasizing traditional performance (MSR-IIA, β = 0.04) and optimistic performance (MSR-IIB, β = 0.08) respectively, have been identified and are presented in Table 2.3-3. Both commercial electric plants are assumed to operate as ignited, steady-state DT systems. Both MSR coil configurations consist of 36 modular coils with modest lateral deformation, in order to reduce individual coil mass, the ratio of peak coil field and on-axis field, the toroidal-field ripple and lateral coil force components. Relatively few MSR toroidal-field periods allow radial rotational transform profiles in the ranges 0.5-0.4 (MSR-IIA) and 0.7-0.9 (MSR-IIB), and provide significant non-zero transform on-axis and shear at the plasma edge (MSR - IIB). Except for the out-of-plane winding, the internal coil technology is comparable to other recent superconducting fusion reactor system designs. Figure 2.3-6 is a top view of one-half of the MSR coil set.

Impurity control is assumed to be provided by a pumped limiter. Routine maintenance and replacement of limiter/first-wall/blanket components would be accomplished without moving modular coils in order to promote high plant availability. Blanket modules are removed from the reactor through spaces between the coils.

2.3.2.1 Critical Issues. The major critical issues for stellarators are β, impurity accumulation and modular coils. The complex geometry in the stellarator makes the calculation of beta limits extremely difficult and costly. Further, there is some speculation that impurities may accumulate in the plasma center, eventually poisoning the plasma. Larger experiments are needed to test these speculations and to complement the calculations. There are proposals for helically twisting axis devices which some claim can achieve higher β.

Finally, it is believed that modular coils are essential for stellarators to be viable power reactors. A small scale test of the plasma performance of a stellarator with modular coils is under construction (see Section 2.1.3.1).

2.3.3 Tandem Mirrors

Two types of tandem mirror conceptual studies are described here. The first type is that of engineering test facilities, whose primary purpose will be to provide a test bed for the development of various technologies (e.g., neutral beams, ICRF and ECRH heating, magnets, etc.) as well as a neutron irradiation facility for materials and neutronic tests in an integrated facility. The second are commercial power reactor studies aimed at evaluating the potential of the tandem mirror concept as a power reactor. A good general reference for tandem mirrors is the National Mirror Fusion Program Plan (14).

2.3.3.1 Engineering Test Facilities. There have been four major studies of the tandem mirror as an engineering test facility. Some general parameters for these studies are given in Table 2.3-4.

MFTF-B+T. The study of MFTF-B+T is just beginning; it would be a modification of MFTF-B wherein a high field axicell would be inserted into the central cell and neutral beams of both tritium and deuterium would be injected (15). The device would produce a neutron wall loading of about 1 MW/m^2. It would run for 10 to 20 hour shots about once a month, limited by neutron activation. MFTF-B+T would test electricity production, thermal hydraulics, tritium breeding, and other reactor features dependent on a high neutron flux. Construction has not yet been authorized.

TDF. TDF, designed by a group led by LLNL and TRW, aimed at testing technology in both high neutron flux (~ 1.4 MW/m^2) and high neutron fluence (~ 5 $MW\text{-}y/m^2$) (16). Operation in the early 1990's was a major goal. Figure 2.3-7

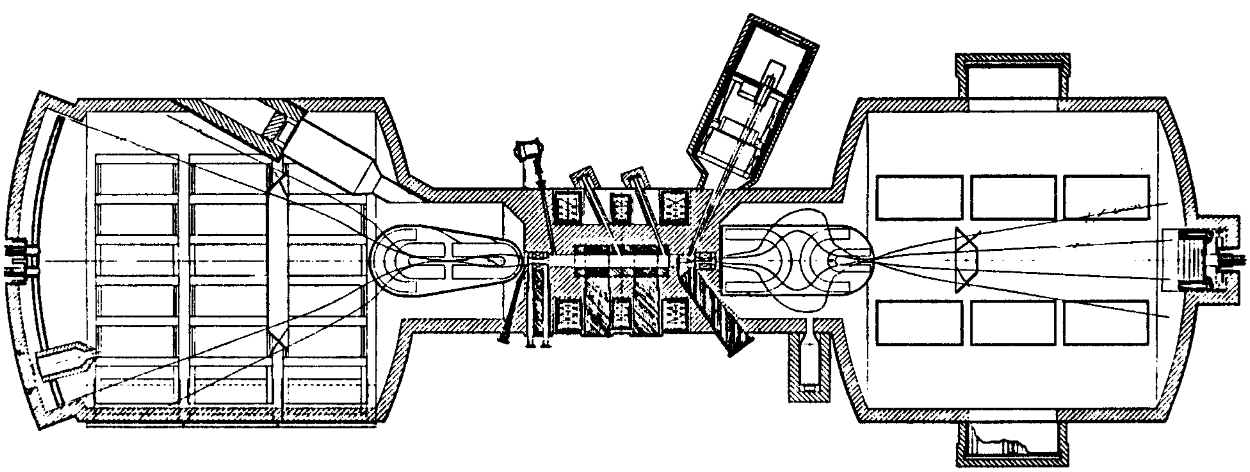

Fig. 2.3-7. TDF Configuration.

TABLE 2.3-4

Tandem Mirror Test Facility Parameters

Parameter	MFTF-B+T	TDF	TASKA	TMNS
General				
Neutron wall load, MW/m^2	1.12	1.4	1.5	0.5
Total input power, MW	24	70	117	114
Fusion power, MW	13	20	86	245
Total ICRF power, MW	---	---	40	---
Total ECRF power, MW	1.7	1.2	15	13.2
Total NB power, MW	22	69	62	101
Highest NB energy, keV	80	80	250	200
On-Axis Magnetic Fields				
Central cell, T	4.5	4.5	2.7	2.5
Barrier solenoid peak, T	12	15	20	---
Yin-yang peak, T	3	3	6.25	9
Plasma*				
CC density, cm^{-3}	3.8×10^{14}	6×10^{14}	1.9×10^{14}	9.9×10^{13}
CC mean ion energy, keV	49	37	45	45
CC plasma radius, m	0.15	0.10	0.32	0.83
CC length, m	5.6	8.0	19.2	51.
CC beta	0.4	0.4	0.5	0.5
Anchor beta		0.5	0.64	0.7

*CC ≡ central cell

shows the TDF configuration. The study was recently completed, and a report is to be issued in 1983. Because technology testing is the prime concern, less emphasis was placed on the tandem mirror concept. Although TDF relies on thermal barriers and a small plugging potential in a yin-yang end cell, the electrostatic potential only serves to contain a low density "warm" ion component in the central cell for microstability. The main, "hot" central cell ion component is generated by neutral beam injection, giving simple mirror confinement there. This concept is useful for providing a high power density and neutron flux, but it does not lead to an economic reactor because of the low energy multiplication.

TASKA. TASKA (Tandem Spiegelmachine Karlsruhe) was a joint study by the Kernforschungszentrum-Karlsruhe, in the Federal Republic of Germany, and by the University of Wisconsin (17). The aim of the project was to examine technology and materials testing on a 1990's time scale in a configuration that would scale to a reactor. The neutron wall loading is 1.5 MW/m^2 with a fluence of ~ 7 $MW\text{-}y/m^2$ over the operational lifetime. Some interesting features were:

- The REGAT materials test module was introduced, in which the first wall would be partially cut away, allowing added neutrons to impact the sides of the module. The result was a higher volume and reduced gradient of high neutron flux.

- A copper insert was included in the superconducting barrier solenoid, increasing the on-axis field to 20 T. This concept is now being pursued in the MARS and TDF reactor studies (see below).

TMNS and FPD Studies. The Tandem Mirror Next Step (TMNS) (18) and Fusion Power Demonstration (FPD) investigate devices intermediate in scope between the technological test facilities previously discussed and power reactors. They assume that physics and technology would be well in hand, and aim at integrating all systems into a small, not necessarily economic, reactor. The TMNS study examined a configuration in which a C-shaped coil is placed outside of a yin-yang, forming a magnetic mirror cell called the A-cell. Some features are attractive, but other studies (19) have shown that, in the reactor regime, the magnetic fields required for the C coil made its costs prohibitive. The study of FPD is only beginning. It would be similar in size and scope to TMNS, but would use the MARS configuration.

2.3.3.2 Conceptual Tandem Mirror Power Reactor Studies. The tandem mirror concept as a power reactor is a driven system since the end plugs require energy input to sustain the plasma there. The central cell plasma can be ignited (i.e., requiring no energy input to maintain the fusion "burn") but the complete system is driven. Consequently, one figure of merit for tandem mirror reactors is Q, which is the ratio of the fusion power produced to the power required to sustain the plasma. For an economic system, it is generally considered that Q should be greater than about 20, although the form of the injected power (i.e., cost of the power sources) is also important. A number of reactor studies have been made since the invention of the tandem mirror concept. The early studies were based on the "standard" tandem mirror configuration and concluded that it was difficult to get Q above 10, even using very optimistic physics assumptions. This led to the invention of the thermal barrier as a way of improving the electrostatic confinement of the central cell without requiring such large amounts of injected power in the end plugs.

Table 2.3-5 gives some general parameters for a standard tandem mirror reactor design and three newer designs utilizing thermal barriers.

LLNL Standard Tandem Mirror Reactor. This is the only full-scale tandem mirror reactor design without thermal barriers (20). Its parameters are given in Table 2.3-5. Despite the high technology (plug injection energy of 1.2 MeV, plug yin-yang field of 16.5 T) and optimistic physics assumptions (plug beta = 1, central cell beta = 0.7) a Q (≡ fusion power/injection power) value of only 5 was achieved--which gave impetus to the invention of the thermal barrier concept.

LLNL Thermal Barrier Tandem Mirror Reactor. This study was the first thermal barrier tandem mirror conceptual reactor design (21). Although the parameters given in Table 2.3-5 do not look particularly encouraging, the work was completed before the idea of hot, mirror-trapped barrier electrons for enhancing the thermal barrier was introduced. The design utilized the inside-barrier configuration, with the thermal barrier formed in the transition between the central cell and the plug.

WITAMIR-I. WITAMIR-I, a thermal barrier tandem mirror reactor design done at the University of Wisconsin, was a full-scale physics and engineering reactor design effort (22) utilizing the thermal barrier concept. The configuration was similar to that of TASKA (Section 2.3.3.1), and is shown in Fig. 2.3-8. An important conclusion was that the tandem mirror reactor could be cost competitive; the total cost per unit power was ~ 2130 \$/kWe in 1980 dollars. This was the first study to use $Li_{17}Pb_{83}$ eutectic as the coolant and breeding material in the blanket; this had advantages for tritium breeding and energy multiplication.

MARS. MARS is an extensive, two-year (1982/83) reactor study now in the midst of its second year (23). The parameters given in Table 2.3-5 are therefore, preliminary. It is a cooperative effort of LLNL, University of Wisconsin, and TRW, Inc., with subcontractors General Dynamics, Ebasco Services, Inc., Grumman Aerospace Corp., and Science Applications, Inc. Figure 2.3-9 gives an engineering view of the interim MARS configuration. The thermal barrier and plug are formed in the minimum-B end cell, as in TMX-U, MFTF-B, and TDF. Thus, MARS is the final link in the chain of machines which is presently considered the main tandem mirror reactor path, and will be based on extensive experimental experience. MARS will use $Li_{17}Pb_{83}$ in the main blanket, but will also examine high temperature production of synfuels in an alternative blanket. MARS is the first reactor study to address the question of alpha particle ash removal in detail. Although the Q-value given in Table 2.3-5 is low, newer concepts yielding Q ≃ 28 are being evaluated for MARS.

GAMMA-R. A preliminary conceptual design of GAMMA-R is being done in Japan. Based on the GAMMA-10 configuration, the barriers will be formed in the transition between the anchors and the plugs. An interesting feature is the use of passive barrier pumping--creating a region where ion drift orbits are unconfined. A major problem will be shielding the yin-yang coils from the high neutron flux.

TABLE 2.3-5

Tandem Mirror Reactor Parameters

Parameter	LLNL Standard TMR	LLNL Thermal Barrier TMR	WITAMIR-I	MARS (1982 Interim Design)
Year of Study	1977	1979	1980	1982/83
Fusion power, MW	2532	1770	3000	3500
Net electric power, MW_e	1000	527	1530	1360
Total injected power, MW	587	163	123	265
Neutron wall load, MW/m^2	2.1	2.9	2.4	5.0
Breeding material	Li	Li_2O	$Li_{17}Pb_{83}$	$Li_{17}Pb_{83}$
Structure	SS	SS	HT-9	HT-9
Total ICRF power, MW	---	---	---	46
Total ECRF power, MW	---	108	49	56
Total NB power, MW	587	55	74	163
Highest NB energy, keV	1200	400	500	475
Energy Multiplication, Q	4.3	11	24	13
Magnetic Fields				
Central cell, T	2.4	2.8	3.6	4.7
Solenoid peak, T	---	12	14	24
Yin-yang peak, T	17.7	6	6	6
Plasma*				
Density, cm^{-3}	1.1×10^{14}	1.4×10^{14}	1.5×10^{14}	4.2×10^{14}
Ion temp., keV	30	25	32.5	35
Plasma radius, m	1.22	1.04	0.72	0.43
Length, m	101	56	165	150
On-axis beta	0.7	0.75	0.4	0.7
Anchor beta	1.0	0.6	0.64	0.65

* central cell values

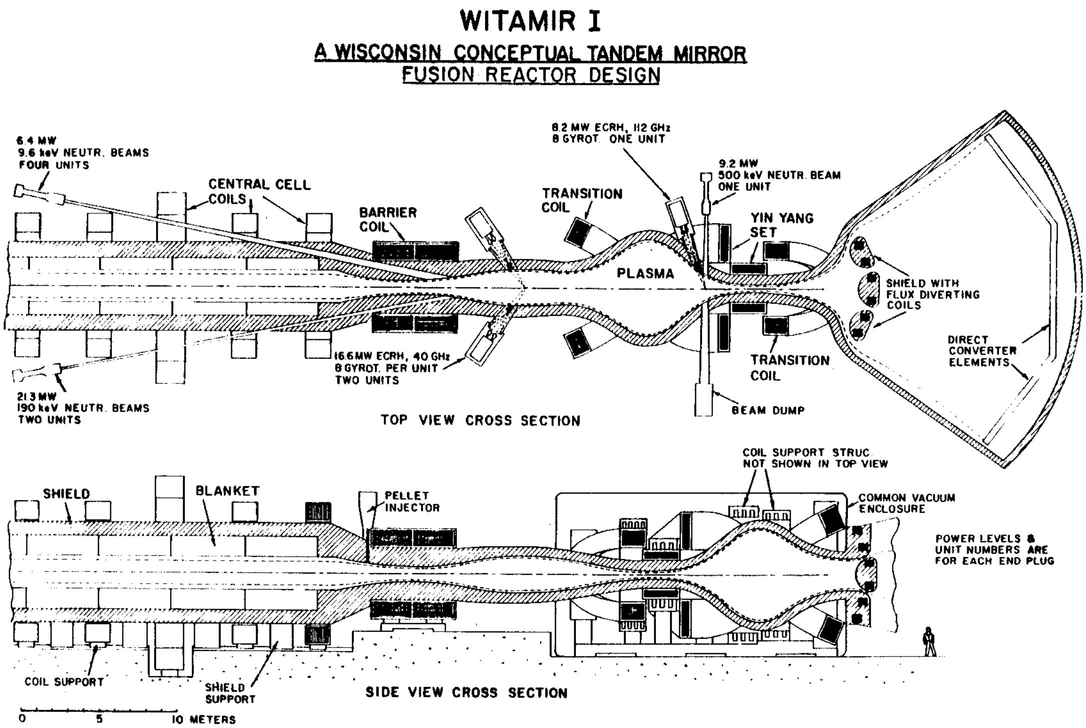

Fig. 2.3-8. WITAMIR-I Configuration.

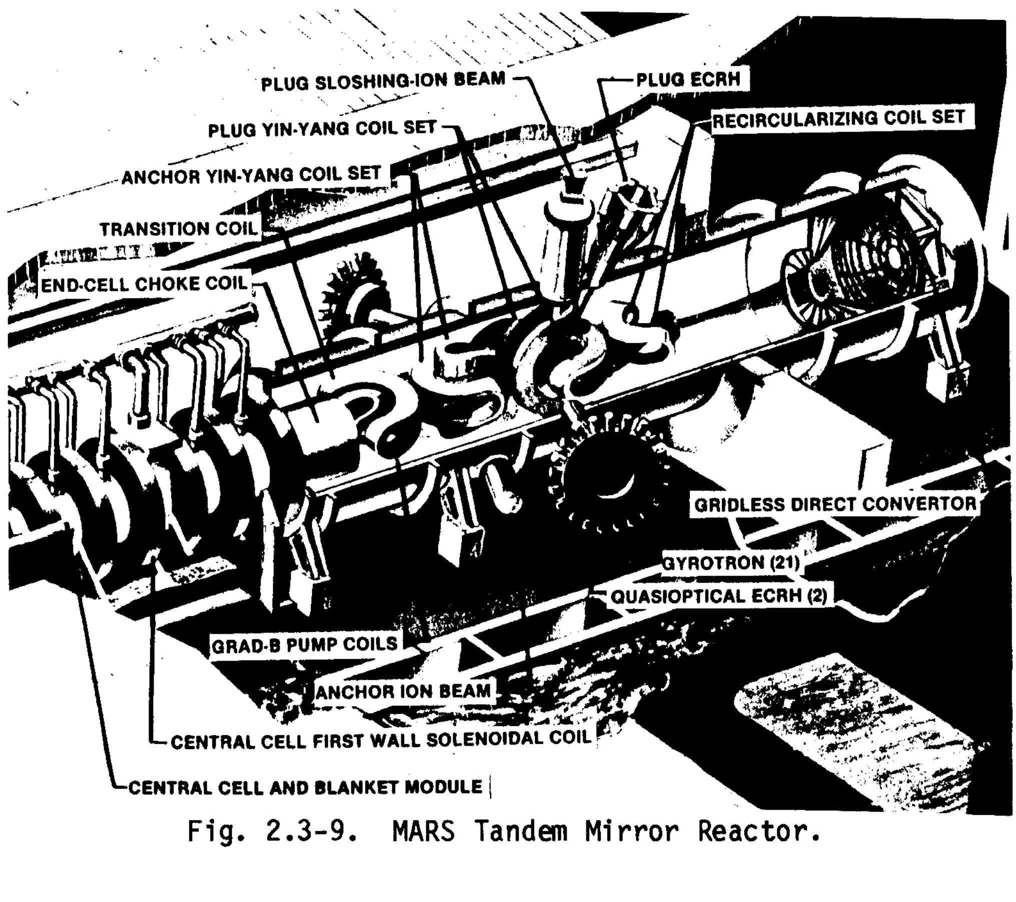

Fig. 2.3-9. MARS Tandem Mirror Reactor.

2.3.3.3 Critical Issues. Figure 2.3-10 shows the confinement parameter ($n\tau$) versus central cell ion temperature for present experiments, the next generation devices, test facilities, and reactors. All of these devices lie within a general band leading to the reactor regime. In comparison with tokamaks, the data base for reactor design is not as well developed. This is due to the newness of the concept and the time required to build facilities. Consequently, the tandem mirror reactor studies are largely based on theoretical concepts with little experimental verification. Providing a solid experimental foundation for these concepts is a crucial task. If the present concepts prove to be valid, then the tandem mirror has many potential advantages for a reactor. These include truly steady-state operation, lack of plasma disruptions, and reduced plasma-wall interaction problems. The open geometry

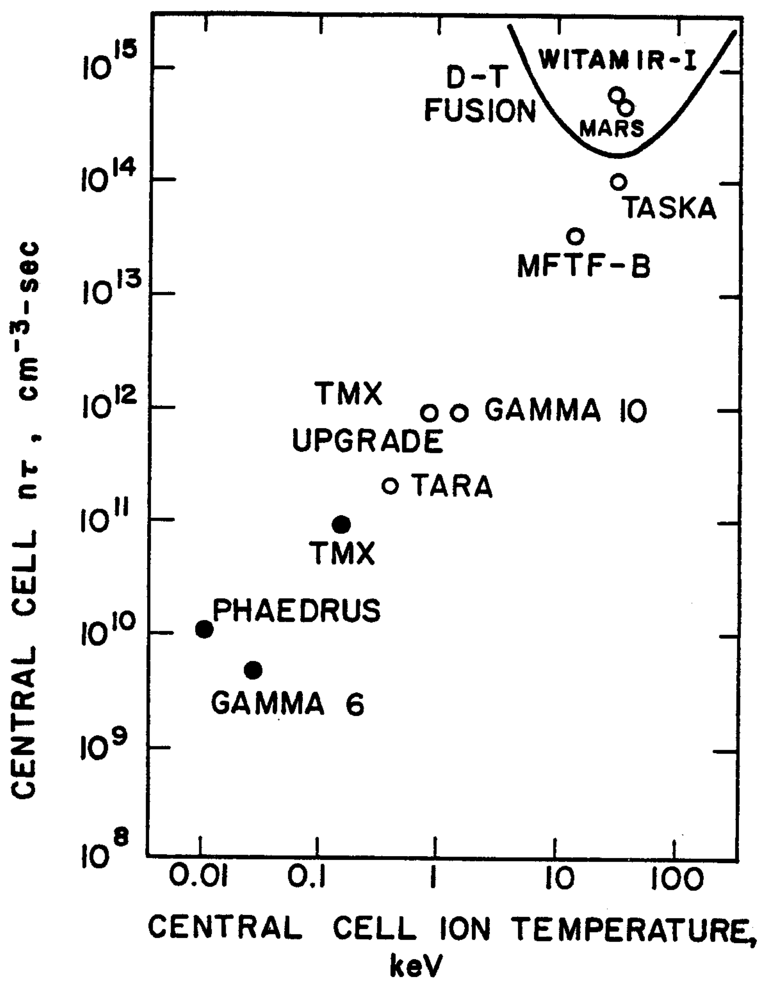

Fig. 2.3-10. $n\tau$ versus central cell ion temperature for present (•) and future (○) tandem mirrors.

should have a considerable advantage for maintainability, in comparison to toroidal geometry, although this has not yet received sufficient study. The design of the end plugs, however, is becoming more complicated. Maintainability of the end plugs has also not yet been adequately addressed.

Most of the physics issues pertinent to reactors also impact the experiments described in Section 2.1.2 and are discussed there. The chief exceptions are alpha particle and ECRF heating physics. No plasmas with significant alpha particle populations presently exist in the laboratory, but a strong theoretical effort addresses the relevant questions: ash removal and alpha particle driven instabilities. Experimental study awaits the construction of a TDF, TASKA, FPD, or an upgrade of MFTF-B. No adequate, relativistic ECRF heating theory exists for the hot electrons in the thermal barrier; experiments will probably lead theory in understanding the behavior of hot, mirror-trapped electrons.

Technological issues include:

- Materials questions related to neutron damage await a high flux, high fluence, reasonable volume test device like TASKA or TDF.
- High power, high frequency, reasonable cost ECRF sources must be developed. Present tandem mirror reactor concepts make substantial use of gyrotron microwave sources for electron heating (see Section 2.2.1.2).
- High energy, continuous neutral beam injectors are needed to sustain the end plug plasma. These will probably require negative-ion based sources, which require considerable development (see Section 2.2.1.1).
- Hybrid magnets, with copper coils inside superconducting coils are required by TASKA and MARS. No insurmountable difficulties are foreseen with the development of such magnets except for the development of radiation damage resistant insulators. Lower field, purely superconducting magnets are a backup possibility but the performance would suffer.

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Section 3

INERTIAL CONFINEMENT FUSION

3.1 INTRODUCTION

3.1.1 Program Overview

The inertial confinement fusion program in the United States has the dual goal of both military and civilian applications. The military application is near term and is the development of a laboratory test bed for the study of nuclear weapons related physics and nuclear weapons effects. The eventual application of ICF to commercial electric power production is viewed as a secondary long range issue. Within the last five years, amidst ever-tightening budgets, there has been less and less emphasis placed on commercial applications. In fact the U.S. House of Representatives Armed Services Committee has specifically forbidden the spending of any ICF funds for the study of non-military applications. This policy decision is demonstrated by the budget figures in Table 3.1-1.

For the near term physics experiments, planning for the future is restricted to a very short time frame. The DOE policy today is to build the lasers and pulsed power machines that are currently in the budget (all will be operational by 1987) and wait for conclusive results from these before deciding on the next step. The anticipated operating budget for the next five years is there-

TABLE 3.1-1

U.S. Funding of Inertial Confinement Fusion (M$)

	FY-81	FY-82	FY-83
Operating	139	127	105
Construction	69	82	13
Support Research Part of Operating Budget (including applications studies)	17.2	3.3	0.5

fore flat. Added to this policy is the fact that two of the three future driver systems have been scaled back from their originally intentioned energy levels as shown in Table 3.1-2. Only PBFA-II at Sandia Laboratory is actually being built to design specifications originally proposed. These programmatic decisions have placed federally funded ICF research directed toward commercial applications at a standstill. They also seriously jeopardize the timely demonstration of high target gain for either military or civilian applications.

This policy decision is unfortunate in light of the potential of ICF as the power source for fusion reactors. The studies that have been done to date demonstrate that ICF reactors have some unique and attractive features, such as the nearly complete separation of the sophisticated driver machine from the reaction chamber. The nuclear island of an ICF reactor could more closely resemble the pressure vessel of current light water reactors than any other fusion scheme.

The link to nuclear weapons development was explicitly confirmed in 1980 with the release of a declassified statement regarding both ICF and nuclear weapons:

> *"In thermonuclear weapons, radiation from a fission explosive can be contained and used to transfer energy to compress and ignite a physically separate component containing thermonuclear fuel. In some ICF targets, radiation from the conversion of focused energy (e.g., laser or particle beam) can be contained and used to transfer energy to compress and ignite a physically separate component containing thermonuclear fuel."*

Memo to Los Alamos National Laboratory Employees
from J.W. McDonald, Classification Office.

This statement is a significant step in the declassification of ICF target research. However, at this time, all details of ICF target design remain

TABLE 3.1-2

Driver Systems Under Construction

	Original Energy	Funded Energy
NOVA (Glass laser, LLNL)	200 kJ	100 kJ
ANTARES (CO_2 laser, LANL)	100 kJ	40 kJ
PBFA-II (Pulsed power, SNL)	3.5 MJ*	3.5 MJ*

* energy to diode

classified as Secret Restricted Data. Because classified research in this area of target design and fabrication represents the greatest fraction of work at the major ICF laboratories within the U.S., Great Britain and France, this report will not be able to summarize much of the progress made during the past five years. The reader should keep this in mind, especially in this chapter.

A rising star in the world ICF community is Japan. Within the last five years, Japan has emerged as a major participant in ICF research. The Institute for Laser Engineering at Osaka University has built world class Nd:glass and CO_2 lasers; it is actively pursuing target design and implosion studies; and it is liberally publishing the results from this research.

In addition to the U.S., Great Britain, France, and Japan there is a substantial, although unquantifiable, amount of ICF work conducted in the Soviet Union. Information regarding the U.S.S.R. program is scant and there is little communication between Soviet and free-world scientists in comparison to the extensive collaboration in magnetic fusion.

In the U.S., ICF research is conducted through a "lead laboratory" framework that was instituted within the past five years. The federal ICF office in Washington, D.C. has relinquished much of the control of the program to the three major nuclear weapons laboratories: Lawrence Livermore National Laboratory, Los Alamos National Laboratory, and Sandia National Laboratory. These three laboratories are responsible for the three principal approaches to driving ICF targets to fusion conditions:

- Nd:glass lasers (wavelength conversion) — LLNL
- CO_2 gas lasers, heavy ion — LANL
- Pulsed power — SNL

The major ICF experimental facilities within the U.S. are shown in Table 3.1-3. The other laboratories serve as support organizations to the three principal mission oriented laboratories. On the worldwide scene there are many laboratories with lasers of varying size that are performing ICF related research. A list of these is given in Table 3.1-4.

Amid the gloom of the preceeding programmatic review it must be emphasized that a great deal of progress <u>has</u> been made in the understanding of ICF physics over the past five years. Although the future of the civilian program, in the U.S. at least, is discouraging, the past has seen many exciting discoveries and achievements. These will be reviewed in the remainder of this section. To place these achievements in perspective we will first review, in some detail, the requirements for reaching commercially viable ICF based electric power plants. This can then be used as the "yardstick" to measure the progress over the last five years and the current status of ICF.

Table 3.1-3

Major ICF Facilities in the United States

Lawrence Livermore National Laboratory	Energy (kJ)	Power (TW)	
ARGUS (shut down)	2	5	Nd:glass
SHIVA (shut down)	10	30	Nd:glass
NOVETTE	15	15	Nd:glass
NOVA	100	100	Nd:glass
Los Alamos National Laboratory			
HELIOS	10	10	CO_2 gas
ANTARES	40	40	CO_2 gas
Sandia National Laboratory			
PROTO-I	22	1.1	Pulsed power
PROTO-II	500	10	Pulsed power
PBFA-I	900	30	Pulsed power
PBFA-II (under construction)	3500	100	Pulsed power
KMS Fusion, Inc.			
CHROMA-I	0.9	1.9	Nd:glass
Naval Research Laboratory			
PHAROS-II	1.3	0.7	Nd:glass
GAMBLE-II	50	0.25	Pulsed Power
University of Rochester			
GDL	---	0.7	Nd:glass
OMEGA	4	12	Nd:glass

3.1.2 Requirements for Commercially Viable ICF Electric Power Plants

The eventual commercialization of ICF depends upon two related, yet distinct subjects: (1) target and driver performance and (2) economics. We will emphasize the first of these two issues but will show how economics is related to these considerations.

The power flow in an ICF reactor power plant is shown schematically in Fig. 3.1-1. The major features include: (1) the driver system that converts electrical power into driver beam power with an efficiency η_D, (2) the fusion target that multiplies the input driver energy into fusion energy with a multiplication factor or gain G, (3) the blanket system that captures the fusion energy (principally in the form of neutrons), multiplies this energy by a blanket multiplication factor M_B (usually 1.-1.4) and converts it to thermal energy, and (4) the electrical conversion system that converts thermal energy in the form of steam into electrical energy at the thermal efficiency η_{th}. A fraction f_R of this gross electrical power must be recirculated to operate the driver and other support systems, and the remainder is available for sale to the customers. The economic viability of this system depends upon the re-

Table 3.1-4

Major World Energy Drivers for ICF Research

1. Nd:Glass Laser

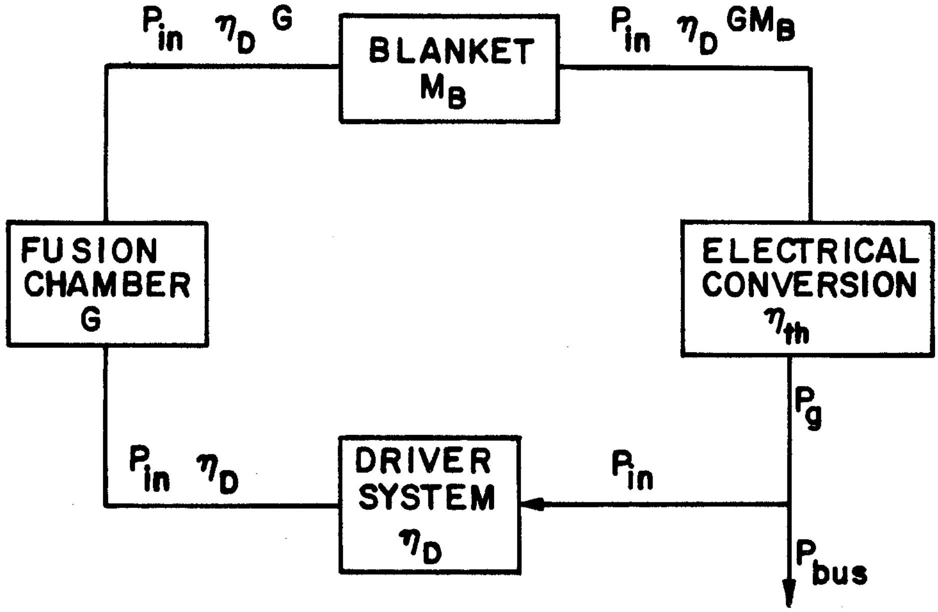
Country	Lab. Name	Number of Beams	Output Power (TW)	Output Energy (kJ)	Pulse Length (ns)	Remarks
USA	LLNL					
	Argus	2	5	2	0.03~1.0	10^9 Neutron
	Shiva	20	30	10	0.1~2.0	$3x10^{10}$ Neutron, 20 g/cc
	Novette	2	15	15	0.1-2.0	40 g/cc (1983)
	Nova	10	100	100	0.1~5.0	200 g/cc (1984)
	LLE					
	U. Rochester					
	GDL	1	0.7	---	0.1	
	Omega-X	24	12	4	0.03~0.1	10^{10} Neutron, 6 g/cc
	NRL					
	Pharos-II	2	0.7	1.6	0.1~1.0	
	KMS					
	Chroma-I	2	0.9	1.9	0.1	
USSR	Kurchatov					
	Mishen	4	---	1	1.0	
	Lebedev					
	Delfin	216	33	100	0.2~3.0	
	Aurora	20	---	50~500	0.03~10	} Under Construction
	UMI-35			100		
JAPAN	ILE Osaka					
	Gekko-II	2	0.4	0.12	0.1~1.0	
	Gekko-IV	4	4	1	0.05~1.0	10^8 Neutron, 5 g/cc
	Gekko M-II	2	7	3	0.1~1.0	
	Gekko VII	12	40	20	0.1~1.0	(1983)
	IPP Nagoya					
	Halna	1	0.1	---	0.1	
	ELI					
UK	Rutherford					
	Vulcan	6	3.6	1.2	0.1~1.0	
	AWRL					
	---	2	1	1	0.05~1.0	
FRANCE	Limeil					
	P102	2	0.6	---	0.08	
	Octal	8	2	1	0.1~1.0	
	Ecole Poly. Tek.					
	Greco	1	0.25	0.25	0.1~2.5	

Table 3.1-4 (continued)

Country	Lab. Name	Number of Beams	Output Power (TW)	Output Energy (kJ)	Pulse Length (ns)	Remarks
ITALY	Frascati					
	---	2	---	0.25	2.0	
POLAND	Kalisky Inst.					
	---	4	---	0.1	3.0	
AUSTRALIA						
	LIT-IV	4	1.2	0.12	0.005~0.01	

2. CO_2 Laser and Iodine Laser		Number of Beams	Output Power (TW)	Output Energy (kJ)	Pulse Length (ns)	Remarks
USA	LANL					
	DBS	2	2.5	2.5	1	10^{10} Neutron,
	Helios	8	10	10	1	} 20 g/cc,
	Antares	2x(12)	40	40	1	(1983)
JAPAN	ILE Osaka					
	Lekko-II	2	1	1	1	10^6 Neutron
	Lekko-VIII	8	10	10	1	
CANADA	NRC					
	Coco II	1	0.2	0.2	2	
ITALY	Frascati					
	Shimera	2	0.2	0.2	1	
POLAND	Kalisky Inst.					
	---	8	6	6	1	
FRG	IQO Garching					
	Asterix III	1	1.1	0.4	0.35	Iodine laser

3. Particle Beam		Number of Modules	Output Power (TW)	Diode Voltage (MV)	Pulse Length (ns)	Remarks
USA	Sandia					
	Proto II	1	10	1-3	50	0.5 MJ
	PBFA-I	36	30	2-4	30	0.9 MJ
	PBFA-II	36	100	2-28	35	(1987) 3.5 MJ
USSR	Kurchatov					
	Angara V-M	1	1	2.0	80	0.1 MJ (> 1983)
	Angara V	50	50	2.0	80	5 MJ (proposed)
FRANCE	Valduc					
	Sidnix	1	1	1.0	80	50 KJ
FRG	Karlsruhe					
	Kalif	1	1	5.5	45	55 KJ
JAPAN	ILE Osaka					
	Reiden IV	1	1	1.0	60	50 KJ
	Reiden IV-H	1	1	3.0	60	50 KJ



η_D = DRIVER SYSTEM EFFICIENCY

G = TARGET GAIN

M_B = BLANKET ENERGY MULTIPLICATION

η_{th} = ELECTRICAL CONVERSION EFFICIENCY

P_{in} = ELECTRICAL POWER INPUT TO DRIVER

P_g = GROSS ELECTRICAL POWER

P_{bus} = BUS BAR POWER

$f_R = P_{in} / P_g = 1/\eta_D\ G\ \eta_{th}\ M_B$

Fig. 3.1-1. Power flow diagram for an ICF power plant.

circulating power fraction. If f_R is too large, then the system will be uneconomical. An expression for f_R is shown in Fig. 3.1-1. The recirculating power fraction is plotted as a function of the parameter $\eta_D G$ in Fig. 3.1-2. Prior to 1977 it was thought that values of $f_R = 0.25$, hence $\eta_D G = 10$, would be adequate for economical operation. However, more detailed economic analyses done by the University of Wisconsin Fusion Program have shown that $\eta_D G \approx 20$ and $f_R = 0.12$ are required for ICF systems to be competitive with other similarly analyzed magnetic fusion systems. This discussion calls out the strongest link between target and driver performance and economics. The product of target performance (i.e., gain) and and driver efficiency must equal about 20 to have interesting (i.e., economic) reactors (1, 2).

We next turn to a brief discussion of driver performance. Figure 3.1-3 shows an $\eta_D G = 20$ contour and places the various expected driver efficiencies on the curve for lasers and ion beam drivers. Lasers are basically low efficiency devices and require large target gain to be acceptable for commercial systems. A possible exception is the free electron laser discussed in Section 3.2.3.4. Ion beams have potentially higher efficiency and therefore allow lower target gains. This fact has led reactor designers in the past five years to consider ion accelerators (particularly heavy ion accelerators) as potential ICF reactor drivers.

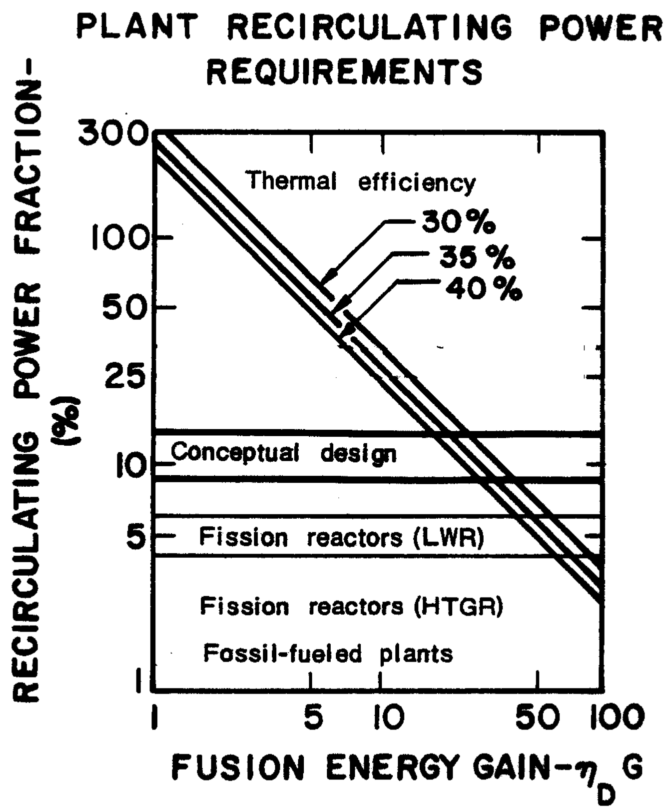


Fig. 3.1-2. Plant recirculating power vs. fusion energy gain - $\eta_D G$. $M_B = 1.0$.

The target performance or gain is of course functionally independent of the driver efficiency. However, the target gain does depend upon the size of the driver (i.e., the amount of driver input energy). The best estimate of this dependence is shown in Fig. 3.1-4. Also plotted is the theoretical maximum gain achievable. The combination of Fig. 3.1-3 and Fig. 3.1-4 shows that laser drivers, as we know them today, will have a difficult time meeting the requirements of commercial ICF. Ion beams, on the other hand, offer a potentially attractive alternative. Advanced drivers for eventual commercial systems that are efficient and can be operated in a repetitive mode are a low priority in the military applications oriented ICF program of today and consequently there is little attention paid to this part of the total picture.

The common thread between military and commercial applications is high target gain. It is to this end that most of the experimental and theoretical effort of the past five years has been devoted. Figure 3.1-5 shows the important issues associated with target implosion, ignition and high gain. This figure will serve as the focal point for much of the remainder of this chapter. The basic features of all forms of ICF are the same. The driver energy is deposited in the outer layers of a spherical shell target. The absorbed energy ablates material from the surface and the equal but opposite reaction drives an implosion of the inner part of the shell, including the DT fuel. In the case of laser beams (which have accounted for > 99% of the ICF target experiments done during the past five years) the laser light cannot penetrate any further into the plasma than the so-called critical density. For 1 μm wavelength lasers (Nd:glass) this corresponds to 10^{21} cm^{-3} and for 10 μm lasers (CO_2) this critical density is 10^{19} cm^{-3}. The energy is carried from the region of critical density into the higher density region, finally to the ablation front, by electron thermal conduction. The imploding material ahead of the ablation front must be kept cold (i.e., on a low isentrope) so that the work needed to compress it is minimized. For an efficient "burn" the fuel must be compressed to ~ 1000 times liquid DT density. However, the implosion velocity must be ~ 3 x 10^7 cm/s so that the DT fuel will be shock heated to a 4 keV temperature when it converges to the center of the target. The implosion must be carefully programmed so that the compression is done with the least amount of work on the fuel and only the central 1% of the DT is raised to ignition temperatures. To do otherwise would require too much energy and would thus reduce the gain below acceptable levels. Once the fuel is put into this ignition configuration it will burn in a self-sustained fashion almost instantaneously. Hence all ICF target experiments are directed toward the ultimate goal of reaching this ignition configuration.

Many things can go wrong with this complex dynamical system. These include:

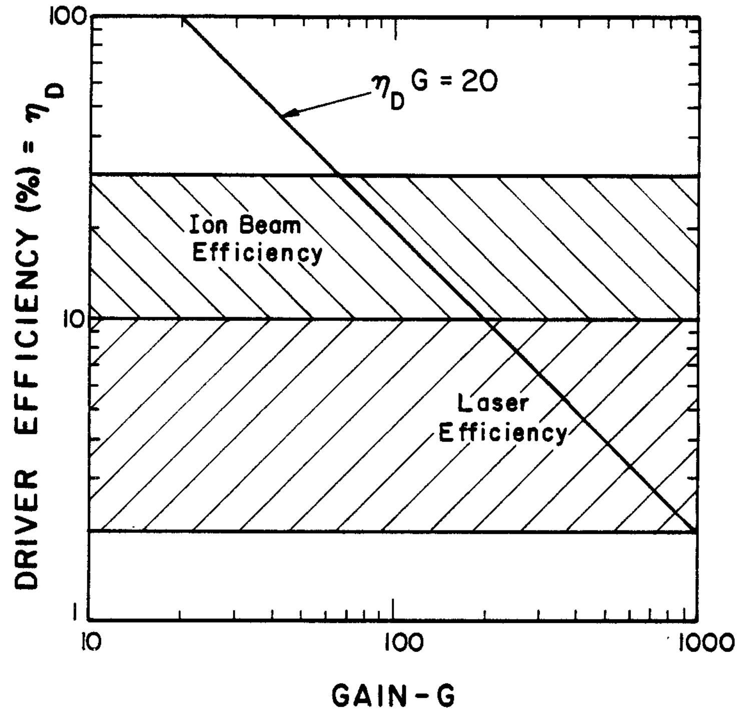


Fig. 3.1-3. Fusion energy gain = 20 contour and acceptable driver efficiencies.

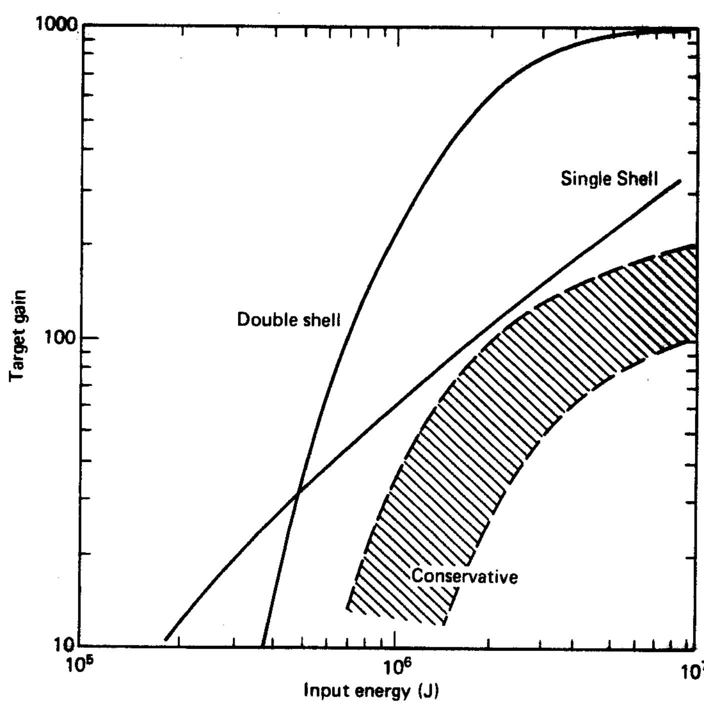


Fig. 3.1-4. Target gain predictions.

IMPORTANT TARGET PHYSICS FOR HIGH GAIN

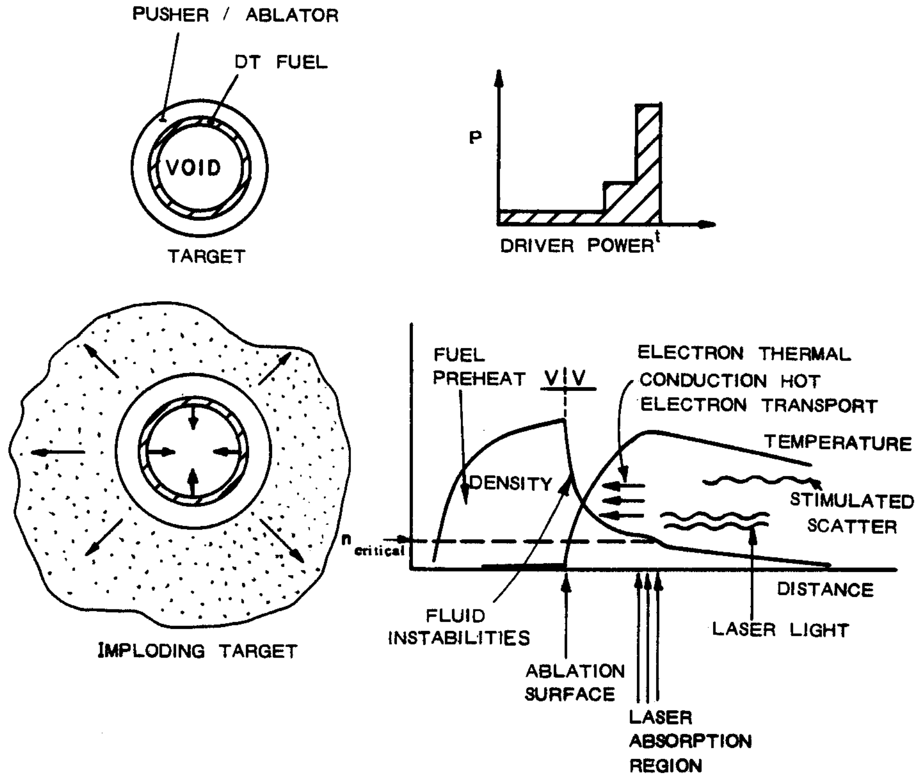


Fig. 3.1-5. Important target physics issues for high gain.

- Stimulated reflection of the laser light from the target.
- Creation of high energy electrons by the laser light absorption through complex plasma-EM wave coupling.
- "Anomalously" inhibited transport of electron thermal energy from the critical density to the ablation front.
- Hydrodynamic fluid instabilities that destroy the very high degree of symmetry required to reach 1000 x liquid density.
- Nonuniformities in the energy deposition that destroy the implosion symmetry.

Table 3.1-5 gives the values of parameters required for high target gain and the values currently achieved. Many of the required values have been achieved or nearly achieved over the past five years, the period of interest for this report. Those values that are far from achievement are all related to the driver size. We require a 10 to 100 fold increase in driver energy over that of Nova-I to reach the high gain goal. This is at least several times more energy (and hence more money) than was anticipated prior to the five year period we are reviewing here. This fact alone could be held responsible for the programmatic slowdown that ICF has taken. Details of the experiments and analysis leading to these achievements will be discussed in the following sections.

TABLE 3.1-5

Parameters Required for High Gain and Those Achieved to Date

Parameter	High Gain	Currently Achieved
Gain	100-1000	6×10^{-5}
Ion temperature	4-8 keV	4-8 keV
DT density	~ 1000 x liquid density	~ 100 x liquid density
ρR confinement	3 g/cm^2	~ 0.03 g/cm^2
Driver energy on target	1-10 MJ	10-20 kJ
Driver power on target	100-1000 TW	5-20 TW
Driver intensity	100-1000 TW/cm^2	100-1000 TW/cm^2
Absorption efficiency	~ 80%	~ 30-80%
Fuel isentrope	~ 1 eV	~ 10 eV
Irradiance uniformity*	~ 1%	~ 5-10%
Surface finish quality	~ 100-1000 Å	~ 1000 Å

* not important for classified targets.

REFERENCES FOR SECTION 3.1

1. SOLASE - A Conceptual Laser Fusion Reactor Design, University of Wisconsin Fusion Engineering Program Report UWFDM-220, December 1977.

2. B. Badger et al., HIBALL - A Conceptual Heavy Ion Beam Driven Fusion Reactor Study, University of Wisconsin Fusion Engineering Program Report UWFDM-450, 'une 1981.

3.2 PHYSICS PROGRESS

3.2.1 Driver Beam Target Interaction

The interaction between incident driver beams and the material in the outer shells of targets is a topic of great importance to inertial confinement fusion. It has been vigorously studied for many years at government laboratories, universities and private industry in the United States and throughout the world. Most of the work has been in the study of the absorption and scattering of laser light in the plasma corona which surrounds a laser fusion target and, in the last five years a great deal has been learned through experimental and theoretical investigations. Fundamentally different in nature from laser-target interactions, light and heavy ions deposit their energy in dense material in ways very different from those found to be important to laser fusion. To date, there has been theoretical investigation of the stopping of intense ion beams in target matter and some experimental results have been reported.

3.2.1.1 Laser-Target Interactions. In the past five years, there have been substantial gains made towards understanding laser-target interactions. In 1977, many of the diagnostic techniques which are standard today were just being developed and the operation of lasers with pulse energies greater than 1 kJ was just beginning. The intervening years have seen a great number of interesting experimental results which have stimulated the development of theoretical models for laser-plasma interactions. The advances that have been made include: (1) understanding of the importance of resonance absorption and its connection with the generation of energetic electrons and ions, (2) the occurence of self-generated megagauss magnetic fields and the subsequent effects on thermal transport, (3) plasma instabilities leading to stimulated scattering and absorption and filamentation of the incident laser, (4) conversion of laser energy to x-rays, and (5) the modification of the plasma density by the incident laser and its effect on the absorption and scattering processes. The dependence of these processes on the frequency of the laser and on the geometry of the experiment, including the irradiation symmetry, has been established. A major consequence of this work has been the understanding that laser wavelengths between 0.25 µm and 0.53 µm have substantially better coupling to the target than those in the 1 µm to 10 µm range: this effectively removes CO_2 lasers as ICF drivers.

The material reported in this section may all be found in a recent review of the subject (1) and the proceedings of the May 1982 Japan-U.S. seminar on the subject (2).

We will begin with a brief overview of the absorption and scattering of laser light by the plasma around the target and a description of the state of this plasma during the laser irradiation. Figure 3.2-1 shows an electron density profile of the laser-produced plasma. In the region of high density, energy is transported inward by electrons. Light cannot propagate through matter above the critical density, where the laser frequency is equal to the plasma frequency. Resonance absorption heats the plasma at the critical density.

While the light is approaching the critical surface it may be affected by the underdense plasma where the plasma frequency is less than the laser frequency. In the underdense region, the light may be Raman or Brillouin scattered or may be absorbed via plasmon decay at the point where the density is 1/4 of the critical density. All of these effects are discussed in more detail in what follows.

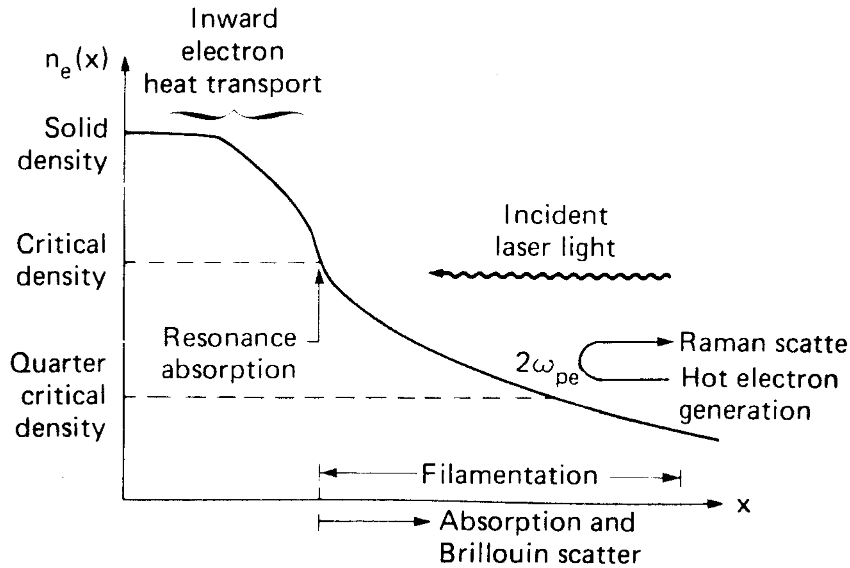


Fig. 3.2-1. Schematic picture of electron density profile in a laser-produced plasma, showing location of major plasma-physics coupling processes. The "critical density" $n_c = m_e\omega_L^2/4\pi e^2$ is the highest electron density accessible to laser light of frequency ω_L.

Resonance absorption has been understood for many years to be an important process in laser-target interactions for 10 μm laser light but is much less important at short wavelengths of interest presently, where collisional absorption dominates. Resonance absorption occurs through the excitation of local plasma oscillations matched to the laser frequency at the critical surface. This "collisionless" absorption is expected to dominate over collisional inverse Bremsstrahlung when the temperature of the plasma is high because the latter vanishes in collisionless (high temperature) plasmas. This has been experimentally observed by measuring the dependence of laser absorption on the polarization and the angle of incidence of the laser light to the target. Resonance absorption only can occur when incident laser light, which is obliquely refracted off of the plasma density gradient, has a component of its electric field vector parallel to the density gradient at the point where the refracting beam is closest to the critical surface (Fig. 3.2-2). Thus there can only be resonance absorption when the incident laser is polarized in the plane of the incident and refracted wave (p-polarization), which is also the plane of the density gradient, and when the laser is not incident parallel to the density gradient. Five years ago this behavior was seen in absorption measurements of lasers on planar targets, thus verifying that resonance absorption is responsible for the collisionless absorption at the critical density.

The experiments of that time also showed that energetic ("hot") electrons and ions are emitted from plasmas under intense laser irradiation. It was under-

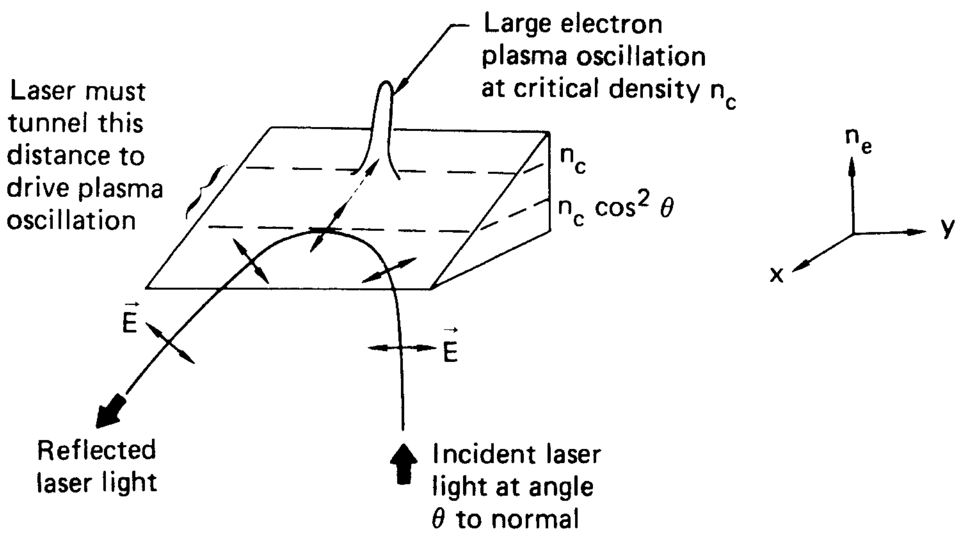


Fig. 3.2-2. Schematic configuration for resonance absorption of p-polarized light, where E is in the plane of k_L and ∇n_e.

stood from experiment and theory that these particles, in part at least, are generated by the resonance absorption process. There is still work in progress on the explanation of the approximately Maxwellian energy distribution of these particles and representative of the hot electron temperature which has been experimentally found to scale like $(I/\omega_L^2)^n$ where n is between 0.33 and 0.39. Thus, the hot electron temperature decreases as the laser frequency increases, as is shown in Fig. 3.2-3.

Five years ago it was known that these hot electrons contribute to the generation of large d.c. magnetic fields and in the intervening years these self-generated magnetic fields have been measured through Faraday rotation techniques which allow better spatial resolution. It has been found that these fields are dependent upon the geometry and time history of the laser irradiation. The latter effect can be explained by a large return current required to preserve charge neutrality in the region of the laser focal spot and by the existence of a mechanism for large current generation other than resonance absorption. These other mechanisms include currents generated by gradients in the electron density and temperature. A circular focal spot on a planar target generates toroidal magnetic fields around the focal spot. Magnetic fields generated by any of these processes may be in excess of 1 megagauss. The computational methods presently available are not refined enough to determine what mixture of mechanisms corresponds to the observed generated fields. In addition to these large scale magnetic fields, which have dimensions on the order of the focal spot size, there are also smaller scale magnetic fields proposed which have spatial scale lengths of from a laser wavelength to an electron mean free path and which would be the result of various instabilities in the plasma. While they have not been observed because of their small spatial size, if they exist they may play an important role in electron transport.

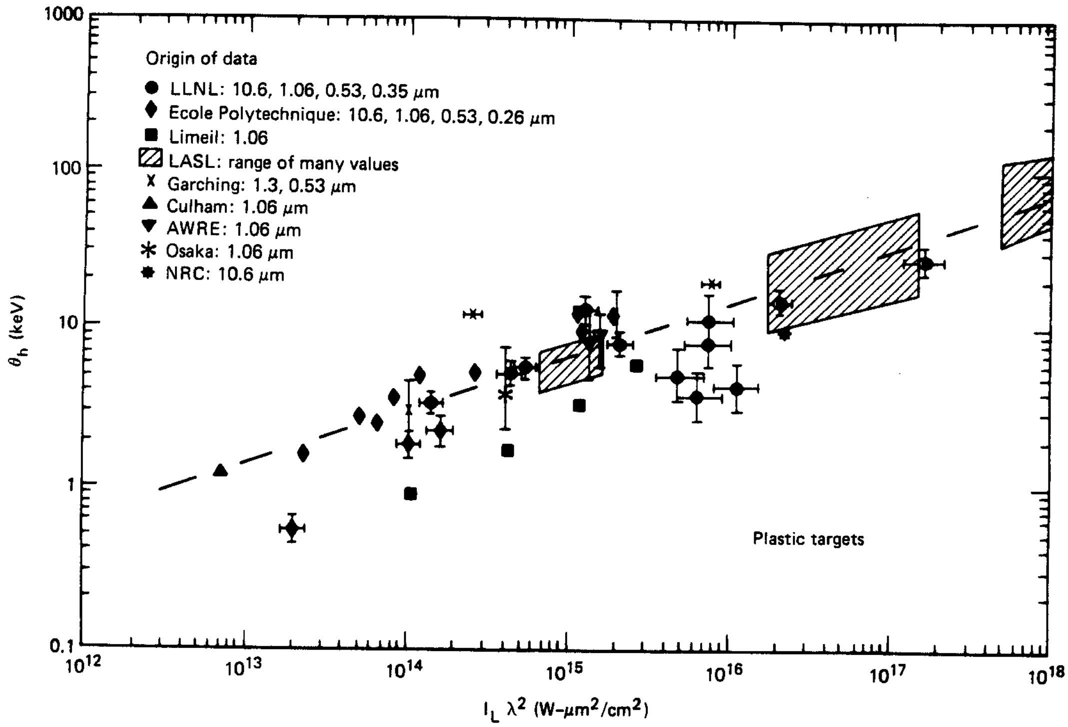


Fig. 3.2-3. The suprathermal electron temperature is a function of the laser intensity times the wavelength squared.

In the last five years, one of the most serious problems in laser plasma interactions has been wide discrepancies between experimental and computer simulation of the transport of thermal electrons. As explained in Section 3.2.2, the simulation codes only agree with experiment if the conductivity is arbitrarily reduced to 3-5% of the classically predicted value. Possible physical explanations of this "flux inhibition" include large self-generated d.c. magnetic fields, ion-acoustic turbulence, Weibel instabilities and improvements needed in the "classical" expressions for thermal and hot electron heat fluxes. Magnetic fields can inhibit the flow of electrons at energies much greater than 1 keV when the density is below a few times the flow of the critical density (where plasma frequency = ω_L) while thermal electrons with energies $\simeq$ 1 keV may be inhibited at electron densities below 10^{21} cm^{-3}. This means that the large scale magnetic fields should be able to cause very non-isotropic electron flow in the target. This effect has been recently studied in a series of experiments performed at Los Alamos National Laboratory. Ion-acoustic turbulence could be generated by the electron heat flow which would lead to strong scattering of the electrons off of the clumps of ion charge of the turbulence. This will most probably occur when the electron density is less than 10^{21} cm^{-3} (outside of the critical surface). The understanding of this phenomena requires more theoretical work. Weibel instabilities, which can also be generated by the flow of electrons, create small scale magnetic field perturbations which can inhibit the flow of electrons. This theory also needs work before it can definitely be taken as a mechanism of slowing electron transport. Finally, corrections to the "classical" transport expression due to the non-Maxwellian shape of the total electron distribution function and other kinetic effects may substantially reduce the electron transport. In summary, all of the mechanisms mentioned here may contribute to reducing the calculated electron fluxes to agree with the measured values but more theoretical work is needed to verify the most important parameters.

There are a few mechanisms possible for enhanced scattering and absorption of the laser light in the underdense plasma. These processes have been considered for a number of years, but now that target designs tend towards longer pulse width laser irradiation, they have become even more important to the overall efficiency of coupling lasers to targets. The mechanisms most regularly considered are stimulated Brillouin scattering, stimulated Raman scattering and two-plasmon decay. The linear theories of these processes are well understood but the modeling of nonlinear effects needs more development. Stimulated Brillouin and stimulated Raman scattering are the result of coupling between the incident laser wave, plasma waves in the plasma outside the critical surface and a reflected light wave while the two-plasmon decay instability involves the transformation of the incident laser light wave into a plasma wave in the presence of another plasma wave outside the 1/4 critical surface (where the plasma frequency is 1/2 of the laser frequency). These three processes are all retarded if the bandwidth of the laser is increased or if the density gradients in the region outside the critical surface are increased. Because of this last effect, increasing the laser pulse width while reducing the intensity, which causes more gradual changes in the density profile, should make these processes more important.

Another effect which has received considerable attention is the self-focusing or filamentation of laser beams by the plasma before they reach the critical surface. The basic idea is that the high radiation pressure in the center of the laser beam can rarify the plasma around the target in the center of the beam path. This in turn causes an increase in the index of refraction and a focusing of the laser beam. This process could cause large inhomogeneities in the irradiation of the critical surface which could cause an unsymmetric implosion of the target.

All of the physical issues described above combine to produce an overall absorption efficiency. These have been measured at different laser frequencies and are plotted in Fig. 3.2-4 against laser intensity for a number of laser wavelengths and pulse widths. It can be easily seen that longer pulse widths and shorter wavelengths lead to higher absorption efficiencies. It has also been found that the production of hot electrons is reduced at short wavelengths, the temperature of the electrons is lowered and the range is shortened, so that the problem of preheating the target before implosion is improved for short wavelength lasers.

The absorption efficiency leads to the mass ablation rate from the target. This has also been measured as a function of wavelength and is shown in Fig. 3.2-5 plotted against laser intensity for a few laser wavelengths and for both

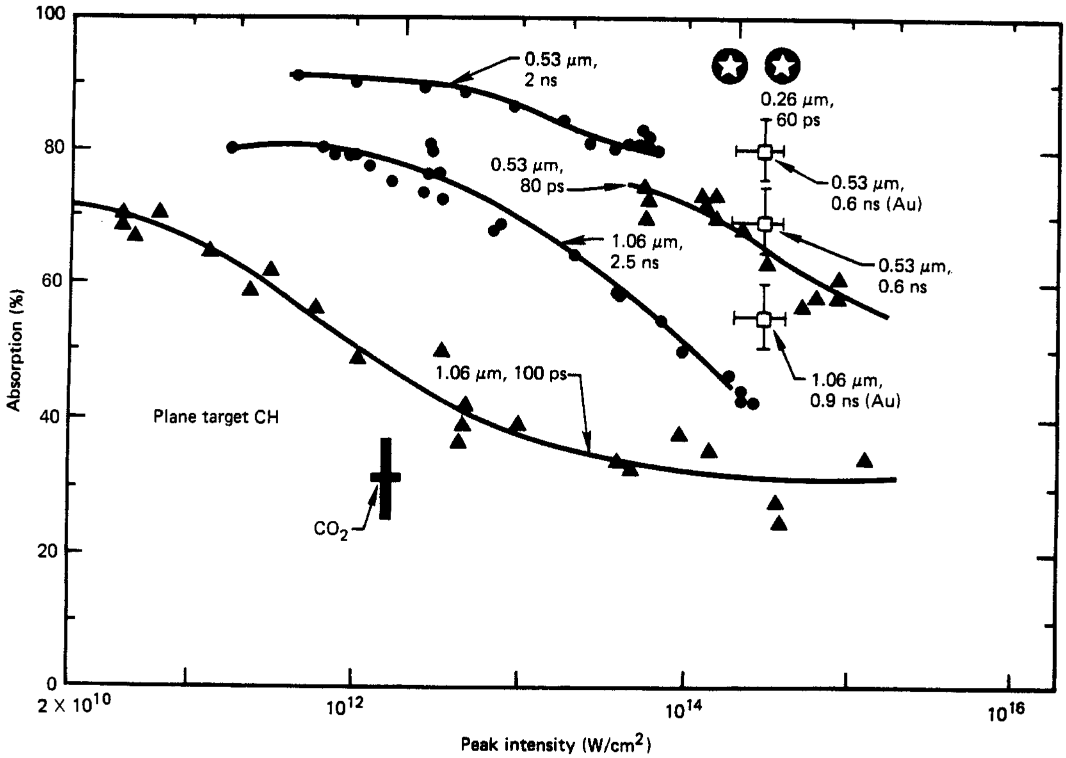


Fig. 3.2-4. Short wavelengths, long pulse lengths, and low intensities yield high absorption efficiency.

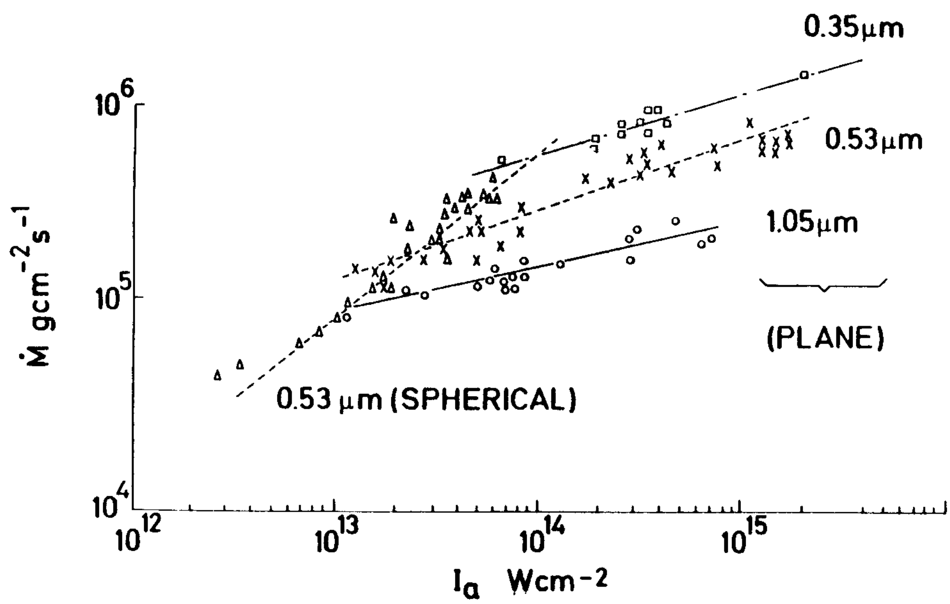


Fig. 3.2-5. Mass ablation rate vs. laser intensity for spherical and planar targets and various laser wavelengths.

planar and spherical targets. It is clearly seen that the mass ablation rate increases at shorter wavelengths. The ablation pressure which drives the implosion is proportional to the mass ablation rate so that short wavelength lasers will drive target implosions more strongly than at longer wavelengths for the same intensity. This is shown in Fig. 3.2-6.

It is for the reasons of increased absorption and reduced target preheat by hot electrons that lasers with wavelengths of 0.53 μm and 0.35 μm are preferred over 10.6 μm wavelength CO_2 lasers. Additionally, if one wishes to create x-rays for indirectly irradiated targets, the conversion of laser photons

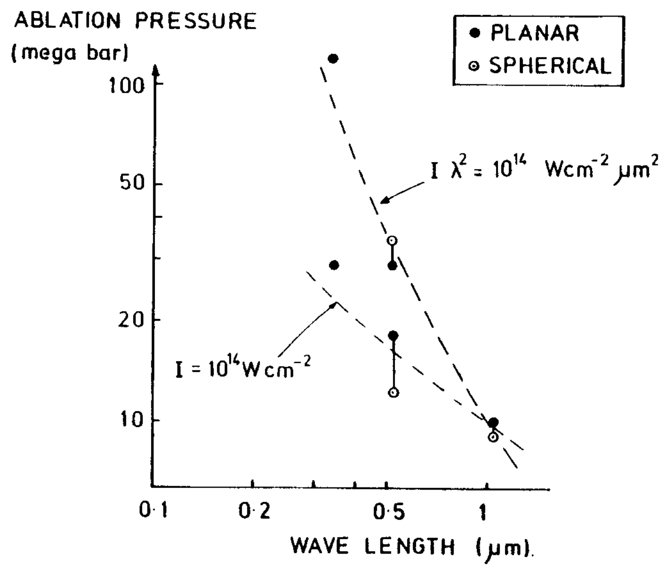


Fig. 3.2-6. Ablation pressure vs. laser wavelength for constant intensity and constant intensity times wavelength squared.

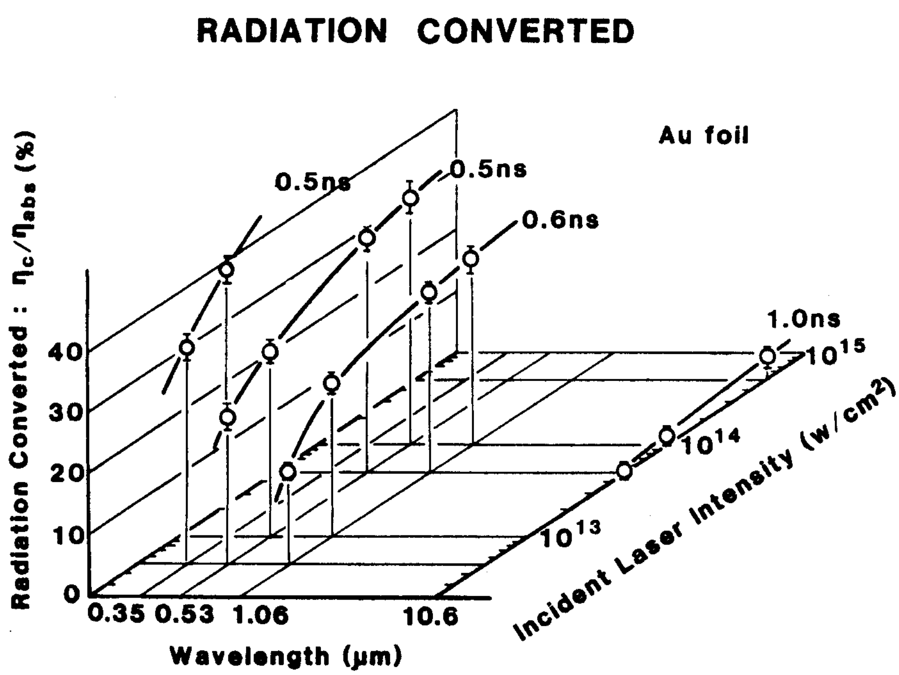


Fig. 3.2-7. X-ray conversion efficiency vs. laser wavelength and intensity.

to x-rays is also more efficient at short wavelengths (see Fig. 3.2-7). The disadvantage of short wavelengths is that they may require better uniformity of target illumination by the laser unless indirectly driven targets are used (see 3.2.2.2).

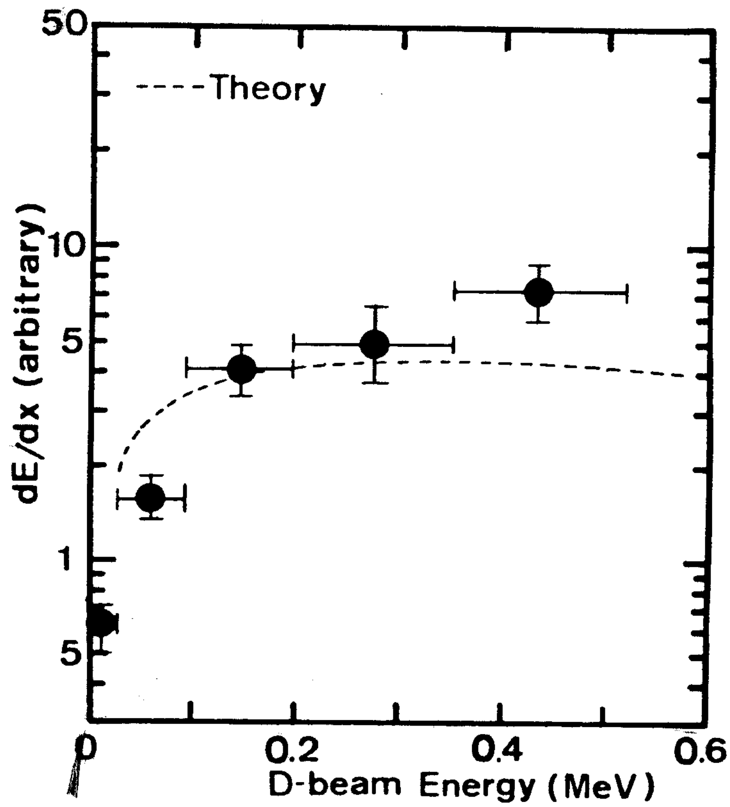


Fig. 3.2-8. Deuterium energy deposition vs. energy.

3.2.1.2 Light Ion Beam-Target Interactions. The issue of light ion beam-target interactions is not a primary question as it is in the case of laser driven ICF. Experiments done at Sandia National Laboratory at current densities of 500 kA/cm^2 and at the Naval Research Laboratory at 50-250 kA/cm^2 indicate that at high intensity the stopping of light ions in the megavolt energy range follows a classical expression (3). Experiments at Osaka University in Japan (4) have reached the same conclusion and a plot of energy deposition against ion energy for deuterium is shown in Fig. 3.2-8 along with the theoretical curve. Estimates of the ablation pressure generated in foil targets by deuterium ion beams have been obtained from experiments at Osaka and it has been found to follow the scaling law $P_a = 3 \times 10^{-3} I^{0.7}$ bar (I in W/cm^2). This indicates that the intensity needed to generate the 2×10^7 bar required for fusion is $\sim 10^{14}$ W/cm^2. This implies a current density of more than an order of magnitude higher than those used in any of the previously mentioned energy deposition studies. The question of anomalous effects due to the passing of this high current density beam through the outer reaches of the target plasma has been studied theoretically and no harmful effects are expected at these higher current densities, though the question will not be closed until experiments can be done at high intensities.

3.2.1.3 Heavy Ion Beam-Target Interaction. The interaction of heavy ion beams with targets has received some theoretical attention but very little experimental work has been done. Classical energy deposition of heavy ions in targets is generally assumed. The generation of free electrons through ionization of beam ions or target material atoms is a possible source of an electron beam which, if intense enough could generate magnetic fields affecting the beam or thermal transport in the target. These electron beams could also lead to the onset of plasma instabilities, causing anomalous ion beam propagation and stopping and anomalous heat transport in the target. They might also prematurely heat the target fuel, degrading the performance of the target.

3.2.2 Target Implosions

In the last five years we have seen the realization that target gain is unlikely to be nearly as

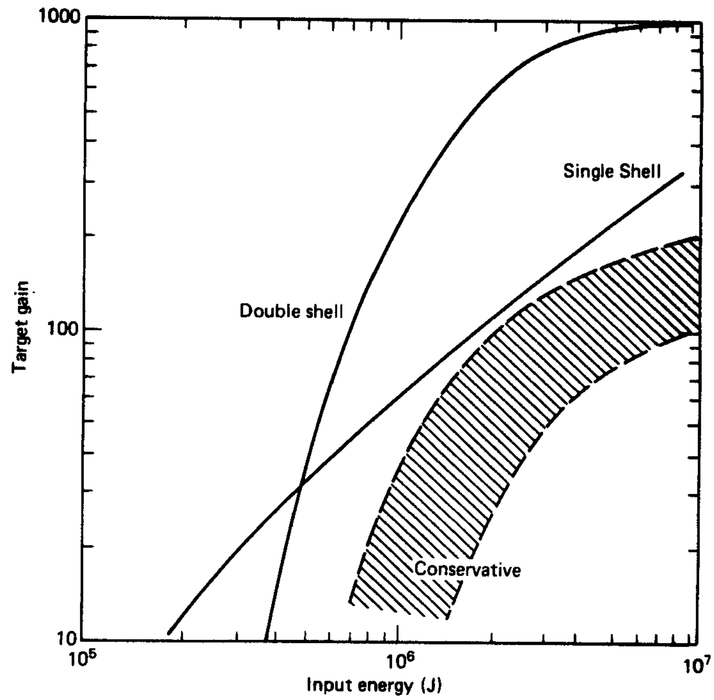


Fig. 3.2-9. Target gain predictions.

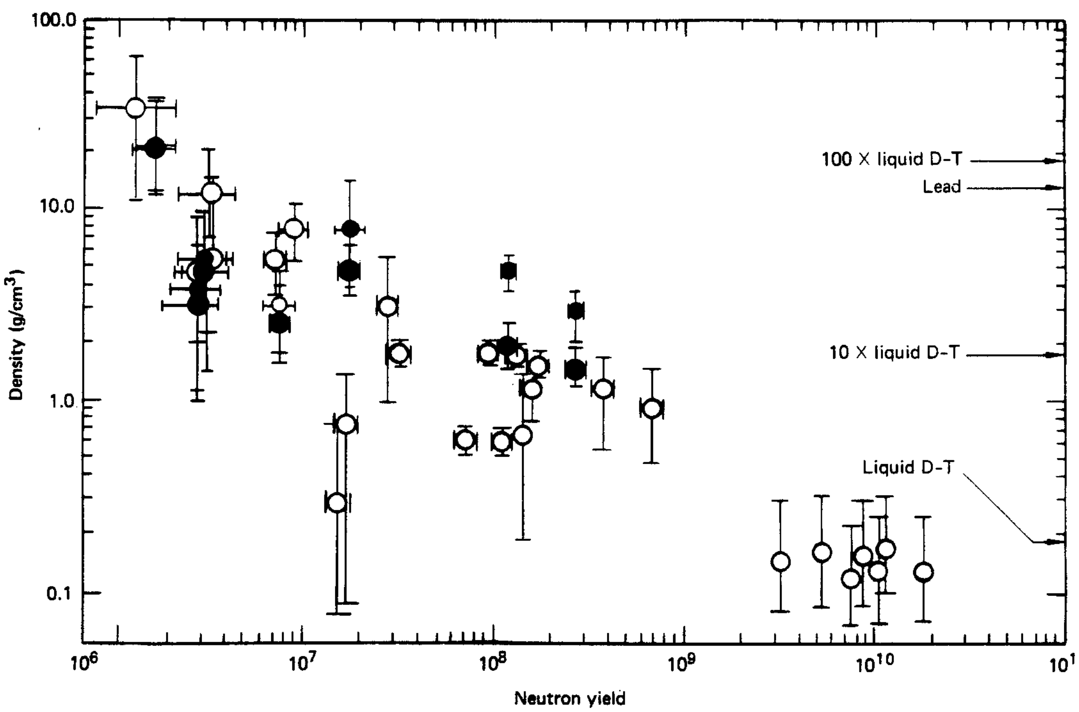


Fig. 3.2-10. High fuel density and fuel temperatures.

large as the theoretical maximum. This is clearly demonstrated by Fig. 3.2-9. These curves were first obtained as a result of many parametric computer calculations using the LASNEX target design code. Later Bodner (5), in an elegant paper, was able to distill the dependence of target gain into the most critical parameters associated with the implosion: laser energy absorption efficiency, hydrodynamic implosion efficiency, implosion symmetry, ignition temperature, and fuel isentrope. He was able to reproduce these curves using his analytic model and put the target gain question at a new level of understanding.

The size of lasers over the past five years did not allow the simultaneous achievement of high fuel density and temperature. However, each of these was investigated separately (6) and the results are displayed in Fig. 3.2-10. High temperature experiments were done first and were concluded at the tail-end of the so-called "neutron derby" that took place between 1974 and 1978. These experiments consisted of focusing very intense (> 10^{15} W/cm^2), short (~ 100 ps) laser pulses onto glass microballoon targets filled with 10-100 atmospheres of DT gas. A very strong shock wave non-isentropically heated the fuel to temperatures in excess of 4 keV and produced 10^8-10^{11} neutrons. These experiments gave a good indication that spherical implosion velocities greater than 3 x 10^7 cm/s could be achieved. They also demonstrated a weak dependence of neutron yield on laser wavelength. However, these results were not directly applicable to the ideal implosion scenario described earlier. These targets were operating in the "exploding pusher mode" (7). This is shown schematically in Fig. 3.2-11. The intense laser light creates copious amounts of very high energy electrons that penetrate the glass shell, uniform-

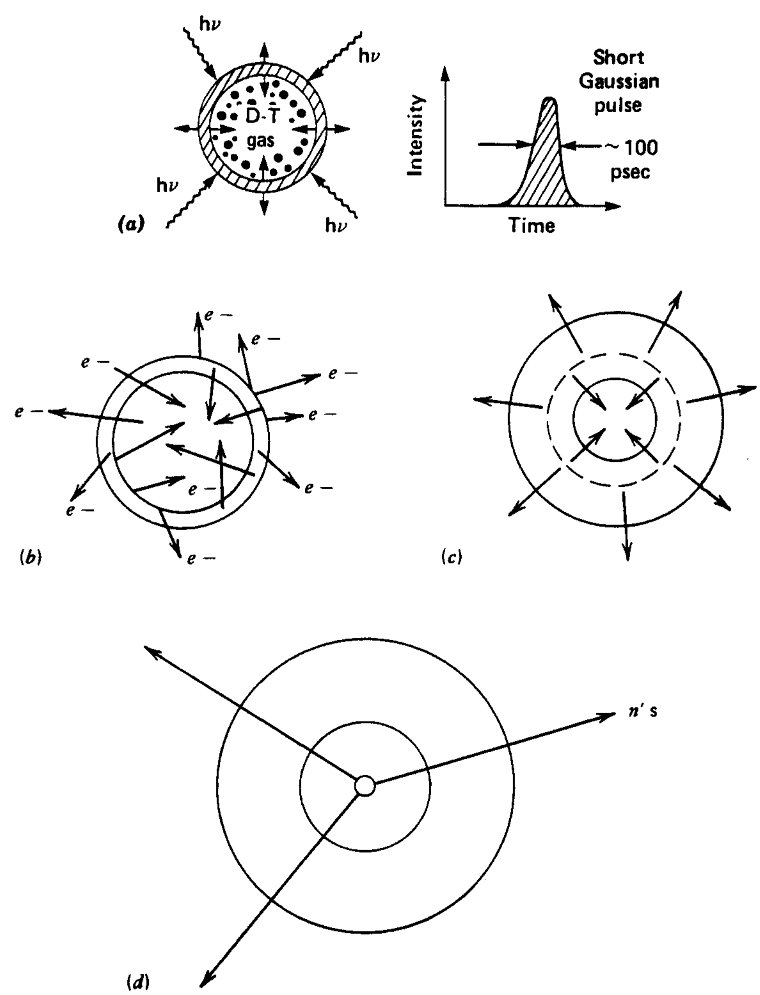


Fig. 3.2-11. The exploding pusher mode of target implosion.

ly heating it in a time that is much shorter than the implosion time. The shell subsequently expands (explodes), half inward and half outward. The inward half actually decompresses from its initial solid density driving a shock wave into the DT. The DT fuel is volume compressed by a factor of ~ 100 and is heated to a high temperature. By the end of this "exploding pusher" campaign, this type of implosion dynamics was very well understood. The complex hydro codes such as LASNEX were able to predict this behavior reliably and simple analytical theories were able to fit the data to a high degree of accuracy as shown in Fig. 3.2-12.

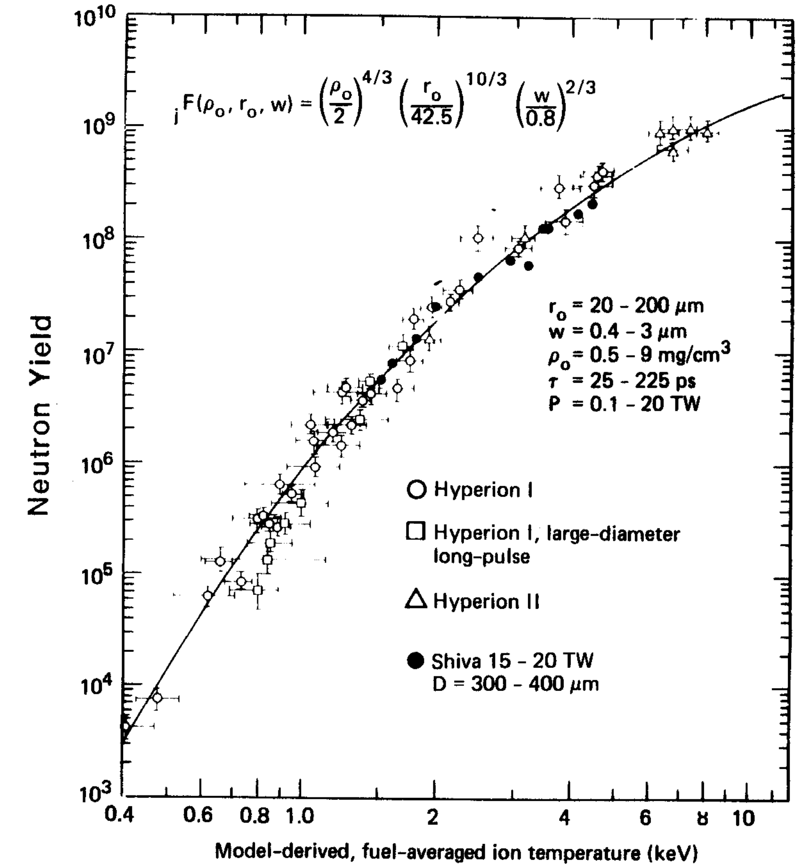


Fig. 3.2-12. Exploding pusher target scaling.

These high intensity (> 10^{15} W/cm^2) experiments led to a greater understanding of the resonance absorption process and the generation of hot or suprathermal electrons (see Section 3.2.1). They proved that DT fuel could be raised to high temperatures using laser drivers. Such experiments also exercised the calculational, diagnostic and target fabrication techniques crucial to the development of ICF.

To bring the fuel to higher density the laser intensity was lowered to reduce the temperature of the hot electrons. The glass microballoon wall thickness was increased or the balloon was coated with a thick plastic layer to partially attenuate the hot electrons thus reducing the fuel preheat. More DT fuel was also put into the glass shell. Relatively short pulses were still used and the shells were impulsively imploded, although this was still not a true ablation process. This type of experiment was used to implode the DT fuel to a density of nearly 100 x liquid density (6). This was a major achievement in the path toward high gain.

High gain target designs always use a spherical shell of DT fuel frozen on the inner surface of a pusher layer. Experiments were done to compare the neutron yields and final fuel densities of glass microballoons filled with gaseous DT fuel and with DT fuel frozen on the inner surface of the shell (8). The density results are recorded in Fig. 3.2-13 for 1.06 μm laser light. Solid

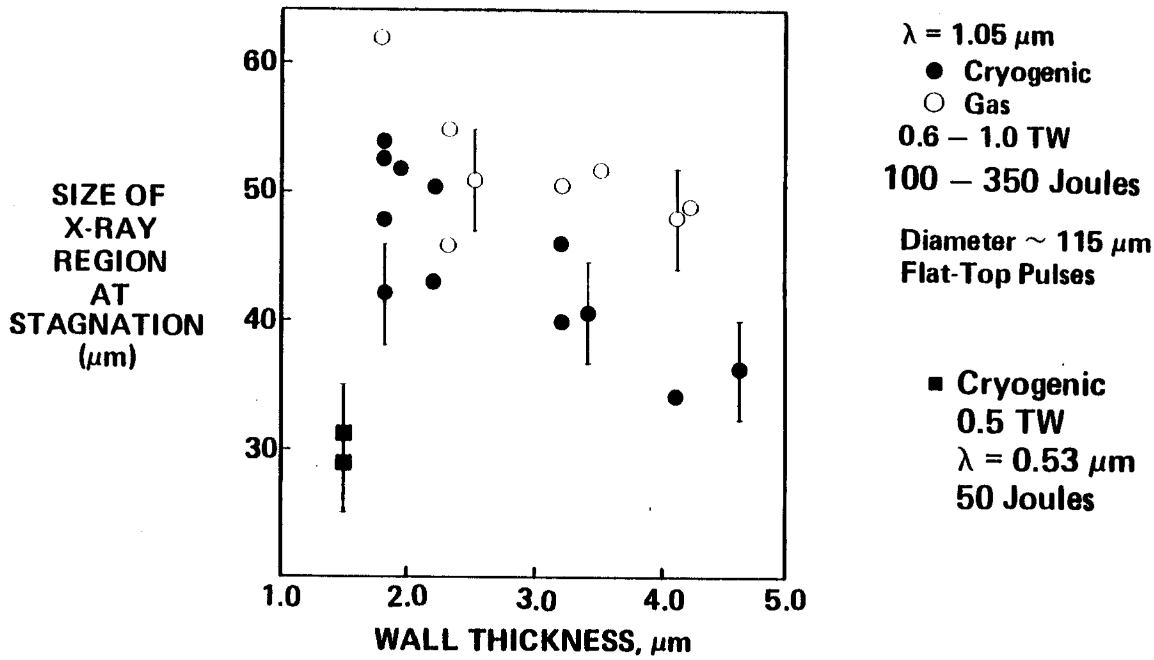


Fig. 3.2-13. Fuel density for gaseous and cryogenic fuel.

fuel results in marginally higher fuel density. Also shown in Fig. 3.2-13 are results using 0.53 μm light where there are fewer hot electrons. In this case the fuel density is considerably higher. This tends to indicate that these high density experiments continue to be dominated by suprathermal electrons and are not a true test of the ablative implosion required for high gain ICF.

Experiments to test the theories of ablation driven implosions have been performed on slab targets (9). These were low intensity (< 10^{15} W/cm^2) long pulse experiments as shown in Fig. 3.2-14. The low intensity insures that light absorption is via inverse bremsstrahlung rather than resonance absorption, hence there are few hot electrons. The mass ablation rate was measured along with the hydrodynamic efficiency of the acceleration process. These are shown in Fig. 3.2-14 along with the results of the simple "rocket model" of undergraduate mechanics. It shows that the model fits the data quite well. This is an important achievement. It shows that lasers can indeed be used to ablatively accelerate solid density plasmas up to high velocity. Furthermore, when the laser-plasma coupling processes deposit the laser energy into a thermal plasma, the hydrodynamic response can be modeled accurately by a simple, well-understood model.

Thus far, we have discussed the critical issues of laser energy absorption efficiency, hot electron generation, associated wavelength dependence, implosion velocity, high fuel density and ablative acceleration scaling. The one remaining critical issue is implosion symmetry. The implosion symmetry can be destroyed by two general mechanisms: (1) rapid growth of fluid instabilities

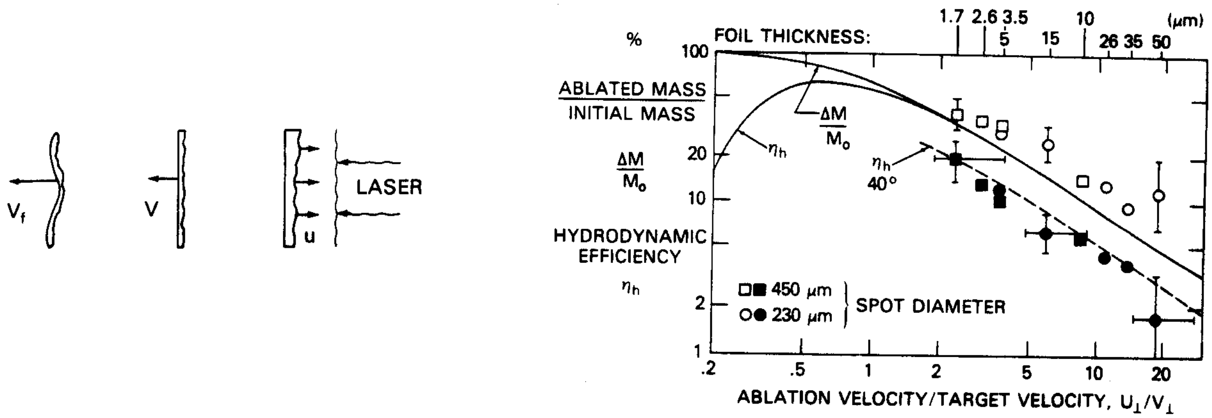


Fig. 3.2-14. Ablative acceleration of slab targets.

that amplify small imperfections in the target shells during perfectly spherical implosions, and (2) nonuniform driving pressure due to nonuniform energy deposition from the laser or ion beams. The first of these problems must be overcome by sufficient tolerances on the surface finish and density variation of the target shells during the fabrication process. The second problem has several alternative potential solutions.

3.2.2.1 Direct Drive. Everything that has been discussed in this report is applicable to so-called direct drive ICF. By this we mean that the laser or ion beams directly irradiate the target and the implosion is "driven" by the electron thermal conduction mechanism. In this case, the solution to the symmetrical implosion problem is to very carefully design the laser beams so that when they overlap on the target, they form a uniform irradiance pattern. Any small perturbations (~ 1-5%) in the irradiance can possibly be "washed out" in the heat diffusion region between the critical density and the ablation surface as shown in Fig. 3.2-15. Thus, irradiance uniformity requirements become more stringent for shorter wavelengths because the smoothing region thickness is reduced for the higher critical densities associated with short wavelength lasers. Ion beams have the most severe requirements since there is no critical density for them and the energy deposition region and ablation surface coincide. Hence, short wavelength lasers and ion beams provide the highest absorption efficiency and hydrodynamic implosion efficiency but also have the highest probability of driving non-symmetric implosions. Using the types of target considerations discussed in this report, the only remedy for this is in the development of very well-characterized driver beams. This is the approach to ICF that is being pursued at the University of Rochester and is supported by work at the Naval Research Laboratory.

3.2.2.2 Radiation Drive. Another alternative is the use of a so-called radiation driven target. In this target design the driver beam energy is absorbed and converted to thermal x-rays. These thermal x-rays are trapped in an enclosure called a hohlraum and are used to drive the implosion of a separate

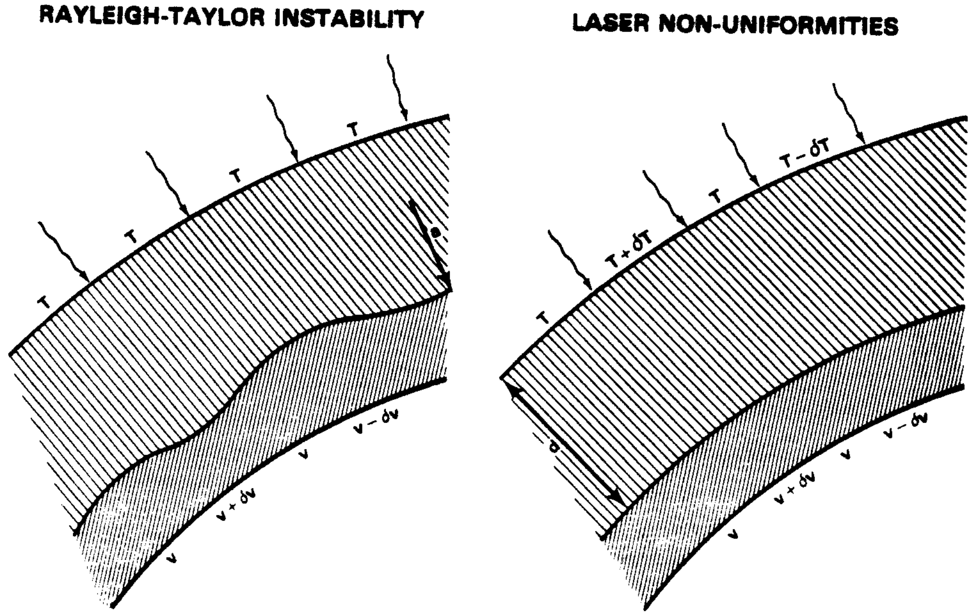


Fig. 3.2-15. Smoothing of nonuniform irradiance in the design between the critical density surface and the ablation surface.

capsule containing the DT fuel. This approach offers the promise of more symmetric implosions (10). Target designs utilizing this principle of x-ray conversion are classified by the U.S. government and no details about them can be included in this report. However, the concept itself was declassified during the past five years. This is the approach to high gain target design that is being pursued at LLNL for short wavelength lasers and at SNL for light ion beams. It is also being pursued at LANL for long wavelength lasers but for different reasons.

The use of x-rays to drive the target implosion opens up a whole new area of ICF. Any suitable x-ray source, with the proper characteristics, could potentially be utilized to implode targets. Such a new scheme has been proposed by SNL in the form of imploding foils (11). Rather than attaching a diode to their pulsed power machines they have proposed attaching a hollow cylindrical can. When the machine is pulsed, the current driven through this can creates magnetic fields that implode the can. When the can collapses it converts its kinetic energy into thermal energy and radiates x-rays. These x-rays can then possibly be used to implode a target. Again, the details of this approach cannot be discussed in this report. However, it is quite possible that PBFA-II will be configured to operate in this way rather than accelerate ions. This imploding foil approach does not readily extend to reactor applications. Hence, all three of the major ICF laboratories in the U.S. are pursuing an approach to ICF target implosions that is classified. Consequently, much (probably over 50%) of the experimental and theoretical work over the past

LASNEX PHYSICS

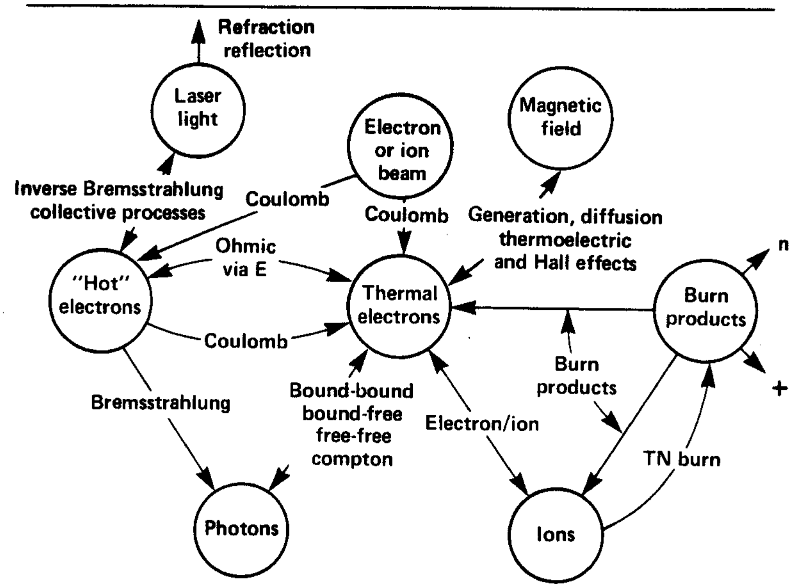


Fig. 3.2-16. Block diagram of LASNEX computer code.

five years will not be reviewed in this report. No complete picture of the status of ICF can be drawn without this information. For this report we have adopted the philosophy that we will simply review the available information and call out the fact that it is an incomplete story.

3.2.2.3 Code Development. The principal target design tool for ICF is a radiation-magnetohydrodynamics computer code called LASNEX (12). A schematic diagram of the interrelated physics modules in the code is shown in Fig. 3.2-16. This code is under constant improvement and change by groups at LLNL, LANL and SNL. Roughly 20 man-years per year of effort goes into maintaining this 100,000 line program at the 3 laboratories. Improvements over the last five years are difficult to completely document, however, they include: an improved treatment of mixed Eulerian-Lagrangian hydrodynamics, fully two-dimensional solution of the various diffusion equations used to transport electrons and photons, ray tracing options to follow laser and ion beams, and improved atomic physics modeling of the state of the plasma for the purpose of computing its thermodynamic and radiative properties. At LLNL the code has been rewritten for implementation on the Cray computer. This new version adheres to structured programming methods and uses such state-of-the-art concepts as a dynamic memory management system for handling variables. The user interface for the LASNEX code is one of its strongest features. It is easy to use, fully interactive in the timesharing mode, and provides a rich variety of graphical output options. Over the past five years the LASNEX code has become the primary tool for ICF target design activities.

3.2.2.4 Diagnostics. An area that has seen tremendous progress is the diagnostics associated with the ICF experiment (6). Many remarkable instruments such as the x-ray streak camera with 15 ps resolution, grazing incidence x-ray microscopes with 1 μm resolution and micro-Fresnel zone plates with 1 μm resolution have been developed over the past five years. The extremes of spatial (~ 1 μm) and temporal (~ 10 ps) resolution were necessitated by the small targets used for implosion studies. Computer assisted alignment of multi-beam laser systems was a prerequisite to their very development for the ICF program. Computer controlled data collection from hundreds of separate channels is necessitated by the infrequent and expensive target shot rate. Only 4-8 targets per day can be shot, in part due to the high cost of target fabrication. Specific details of the diagnostic development are too involved for inclusion in this review, but excellent reviews of this subject are available.

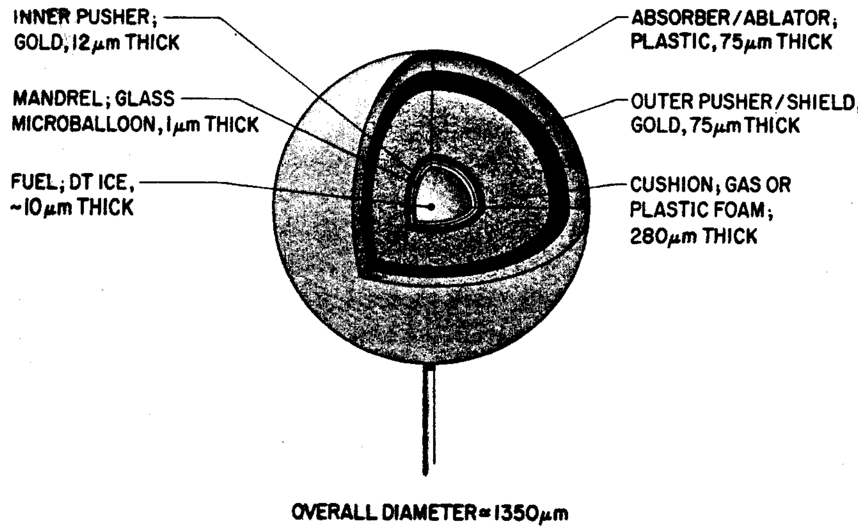


Fig. 3.2-17. A multicoated, multishell laser fusion target.

3.2.2.5 Target Manufacturing. The success of ICF power reactors will depend on the development of manufacturing processes to produce reactor quality targets. Fusion reactor targets are fairly complex. They may be either single-shelled, consisting of a hollow cryogenic DT core coated with multiple layers, or double-shelled targets in which there is a void between shells. The production of ICF targets and the analysis of the target factory concept have been extensively studied by Charles Hendricks and co-workers (13).

Several approaches have been proposed to produce ICF targets (see Fig. 3.2-17). Presently, solid and hollow hydrogen spheres and glass shells can be produced at high rates and low costs. Coatings have been applied to glass microspheres using magnetron sputtering, electroplating and vapor deposition techniques. Laser microdrilling and micromachining is also being investigated. A method for forming a uniform DT condensate layer inside of a glass microballoon has been demonstrated. Research laboratories are being constructed at Lawrence Livermore National Laboratory and Los Alamos National Laboratory to study and develop the technology of target production.

The manufacture of reactor-grade targets at the rates necessary for ICF power production will require many further technological developments. The production rates and allowable costs per target depend upon the achievable target yield. Typically, production rates of 1-20 s^{-1} must be achieved at an allowable cost of about ten cents per target. Target fabrication must be completely automated to achieve these goals. The targets must also undergo careful quality control to insure that the driver energy is not used on defective targets. Some parameter tolerances for reactor-grade targets are: diameter ±1 μm, wall thickness ±0.05 μm, sphericity < 1% of the diameter, fuel fill ±10% and surface finish with defect heights < 100 Å. To attain this level of quality control, computerized analysis tools must be developed to monitor the target at specified points in the production process. Rejected targets which contain valuable DT fuel must be recycled. The surface finish requirement makes levitated transport of targets necessary. Electrostatic, electrodynamic, gas jet, acoustic, magnetic and focused laser beam levitation schemes have been studied.

The target factory should be designed to minimize the tritium inventory. The tritium inventory is dependent on the process time for a target, the point at which the DT fill step occurs and the efficiency of production steps handling fuel-filled targets. Estimates of the tritium inventory in a typical target facility are on the order of tens of kilograms.

3.2.3 Driver Development Physics

In the past few years there have been significant gains in the physics understanding and the technology of ICF drivers. Great improvements have come in the design and operation of Nd:glass lasers, which are now yielding many of the physics results described in Sections 3.2.1 and 3.2.2. In addition, development is proceeding in the more reactor relevant particle beam drivers, especially light ions. Heavy ion beam driver development has suffered from a lack of funding and no major experimental devices have been built. There is, however, significant interest in heavy ion drivers in the ICF community and many new ideas for reactor drivers have been proposed. The development of all of these drivers is the topic of this section.

3.2.3.1 Nd:Glass Lasers. The solid state Nd:glass lasers are currently heavily used in experiments studying laser plasma interactions and implosions of both directly and indirectly driven targets. These lasers operate at a 1.06 μm wavelength, but in the past five years it has been demonstrated that the 1.06 μm light can be efficiently converted to 0.53 μm and 0.35 μm light. There are many Nd:glass laser systems presently operating in the world which are being used for studying laser plasma interactions and they have been used with this frequency conversion to study the wavelength scaling mentioned in Section 3.2.1.1.

Though the technology of Nd:glass lasers has been greatly improved in recent years, there are some very important problems which make them seem less attractive for ICF reactors than the so-called advanced lasers, e.g. rare-gas-halide and free electron lasers. The first is that their efficiency is very low so that for an economically viable fusion power plant the target gain would have to be very high. The second is that since the amplifying medium is a solid, heat transfer problems limit the repetition rate to much lower than what would be required for a reactor. There are currently extensive research programs concentrating on basic physics and technology issues which may eventually give some help in these areas. These drawbacks with Nd:glass do not preclude using this type of laser to develop target designs and demonstrate that targets can be imploded to ignition by lasers.

Research in solid state Nd:glass lasers is being conducted and large lasers exist or are under construction at Lawrence Livermore National Laboratory (14), KMS Fusion, The University of Rochester, Osaka University in Japan (15), and several other labs around the world. The largest laser constructed and operated to date is the 15 kJ Shiva laser system at Livermore. The 120 kJ Nova system should be constructed in the mid-1980's.

3.2.3.2 CO_2 Lasers. CO_2 lasers have been built for use in laser-plasma interaction experiments and in laser driven target implosion experiments in the U.S., Canada, Japan, and elsewhere. Los Alamos National Laboratory has made the greatest advances in the technology of large CO_2 lasers (16). These advances have occurred in the building and operating of the Helios laser, in the building of the Antares CO_2 laser which is still in progress, and in the advanced laser program at Los Alamos.

Technological advances have been in the optics, power amplifiers, energy storage and control systems of large CO_2 lasers through the construction of Antares. The optical system for Antares will contain elements with new LiF low reflectance coatings and computer controlled alignment and calorimetric calibration of the laser power will be used. Among the large optical components used in Antares will be anti-reflecting salt windows and copper mirrors made with new electroplating or painting techniques. Front-end development has continued to produce better pulse shaping and control of line content. The power amplifiers in Antares are driven with electron beams where the control of breakdown in the electron gun has seen advancement.

In the operation of the Helios laser much has been learned about the nonlinear optical effects and the control of parasitic oscillations in CO_2 lasers. The use of a saturable absorber gas has been found to be a useful method of controlling parasitic oscillations and much has been learned about this technique in recent years. Pulse shape distortion and laser beam self-focusing are potential problems in any high power laser which have been studied on Helios.

The Advanced Lasers Program at Los Alamos has studied general issues of technology and physics which are important to large CO_2 lasers. Saturable absorbers as a means of controlling parasitic oscillations, energy transfer kinetics and discharge and kinetics modeling in electron beam driven CO_2

amplifiers are all topics which have seen significant research in recent years. Additionally, plasma shutters have been investigated as a method of isolation of the laser from the light reflected off of the target, a potentially dangerous problem for these lasers. Damage limits from the highly intense repetitive illumination to the copper mirrors have been investigated.

3.2.3.3 Rare-Gas-Halide Lasers. Several of the rare-gas-halide (RGH) lasers are efficient and possibly scalable to large energies. The one most commonly considered is KrF, with a wavelength of 0.248 μm. For reasons of laser target coupling (see Section 3.2.1.1) this looks like an advantageous choice of wavelength. These lasers do not store much energy thus they would be more useful as amplifiers for an incident laser. In addition, the pulse length for efficient use of the amplifier should be greater than 400 ns, much longer than the pulse widths required for laser fusion. Thus, mechanisms of compressing the pulse must be investigated. The detailed kinetics of lasing RGH mediums is also an important area of research. Experimental and theoretical work is continuing with the hope that an efficient, short pulse RGH laser with a pulse energy in the MJ range will become a real choice as an ICF driver. However, at this point in time, RGH lasers are still less developed than the mainline Nd:glass and CO_2 lasers but this may change as large KrF lasers are built because of their high efficiency and favorable wavelength.

In the United States, Lawrence Livermore National Laboratory is the lead lab for advanced lasers (17), which includes RGH and free electron lasers (see Section 3.2.3) though Sandia National Laboratory and Los Alamos National Laboratory are also participating in this research as are labs in Japan (18) and other countries. In the past five years there have been advancements made in the design and understanding of pulse compressor systems. The two main candidates of pulse compressors are backward wave Raman scattering and pulse stacking. These ideas have been investigated experimentally, for example, at Lawrence Livermore with the RAPIER KrF laser system, which is designed to provide 25 J at 50 ns pulse widths or 500 J at 150 ns. Also at Livermore, there has been a conceptual study for the design of a 2 MJ KrF laser for ICF reactor scenarios. The Department of Energy Advanced Lasers Program plan of 1980 called for the demonstration of a 10 kJ RGH laser and the development of a 100 kJ amplifier for a RGH laser system by 1985. This plan has been dropped since then though LANL has an ongoing KrF development program.

3.2.3.4 Free Electron Lasers. Free electron lasers are not yet competitive with Nd:glass or CO_2 lasers as ICF drivers in the 0.3 μm to 10 μm wavelength range but they hold great promise for the future (19). They would be tunable to any wavelength and they are possibly very efficient compared to Nd:glass lasers. There is currently theoretical and experimental work in progress in the U.S. and in other countries on free electron lasers.

Free electron lasers convert the kinetic energy of injected electron beams into laser radiation. This is accomplished with the use of an electron beam, an input laser beam and a spatially periodic applied magnetic field, called a "wiggler." A schematic picture of a free electron laser is shown in Fig.

3.2-18. An electron beam incident from the left in the z-direction is given an oscillatory component of velocity in the y-direction by the $\vec{V} \times \vec{B}$ force from the magnetic field of the wiggler which is in the x-direction. The input laser also enters from the left in the z-direction with its electric field polarized in the y-direction, parallel to the induced velocity of the electron beam so the oscillations of the electron beam and the laser electric field are in phase and on the same wavelength so that energy can be transferred from the electron beam to the laser. Thus the electron beam and wiggler together act as a laser amplifier. To help achieve a high efficiency, the unused beam energy may be recovered using a decelerating field beam dump.

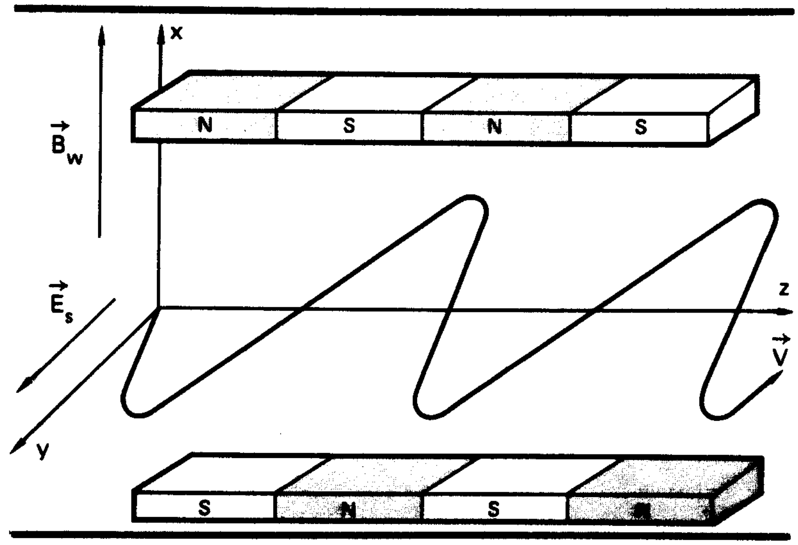


Fig. 3.2-18. Energy is transferred to the FEL by giving the electrons a velocity component parallel to the laser field.

The free electron laser concept is still undergoing mainly theoretical examinations though in the last five years small experiments have been built. It is possible that this may become a very important type of driver for laser fusion targets in the future because of the tunability and high efficiency. A preliminary conceptual design of such a driver was performed at Lawrence Livermore National Laboratory for an output energy of 1.5 MJ where 30% to 40% of the electron beam energy is transferred to the laser and the overall efficiency of the system including that of the induction linac or betatron used to generate the electron beam is between 14% and 18%.

3.2.3.5 Heavy Ion Fusion Drivers. Two types of driver seem suitable for heavy ion fusion, those based on an rf linac accelerator followed by storage ring accumulation and bunching, and those based on an induction linac. The rf linac-storage ring schemes use more conventional technology. A number of such scenarios have been designed an example of which is shown in Fig. 3.2-19 for the HIBALL system. Their performance can be calculated with relatively high confidence. The heavy ion induction linac seems to promise a simpler and less expensive driver but with greater technological risk. Electron induction linacs have been built and operated satisfactorily. However, heavy ion linacs involve new problems, and are at present only in the conceptual stage.

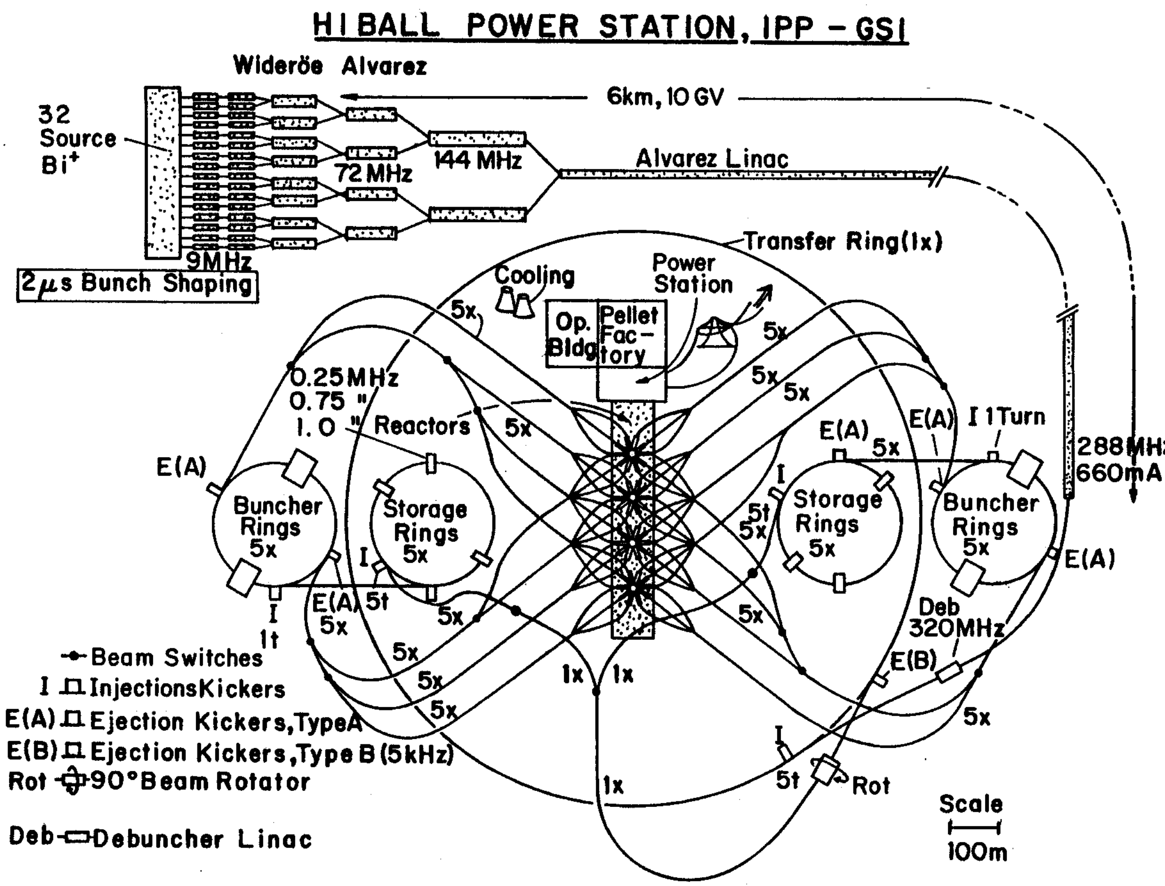


Fig. 3.2-19. HIF drivers.

The best, most comprehensive, and recent source of information on heavy ion fusion drivers is the Proceedings of the Symposium on Accelerator Aspects of Heavy Ion Fusion, Darmstadt, March 29, 1982. Probably the most complete power plant design study using the rf linac-storage ring approach is the HIBALL study, described in Section 3.3. One recent development in this type of driver is the application of RFQ structures (rf quadrupoles) to the early linac stages. These allow lower injection energies, because of their better focusing at low energies. They thus eliminate the need for a high voltage DC injector. By using adiabatic bunching in the early stages, they promise better capture efficiency and better beam quality.

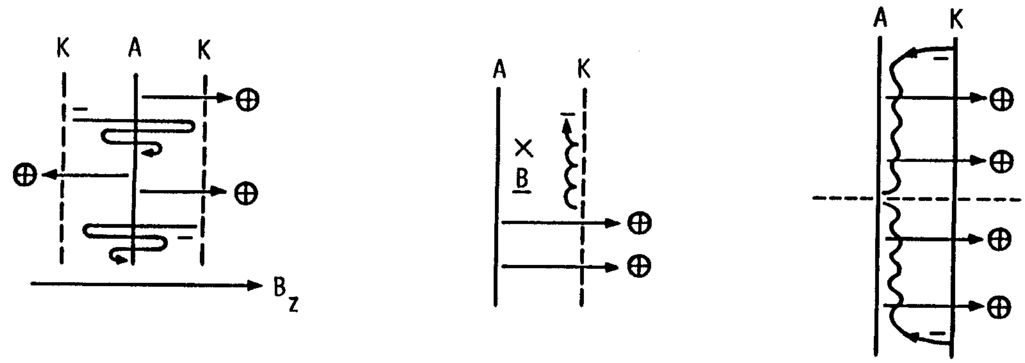
One of the most active groups in developing heavy ion induction linac drivers is the group at Lawrence Berkeley Laboratory. They have recently studied multiple beam induction linacs. These will allow the acceleration of separate beams of ions from source to target in the same linac structure. In this way, the increased space charge limits associated with multiple beams, both at the injector and at the target, can be achieved without the emittance dilution associated with combining and separating beams as required in other scenarios.

Several new magnetic and insulating materials have been developed in the past few years which can improve the price and performance of induction linacs. New magnetic materials, particularly Metglas (R) developed by Allied Chemical Corporation, have low electrical conductivities. Their use will reduce eddy current losses in induction linacs. Cheaper insulating materials suitable for use in linacs have also been developed.

Although no large heavy ion induction linacs have been built, design studies have been done on heavy ion induction linac drivers. During the past few years a number of people have carried out theoretical studies of the design and dynamics of such linacs, and particularly of the space charge phenomena that will occur in these machines. Experiments in this field have been severely restricted by budget limitations, particularly in the United States. LBL has constructed a model injector for a heavy ion induction linac, and initial tests are encouraging. LBL is also building a model induction module to follow the injector. Cost restrictions have required design compromises which make the module both more difficult to design and build and somewhat further removed from an ultimate driver design.

3.2.3.6 Diode Development. Probably the critical piece of hardware in a light ion beam driver system is the diode. To date there are three major types of diodes (20): reflex diodes, magnetically insulated diodes and pinched electron beam diodes. In the last five years work has progressed on these diode designs to the degree that the achievable ion power has increased from less than 10^8 W to more than 10^{12} W in that time span.

In ion diodes, the key to high power and efficiency is the effectiveness in stopping the flow of electrons from the cathode to the anode. If these electrons are allowed to flow freely as the ions flow from the anode to the cathode, a maximum of only 2.3% of the energy will go into ions. The main types of diodes use three different methods of impeding the flow of electrons from the cathode to the anode. The methods are shown schematically in Fig. 3.2-20. In reflex diodes, Fig. 3.2-20a, the electrons from a cathode pass through an anode, losing energy, and do not reach the cathode on the opposing side of the device. The electrons oscillate around the anode losing energy on each pass. The electron flow is inhibited because of this cloud of oscillating electrons around the anode, while the ions pass through the cathode and their flow is not inhibited. In magnetically insulated diodes, shown in Fig. 3.2-20b, an externally applied magnetic field is put in-between the cathode and anode and is perpendicular to the diode electric field. The electrons cannot traverse the anode-cathode gap while the magnetic field is not strong enough to affect the ion flow. A buildup of negative charge from the electrons can be avoided by allowing the electrons to leave the diode through an $\vec{E} \times \vec{B}$ drift shown in the figure. In pinched-electron diodes, Fig. 3.2-20c, the self-induced magnetic fields pinch the electron beam from the cathode together. This impedes the electron flow. This diode works best when the radius



(a) Reflexing-electron (b) Magnetically-insulated (c) Pinched electron beam

Fig. 3.2-20. Three methods of impeding electron flow in pulsed power diodes.

of the diode is much larger than the gap between the anode and the cathode. Various diodes, which are based on these three basic types, have been used in experiments at SNL, NRL, Cornell University and Osaka University and parameters for some of them are shown in Table 3.2-1. Here, diode voltage, diode current, current density, beam divergence, beam brightness, and power brightness are shown.

The beam power brightness is a measure of how unidirectional the energy flow is and is in units of power per unit area per unit solid angle. A large power brightness for ICF is needed to insure that a large enough power density reaches the target, if ballistic transport is assumed, or reaches the entrance of the channel if channel propagation is assumed. A typical minimum value needed for ICF is 100 TW/cm^2/steradian. It should be noted from Table 3.2-1 that the maximum power brightness achieved is more than an order of magnitude below that needed for ICF. In the last five years the achievable beam divergence has been reduced from 10° to 0.4° and the beam power brightness has increased by more than four orders of magnitude. Further increase in beam brightness is expected to come with an increase in diode voltage as shown in Fig. 3.2-21.

One way that the performance of diodes can be improved is by developing a high density low divergence source of ions at a specific ionization state. Many of today's diodes use a surface flash-over technique to generate ions but work is underway on laser vaporization and ionization techniques at Osaka and other places which could create a higher source density and a higher fraction of ions in a single ionization state.

To be usable in a reactor, a diode must be able to fire on the order of once per second. This is a problem that needs a great deal of work in the next one or two decades. None of the diodes used today are designed to be pulsed that rapidly. In fact many of today's diodes may only be used once before they

TABLE 3.2-1

Ion Beam Diode Performance

Machine	Diode	Ion	V (MV)	I_1 (kA)	J_1 (kA/cm^2)	Beam Divergence $\Delta\theta_{1/2}$ (°)	Beam Brightness $B = J_1(\pi\beta^2\gamma^2\Delta\theta^2)$ (MA/cm^2/sr)	Power Brightness $\beta = J_1V/(\Delta\theta)^2$ (TW/cm^2/sr)
OMNIPULSE	pinch reflex	p	0.13	6	0.02	10	0.75	0.000085
	mag. ins.	p	0.2	4	0.05	4	7.7	0.0021
	mag. ins.	p	0.15-0.2	5	0.07	3	19	0.0051
	field incl.	p	0.35		0.03-0.05	5.4-4	4.4	0.0036
NEPTUNE	cylindrical	p	$\leqslant$ 0.3	$\leqslant$ 100	0.09	3-5	16.3	0.0098
PROTO I	rad. appl. B	p	0.8-1.4	360	1	3.5	29	0.38
HYDRA	spherical	p	1	50	~ 1	2.1 ± 0.3	111	0.74
	appl. B_θ	p	0.27		0.1	$\leqslant$ 2	45	0.022
GAMBLE II	pinch reflex	d	1.3	700	5	~ 3(1)	221	2.6
	cage B_θ		0.35	8.2	0.22	< 2	77	0.063
NEPTUNE	planar mag.	p	0.8	50-65	0.2	$\leqslant$ 1	123	0.53
LONGSHOT		p	0.8-1.2		$\leqslant$ 0.1	2.5	6.5	0.063
CASTOR	mag. ins.		0.3-0.6			~ 1		
NEPTUNE	mag. ins.	p	0.6-0.8	70-110	~ 0.25	$\leqslant$ 0.4	957	4.1
GAMBLE II	pinch reflex		1.2	600	30	4.6	579	5.6
PROTO I	rad. appl. B	p	1.2	400	~ 5	3.5	167	1.6
HYDRAMITE	AMPFION		1.7	200	~ 0.3	1(2)	86	1.7

(1) angle subtended by target of radius 0.5 cm at a distance of 10 cm.
(2) local divergence only; actual focal spot spread due to non-uniform plasma fill.

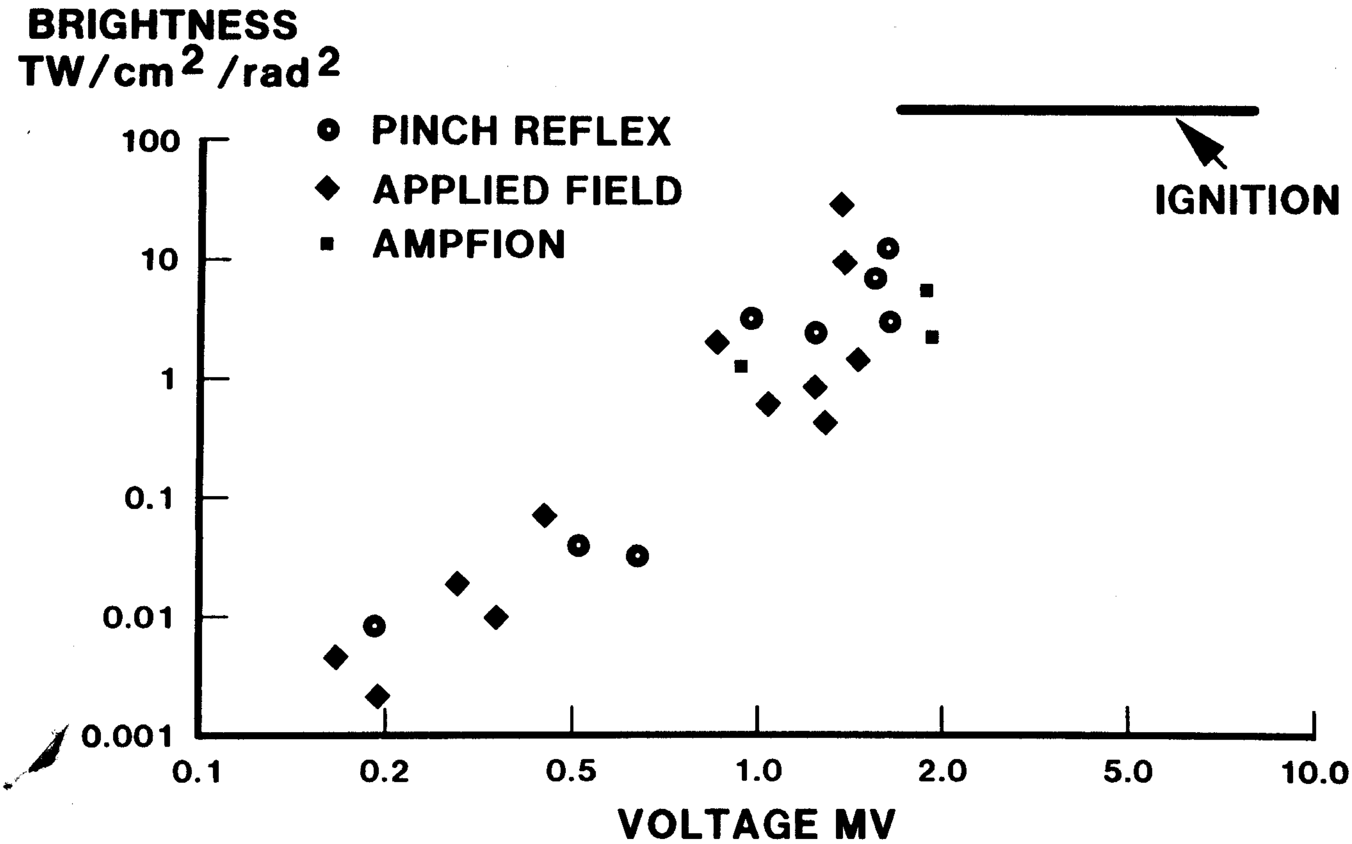


Fig. 3.2-21. Plot of Beam Brightness vs. Voltage.

have to have critical parts replaced. Currently there is very little work underway to solve this problem as the major labs involved in light ion beam fusion are addressing the problems of power brightness, increasing the total energy, and shaping the pulse as well as just understanding the basic physics of diodes.

3.2.4 Beam Propagation and Focusing

One area of active research in light and heavy ion beam fusion is beam propagation from the final focusing element of the driver to the target. This is not as much of a problem for laser fusion as it is for ion beam fusion because it is fairly easy to focus the laser beam with a system of large mirrors or lenses as long as the gas density in the reaction chamber is below a few times 10^{15} particles/cm^3.

3.2.4.1 Light Ion Beam Propagation and Focusing. One of the biggest problems in light ion beam fusion is focusing the mega-amps of light ion current onto a small target. If there are only the positively charged ions propagating to the target, the space charge forces repelling those charges prevent the beam from coming to a small focus at the target. Thus, in most schemes for focusing light ions onto a target, electrons are mixed with these ions to reduce this space charge force. Ions propagating by themselves will lead to an electrical current which will also create magnetic forces which can de-focus the beam or limit the current that can be carried in a beam. Any method of focusing the beams must overcome the problems caused by these two forces.

One method is propagation in a plasma channel, which is shown schematically in Fig. 3.2-22 (20). The reaction chamber is filled with a gas at a density of 10^{17} to 10^{18} particles/cm^3 and a path of least resistance for the beam is formed in the gas between the diode of the driver and the target by rarifying and ionizing the gas along that path. This is achieved by first defining the path of this channel by slightly ionizing the gas with a laser or by exploding a small wire by discharging a large capacitor across it. A current is discharged through the channel to form confining azimuthal magnetic fields which will guide the ions down the channel and which is frozen into the conducting plasma of the channel. Then ions are accelerated by the pulse power driven diode and are injected into the diode end of the channel. The ions are kept inside the channel by the magnetic fields and they undergo betatron oscillations as they move down the channel to the target.

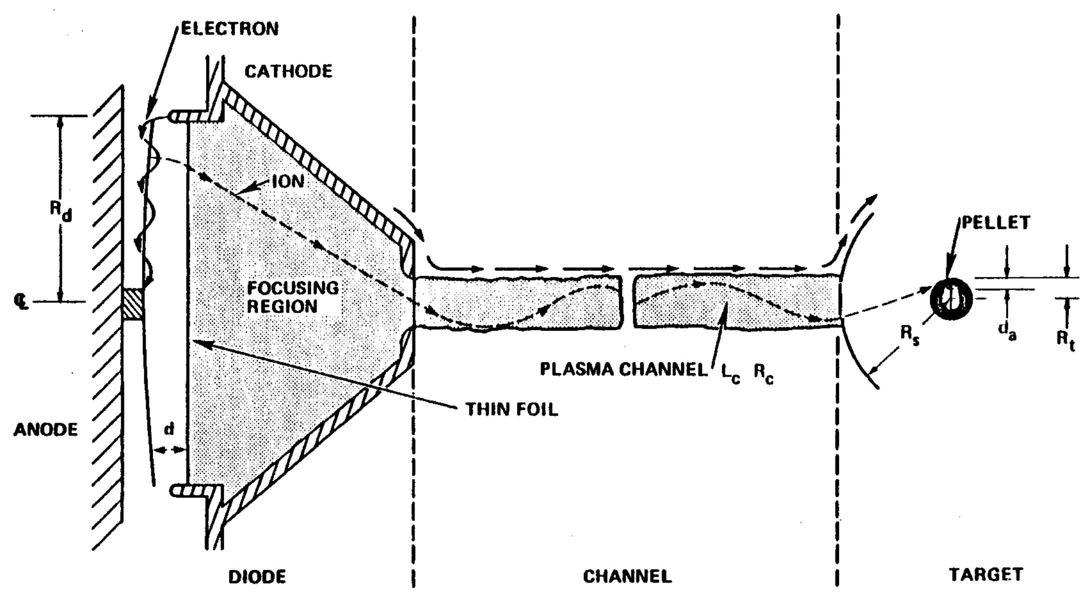


Fig. 3.2-22. Schematic drawing of ion beam transport via a preformed plasma channel from the diode to the target.

This scenario has been investigated experimentally and theoretically in the last five years at NRL and Sandia. Ion propagation over several meters has been demonstrated at both labs. There are uncertainties about the best type of gas, but for the best hydrodynamic behavior of the channel a mass density of about 4×10^{-5} g/cm^3 seems proper. There are also questions about the best radius of the channel which have not been resolved. It is thought that in a reactor the atomic number of the gas must be higher than 10 to protect the first wall of the reactor from the majority of the x-rays generated during the explosion of the target. There is also the question of the propagation of the x-ray and ion debris energy from the target back down the plasma channel which is under study at the University of Wisconsin but has not yet been resolved. Creation of channels with low energy lasers is under investigation at the University of Toronto and the University of New Mexico. In a reactor scheme, using lasers instead of wires to guide the formation of the channel is preferred because it is possible to repeat the process rapidly. Work has been done at Sandia in the past few years on the question of ion propagation near the target where the magnetic fields of adjacent channels interfere destructively with each other.

Another method of propagation is the so-called self-pinched mode (21). The beam is injected into a gas at a density of from 4×10^{16} to 2×10^{18} particles/cm^3. The ions early in the pulse ionize the background but lose their energy or diverge out of the beam in the process. In the process of ionizing the gas, pinching magnetic fields are generated by the leading edge of the pulse which pinch the trailing edge. In effect, the beam creates its own plasma channel. Scientists at JAYCOR/NRL have been studying this scheme for the past few years and are still working in this area.

Ballistic focusing has also been considered where charge and possibly current neutralization are achieved through the injection of electrons along with the ions into a vacuum (22). The beam would have to be focused before entering the reaction chamber by a specially designed focusing diode or a system of final focusing magnets. This focusing method has been investigated experimentally and theoretically at TRW and at other labs (20).

3.2.4.2 Heavy Ion Beam Propagation and Focusing. The propagation of heavy ion beams from the final focusing magnets of the driver to the target has been most thoroughly studied for propagation in the ballistic and self-pinched modes (23). Less frequently considered is the propagation in preformed plasma channels. Figure 3.2-23 shows at what ranges of densities, different physical effects dominate the propagation, but it should be noted that the boundaries of these ranges are somewhat vague and depend on details of the ion beam reaction chamber.

At very low background gas density, where there are no beam disrupting plasma instabilities and the number of beam ions which change their charge due to collisions with the gas atoms is low, the beam ion only can be deflected by electrostatic forces due to other beam ions. If the beam system is properly designed, the problems caused by this force can be avoided. This problem is not severe because of the large inertia of the heavy ions. The ions then

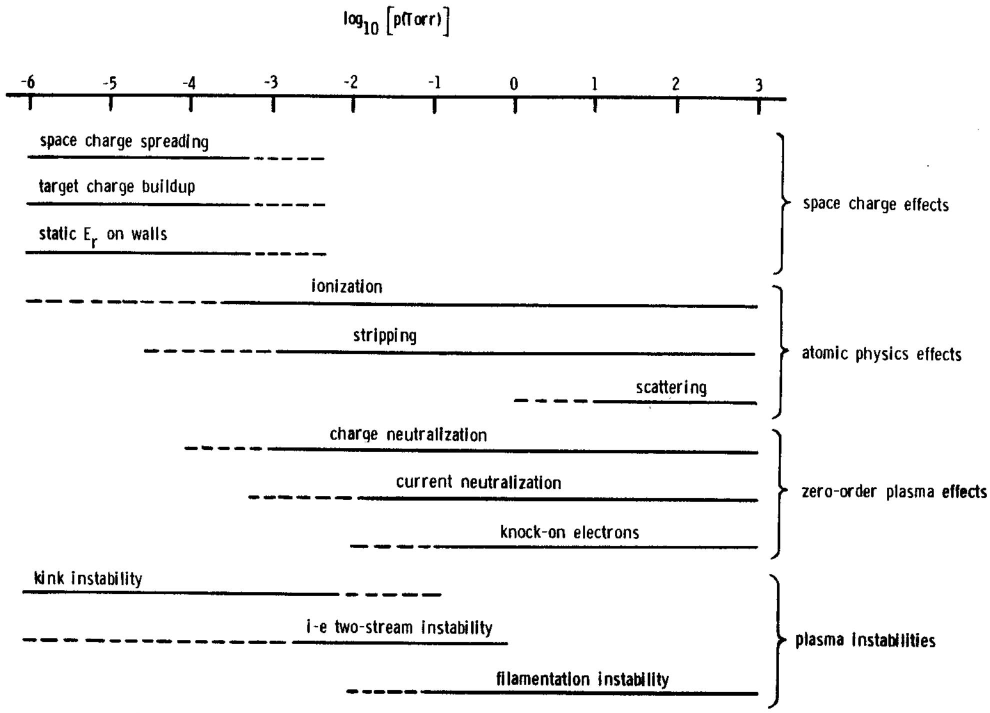


Fig. 3.2-23. Transport effects as a function of gas pressure. This is the pressure the gas would have if it were at 0°C.

travel in almost straight lines after leaving the driver and may thus be focused onto the target by a system of final focusing magnets which are positioned at the end of the driver. Computer simulations of beam propagation in the ballistic mode have been performed at NRL and Max-Planck-Institut für Plasmaphysik (IPP) in West Germany and final focusing studies have been done at the University of Giessen, also in West Germany (24). Experimental work is lacking because there are presently no heavy ion beam accelerators with the kA currents needed to conduct a meaningfully experimental demonstration of propagation. There has been recent work, both experimental and theoretical, on charge changing collisions between beam ions and gas atoms done at ANL, LBL, ORNL and some foreign labs.

Propagation in the self-pinched mode and in preformed plasma channels is the same as for light ions and occurs at high enough densities that the deleterious plasma instabilities are damped by collisions. This density is very difficult to determine. Theoretical work on the self-pinched mode is being performed at LBL and at JAYCOR/NRL.

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3.3 ICF DRIVER FACILITIES

In the past five years, several large experimental drivers have been built in the U.S. and around the world (1-11). They have been mainly of three types: Nd:glass lasers, CO_2 lasers, and pulsed power machines for driving ion and electron diodes. The lasers and the pulsed power drivers operate in rather different regimes. Lasers are highly focusable and may have short pulses which lets them put high power densities on a target but they are still limited in their total energy. Pulsed power machines can deliver large total energy but the beams are not yet very focusable and the pulse widths are much longer than for lasers so that the power densities that they can deliver are relatively low. In this section we list the parameters of the presently or soon to be operating drivers of these three types. There are drivers of other types, most notably heavy ion drivers, which we will not list here but their development is discussed in Section 3.2.3.

3.3.1 Nd:Glass Lasers

The relevant parameters for the major Nd:Glass laser systems (1-3, 9, 10) are given in Table 3.3-1. The pulse widths are comparable for all of those lasers listed. The Shiva, Nova and Novette lasers stand out because their powers and total energies are much larger than those of the other lasers. These lasers need the large powers and energies because they are designed to be used in pellet implosion experiments. One should notice that at the present time none of these three are operating because Shiva has been terminated and some of its parts are being used in the construction of Novette which will begin operation in late 1982 or early 1983. Nova will not be built for a number of years. Because there are none of the large Nd:glass lasers operating, few implosion experiments have taken place in 1982 and those have been done on much smaller lasers. Once Novette begins operation there may again be some interesting implosion experiments.

The other laser systems listed in Table 3.3-1 are mainly used for laser-target interaction studies. They have been used and are still being used to provide much of the physics understanding described in Section 3.2.1.

Most of these laser systems have been used with frequency doubling and tripling systems to operate at 0.53 μm and 0.35 μm. For high intensity light, this frequency conversion can be achieved with more than 70% efficiency.

3.3.2 CO_2 Lasers

Parameters for some of the world's major CO_2 laser systems (2, 3, 12) are given in Table 3.3-2. The Antares laser system will not be completed until 1983. The total energy of the two large lasers, Helios and Lekko VIII, is 10 kJ which compares favorably with the 15 kJ of Shiva. However, one must remember that the coupling between CO_2 lasers and targets is much less efficient than for Nd:glass lasers. Thus the 10 kJ of CO_2 light put onto the target by these lasers will not put nearly as much energy into implosion as the 15 kJ of Shiva. Implosion experiments have been carried out on Helios and the Japanese Lekko VIII laser.

As in Nd:glass lasers, the lower energy lasers are mainly used for laser target interaction experiments. The Helios laser has been found to be very useful for studying the generation of magnetic fields and inhibited transport in target material described in Section 3.2.1.1.

TABLE 3.3-1

Major Nd:Glass Laser Facilities In the Western World

	Argus	Shiva	Novette	Nova	Omega	Chroma-I	Pharos-II	GEKKO IV
Laboratory	LLNL	LLNL	LLNL	LLNL	U. Rochester	KMS	NRL	Osaka U.
Dates of Operation	1976-1981	1977 1981	1982-	1984	1978-	1979-		1978-
Pulse Width (ps)	30-1000	100-2000	100-5000	100-5000	30-100	100	100-1000	50-1000
No. of Arms	2	20	2	10	24	2	2	4
Total Power (TW)	5	30	15	80-120	12	1.9	0.7	4
Total Energy (kJ)	2	2-15	15	80-120	4	0.9	1	1

TABLE 3.3-2

Major CO_2 Laser Facilities in the Western World

	Helios	Antares	Lekko II	Lekko VIII	Coco II
Laboratory	LANL	LANL	Osaka U.	Osaka U.	NRC Canada
Pulse Width (ps)	1000	1000	1000	1000	2000
No. of arms	8	24	2	8	1
Total Power (TW)	10	40	1	10	0.2
Total Energy (kJ)	10	40	1	10	0.2

3.3.3 Pulsed Power Drivers

Some of the world's pulsed power drivers (2-8, 11) are listed in Table 3.3-3. In addition to these, which are all in the U.S., there are machines with about 1 TW of power at Osaka University, at Karlsruhe in West Germany, in France, and in the U.S.S.R. Notice that the total energy in these devices is large compared to the lasers mentioned in this section but there has been difficulty in focusing this energy onto a target. Also notice that the pulse widths are more than an order of magnitude higher than those for lasers. The efficiencies for some of the devices are given and they are much higher than any presently available high power lasers. Efficiencies are not given for PBFA-I because SNL is still in the process of choosing a diode, or for PBFA-II because it is still being built. The parameters listed in this table are dependent on the diode and target chamber because the diode and beam add inductive loads to the driver and affect its performance.

Most of the published results for all of these drivers deal with diode experiments and imploding foil experiments. It is for implosion experiments that PBFA-I and PBFA-II are very promising because they are the first drivers of any type which will deliver the energy expected to be needed for a successful implosion of a fusion reactor target.

TABLE 3.3-3

Major Pulse Power Drivers in the U.S.

	Proto-I	Proto-II	PBFA-I	PBFA-II	Gamble-II	Aurora
Laboratory	SNL	SNL	SNL	SNL	NRL	Harry Diamond Labs
Diode Voltage (MV)	1-2	1-3	2-4	2-28	0.8-1.4	5
Current (MA)	0.5	10-3.3	15-7.5	50-6.3	0.3-0.7	0.07
Power (TW)	0.5-1	10	30	100	0.24-1.0	0.4
Pulse Width (ns)	25	50	30	35	50	140
Total Energy (MJ)	0.12-0.25	0.5	0.9	3.5	0.05	0.05
Diode Efficiency (%)	75 - > 90	80	---	---	40-70	30

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3.4 REACTOR STUDIES

3.4.1 General Information

The major feature of ICF reactor designs is the separability of the driver from the reactor vessel and blanket. In magnetic fusion, the magnets, plasma engineering, heating systems, blankets and shielding are all interrelated and compromises must be made in one area to allow improvements in another area. The operation of the driver in an ICF reactor is usually almost independent of the remainder of the power plant. The choices of driver and beam propagation method set ranges of operating parameters for the reactor vessel gas conditions and the target injection but, within these ranges, the designers are free to adjust the design to enhance first wall lifetimes, tritium breeding and heat recovery. This is a major advantage of ICF over magnetic fusion and makes dealing with the governmental problems mentioned at the beginning of this chapter worthwhile. Specifically, one design feature which has appeared in various forms in several studies is the placement of a liquid metal between the target explosion and the first structural wall. This has improved first wall lifetimes through reduced thermal, physical and neutron loads on the first wall. It also has safety implications as mentioned in Chapter 5.

Other important advantages ICF reactors would have over magnetic machines include easy access to blanket modules, easier startup and shutdown and fewer safety, lifetime and physical structural problems associated with magnets, if there are any magnetics. There are, however, some possible disadvantages of ICF designs including target manufacturing, target storage, driver rep-rate (except heavy ions), availability and reliability and cleanup of radioactive target debris.

To investigate the feasibility of ICF reactors, reactor conceptual design studies were started in the early 1970's. By 1977, many concepts had been proposed and studied and reactor designs had been published, for example the laser fusion reactor design SOLASE. The HYLIFE study was started in 1977 and although there has never been a complete report issued, various publications have been made, with some as late as 1981. Heavy ion beam accelerators have been considered as prime candidates for drivers, also in the mid seventies, but the first complete reactor study by Westinghouse (this study has a laser driven version also) was started in March, 1979 and completed in February, 1981. It was soon followed by the HIBALL design, which was started in January 1980 and completed in June 1981. Light ion beams have also been contenders for quite some time and have received more attention since electron beams were declared unsuitable as drivers. The EAGLE study was started in early 1980 with Phase I completed in April, 1982. Table 3.4-1 gives the main parameters of these ICF conceptual design studies.

3.4.2 SOLASE

A comprehensive conceptual design of a laser driven fusion power reactor, SOLASE, was completed by the University of Wisconsin Fusion Engineering Program in 1977 (1). Table 3.4-1 identifies key parameters and general physical characteristics are shown in Fig. 3.4-1.

The laser system is generic wherever possible but based upon the CO_2 laser if specific values are necessary. The spherical reaction chamber contains a noble gas at very low pressure. This buffer gas protects the first structural wall by absorbing or attenuating target - generated ions and x-rays and then reradiates energy to the wall over an extended period, precluding excessive temperature rises and potential material degradation. The first wall and blanket system is constructed of graphite fiber composite in a cellular con-

TABLE 3.4-1

Pertinent Parameters of Main ICF Conceptual Design Studies

	LASERS			HEAVY ION BEAMS		LIGHT ION BEAMS	
	SOLASE	HYLIFE	WESTINGHOUSE (LASER)	WESTINGHOUSE (HIB)	HIBALL	EAGLE (A)	EAGLE (B)
Type of Driver	CO_2	KrF	CO_2	RF Linac	RF Linac	Sin. St. Diode	Lin. Accel.
Driver Efficiency (%)	7	5	10	30	26.7	20	25
Driver Radiation	10.6 μm	0.27 μm	10.6 μm	$^{131}Xe^{1+}$	$^{209}Bi^{2+}$	D	Ne
Ion Energy (GeV)	N/A	N/A	N/A	1.0	10	6.3	150
Driver Energy (MJ)	1.0	4.5	2.0	2.0	4.8	5.0	5.8
Beam Power (TW)	1000	1000	2000	150	240	300	200
Type of Target	Cryog.	Cryog.	Cryog.	Cryog.	Cryog.	Cryog.	Cryog.
No. of Shells in Target	1	1	2	1	1	1 or 2	1 or 2
Target Yield (MJ)	150	1800	350	350	396	300	300
Target Gain	150	400	175	175	83	60	60
Repetition Rate (Hz)/Cavity	20	1.5	10	10	5*	3	3
Chamber Shape	Spher.	Cylind.	Spher.	Spher.	Cylind.	Cylind.	Cylind.
Chamber Dimension (m)	R = 6	R = 5, h = 8	R = 10	R = 10	R = 5, h = 10	R = 4, h = 10	R = 4, h = 10
Gas in Chamber	Ne or Xe	Li	Xe	Xe	Pb + Li	He + Xe	Xe
Gas Density (n/m^3)	2.4×10^{21}	7×10^{20}	3.6×10^{21}	1.6×10^{19}	1.0×10^{17}	2×10^{25}	2.2×10^{23}
Beams per Chamber	12	2	108	20	20	24	5
Breeder/Coolant	Li_2O	Li	Li	Li	$Li_{17}Pb_{83}$	Li	Li
Structural Material	Graphite	2-1/4 Cr-1 Mo	HT-9	HT-9	HT-9	HT-9	HT-9
First Wall Protection	Buffer Gas	Li Jets	Ta Coating	Ta Coating	INPORT Units	Li Spray	Li Spray
Maximum Coolant Temp. (°C)	600	500	500	500	500	783	783
DT Power (MW_{th})	3000	2700	3500	3500	2000*	900	900
Net Elect. Out. (MWe)	965	1004	1207	1207	942*	290	290

*Values are for one chamber. Plant has four identical chambers served by a single driver.

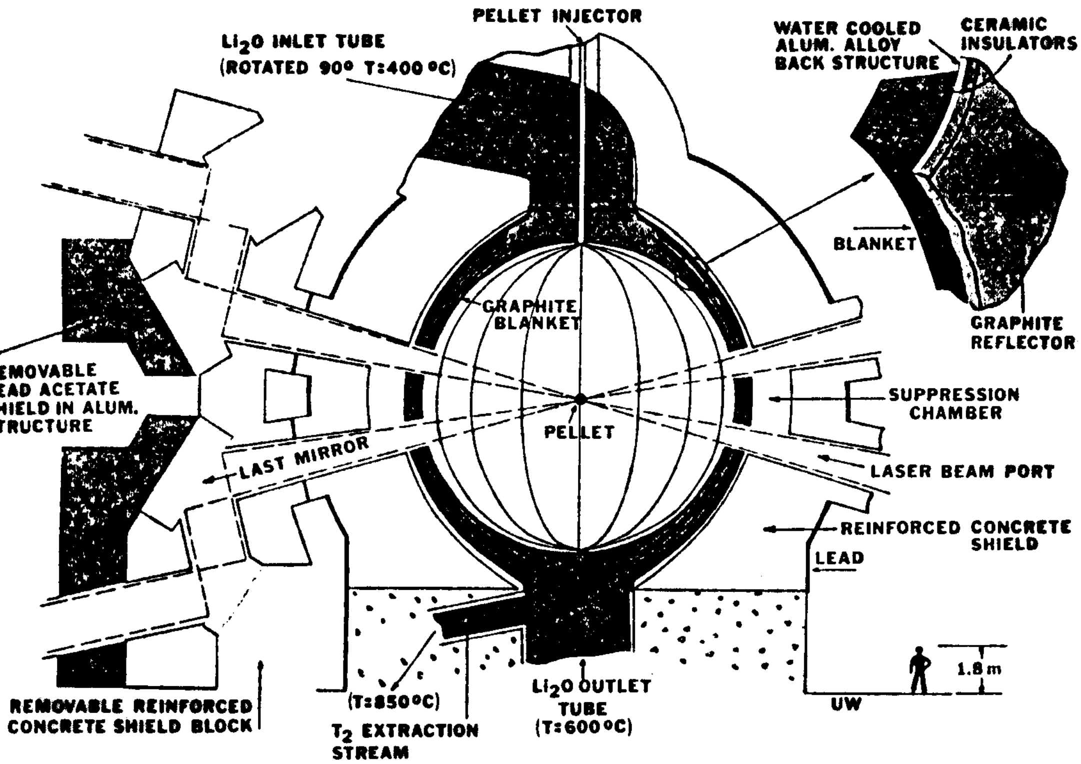

Fig. 3.4-1. SOLASE conceptual laser fusion reactor.

figuration; the entire blanket is comprised of sixteen segments. For tritium breeding and heat transport, lithium oxide in particulate form (100-200 μm diameter) flows through the blanket zones at different rates determined by the radial variation in energy deposition. Calculated temperatures for the concave and convex surfaces of the first wall are 1800 and 1400°C, respectively. At these temperatures, radiation swelling of the graphite is not expected to be a significant problem. Other noteworthy features of the blanket include an acceptable breeding ratio (1.33) and a very low level of induced radioactivity. The graphite components are supported by an aluminum alloy structure and thus radioactivity of the entire assembly drops rapidly after shutdown, facilitating module replacement.

Balance of plant studies and economic analyses have shown the basic SOLASE design to be sound. A viable first wall/blanket system has also been developed. However, uncertainty exists concerning cavity gas response. The state-of-the-art for such calculations has advanced significantly and recent analysis indicates that under the original cavity conditions, the buffer gas would not retain energy for the required period of time. But there is also evidence to indicate that such a system can function at high pressures. Thus, the gas protection scheme continues to offer promise but there is need for experimental and theoretical analysis, particularly for gases and mixtures at higher pressures.

3.4.3 HYLIFE

The HYLIFE (2) (acronym for high yield lithium injection fusion energy) conceptual laser fusion study was initiated in 1977 at LLNL (with other university and industrial participants) and has gone through several iterations culminating in a design with the parameters listed in Table 3.4-1.

The chamber is cylindrical 5 m in radius, 8 m high and is constructed of 2-1/4 Cr-1 Mo steel. It has a yield of 1800 MJ, a very high gain of 400 and at a repetition rate of 1.5 Hz, provides 2700 MW_{th} of DT power which converts to a net power output of 1004 MWe. The driver is a KrF laser at 0.27 μm, with two bundles of beams. The final mirrors are located at the far end of the containment building 60 m from the chamber center.

First wall protection is provided with a hexagonal array of lithium jets, 20 cm in diameter injected at a velocity of 9.5 m/s through a nozzle plate at the top as shown in Fig. 3.4-2. The lithium jets have a packing fraction of 50% for an effective thickness of one meter. The attenuated integrated neutron flux on the structure is only 0.3 MW/m^2 allowing the chamber to operate 30 years at an availability of 70%. The bottom of the chamber has a pool of lithium with a perforated plate beneath it leading to outlet headers. Upper parts of the chamber are protected with criss-crossed lithium jets. Laser beams penetrate the chamber between jets and the pellets are injected horizontally through one of the beam tubes. Following the shot, the hot gases flow through the array of jets, minimizing wall stresses which may occur due to the impact of accelerated lithium. A substantial fraction of the neutron kinetic energy is dissipated in expansion of the fluid and in liquid-liquid interactions among the colliding jets. Dissassembly of the jets provides an enormous area which acts to condense the lithium vapor. The repetition rate is determined by the

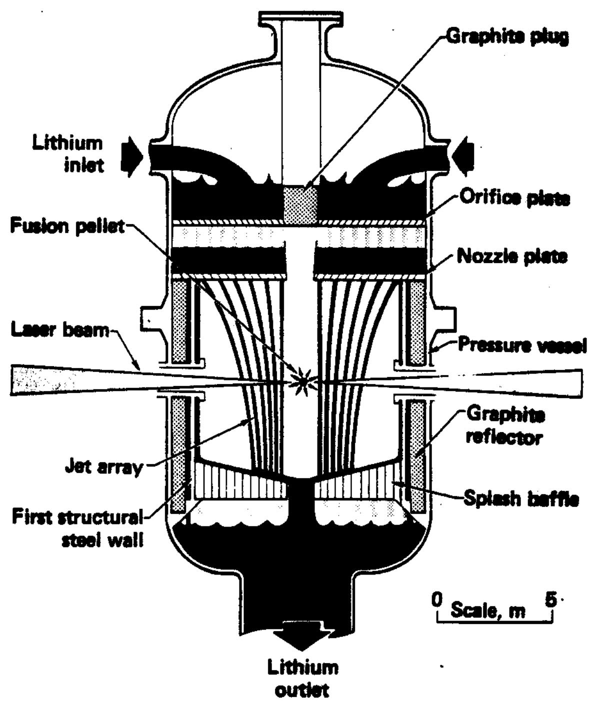

Fig. 3.4-2. Cross section of the HYLIFE chamber.

length of time it takes for the lithium droplets to settle out and the lithium jets to be reestablished.

3.4.4 HIBALL

The heavy ion beam design HIBALL (3) performed by scientists at the University of Wisconsin and in the Federal Republic of Germany, proposes the use of four reactor chambers, each fired at a repetition of 5 Hz, to take advantage of the high duty cycle available from the linear accelerator. One reactor chamber is illustrated in Fig. 3.4-3. Ballistic focussing is used in the chamber. The first metallic wall of the chambers, made of HT-9, is 7 meters from the target. It is protected from the target x-rays, ions, and neutrons by an array of porous SiC tubes through which $Li_{17}Pb_{83}$ is flowing, Fig. 3.4-4.

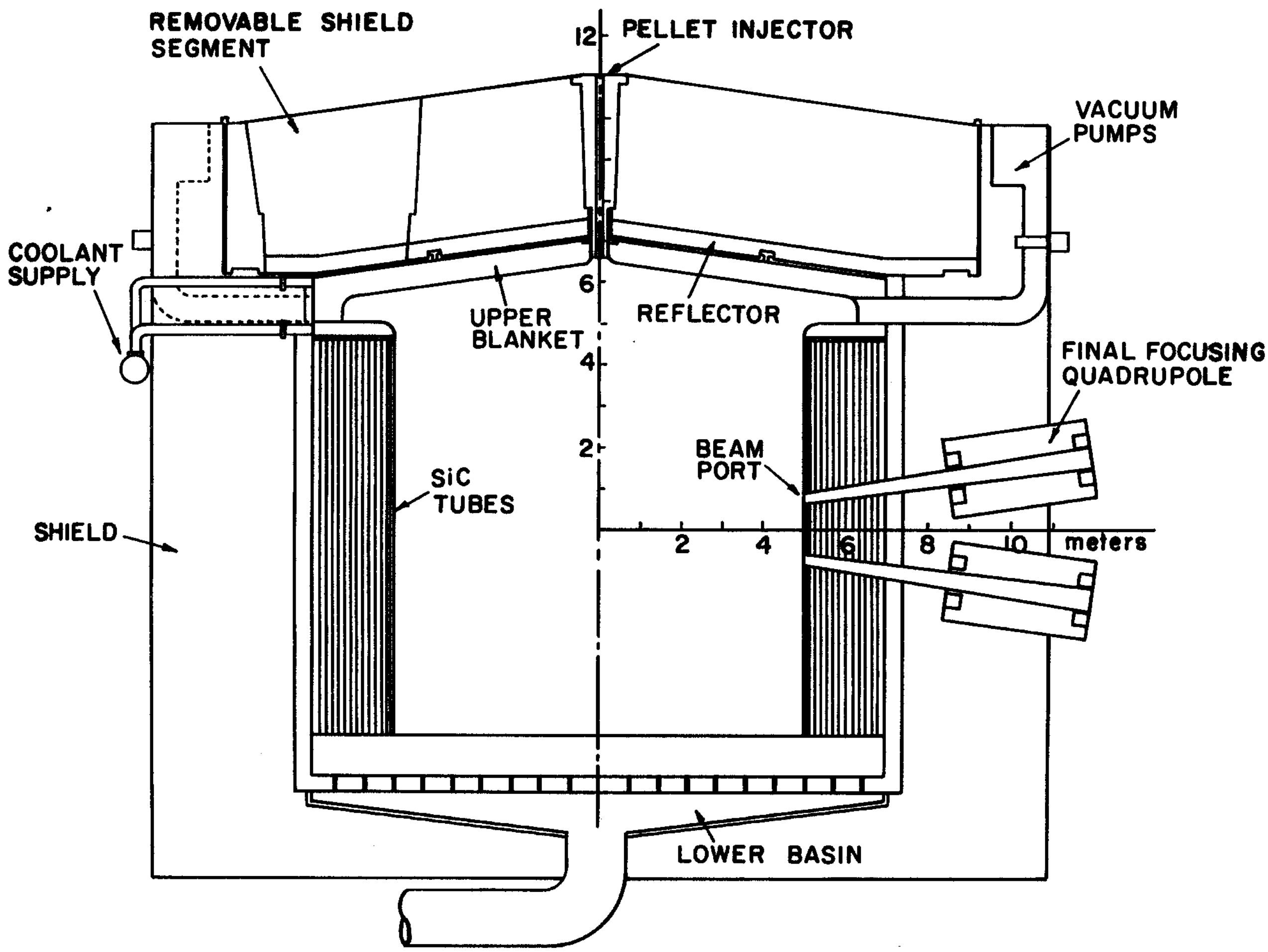

Fig. 3.4-3. Cross section of the HIBALL reactor.

These tubes, called INPORT tubes are protected from the short range x-rays and ion debris by a thin coating of $Li_{17}Pb_{83}$ flowing through the porous tube wall. This coating is partially vaporized on each shot and recondenses on the tube wall in the 200 ms between shots. The maximum coolant temperature is 500°C insuring that the cavity pressure is less that 10^{-5} torr (normalized to 0°C) before the next shot as required for beam propagation. The low solubility of tritium in $Li_{17}Pb_{83}$ results in a tritium inventory of less than 100 g in the blanket and fuel processing system. This low solubility also allows all tritium extraction to be done in the cavity itself. By using the INPORT tubes the radiation damage to the first structural wall is at a level such that it is expected to last the life of the plant.

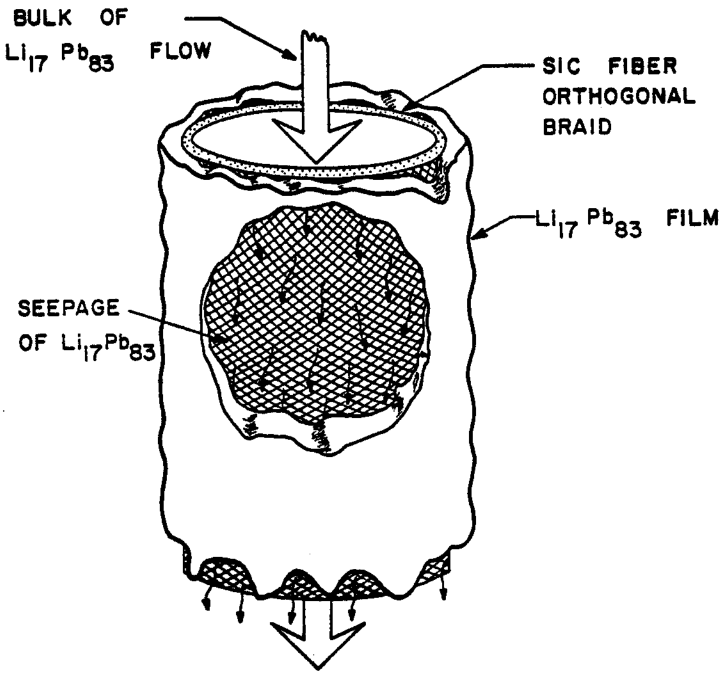

Fig. 3.4-4. INPORT unit.

A preliminary economic analysis of HIBALL indicates that the capital cost (in 1981 dollars) is \$1795/kWe and the corresponding busbar cost of electricity is 41 mills/kWh. These figures are comparable to those calculated for previous tokamak and tandem mirror reactor studies.

3.4.5 Westinghouse Inertial Confinement Power Plant

This ICF design study (4) was undertaken to investigate the comparative merits of commercial power plants based on laser and heavy ion beam drivers and to examine the associated technological problems that need to be solved. The driver and targets were sized to produce the same fusion energy in both systems. While the pressure in the chamber is different for the drivers (< 10^{-1} torr for the laser and < 5 x 10^{-4} torr for the heavy ion beam) the same first wall and blanket concept was used. The first wall is constructed of a tantalum coating on thin wall HT-9 tubes and is unprotected from the target explosion. Tantalum was chosen because of its high melting temperature and because it is a constituent of the target itself. Liquid lithium flows through the first wall tubes to remove the surface heat flux generated by the x-rays and ions as well as the volume heat load generated by the fusion neutrons. A flowing liquid lithium region outside of this tube bank is used to remove the bulk of the neutron energy. Manifolds around the beam ports are used to bring the coolant into and out of the chamber. The results of the calculations indicate that even in the more severe heavy ion case the temperature of the tantalum coating remains below its melting point with a coolant flow of 20 m/s in the first wall tubes. The tritium breeding ratio is ~ 1.2 and the bred tritium is removed from the lithium by a yttrium bed. Figure 3.4-5 shows the schematic of the reactor chamber and illustrates the Ta coated first wall concept.

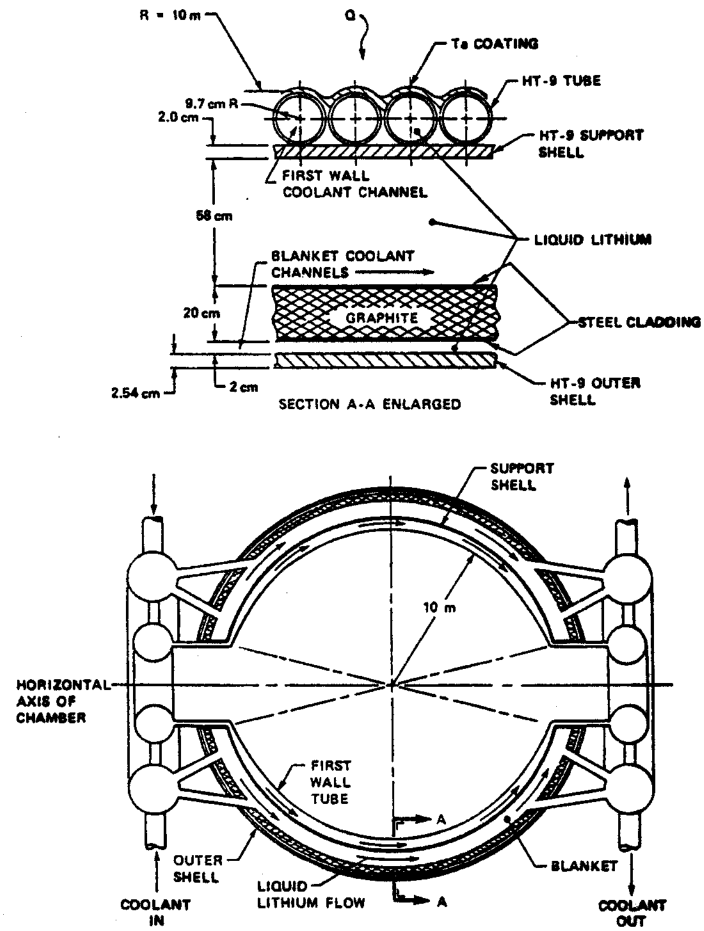

Fig. 3.4-5. Schematic representation of the tubular first wall and blanket concept.

3.4.6 EAGLE

EAGLE is the result of a preconceptual design of a light ion driven engineering test reactor (ETR) prepared by Bechtel Group, Inc. and Physics International for the Electric Power Research Institute (5). Such an ETR would be used for the demonstration of the integrated performance of repetitive light ion beam generators, fusion targets and reaction chamber.

The principal parameters of the reactor are listed in Table 3.4-1. It can be seen that two drivers have been proposed and given equal status at this time. The first is a low voltage (10 MV) capacitive pulsed single stage light ion diode generator. In this case the chamber gas is relatively high density helium for plasma channel formation seeded with xenon for attenuation of target generated x-rays. The second ion beam generator considered is a pulsed linear induction accelerator (PULSELAC). This would use a relatively low density chamber gas of xenon, a requirement for propagation of the self-pinched beams. For each of the drivers, both single and double shell targets have been proposed. Common to these is a spherical shell of frozen DT surrounded by a low Z shield seeded with a high Z material and an ion deposition layer. For the double shell target, a second spherical shell of DT covered by a high Z tamper is suspended within the basic DT shell. Both designs are nominally one centimeter in diameter.

The same reactor chamber would be used for either driver system (Fig. 3.4-6). A novel feature of this design is a one meter thick annular spray of lithium intended to absorb energy from the chamber gas. (This is the basis for the EAGLE acronym - Energy Absorbing Gas, Lithium Ejector.) The lithium falls into a pool at the bottom of the chamber and is then circulated to a cleanup system and intermediate heat exchangers; the secondary heat transport system uses sodium.

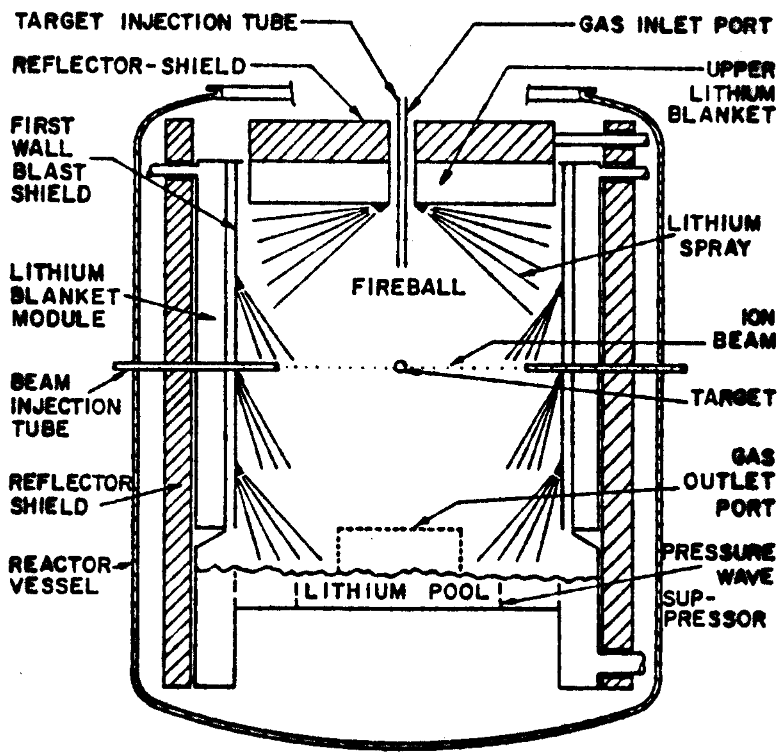

Fig. 3.4-6. EAGLE preconceptual light ion reactor.

A number of areas of major uncertainty have been identified and are itemized on a comparative basis as Critical Issues (see Table 3.4-2). A secondary phase of the EAGLE program, currently in progress, proposes more specific development of requirements for power, target production, tritium breeding, plant availability and cost, eventually leading to a more refined ETR conceptual design.

3.4.7 Critical Issues

Aside from certain generic critical issues such as target fabrication, protection of final focusing elements and shock wave effects, there are issues which are specific to the various designs. These issues are called out in Table 3.4-2.

The outstanding critical issues for SOLASE have to do with the gas protection, graphite structure and the Li_2O breeding material. First wall temperature, gas breakdown and target survivability are all associated with the relatively high chamber gas pressure. Blanket fabrication, erosion and breeding material transport problems are issues linked to the choice of graphite structure and

TABLE 3.4-2

Table of Critical Issues for ICF Reactor Designs

	SOLASE	HYLIFE	WESTING. LASER	WESTING. HIB	HIBALL	EAGLE
Vapor Recondensation		X			X	X
Liq. Metal Jet Reestablish.		X				
Buffer Gas Effects	X		X			X
Liquid Metal Layer Stab. & Wetting Charac.					X	
Metal Coating-Fabr. & Integrity			X	X		
Blanket Fabrication	X		X	X		
Liq. Metal Corrosion		X	X	X	X	X
Breeding Mat. Transport	X					
Target Survivability	X		X			X
Driver Efficiency	X	X	X			
Driver/Target Coupling	X	X	X			
Beam Propagation				X	X	X
Gas Breakdown	X		X			
Driver Development	X	X	X		X	X

Li_2O breeding material. Issues associated with the laser driver, such as low efficiency, and poor coupling to the target are generic to all laser systems.

The most important critical issue for HYLIFE is the low repetition rate necessitated by the time required to settle out the mist and reestablish the liquid metal jets. Liquid metal corrosion and corrosion material transport is an issue for all systems utilizing liquid metals. The KrF laser also suffers from a low efficiency and the need for extensive development.

The Westinghouse design depends on distance and Ta coating for first wall protection and thus, critical issues have to do with fabrication of the tubes, Ta coating and its integrity. The laser version also has the problems generic to CO_2 lasers. The HIB version has a critical issue with beam propagation in the chamber.

The critical issues for HIBALL are vapor condensation and liquid metal layer stability. The cooling, condensation and re-evaporation of the vapor are difficult to model especially when radiation cooling is no longer predominant. Thus, the calculation of the time required to reestablish the initial conditions between shots is difficult. Protection of the INPORT units and the upper blanket depends on the liquid metal layer stability which in turn depends on the wetting characteristics of SiC by $Li_{17}Pb_{83}$.

The predominant critical issue for EAGLE is the need to develop light ion beam generators that are efficient and reliable. For pulsed linear accelerators, beam stability is critical. Other issues for EAGLE are target survivability due to the high density and tritium management, and the viability of energy transfer to the Li "mist" in the chamber.

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Section 4

ALTERNATIVE APPLICATIONS

4.1 HYBRIDS

4.1.1 Physics Principles and Reasons for Consideration

There have been a significant number of papers published studying the desirability of nuclear reactors in which both fission and fusion are employed (see for example Ref. (1)). In this section the rationale for this combination will be repeated and examples presented of the applications of this idea.

In a fusion reactor, such as one based on the D-T fuel cycle, one neutron is produced for every D-T fusion reaction and this neutron carries most of the energy of the reaction (14.1 MeV out 17.6 MeV from the reaction). The energy is recovered by slowing down and absorbing the neutron in the blanket surrounding the reacting region. The blanket must also perform the function of generating enough tritium, at least one tritium nucleus per fusion reaction, to provide additional fuel for the reactor. This process is accomplished by directly generating more than one tritium per initial neutron. For example, this can be accomplished by the $^{7}Li(n,n'T)\alpha$ reaction plus the $^{6}Li(n,T)\alpha$ reaction or indirectly by generating more than one neutron per source neutron, then absorbing these neutrons in tritium producing reactions such as the $^{6}Li(n,T)\alpha$ reaction. In a pure fusion device the neutron multipliers are usually considered to be beryllium or lead through their (n,2n) reactions. With proper design these blankets can have a tritium breeding ratio substantially greater than one or, to put it another way, they can have more neutrons available than necessary to breed new fuel for the reactor. In fact, enough extra neutrons are produced such that if they were used to produce excess tritium, fuel doubling times can be on the order of weeks or months rather than years as is the case for a fission reactor. Consequently, a fusion reactor may be viewed as having a surplus of neutrons accompanied by a relatively small amount of energy. Fission reactors, because of the requirement that at least one of the neutrons from fission is needed to maintain the chain reaction and unavoidable parasitic capture, have a relatively small number of excess neutrons available for breeding. Thus they are considered to be energy rich (~ 200 MeV/fission event) and neutron poor.

The basic idea of the hybrid is to combine fusion and fission to take advantage of the best features of each and, if possible, arrive at something superior to both. For example, the low power density of fusion can be compensated by the high power density of fission while the low neutron production of fission can be compensated by the high neutron production from fusion. Furthermore, the two functions need not be combined in the same physical unit. If the excess neutrons in the fusion system are used to make fissile fuel, as in a breeding reaction, the fuel can be burned in a separate fission reactor. This concept leads to one of the important figures of merit of a hybrid reactor, namely the support ratio. The support ratio is defined as the amount of external fission power that can be produced from fuel from the hybrid per unit of power from the hybrid. Thus, the support ratio is based on the total thermal power (including blanket multiplication) rather than the fusion power of the hybrid. If all plants, fission and hybrid, were to have the same thermal rating then the support ratio is equivalent to the number of fission reactors which could be supplied by one fusion reactor. The support ratio is not only a function of the type of fusion reactor fuel (e.g., D-D or D-T) but also of the type of fission reactor considered and the fuel (e.g., ^{239}Pu or ^{233}U) to be used.

A typical hybrid fuel cycle scenario is shown in Fig. 4.1-1. The fertile fuel is processed into a form suitable for use in the hybrid where it is enriched

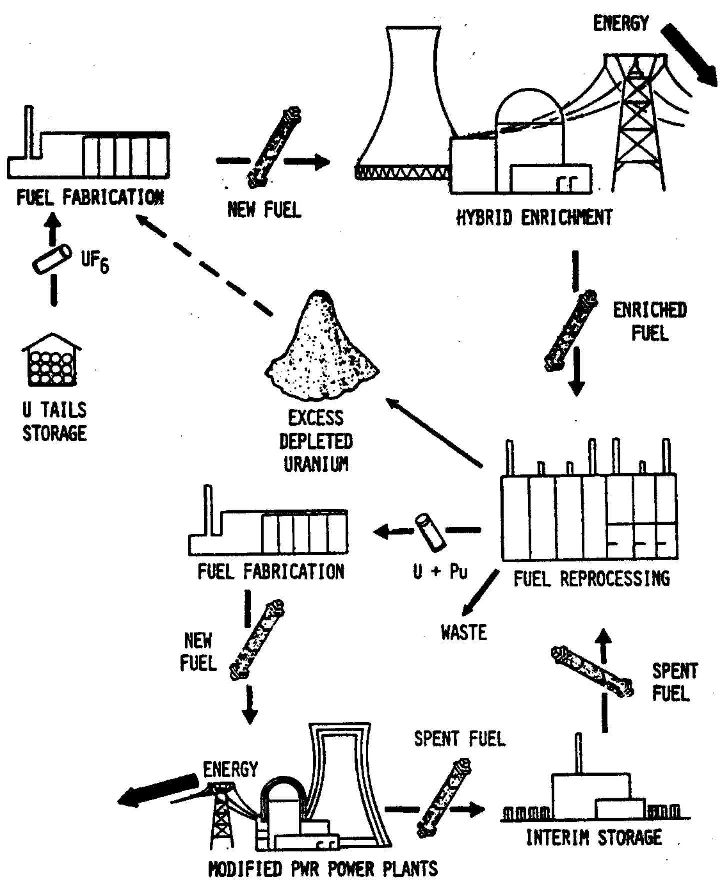


Fig. 4.1-1. Hybrid fuel cycle (from Ref. (4)).

in fissionable isotope content. The enriched material then enters a conventional LWR fuel cycle where it serves as the fissile feedstock. When first conceived, the hybrid concept was intended to enhance the attractiveness of fusion reactors which had low power densities. The fission energy then boosted the net energy available from the reactor to more acceptable levels. Current fusion devices have been designed with much higher power densities than the early systems and therefore the need to boost the power density is not so important. Hence, hybrids are now viewed mainly as fissile fuel production facilities.

The deployment of the hybrid could provide a means by which conventional light water reactors could be assured of a fuel supply with no further reliance on natural uranium resources, relying instead on ^{232}Th or the presently available stocks of fertile ^{238}U. This would also have the advantage of stabilizing the cost of reactor fuel. In addition, if the support ratio were large enough the economics of the fusion reactor might be less of a concern since its cost would be spread over a large number of fission plants and at the same time the design requirements of the hybrid might be eased. That is, the reactor would be designed to utilize a less efficient but more easily attainable mode of operation. The availability of the fusion reactor would still have to be acceptable but its use in a hybrid concept might ease operating constraints since only the net energy generated is important.

4.1.2 Types of Hybrid Reactors

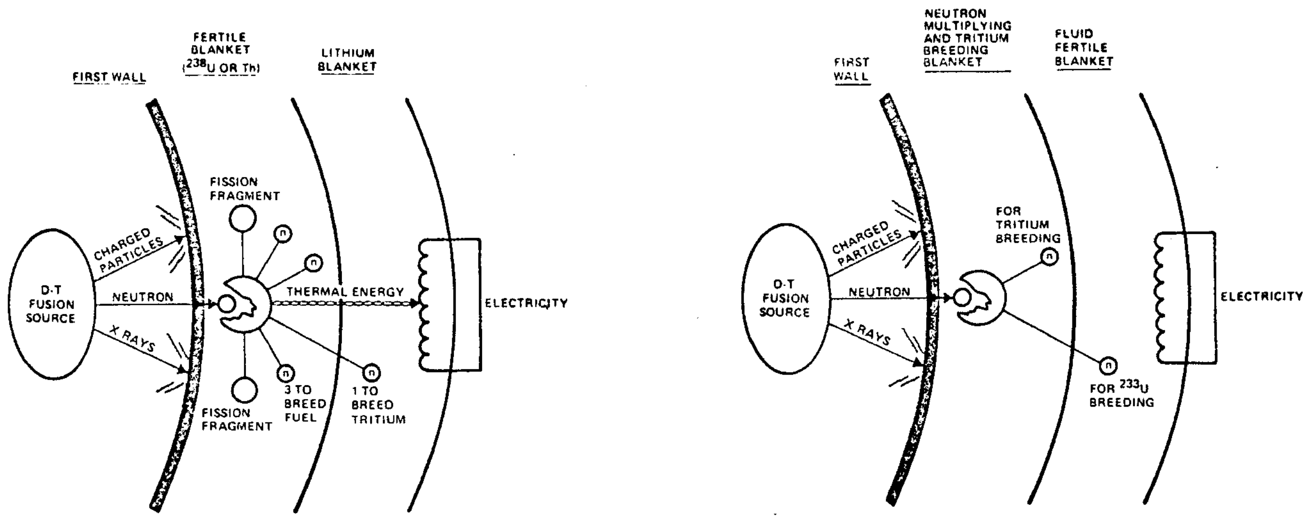
In the previous discussion it was indicated that it was necessary to take advantage of (n,n'T) or (n,2n) reactions to obtain the excess neutrons for use in a hybrid. Beryllium or lead were suggested as neutron multipliers. However, high energy neutrons interacting with fertile materials are also multiplied through fast fission and (n,xn) reactions (~ 4.5 neutrons/fast fission). In this case, however, the multiplication of neutrons is accompanied by the energy from fission. The fact that fertile materials act as neutron multipliers as well as energy producers has led to two somewhat different approaches to hybrid reactors. They are:

1. Reactors with fast fission blankets.

2. Reactors with fission suppressed blankets.

These two approaches are illustrated in Fig. 4.1-2 (1). In the fast fission blanket concept the D-T fusion source is surrounded by a blanket of fertile material. The neutrons from the fusion reactions induce fast fissions in the blanket as described above. Of the neutrons produced at least one must be captured in Li to produce fusion fuel (tritium) while the rest are available for breeding fissile fuel.

In the fission suppressed approach the non-fissioning neutron multiplier and the tritium breeding material surrounds the fusion source. This region also moderates the neutrons below the fast fission threshold. One neutron is available for tritium breeding and the remainder for fissile fuel production.



a) b)

Fig. 4.1-2. Fusion breeder concepts: (a) fast fission blankets (fusion-fission hybrids), and (b) suppressed-fission blankets (from Ref. (1)).

Since most of the neutrons entering the fertile outer region of the blanket are below the fast fission threshold, energy production in this region is minimized. In fact, if this region is designed with mobile fuel which is continually reprocessed the concentration of fissile fuel may be kept low. Although ^{233}Th decays to ^{233}Pa in 22 minutes, the decay of ^{233}Pa to fissile ^{233}U requires 27 days and allows time for removal before it becomes a fissile atom, thus the thermal fission is minimized at the same time.

Both types of blankets offer good fuel breeding performance and each has advantages and disadvantages. The fast fission blanket design, because of the energy produced from fission, has high energy multiplication. Consequently the plant could produce a significant amount of power as well as producing fuel. Thus the fusion system can be operated with a low Q and still provide satisfactory overall plant performance. The fission blankets have high power densities and quickly build up an inventory of fission products. Therefore, concerns with respect to safety issues similar to those encountered in fission reactors will be very important. In addition, if stationary fuel is used in the blanket the production of fissile fuel results in an increasing number of thermal fissions. The resultant increase in power, which may be as much or greater than a factor of two, presents problems in the design of the heat removal and secondary systems or requires a fuel management procedure to limit the change in power with time.

Fission suppressed hybrids produce much more fuel per unit of thermal energy than fast fission blanket hybrids (the support ratio was defined in terms of thermal power from the hybrid). These reactors have a higher support ratio than fast fission hybrids. Since the concentration of fissile fuel in the reactor is kept low, the power density and thus the fission product and actinide concentrations are also low and many of the cooling and safety problems associated with the fast fission blanket are alleviated. To retain the full advantages of the fission suppressed system, thermal fission must be avoided; this implies continuous refueling and reprocessing. The development of these latter processes offers problems of their own which may be difficult to solve.

The higher support ratios and reduced safety concerns have caused much of the recent hybrid work in the 1978-82 period to be concentrated on the fission suppressed concept. While in principle any fusion device could operate as a hybrid, the problems associated with access and utilization of the space surrounding the plasma in the tokamak reactor have resulted in most studies concentrating on some version of a mirror reactor or the various ICF systems. Variations on the basic fission suppressed systems have been investigated. For example, it has been proposed that the fertile fuel in the form of macroscopic particles be circulated at a rate slow enough that reactor level enrichment is obtained with one pass. The enriched particles may be made into reactor fuel with no chemical processing required. All other considerations being equal, the utility of these alternative designs will depend on the cost of the fuel and its suitability for processing into an acceptable form.

4.1.3 Studies and Reviews

Several designs for hybrid reactors have appeared in the past five years (2-7). Each of these has had a somewhat different approach and different characteristics. A brief description of three of these designs is presented to illustrate some of the thinking that has gone into the possibilities of hybrid use.

4.1.3.1 Commercial Tokamak Hybrid Reactor. The Commercial Tokamak Hybrid Reactor (CTHR) (4) design of the Westinghouse group, sponsored by the U.S. Department of Energy, is based on a tokamak of 6 m major radius, 1.4 m minor radius, neutron wall loading of 2 MW/m^2 and a fusion power of 1200 MW. The reactor used ^{238}U as the fertile fuel in a fast fission blanket and was self-sufficient in tritium production. Two different blanket concepts were considered: (1) UC fueled stainless steel clad and structure, He cooling (UC-SS-He); and (2) UO_2 fueled, zircalloy clad, stainless steel structure, boiling water cooling (UO_2-Zr-BW). The motivation for studying the latter was that it utilized many features of proven light water reactor technology. In addition the use of Li_2O and LiH with lead as a neutron multiplier or graphite as a reflector was considered for the tritium producing section of the blanket.

A schematic of the UC-SS-He concept is shown in Fig. 4.1-3. The fertile material is loaded in 17.5 cm canisters ~ 1.5 m long oriented in the toroidal direction. Cooling is provided by passing the He through 1 cm cooling channels in the canister and through a coolant channel between the fuel and the canister shell. The pitch between the coolant channels in the canister is increased in the direction away from the plasma commensurate with the spatial power distribution.

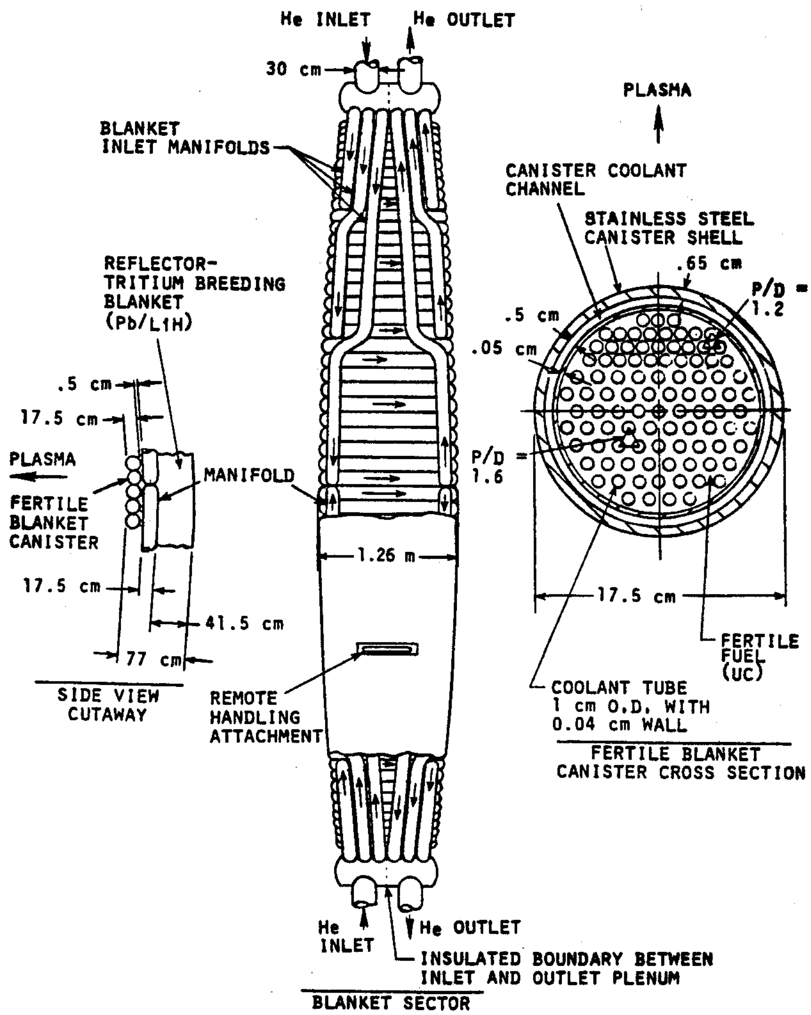


Fig. 4.1-3. CTHR UC-SS-He Blanket Concept (from Ref. (4)).

The UO_2-Zr-BW blanket is shown schematically in Fig. 4.1-4. Here the fertile UO_2 is loaded into zircalloy tubes (1.45 cm O.D., 2.5 m long). The fuel rods are arranged in a triangular lattice with pitch to diameter ratio of 1.1. The tubes are oriented in the poloidal direction to permit upward flow of the two phase boiling water which serves to cool both the fuel and the stainless steel first wall. As in the UC-SS-He design the tritium producing regions are behind the fertile fuel.

Both blanket concepts are operated on a batch basis and require four years of operation to achieve an average enrichment of ~ 3%. Also both concepts have steam cycles attached to recover the considerable amount of energy generated. Since an important consideration in the deployment of hybrids is fuel costs,

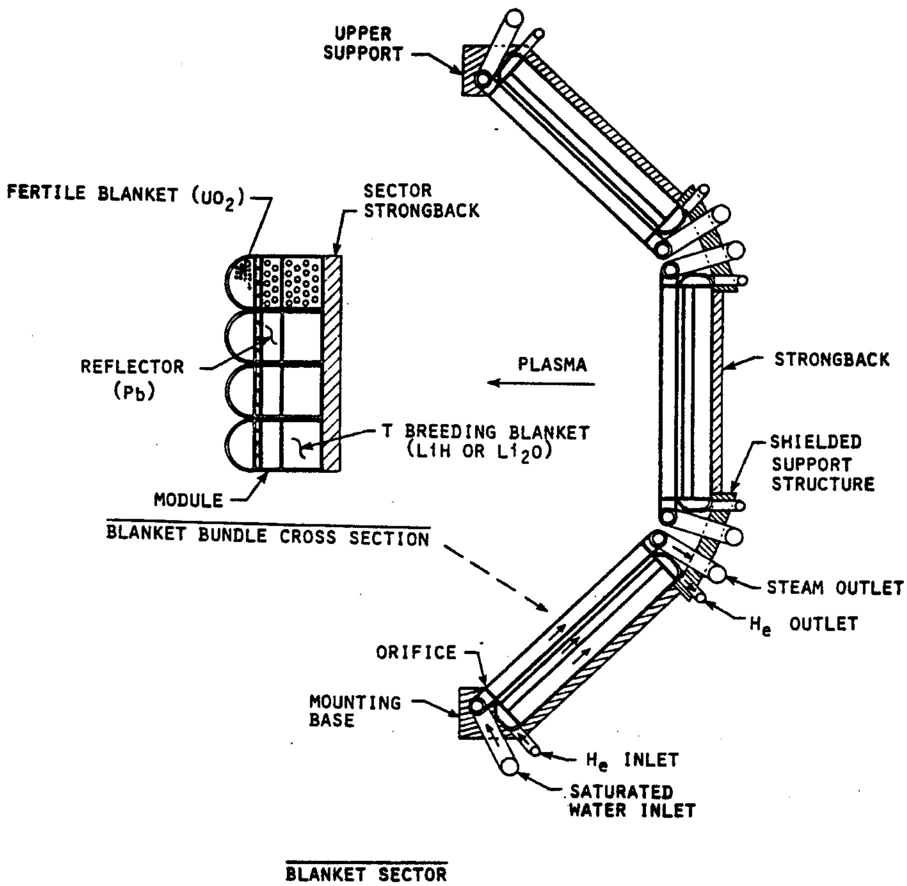


Fig. 4.1-4. CTHR UO_2-Zr-BW Blanket Concept (from Ref. (4)).

an economic analysis was also performed. Table 4.1-1 gives a brief summary of some of the characteristics of the two systems. Note that because of the use of a fast fission blanket, the thermal power increases significantly with time as the plutonium is produced. The cost of Pu from the two systems is similar at 90-100 \$/g but the UC(LiH) blanket has a support ratio of 8 versus 4.4 for the $UO_2(Li_2O)$ blanket.

4.1.3.2 SOLASE-H. The SOLASE-H (5) design prepared by the University of Wisconsin group, sponsored by EPRI, took an approach to the use of hybrids rather different from that of the CTHR. A laser driven ICF reactor was used as the fusion source and was based on the SOLASE design (see Section 3.4.2). The reactor chamber was cylindrical with a radius of 6 m and a height of 12 m. The fusion power was 1240 MW resulting in a neutron wall loading of 1.9 MW/m^2. The blanket design was developed as a consequence of trying to find an answer to the question of whether the hybrid had a place in a nuclear future that would not allow reprocessing in a fuel cycle unless it was diversion resis-

TABLE 4.1-1

Characteristics of CTHR Blanket

(Values are after two years of operation unless otherwise specified)

	UC (LiH)	UO_2 (Li_2O)
Tritium breeding ratio	1.20	1.13
Pu-atoms/fusion neutron	0.97	0.54
Heavy metal inventory (THM)	410	255
Fissile enrichment after 4 years	2.89	2.96
Fuel burn up after 4 years (MWd/MT)	12,700	
Fissile production (kg/yr)	2,730	1,524
Thermal power (MW)	6,100 (4580 BOL, 7620 EOL)	5,840
Fuel costs (\$/g ^{239}Pu)	88	97
Number of 1 GWe LWRs supported	8	4.4

BOL = Beginning of Life, EOL = End of Life

tant. The approach taken was to investigate the possibility of inserting a 17 x 17 PWR fuel bundle loaded with fertile material only and using the hybrid to enrich the fuel in place for direct insertion into a LWR. No reprocessing would be necessary and the irradiated fuel would be self-protected by its own induced radioactivity. Thorium in the form of thorium-oxide in zircalloy cladding was chosen as the fertile material. Since the reactor cavity is cylindrical the design was based on fully fabricated fuel elements stacked around the periphery of the cavity. A schematic of SOLASE-H is shown in Fig. 4.1-5. Lead was used as a neutron multiplier and sodium was used as a coolant to avoid moderating neutrons, thereby minimizing thermal fission in the bred fuel. The fuel bundles are surrounded on all four sides by lithium filled rods which serve to breed tritium and to prevent the diffusion of low energy neutrons into the fuel. Outside this lithium region is a zone of lead filled rods which act both as neutron multipliers and as fast neutron reflectors. To accommodate this thermal neutron trap-reflector region on all sides of the fuel bundles, the bundles are not adjacent to each other around the circumference of the reactor cavity. This particular design was chosen after an extensive survey of alternatives and was found to give an optimum with regard to

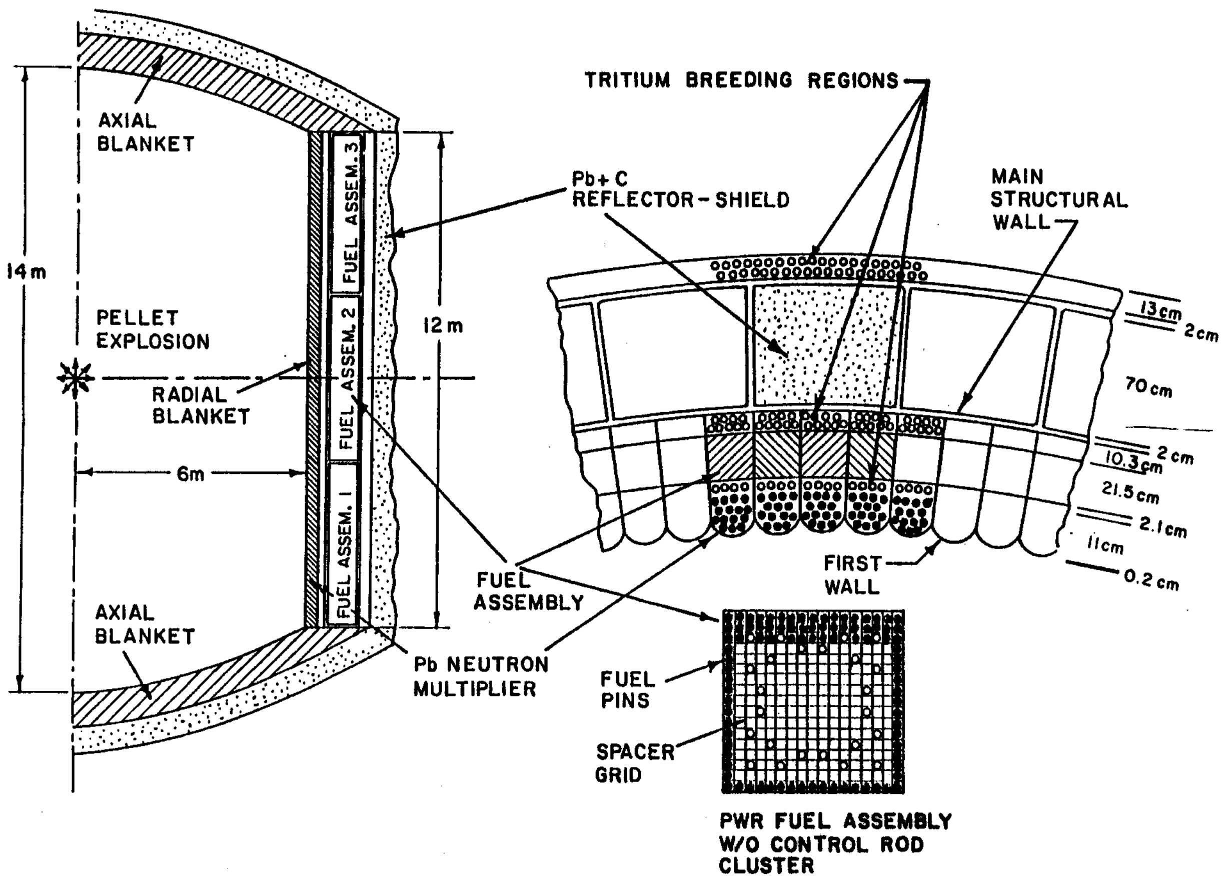


Fig. 4.1-5. SOLASE-H Blanket Concept (from Ref. (5)).

fuel production and uniformity of enrichment over the fuel bundle. Tritium breeding takes place in the lithium rods and in the top and bottom regions of the chamber. To further assure uniformity of enrichment the fuel bundles are rotated 180° halfway through their residence in the hybrid. With this design it was found that an enrichment of 4% would be obtained in 2.6 years. The maximum to average enrichment was 1.1 with the maximum being near the fuel bundle edge. The power increase during the enrichment process was held to 19% by assuming a 4 batch operation. The characteristics of the design are summarized in Table 4.1-2.

4.1.3.3 Fission Suppressed Tandem Mirror Hybrid Reactor. The Fission Suppressed Tandem Mirror Hybrid is a recent design study prepared jointly by groups from the Lawrence Livermore National Laboratory, TRW, Inc., General Atomic Company, Westinghouse Electric Corporation, and the Oak Ridge National Laboratory under the sponsorship of the U.S. Department of Energy (7). In

TABLE 4.1-2

Characteristics of the SOLASE-H Blanket

Tritium breeding	1.08
^{233}U atom/fusion neutron	0.43
Fuel inventory	270 fuel bundles
Fissile enrichment	4% in 2.6 years
Fuel burn-up @ 4% enrichment (MWd/MT)	4300
Fissile production (kg ^{233}U/yr)	2030
Thermal power (MW)	2400-2900
Number of 1 GWe LWRs supported	1.4, no reprocessing

this study a fission suppressed hybrid was designed using a tandem mirror as the fusion driver. Two alternative blanket concepts were investigated and issues such as safety, fuel reprocessing, other fuel cycle issues, economics, and deployment were studied. Primary emphasis was placed on the use of ^{232}Th to produce fissile ^{233}U. In this description the emphasis will be placed on the features of the fission suppressed blanket.

This design is based on a tandem mirror fusion driver having the following basic characteristics:

Fusion power (MW)	3000
Neutron wall loading (MW/m^2)	2
Wall radius (m)	1.5
Central cell length (m)	129
Plasma Q	15.

Two different blanket designs were considered. The first utilized liquid lithium for tritium breeding with minimum neutron loss and molten salt as a carrier for the fertile material. The second used beryllium as the primary neutron multiplier, thorium oxide as fertile fuel, $Li_{17}Pb_{83}$ as a tritium breeding and heat transfer material, and He as coolant.

A central cell blanket module for the lithium-molten salt (Li/MS) design is shown in Fig. 4.1-6. The liquid lithium flows through a manifold arrangement into a 50 cm thick inner region, axially through this region, then exits through another set of manifolds. The lithium has two functions: (1) to re-

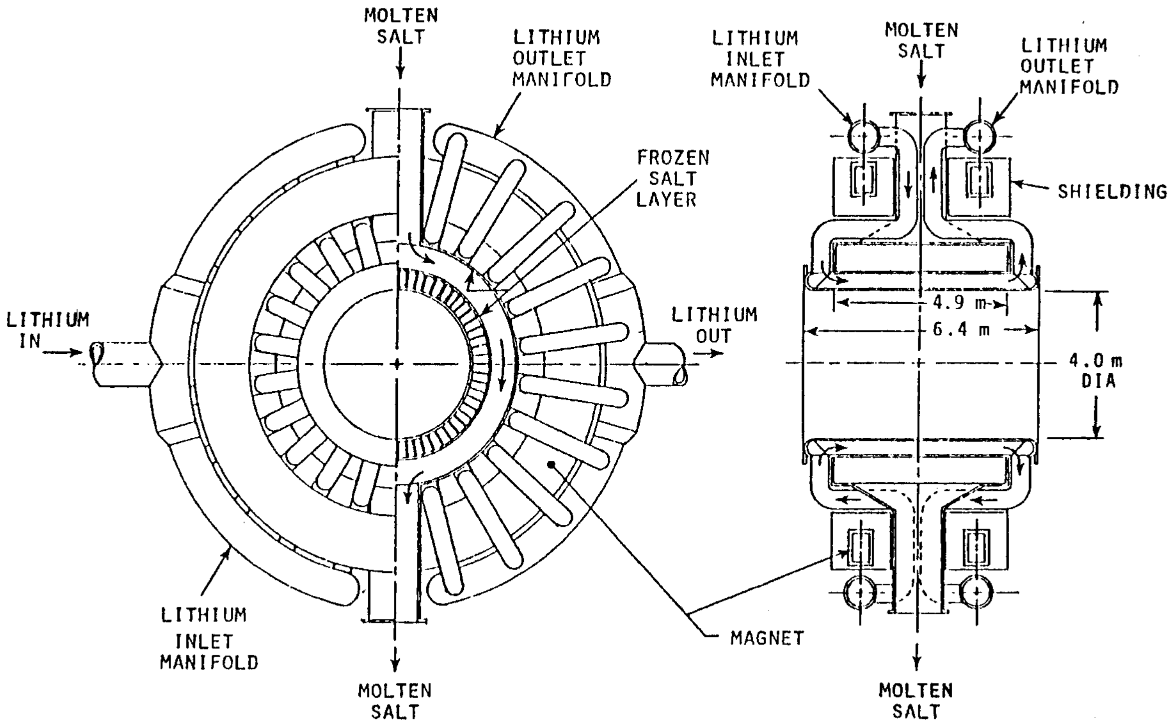


Fig. 4.1-6. Reference liquid metal cooled-molten salt blanket module concept for TMHR reactor (from Ref. (7)).

move the heat deposited in and on the first wall as well as that generated from neutron capture in the lithium itself, and (2) to accomplish the tritium breeding. Since a natural lithium region of this thickness would breed too much tritium, the lithium is depleted to 0.2% in ^{6}Li. With this enrichment the calculations give a tritium breeding ratio from ^{7}Li of 0.661 and 0.389 from ^{6}Li for an overall breeding ratio of 1.05. The exact value of ^{6}Li enrichment is subject to some uncertainty due to recent measurements of the ^{7}Li(n,n'T) cross section. The lithium enters at 220°C, exits at 390°C and removes 2233 MW. The MHD pressure drops in the flowing lithium are minimized by the axial flow in the blanket region and by the use of large cross sectional flow (i.e., low coolant velocity) and electrically insulated duct walls in the manifold region.

The outer 80 cm region contains the thorium bearing molten salt used to breed the ^{233}U. It is slowly circulated both to remove the energy deposited in it and to allow on-line fuel processing. At the reference design conditions the net fuel production is 0.519 ^{233}U per fusion neutron. The molten salt inlet and outlet temperatures are 550°C and 650°C and the power generated in this region is 1425 MW.

Fast and thermal fission is suppressed by moderating the neutrons to lower energy in the 50 cm lithium region, keeping the concentration of fissile fuel low through the on-line reprocessing (concentration of ^{233}U/concentration of $^{232}Th \sim 0.11\%$) and by the low concentration of ^{232}Th in the molten salt. To reduce materials compatibility problems on the intermediate wall a layer of molten salt is allowed to freeze on this surface and other surfaces common to the two fluids.

The beryllium/thorium oxide blanket design is shown in Fig. 4.1-7. The first zone of this design uses nonstressed beryllium blocks immediately behind the first wall. These blocks serve as the neutron multiplier. The volume between the blocks is filled with a liquid suspension including the lead-lithium eutectic and thorium oxide particles of approximately equal density. Also located between the beryllium blocks are concentric helium coolant tubes which remove the heat generated in the blocks and the suspension. The second zone, composed of SiC blocks, serves as a reflector and is also cooled by helium carrying tubes.

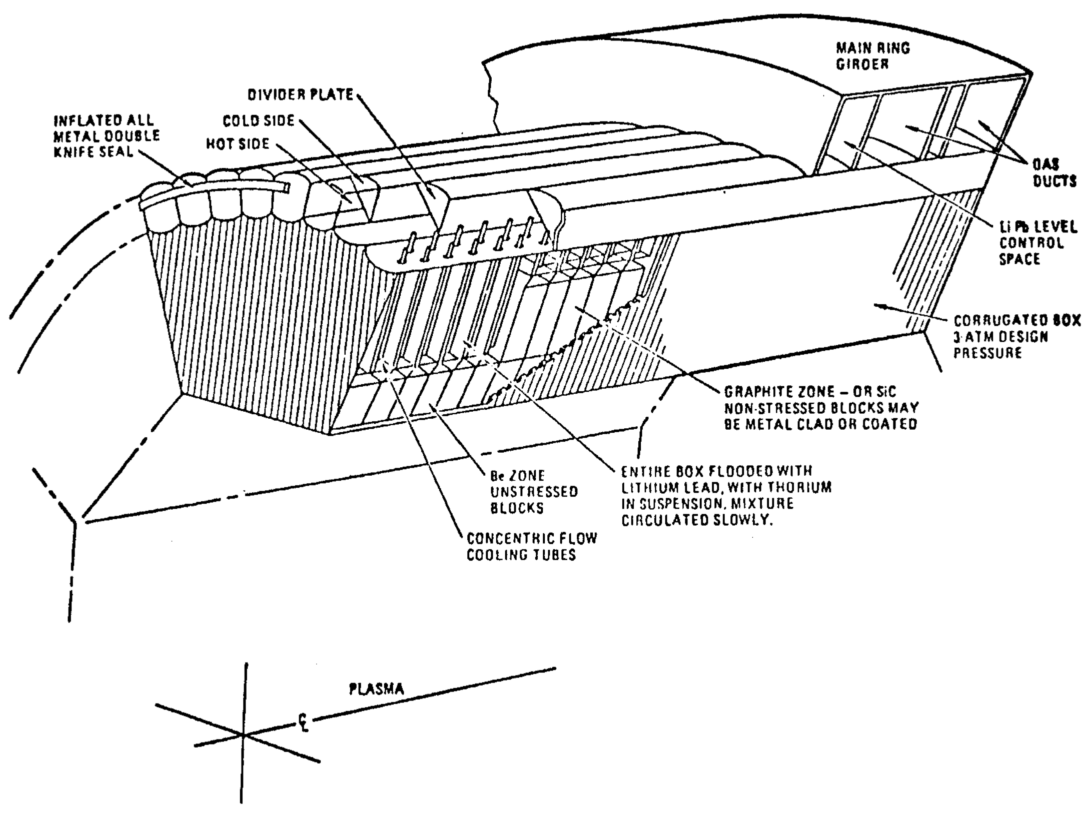


Fig. 4.1-7. Reference beryllium/thorium oxide suspension blanket (from Ref. (7)).

The use of beryllium as a neutron multiplier gives this design excellent breeding performance (fissile breeding ratio of 0.74 with a tritium breeding ratio of 1.05) along with a very low fission rate (0.03/fusion). Fission suppression has been achieved by keeping the fertile concentration low, thus allowing the $^9Be(n,2n)^8Be$ reaction to dominate at higher energies, through the use of the moderating properties of the beryllium and by on-line reprocessing to keep the fissile content of the blanket low. In addition, the fuel residence time in the blanket (1-2 months) is short enough that most of the bred fuel will not have decayed from ^{233}Pa to ^{233}U.

A more complete analysis of these two approaches to blanket design shows that each has its own development problems, both in the reactor and in the fuel reprocessing, safety advantages and disadvantages, and fuel characteristics. The authors have made an analysis of the economics of the reactor and report the results in three ways: support ratio, electricity costs and equivalent U_3O_8 costs. In terms of the support ratio the results are summarized in Table 4.1-3.

The higher support ratio for the Be/ThO_2 system is due to the 50% higher fissile fuel production in the blanket.

In terms of electricity cost the authors estimate that for a denatured uranium fuel cycle the system electricity costs would be ~ 13% and 9% higher than current LWR costs for the Li/MS and Be/ThO_2 blankets.

In their third type of economic comparison they indicate that the use of this tandem mirror hybrid reactor with the Li/MS blanket would result in a fuel cost equivalent to U_3O_8 at 201 $/kg while with the Be/$ThO_2$ blanket the equivalent cost would be 168 $/kg.

The report also analyzes the uncertainties in these costs and concludes they could result in less than a 10% increase in the cost of system electricity.

4.1.4 Status and Conclusions

It is clear from this brief review that the issues surrounding the hybrid transcend the design of the reactor itself. The concerns and problems with reprocessing, safety, proliferation, the nature of the fuel, refabrication, operation of the client fission reactors, deployment scenarios, and economics all have determining roles even given that the hybrid can be built. Much of the recent work in hybrid reactors has been devoted to these issues in an effort to establish their importance, more precisely define them, evaluate their impact and develop solutions or alternative approaches. A large fraction of this work is summarized in Ref. (1) and its bibliography. Because of its close connection with operators of potential client fission reactors, EPRI has been particularly active in the efforts to evaluate the various aspects attendant to the introduction of hybrids. On the other hand, the U.S. DOE program has devoted very little resources to this area and shows signs of reducing it further in the coming years. Notable among recent work sponsored by EPRI is the proceedings of a workshop on the Technical Feasibility of Hybrids (8)

TABLE 4.1-3

Characteristics of the Fission-Suppressed Tandem Mirror Hybrid Reactor (7)

	Blanket Type	
	Li/MS	Be/ThO_2
Tritium breeding ratio	1.05	1.05
^{233}U atoms/fusion neutron	0.49	0.73
Blanket thermal power (MW)	3685	4460
Fissile production (kg ^{233}U/yr)	6360	9475
Support ratio:		
a) Denatured-uranium fuel cycle	10.15	13.4
b) Denatured-thorium fuel cycle	13.7	17.5

a) 30% of LWRs in this system burn excess plutonium generated in 3% ^{233}U/97% ^{238}U LWR fuel.

b) 14% of LWRs in this system burn excess plutonium generated in 3% ^{233}U, 19% ^{238}U, 79% ^{232}Th LWR fuel.

and a report on the energy and economic consideration of hybrids (9). These references expand in greater detail on many of the issues mentioned above.

In conclusion, the past five years have seen considerable progress in understanding the role of hybrids and problems associated with them. This better understanding of the part hybrids can play has led to better defined designs and to further exploration of the suppressed fission concept. The effects of the utilization of the products of the hybrid have also become clearer. Recovery, reprocessing, and fabrication technical issues may be important to a practical fuel cycle, but have not yet been explored in sufficient depth to make a proper evaluation. As the systems themselves become better defined, it will become possible to make judgements on the economic consequences of hybrids and to develop scenarios for their introduction. The next five years could further answer these questions and perhaps determine whether hybrids make not only technical and economic but also practical sense in the mix of technologies available for the production of energy. This probably cannot be done on the current level (~ 1 M$/y) of overall support and unless new emphasis is forthcoming, it may be a long time before the critical issues are resolved.

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4.2 SYNFUELS

4.2.1 Major Approaches

There has been a small program in the U.S. to investigate alternate uses of fusion energy outside of making electricity. One area of investigation has been the production of oxygen and/or hydrogen. The potentially high temperatures associated with fusion have spurred scientists to investigate such concepts but the effort has been minor even compared to hybrids.

There are basically two major approaches to the synfuel process. The design carried out by a team consisting of LLNL and University of Washington is based on thermochemical cycles (1), while the design carried out by BNL and Westinghouse is based on high temperature electrolysis (HTE) (2).

A thermochemical cycle for hydrogen production is a process in which water is used as a feedstock along with a high temperature heat source to produce H_2 and O_2. The water splitting process is accomplished through a closed loop sequence of chemical reaction steps in which the chemical reagents are continuously recycled. Practical thermochemical cycles require input temperatures of ~ 1200 K for the highest temperature chemical step, and operate at a conversion efficiency of ~ 40%. Although there are about 30 thermochemical cycles under various stages of development, only three are tested. Table 4.2-1 lists the principal chemical steps of these three processes. The LLNL study is based on the sulfur-iodine cycle and the simplified flow diagram of this cycle is shown in Fig. 4.2-1.

The HTE process is attractive due to the high conversion efficiency, ~ 50% to 65% depending on operating conditions and power cycle. The basic flow diagram of HTE is shown in Fig. 4.2-2. HTE cells have operated satisfactorily for thousands of hours at ~ 1000°C.

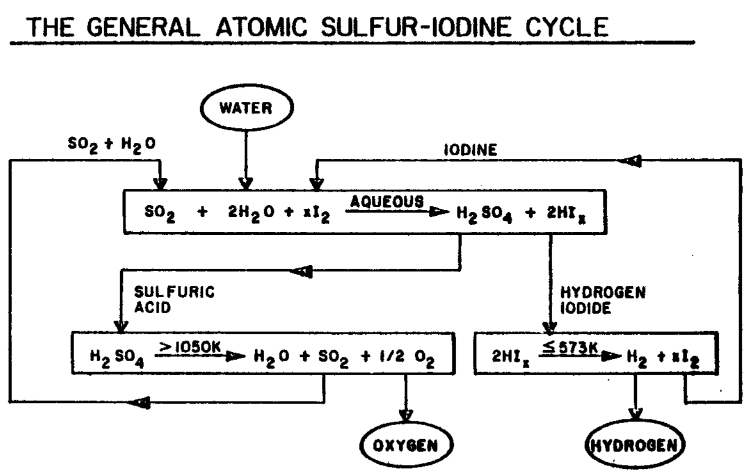

Fig. 4.2-1. The General Atomic Sulfur-Iodine Cycle.

TABLE 4.2-1

Thermochemical Cycles Whose Chemistry and Closed Loop Operation Have Been Verified in the Laboratory

Sulfur-Iodine Cycle

$$2\ H_2O + SO_2 + X\ I_2 \xrightarrow{\text{AQUEOUS}} H_2SO_4 + 2\ HI_X$$

$$2\ HI_X \xrightarrow{\leq 573^0K} X\ I_2 + H_2$$

$$H_2SO_4 \xrightarrow{> 1100^0K} H_2O + SO_2 + 1/2\ O_2$$

Sulfur Cycle (Part Electrochemical)

$$2\ H_2O + SO_2 \xrightarrow[\text{ELECTROLYSIS}]{\text{AQUEOUS}} H_2 + H_2SO_4$$

$$H_2SO_4 \xrightarrow{\text{HIGH T}} H_2O + SO_2 + 1/2\ O_2$$

Sulfur-Bromine Cycle (Part Electrochemical)

$$2\ H_2O + SO_2 + Br_2 \xrightarrow[\text{320-370 K}]{\text{AQUEOUS}} H_2SO_4 + 2\ HBr$$

$$2\ HBr \xrightarrow[\text{ELECTROLYSIS}]{\text{AQUEOUS}} Br_2 + H_2$$

$$H_2SO_4 \xrightarrow{\text{1000-1100 K}} H_2O + SO_2 + 1/2\ O_2$$

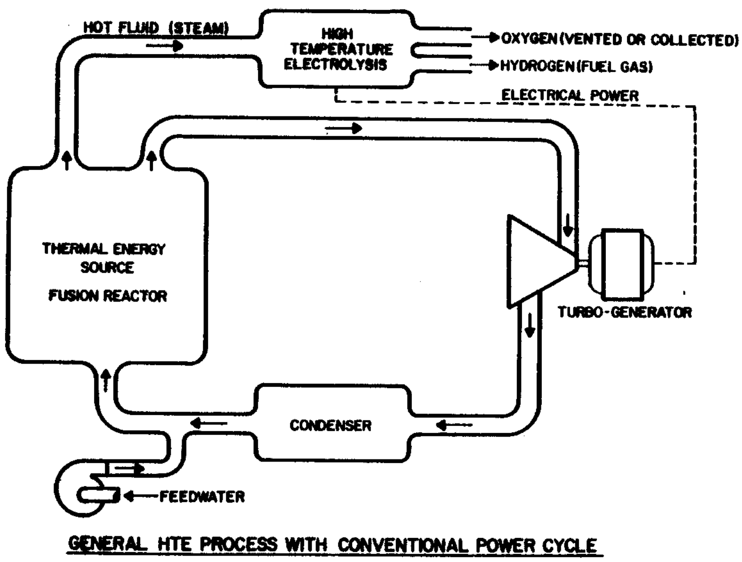

Fig. 4.2-2. General HTE Process with Conventional Power Cycle.

4.2.2 Work Performed Over the Past Five Years

The work associated with the fusion/synfuel process is carried out at LLNL (1) and BNL (2). The major accomplishments of the LLNL design are:

1. Utilized a tandem mirror reactor as a synfuel driver.

2. A cauldron blanket module was designed which includes a NaLi pool. The sodium vaporizes and transfers the energy deposited in the blanket to the heat exchanger in a dome on top of the NaLi pool. This energy is transported to remote processors for the thermochemical production of hydrogen.

3. A flowing microsphere blanket was designed which uses flowing microspheres of Li_2O as the coolant and breeding materials for the reactor. The Li_2O particles can be fed either to a shell and tube heat exchanger or a direct contact heat exchanger to extract the heat.

4. Invention of the Joule-Boosted Decomposer Concept. One of the most significant advances in the synfuels program at LLNL was the introduction of the Joule-Boosted Decomposer concept. This concept uses an electrically heated, commercial SiC furnace element in place of a heat exchanger to obtain the highest temperature in the thermochemical process.

5. Developed a "tritium-free" hydrogen plant.

The major accomplishments of the BNL designs are:

1. BNL has finished a major design study, HYFIRE, using the STARFIRE tokamak design as the driver.

2. In 1982, a high temperature electrolyzer has been designed to operate at 1100°C and 1300°C. The conversion efficiency is ~ 50% to 53%. The estimated synfuel cost is ~ \$6 to \$8 per million BTU.

3. A small electrolyzer has been constructed and operated for 50 hours at 1000°C. The hydrogen production rate is 5 cc/s. There is no major discrepancy from theoretical calculations.

4. Different materials have been tested at ~ 1000°C to 1500°C in a steam or steam/H_2 environment to observe erosion and chemical reaction. Al_2O_3 and ZrO_2 appear to have the best high temperature characteristics.

4.2.3 Status At the End of 1982

The LLNL synfuel group is working with the MARS group on a synfuel process design. The new blanket uses a two zone design in which each zone supplies a parallel stream to the thermochemical plant. The temperatures of the zones are 1200°K and 740°K, respectively. The high temperature zone achieves the SO_3 decomposition step on a four stage fluidized bed decomposer operating at T_{max} on the reactant side of 1100°K.

BNL has finished their conceptual design study, HYFIRE-I (2). The work is continued in more detailed calculations. In particular, Monte Carlo neutronic calculations of the blanket, T_2 permeation calculations, and tritium isolation method from the breeding blanket to the high temperature blanket, and the high temperature recouperator designs are the more critical areas associated with the HTE process.

The future research effort in synfuels is very much in doubt. At the time of this writing most U.S. programs in this area are being phased out.

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4.3 ADVANCED FUELS

4.3.1 Key Issues

The term "advanced fuels" encompasses all fusion fuel cycles other than D-T for a significant portion of their energy (1, 2). In practice, the goal is to find reactions where most of the energy is released in charged particles, thus reducing radioactivity problems associated with neutrons. In addition, the complication of tritium breeding is eliminated and the associated tritium inventory is greatly reduced or eliminated. The difficulty is that advanced fuels have a reduced power density, which must be balanced against simpler, longer life blankets and potential improvements for environmental and safety characteristics. In order to effectively utilize advanced fuels, efficient means of converting the charged particle energy are required. Table 4.3-1 lists a variety of advanced fuel reactions and their reaction energies and Fig. 4.3-1 shows the reaction cross sections for selected advanced fuels.

Radiation is a particularly severe problem for proton-based fuels. Since much higher ion temperatures are required for efficient fusion, collisions tend to give high electron temperatures. This results in high losses to synchrotron and bremsstrahlung radiation. In addition, the multiple reaction branches can lead to high levels of "impurity" radiation. A mitigating factor is that moderate energy reaction products tend to give more energy to ions than to electrons as they slow down. On the other hand, if a very efficient blanket for conversion of high heat loads could be found, having the energy in the form of radiation may not be such a severe constraint. For D-based fuels, the low power density is a more important consideration. High-β systems are therefore very beneficial for raising the power density at a given magnetic field strength.

Since the reaction rates tend to be borderline for economic reactor operation, reaction kinetics is very important. An important consideration here is that the high energy fusion reaction products can have cross sections for nuclear scattering comparable to those for Coulomb scattering. These so-called knock-on nuclear scattering collisions are thought to enhance some reaction rates by as much as 50%. Such calculations are at an early stage of development, but they do give some reason for optimism that the enhancement may be real.

The charged particle kinetic energy from the fusion events must be converted into other forms. In a toroidal system, this can be accomplished by allowing the field to expand under particle pressure. However, such a system is inherently pulsed, which may induce materials problems due to fatigue. In a mirror system, a direct convertor at the ends of the machine can decelerate ions, and collect them on electrodes. This process is very efficient, but heat loads and erosion on the grids can become a problem. In moving ring reactors of the FRM or Spheromak type, it is possible to utilize the magnetic expansion of the exiting plasmas for direct conversion.

An interesting new suggestion is that ions might be spin-polarized and injected into a reactor (3). Theoretical calculations show that the D-T reaction rate may be enhanced about 50%. More pertinent to this section, the

TABLE 4.3-1

Selected Advanced Fuel Reactions

REACTION	ENERGY (MeV)
$D + T \rightarrow n + {}^4He$	17.6
$D + D \rightarrow p + T$	4.0
$\rightarrow n + {}^3He$	3.3
$D + {}^3He \rightarrow p + {}^4He$	18.3
$T + T \rightarrow 2n + {}^4He$	11.3
$\rightarrow D + {}^4He$	14.3
${}^3He + {}^3He \rightarrow 2p + {}^4He$	12.9
$p + {}^6Li \rightarrow {}^3He + {}^4He$	4.0
$D + {}^6Li \rightarrow 2\ {}^4He$	22.4
$\rightarrow p + {}^7Li$	5.0
$\rightarrow n + {}^7Be$	3.4
$\rightarrow p + T + {}^4He$	2.6
$\rightarrow n + {}^3He + {}^4He$	1.8
${}^3He + {}^6Li \rightarrow p + 2\ {}^4He$	16.9
$\rightarrow D + {}^7Be$	0.1
${}^6Li + {}^6Li \rightarrow n + {}^4He + {}^7Be$	1.9
$\rightarrow 3\ {}^4He$	20.8
$\rightarrow D + {}^{10}B$	3.0
$\rightarrow p + {}^{11}B$	12.2
$\rightarrow n + {}^{11}C$	9.5
$p + {}^{11}B \rightarrow 3\ {}^4He$	8.7

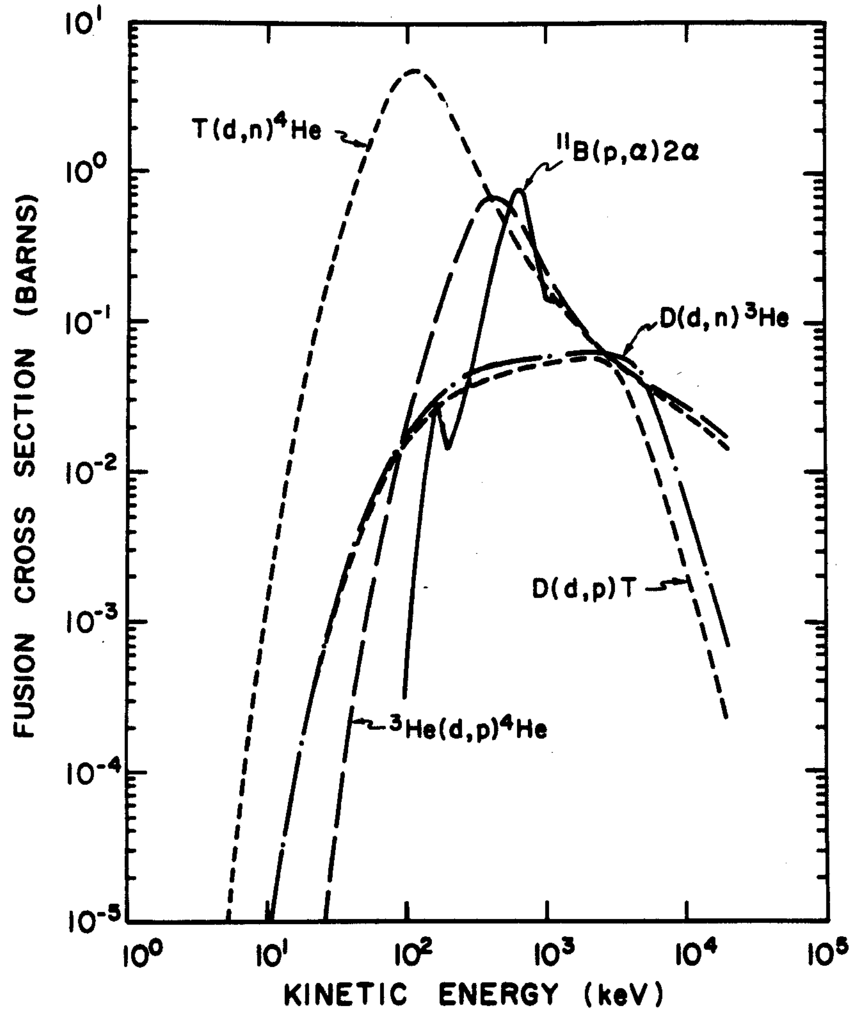

Fig. 4.3-1. The microscopic cross section for several fusion reactions (1 barn = 10^{-24} cm^2).

neutron branch of the D-D reaction may be suppressed, leading to totally neutron-free D-^{3}He neutrons or neutron free D-^{6}Li reactors. Another intriguing feature is the possibility of causing the neutrons to be emitted nonisotropically. This would allow more design flexibility. The polarized sources required appear to be technologically feasible. Unfortunately, present theory indicates that the ions might be quickly depolarized if magnetic fluctuations resonant with the spin precession of the ions are present or the ions recycle off the wall.

4.3.2 Advanced Fuel Reactions

The "mainstream" advanced fuels will be discussed here. However, the data base for reaction cross-sections is by no means complete and more work is required on reaction kinetics, so other reactions should not be totally ruled out as yet.

D-D. The catalyzed D-D reaction, in which all fusion products are assumed to be subsequently burned, has been the subject of the most advanced fuel work so far. The fuel is abundant and tritium need not be bred. However, plasma power density is low; the total neutron flux is comparable to that of D-T and high beta and plasma temperature operation are needed.

D-^{3}He. While this reaction, run lean in D, can be almost neutron free, it suffers from the lack of a good ^{3}He source. The reaction is generally presented as appropriate for a "satellite" reactor near a D-D reactor which would provide the ^{3}He. Suggestions to mine Jupiter's moons for ^{3}He have not yet been taken seriously.

D-^{6}Li. The fuel cycle based on D-^{6}Li is very complex, with many neutrons produced in the D-D and ^{6}Li-^{6}Li branches. The original thought was that the D + ^{6}Li $\rightarrow$ ^{4}He + ^{4}He + 22 MeV branch was dominant, but this is presently believed to be untrue. Because of the multitude of reaction branches, solid numbers for the reaction rate and number of neutrons produced are difficult to obtain.

p-^{6}Li. This reaction has many branches and secondary reactions, but may have a low total neutron yield. Unfortunately, even the "propagating" cycle, where all fusion products are burned, remains doubtful for ignition. Again, many required cross-sections are uncertain.

p-^{11}B. The p + ^{11}B $\rightarrow$ 3 ^{4}He + 8.7 MeV offers prospects for an almost totally neutron-free reactor. However, the reaction rate, even including nuclear elastic scattering, is presently thought to be too low for economic reactor operation and may not even ignite.

4.3.3 Advanced Fuel Reactor Studies

Various fuel cycles and types of reactors have been studied over the past few years, ranging from tokamaks to mirrors. The major ones are described here; Table 4.3-2 contains parameters and a comparison to D-T reactors for the toroidal machines.

WILDCAT (4), designed at Argonne National Laboratory, is a D-D version of the STARFIRE conceptual tokamak rector design. Although WILDCAT presumed more

TABLE 4.3-2

DT and Advanced Fuel Reactor Parameters

PARAMETER	TOKAMAK STARFIRE DT	TOKAMAK WILDCAT DD	RFPR DT	RFPR DD	CRFPR DT	CRFPR DD	TANDEM MIRROR* WITAMIR-I DT	TANDEM MIRROR* SATYR DD
Net Power, MWe	1200	810	750	750	1250	1000	1530	900
Net Plant Efficiency	0.30	~ 0.35	0.25	0.27	0.30	0.25	0.39	0.27
Neutron Wall Loading, MW/m^2								
at 14.1 MeV	3.6	0.55	2.7	0.86	14.5	8.6	2.4	0.35
at 2.45 MeV	---	0.10	---	0.17	---	4.2	---	0.06
Magnetic Field, T	11	14	1.7	4	3.5	9.6	3.6	4.0
Plasma Volume, m^3	781	1887	564	781	45	15	269	919
Plasma Density, cm^{-3}	8.1×10^{13}	2.0×10^{14}	2.1×10^{14}	7.1×10^{14}	3.7×10^{14}	2.8×10^{15}	1.5×10^{14}	2.1×10^{14}

*Tandem mirror parameters are given for the central cell.

optimistic physics and technology than STARFIRE (β = 0.11 vs. 0.07, B = 14.4 T vs. 11.1 T) costs were about a factor of two higher. The main advantages were the lack of tritium breeding, lower tritium inventory and first wall life approaching the plant life. SATYR (5), done by UCLA, is a D-D thermal barrier tandem mirror reactor study. An interesting, axisymmetric coil design was introduced, but the reactor did not compare favorably to a D-T version of the machine. Only the LANL work on a D-D reversed field pinch reactor (RFPR) and a D-D compact reversed field pinch reactor (CRFPR) seems to hold promise for advanced fuels. For these concepts, the plasma power density for D-D can be maintained by increasing magnetic fields without straining magnet technology. Indeed, the D-D CRFPR helps bring neutron wall loads down closer to manageable values.

SAFFIRE (6), a D-^{3}He field reversed mirror (FRM) reactor study by the University of Illinois, examines the possibility of burning the ^{3}He "exhaust" from a DT reactor in a "satellite" FRM reactor. The magnetic field geometry is similar to that of other field-reversed concepts such as CRFPR, and the SAFFIRE study also gives encouraging results. A p-^{11}B octupole reactor study has been done by TRW with EPRI funding. The p-^{11}B fuel cycle was chosen to avoid neutron damage to the internal octupole rings. The reaction rate was found to be insufficient for an economic reactor. Work has also been done on the possibility of using D-D or D-^{3}He pellets with D-T cores in an ICF reactor, the A-FLINT pellet concept, by the University of Illinois.

The elimination of tritium breeding in the blanket can permit more neutron efficient blankets for both hybrids and production of synthetic fuels. These applications are discussed in Sections 4.1 and 4.2.

In addition to the reactor studies, work has been done on details of synchrotron radiation, reaction kinematics, direct convertor design, and reactor systems, all of which also have applicability to D-T fusion.

4.3.4. Status as of the End of 1982

There is presently a lull in advanced fuel activity. Most advanced fuels do not alleviate radioactivity to the extent expected (although they do eliminate the need to breed tritium), and it is difficult to find economic reactor concepts. Although RFPR and CRFPR designs appear promising, there exists only a small experimental data base. In general, considerable innovation still seems to be required in order to make advanced fuels appear competitive with the D-T reaction.

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Section 5

SAFETY AND ENVIRONMENTAL ISSUES

The emergence of any new technology not only produces great benefits to a society but also generates potential new risks to that society. One of the challenges of engineering is to maximize the benefits and minimize the risks within the constraints of economic viability and public acceptance. Fusion power systems used for electrical energy production have the great benefit of a near limitless source of energy for the future. Its major risk is that it produces and utilizes radioactive materials in its operation. The engineering challenge in regard to environmental and safety issues is threefold. First, the level of radioactivity released to the environment during normal operation must be kept within acceptable limits, and second, the probability of an accidental release of radioactivity to the environment due to plant transients or off-normal events must be kept within acceptable bounds and the consequences assessed. Third, fusion power must accomplish the other two objectives in an economically competitive manner because it is not the only limitless energy source. This section discusses what has happened in both of these areas during the past few years; i.e. the environmental risks during normal operation and the safety concerns in the event of an accident. In order to put this topic in perspective, a brief review of fusion reactor safety philosophy is presented first.

5.1 FUSION SAFETY

Historically, reactor safety has been based on the concept of preventing the release of radioactive material to the environment following an unanticipated transient or accident. The method of prevention for current light water reactors (LWR) and fast breeder reactors has been a defense in depth; i.e. multiple containment barriers between the radioactive materials and the environment. For a LWR the primary source of radioactivity resides in the fuel, and defense in depth involves: (1) the fuel rod cladding and assuring its integrity during operation, (2) the primary coolant system and preventing leaks, and (3) the containment building and maintaining the integrity during an accident. For a fusion energy system the concept of defense in depth could be applied; however, the nature, quantity and location of the radioactive materials is different.

In the LWR the primary sources of radioactivity are the fuel and the associated products produced in the fission process which reside in the zirconium clad fuel rods (e.g., 20 billion curies of radioactivity at shutdown for an

1100 MWe plant). The amount of radioactivity present in the surrounding structure, caused by neutron activation, and in the coolant, caused by corrosion product activation, is insignificant compared to this residual radioactivity in the fuel (more than two orders of magnitude smaller).

In the present designs for a fusion power plant the only radioactive "fuel" is the tritium which is in the plasma and the associated fueling systems. The amount of tritium present in the fusion reactor actually represents a small amount of radioactivity; i.e. relatively insignificant compared to a LWR (~ 1 kg of tritium or 10 million curies, a factor of 1000 less than the activated structure). The tritium is enclosed within the vacuum boundary, and this with the outer containment building comprises the multiple barriers of isolation from the environment. Because the neutrons produced by the fusion reaction are so energetic (14.3 MeV) the largest amount of radioactivity in the fusion system is found in the surrounding structure (blanket, reflector, and shield) where neutron activation induces radioactivity. The amount of radioactivity is large compared to that in a LWR surrounding structure, but still lower than that from the fuel and fission products in a LWR (e.g., for the WITAMIR reactor design, (1) about 2 billion curies at shutdown). The structural material activated is within the vacuum boundary for the fusion reactor (i.e., blanket and reflector) and also makes up the vacuum boundary (shield). One exception to this generalization is that ICF reactors with liquid metal "first walls" may have considerably reduced structural radioactivity. The amount of radioactive corrosion products is approximately similar to that in a LWR. There are important differences in the amount and location of radioactivity, and this causes the safety concerns in fusion to be different. One should note that spent fuel for a LWR and tritium fuel storage for a fusion design are not considered here. Both of these radioactive source terms are of some safety concern, but probably would not directly affect the design of the reactor and its containment.

A second consideration that one must remember is that in current LWR's the efforts in reactor safety for reducing the probability of an accident or for mitigating its consequences are being performed for very specific reactor designs. In a sense, the safety investigations are confirmatory in nature where specific designs are evaluated with respect to accidents and modifications in the design are made if needed.

In sharp contrast to this, safety research for a fusion power system can only focus its attention on generic issues, because the detailed fusion reactor designs are not available. In a sense this is quite beneficial because a broader design approach to safety could be undertaken. In this approach the fusion power system would have a set of goals for reliability--e.g. the fusion device will be designed to have an availability of 70-80%; and for safety--e.g. the probability of an accident involving a loss of the coolable geometry and the subsequent release of a significant fraction of its radioactivity will be designed to be less than 10^{-6} per reactor per year. Then, as part of the plant design, system reliability can be increased, probable accident paths can be identified, and consequences of fusion power system accidents can be assessed. Current trends in fusion reactor safety seem to stress the assessment of consequences more than improvement of the reactor design from a reliability standpoint.

5.1.1 Major Potential Problems

In delineating the major generic safety concerns for a fusion power plant, steady-state operation and startup and shutdown of the system are excluded. The focus of the discussion will be on unanticipated transients and accidents that may lead to a release of radioactivity to the environment. If one accepts a safety design goal of a minimal release of radioactivity (e.g., the stated goal of 10^{-6} per reactor per year) then the approach is to design a system which minimizes the probability of a loss of the coolable geometry for the reactor, and investigates the system interactions and physical processes that may be involved in causing a radioactivity release. Calculation of accident consequences fits into this framework primarily in the sense of assessing the impact of various radioactive source terms on the environment.

A possible list of these system interactions or physical processes is given below:

1. Power-cooling mismatch when the fusion power system is still at some level of power, e.g. a loss of flow accident while at power;

2. Power-cooling mismatch when the fusion power system is shut down, e.g. a loss of heat sink and failure to remove the decay heat;

3. Local quenching of the plasma on the first wall, e.g. a plasma disruption event;

4. Physical processes which can lead to overpressurization of containment by combustion or pressurization, e.g. lithium-water reactions in the blanket or coolant causing local heating and hydrogen generation; loss of helium coolant in the magnets;

5. Power-cooling mismatch in the magnet systems, e.g. a magnet quench or loss of cryogenic heat sink;

6. External events, e.g. earthquakes or tornadoes;

7. Operational errors.

This list, although not complete, gives an overall picture of what one should consider in the design of the fusion power plant system. Some of the issues given above have a greater impact on the design and provide redundant systems to assure a high reliability in reactor shutdown, blanket and reflector cooling, and magnet stability. Others are physical processes that should be understood in terms of their impact on how containment integrity might be threatened by possible chemical and physical reactions. In this sense the design would be affected by the choice of materials one makes for the blanket, structure, and coolant, and on what loads are to be designed for in the containment and pipe support structure. Figure 5.1-1 presents a description of some of the safety concerns in each area.

5.1.2 Previous Work in Fusion Reactor Safety

Early work in fusion safety (2) considered a number of first generation and second generation fusion reactor designs (i.e., UWMAK-I, II, III, EPR) in order to identify potential safety hazards, significant accident sequences, and major safety-related information gaps. This work identified potential generic hazards in the areas of tritium release, transport of induced radioactivity in the blanket/reflector, liquid-metal interactions, and magnet design. For most of the designs three generic three generic accident sequences were identified; the power-flow mismatch, exothermic chemical reactions in the reactor system--or containment, and magnet quench.

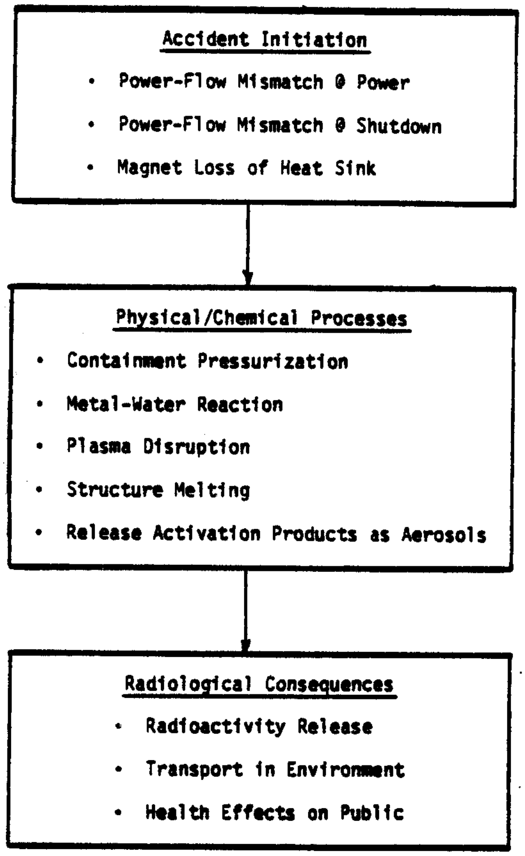

Fig. 5.1-1. Major Issues in Safety and Their Interrelation.

Subsequent to this study, the efforts on fusion reactor safety were consolidated, and in 1979 the EG&G Idaho organization was designated the lead laboratory in fusion safety. Since that time research efforts have been centered at Idaho with a number of research institutions participating in the work (e.g., ANL, HEDL, MIT, UCLA). To briefly summarize the work over the last three years (1980-82) four subject areas are discussed: (1) thermal-hydraulics analyses, assessment.

5.1.2.1 Thermal-Hydraulics. Thermal-hydraulic analysis of a fusion reactor system provides quantitative data concerning flow and temperature variations in the blanket/reflector regions during an unanticipated transient or accident, and helps scope out the resultant progress of an accident. During the last three years an advanced thermal-hydraulics analysis computer program (ATHENA) was developed at EG&G Idaho (3-5). It is based on the LWR computer program RELAP5, with the addition of a plasma kinetics model. It incorporates a modular structure that permits different fusion kinetics models to be used

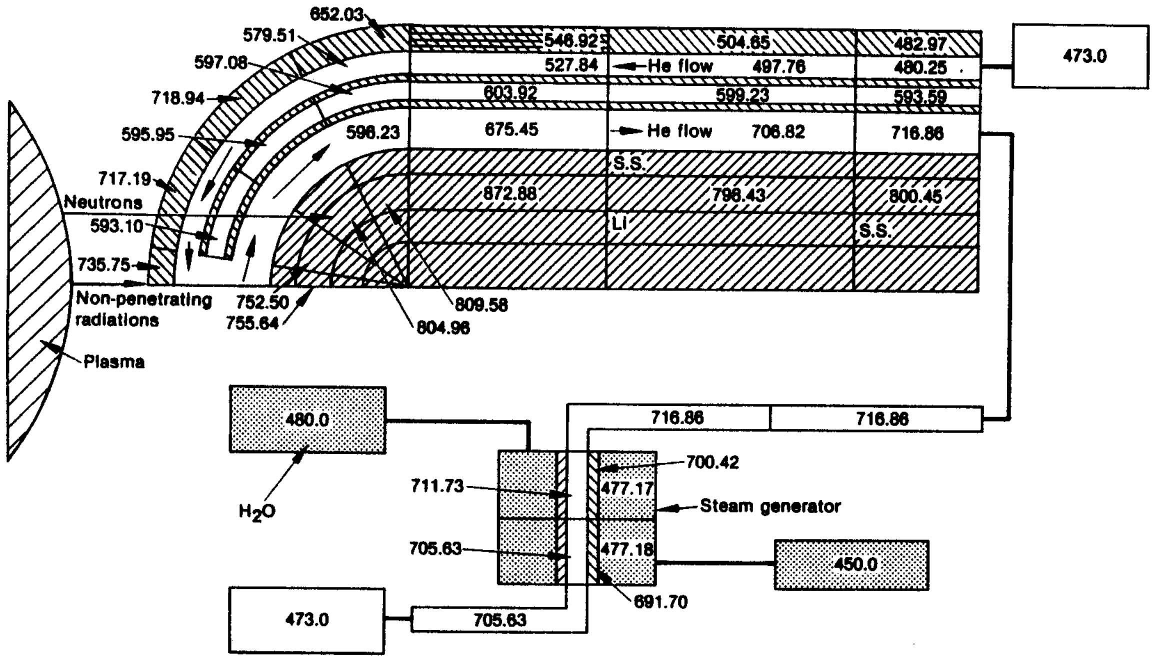

Fig. 5.1-2. Nodes and Steady-State Temperatures (K) Used in Model.

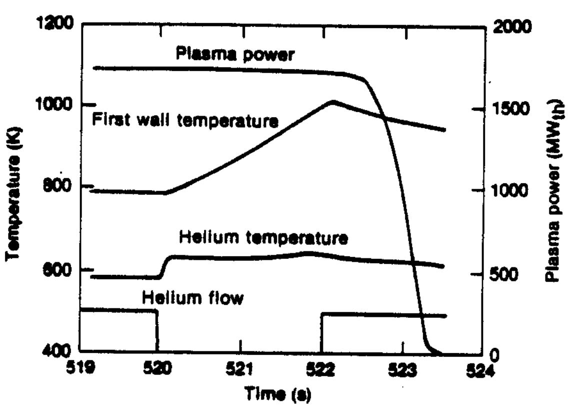

Fig. 5.1-3a. Two-Second Coolant Interruption.

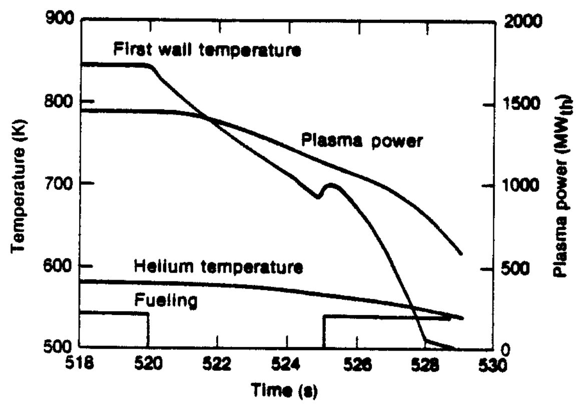

Fig. 5.1-3b. Five-Second Fueling Interruption.

for a tokamak, mirror or other plasma confinement concept, different blanket designs and materials, and variable flow paths. The thermal-hydraulics model allows for multiphase flow, transient response to energy inputs and nonequilibrium effects. A number of transients have been analyzed with sample results shown in Figs. 5.1-2 and 5.1-3. The blanket design analyzed was high-pressure helium flowing through a blanket module composed of stainless steel and a solid lithium breeder (Fig. 5.1-2). In Fig. 5.1-3a, the helium coolant flow to the blanket is stopped for two seconds; as the first wall temperature increases the plasma becomes polluted with additional impurities, and, after a few seconds, it cannot remain ignited without more auxiliary heating. In Fig. 5.1-3b fueling was stopped for two seconds. While the fuel is off, plasma power gradually decreases. The lack of fueling also causes the first wall and helium temperatures to decrease. When the fuel is restored, plasma power increases briefly, but the large quantity of cold fuel injected into the plasma at one time causes the plasma to cool and fail to remain ignited. Work is continuing using this computer model and other heat transfer models to look at thermal-hydraulic transient response during fusion reactor transients.

5.1.2.2 Materials Compatibility. Liquid lithium, molten lithium alloys, or lithium chemical compounds are possible candidates for both blanket and coolant materials because of its properties and neutron absorption/tritium-breeding characteristics. For liquid lithium, the areas of concern are lithium-atmosphere interactions (N_2 or O_2), lithium-water interactions, and lithium-concrete interactions. Previously small-scale experiments in these areas (2-5) have been conducted by Hanford Engineering Development Laboratory (HEDL). Analysis of the lithium-atmosphere interactions (either in a pool or spray) have been done using the LITFIRE computer program developed at MIT (6). Currently the experimental efforts have been broadened in scope to cover possible blanket/coolant alternative materials; i.e. lithium alloys and lithium chemical compounds interacting with the atmosphere, water or concrete.

The liquid lithium spill experiments were begun in late 1977. They consisted of a lithium pool or a spray (~ 10 kg) being introduced into a 14 m^3 test vessel with an atmosphere of air or nitrogen. In addition lithium was deposited on concrete to measure its chemical reactivity with this construction material. Current results of these scoping tests are presented in Refs. (4) and (5) and compared to sodium. Elemental lithium is much more reactive than sodium. Scoping tests performed by Argonne National Laboratory for DOE dropping liquid lithium into water indicated that the lithium underwent rapid oxidation and the hydrogen produced ignited and resulted in a chemical explosion.

Current alternative blanket/breeder material candidates include lithium oxide, lithium aluminate, lithium silicate, lithium zirconate, and lead-lithium alloys ranging from $Li_{77}Pb_{23}$ to $Li_{17}Pb_{83}$. The coolant often used in conjunction with these materials is water. HEDL is now performing scoping experiments to investigate the compatibility of these materials with water. Two types of tests are being performed: 5 g of blanket material at 600°C added to an excess of water at 90°C, and 1 g of water at 98°C added to an excess of blanket material at 600°C. The preliminary results of these tests suggest that lithium aluminate, silicate and zirconate are quite compatible with water. Hydrogen is released at a very modest rate for $Li_{17}Pb_{83}$ and lithium oxide. More vigorous interactions were observed for $Li_{77}Pb_{23}$. The results suggest it is prudent to avoid lithium and $Li_{77}Pb_{23}$ alloy as a blanket material to prevent large energy and hydrogen releases under accident conditions. Perhaps the most important result of this work is the demonstration that liquid $Li_{17}Pb_{83}$ coolants represent a relatively small hazard compard to liquid Li.

5.1.2.3 Magnet Safety. To ensure, through computation and experiment, that a large superconducting magnet is cryostable is an important part of magnet design. It is important to know the limits on the growth of the normal region of the superconductor under accident conditions. In this area of magnet safety, ANL and MIT have begun analytical and experimental research programs to address these technical issues:

1. Energy deposition and flow--e.g. quench and short circuit effects on magnet temperature, or coil or terminal failure by burnout.

2. Quench detection and discrimination.

3. Indication of impending failure.

At this time an experimental program is just beginning to investigate various physical effects, e.g. coupled circuit discharge characteristics with and without quench or short circuit. Also a computer program TASS has been written by ANL to calculate the behavior of a normal region in a superconducting magnet with pool-boiling cooling. In using the code, the thermal, electrical, and geometrical properties of the magnet are first specified. Then either an initial temperature distribution or an initial distribution of heating for a specified time period are defined. The program then steps through time, and determines whether the normal region grows or collapses for the given current. It is found that if the initial heat or temperature pulse is localized sufficiently in time (< 10 ms) and position (< 1 m), then only the total energy deposited in the conductor influences the subsequent behavior of the normal region. In the limited opportunities to date for comparing the stability calculations with experiments, the agreement has been fair.

5.1.2.4 Plasma Disruptions. A plasma disruption can occur in a tokamak fusion reactor when instabilities cause the plasma to contact the first wall or a limiter (or divertor). These disruptions can potentially limit the life of these plasma confinement barriers. The surfaces of these components are subject to melting and/or vaporization during the disruption. Currents induced by this event can cause forces and torques within the structural components that may be of a safety concern during operation. In order to design against such occurrences studies have begun (e.g., Ref. (8)) which attempt to describe the response of the structure to a disruption event.

In the INTOR study (8), for example, the plasma disruption event was analyzed in two phases; first the thermal response of the wall material was calculated for a range of disruption conditions and second the hydrodynamic stability of any melt layer produced by the disruption was analyzed. The primary disruption parameters were the energy deposition per unit area, the disruption time, the frequency of disruptions and the wall material and its initial conditions.

The vaporization and melting characteristics of candidate surface materials have been evaluated for a range of disruption conditions (energies of 170 to

270 J/cm^2 and times of 5 to 20 ms). The materials examined were the refractory compounds (SiC, TiC, and BeO), the metals (Be, W, Mo) and stainless steel along with graphite. For example, calculations indicated that BeO exhibits the greatest vaporization and melting losses while SiC exhibits the least. In general, all of these refractory materials exhibited lower vaporization and melting than the low- and medium-Z metals. The INTOR study chose a rather optimistic (low) energy deposition level and deposition time. If one allows for higher peaking factors (producing up to 1000 J/cm^2) and shorter deposition times (~ 1 ms), then vaporization and melting become severe problems.

The primary result of the INTOR melt layer stability analysis was that for the reference conditions (270 J/cm^2, 20 ms), the melt layer is predicted to be stable on the limiter. However, for shorter disruption times and higher energy densities, instabilities were predicted, and, therefore, melt layer loss remains a concern.

5.1.2.5 Fusion Risk Assessment. An actual risk assessment of a fusion power system would require detailed system designs from which the possible accident sequences and their probabilities could be determined. Since the current designs for large fusion power plants are conceptual in nature, determination of a comprehensive set of reactor accident sequences and probabilities may not be feasible. EG&G and MIT have investigated an alternative approach to this problem of hazard evaluation where the risk requirements for fusion reactors are found by requiring that the risk associated with fusion reactor designs be less than that from current LWR systems.

Previous work in this area by Kazimi (7) has taken the approach of focusing on selected conceptual fusion designs, like UWMAK-I or UWMAK-III, and determining the health and economic consequences of severe reactor accidents in a manner similar to that in WASH-1400, the Reactor Safety Study. From these analyses and a comparison of the consequences to those in a LWR the required probability of the radioactive source term was found. This could be then used to specify the necessary reliability requirements of specific fusion systems to keep the risk within acceptable bounds.

Some of the conclusions from this work are:

1. The reliability requirements for the blanket/reflector region of the reactor are highly dependent on the design and the materials used (5). For example, the use of material with a high resistance to oxidation and/or low potential for activation is important.

2. The consequences of a massive tritium release to the environment (e.g., 10^7 curies) is not as important as the release of structural activation products (as little as 1%).

3. For fusion reactor designs to have an equivalent risk to current LWR designs from severe accidents imposes overall system reliability requirements that are an order of magnitude smaller than those calculated in WASH-1400. This is because

the severe accident source terms are inherently much smaller for a fusion reactor.

REFERENCES FOR SECTION 5.1

1. B. Badger et al., WITAMIR-I, A Tandem Mirror Reactor Study, UWFDM-400, University of Wisconsin (September 1980).

2. W.E. Kastenberg et al., Safety of Fusion Reactors, PPG-342, UCLA (October 1977). See also EPRI-ER-546, -547, -548, July 1978.

3. J.G. Crocker et al., Fusion Reactor Safety Research Program Annual Report, FY-79, EGG-2018, EG&G Idaho (August 1980).

4. J.G. Crocker et al., Fusion Reactor Safety Research Program, FY-80, EGG-2106, EG&G Idaho (June 1981).

5. J.G. Crocker et al., Fusion Reactor Safety Research Program, FY-81, EGG-2205, EG&G Idaho (July 1982).

6. S.J. Piet et al., Potential Consequences of Tokamak Fusion Reactor Accidents; The Materials Impact, PFC/RR-82-19, MIT Plasma Fusion Center (June 1982).

7. M.S. Kazimi et al., "Radiological Aspects of Fusion Reactor Safety: Risk Constraints in Severe Accidents," J. of Fusion Energy, VI, No. 1, p. 87 (Jan. 1981).

8. W.M. Stacey et al., U.S. FED-INTOR Activity, Critical Issues, USA-FED-INTOR/82-1 (October 1982).

5.2 ENVIRONMENTAL IMPACT

The environmental issues associated with fusion reactors have been recognized at least in a qualitative way almost since the first proposals for fusion power plants were made and the perceptions of them were often cited as part of the rationale for the development of fusion power. These issues have had an impact on the conceptual designs that have been made. Typically the designers might try to determine or reduce source terms, e.g. determining the radioactivity in the structure or reducing the radioactivity by the use of low activation materials, to design and simplify maintenance activities and thus reduce occupational radiation exposure, or to propose the use of specific materials in quantities that do not have an adverse effect on available resources. These efforts have resulted in systems that qualitatively might have acceptable environmental impact.

No complete assessment of the environmental impact of fusion power plants or of any particular design concept has been published although aspects of the problem have been considered and reported. At the present time (January 1983) the U.S. Department of Energy has contracted with the Fusion Environmental Assessment Program at the Oak Ridge National Laboratory for a generic environmental impact statement on magnetic confined fusion. This study will deal with one particular design concept (STARFIRE) but enough alternatives will be considered to make it applicable to most magnetic fusion systems.

While it is not an extensive assessment, considerable attention was given to environmental concerns in the INTOR workshops (1, 2). The INTOR design concept (see Section 2.3.1.2) is a relatively low fusion power device (~ 600 MW) and is not a power reactor, but many of the environmental effects are similar in kind if not magnitude to those of a commercial system. Much of this section is based on extrapolations from the INTOR work (2).

The effects of fusion power plants on the environment come about from: (1) the exposure of the plant staff and the general public to small amounts of tritium and other radionuclides, (2) the disposal of radioactive components and wastes, (3) the presence of intense magnetic fields, (4) the use of resources of limited availability, and (5) all the effects common to any large power plant based on a Rankine or similar thermodynamic cycle.

These impacts are inherent to fusion systems based on the deuterium-tritium reaction in a magnetically confined plasma. They are due to the presence of 14.1 MeV neutrons from the reaction which interact with the surrounding structure, leading to induced radioactivity, and deposit their energy over a relatively large volume resulting in a low power density. At the same time, kilogram quantities of tritium must be bred, extracted and processed to provide a continuing source of fuel. The plasma must be confined which leads to large superconducting magnets, which use scarce materials (e.g. Nb), must be cooled by liquid helium, and generate stray magnetic fields. Components near the plasma are subject to a harsh environment (high neutron fluences, surface bombardment, high temperatures) and will have to be replaced at regular intervals. These basic factors will be present in any system and thus all will have similar impact. The actual magnitude of the impact will depend on which of the many reactor concepts, i.e. tokamak, mirror, etc., is used and the way its use is implemented through a particular design.

5.2.1 Radiation Impact

The presence of large quantities of tritium which must undergo continual processing combined with its physical and chemical properties lead to containment and contamination problems which affect both the plant work force and the public. Consideration of dose to the public has led to design goals for fusion plants of routine tritium releases in the range of 10-20 Ci/d. An analysis of the release of 5000 Ci/yr from a 100 m stack yields a committed dose (50 year integration time) an order of magnitude less than the 5 mrem/yr used in many radiation guidelines. A ground level release could result in a similar dose at distances greater than 80 m. The analysis was extended to find that the global dose commitment was 10 man-rem per year of operation.

Since tritium is likely to be the most mobile radioactive nuclide in the system and therefore may be most susceptible to release in accident conditions, calculations have been reported for large releases. The INTOR calculation indicates that the release of 1 kg of tritium from a 100 m stack results in a committed dose of less than the 25 rem limit for a once-in-a-lifetime occurrence for distances greater than 500 m from the release point.

These calculations are based on assumed release rates. They therefore are in the nature of design goals rather than expected releases from a given system. However, they are representative of what is felt may be obtained and indicate that exposure to tritium can result in an acceptable impact.

The exposure to other nuclides released during routine operation and accident situations is more difficult to quantify. The quantity and species of nuclei released depend on the specific design. For example the releases from a system involving water and 316 stainless steel are likely to be quite different from one incorporating a liquid lead-lithium alloy and a ferritic steel such as HT-9 and different still from a system designed for low residual activity. As an example of what might be expected, the INTOR analysis indicated that the dose due to other nuclides was a fraction of that anticipated for the release of tritium. Thus it may be anticipated that the release of radioactivity from a fusion plant is probably acceptable.

The waste products from a fusion reactor must also be considered. The wastes are of two general types. Low level wet and dry solid wastes would be generated in quantity and specific activity comparable to those generated in current light water reactors. These wastes can be stored and disposed using current techniques and present no special problems. The high level wastes generated by blanket replacement present some of the same difficulties as do high level wastes from light water reactors. Disposal problems may be easier since the wastes are in a stable form to start with (corrosion resistant metals) and contain no fission products or actinides. Present practices and regulations should be adequate for handling and disposal in an acceptable manner. Again the problem may be made much more tractable with the choice of suitable low activation materials in the design.

5.2.2 Magnetic Field Impact

The magnetic fields surrounding the reactor become small as the distance from the magnets is increased. The fields anticipated outside the reactor building are in the range of 2-5 x 10^{-5} tesla. The field in the control room might be ~ 10^{-4} tesla, achieved by either using local shielding or by establishing exclusion areas. The health effects of magnetic fields are not well understood; however, interim exposure guidelines have been established which are met at the field strengths indicated above. Radiofrequency magnetic fields from plasma heating devices would also be contained and shielded in INTOR so that personnel exposure would be below guideline levels.

5.2.3 Critical Material Usage

Many of the materials which would be used in a fusion system are similar to those used in any large central power station - concrete, steel, copper, etc. The quantities used for the nuclear steam supply system might be somewhat larger because of the lower power density but the basic concerns would remain the same. However, certain materials are specific to fusion systems and require further consideration. These materials are: deuterium, tritium, lithium (fusion fuel and fertile material) and helium and niobium (magnets).

Deuterium is in plentiful supply since it occurs in water at a concentration of about 150 ppm. Facilities currently exist to supply it for heavy water reactors. Tritium is not available in nature in useful amounts. Startup quantities are thought to be available from existing production facilities or heavy water reactors. Subsequently tritium would be self-generated from the fusion reactor blanket. The supply of lithium seems to be adequate although additional production facilities would have to be built for a deployed fusion economy.

Liquid helium (4.2 K) is required for cooling the superconducting magnets. The present supply is adequate but shortages may exist in the future unless the program of stockpiling helium from natural gas is not reestablished. Helium is present in the atmosphere at about 0.65 ppm and could be recovered at a cost significantly greater than at present.

The superconducting material in the magnets requires niobium. There is no acceptable substitute at present and almost all of the U.S. requirements are met by import. Thus while the worldwide supply appears to be adequate it is subject to economic and political pressures by the suppliers. Domestic reserves exist in the U.S. and extraction capability could be developed to mitigate these potential external influences.

There is a general resource problem with respect to materials such as manganese, chromium, cobalt, etc., which are obtained from foreign sources. However, these problems exist for any technology which uses these elements in structural alloys and fusion would represent only a small fraction of those requirements.

This summary refers specifically to magnetic fusion. Nothing equivalent has been done for inertial fusion. However, in general the same conclusions are likely to be reached. Certain impacts will be somewhat reduced, e.g. magnetic effects. Others may be increased, e.g. the land area required for some of the heavy-ion designs (HIBALL) are rather larger than envisioned for magnetic devices. Radiological impacts are expected to remain about the same.

REFERENCE FOR SECTION 5.2

1. International Tokamak Reactor, Zero Phase, STI/PUB/556, International Atomic Energy Agency, Vienna (1980).

2. International Tokamak Reactor, Phase One, STI/PUB/619, International Atomic Energy Agency, Vienna (1982).

Section 6

MAJOR PROGRAMMATIC REVIEWS OF FUSION RESEARCH

The time period just before the 1978-82 scope of this review was characterized by a general optimism and "let's get on with it" attitude that has not been duplicated in the past 5 years. For example, the U.S. fusion office in Washington was operating, in 1977, under a program that called for the operation of the following facilities:

Experimental Power Reactor-I (10 MW Electric)	1986
Experimental Power Reactor-II (100 MW Electric)	1989
Demonstration Power Plant (500 MW Electric)	1998

Equally optimistic programs were being proposed in Japan and the Soviet Union at that time. Because of the financial implications of embarking on such an ambitious program, there have been at least seven major reviews of the U.S. and European fusion programs in the 1978-82 period (Table 6.1-1). These reviews have generally concentrated on the magnetic fusion aspects but, in at least one case, some consideration was given to the inertial confinement program of the U.S. In the U.S., several of the reviews paved the way for the passage of the most aggressive piece of fusion legislation ever proposed, the Magnetic Fusion Engineering Act of 1980. We will summarize each of the reviews in chronological order.

6.1 FOSTER PANEL

On February 27, 1978, Dr. John M. Deutch, the Director of Energy Research for DOE, requested that Dr. John S. Foster, Jr. head a panel of scientists to review both the magnetic and inertial confinement fusion programs. Their findings are summarized in a DOE report dated June 1978 (1) and some of the major conclusions are listed below.

TABLE 6.1-1

Summary of Major Fusion Program Reviews 1978-82

Country	Date of Review	"Name" of Review	Remarks
U.S.	1978	Ad Hoc Experts Group on Fusion (Foster Panel)	Magnet and Inertial
U.S.	1979	Atomic Industrial Forum Committee on Fusion	Magnetic only
U.S.	1979-80	Advisory Panel on Fusion Energy (Hirsch Panel)	For U.S. House of Representatives, magnetic fusion only.
IAEA	1979	INTOR Fusion Experts Committee	For World Tokamak Fusion Program
U.S.	1980	Fusion Review Panel of the Energy Research Advisory Board (Buchsbaum Panel)	Magnetic Fusion Only
U.S.	1980	Magnetic Fusion Engineering Act[a]	Legislative Directive
Euratom	1981	European Fusion Review Panel (Beckurts Panel)	European Magnetic Fusion Program Only
U.S.	1982-?[b]	Magnetic Fusion Advisory Committee (MFAC)	Five panels on magnetic fusion only. a. Tokamaks and mirrors. b. Alternate concepts. c. Upgrades of TFTR. d. Upgrades of MFTF-B. e. University role in fusion.

(a) Not a Review but a Major Statement on Fusion.

(b) The MFAC activity presumably will extend to at least FY-84.

1. Program Objectives

"The objective of the programs should be to determine the highest potential for a commercial fusion energy source at the earliest practical date."

"Demonstration of scientific and technological feasibility should remain the near term aim of the program. Its achievement should be a necessary, but not sufficient, step in the decision to proceed with the construction of an engineering prototype reactor. That decision should include the evaluation of the suitability of the various contending approaches for a reactor as well as the attainment of required technology."

2. Strategy

"Adopt a modification of strategy that reduces the risk of broadening and strengthening the technical base from which to choose the best fusion approach to practical energy production."

"Pursue vigorously several physics approaches and carry out in parallel, engineering and materials test programs until at least one potentially economic competitive design is identified."

3. Magnetic Confinement

"Obtain a thermonuclear burn in tokamaks as quickly as practical. However, commitment to construction of a next generation tokamak beyond TFTR, should not be made until results from TFTR and other related experiments justify it."

4. Inertial Confinement

"Pursue the development of alternate drivers which offer the potential for achieving performance parameters required for eventual commerical use taking into account target coupling."

"Review the current classification policy, its impact on the program, and the protection of information which should be classified."

5. Management

"Implement a coordinated management of the MFE and ICF programs."

"Expand and evolve appropriately the participation of universities, industry, and users as the program develops, e.g., in the areas of:

drivers/heaters,

system engineering,

alternate concepts/drivers, and

physics investigations."

6. Areas of Further Study

"Determine the role of the fuel producing fusion hybrid in the overall fusion program."

"Determine the impact and potential of international cooperation on the development of the U.S. fusion program and recommend appropriate actions."

The conclusions of the Foster Panel were widely interpreted as a call to diversify the magnetic fusion program away from such heavy emphasis on the tokamaks and laser approaches and to revise the governmental management structure to lay the base for a transition to engineering development. The Department of Energy used the conclusions of the Foster Panel to set a new schedule for commercializing fusion as outlined below (published in September 1978) (2):

Engineering Test Facility	1992-5 (magnetic) 1995-8 (inertial)
Experimental Power Reactor	2005
Demonstration Power Plant	2015

REFERENCES FOR SECTION 6.1

1. Final Report of the Ad Hoc Experts Group on Fusion, U.S. Dept. of Energy, Washington, DC, June 1978, DOE/ER-0008.

2. The Department of Energy Policy for Fusion Energy, U.S. Dept. of Energy, Washington, DC, Sept. 1978, DOE/ER-0018.

6.2 ATOMIC INDUSTRIAL FORUM (AIF) COMMITTEE ON FUSION

A committee of industrial scientists and managers were requested by the AIF in 1979 to review the U.S. fusion program particularly with respect to how industry might play a more meaningful role. The committee concentrated on magnetic fusion and in general was quite critical of the manner in which the U.S. DOE was conducting the program. It also disagreed with the Foster Panel's call for "defocussing" the program and preferred instead to concentrate the program on the lead concept, the tokamak. A few of the more notable recommendations and findings as taken from the Dec. 13, 1979 report (1) are given below:

> *"A national goal be established aimed at the construction and operation of a fusion energy facility producing net power before the end of the century."*
>
> *"That in furtherance of this goal the government move forward now with a program, identified as a line item in the Federal budget, for the site selection, design, and construction of an Engineering Test Facility (ETF)."*
>
> *"That specific government funding be earmarked for industrial participation in the fusion program and that this funding be insulated from encroachment by other program demands."*

The committee also had some observations on the DOE management policies.

> *"In effect, existing policies and procedures have biased governmental decisions toward performing work in-house, thus undercutting the growth of a healthy industrial base and discouraging the commitment of industrial resources to the program."*
>
> *"The lead laboratory concept of project management which has been adopted by DOE tends to discourage industrial involvement since the national laboratories are often placed in a competitive position with industry on the disposition of funding for R & D projects. In such circumstances, it is inappropriate for the government to ask the laboratory to engage in peer review of industrial proposals."*

In summary, the AIF committee seemed to be calling for an accelerated, focused program in which they could effectively compete and to which they could contribute in a meaningful way.

REFERENCE FOR SECTION 6.2

1. Fusion Energy at the Crossroads: Role of the Private Sector, Atomic Industrial Forum, Inc., Washington, DC, December 31, 1979.

6.3 FUSION ADVISORY PANEL (HIRSCH PANEL)

In July 1979, Congressman Mike McCormack, Chairman of the Energy Research and Production Subcommittee of the House Science and Technology Commmittee, formed a Fusion Advisory Panel, chaired by Dr. Robert L. Hirsch. This panel was charged with reviewing the technical and engineering credibility of fusion and to assess whether a slippage in the target date for a fusion demonstration plant (1) was necessitated by technical difficulties or by funding limitations.

After several meetings the Panel concluded that (2):

> *"The magnetic confinement program has reached, and in many cases surpassed, the goals publicly set forth in past years. Magnetic fusion research has consistently been on schedule and very close to cost, even during recent inflationary times."*
>
> *"...magnetic fusion program is without a doubt ready to proceed much more aggressively than presently projected by DOE."*
>
> *"...electric power from fusion should be attainable before the turn of the century [and] the total programmatic cost for an accelerated program will be lower than for the present stretched out schedule."*

The Panel recommended (in July 1979) that DOE be requested to prepare an accelerated program plan. By December 1981, DOE had prepared such a plan with milestones as listed below.

Plan	FETF Operation	DEMO Operation	Total Cost (billions)
Base Program	1995	2010	$14.3
H.S. & T. 2000	1990	2000	$11.9
H.S. & T. 1995	1988	1995	$12.1

FETF - Fusion Engineering Test Facility.
H.S. & T. - House Science and Technology Committee

DOE officials acknowledged that if the current "base" plan of that time were adopted (400 million dollars per year for the 1982-84 time period) it would be at least 2010 before a demo will be on-line and 2023 before a "significant" amount of energy would be generated by fusion. Increasing the funding to 585 million dollars per year (1982-84) would move the DEMO operation date to 2000 and that a "crash" program, essentially doubling the budget of 1979 to 870 million dollars per year, would get a DEMO by the year 1995. Furthermore, it appeared that the program producing the DEMO by the year 2000 is the most economical plan, saving nearly 2.5 billion dollars.

The Hirsch Panel report led directly to the Magnetic Fusion Engineering Act of 1980 which is viewed as the most aggressive program put forth in recent years.

The Fusion Advisory Panel finally concluded by saying *"the pace of the development of fusion power is now primarily in the hands of Congress and the President, not in the hands of the technologists"*.

REFERENCES FOR SECTION 6.3

1. The Department of Energy Policy for Fusion Energy, U.S. Dept. of Energy, Washington, DC, Sept. 1978, DOE/ER-0018.

2. Fusion Energy: An Overview of the Magnetic Confinement Approach, Its Objectives, and Pace, U.S. Government Printing Office, Washington, DC, Dec. 1980, p. 166.

6.4 THE INTERNATIONAL TOKAMAK REACTOR STUDY (INTOR)

In September 1978, the Director General of the IAEA approved the recommendation of the International Fusion Research Council (IFRC) to establish a workshop to evaluate the possibility of international cooperation on constructing the next large fusion device. In 1979, representatives of the U.S., U.S.S.R., Japan and the European community met in several workshops in Vienna and together with the support of roughly 100 scientists in each country developed the following consensus on the state-of-the-art for a Tokamak Engineering Test Facility (INTOR) (1).

> *"A substantial physics and technology data base for INTOR exists today, and this data base will be expanded over the next few years by currently planned programs. However, certain crucial information will not be developed by currently planned programs. Much of this missing information could be developed on the INTOR time scale by the expansion and/or acceleration of existing R&D programs and by the establishment of new R&D programs. On this basis, it is concluded that it is scientifically and technologically feasible to undertake the construction of an INTOR-like device to operate in the early 1990's, provided that the supporting R&D effort is expanded immediately to provide an adequate data base within the next few years in a few critical areas. Furthermore, it is concluded that the construction of an INTOR-like device to operate in the early 1990's is the appropriate next major step in the development of fusion power."*

On the basis of this recommendation, the INTOR group met in Vienna in 1980, 1981 and 1982 to provide the details for such a collaborative effort. Unfortunately the INTOR project was halted in 1982 due to political actions in the IAEA and the study was slated to restart in mid-1983. Beyond 1983, the level of activity is uncertain.

REFERENCE FOR SECTION 6.4

1. General Characteristics and Assessment of the Scientific/Technical Feasibility of the Next Major Device in the Tokamak Fusion Program, U.S. Dept. of Energy, Washington, DC, Sept. 1979, DOE/ET-0117/1.

6.5 FUSION REVIEW PANEL OF THE ENERGY RESEARCH ADVISORY BOARD (BUCHSBAUM PANEL)

In February 1980, Dr. Edward A. Frieman, Director of Energy Research, requested that the Energy Research Advisory Board (ERAB) review the DOE Magnetic Fusion Program. Of particular concern to the DOE was the judicious choice of the next major steps to be taken in proceeding from the current generation of experimental devices towards demonstration of economic power production from fusion. Of equal concern is the overall soundness of the DOE Magnetic Fusion Program as stated previously (1); its pace, scope, and funding profiles. A review committee headed by Dr. S. Buchsbaum was formed. The committee finished its investigations by August 1980 and issued its report (2). Some of the more important conclusions are given below.

1. The Magnetic Fusion Program should construct a device which contains a burning plasma and includes technologies fundamental to a commerical reactor. This Fusion Engineering Device (FED) should be placed at a Center for Fusion Engineering, to be built within 10 years and cost less than 1 billion (1980) dollars. It was concluded that the then current ETF design was too ambitious for the fusion program.

2. * The program should proceed with the MFTF-B device at LLNL.

   * The program should build, in addition to the FED, several other tokamak facilities which will address physics issues that have arisen since the TFTR was built.

   * EBT-P construction should wait until experimental results from existing facilities are available.

   * Work on alternate concepts should continue but only a few should be selected to go to the Proof of Principle stage.

   * DOE should support a strong program on fusion fuel cycles other than DT.

In summary, the Buchsbaum Panel felt that the community had gone too far (both technically and financially) in the design of the next step (ETF) and was asking that a more modest step be taken. It was urging that more attention be paid to physics issues which have originated over the past few years and that the community be more cautious in the area of alternatives such as EBT.

REFERENCES FOR SECTION 6.5

1. The Department of Energy Policy for Fusion Energy, U.S. Dept. of Energy, Washington, DC, Sept. 1978, DOE/ER-0018.

2. Report of the Fusion Review Panel of the Energy Research Advisory Board, U.S. Dept. of Energy, Washington, DC, June 1980.

6.6 MAGNETIC FUSION ENGINEERING ACT OF 1980 (MFEA-80)

As a result of the Fusion Advisory Panel's report to congress and a very active informational effort by Fusion Power Associates (a voluntary nonprofit organization), Representative Mike McCormack introduced the Fusion Energy, Research, Development, and Demonstration Act of 1980 (H.R. 6308) on January 29, 1980. The purpose of the bill, which had 160 co-sponsors, was to establish, as a national commitment, the goal of using the fusion process to successfully generate electricity in a demonstration power plant before the end of the century. The bill was reported from the House Science and Technology Committee on June 17, 1980 and passed the House on a 365 to 7 vote on August 25, 1980.

Similar legislation was introduced in the Senate by Senator Paul Tsongas (S.2926) on July 2, 1980. The bill, with its 23 co-sponsors, was amended and reported from the Senate Energy and Natural Resources Committee on September 15, 1980. It was unanimously passed by the Senate on September 24, 1980 and was signed into law by President Carter on October 7, 1980 (Public Law 96-386).

The MFEA-80 calls for, among other things,

> Section 2(b)(3) *"to achieve at the earliest practicable time, but not later than the year 1990, operation of a magnetic fusion engineering device based on the best available confinement concept."*
>
> Section 2(b)(4) *"to establish as a national goal the operation of a magnetic fusion demonstration plant at the turn of the twenty-first century."*
>
> Section 2(b)(5) *"to foster cooperation in magnetic fusion research and development among government, universities, industry, and national laboratories."*

Section 2(b)(6) *"to promote the broad participation of domestic industry in the national magnetic fusion program."*

Section 2(b)(7) *"to continue international cooperation in magnetic fusion research for the benefit of all nations."*

With regard to financing the program the MFEA-80 states:

"The Congress hereby finds ... (a) accelerations of the current magnetic fusion program will require a doubling, within seven years, of the present funding level without consideration of inflation and a 25 per centum increase in funding in each of fiscal years 1982 and 1983."

The Secretary of Energy is also directed, in the MFEA-80, to *"develop a plan for the creation of a national magnetic fusion engineering center for the purpose of accelerating fusion technology development via the concentration and coordination of major magnetic fusion engineering devices and associated activities at such a national center".*

The financial implications of the MFEA-80 are given in Table 6.6-1 below.

In summary, MFEA-80 calls for a very aggressive program similar to that proposed by the Hirsch Panel in 1977. The MFEA-80 was backed by numerous studies, most of the scientific community, and the legislative branch of the U.S. government. Subsequent actions by the current administration and by the Department of Energy, however, have not been consistent with the Act and in

TABLE 6.6-1

Summary of Funding Levels for Magnetic Fusion

	Millions $	
FY	Levels Called for in MFEA-80	Actual Levels
1981	434.5	393.6
1982	490.7	453.8
1983	633.8	447.1
1984	664.5	467 (prelim.)
1985	720.1	

fact, the MFEA-80 seems to have little effect on the overall U.S. magnetic fusion program as indicated by Table 6.6-1.

6.7 EUROPEAN FUSION REVIEW PANEL (BECKURTS PANEL)

The Fusion Review Panel, chaired by Professor K. H. Beckurts, was set up by a Decision of the Commission of the European Community on November 26, 1980 (immediately after the passage of the MFEA-80 in the U.S.). The committee reviewed the European program between January and June of 1981. Some of the main recommendations of that committee are summarized below (1).

Area	Recommendation
Program Strategy	* Pursue a program where approximately 80% of the resources are directed toward the tokamak.
	* Complete the first stage of the tokamak program (JET).
	* Establish a design team to examine the Next European Tokamak (NET).
	* Devote no more than 15% of the program to alternate concepts (mirrors, RFP, stellarators, ICF, etc.) and encourage collaboration with other world fusion programs, especially the U.S.A.
	* Review world results in the tokamak field before deciding whether to go ahead with the second stage of the tokamak program (NET).
JET Project	* Push ahead as fast as possible up to the stage of introducing tritium in about 1989. The final decision to insert tritium must be carefully examined.
General Tokamak Program	* Implement the TORE SUPRA, FTU, and ASDEX-Upgrade projects.
Alternate Magnetic Confinement Systems	* Implement the RFX Project.
	* Implement 2 stage development of stellarators.
	* Monitor U.S. activity in mirrors.

Inertial Confinement	*	Monitor world progress in inertial confinement.
Fusion-Fission Hybrids	*	Do not implement a specific program in this area at the present time.
Budget	*	A real expansion of approximately 9% per year over the 1982-86 period. This indicates a total program of 1500 MIO ECU at 1981 prices (U.S. equivalent = 1.635 billion dollars).
International Cooperation	*	Continue to participate in INTOR.
	*	Seek expanded collaboration with other major countries in world fusion programs.

In summary, the Beckurts report takes a much more conservative approach to the European fusion program than was taken in the U.S. This is illustrated by the following quote, *"However, in view of the nature of this work, the uncertainites involved, and the role which nuclear fusion energy is likely to play in the foreseeable future, there seems no reason to treat it as a crash programme".* It is clear from this report that the Beckurts Panel was suggesting that the concentration (80% of the funding) of the European program on the tokamak should continue and that alternate magnetic fusion approaches should play a minor role. The report also advises that inertial fusion should simply be monitored with only low level basic studies pursued. The proposed expansion of the program is very modest and, in the words of the Beckurts Report *"...it appears unlikely that commerical fusion power will be in general use within the next 50 years and by that time the worldwide expenditure on research, development, and demonstration may well have exceeded 100 BIO ECU."*

REFERENCE FOR SECTION 6.7

1. Report of the European Fusion Review Panel, Commission of the European Communities, Brussels, EUR-FU-BRU/XII-715/81, June 1981.

6.8 MAGNETIC FUSION ADVISORY COMMITTEE

In May of 1982, Dr. Alvin W. Trivelpiece, Deputy Director of the U.S. DOE, established a Magnetic Fusion Advisory Committee (MFAC) headed by Professor Ronald C. Davidson. The first assignment to this committee was made on June 1, 1982 when Dr. Trivelpiece asked the MFAC to address three critical issues within the U.S. fusion program.

1. a. Assess the makeup and pace of the steady state tokamak and mirror programs with respect to how they may qualify for the ETR.

   b. Assess the present experimental facilities now planned to see if they can advance each concept to the ETR stage. Also make recommendations for a program that would allow the concepts to be ready for the ETR stage within the budget levels specified (see below).

   c. Recommend the priorities that need to be given to each program with respect to building a technical base prior to an ETR decision.

2. Review the Stellarator, Elmo Bumpy Torus, and Reversed Field Pinch and recommend their priorities as backups to the steady state tokamaks and mirror reactor development programs.

3. Consider the possible programmatic roles of the TFTR facility after feasibility experiments and make recommendations as to its use as

   a. an engineering development facility

   b. an alpha particle physics experiment

   c. a reactor level hydrogen experiment

   d. other.

The MFAC committee was to consider the 3 questions above for 3 different constant budget scenarios (400, 500, and 600 M$) for the forseeable future.

The reason that it was necessary to form the MFAC to address these questions was the decision to drop the Fusion Engineering Demonstration reactor from active consideration as the fusion technology development facility. It was stated, *"the high cost of the FED precludes construction at this time"*. The cost referred to was 1.2 billion dollars as estimated by the FEDC in Oak Ridge to build the reactor.

The MFAC divided itself initially into 3 groups to address the questions posed by Dr. Trivelpiece and all had completed their work by December 30, 1982. A brief summary of their findings is given below.

MFAC-I

* *"At present, the tandem mirror and tokamak concepts can be embodied in viable reactor designs of roughly similar characteristics. In both reactor designs, there are scientific and technological assumptions that remain to be validated by*

experiment. Both designs also have the potential for significant further improvement through technical innovations that are in the exploratory phase."

* The existing data base for the tokamak implies the feasibility of net power production and the current experimental program is addressing the achievablity of high power density and long pulse operation.

* Tandem mirror research is in an earlier stage of development and the demonstration of the thermal barrier is particularly important.

* A significant upgrade of TFTR should be considered for the 1990's period.

* It is essential that the MFTF-B be completed in a timely fashion.

* The relative promise of the tandem mirror and tokamak reactors can be assessed in a preliminary way by the end of 1984 and more substantially by the end of 1987. The program funding balance should be readjusted on the basis of their technical assessments.

* The decision to proceed with construction of either a tokamak or tandem mirror ETR could be made after the 1987 assessment with DT operation scheduled for the late 90's. Competing conceptual designs should be undertaken soon, with intensified efforts as appropriate following the 1984 assessment.

* Substantial incremental funding will be needed, beginning in FY-88, in order to move forward to an ETR and to the demonstration of commerical feasibility.

* The technical status and rate of progress in the tokamak and mirror areas are favorable to the timely development of an attractive commercial reactor provided that the funding is program driven. In FY-84 a 600 M$ level approximates a program driven case; the 500 M$ level implies a significant curtailment of productivity and the 400 M$ level would require dismantling key elements of the present goal oriented national magnetic fusion program.

<u>MFAC-II</u>

Conclusions here were divided along the three budget scenarios.

<u>Reduced Budget Case</u> (400 M$)

* The alternative concepts be actively pursued but no new facilities be built.

* One mainline (tokamak or mirror) program should be aggressively pursued while the other is reduced to provide money for alternate concepts.
* The EBT-P project should be cancelled but the EBT program continued.
* Present RFP program should be continued but no upgrade permitted.
* Stellarator program should be modestly supported by doing computational studies and analyses.

<u>Constrained Budget Case</u> (500 M$)

* Proposed RFP program should be pursed with vigor.
* The MFAC panel voted 6 to 5 to continue the EBT-P program.
* The "advanced" stellarator program should be started but with only a single experiment.

<u>Enhanced Budget Cost</u> (600 M$)

* Implement Enhanced RFP Program.
* Implement Reference EBT Program.
* Implement the Reference Stellarator Program.

<u>MFAC-III</u>

* The MFAC agreed that the next major step beyond TFTR should be the demonstration of ignition and a long pulse equilibrium.
* The mission of the TFTR Upgrade should be raised to the level of a Tokamak Fusion Core Demonstration (TFCD) which means that the scope of the tokamak Engineering Test Reactor (ETR), as the last step before commercialization, can be enhanced correspondingly.
* In order to provide a reactor-relevant configuration on the road to a tokamak ETR, a superconducting-coil device such as DCT-8† is highly desirable. However, lower-cost options for the achievement of ignition and long-pulse burn, using copper coils, should continue to be investigated. The technical merit of copper-coil options relative to superconducting-coil options should be determined in the light of budgetary constraints and technological needs.

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†DCT-8 is a specific device proposed by PPPL.

* While the construction costs for an equilibrium burn experiment of the DCT-8 type are substantially smaller than those for the FED or for a tokamak ETR, it is clear that incremental funding above the base program level will be required for timely implementation.

* Less ambitious TFTR improvement and upgrade options should continue to be studied in order that a backup may be provided in the event that budgetary stringency precludes proceeding with DCT-8.

* Since the nuclear engineering requirements of fusion reactors cannot be met solely by the proposed upgrades of the TFTR, a serious parallel effort involving complimentary facilities must be undertaken to establish the nuclear data base for ETR.

As this report is going to press, Dr. Trivelpiece has requested that two more panels be formed. The fourth panel (MFAC-IV) will assess the possible upgrades of MFTF-B and the fifth panel (MFAC-V) will assess the role of Universities in the fusion program.

The total integrated MFAC report (in which all panel recommendations could be put together on the reference budget cases) was not available at the time of this report. However, it appears to be very difficult to follow the recommendations in any of the five cases, except for possibly the 600 M$ case.

6.9 CONCLUSIONS

As the fusion community entered into the 1978-82 review period, the hope was still held that usable power could be generated by the turn of the century from magnetic fusion. This hope was first dashed in the U.S. by the Foster Panel (1978) and attempts to revive it were made in 1979 and 1980. With the passage of the Magnetic Fusion Engineering Act of 1980, the U.S. fusion program seemed back on track to building a large scale fusion device by the turn of the century. However, budgetary and DOE policy constraints in the 1981-1982 period forced the program back into a mode where useful power from magnetic fusion was again pushed off well into the 21st century. One can only hope that the pendulum will swing back again over the next 5 years.

APPENDIX:

LIST OF ACRONYMS AND GLOSSARY OF SELECTED FUSION TERMS

LIST OF ACRONYMS

AC	alternating current
AIC	Alfvén ion cyclotron
AIF	Atomic Industrial Forum
ALC	axial loss cone
AMBAL	Soviet tandem mirror experiment
ANS	American Nuclear Society
ASDEX	Axisymmetric Divertor Experiment
AWRE	Atomic Weapons Research Establishment
AWRL	Atomic Weapons Research Laboratory
Alcator	Alta Campu Torus
BA	budget authorization
BIO	billion
BNL	Brookhaven National Laboratory
BO	budget outlay
BOL	beginning of life
BW	boiling water
CANDU	Canadian heavy water reactor
CC	central cell
CECE	Combined Electrolysis and Catalytic Exchange
CFR	Code of Federal Regulations
CLEO	Conference on Lasers and Electro-Optics
CPMP	Comprehensive Program Management Plan
CRFPR	Compact Reversed Field Pinch Reactor
CTHR	Commercial Tokamak Hybrid Reactor
CTX	Compact Torus Experiment
CX	charge exchange
D-D	deuterium-deuterium
D.C.	direct conversion
DC	direct current
DCLC	drift cyclotron loss cone
DCT	Direct Current Tokamak
DEMO	Demonstration reactor
DITE	Divertor and Injection Tokamak Experiment
DOE	Department of Energy
dpa	displacements per atom
D-T	deuterium-tritium
DTHR	Demonstration Tokamak Hybrid Reactor
EAGLE	Energy Absorbing Gas, Lithium Ejector
EBR	Experimental Breeder Reactor
EBT	Elmo Bumpy Torus
EC	European Community
ECH	electron cyclotron heating
ECRH	electron cyclotron resonance heating
ECU	European Currency Unit
EM	electromagnetic
EOL	end of life
EPR	Experimental Power Reactor

EPRI	Electric Power Research Institute
ERAB	Energy Research Advisory Board
ERDA	Energy Research and Development Administration
ETF	Engineering Test Facility
ETR	Engineering Test Reactor
FCU	Fuel Cleanup Unit
FED	Fusion Engineering Device
FEDC	Fusion Engineering Design Center
FEL	free electron laser
FETF	Fusion Engineering Test Facility
FINTOR	Frascati, Ispra, Napoli Tokamak
FPD	Fusion Power Demonstration
FRC	Field Reversed Configuration
FRG	Federal Republic of Germany
FRM	Field Reversed Mirror
FT	Frascati Tokamak
FTU	Frascati Tokamak Upgrade
FW/B/S	first wall/blanket/shield
FY	fiscal year
GA	General Atomic Co. (now GA Technologies, Inc.)
GAMMA-10	Japanese thermal barrier tandem mirror device
GDL	Gas Dynamic Laser
GDR	German Democratic Republic
GEIS	Generic Environmental Impact Statement
GSI	Gesellschaft für Schwerionenforschung (Darmstadt, FRG)
HEDL	Hanford Engineering Development Laboratory
HFCTR	High Field Compact Tokamak Reactor
HIB	heavy ion beam
HIBALL	Heavy Ion Beams and Lithium Lead
HIF	heavy ion fusion
H.R.	House of Representatives
HS&T	House Science & Technology
HTE	high temperature electrolysis
HTGR	High Temperature Gas Reactor
HV	high voltage
HYFIRE	Brookhaven fusion/synfuel design study
HYLIFE	High Yield Lithium Injection Fusion Energy
IAEA	International Atomic Energy Agency
ICCS	Internally Cooled Cabled Superconductor
ICF	Inertial Confinement Fusion
ICH	ion cyclotron heating
ICRF	ion cyclotron range of frequencies
ICRH	ion cyclotron resonance heating
IEEE	Institute of Electrical and Electronics Engineers
IFRC	International Fusion Research Council
ILE	Institute for Laser Energetics (Osaka University)
IMS	Interchangeable Modular Stellarator
INESCO	International Nuclear Energy Systems Company
INPORT	Inhibited Flow - Porous Tube
INTOR	International Tokamak Reactor
IPP	Institut fuer Plasmaphysik (Garching, Federal Republic of Germany)

IPP	Institute for Plasma Physics (Nagoya University)
IQO	Institut fuer Quantenoptik (Garching, FRG)
ISS	Isotopic Separation System
ISX	Impurity Study Experiment
JAERI	Japan Atomic Energy Research Institute
JET	Joint European Torus
JFT	Japanese Fusion Tokamak
KfK	Kernforschungszentrum Karlsruhe GmbH
LAMEX	Large Axisymmetric Mirror Experiment
LANL	Los Alamos National Laboratory
LASNEX	laser target implosion simulation computer code
LBL	Lawrence Berkeley Laboratory
LCT	Large Coil Task
LH	lower hybrid
LHH	lower hybrid heating
LHRF	lower hybrid range of frequencies
LIB	light ion beam
LIBRA	Light Ion Beam Reactor
LLE	Laboratory for Laser Energetics (University of Rochester)
LLNL	Lawrence Livermore National Laboratory
LN	low nitrogen
LWR	Light Water Reactor
MARS	Mirror Advanced Reactor Study
MFAC	Magnetic Fusion Advisory Committee
MFE	magnetic fusion energy
MFEA-80	Magnetic Fusion Energy Engineering Act of 1980
MFTF	Mirror Fusion Test Facility
MHD	magnetohydrodynamics
MIO	million
MIT	Massachusetts Institute of Technology
MMX	Multiple Mirror Experiment
MS	molten salt
MSR	Modular Stellarator Reactor
NB	neutral beam
NBI	neutral beam injection
NBT	Nagoya Bumpy Torus
NET	Next European Tokamak
NRC	National Research Council
NRL	Naval Research Laboratory
NSF	National Science Foundation
NUWMAK	University of Wisconsin tokamak reactor design
NYU	New York University
OHTE	Ohmically Heated Toroidal Experiment
ORMAK	Oak Ridge Tokamak
ORNL	Oak Ridge National Laboratory
OSA	Optical Society of America
PBFA	Particle Beam Fusion Accelerator
PDX	Poloidal Divertor Experiment
PF	poloidal field
PINI	Plug-In Neutral Injectors
PLT	Princeton Large Torus

PPPL	Princeton Plasma Physics Laboratory
PWR	pressurized water reactor
R&D	research and development
REB	relativistic electron beam
REGAT	Reduced Damage Gradient Test
RF	radio frequency
RFP	Reversed Field Pinch
RFPR	Reversed Field Pinch Reactor
RFQ	radio frequency quadrupole
RFX	Reversed Field Experiment
RGH	rare gas halide
SAFA	Society Against Fusion Acronyms
SAFFIRE	University of Illinois field reversed mirror design
SATYR	UCLA tandem mirror reactor design
SNL	Sandia National Laboratory
SOLASE	University of Wisconsin laser fusion reactor design
SPTR	Swimming Pool Test Reactor
SS	stainless steel
STARFIRE	Argonne National Laboratory tokamak reactor design
STM	Symmetric Tandem Mirror
TAERF	Texas Atomic Energy Research Foundation
TARA	MIT thermal barrier tandem mirror experiment
TASKA	Tandem Spiegelmaschine Karlsruhe
TDF	Target Development Facility
TDF	Technology Demonstration Facility
TF	toroidal field
TFR	Tokamak Fusion Reactor (France)
TFTR	Tokamak Fusion Test Reactor
THM	tonnes heavy metal
TMHR	Tandem Mirror Hybrid Reactor
TMNS	Tandem Mirror Next Step
TMR	tandem mirror reactor
TMX	Tandem Mirror Experiment
TMX-U	Tandem Mirror Experiment Upgrade
TN	thermonuclear
TNS	The Next Step
TORMAC	Toroidal Magnetic Cusp
TPFS	Tokamak Poloidal Field System
TRACT	Triggered-Reconnection Axially Compressed Torus
TSTA	Tritium Systems Test Assembly
UWMAK	University of Wisconsin tokamak reactor designs
UWTOR	University of Wisconsin stellarator reactor design
W VII-A	Wendelstein VII-A stellarator
WEURF	Wisconsin Electric Utilities Research Foundation
WILDCAT	Argonne D-D tokamak reactor design
WITAMIR	Wisconsin Tandem Mirror

GLOSSARY OF SELECTED FUSION TERMS

ablator - A layer of pellet material which, due to absorption of beam energy, is changed into a plasma that accelerates radially outward to create a reaction force that drives an implosion of interior pellet material.

adiabatic compression - Compression (of a gas, plasma, etc.) not accompanied by gain or loss of heat from the outside. For a plasma in a magnetic field, a compression slow enough that the magnetic moment (and other adiabatic invariants) of the plasma particles may be taken as constant.

adiabatic invariant - Parameter of the motion of a charged particle in a magnetic field, which remains constant when the variations of the magnetic field in space and time are sufficiently slow.

advanced fuels (fusion) - Fusion fuels other than a deuterium tritium mixture which may have advantages as fuels in spite of the increased requirements to achieve fusion conditions.

Alcator - Toroidal confinement device at Massachusetts Institute of Technology designed and operated to produce plasmas with relatively high current and particle densities. Typical design parameters involve a relatively small major radius and high magnetic fields, both for the toroidal magnet and the air core transformer systems.

Alfvén waves - Waves, of a much lower frequency than the ion cyclotron frequency, occurring in a plasma or in a conducting fluid immersed in a magnetic field, characterized by a transverse motion of the lines of force together with the plasma. These transverse hydromagnetic waves propagate at a velocity which depends on the strength of the magnetic field and the particle density.

aspect ratio - The ratio between the major and minor radius (R/a) of the plasma axisymmetric toroidal confinement device.

barn - (Symbol b) Unit area used in expressing the cross sections of atoms, nuclei, electrons, and other particles. One barn is equal to 10^{-24} square centimeter. (See cross section.)

beam ducts - Channels penetrating the fusion reactor vessel, allowing transport of driver beams to the target.

beta value - Ratio of the outward pressure exerted by the plasma to the inward pressure which the magnetic confining field is capable of exerting. Equivalent to the ratio of particle energy density to magnetic field energy density.

blanket - Region surrounding a fusion reactor core within which fusion neutrons are slowed down, heat is transferred to a primary coolant, and tritium is bred from lithium. In hybrid applications, fertile materials (U-238 or Th-232) are located in the blanket for fissile-fuel breeding purposes.

blanket energy multiplication (M) - Energy generation in blanket per fusion neutron divided by kinetic energy of fusion neutron.

breakeven - See Lawson criterion.

bremsstrahlung - Radiation emitted as a result of deflection (e.g., through near collisions) of rapidly moving charged particles.

central cell (or solenoid plasma) - The long central part of a tandem mirror in which the plasma is electrostatically confined by the end mirror cells.

charge exchange - Process in which there is a transfer of charge between two bodies during a collision between them (e.g., the collisional transfer of an electron from a neutral atom to a singly charged positive ion, the latter becoming neutral and the former charged).

closed magnetic configuration - A collection of magnetic field lines which remain entirely within a plasma confinement region.

coil, baseball - A coil wound in the shape of a baseball seam to produce an absolute minimum B field for mirror confinement.

coil, compression - A coil that produces a time varying magnetic field which adiabatically compresses a plasma.

coil, poloidal field - A set of conductors that produces a magnetic field perpendicular to the minor axis of a toroidal device.

coil, toroidal field - A coil that produces a magnetic field which encircles the major axis of a toroidal device.

coil, yin-yang - Nested coil pair used in mirror devices to produce a magnetic well or minimum B configuration.

compression ratio, ICF - Ratio of the fuel zone density at peak of implosion to that prior to implosion.

confinement, electrostatic - Use of electric fields to contain a plasma.

confinement, inertial - Use of inertia (i.e., finite time is required to accelerate to finite mass) to prevent escape of fusing particles.

confinement, magnetic - Use of magnetic fields to contain a plasma.

containment vessel - Gas-tight shell or other enclosure around a reactor.

coolant - Substance circulated through a nuclear reactor to remove or transfer heat. Common coolants are water, air, carbon dioxide, liquid sodium and sodium potassium alloy (NaK).

coulomb collision - Collision between two charged particles. Interaction of their electric fields results in deflection of each of the particles from its initial path.

coulomb force - Force of repulsion (or attraction) exerted by one electrically charged body upon another. Also called "electrostatic force."

cross section (for a given event) - Quantity proportional to the probability that such an event will occur.

cross section - (Symbol σ (sigma)) A measure of the probability that a nuclear reaction will occur. Usually measured in barns, it is the apparent (or effective) area presented by a target nucleus (or particle) to an on-coming particle or other nuclear radiation, such as a photon of gamma radiation.

cusped geometry - Magnetic configuration in the form of cusps, such that the lines of magnetic force are everywhere convex toward the center of the configuration. Such a configuration is of particular interest for the confinement of plasma, since it is theoretically stable against the development of hydromagnetic instabilities.

cyclotron resonance heating - Mode of heating of a plasma by resonant absorption of energy based on the waves induced in the plasma at the cyclotron frequency of ions or electrons or at a harmonic frequency of the former.

deuterium atom - An isotope of the hydrogen atom with one proton and one neutron in its nucleus and a single orbital electron.

diffusion - Interpenetration of one substance into another as a result of thermal motion of the individual particles (e.g., diffusion of a plasma across a magnetic field as a result of collisions).

direct conversion - Generation of electricity by direct recovery of the kinetic energy of the charged fusion reaction products.

distribution function - Density function or number of particles per unit volume of phase space: a function of the three space coordinates and the three velocity coordinates. A point in phase space represents a given position on ordinary space and a given velocity in velocity space. Therefore, the distribution function evaluated at such a point is the number of average density of particles per cubic length and cubic velocity that have the same position and velocity which is represented by the point. In reality one never counts the exact number of particles in a unit volume, and the distribution function represents the average density over a reasonably long time or at any particular time represents the most "probable" distribution of particles.

DITE - Divertor and Injection Tokamak Experiment at Culham Laboratory in UK will make an extensive study of (i) neutral injection heating in tokamaks and (ii) the control of impurities and recycling using a bundle divertor.

divertor - Component of a toroidal fusion device that serves to divert charged particles in the outer shell of the discharge into a separate chamber where they strike a barrier, become neutralized, and are pumped away. In this way, energetic particles in the outer shell are prevented from striking the walls of the main discharge chamber and releasing secondary particles that would cool the discharge.

Doublet devices - Non-circular cross-section tokamak devices with a kidney-shaped cross section.

drift cyclotron loss cone (DCLC) instability - Microinstability at ion gyrofrequency driven by the loss-cone nature of the ion distribution (in mirror fields) in the presence of a radial density gradient.

drift surface - Surface on which the guiding center of a particle is constrained to move under the laws of adiabatic invariance.

driver, ICF - The system used to produce the energy required for implosion, e.g. the laser, electron or ion accelerators depending on the particles used for energy transport to the target.

e-beam fusion - The concept of imploding a pellet with electron beams to induce inertial confinement fusion.

EBT, Elmo Bumpy Torus - A toroidal device consisting of a number of connected simple magnetic mirror sections in which annuli of heated electrons are created between each pair of mirror coils using intense microwave radiation at the electron cyclotron frequency.

EBT-S - Elmo Bumpy Torus is 24-sector toroidal magnetic trap at Oak Ridge National Laboratory that used microwave heating to produce and maintain a steady-state hot plasma.

ECRH - Initials used to denote electron cyclotron resonance heating in which only electrons gain energy by action of an applied RF (radio-frequency) field operating at the electron cyclotron frequency.

electron volt (eV) - Unit of energy, equal to the energy acquired by a singly charged particle in passing through a potential difference of one volt. 1 eV = 1.6×10^{-19} joule.

electrostatic waves - Longitudinal waves appearing in a plasma on account of a perturbation of electric neutrality. In the case of a cold unmagnetized plasma, and also for large wavelengths, the frequency of these waves is, by definition, equal to the plasma frequency.

energy balance - Comparison of the energy put into a system (e.g., a hot plasma) and the energy dissipation from the system by one mechanism or another (e.g., by an increase in the plasma temperature, by radiation, or by various mechanisms of particle loss from the plasma).

energy confinement time - The total thermal energy divided by the bulk rate at which energy is lost from a magnetically confined plasma due to all energy loss mechanisms. Sometimes this is calculated for a particular species (electrons or ions) of a particular channel of power loss.

equilibrium (macroscopic or MHD) - The complete balance of forces in a magnetically confined plasma.

fertile material - Nuclide that will convert to fissile material on neutron capture and radioactive decay (e.g., U-238 or Th-232).

first wall - First physical boundary that surrounds a plasma.

fissile material - While sometimes used as a synonym for fissionable material, this term has also acquired a more restricted meaning, namely: any material fissionable by neutrons of all energies, including (and especially) thermal (slow) neutrons as well as fast neutrons; for example, ^{235}U and ^{239}Pu.

fissile breeding ratio, fusion ($^{239}Pu/n$ and/or $^{233}U/n$) - Net fissile atom production per fusion neutron.

Fokker-Planck equation - Equation that describes the motion of a free particle in velocity space and which is applicable to plasmas when the cumulative effect of weak deflections resulting from relatively distant encounters is more important than the effect of occasional large deflections.

fusion-fission hybrid - Reactor in which energy is produced by both fusion and fission reactions. Fusion neutron source is typically surrounded by a subcritical blanket containing fissile material. If fertile material is also contained in the blanket, the reactor will produce additional fissile material.

hybrid reactors - See fusion-fission hybrid.

ICRH - Initials used to denote ion cyclotron resonance heating. See cyclotron resonance heating.

ignition temperature - Temperature at which the energy deposited in a plasma through the fusion process just equals the energy losses (e.g., through radiation processes).

inertial confinement fusion, ICF - Retention of fuel by inertial forces in a reaction volume for a time sufficient for fusion reactions to take place.

inertial confinement parameter (ρR) - Product of density times radius of a compressed pellet.

instability, plasma - State of a plasma in which any small perturbation amplifies itself to a considerable alteration of the equilibrium of the system.

Ioffe bars - A set of conductors placed near a plasma in an open ended device to distort the field into a more favorable minimum B configuration.

ion-beam fusion - The concept of imploding a pellet with ion beams to induce inertial confinement fusion.

ion cyclotron resonance heating (ICRH) - Heating of a plasma by resonant absorption of energy from waves induced in the plasma at or near the ion cyclotron frequency.

ISX tokamak - Flexible, medium-size tokamak designed for easy access and rapid changing of the vacuum system and poloidal field system.

JET tokamak - Joint European Torus is a large tokamak that is commonly owned by the European Communities. It is being built at the Culham Laboratory and will be comparable to TFTR.

JT-60 - Large Japanese tokamak. Purpose is to extend the tokamak parameters closer to reactor parameters and to investigate plasma confinement and heating and associated technological problems.

laser - Light Amplification by Stimulated Emission of Radiation.

laser fusion - Nuclear fusion process that occurs when a small pellet of fuel material is compressed by a burst of laser light (see confinement, inertial).

Lawson criterion - Condition that the product of number density and confinement time of a plasma must equal approximately 10^{14} cm^{3}-sec at a temperature of about 70,000,000 degrees to produce net power in a fusion reactor.

limiter - Material extension into a plasma chamber designed to separate the plasma from the wall.

loss cone - In the velocity space related to a magetic mirror, the cone having an axis of symmetry parallel to the magnetic field and an apex angle alpha defined by $\sin \alpha = 1/\sqrt{R}$, R being the mirror ratio. Particles whose velocity vectors lie in the loss cone will not be reflected by the mirror.

magnetic mirror - Magnetic field that is generally axial with a local region of increased intensity causing convergence of the field lines. A particle moving into the region of converging magnetic field lines will be reflected if the ratio of its energy parallel and perpendicular to the magnetic field satisfies the relationship:

$$\frac{E_{\parallel}}{E_{\perp}} < \left(\frac{B_m - 1}{B_o}\right)$$

where B_m and B_o are the magnetic field strengths at the mirror and at the original point, respectively.

magnetic pressure - Pressure a magnetic field is capable of exerting upon a plasma, equivalent to the energy density of a magnetic field.

magnetic pumping - Term given to type of plasma heating in which plasma is successively compressed and expanded by means of rapidly fluctuating external magnetic field.

magnetic well - See minimum B configuration.

magnetohydrodynamics (MHD) - Science dealing with the motion of electrically conducting fluids (liquids and gases) interacting with a magnetic field.

Maxwell-Boltzmann distribution - Distribution of particle velocities (or energies) that occurs in any gas or plasma when it is in thermal equilibrium at a given temperature.

MeV - One million electron volts (see electron volt).

MFTF - (Mirror Fusion Test Facility) Large mirror machine experiment at Lawrence Livermore Laboratory that uses a superconducting magnet of the yin-yang design. Technical problems are addressed that must be solved before reactors can be attained.

minimum B configuration - Name given to a magnetic configuration that increases everywhere in strength with increasing distance from the plasma it is confining. In such a configuration the plasma finds itself in a region of minimum magnetic potential.

mirror, standard - Open field line system consisting of plasma confined in a magnetic well between a single pair of coils with plasma pressure less than that required for field reversal.

mirror, tandem - Combination of two mirror-confined plasmas at the ends of a longer plasma confined by a solenoidal magnetic field. The lower temperature ions in the solenoid are primarily confined electrostatically (axially) by the positive ambipolar potentials of the two end mirror plasmas.

neoclassical - Term used to characterize results of elementary calculation of collisional diffusion in finite toroidal geometries.

neutral injection concept - Concept similar to molecular ion injection concept, but with the molecular ions replaced by fast neutral atoms which are subsequently ionized inside the magnetic container.

neutron multiplication, fusion - One plus the number of neutrons added by neutron multiplying reactions per source nuetron.

neutron wall loading - Energy flux carried by fusion neutrons into the first physical boundary that surrounds the plasma.

neutron yield, ICF - The number of neutrons produced per pellet implosion.

PDX - (Poloidal Divertor Experiment) Large high-current divertor tokamak at Princeton whose primary objective is to determine the effectiveness of magnetic limiters and poloidal divertors in controlling impurities in hot reactor-like plasmas.

pellet debris, ICF - Charged particles emerging from target following disassembly.

pellet gain, ICF - The ratio of fusion energy produced by a pellet to driver energy incident on the pellet.

pellet, ICF - A small sphere containing fuel and other materials required to enhance the energy production.

pellet yield, ICF - The amount of energy released by thermonuclear reactions in a pellet.

plasma, cold - Model of plasma in which the temperature is neglected with respect to the effects of interest.

plasma, collisionless - Model of plasma in which the density is so low or temperature so high that close binary collisions have practically no significance because the time scales of interest are smaller than the collision time.

plasma containment - Operation intended to prevent, in an effective and sufficiently prolonged manner, the particles of a plasma from striking the walls of the container in which this plasma is produced.

plasma frequency - Natural frequency of oscillation of a plasma, caused by the collective motion of the electrons acting under the restoring force of their space-charge attraction to the relatively stationary ions. This frequency is proportional to the square root of the electron density.

PLT - (Princeton Large Torus) Large toroidal apparatus of the tokamak type, to operate without a copper shell. The purpose is to study heating and confinement of high-temperature plasmas with a plasma diameter roughly the geometric mean of present devices and proposed fusion reactors.

poloidal field - Magnetic field that encircles the plasma axis in toroidal devices.

poloidal field windings - Sets of windings in toroidal devices which are aligned along the plasma axis and produce poloidal fields. These include ohmic heating, shaping, vertical, equilibrium, and divertor windings.

power density - Rate of heat generated per unit volume of a reactor core.

preheat, ICF - Heating of the pellet core before arrival of the inward moving implosion front resulting in reduced compression relative to the non-preheated result.

Project Sherwood - Name often used to designate the U.S. program in controlled fusion during the 1950s and 1960s.

Q, engineering - Output energy of the system divided by the input energy to the system.

Q, plasma - Fusion energy output per unit of energy input into the plasma.

rad - (Acronym for radiation absorbed dose.) Basic unit of absorbed dose of ionizing radiation. Dose of one rad means absorption of 100 ergs of radiation energy per gram of absorbing material.

radiation damage, bulk - General term describing changes in chemical and/or metallurgical properties of structural components of fusion reactors caused by atomic displacement and nuclear transmutation (e.g., via (n,α) or (n,p) reactions) events occurring as the result of higher energy neutron environment.

radiation damage, surface - General term describing damage to the surface of the containment structure which directly interfaces with the thermonuclear plasma; includes such phenomena as radiation blistering, charged-particle (or neutron) sputtering, and spallation or exfoliation of layers of the surface.

recirculating power fraction (fusion) - Fraction of gross electrical energy produced used to run the plant.

rem - (Acronym for roentgen equivalent man.) Unit of dose of any ionizing radiation that produces the same biological effect as a unit of absorbed dose of ordinary x-rays.

reversed field pinch, RFP - A toroidal system similar in basic configuration to the tokamak, but with a much larger plasma current (of the order of ten times that in a tokamak) which creates a large poloidal field that compresses and traps the toroidal field inside the plasma and results in a reversal in the direction of the toroidal field outside the plasma. Conducting walls are required to produce eddy currents needed for stability.

scattering - Deflection of one particle as a result of collision with another. Elastic scattering is a scattering process in which the total kinetic energy is unchanged.

shaping field windings - Set of poloidal field windings in a tokamak which provides a magnetic field topology designed to constrain horizontal and vertical motion of the plasma as well as, in some applications, to produce a noncircular plasma cross section or a divertor separatrix.

shock heating - Heating produced by the impact of a shock wave.

shock wave - Wave produced (e.g., in a gas) as a result of a sudden violent disturbance. To produce a shock wave in a given region, the disturbance must take place in a shorter time than the time required for sound waves to traverse that region.

spheromak - A configuration similar to a collapsed form of a tokamak with a small aspect ratio in which toroidal and poloidal currents flow in the plasma itself and the toroidal magnetic field vanishes outside the plasma.

stellarator - Apparatus designed for the containment of a plasma inside a tube closed upon itself by using the combination of an axial magnetic field and of an additional field created by the helical windings. This magnetic configuration presents a rotational transformation in itself and permits containment in the absence of an axial current in the plasma.

support ratio, fusion-fission - Power of fission burners fueled by fissile breeding fusion reactors, divided by power of fusion reactors.

superconductor - Type of conductor that permits an electrical current to flow with zero resistance.

T-20 - A large Russian tokamak that will operate under reactor conditions.

tamper (also pusher), ICF - A layer of dense material surrounding the fuel that increases the burn time by increasing the disassembly time, i.e. it enhances the inertial confinement effect. It affects the compression by increasing the density of the fuel and also serves as a preheat shield.

temperature, kinetic - Measure of the energy of random motion of an assembly of particles in thermodynamic equilibrium. Specifically, temperature (T) appropriate to the Maxwellian distribution assumed by a system of particles upon equipartition of energy among the three translational degrees of freedom. The mean particle energy is then 3/2 kT, where k is Boltzmann's constant.

TFR tokamak - An iron core French tokamak that produces a 1 keV ion temperature plasma with a density in the range of 10^{13} particles$\cdot$cm^{-3} and a confinement time of 10 ms.

TFTR - (Tokamak Fusion Test Reactor) Closed-geometry device at Princeton University operating as a standard tokamak capable of modest compression. In addition to operating as a hydrogen experiment, it will be capable of injecting high-energy neutral deuterium into a tritium plasma in the two-energy component mode, producing a D-T plasma under reactor conditions.

thermodynamic equilibrium - Very general result from statistical mechanics which states that, if a system is in equilibrium, all processes that can exchange energy must be exactly balanced by the reverse process so that there is no net exchange of energy. For instance, ionization must be balanced by recombination, bremsstrahlung by absorption, etc. If a plasma complies with this statement, the distribution function of particle energies and excited energy levels of the atoms can be obtained from the Maxwell-Boltzmann distribution which is a function only of the temperature. Saha's equation is a special application of this result and gives the distribution function or density of ions and electrons.

thermonuclear burn efficiency - The fraction of fuel that reacts. It is proportional to the ratio of characteristic disassembly time to thermonuclear burn time.

thermonuclear burn wave, ICF - A radially advancing surface which encloses the region of highest reaction rate in the fuel of a pellet.

tokamak - Name given to a specific concept in the field of controlled fusion, involving confinement and heating of a plasma in a toroidal configuration. A large current induced in the plasma provides the rotational transform necessary for confinement while simultaneously heating the plasma.

TORMAC - Hybrid confinement system operating at high beta. A region of closed toroidal magnetic flux with high-beta plasma is separated by a narrow sheath from the surrounding field which contains externally produced poloidal components arranged in a toroidal line-cusp configuration. Plasma migrating to the outer sheath is temporally mirror-confined before being removed in a divertor system.

toroidal field coils - Coils in a toroidal system which provide the major confining field. Each turn completely surrounds the minor axis of the plasma.

toroidal system - Name given to the general class of "doughnut-shaped" magnetic fields in which lines of force close on themselves. Stellarators, tokamaks, and multipole devices are examples of this class of devices.

torsatron - A torsatron is a modification of the stellarator concept. It has a toroidal non-axisymmetric configuration, and rotational transform is provided by external windings. Unlike a stellarator, however, both toroidal and poloidal fields are generated by helical fields alone, with half the number of helical conductors required for a stellarator.

tritium breeding ratio, (T/n) fusion - Number of tritons produced per fusion event.

turbulence - Violent macroscopic fluctuations which can develop under certain conditions in fluids and plasmas and which usually result in the rapid transfer of energy through the medium.

turbulent heating - Technique of using turbulence induced by large electric fields to rapidly heat a plasma.

wall loading - Fusion reactor thermal output power divided by the area of the wall facing the plasma.

yin-yang coil - See coil, yin-yang.