

Summary of Opportunities in the Fusion Energy Sciences Program

Prepared by the
Fusion Energy Sciences Advisory Committee
For the
Office of Science of the U.S. Department of Energy

SUMMARY OF OPPORTUNITIES IN THE FUSION ENERGY SCIENCES PROGRAM

November 1999

Prepared by
the Fusion Energy Sciences Advisory Committee
for the Office of Science of the U.S. Department of Energy

Fusion Energy Sciences Advisory Committee Panel

Charles C. Baker,^{*} University of California, San Diego
Stephen O. Dean, Fusion Power Associates
N. Fisch,[†] Princeton Plasma Physics Laboratory
Jeffrey P. Freidberg,^{*} Massachusetts Institute of Technology
Richard D. Hazeltine,^{*} University of Texas
G. Kulcinski, University of Wisconsin
John D. Lindl,^{*} Lawrence Livermore National Laboratory
Gerald A. Navratil,^{*} Columbia University
Cynthia K. Phillips,^{*} Princeton Plasma Physics Laboratory
S. C. Prager, University of Wisconsin
D. Rej, Los Alamos National Laboratory
John Sheffield,^{*} Oak Ridge National Laboratory, Coordinator
J. Soures, University of Rochester
R. Stambaugh, General Atomics
R. D. Benson, ORNL Support
J. Perkins, LLNL Support
J. A. Schmidt, PPPL Support
N. A. Uckan, ORNL Support

^{*} Member of FESAC.

[†] Ex-Member of FESAC.

Fusion Energy Sciences Advisory Committee 1998–1999

Members

Dr. Charles C. Baker, University of California, San Diego
Dr. Richard J. Briggs, Science Applications International Corporation
Dr. Robert W. Conn, University of California, San Diego
Professor Jeffrey P. Freidberg, Massachusetts Institute of Technology
Dr. Katharine B. Gebbie, National Institute of Standards and Technology
Professor Richard D. Hazeltine, University of Texas at Austin
Professor Joseph A. Johnson III, Florida A&M University
Dr. John D. Lindl, Lawrence Livermore National Laboratory
Dr. Gerald A. Navratil, Columbia University
Dr. Cynthia K. Phillips, Princeton Plasma Physics Laboratory
Dr. Marshall N. Rosenbluth, University of California, San Diego
Dr. John Sheffield (Chair), Oak Ridge National Laboratory
Dr. Tony S. Taylor, General Atomics

Ex-Officios

Dr. Clement Wong (GAT), American Nuclear Society
[Previously, Dr. Wayne Houlberg (ORNL)]
Professor Miklos Porkolab (MIT), American Physical Society
Dr. Ned R. Sauthoff (PPPL), Institute of Electrical and Electronics Engineers

Staff

Robert Benson (ORNL)

PREFACE

This document has been prepared in response to a charge to the Fusion Energy Sciences Advisory Committee (FESAC) from Dr. Martha Krebs, Director of the Department of Energy's Office of Science:

... to make final a program plan for the fusion energy science program by the end of 1999 (FY). Such a program plan needs to include paths for both energy and science goals taking into account the expected overlap between them. The plan must also address the needs for both magnetic and inertial confinement options. It will have to be specific as to how the U.S. program will address the various overlaps, as well as international collaboration and funding constraints. Finally, this program plan must be based on a 'working' consensus (not unanimity) of the community, otherwise we can't move forward. Thus I am turning once again to FESAC.

I would like to ask FESAC's help in two stages. First, please prepare a report on the opportunities and the requirements of a fusion energy science program, including the technical requirements of fusion energy. In preparing the report, please consider three time-scales: near-term, e.g., 5 years; mid-term, e.g., 20 years; and the longer term. It would also be useful to have an assessment of the technical status of the various elements of the existing program. This document should not exceed 70 pages and should be completed by the end of December 1998, if at all possible. I would expect to use this work, as it progresses, as input for the upcoming SEAB review of the magnetic and Inertial Fusion Energy Programs.

A FESAC Panel was set up to prepare the document. The Panel decided to follow the approach used in the preparation of the reports from the Yergin Task Force on Strategic Energy Research and Development of June 1995 and from the National Laboratory Directors on Technology Opportunities to Reduce U.S. Greenhouse Gas Emissions of October 1997. As a first step, a two-page description of each of the main topical areas of fusion energy sciences was obtained from key researchers in that area. The descriptions give the status and prospects for each area in the near-term, midterm, and longer term, discussing both opportunities and issues. These two-pagers are published as a separate report. The two-pagers were used as background information in the preparation of this overview, *Opportunities in Fusion Energy Sciences Program*. FESAC thanks all of those who participated in this work. These two reports have been published and appear on the World Wide Web at the following URL: http://www.foe.er.doe.gov/More_HTML/FESAC_Charges_Report.html. This document is the 70-page summary of these reports. The preparation of this summary document was undertaken mainly by C. C. Baker, R. Goldston, R. D. Hazeltine, J. D. Lindl, C. K. Phillips, D. Rej, N. R. Sauthoff, and J. Sheffield.

Page Intentionally Blank

CONTENTS

	Page
PREFACE.....	iii
EXECUTIVE SUMMARY.....	vii
1. INTRODUCTION.....	1-1
1.1 THE SCIENCE OF FUSION.....	1-1
1.2 THE STRATEGIC ROLE OF FUSION ENERGY RESEARCH.....	1-1
1.3 TWO PATHWAYS TO FUSION ENERGY.....	1-2
1.4 THE DOE AND WORLD FUSION PROGRAMS.....	1-3
1.5 THE FUTURE PROGRAM.....	1-4
2. FUSION ENERGY SCIENCE AND TECHNOLOGY.....	2-1
2.1 INTRODUCTION.....	2-1
2.1.1 Fusion Fuel Cycles.....	2-1
2.1.2 Environmental and Safety Aspects of Fusion Energy Production.....	2-2
2.1.3 Progress in Fusion Energy Research.....	2-2
2.2 MAGNETIC FUSION ENERGY.....	2-4
2.2.1 Introduction.....	2-4
2.2.2 Physics of Magnetic Confinement.....	2-5
2.2.3 Path to Magnetic Fusion Energy.....	2-10
2.2.4 Opportunities in MFE.....	2-19
2.3 THE INERTIAL FUSION PATHWAY TO FUSION ENERGY.....	2-25
2.3.1 Introduction.....	2-25
2.3.2 ICF Target Physics.....	2-26
2.3.3 An IFE Development Pathway for Lasers and Ion Beams.....	2-35
2.3.4 IFE Drivers.....	2-37
2.3.5 IFE Fusion Target Concepts and Design.....	2-41
2.4 TECHNOLOGY OPPORTUNITIES.....	2-43
2.4.1 Overview and Recent Progress.....	2-43
2.4.2 The Technology Portfolio.....	2-45
2.4.3 IFE Chamber and Target Technology R&D.....	2-48
3. SCIENTIFIC CONTEXT OF FUSION RESEARCH.....	3-1
3.1 INTRODUCTION.....	3-1
3.1.1 Plasma Science.....	3-1
3.1.2 Conceptual Tools.....	3-1
3.1.3 Evolution of Fusion Science.....	3-2
3.2 MAJOR TOPICAL AREAS IN PLASMA SCIENCE.....	3-2
3.2.1 Hamiltonian Dynamics.....	3-2
3.2.2 Long Mean-Free Path Plasmas.....	3-3
3.2.3 Turbulence.....	3-3
3.2.4 Dynamo and Relaxation.....	3-4
3.2.5 Magnetic Reconnection.....	3-4
3.2.6 Wave Propagation.....	3-5
3.2.7 Nonneutral Plasmas.....	3-5
3.2.8 Electrostatic Traps.....	3-6
3.2.9 Atomic Physics.....	3-6
3.2.10 Opacity in ICE/IFE.....	3-7
3.2.11 Plasma Diagnostics.....	3-7
3.2.12 Computer Modeling of Plasma Systems.....	3-7
3.2.13 Advanced Computation.....	3-8

3.2.14	Dense Matter.....	3-9
3.2.15	High Energy Density Laboratory Astrophysics.....	3-9
3.3	MAJOR TOPICAL AREAS IN ENGINEERING SCIENCE.....	3-9
3.3.1	Bulk Materials Science.....	3-9
3.3.2	Surface Materials Science and Atomic Physics	3-10
3.3.3	Heat Transfer at Liquid/Vacuum Interfaces	3-10
3.3.4	Ablation, Radiation Gas Dynamics, and Condensation	3-11
3.3.5	Neutron and Photon Transport in Materials	3-11
3.3.6	Pebble Bed Thermomechanics.....	3-11
4.	NEAR-TERM APPLICATIONS.....	4-1
4.1	INTRODUCTION	4-1
4.2	OPPORTUNITIES.....	4-1
4.2.1	Microelectronics and Flat Panel Displays	4-1
4.2.2	Materials and Manufacturing	4-2
4.2.3	Environmental Applications.....	4-3
4.2.4	Biomedical and Food-Safety Applications.....	4-4
4.2.5	Plasma Propulsion	4-5
	LIST OF ACRONYMS.....	L-1

EXECUTIVE SUMMARY

Recent years have brought dramatic advances in the scientific understanding of fusion plasmas and in the generation of fusion power in the laboratory. Today, there is little doubt that fusion energy production is feasible. The challenge is to make fusion energy practical. As a result of the advances of the last few years, there are now exciting opportunities to optimize fusion systems so that an attractive new energy source will be available when it may be needed in the middle of the next century. The risk of conflicts arising from energy shortages and supply cutoffs, as well as the risk of severe environmental impacts from existing methods of energy production, are among the reasons to pursue these opportunities.

Fusion is a scientific and technological grand challenge. It has required the development of the entire field of high-temperature plasma physics, a field of science that contributes to the description of some 99% of the visible universe. Plasma physics also provides cross-cutting insights to related fields such as nonlinear mechanics, atomic physics, and fluid turbulence. Quality science has always been the key to optimizing fusion systems. Throughout the history of fusion energy research, the combination of exciting, challenging science and the lofty energy goal has attracted gifted young people into fusion research, many of whom have gone on to make important contributions in related scientific fields and in the commercial technology arena.

The DOE Fusion Energy Sciences Program is exploring multiple paths for optimizing fusion systems, taking advantage of both the strong international program in magnetic fusion energy and the strong DOE Defense Programs effort in inertial confinement fusion. As in other fields, the advancement of plasma science and technology requires facilities in a range of sizes, from the largest devices that press the frontier of high-temperature plasmas to smaller experiments suitable to begin the exploration of innovative ideas for fusion optimization. The very largest facilities may require international collaboration, while the smallest are natural for university-scale investigation. Specific questions of plasma science and fusion technology set both the required number and the required scale of the experimental facilities in the program.

The large international magnetic fusion program, at over a billion dollars per year, is an indication of the world-wide commitment to the development of a practical magnetic fusion power system. This global investment also provides dramatic leverage for U.S. research. Furthermore, world-wide efforts to develop low-activation materials indicate that fusion energy systems will be environmentally attractive. Extraordinary progress in understanding magnetically confined plasmas, coupled with the recent achievement of over 10 MW of fusion power production (and over 20 MJ of fusion energy), has opened up new and important research vistas. The scientific advances made on the large tokamak facilities throughout the world, and on the smaller alternate concept experiments, have spurred the development of a set of promising innovative ideas for new approaches to optimizing magnetic confinement systems. These advances have simultaneously made possible the evolutionary development of an attractive “advanced-tokamak” concept. There are today compelling, peer-reviewed, near-term opportunities for investment in innovative confinement experiments (at a range of scales), in new tools for U.S. tokamak facilities to address advanced-tokamak issues, and in collaborations on the most powerful experimental facilities overseas. These investments will enable a broad, coordinated attack on key scientific and technical issues associated with the optimization of magnetic confinement systems and the achievement of the most attractive power plant concept. In the longer term, there may also be an opportunity to undertake or participate in a burning plasma experiment, most likely in an international context. The science necessary to take this step confidently is already available. Key plasma technologies are needed to support all these efforts, and technological innovation will continue to play a critical role in ensuring the attractiveness of the ultimate fusion product.

Progress on the physics of inertial confinement fusion and construction of the National Ignition Facility (NIF) in DOE Defense Programs provide the United States with an opportunity to develop a complementary approach to fusion energy with some unique potential benefits. Separation of most of the high-technology equipment from the fusion chamber will simplify maintenance of inertial fusion systems.

The driver systems, which are external to the fusion chamber, are in some cases extensively modular so that partial redundancy could permit on-line maintenance. Some fusion chamber concepts have solid walls that are protected from neutron flux by thick fluid blankets (an idea now being pursued synergistically with magnetic fusion energy), leading to long chamber lifetime (which would reduce the need for advanced materials development) and low environmental impact. Ignition on the NIF represents a grand scientific challenge involving integration of laser-matter interaction under extreme conditions, control of hydrodynamic instabilities, radiation transport, and atomic physics—all under conditions similar to those at the center of stars. Exciting opportunities exist, in parallel with the construction and operation of NIF, to demonstrate the principles for a range of potentially attractive drivers for repetitively imploding fusion targets, to address associated fusion target chamber technologies and to examine techniques for the mass manufacture of precision targets. New innovative driver and target concepts are also being developed, providing opportunities for new science and a potentially more attractive ultimate power plant.

The strengthening of basic plasma science and technology research, for which fusion has been the single strongest driver, is another important investment opportunity. The scientific understanding of magnetically confined plasmas has helped form much of the basis for advances in the broad field of space plasma physics and for the understanding of solar and stellar magnetism. Important contributions have also been made from inertial confinement fusion to astrophysics, in particular, to the understanding of supernova explosions and the structure of dense gas planets. Commercial technological spin-offs benefiting from plasma research range from plasma etching of computer chips to satellite positioning with plasma thrusters and from lithography using extreme ultraviolet light emitted by dense plasmas to the use of lasers for non-invasive surgery.

The Department of Energy has major initiatives in advanced computational simulation both underway and proposed. Fusion Energy Science was a pioneer in the use of nationally networked supercomputing and intends to be a major participant in these new initiatives. Advanced computing power can open the way to much more detailed 3-D simulations of the wide range of magnetically confined plasma configurations, of inertial fusion capsule implosions, and of high-current ion beams. The DOE initiatives in advanced computing provide a unique opportunity to accelerate the cycle of theoretical understanding and experimental innovation in fusion energy science.

In summary, fusion energy is one of only a few truly long-term energy options. There have been dramatic recent advances in both the scientific understanding of fusion plasmas and in the generation of fusion power in the laboratory. As a consequence, there are now exciting and important opportunities for investment in magnetic fusion energy, inertial fusion energy, plasma science and technology, and advanced simulation. These opportunities address the scientific and technological grand challenge of making fusion a practical and attractive new energy source for humankind.

1. INTRODUCTION

1.1 The Science of Fusion

Since its inception in the 1950s, the vision of the fusion energy research program has been to develop a viable means of harnessing the virtually unlimited energy stored in the nuclei of light atoms. This vision grew out of the recognition that the immense power radiated by the sun is fueled by nuclear fusion in its hot core. High temperatures are a prerequisite for driving significant fusion reactions.

Note that at low enough temperatures nearly all materials are solids. As the temperature is raised they become liquids, and as molecular bonds are disrupted at further increased temperature, they become gases. Solids, liquids, and gases are the most common forms of matter at the temperatures normally found on earth. However, around 10,000 degrees Celsius or more, collisions in the gas begin to release electrons from the atoms and molecules. Further collisions create a mixture of a large number of free electrons and positively charged ions. This fascinating fourth state of matter is known as plasma. It is only in this fourth state of matter that the nuclei of two light atoms can fuse, releasing the excess energy that was needed to separately bind each of the original two nuclei. Because the nuclei of atoms carry a net positive electric charge, they repel each other. Hydrogenic nuclei, such as deuterium and tritium, must be heated to approximately 100 million degrees Celsius to overcome this electric repulsion and fuse.

A plasma is the most pervasive form of visible matter in the universe, comprising the major constituent of stars as well as the interstellar medium (see cover). Only in exceptional environments such as the surface of a cool planet like the Earth can other forms of matter dominate. Laboratory-generated plasmas are utilized in a number of diverse industrial applications: the sterilization of certain types of medical supplies; the plasma glow discharge as a mainstay of the electronic chip manufacturing industry; and plasma thrusters for position control of satellites.

It is in the pursuit of fusion energy that plasma physics plays one of its most critical roles. The U.S. campaign for controlled thermonuclear power that arose from Project Sherwood in the 1950s has produced a dramatic flowering of the theory of plasmas—including large-scale computer simulations—as well as a wide range of laboratory devices that can be used to create, heat, confine, and study plasmas.

Most importantly, the plasma conditions of fusion plasmas overlap those of both astrophysical and earth-bound plasmas, allowing research in the various areas to be mutually supportive. The resulting gains in the understanding of plasmas have brought enormous benefit to a wide range of fields of science and applied technology and have brought us many steps closer to achieving a controlled thermonuclear burn of tremendous practical value to humankind.

1.2 The Strategic Role of Fusion Energy Research

Energy availability has always played an essential role in socioeconomic development. The stability of each country, and of all countries together, is dependent on the continued availability of sufficient, reasonably priced energy. Per capita energy consumption in the various regions of the world is correlated with the level of wealth, general health, and education in each region. World energy consumption has increased dramatically over time and is projected to continue increasing, in particular to meet the need for greater per capita energy consumption in the developing world. The growth in energy demand will be exacerbated by the almost doubling of the world's population expected to occur, mainly in the developing countries, within the next 50 years. The fraction of energy used in the form of electrical power is also expected to grow during this time period.

While there are significant global resources of fossil and fission fuels and substantial opportunities for exploiting renewable energies, numerous countries and some of the developing areas experiencing major population growth are not well endowed with the required resources. Further, utilization of some resources may be limited because of environmental impact. A sustainable development path requires that the industrialized countries develop a range of safe and environmentally benign approaches applicable in the near, medium, and long term. Continuing to meet the world's long-term energy requirements raises challenges well beyond the time horizon of market investment and hence calls for public investment. It is becoming increasingly apparent that by continuing to burn fossil fuels even at the present rate, without substantial mitigation of the carbon dioxide emissions, mankind is conducting a major experiment with the atmosphere, the outcome of which is uncertain but fraught with severe risks. Prudence requires having in place an energy research and development (R&D) effort designed to expand the array of technological options available for constraining carbon dioxide emissions without severe economic and social cost.

Fusion offers a safe, long-term source of energy with abundant resources and major environmental advantages. The basic fuels for fusion—deuterium and the lithium that is used to generate tritium—are plentifully available. Even the most unlikely accident would not require public evacuation. During operation, there would be virtually no contributions to greenhouse gases or acidic emissions. With the successful development of materials, tailored to minimize induced radioactivity, the wastes from fusion power would not require isolation from the environment beyond 100 years and could be recycled on site.

With successful progress in fusion science and with the development of the necessary technologies, fusion is expected to have costs in the same range as other long-term energy sources, and fusion power plants could provide a substantial fraction of world electricity needs. With appropriate research support, fusion will be able to provide an attractive energy option to society in the middle of the next century. Fusion could begin to be deployed at a time when the utilization of other sources of energy is uncertain and when the climate issue is likely to have become more critical than today. Accordingly fusion energy science and ultimately fusion technology should be pursued vigorously in the U.S. and world programs.

1.3 Two Pathways to Fusion Energy

Two complementary pathways toward a fusion energy power plant have emerged, both of which offer the potential basis for a viable fusion energy power plant. In one approach, Magnetic Fusion Energy (MFE), the tendency of the plasma charged particles to follow along magnetic field lines, is exploited in the creation of “magnetic bottles.” Magnetic fields restrict the outward motion of the charged particles and the plasma that they constitute. By curving the magnetic field lines into a closed form (making a doughnutlike *toroidal* configuration), a plasma can be confined while it is heated to the temperature needed for a steady-state, self-sustaining fusion burn to be initiated (see Fig. 1.1), like in the sun.

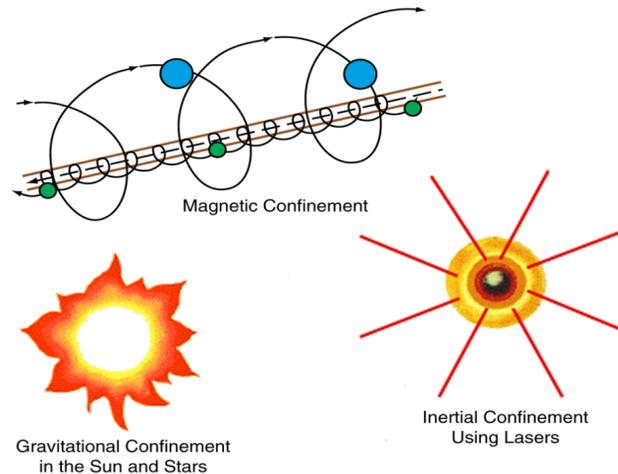


Fig. 1.1. Fusion plasma confinement approaches.

With the advent of high-powered lasers in the 1970s, a second approach emerged—Inertial Fusion Energy (IFE). In IFE, a tiny hollow sphere of fusion material is rapidly imploded to very high density (see Fig. 1.1). A central low-density region, comprising a small percentage of the fuel, is heated to fusion temperatures and initiates an outwardly propagating burn wave that fuses a significant fraction of the remaining fuel, during the brief period while the pellet is still held together by its own inertia. This

approach utilizes physical processes similar to those present in thermonuclear explosions. Steady power production is achieved through rapid, repetitive fusion microexplosions.

During the evolution of fusion research, diverse plasma confinement concepts have been proposed and studied, and a number of these have evolved into promising approaches for fusion energy production. Because most of the concepts fit into a small number of general categories, each can be considered not only as a potential power plant in its own right but also as a contributor to the general scientific knowledge base and to the building blocks needed for developing an energy system. There are many ways of characterizing fusion concepts—steady-state or pulsed, externally controlled or self-ordered, symmetric or non-symmetric, and thermal or non-thermal energy distribution. Concepts have been conceived with various combinations of these characteristics. Nevertheless, in the simplest terms, there are two main approaches to fusion energy. Each has two subcategories, as characterized in Fig. 1.2.

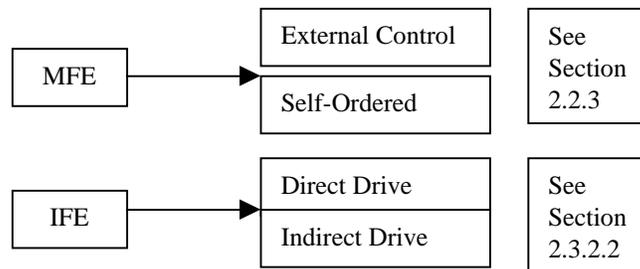


Fig. 1.2. Main approaches to fusion.

At this stage in the development of fusion energy, it is premature to choose between these two pathways to commercial fusion energy. Substantial progress has been made along both pathways toward realizing an energy gain from fusion of deuterium and tritium at temperatures around 100 million degrees. In MFE production of up to 20 MJ/pulse has been obtained with a fusion gain of 0.6, while in IFE more modest energy production (~400 J/pulse, and gain = 0.01) has been obtained in laboratory experiments, with higher energy production in classified underground tests. Both may lead to attractive fusion energy options, and within each approach specific technical implementations need to be investigated to provide the optimal system. Indeed, a lesson can be drawn from the history of rocket science, in which parallel development of both solid and liquid boosters was pursued.

1.4 The DOE and World Fusion Programs

The Office of Fusion Energy Sciences (OFES) in the Science element of the Department of Energy (DOE) leads the U.S. research program in fusion energy sciences and, in collaboration with the National Science Foundation (NSF), the effort on basic research in plasma science. The OFES program has focused primarily on MFE concepts. Inertial confinement fusion (ICF) research is supported primarily by the ICF program within the Defense Programs (DP) element of DOE, for national security needs. In the late 1980s, responsibility for scientific research into aspects of IFE, that are not relevant to national security needs and that do not involve classified information, were consolidated within OFES. In FY 1999, the OFES budget totals \$223M, with \$210M spent on MFE and MFE-related basic research and about \$10M on non-defense aspects of IFE. The DP program on ICF, focused on scientific stockpile stewardship, is funded at \$508M in FY 1999, with \$284M for the construction of the National Ignition Facility (NIF) and \$224M spent on the base ICF program, laser development, and related basic research.

The U.S. fusion research effort is imbedded in a larger international program. The international research program in MFE is presently supported at over \$1 billion annually and represents enormous potential leverage for the U.S. domestic program. Currently Europe and Japan each invest respectively 2.5 and 1.5 times the resources in magnetic fusion energy research as does the United States. Each operate billion-dollar class tokamak experiments. Japan has just completed construction of a similar-scale stellarator device, while Germany has such a device under construction. Interesting, but much smaller, IFE programs

exist in Japan and Europe. The Japanese IFE program focuses on laser-driven fusion, and the German program focuses on ion beams. The French have initiated construction of a NIF-like laser system.

1.5 The Future Program

Through the middle part of this decade, the OFES fusion energy program was focused nearly exclusively on the fusion energy goal, with resources devoted primarily to developing the tokamak concept to the stage at which burning plasma physics could be investigated. About 3 years ago, severe constraints on the availability of federal research funds coupled with a perceived near-term abundance of energy resources resulted in a major budget cut for the fusion energy research funded by DOE–OFES. The Fusion Energy Advisory Committee (FEAC) conducted a review of the program and concluded that, as a result of the constrained budgets, the program should be redirected away from “the expensive development path to a fusion power plant” toward a program focused “on the less costly critical basic science and technology foundations.” The directions for the future of the DOE program, as recommended by FEAC, follow.

MISSION: Advance plasma science, fusion science, and fusion technology—the knowledge base needed for an economically and environmentally attractive fusion energy source.

POLICY GOALS:

- **Advance plasma science in pursuit of national science and technology goals.**
- **Develop fusion science, technology, and plasma confinement innovations as the central theme of the domestic program.**
- **Pursue fusion energy science and technology as a partner in the international effort.**

While fusion energy science is supported within the United States as a science program, it is necessary nonetheless to consider this program in the context of its ultimate goal, the ability to proceed to the development of a practical energy source. The structure for the development aspect of fusion is provided by the roadmap, shown in Fig. 1.3, prepared by members of the U.S. fusion community. It includes the portfolio MFE and IFE approaches within a unified framework, designed to build on the successes in each of these programs—the **experimental results of the last decade indicate that fusion can be an energy source, and the challenge now is to optimize the science to make each stage of development, as well as the ultimate product, practical and affordable.** This is the central focus of the roadmap and the associated portfolio.

Within the fusion Portfolio Approach, confinement configurations advance through a series of stages of experimental development. These stages are “Concept Exploration” and “Proof-of-Principle,” followed by “Performance Extension.” Success in these stages then should lead to a stage of “Fusion Energy Development” and “Fusion Energy Demonstration.” At each stage of development the opportunities increase for developing the building blocks of a fusion power plant and for increasing scientific understanding. The facilities have, successively, a greater range and capability (dimensional and dimensionless parameters) for exploring plasma conditions and are more demanding on technology requirements. Briefly, the steps are as follows:

- **Concept Exploration** is typically at <\$5M/year and involves the investigation of basic characteristics. Experiments cover a small range of plasma parameters (e.g., at <1 keV) and have few controls and diagnostics.
- **Proof-of-Principle** is the lowest cost program (\$5M to \$30M/year) to develop an integrated understanding of the basic science of a concept. Well-diagnosed and controlled experiments are large enough to cover a fairly wide range of plasma parameters, with temperatures of a few kiloelectron volts, and some dimensionless parameters in the power plant range.
- **Performance Extension** programs explore the physics of the concept at or near fusion-relevant regimes. Experiments have a very large range of parameters and temperatures >5 keV, with most dimensionless parameters in the power plant range. Diagnostics and controls are extensive.
- **Fusion Energy Development** program develops the technical basis for advancing the concept to the power plant level in the full fusion environment. It includes devices with power-plant-like fusion gain, integrated fusion test systems, and neutron sources.
- **Demonstration Power Plant** is constructed and operated to convince electric power producers, industry, and the public that fusion is ready for commercialization.

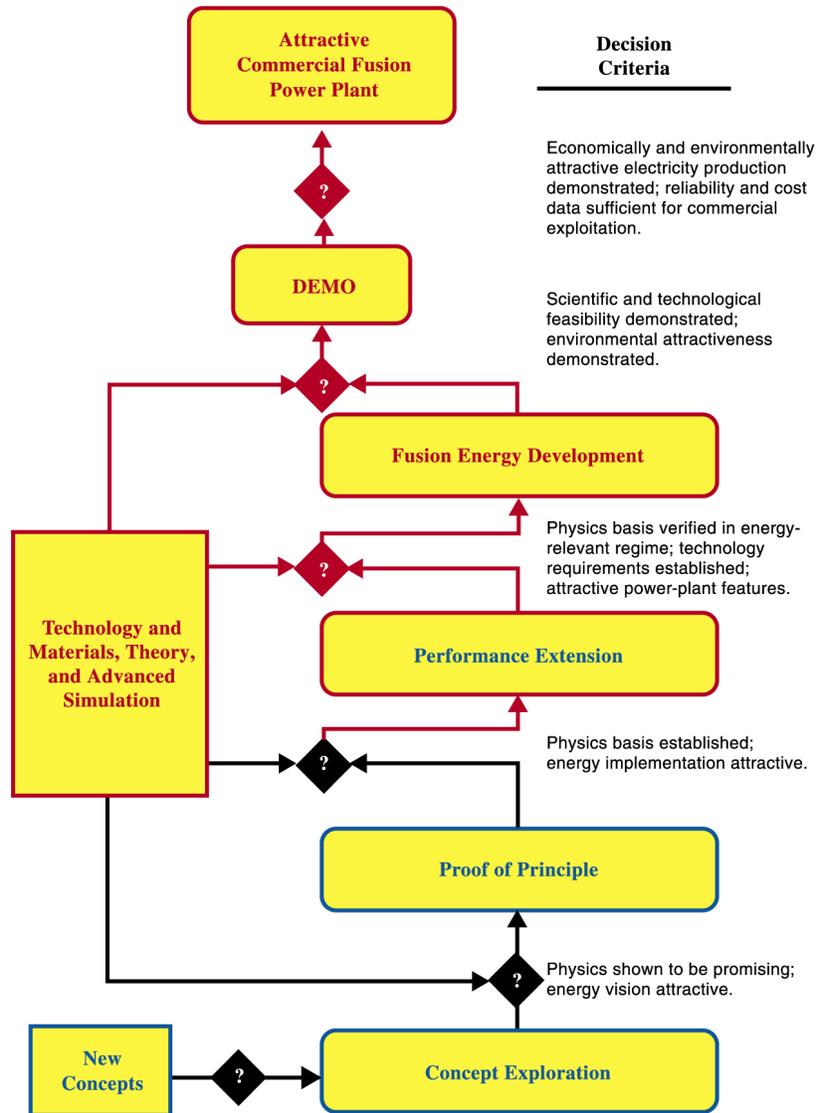


Fig. 1.3. Roadmap for fusion energy.

Page Intentionally Blank

2. FUSION ENERGY SCIENCE AND TECHNOLOGY

2.1 Introduction

Fusion is one of only a very few long-term energy options. The mission of the fusion energy science program is to develop the knowledge base for an economically and environmentally attractive energy source that could be available in the middle of the next century. Fusion energy research also provides important near-term scientific and technological benefits to society.

2.1.1 Fusion Fuel Cycles

The rate of fusion production for deuterium (D) and tritium (T) ions becomes substantial for temperatures above roughly 50 million degrees (~5 keV). Each fusion reaction produces 17.6 MeV of energy per reaction—3.5 MeV associated with an alpha particle and 14.1 MeV with a neutron. The optimum ion temperature for maximizing D-T fusion production is around 10 keV. Fusion of deuterium with deuterium and deuterium with helium-3 (^3He) has a substantially lower rate, for a given plasma pressure, and needs higher temperatures of around 30 keV (Table 2.1). Therefore, to date, most studies have concentrated on the D-T cycle, because high fusion power densities are much easier to achieve. Because tritium is not available naturally, it will be necessary to generate it in a fusion power plant to sustain the fusion cycle. Analysis and research indicate that adequate generation may be achieved by absorbing the fusion neutrons in a blanket surrounding the plasma, which contains lithium. Note that both the D-T cycle and the D-D cycle produce energetic neutrons, 80% neutron power fraction for D-T and ~50% for D-D fuel! Because these neutrons damage the structures surrounding the plasma, an important R&D program has been devoted to developing radiation-resistant and low-activation structural materials.

Table 2.1. Fusion reactions

Fusion reactions	Energy in ions (MeV)	Total (MeV)
$\text{D} + \text{T} \rightarrow ^4\text{He}(3.2 \text{ MeV}) + \text{n}(14.06 \text{ MeV})$	3.52	17.58
$\text{D} + \text{D} \rightarrow ^3\text{He}(0.82 \text{ MeV}) + \text{n}(2.45 \text{ MeV})$	0.82	3.27
$\text{D} + \text{D} \rightarrow \text{T}(1.01 \text{ MeV}) + \text{p}(3.03 \text{ MeV})$	4.04	4.04
$\text{D} + ^3\text{He} \rightarrow ^4\text{He}(3.67 \text{ MeV}) + \text{p}(14.67 \text{ MeV})$	18.34	18.34
$^3\text{He} + ^3\text{He} \rightarrow 2\text{p}(8.57 \text{ MeV}) + ^4\text{He}(4.29 \text{ MeV})$	12.86	12.86
$\text{p} + ^{11}\text{B} \rightarrow 3^4\text{He}(8.66 \text{ MeV})$	8.66	8.66

The anticipated engineering, safety, and environmental advantages of reduced neutron production motivate research on the use of “advanced” fuel cycles, which produce fewer neutrons despite the much greater physics obstacles compared to D-T. For example at any fixed plasma pressure, the advanced fuels produce less than 2% as much fusion power density as D-T, potentially leading to quite large systems. The two advanced fuels generally considered most important are D- ^3He (1–5% of fusion power in neutrons from D-D reactions) and p- ^{11}B (no neutrons); see Fig. 2.1. (Note that 50% of the tritium produced by D-D reactions is assumed to react with deuterium before leaving the plasma.) Although p- ^{11}B and ^3He - ^3He produce no neutrons, calculations indicate that plasmas with comparable electron and ion temperatures produce bremsstrahlung radiation power very close to the total fusion power. Energy production from p- ^{11}B , for example, will require low charged-particle heat transport and low nonbremsstrahlung radiation losses as well as rapid expulsion of fusion reaction products. It will be a challenge to develop more than a heavily driven, low-gain energy amplifier.

The requirements for IFE make it unlikely that advanced fuels could be used for net energy production, and they should only be considered for use in MFE. Sufficient ^3He has been identified on Earth to

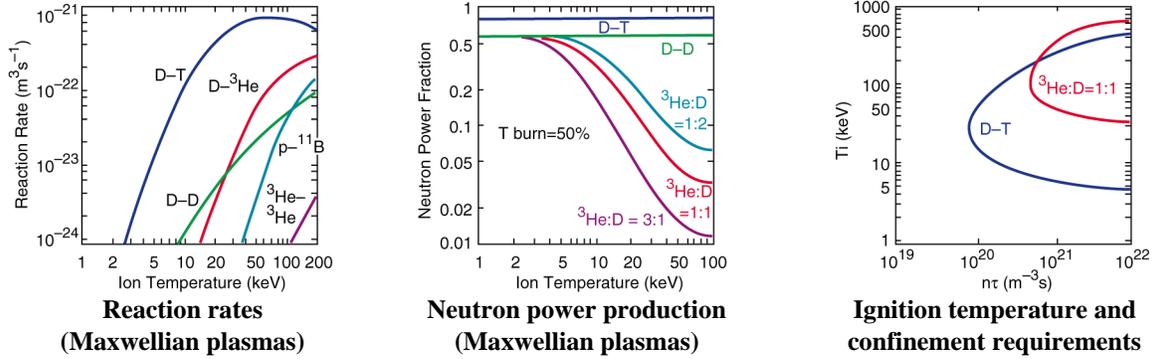


Fig. 2.1. Characteristics of fusion reactions.

conduct a D- ^3He fusion research program up to and including the first 1000-MW(e) power plant. Advanced fuels such as p, D, and ^{11}B are plentiful on Earth, but large-scale deployment of D- ^3He power plants would require developing the large resource ($\sim 10^9$ kg) on the lunar surface.

2.1.2 Environmental and Safety Aspects of Fusion Energy Production

The environmental and safety characteristics of fusion power production offer the prospect of significant advantages over present major sources of energy. The basic fuels for fusion — deuterium and the lithium that is used to generate the tritium fuel—are plentifully available, and there would be virtually no contributions to greenhouse gases or acidic emissions. However, these benefits will not come automatically. Tritium and neutron activation products in fusion power plants, using the D-T or D-D fuel cycles, will present significant radiological hazards. The safety-conscious choice of materials can result in minimization of activation products and tritium inventories. The radiological inventory in a fusion power plant can be much lower than that in an equivalent fission reactor, and the time-integrated biological hazard potential can be lower by factors approaching 100,000. The stored energy of the fusion fuel contained in the plasma, equivalent to only a few minutes of power production, is vastly less than that in a fission power system, whose active fuel inventory is typically adequate for 1 to 2 years of operation. Further, the use of low-activation materials will allow fusion components to be recycled or disposed of as low-level waste and not be a burden to future generations. A promising approach using liquid walls, which surround the fusing plasma, is currently under study and has the potential to provide an additional reduction in activation products.

The comparison of the decay of the radioactive inventory in a reference fission reactor and reference fusion power plants, using low-activation wall materials, in Fig. 2.2, shows the potential advantage of fusion power. After a period of 100 years, the radioactivity remaining from a fusion system can be millions of times less than that from fission. In the simplest terms, this translates into no need for the storage of waste over the geological time periods contemplated for repositories such as Yucca Mountain.

2.1.3 Progress in Fusion Energy Research

The status of fusion energy research is summarized in Fig. 2.3; it shows the present and historical levels of achievement for D-D and D-T plasmas in overall energy gain, Q , and the Lawson $nT\tau$ figure of merit, relative to the requirements for a fusion energy source ($Q > 10$). There has been considerable progress in the past 20 years in advancing to near break-even conditions in D-T plasmas, setting the stage for reaching the fusion energy range of $Q > 10$ in the next generation of experiments in both MFE and IFE.

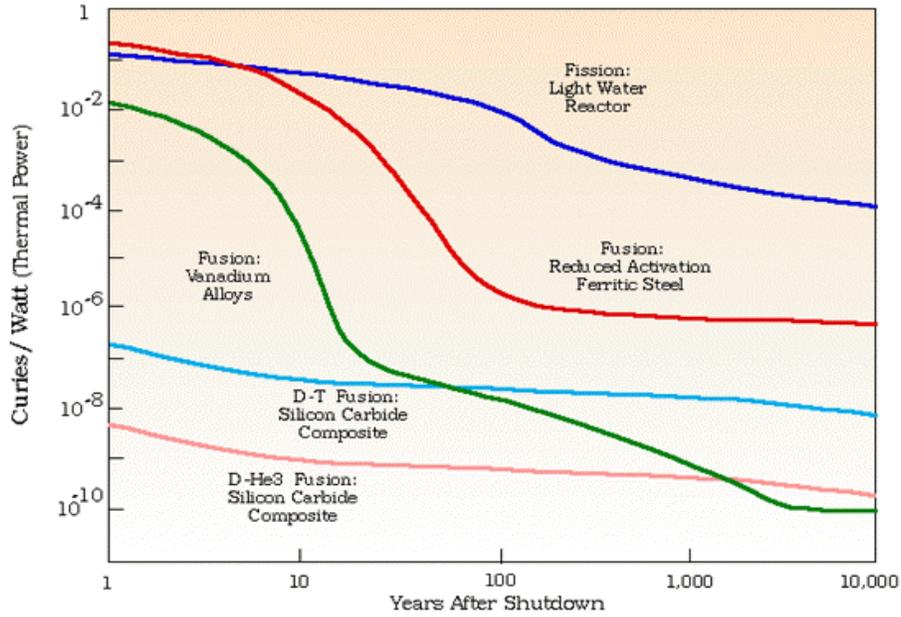


Fig. 2.2. Comparison of fission and fusion radioactivity after shutdown.

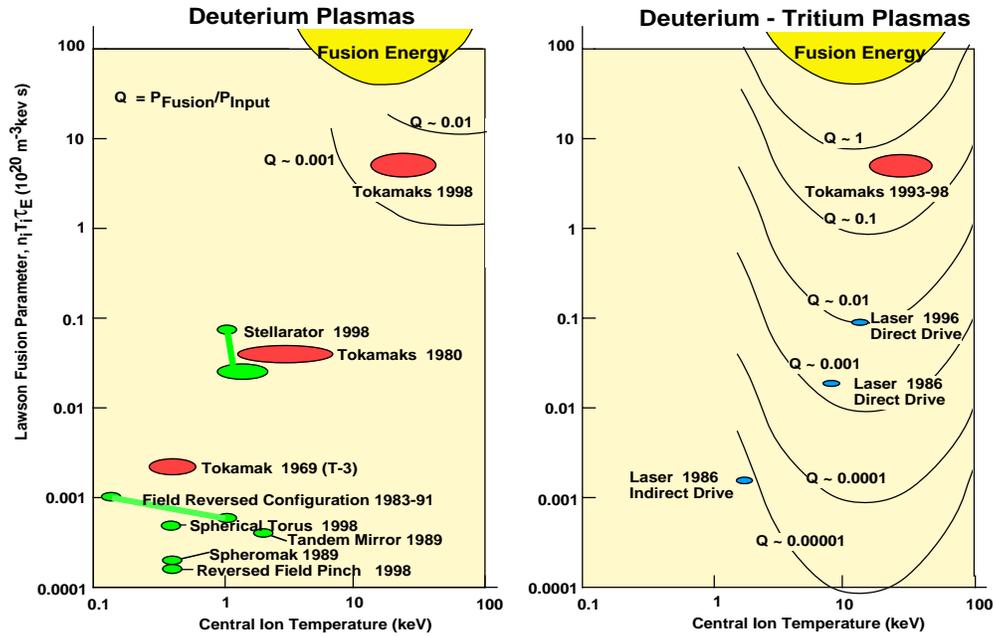


Fig. 2.3. Summary of progress in fusion energy gain achieved in experiments.

2.2 Magnetic Fusion Energy

2.2.1 Introduction

Power Plant. An MFE power plant, using D-T fuel, is shown schematically in Fig. 2.4. It consists of five major components surrounding the magnetically confined fusion plasma core including (i) a magnetic coil set for generation and control of the confining magnetic field; (ii) plasma heating and current drive systems; (iii) a first wall and blanket system for energy recovery and tritium fuel breeding; (iv) power and particle exhaust/recovery system; and (v) a steam plant to convert the fusion-generated energy recovered as heat in the blanket into electricity. While both pulsed and steady-state MFE plasma core concepts exist as candidates for power plants, the leading approaches seek to exploit the benefits of continuous operation coupled with low levels of recirculating power ($\eta_R < 20\%$).

Numerous studies have been made of potential fusion power plants, culminating most recently in the series of studies by the ARIES Team.* These systems studies involve detailed, bottoms-up designs and cost estimates for a number of approaches to MFE. The studies are valuable in identifying the critical plasma core and fusion technology issues which affect plant economics, availability and reliability, safety, and environmental impact.

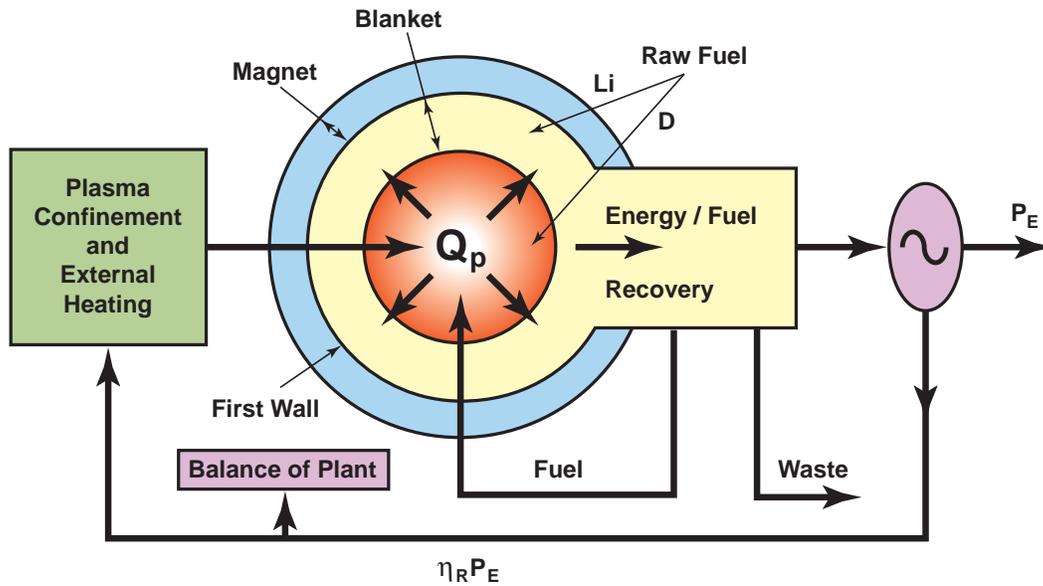


Fig. 2.4. Schematic diagram of an MFE power plant.

The main features of such plants are captured in the equation for the net electric power produced.

$$P_{\text{net}} = (f_{\text{bl}} \cdot f_{\text{te}}) \cdot P_{\text{fus}} [1 - \eta_R] \text{ MW}_e$$

Nuclear technologies
Balance of plant
Plasma core
Plasma technology

Here, P_{net} (MW_e) is the net electric power output from the power plant; f_{bl} is the exothermic energy gain in the breeding blanket (typically about 1.10); f_{te} is the thermal to electric conversion efficiency; P_{fus}

*F. Najmabadi et al., *Starlite Study*, University of California–San Diego Report UCSD-ENG-005, 1997.

(MW_f) is the fusion power produced; and η_R is the fraction of power that is recirculated to run the power plant.

Plasma core. Since its inception in the 1950s, the MFE program has focussed primarily on the physics of the plasma core. It is in the plasma core that the interplay of plasma confinement, stability, density, and temperature must be optimized in order to provide sufficient fusion power production and low recirculating power requirements to sustain the plasma. A principal issue is maintaining a nearly pure D-T plasma, and a magnetic divertor is the preferred option for this task.

The recirculating power consists of two parts: the power recirculated to operate the fusion device and its plasma and the conventional balance of plant, cooling pump power, instrumentation and controls, air conditioning, etc. In the magnetic fusion case, the fusion device requires power mainly for the magnets (generally superconducting to reduce power demand) and their cooling, for plasma heating and current drive, and for fueling and exhaust gas handling systems.

Nuclear technologies. The tritium-breeding blanket, neutron shield, and material wall that faces the plasma handle the bulk of the neutron energy and heat the cooling fluid for electricity generation. With the tritium plant and equipment for maintenance and radioactive materials handling, they are the nuclear technologies.

Materials. The development of radiation-resistant materials is an important part of fusion energy R&D. Good progress has been made in understanding the science of optimizing materials to handle the intense flux of 14-MeV neutrons generated in the D-T plasma. There is also understanding of which elements are preferred for making these materials, to minimize induced radioactivity. However, more work is needed to develop and demonstrate materials with the ability to handle a high fluence (>15 MW·y/m² of 14-MeV neutrons).

Safety and environmental concerns are a major driver for R&D and design, for example, leading to an emphasis on low-activation materials and extensive tritium systems testing.

2.2.2 Physics of Magnetic Confinement

The requirement for fusion energy production in a magnetically confined plasma in steady-state is set by the nuclear cross-section for fusion reactions, σ_f , which determines the fusion power production, and the thermal insulation provided by the confining magnetic field, which determines in large part the power needed to sustain the plasma. The fusion power density, p_f , produced by a D-T plasma of density, n_{DT} , is given by

$$p_f = n_D n_T \langle \sigma_f v \rangle W_f ,$$

where $\langle \sigma_f v \rangle$ is an average over a Maxwellian distribution of the D-T fuel velocity times the fusion cross section, and W_f is the energy release per fusion reaction (17.6 MeV for D-T).

We can rewrite p_f as

$$p_f = n_D n_T T^2 [\langle \sigma_f v \rangle / T^2 W_f] \propto p^2 \propto \beta^2 B^4 .$$

Since $\langle \sigma_f v \rangle / T^2$ is maximized, and therefore roughly constant in the range of 10 keV to 25 keV, the fusion power density in this temperature range depends only on the square of the plasma pressure, p . Because equilibrium and stability requirements limit the maximum ratio of plasma pressure to applied magnetic field pressure, $B^2/2\mu_0$, it is useful to express the fusion power density in terms of β , which is the dimensionless ratio of the pressure in the plasma, $p = nT$, and the magnetic pressure, $B^2/2\mu_0$. The plasma β is a measure of the efficiency with which the applied field is used to confine the plasma.

The power density needed to sustain this rate of fusion energy production, p_{loss} , can be related to the thermal insulation provided by the confining magnetic field. The timescale for global energy loss, $\tau_E \sim a^2/\chi$ is determined by the thermal diffusivity, χ , and a characteristic linear dimension, a , of the plasma perpendicular to the magnetic field. Hence,

$$p_{\text{loss}} = 3/2 nTV/\tau_E .$$

To maintain steady-state operation in the fusion plasma core, this level of P_{loss} must be offset by a combination of externally supplied heating power, P_{ext} , and the self-heating of the fusion plasma due to the

slowing down of the electrically charged fusion reaction products which make up a fraction, f_c , of the total fusion power. The energy gain, Q , of the fusion plasma is then simply the ratio of $P_{\text{fus}}/P_{\text{ext}}$.

$$Q \sim n_D n_T T^2 W_{fT_E/nT} \sim nT\tau_E .$$

Practical magnetic fusion power plants require Q to be large (typically >15), which implies that $P_{\text{loss}} \sim f_c P_{\text{fus}}$. Therefore, a minimum value of $\rho\tau_E$ or $nT\tau_E$ must be achieved ($\sim 10^{22} \text{ m}^{-3} \text{ keVs}$ for D-T fuel). Since $nT\tau_E \propto \beta/\chi [a^2 B^2]$, the ratio of β/χ and the magnitude of B applied to confine the plasma determine both the physical size of the fusion plasma core and the total fusion power output, P_{fus} .

2.2.2.1 Plasma Science Areas in MFE

The National Research Council in its 1995 report on the field of plasma science* divided the field into four broad areas, each of which contains critical scientific issues which must be addressed to reach the goal of practical magnetic fusion energy. These four areas are

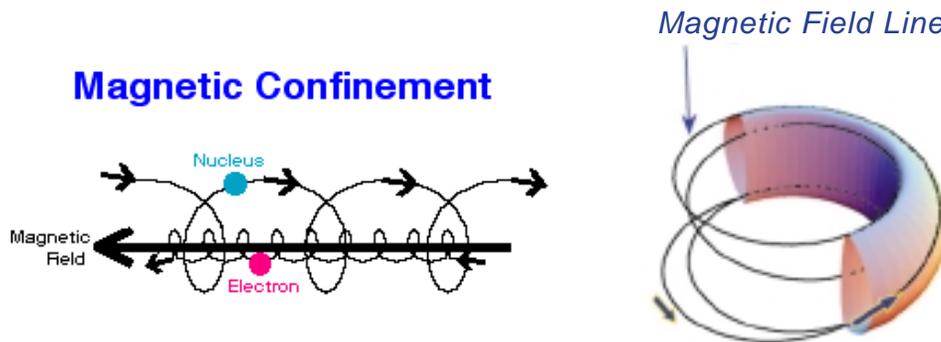
- **Transport and Turbulence:** energy, particle, and momentum transport;
- **Magnetohydrodynamics (MHD):** equilibrium, stability, magnetic reconnection, dynamo physics;
- **Wave-Particle Interactions:** plasma heating and current drive; and
- **Plasma Wall Interactions, Sheaths, and Boundary Layers.**

Transport and Turbulence

Major Research Challenge: *What are the fundamental causes of heat loss in magnetically confined plasmas, and how can heat losses be controlled, in order to minimize the required size of a fusion power system?*

Magnetic fields constrain charged particles to execute cyclotron and drift motion in the plane perpendicular to \mathbf{B} while allowing them to move relatively freely along the magnetic field lines as a consequence of the Lorentz force law of electrodynamics [$\mathbf{F} = q(\mathbf{E} + \mathbf{v} \times \mathbf{B})$] [see Fig. 2.5(a)]. This relatively unconstrained flow along the magnetic field leads to a large thermal conductivity in the direction parallel to \mathbf{B} (especially for the more mobile electrons in the plasma). For this reason, configurations that allow the hot confined plasma to be connected along “open” magnetic field lines to a material boundary have largely been abandoned in favor of toroidal systems which produce a nested set of magnetic surfaces [shown schematically in Fig. 2.5(b)] where the motion along the magnetic field is confined to a closed toroidal surface. The dominant loss mechanisms from a fusion plasma core include synchrotron radiation, bremsstrahlung, and thermal conduction and convection.

While the first two are well understood quantitatively, the third and fourth remain outstanding research problems in the field. Classically, the Coulomb collisions between charged particles will lead to cross-field thermal transport. A first-principles calculation of particle, heat, and momentum transport in a



*Plasma Science Report, National Research Council, 1995.

(a)

(b)

Fig. 2.5. Schematic illustration of (a) the motion of a charged particle in a magnetic field and (b) the nested magnetic surfaces in a toroidal configuration.

toroidal magnetic confinement system (neoclassical theory) was well established in the 1970s. Processes dominated by flow along the magnetic field lines, such as electrical resistivity and the bootstrap current, have confirmed major elements of neoclassical theory. However, the level of neoclassical particle and energy transport is quite small at typical fusion plasma parameters in a toroidal system of $T \sim 10$ keV, $n \sim 10^{20} \text{ m}^{-3}$, and $B \sim 5$ T, with neoclassical predictions for the thermal conductivity, $\chi < 0.1 \text{ m}^2/\text{s}$. This low level of transport has been rarely observed in experiments, since gyro-radius scale-length plasma turbulence typically becomes the dominant energy loss mechanism from the plasma core, resulting in an increase of the value of χ to the range of $1 \text{ m}^2/\text{s}$ to $10 \text{ m}^2/\text{s}$.

For more than 15 years now, regression analysis of the large database of confinement results from fusion experiments around the world that span a wide range of critical plasma parameters (B , n , T , etc.) has led to the development of empirical confinement scaling laws which are predictive for moderate extensions beyond the range of the database; see Sect. 2.2.3.1 (Fig. 2.10). Models for transport due to plasma turbulence have now become reasonably predictive as well. This area of research is presently in transition from the use of well-established empirical scaling laws to predictive, first-principle models of plasma transport (see Sect. 2.4).

Recent Major Scientific Advance: *Strong plasma flow velocity shear can greatly suppress the level of plasma turbulence, with a consequent reduction in the ion thermal conductivity to as low as $0.1 \text{ m}^2/\text{s}$, in line both with theoretical predictions on turbulence suppression and with the predictions for collisional transport as shown for results from the DIII-D tokamak in Fig. 2.6.* A key research question being pursued in a number of toroidal configurations is how to exploit this particular discovery to produce significantly improved global energy confinement and pressure profiles that have improved MHD stability properties.

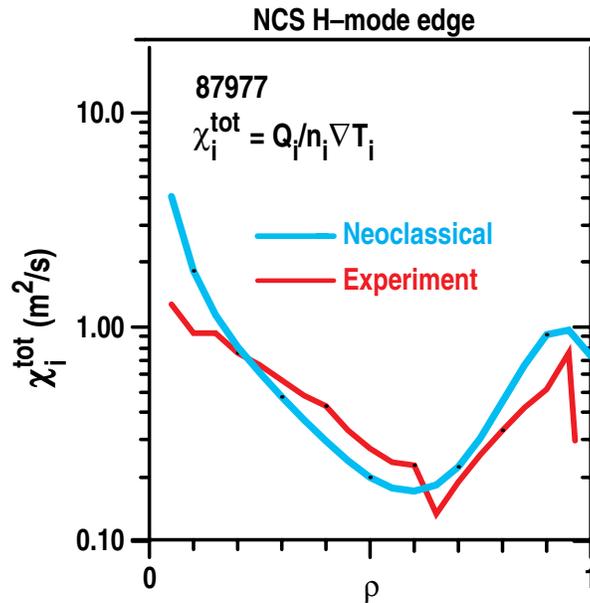


Fig. 2.6. Reduction in ion thermal conduction to neoclassical levels by suppression of the level of plasma turbulence.

MHD

Major Research Challenge: *What are the fundamental causes and nonlinear consequences of plasma pressure limits in magnetically confined plasma systems, and how can a fusion system's plasma pressure and hence power density be optimized, with minimum off-normal events?*

It is the MHD stability properties of the fusion plasma configuration that determine the maximum β , beyond which the configuration becomes unstable. The theory of ideal MHD which treats the plasma as a

perfect electrical conductor is well established and provides quantitative prediction, within 10% to 20%, for equilibrium and stability against global modes external to the plasma. The use of ideal MHD and resistive MHD (which includes finite plasma electrical conductivity) is reasonably predictive for analysis of stability against modes largely internal to the plasma, but there remain some open issues (see Sects. 4.2.4 and 4.2.5). The achievement of this present state of maturity in the field of MHD was only possible through the development of large-scale two- and three-dimensional (2- and 3-D) computational models, of highly accurate diagnostic measurement techniques for determination of the pressure and magnetic field distribution inside a fusion plasma, and the development and deployment of high-power plasma heating systems. The MHD modeling work drove the pioneering effort by the MFE program in the 1970s and 1980s to create the Magnetic Fusion Energy Computer Center (MFECC) and its successor, the National Energy Research Supercomputer Center (NERSC).

The primary phenomena which limit the β of toroidal MFE configurations are: (i) long-wavelength ($\lambda = R/n$, where $n = 0$ to 3) displacements of the plasma driven by current and pressure gradients; (ii) short-wavelength modes ($n \rightarrow \infty$) driven by pressure gradients; and (iii) magnetic reconnection (tearing modes) which forms magnetic island structures in toroidal plasmas. Each of these phenomena are now well understood, and each has been or is being addressed. The $n = 0$ mode is now routinely stabilized by combination of a resistive wall and active feedback control. This approach is now being extended to the general class of $n = 1$ modes. High- n modes can be stabilized by control of the local magnetic shear. This shear stabilization is the basis for the phenomenon of “second stability” that was predicted in the late 1970s and verified experimentally in the mid-1980s. Finally, magnetic reconnection and its consequences remain important issues for toroidal plasma, with new experiments utilizing active feedback and current profile control now being carried out.

Recent Major Scientific Advance: *Volume averaged toroidal β of 40% and central toroidal β of 100% were achieved on a Concept Exploration (CE) scale Spherical Torus, consistent with theory (see Fig. 2.7).*

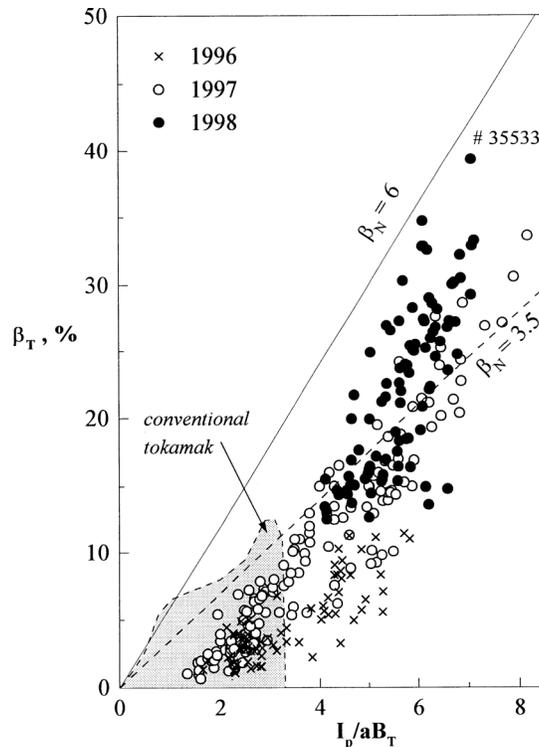


Fig. 2.7. Volume averaged toroidal β achieved in the CE stage START experiment at Culham Laboratories plotted vs the plasma current normalized to the minor radius, a , times the toroidal magnetic field, B_T .

Wave-Particle Interactions

Major Research Challenge: *What are the fundamental causes and nonlinear consequences of wave interactions with non-thermal particles, which can be used both to minimize any negative consequences of fusion products in magnetically confined plasmas, and ultimately to take advantage of the free energy represented by the fusion product population?*

The heating of magnetically confined plasmas by ohmic dissipation, radio frequency (RF) waves, and energetic neutral hydrogen atom beams is well understood. Systems that reliably deliver 20 MW to 40 MW of power to a fusion reactor regime plasma for up to 10 s have been used in research for many years and are routinely used to heat plasmas to fusion reactor conditions ($T \sim 10$ keV) in experiments around the world.

The non-inductive sustainment of the plasma electric currents needed for maintaining a steady-state toroidal plasma equilibrium has been demonstrated at modest power and plasma performance levels with a wide range of RF techniques. The most developed system, lower hybrid current drive (LHCD), is based on directed momentum input from externally-launched waves in the lower hybrid range of frequencies. The longest duration experiments using LHCD are now able to sustain multi-kiloelectron-volt plasmas for several minutes in large devices, while a smaller experiment with superconducting external coils has operated continuously for more than 3 hours sustained entirely by the RF waves. However, the current drive efficiencies achieved using RF waves are limited, and power plant designs typically limit the fraction of RF current drive to about 10% of the total electric current in the plasma to meet recirculating power goals. Other options for sustaining and controlling the distribution of the plasma electric current are (i) use of the self-generated pressure-driven current in the plasma, which reduces the requirement for externally supplied current drive power; (ii) injection of magnetic helicity ($A \cdot B$) from the outside of the system by, for example, application of an electric field across the exterior magnetic field lines; and (iii) application of a rotating external magnetic field to impart momentum to electrons in the magnetically confined plasma. The first option of pressure-driven current is being actively explored with the critical challenge being the existence of a self-consistent solution of the requirements of MHD stability and the local particle and thermal transport which determines the plasma pressure profile. The second option is about to undergo a Proof-of-Principle (PoP) test after success in CE scale experiments, while the third option is now being studied at the CE scale.

Finally, an important fundamental area of wave-particle interactions is the effect of charged fusion reaction products in a fusion energy producing plasma ($Q > 5$) or a so-called burning plasma. The slowing-down process observed in $Q < 1$ experiments and simulated with ions generated by energetic neutral particle heating beams appears generally to be classical and is well understood. However, theory predicts that these suprathermal particles may excite Alfvén wave eigenmodes in a toroidal system, which may lead to high loss levels of the fusion reaction products before they impart their energy to heat the plasma. These effects have been seen in experiments. Another possibly significant loss mechanism of energetic fusion reaction products occurs through stochastic particle orbit effects due to periodic non-uniformities in the confining magnetic field. These effects have been quantitatively modeled and experimentally observed. Both of these loss processes are taken into account in power plant design and do not appear to be severely limiting, but this conclusion remains to be verified in more power-plant-like plasmas. In addition, the effects of these reaction products on the plasma current and electric field distribution directly and through pressure gradient modification may affect the local transport and equilibrium in significant ways. Finally, means have been proposed to enhance the fusion power gain by exploiting the free energy in the population of fusion products to, for example, drive current or heat ions to temperatures above those of the electrons.

Recent Major Scientific Advance: *Detailed internal measurements have confirmed classical energy slowing-down and good radial confinement of α particles in a high-power D-T plasma (see Fig. 2.8).*

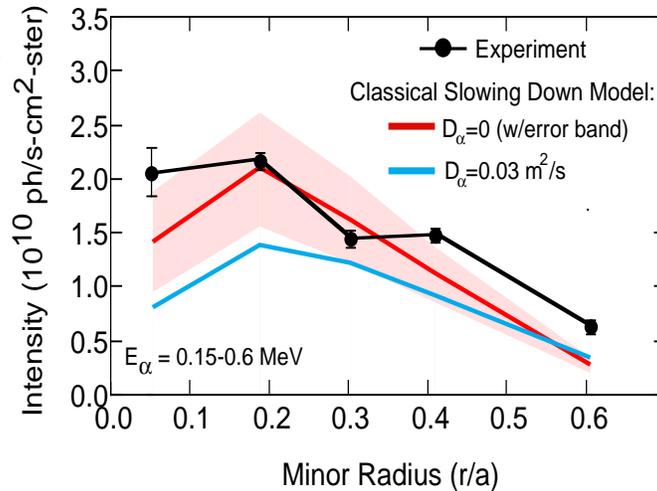


Fig. 2.8. Measured radial distribution of fusion α particles compared with a classical slowing-down model prediction assuming no anomalous diffusion of α particles ($D_{\alpha} = 0$). Assumption of a very small rate of anomalous α -particle transport is seen to be inconsistent with the data.

Plasma-Wall Interactions

Major Research Challenge: *What are the fundamental mechanisms of parallel transport along open magnetic field lines, and how can the heat flux along these field lines be dissipated before it strikes material surfaces?*

All magnetic fusion devices must deal with the power and particle handling interface at a material surface which surrounds the fusion plasma core. The leading approach to this interface is the use of a magnetic “divertor,” which occurs naturally in many toroidal systems and transports particles and power out of the region of closed nested magnetic surfaces to open magnetic field lines that allows the plasma to flow to a material plate. The flux of energy to the cooled plate is reduced by spreading out the power through radiation and geometric expansion of the area over which the power is delivered to the plate. The latter is achieved by divergence of magnetic field lines near the divertor x-points and through the use of plates strongly tilted relative to the field lines. Predictive 2-D numerical models, including plasma and atomic physics effects, have been developed and benchmarked against detailed experimental measurements. This has been a challenging problem both in plasma science and fusion technology. The combination of improved scientific understanding and significant advances in the technology of high heat flux components now allows projections of normal heat flux levels in a power plant environment below 5 MW/m² with allowable steady-state heat fluxes exceeding 10 MW/m². Critical issues which are being addressed include the protection of metallic plasma facing components under off-normal conditions (so-called “disruptions”), where high peak heat fluxes and energetic particle generation may be produced, and provision for adequate helium ash removal to prevent helium buildup in the fusion plasma core.

Recent Major Scientific Advance: *The measured spectrum of hydrogen light from plasma on diverted field lines confirmed the dominance of recombination as the mechanism of plasma extinction in conditions where heat flux is dramatically reduced.*

2.2.3 Path to Magnetic Fusion Energy

The Portfolio Approach

The Portfolio Approach manages risk and cost, balancing the opportunities for in-depth exploration of fusion science and fusion technology in more advanced MFE concepts, while at the same time, maintaining scientific breadth and encouraging innovation in the magnetic confinement configurations explored. It will produce the knowledge base needed to build and operate demonstration fusion energy sources

(DEMO stage) which take full advantage of the advances and innovations in fusion science and technology in the near-term and midterm time frame.

A central driver behind the portfolio-management approach applied to MFE concepts is the strong scientific synergy across the elements of the portfolio. Scientific advances made in one concept are readily translated to others, and new ideas emerge from synergistic combinations. The breadth of the portfolio pushes the development of new theory, encourages extension and validation of existing theory and models, advances new experimental techniques, and stimulates innovation. Some recent examples of the benefits of breadth are the use of MHD mode feedback control in the Advanced Tokamak (see Sect. 2.2.3.1), the combination of tokamak and stellarator ideas in the Compact Stellarator (see Sect. 2.2.3.1), and the use of helicity injection in the Spherical Torus (see Sect. 2.2.3.2).

The implementation of the Portfolio Approach takes account of three important factors: (i) among the spectrum of magnetic fusion confinement approaches, some are much more advanced than others; (ii) larger facilities are needed to reach fusion plasma parameters; and (iii) the U.S. MFE program is only a part of a much larger international effort to develop practical MFE. Another key element in the success of the Portfolio Approach is the application of advanced scientific computer simulation, which allows new ideas to be tested extensively and allows rapid and complete analysis of experimental data. In parallel with progress in confinement concepts, success in fusion energy will depend on continued progress in supporting technology development and in low-activation materials development and qualification. In the nearer term, this will focus on technology developments that enable the ongoing research programs and on the development of long-life, reduced-activation structural materials to confirm the environmental attractiveness of fusion. In the longer term, the emphasis will shift to developing those technologies needed to optimize the attractiveness of the ultimate fusion power source.

Elements of the Portfolio

Based on the results of the past two decades, the leading candidates for magnetic confinement systems are all toroidal in nature, due to the better combination of energy confinement and stability achieved and projected thus far in this geometry. The different toroidal magnetic configurations are classified by the degree to which the magnetic field structure is externally imposed. The two extremes are

- **externally controlled systems** in which the confining magnetic fields are largely supplied by external coils, and
- **self-ordered systems** in which electrical current flowing in the plasma provides most of the confining magnetic field.

Externally controlled systems offer the potential for steady-state plasma operation by imposing a magnetic field structure designed to prevent instabilities. Self-ordered systems, on the other hand, allow the plasma to generate its own magnetic field structure while relaxing to a configuration which minimizes the amount of free energy available to drive instabilities. There is, of course, a continuum between these two extremes, since all magnetically confined configurations have some currents flowing in the plasma and external coil systems to supply parts of the magnetic field structure.

2.2.3.1 Externally Controlled Configurations

The Tokamak is an axisymmetric toroidal system with the primary magnetic field supplied by external magnets and with closed magnetic surfaces which are generated by a toroidal electric current flowing in the plasma as shown in Fig. 2.9. The tokamak was the first magnetic confinement concept to sustain kiloelectron-volt-level plasmas and is today the most advanced magnetic confinement configuration. It has been the workhorse of the MFE program, providing all of the database on D-T burning plasmas and much of the data on the critical scientific elements of transport, stability, wave-particle interactions, and plasma-wall interactions. Much of the plasma science base discussed in Sect. 2.2.2.1 was developed through tokamak research.

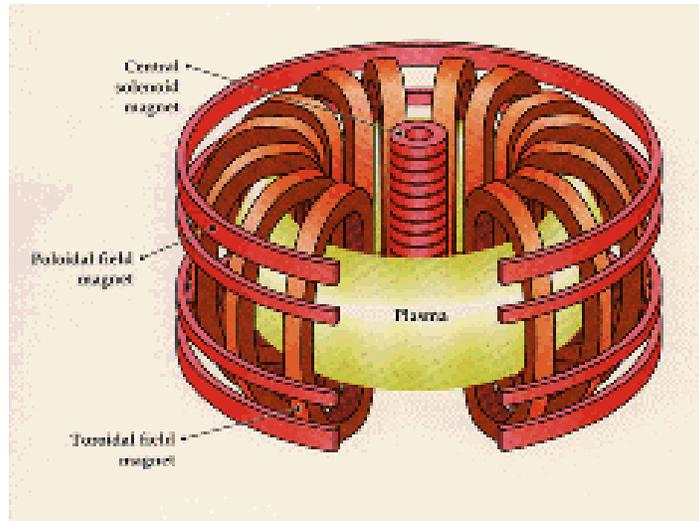


Fig. 2.9. Schematic view of tokamak configuration showing the large toroidal field magnets, the smaller equilibrium and shaping coils, and the toroidal plasma.

Confinement studies in tokamaks have shown that systematic and predictable confinement behavior can be obtained in magnetic confinement devices; see Fig. 2.10. Empirical confinement scaling has been continuously refined but has remained essentially unchanged for over 15 years. The scaling assessment of confinement in standard tokamak operating regimes made in 1982, on the basis of an international set of PoP experiments, predicted the initial confinement performance of the much larger Performance Extension (PE) tokamaks, TFTR, Joint European Torus (JET), and JT-60, to 10% accuracy. Also in 1982, in the ASDEX tokamak in Germany, an improved regime of plasma confinement was discovered—the

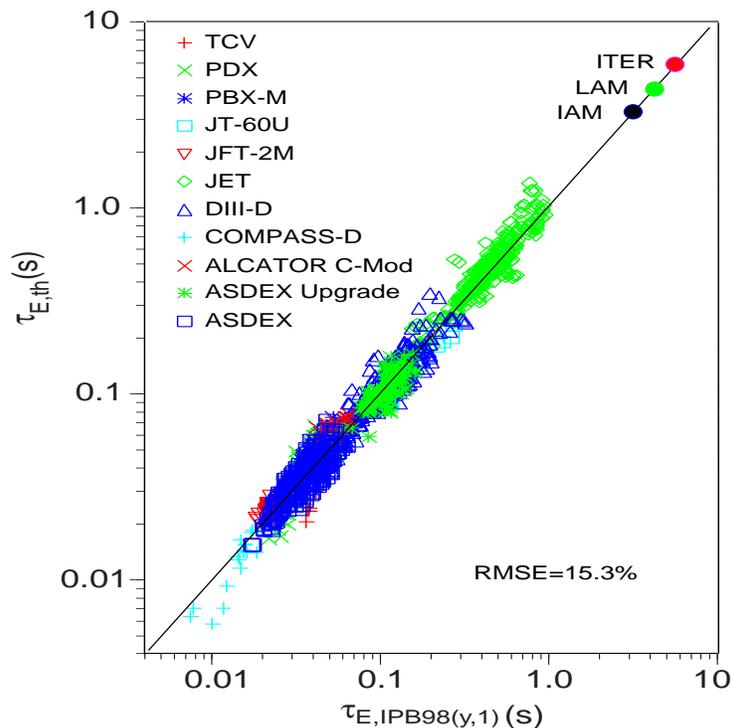


Fig. 2.10. H-mode database for confinement scaling in tokamaks. ITER and reduced cost variants LAM and IAM points are projections to next-step devices at the Fusion Energy Development stage.

H- or High-mode. This first observation of a “transport barrier” in a toroidal device produced a factor of two increase in the overall global energy confinement, compared with the standard or Low-mode behavior.

Volume averaged β values up to 13% have been produced transiently in tokamaks, exceeding power plant requirements, and in excellent quantitative agreement with theoretical predictions, but not yet in regimes consistent with long-pulse or steady-state operation. When a tokamak plasma exceeds the most important stability limit boundaries, such as the β limit, the result is normally a disruption event where most of the plasma energy is lost quickly, followed immediately by a rapid decay of the plasma current. Material surfaces can be damaged from the thermal energy loss, and large structural loads can be induced from the rapid quench of the plasma current; methods to mitigate the effects of disruptions when they occur are being developed with good results to date. However, disruptions remain an important issue for the implementation of a tokamak-based power plant.

The toroidal current in a tokamak is normally driven inductively, making use of a solenoid in the core of the torus. This method of current drive is intrinsically limited in pulse length. Methods to drive the current in steady state, injecting RF waves or beams of energetic neutral atoms that ionize in the plasma, have been successfully employed and are discussed in Sect. 2.2.2.1. The theoretically predicted self-driven or neoclassical bootstrap current has been confirmed and has opened up the prospect of efficient steady-state tokamak operation, with the majority of the current provided internally, leaving only a small part needing an external drive.

The field lines at the edge of a tokamak plasma can be *diverted* away from the main configuration and led into a separate region, where theoretical analysis and experimental results show that the outflowing heat can be handled and the helium ash from the fusion process extracted. It also appears straightforward to pump the helium ash from the plasma at an acceptable rate, taking advantage of this divertor configuration, which is broadly applicable to most toroidal confinement concepts.

Experiments on tokamaks have brought magnetic fusion to the beginning of research involving substantial fusion energy production; see Fig. 2.3. The slowing-down and confinement of α particles has been measured for the first time. Furthermore the effects of fusion plasma self-heating, via the confined α particles, have been observed. In D-T plasmas in the U.S. TFTR device, fusion power of 10.7 MW was produced, corresponding to $Q \sim 0.3$. Subsequently up to 16 MW has been produced at the JET corresponding to $Q \sim 0.6$, with a maximum fusion energy per pulse of 20 MJ. In these cases Q is defined as fusion power divided by heating power. Since the highest fusion power experiments are transient, if the time-derivative of the plasma stored energy, dW/dt , is subtracted from the external heating power, a higher value of Q would be determined as the ratio of $P_{\text{fus}}/P_{\text{loss}}$. By this definition, and extrapolating from D-D to D-T fusion rates, the Japanese JT-60U device has achieved conditions in a D-D plasma which have a projected $Q = 1.25$, in a D-T plasma.

The tokamak concept has advanced to the point at which an integrated fusion energy test at the Fusion Energy Development (FED) stage has been designed internationally (ITER). This ITER device is presently being considered for construction by Europe, Japan, and Russia and if built, will be an historic milestone: it will be the first integrated test of most of the, generally required, physics, technologies, controls, and diagnostics with a power plant relevant D-T fusion plasma.

The Advanced Tokamak (AT) relaxes some of the external control of the tokamak, depending to a large degree on the pressure-gradient-driven current to efficiently maintain its plasma current. This provides a much better prospect for steady-state operation, but the best performance requires operation beyond the β limits predicted in the absence of an ideally conducting wall around the plasma. This will most likely result in the need for active stabilization of long-wavelength MHD “kink” modes, in order for the plasma to experience a real, resistive wall as ideal. Another issue is that slow-growing “neoclassical tearing” modes, driven by inhomogeneities in the bootstrap current, can set a short pulse limit even for moderate β operation, in both the tokamak and AT configurations. If the feedback stabilization studies for long-wavelength kink and tearing modes now being carried out in the laboratory prove successful and current

and pressure profiles can be controlled adequately, the AT is projected to support 1.5 to 2 times the unstabilized plasma β limit.

Advances in techniques to control the plasma and current profile in a tokamak have led to the formation of so-called “transport barriers” in the plasma. The transport reduction within these transport barrier regions has been shown to arise from suppression of turbulence by sheared $E \times B$ plasma flows and the effects of the unusual current profiles on turbulence growth rates. Experiments have demonstrated reduction of the ion thermal transport to the neoclassical non-turbulent level over most of the plasma volume (see Fig. 2.8). The fraction of the plasma current supplied by the self-generated neoclassical bootstrap current has exceeded 80% in some experiments. The outstanding challenge in AT research is to achieve simultaneously: high bootstrap fraction, improved confinement, and extended β limits in long-duration plasma. This area of research is a major effort of the U.S. and foreign tokamak experiments: DIII-D and C-Mod in the United States and JET, JT-60U, Asdex-U, Tore Supra, and many other experiments around the world.

The Stellarator is a configuration in which the external coil set supplies not only the toroidal magnetic field but also much or all of its poloidal magnetic field. The closed magnetic flux surfaces needed for plasma confinement are created by twisting the shape of the external coils as shown in Fig. 2.11. Because stellarators can be designed with no externally driven plasma current, no recirculating power is needed to support the plasma current. Disruption events have not been observed in stellarator experiments in the absence of driven plasma currents and, in most cases, even in the presence of significant currents.

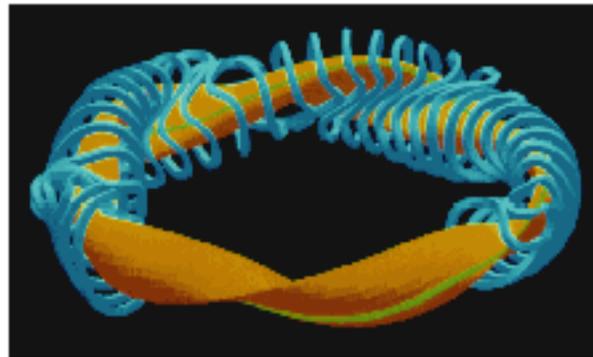


Fig. 2.11. Schematic of a modular stellarator design showing the twisting of the coils which produce nested flux surfaces for confinement.

In its conventional embodiment, the stellarator has a large aspect ratio ($R/a \sim 10$ to 20), resulting in relatively low power density relative to the size of the system. The divertor for power and particle control is more difficult to accommodate in a stellarator configuration because it is no longer axisymmetric and space between the plasma and the coils is limited. However, the large variety of stellarator configurations and the ability to vary the magnetic configuration within one device—change magnetic well and shear and vary the radial electric field—lead to an improved capability to study important basic plasma phenomena in a systematic and controlled manner. During recent years, PoP scale devices have achieved, collectively, $T_e(0) = 4$ keV, $T_i(0) = 1.5$ keV, $n = 3 \times 10^{20} \text{ m}^{-3}$, $\langle \beta \rangle = 1.8\%$, and $\tau_E = 50$ ms. H-mode operation with characteristics similar to tokamaks has also been observed.

Stellarator research is a major thrust of foreign fusion energy programs. A new PE scale device, the LHD stellarator in Japan, uses superconducting coils for steady-state operation and is the largest operating stellarator in the world, comparable in size to the large Performance Extension scale tokamak experiments. In its first few months of operation, LHD achieved an impressive energy confinement time of $\tau_E = 250$ ms. WVII-AS is a PoP device operating since the mid-1990s in Germany, and a new German superconducting coil stellarator of comparable size to LHD, W7-X, is now under construction. A new U.S.

exploratory experiment, HSX, which will study a new form of stellarator symmetry, quasi-helical symmetry, has just begun operation.

The Compact Stellarator (CS) implements new ideas in stellarator symmetries (called quasi-axisymmetry and quasi-omnigeneity), which open up the possibility of much lower aspect ratio stellarator configurations. By employing a significant pressure-driven current, this concept begins to move away from a purely externally controlled configuration. The CS is predicted to be stable against both long- and short-wavelength MHD modes, as well as against neoclassical tearing modes, and so it should require no stabilizing conducting wall nor active feedback control. The stability to neoclassical tearing modes arises from the fact that the global shear in the magnetic field is of the same sign everywhere that makes inhomogeneities in the bootstrap current self-healing. Due to the self-driven bootstrap current, the CS requires no significant external current drive, so power plants based on this configuration are expected to have low recirculating power fraction, similar to the larger aspect ratio stellarator. The quasi-axisymmetric configuration should also exhibit transport reduction via sheared flow in much the same way as the AT. If these theoretical predictions can be verified by experiments, the CS, through its low aspect ratio, should be able to achieve a fusion power density sufficient for a power plant, while offering disruption-free operation and requiring a relatively low recirculating power fraction. A proposal for a CS program has been positively peer reviewed as a new PoP component of the U.S. program and is currently in the conceptual design phase.

2.2.3.2 Intermediate Configurations

The Spherical Torus (ST) is an extension of the tokamak configuration to very low aspect ratio ($R/a < 1.5$) (see Sect. 2.2.5, Fig. 2.16), where the configuration benefits from some of the characteristics of self-ordered systems—simplicity of design and very high beta. To an even greater degree than the AT, the ST concept depends on high levels of pressure-driven bootstrap current and conducting wall stabilization of long-wavelength MHD modes, although requirements for rotational stabilization of the kink mode are predicted to be lower in the ST, and the neoclassical tearing mode may be stable. This configuration provides valuable information on aspect ratio scaling of physics phenomena in toroidal devices. It is also predicted to have naturally large values of plasma rotational shear flow that are believed to be responsible for the suppression of plasma turbulence and the greatly improved confinement seen in tokamak experiments. In exploratory scale experiments on the START device in England, the ST has exhibited low disruptivity and has demonstrated good confinement. High average toroidal field $\beta \sim 40\%$ and central $\beta \sim 100\%$ have been achieved, consistent with the most favorable theoretical predictions. Since the very low aspect ratio leaves little space for an inductive transformer to drive plasma current, a critical issue for the ST concept is to demonstrate effective non-inductive start-up and an efficient combination of current drive and bootstrap current for steady-state operation. Coaxial Helicity Injection, a technique used to initiate and sustain the current in spheromaks (Sect. 2.2.3.3), has been tested at the CE level on an ST and will be further investigated at the PoP level. In addition to its potential as a power plant, the ST is also a promising basis for the design of a volume neutron source (VNS) for component testing and may offer a lower-cost development path to the FED stage. New experiments at the PoP level ($I_p \sim 1$ MA) have started in early 1999 on the NSTX in the United States and on MAST in the United Kingdom.

The Reversed-Field Pinch (RFP) has a self-ordered plasma, and compared to the tokamak or the stellarator, it has a much weaker toroidal magnetic field system linking the plasma. Comparisons with tokamaks and stellarators have helped to clarify common physics issues; for example, the stabilizing effect of a close-fitting conducting boundary used to control low- n MHD modes in an RFP, has led to potentially key methods for improvement in the AT and ST concepts where wall stabilization is now a critical issue. Other areas of common interest include studies of edge turbulence and the role of magnetic vs electrostatic fluctuations in transport. At the CE scale, RFP plasmas have achieved, separately, $T_e(0) = 0.7$ keV, $T_i(0) = 0.4$ keV, $n \leq 5 \times 10^{20} \text{ m}^{-3}$, average $\beta \leq 20\%$, and $\tau_E = 5$ ms. The self-ordering effect of helicity ($\mathbf{A} \cdot \mathbf{B}$) conservation was first demonstrated on the RFP and has had a major effect on the understanding of the evolution of resistively unstable magnetic configurations in space and the laboratory. Recent studies have shown dramatic confinement improvement in the RFP through current profile control that reduces the level of the MHD plasma turbulence, although the tearing of the magnetic field lines still leads to much greater losses than observed in the externally controlled systems. Fusion power plant issues for this device include the nature of confinement scaling in more collisionless plasmas near 10 keV, the embodiment of a divertor, and how to maintain the very large plasma current continuously with low

recirculating power in the absence of any significant bootstrap current. If these problems are resolved favorably, the RFP offers a route to a higher power density power plant than the tokamak, ST, or stellarator. A PoP level proposal for upgrades to the existing CE scale MST experiment, in order to provide capabilities to investigate these issues, was positively peer-reviewed in the United States. The 1-MA PoP-level RFP device in Italy (RFX) and a similar scale device just coming on line in Japan, TPE-RX, form a complementary program.

2.2.3.3 Self-Ordered Configurations

These configurations are focused on globally simple, compact toroidal systems. The magnetic fields in the plasma are produced largely by the internal plasma current, with no coils threading the toroidal plasma, thus giving them a favorable geometry as a power system. The β range expected for these configurations ranges from 10% to as high as 80%. However, these are all exploratory concepts, which present major questions about confinement, gross MHD stability, and sustainment of the plasma current. Their high power density may raise questions about plasma-wall interactions, although divertors can be accommodated in these configurations. In particular, the scaling to power-plant parameters of the complex dynamics of the self-organizational processes which generate the magnetic field structure is not known. Nevertheless, if effective control and current drive systems can be realized, if transport can be reduced, and if power handling can be demonstrated, then these self-ordered configurations may represent an attractive approach to a fusion power plant.

In the Spheromak, the toroidal and poloidal fields, created by the plasma, are approximately equal in size. The device has a simple geometry for incorporating a divertor as shown schematically in Fig. 2.12. In exploratory scale devices central $T_e = 400$ eV and average $\beta \sim 5\%$ have been obtained with about a 2-T magnetic field.

Experiments have shown that the spheromak is subject to continuous resistive MHD modes, similar to those in the RFP, which tear the magnetic fields and reduce plasma confinement. MHD stability against the tilt mode is an issue as well as efficient sustainment of the plasma current. While initial experiments on the use of helicity injection for non-inductive current drive are encouraging, helicity penetration without loss of confinement remains to be demonstrated. A new CE experiment, SSPX, is under construction.

The Field-Reversed Configuration (FRC) is an axisymmetric toroidal plasma with only a toroidal plasma current and only a poloidal magnetic field. The coil and divertor geometry are the simplest of any configuration. FRCs typically operate at high density $n \leq 5 \times 10^{21} \text{ m}^{-3}$, where they have achieved $T_i \sim 1$ keV, an $n\tau_E \sim 10^{18} \text{ m}^{-3}\cdot\text{s}$, and the highest average β of 50% to 80%. An interesting observation is that

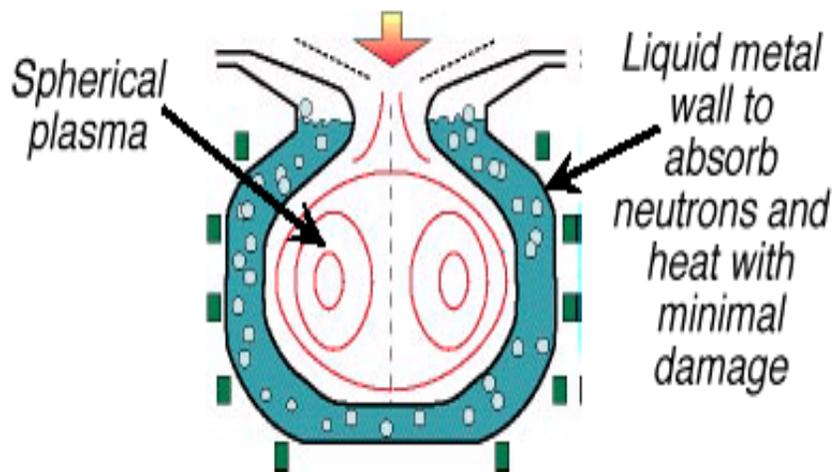


Fig. 2.12. Schematic of a self-ordered spheromak configuration illustrating near spherical reactor geometry using liquid metal blanket and shield.

the FRC plasmas produced in experiments are more globally stable than predicted by ideal MHD theory. It is generally understood that this is a consequence of the large size of ion gyro-orbits relative to the overall system, $\sim 1/4$ in present experiments. Key questions for the FRC include: at what scale (ratio of plasma radius to ion gyroradius) will the configuration suffer from the ideal MHD internal tilt instability; how important are interchange instabilities; and can a minority population of energetic ions stabilize these instabilities at power plant scale size? The physics of transport and confinement scaling for this configuration are not well known. Like the AT, the ST, the RFP, and the spheromak, the FRC has the potential to self-generate most of its plasma current. The remaining problem is how to drive the “seed” current required with the higher plasma density typical in the FRC. A promising approach is the rotamak method, using rotating magnetic fields in the near-field of large antennas—pioneered in Australia in a partially ionized FRC experiment. Use of the rotamak current drive technique in a fully ionized FRC will be investigated on the CE experiment called TCS.

2.2.3.4 Other Configurations

There are a number of other configurations which are interesting as scientific research tools and which may have the potential for some near-term applications in science and/or technology, but which are more speculative in regard to their ability to produce net fusion power.

The Magnetic Dipole fusion concept uses a levitated conducting ring to produce an axisymmetric field. It should operate disruption free, and possibly free of significant turbulence, potentially allowing classical confinement. The externally produced, axisymmetric geometry, similar to a planetary magnetosphere, makes it very appealing as a physics experiment. Such a configuration is projected to be able to contain a plasma with a volume average beta of $\beta \geq 10\%$. The dipole concept is based on a large body of space plasma observations at high β and some limited laboratory results. In regard to fusion applications, the dipole concept needs a superconducting levitated ring within the plasma, which must handle heat and neutron loads from the fusion plasma. A dipole fusion power source will likely require D-He³ fuel to minimize fusion-produced neutron heating of the levitated superconducting ring. This leads to a low system fusion power density. There are also important power-plant issues associated with maintaining the floating ring in the superconducting state. A CE experiment called LDX is under construction.

Strongly Driven Plasmas. The magnetic confinement configurations described above have a primarily thermal particle energy distribution. Through the use of intense particle beams and/or electromagnetic wave heating it is possible to create plasmas with a strong high energy ion component. This leads to enhancement of the fusion production rate over a thermal plasma of the same average energy. A concern is how to maintain efficiently the energetic ion distribution in the face of scattering collisions. Such configurations are interesting for plasma science and as a potential 14-MeV neutron source. A gas dynamic trap has been proposed as a 14-MeV neutron source for fusion technology development, and an energetic particle beam driven FRC has been proposed as a power plant. The former approach is carried out in the Russian Federation at the PoP level; the latter approach remains controversial and requires detailed peer review.

In Magnetized Target Fusion (MTF) a magnetic field embedded in an FRC or other self-organized plasma is rapidly compressed to fusion conditions by a radially-driven metal liner. To date, separate tests have been made of translation of an FRC plasma and of liner compression. The small scale and present availability of DP facilities could allow a rapid, low-cost, test at the PoP level. The energy requirements to achieve a fusion energy gain of ~ 1 are projected to be quite modest. Because of the invasive magnetic coupling required in the reaction chamber, of the high fusion yield, and of the repetition rate required for energy applications, a credible reactor design based on this concept has not yet been formulated. The attainment of high gain without a “hot spot” ignition region as in IFE is problematic, and stability at power-plant scale may be problematic as in the “conventional” FRC. Rapid, repetitive replacement of the liner and removal of the waste materials remaining from the previous implosion are critical concerns for fusion power applications. A PoP stage experiment has been proposed, and positively peer reviewed, with the goal of testing the basic physics of the formation, injection, and implosion elements of this concept.

Inertial-Electrostatic Confinement (IEC) systems make a spherical electrostatic potential well using very energetic magnetically confined electrons. Ions are injected into this large potential well and execute oscillating orbits that are repeatedly focused to the center of the spherically symmetric potential. The defining feature of IEC is the central ion focus with its large ion density due to geometric convergence. CE stage experiments in the PFX-1 experiment at LANL using a magneto-electrostatic extension of the Penning trap have shown that electron focusing occurs, with electron confinement times of about 1 ms and central densities up to 10^{19} m^{-3} . For fusion applications, there are several outstanding physics and technical issues: theory shows fundamental limits to fusion energy gain, $Q < 1$, in static systems, but oscillating fields such as the periodically oscillating plasma sphere (POPS) approach may be able to overcome these limitations; the voltages are high and the electrode spacing small; and space charge effects are a significant problem. The next step in CE studies is to confine ions and demonstrate significant energetic ion lifetime.

2.2.3.5 Commonality and Complementarity in Toroidal Magnetic Confinement

As outlined in the individual concept descriptions of the previous section, there is a significant commonality in key physics issues among the spectrum of magnetic confinement devices, and the field of MFE science has made progress through exploration of these issues on a broad front. Some of the key common issues are summarized in Fig. 2.13. This figure also notes the areas of complementarity—where road-blocks which might arise (e.g., tearing mode control or bootstrap current control) do not span all configurations, and so cannot fundamentally block progress toward fusion energy.

2.2.3.6 The MFE Portfolio

The MFE Portfolio features a range of confinement concepts at varying stages of development and scientific understanding, from CE up to the PE stage. An overview of the distribution of present U.S. MFE

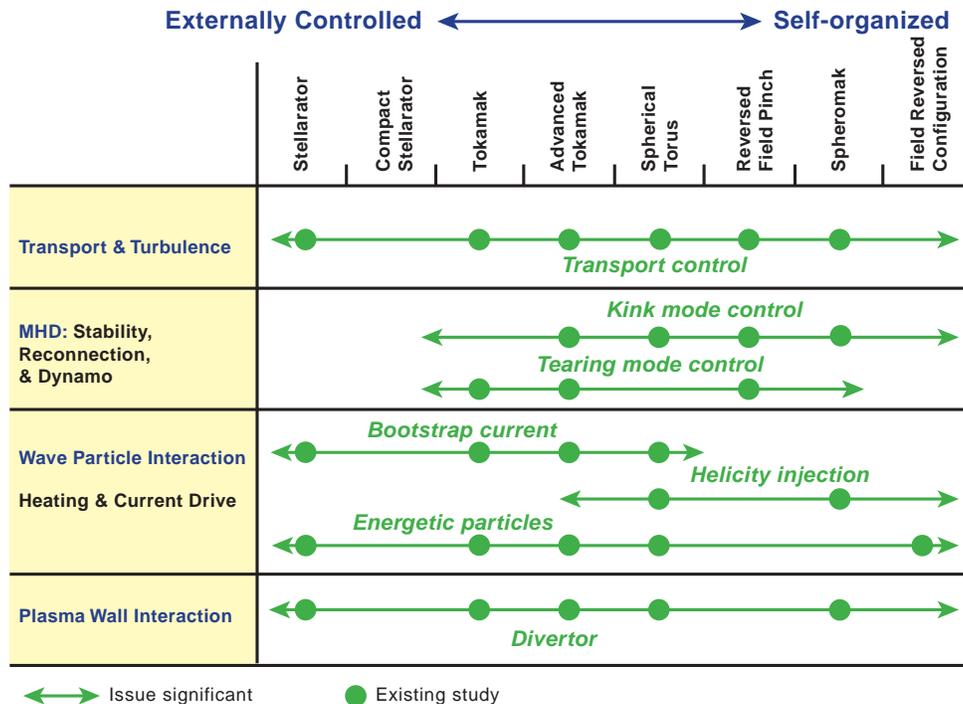


Fig. 2.13. Summary of some common key issues in toroidal confinement devices arranged by plasma science areas with primary fusion figure of merit.

experimental facilities in relation to the world program in fusion is given in Fig. 2.14. As shown in the figure, a broad range of issues is being explored at the relatively inexpensive CE stage within the U.S. program. Only one concept, the ST, is currently being explored at the PoP level, while the tokamak is the only concept being investigated at the PE stage. In contrast, there is a significant investment by the European Union and Japan at the PoP stage in the stellarator, ST, and RFP lines, and at the PE stage in the stellarator and AT. Because of the high cost of the PE and FED stages, leverage against the large international program in fusion energy research is essential.

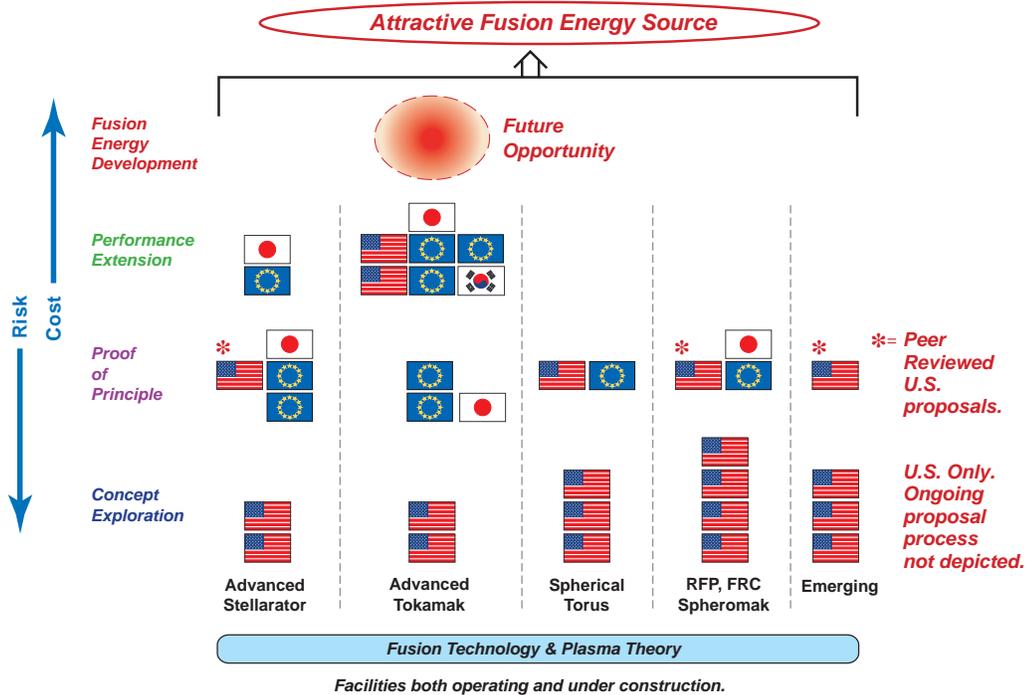


Fig. 2.14. Levels of development and world distribution of major facilities in MFE.

2.2.4 Opportunities in MFE

There are a number of important opportunities in both the near term (~5 years) and the midterm (~20 years) which would substantially advance the MFE program toward the goals of fusion energy and plasma science. These opportunities exist at all stages of concept development, as summarized in Table 2.2 and the following sections.

Near-Term Opportunities

Within the U.S. MFE program, a few speculative confinement configurations which offer a vision for improved fusion systems and broader scientific understanding are being pursued at the relatively inexpensive CE stage. Numerous opportunities exist at this level to begin preliminary tests of other new and unique configurations, particularly in self-organized systems, as well as to enhance the capabilities of existing experiments by adding plasma control and diagnostic systems. Opportunities also exist to provide inexpensive tests of innovative auxiliary systems and to resolve key physics issues which could lead to significant improvement of the more advanced confinement configurations.

Construction of the National Spherical Torus Experiment (NSTX) has been completed and the first integrated high-performance tests of the ST at the PoP level are beginning. The FED path for the ST is

Table 2.2. Opportunities for confinement concept improvement

Configuration	Near term ~5 years	Midterm ~20 years (assuming success with near-term goals)
<i>Externally Controlled Systems</i>		
Compact Stellarator	Build a PoP experiment to test predicted absence of disruptions, high beta limits, and good confinement, and develop turbulence control. Test alternative optimization in CE device.	Build a PE experiment based on the compact stellarator, likely with moderate-pulse D-T operation.
Advanced Tokamak	Demonstrate integrated plasma capabilities for improved beta and confinement, with current and pressure profile control, feedback systems, and a divertor in existing experiments with modest upgrades.	Demonstrate the full range of capabilities at very long pulse, understand burning plasmas at $Q \geq 10$, or do both in the single ITER-RC.
<i>Intermediate Systems</i>		
Spherical Torus	In PoP experiments, demonstrate the physics performance needed for the design of a D-T burning spherical torus.	Build a DTST PE experiment to support the next step of a VNS and/or Fusion Pilot Plant.
Reversed-Field Pinch	Use auxiliary RF and heating to study beta limits, and provide precise non-transient current profile control for magnetic turbulence suppression by upgrading existing device. Test oscillating field current drive. Explore non-circular plasmas in CE device.	Advance to a PE stage experiment in a ~10-MA device possibly with D-T fuel capability.
<i>Self-Ordered Systems</i>		
Spheromak	CE of confinement, reconnection physics, beta limits, feedback control, and divertor operation. Design a long-pulse experiment.	Build a long-pulse PoP for spheromak physics in the kiloelectron-volt range. Develop advanced helicity injection and other current drives. Develop innovative power plant technologies, e.g., liquid walls.
Field-Reversed Configuration	Develop an improved flux buildup and sustainment system. Study the influence of sheared flow and fast particles on stability. Develop a better understanding of confinement.	Build a PoP experiment for testing steady-state operation at multi-kiloelectron-volt temperatures.
<i>Other Systems</i>		
Levitated Dipole	Do CE, comparing theory and experiment, using ECRH to produce a hot electron beta ~100% locally. Use deuterium gas and lithium pellet injection to obtain high density.	Design and construct a PoP experiment.
Driven Plasmas	CE.	Conduct follow-on PoP experiments.
Magnetic Target Fusion	Build a PoP experiment to test performance of the concept. If successful, start preparation for a proof-of-performance experiment on the ATLAS facility.	Capitalize on success of proof-of-performance experiment to carry out PE experiment using ATLAS facility.

potentially highly cost-effective, and if its physics basis can be established, the power-plant implementation has been found to be attractive. Near-term opportunities exist to accelerate the research program on NSTX, including enhancement of the plasma diagnostic capabilities, enabling more in-depth studies of the basic dynamics of an ST plasma.

Three other concepts, which have been positively peer-reviewed for research at the PoP stage, include the CS, the RFP and MTF. The CS offers a potentially attractive fusion system by promising stable, disruption-free operation at very high power density but with very low recirculating power. It would extend the U.S. experimental fusion program to fully 3-D systems, which are more characteristic of naturally-occurring plasmas. A fusion power plant based on the RFP would have a much lower toroidal magnetic field than either the tokamak or the stellarator, thereby offering a potentially attractive power plant implementation with reduced costs for magnetic field coils. Its physics is closely related to that of the solar corona and near-surface regions, making it particularly interesting as a paradigm for understanding the physics of magnetic field generation in astrophysical and laboratory fusion plasmas. MTF—which is intermediate between MFE and IFE—may offer an inexpensive new opportunity for achieving significant fusion energy release and the scientific investigation of a unique plasma regime.

The AT is the focus of a strong domestic and international research program at the PE phase. It offers the potential for fully steady-state operation, and for higher fusion power density than the conventional tokamak, leading to an improved reactor concept. An important cost-effective opportunity would be to provide the key profile-control tools and the associated plasma diagnostics needed for full tests of AT regimes within the U.S. program. The strong investment in this area in Europe and Japan provides a scientific context for this research, leading to accelerated discovery and innovation.

The international fusion program, which is presently supported at over \$1 billion annually, offers a tremendous opportunity for the United States to investigate high-performance plasmas via a more extensive program of collaboration on the more powerful facilities abroad. Attractive billion-dollar-class magnetic fusion facilities operating or under construction abroad include the major tokamaks—JET in England, JT-60U in Japan, and KSTAR in Korea—as well as the major stellarators—LHD in Japan and W7-X in Germany. Specific scientific issues which can now only be addressed on the international facilities include the physics of sub-ignited plasmas, the dynamics of long-pulse, near steady-state plasmas, and the confinement and stability properties of high-performance plasmas in 3-D magnetic configurations.

Midterm Opportunities

The research direction taken by the U.S. program in the midterm will depend in part on the level of support received and the level of success achieved in the near-term program. A range of opportunities will develop at the CE, PoP, and PE level domestically, and at the PE and FED level internationally.

The near-term PoP experiments at the 1-MA level in D-D ST plasmas on NSTX in the United States and on the MAST facility in the European Union should provide key results on the confinement of and stability of high beta ST plasmas in the 2003–2004 time frame. If successful, then the construction and operation of 10-MA ST facility with D-T plasmas would be warranted. The Q and fusion power production of such a device are difficult to estimate without data from the PoP experiments, but the possibility that it may reach $Q \sim 5$ is exciting. Success with such a device could lead to a volume neutron source or fusion pilot plant in the longer term, in order to provide practical early experience with fusion technologies.

The demonstration of magnetic turbulence suppression with non-transient current profile control and sustainment could lead to tokamak-level confinement times in an RFP device. The results from the MST device in the United States and the RFX device in Italy as well as from the TPE-RX experiment in Japan would then provide the scientific and technical basis to proceed with an ~10-MA PE-level device, possibly using D-T fuel. Similarly, success in MTF could lead to a next-step D-T phase to demonstrate $Q \sim 1$.

The database provided by a successful PoP experiment on the CS in the midterm, combined with advances in both the AT and the large conventional-aspect-ratio superconducting stellarator programs,

should provide the basis for proceeding with a PE level device capable of using D-T fuel. Whether such a device would be superconducting and steady-state or employ copper coils and focus more specifically on high performance (e.g., ignition) would depend on prior physics results and available resources.

The FED stage, which precedes DEMO, produces plasmas with significant fusion energy gain ($Q > 5$) in near steady-state conditions, during which the critical fusion technology systems are integrated with a power plant regime fusion plasma core. Because of the time needed to design and construct such a facility, all opportunities for device construction are in the midterm time frame. In the near-term to midterm, the next international MFE steps at the advanced PE or FED stages will be defined and assessed. The tokamak concept is presently at the stage of readiness to pursue burning plasma physics ($Q > 5$). Fusion scientists both in the United States and abroad support moving forward either with RC-ITER, the reduced-cost/reduced technical objectives version of ITER under design by Europe, Japan, and Russia, or with an alternative “modular” strategy discussed below. The timing for a construction decision on ITER could come as early as 2001. However, a firm commitment from Japan, the European Union, and the Russian Federation probably will not come before early 2003. While the United States has withdrawn in FY 1999 from active participation in the extension of the ITER Engineering Design Activity which began in 1992, the other three parties (European Union, Japan, and Russia) remain committed to the project, and new designs have been developed that are expected to have construction costs in the range of half the construction cost of the initial ITER design. This RC-ITER device would accomplish most, perhaps all, of the ITER mission in a less costly experimental facility, including

- the creation and experimental investigation of self-heated plasmas ($Q > 10$),
- the demonstration of a long-pulse AT with $Q > 5$,
- the integrated exploration of related tokamak physics issues,
- the integration of fusion-relevant technologies, and
- the integrated testing of fusion reactor components in a single major facility.

If such a step is taken, it will be a major advance in MFE, and the U.S. fusion program can expect to benefit substantially both in fusion science and in fusion technology. Of generic benefit to most if not all fusion concepts, RC-ITER would represent an attractive opportunity for the United States to participate as a research partner, in the spirit of U.S. participation in the Large Hadron Collider at CERN.

If the RC-ITER is not constructed, it is also possible to move magnetic fusion forward, in the midterm, by the construction of two facilities which divide the mission of ITER into two separate experiments. One would be a D-T fueled small, high-field, limited-pulse $Q > 10$ device at the FED stage, and the other would be an advanced PE stage D-D fueled steady-state AT experiment. The cost of these two experiments would still be high enough (~\$2 billion total) that international collaboration would be essential. Shown in Table 2.3 are three examples of the range in size and performance being considered for a FED stage tokamak experiment. The generic toroidal burning plasma physics information from a small device like Ignitor or FIRE would provide a foundation for understanding burning plasmas in ATs, CSs, and ST, although it would not eliminate the need for such a step in the non-tokamak lines at the FED stage.

Table 2.3. Design objectives of FED experiments

	R(m)	B(T)	I _p (MA)	Gain	P _{fus} (MW)	Burn time (s)
Ignitor	1.32	13	12	>10	200	~5
FIRE	~2	10	~7	~10	200	≥10
ITER-RC	6.2	5.5	13	>10	500	>400

Longer Term Opportunities

The long-term goal of MFE research is the development of optimized confinement configurations as the basis for a decision to advance to the DEMO stage of FED. The contributions of advances and innovations in fusion science and technology in the near term and midterm time frame from elements of the

MFE Portfolio provide scientific understanding and, for the most successful concepts, the primary basis for the plasma core configuration. This step to the DEMO stage would necessarily follow experiments at the FED stage and hence would take place in the long term (>20 years). If the international effort in MFE R&D is maintained, a DEMO power plant could begin producing electricity by the middle of the next century.

In MFE, an extensive set of design studies have been carried out by the ARIES Team for a range of potential power plants including the tokamak (D-T fueled and D-He³ fueled), the RFP, the stellarator, and the ST. These studies have identified the principal development needs for such power plants, for the confinement configurations, and for the commonly needed building blocks, enabling technologies, and the materials required (see Sect. 3). These studies, which are the most complete ever carried out for prospective fusion power systems, provide a fully integrated analysis of power plant options, including plasma physics, fusion technology, economics, and safety. Of course, the actual optimized MFE configurations, which ultimately will be the candidates for a DEMO decision on the long-term timescale, can be expected to continue to evolve based on advances and innovations in the near-term and midterm time frames.

Shown in Fig. 2.15 is a schematic of the ARIES-RS power plant design producing a net 1000 MW of electric power. The ARIES-RS plasma is optimized to achieve a high plasma pressure and a high bootstrap current fraction (90%), which is very well aligned with the required equilibrium current-density profile (Table 2.4). The current-drive analysis showed that about 80 MW of current-drive power is necessary for steady-state operation. This design utilizes a lithium-cooled blanket with a vanadium structure which achieves a high thermal conversion efficiency of 46% (using 610°C coolant outlet temperature and a Rankine steam cycle). Use of vanadium in the high-temperature zones provides sufficiently low levels of afterheat that worst-case loss-of-coolant accidents can be shown to result in a small release of radionuclides (below 1 rem at site boundary), well below the values specified by standards and regulations. The blanket is made of sectors, and rapid removal of full sectors is provided through large horizontal

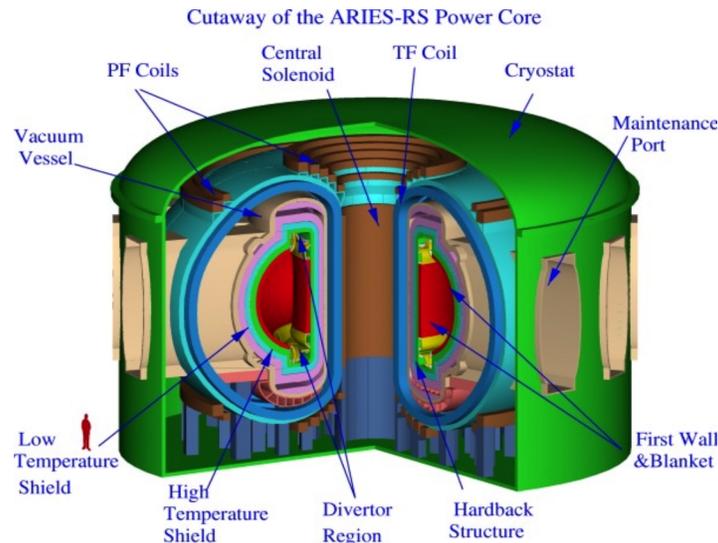


Fig. 2.15. Schematic of the fusion power core of the ARIES-RS AT power plant.

Table 2.4. Parameters of the ARIES-RS and ARIES-ST power plants

	R(m)	B _t (T)	I _p (MA)	Gain	P _{fus} (MW)	Bootstrap fraction (%)
ARIES-RS	5.5	8	11.3	~30	2170	88
ARIES-ST*	3.3	2	29.5	100	2851	95

ports followed by disassembly in the hot cells during plant operation. The simple blanket design with a small number of cooling channels and low mechanical stresses in the structure provides a good basis for high reliability.

The ARIES-ST study was undertaken as a national U.S. effort to provide a preliminary investigation of the potential of the ST concept as a fusion power plant. Similar studies are presently under way in the United Kingdom. The ARIES-ST power plant design (see Fig. 2.16 and Table 2.4) produces 1000 MW of electric power and has an aspect ratio of 1.6 and a major radius of 3.2 m. This configuration attains an average toroidal β of 54% that drives 95% of the plasma current by the neoclassical bootstrap current. While the plasma current is ~30 MA, the almost perfect alignment of bootstrap current density and equilibrium current density profiles results in a current-drive power of only ~30 MW. The on-axis toroidal field is 2 T, and the peak field at the toroidal field coil is 7.6 T. A relatively high recirculating power fraction (33% vs 17% in the ARIES-RS design) is required to drive the normal conductor toroidal field coil. This may be reduced by moving to a larger unit size, but the present design effort was constrained to 1000-MW electric output. The ST configuration allows a very attractive vertical maintenance scheme in which the central column and/or the blanket assembly can be removed for maintenance in a single operation, and then replaced with spares, minimizing downtime.

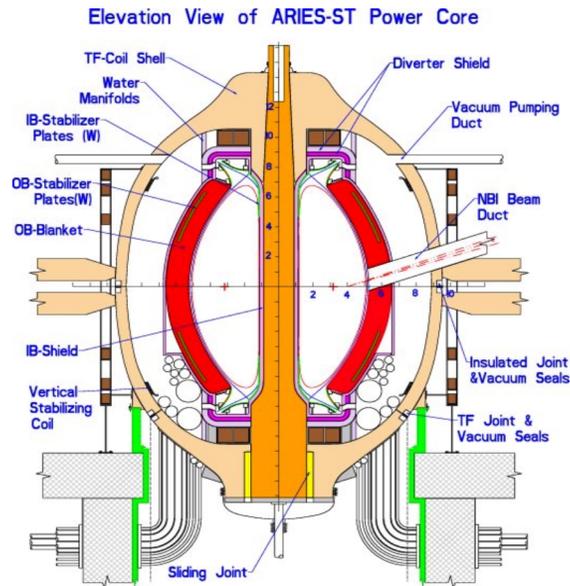


Fig. 2.16. Schematic of the fusion power core of the ARIES-ST power plant design.

2.3 The Inertial Fusion Pathway to Fusion Energy

2.3.1 Introduction

Power Plant. An IFE power plant (see Fig. 2.17) would consist of four major components including a target factory to produce about 10^8 low-cost targets per year, a driver to heat and compress the targets to ignition, a fusion chamber to recover the fusion energy pulses from the targets, and the steam plant to convert fusion heat into electricity.

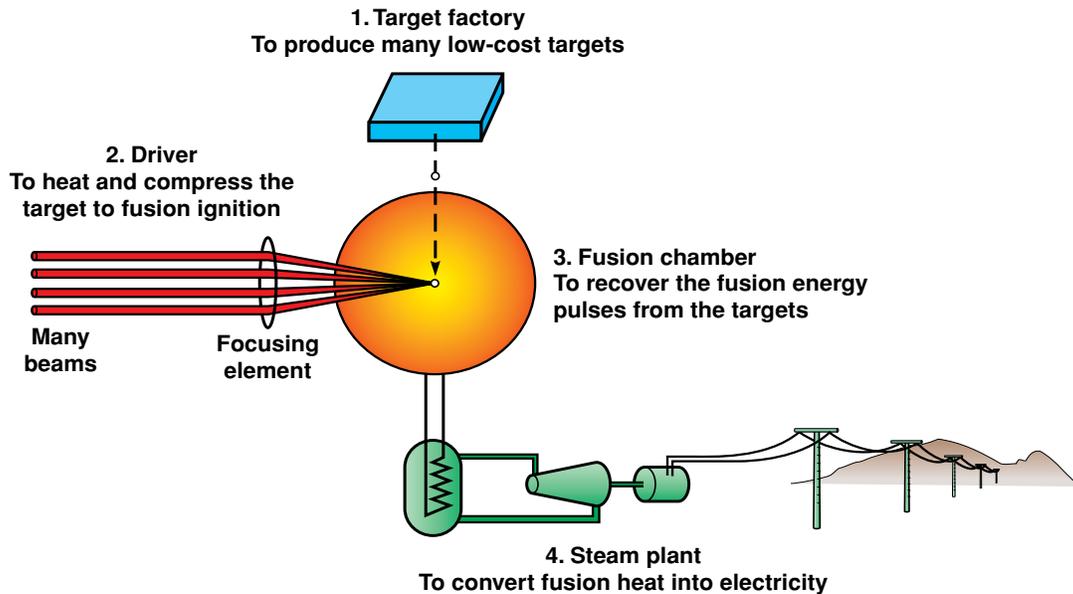


Fig. 2.17. Schematic of an IFE power plant.

Benefits include the fact that most of the high technology equipment (driver and target factory) are well separated from the fusion chamber, leading to ease of maintenance. The major driver candidates (ion accelerators and lasers) are modular so that partial redundancy would allow for on-line maintenance and reduced development cost. Some fusion chamber concepts, such as those that incorporate thick liquid layers, have chamber walls that are protected from the neutron flux. These protected wall chambers can have a long lifetime and low environmental impact, potentially greatly reducing the need for advanced materials development. Laser or ion driver beams can be transported to multiple fusion chambers. This can lead to benefits in the development of IFE and in the cost of electricity at commercial scale, if plant unit sizes in the multigigawatt range are acceptable. To realize these benefits, IFE must meet several challenges.

Targets. Current ICF targets are made by hand and require about 2 weeks of technician time to fabricate. Targets are individually machined, coated, characterized, and assembled. Targets for IFE must be ignited about 5 times per second. To keep the target contribution to the cost of electricity below 1 ¢/kWeh, targets must be produced for less than about \$0.50 each at 1-GWe output. An IFE target mass is less than 1 g, and the cost of materials is minimal. The challenge for IFE is the development of manufacturing techniques that can achieve the required cost and precision.

Fusion Chamber. Inertial fusion is inherently pulsed, and all IFE fusion chambers must deal with the effects of pulsed bursts of neutrons, X rays, and debris. This includes establishing conditions between shots which are suitable for driver beam propagation and cryogenic target injection. A wide variety of fusion chamber concepts has been developed for IFE. These can be divided into those which protect the structural wall from neutrons and those which do not. Those chambers which have structural materials

that are not protected from neutrons, both dry wall and thin film wetted walls, have first wall neutron damage issues and associated R&D needs which are similar to those of MFE, with the additional requirement to handle high pulsed loads. Chambers of this type allow a wide variety of irradiation geometries and concepts exist for all the driver types being considered for IFE. There are IFE chamber concepts which utilize thick layers of liquids or granules inside the solid structural walls. These chambers require targets with driver beam access limited to a narrow range of directions. In general, such targets have reduced gain relative to targets which have uniform irradiation and hence require more efficient drivers. Because of this, current concepts for protected wall chambers are only feasible with ion beam drivers.

Drivers for IFE must achieve an efficiency which depends on the target gain. Central to the economics of any inertial fusion power plant is the fusion cycle gain. This gain is the product of the driver efficiency η (the ratio of the energy delivered to the target and the energy supplied to the driver), the target gain G (the ratio of the thermonuclear yield and the driver energy), the nuclear energy multiplier M (the energy change due to neutron reactions, principally in the lithium-bearing blanket used to produce tritium), and the thermal-to-electric energy conversion efficiency ϵ . In any inertial fusion power plant, the net electricity P_n is related to the gross electricity P_g through the power balance equation:

$$P_n = P_g - P_a - P_d = P_g (1 - f_a - 1/\eta GM\epsilon),$$

where P_a is the power used for auxiliary equipment, and $f_a = P_a/P_g$ is typically a few percent of the gross electricity. P_d is the driver power, and the driver's recirculating power fraction P_d/P_g is the reciprocal of the fusion cycle gain $\eta GM\epsilon$. If the recirculating power fraction becomes large, the cost of electricity (COE) escalates rapidly.

The nuclear energy multiplier M is typically 1.05 to 1.15, and the conversion efficiency ϵ is typically 0.35 to 0.50. If the product $\eta G = 7$ for example, the recirculating power would range from 25% to near 40%. Lasers currently being developed have projected efficiencies of 6–10%, while heavy ion accelerators have projected efficiencies of 25–40%. Hence laser drivers will require targets with higher gain than ion beam drivers for a given recirculating power fraction or driver cost. The COE is given by:

$$\text{COE} = \frac{dC/dt}{P_n A} \approx \frac{dC/dt}{P_g (1 - f_a - \frac{1}{\eta GM\epsilon}) A}.$$

The factor "A" is the plant availability, and dC/dt includes the operating and maintenance cost as well as the capital cost per unit time. For fusion power plant designs which are capital intensive, typically 80% or more of the COE is the capital cost which includes cost for the driver, reactor plant equipment, and balance of plant. In the various IFE designs that have been carried out, the driver costs range from less than 30% to almost 50% of the capital cost. There is a driver size and target gain combination that minimizes the COE. Target gain typically increases for larger driver energy resulting in a higher fusion cycle gain and lower recirculating power. However, the larger driver costs more and increases the capital and operating costs. IFE drivers must also have adequate repetition rate and durability. In the typical IFE chamber, targets would be injected 5–10 times per second. Over the 30-year life of a fusion plant, the driver would need to produce nearly 10^{10} pulses. A driver must be able to deliver a sufficiently high fraction of this number of pulses between maintenance cycles so that plant availability remains high.

2.3.2 ICF Target Physics

2.3.2.1 Introduction

Inertial confinement fusion (ICF) is an approach that relies on the inertia of the fuel mass to provide confinement. To achieve conditions under which inertial confinement is sufficient for efficient thermonuclear burn, high-gain ICF targets have features similar to those shown in Fig. 2.18. A fusion capsule generally is a spherical shell filled with low-density gas ($\leq 1.0 \text{ mg/cm}^3$). The shell is composed of an outer region, which forms the ablator, and an inner region of frozen or liquid D-T, which forms the main fuel.

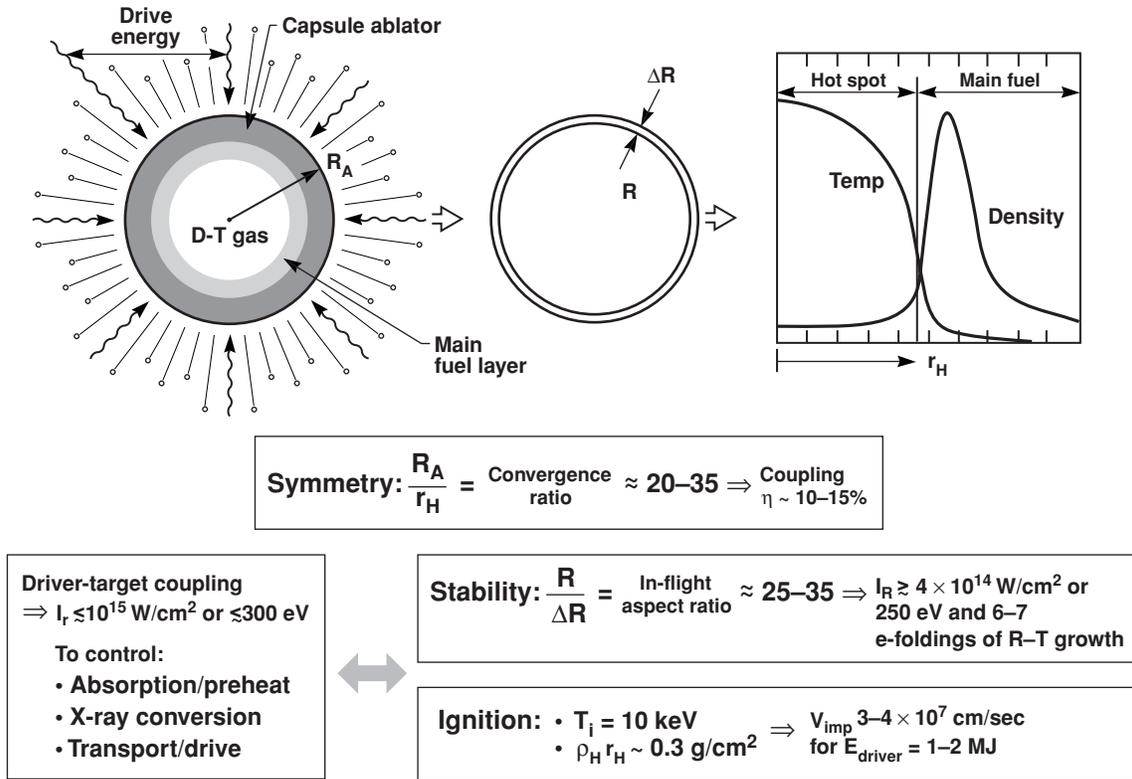


Fig. 2.18. Schematic of ICF imploding capsule with requirements on driver coupling, drive symmetry, hydrodynamic instability, and ignition.

Energy from a driver is delivered rapidly to the ablator, which heats up and expands. As the ablator expands outward, conservation of momentum requires that the rest of the shell move inward. The capsule behaves as a spherical, ablation-driven rocket. The efficiency with which the fusion fuel is imploded typically lies in the range of 5 to 15%. The work that can be done on the imploding fuel is the product of the pressure generated by the ablation process times the volume enclosed by the shell. Hence, for a given pressure, a larger, thinner shell that encloses more volume can be accelerated to a higher velocity than can a thicker shell of the same mass. The peak achievable implosion velocity determines the minimum energy (and mass) required for ignition of the fusion fuel in the shell.

In its final configuration, the fuel is nearly isobaric at pressures up to ~ 200 Gbar but consists of two effectively distinct regions—a central hot spot, containing ~ 2 to 10% of the fuel, and a dense main fuel region, comprising the remaining mass. Fusion initiates in this central region, and a thermonuclear burn front propagates radially outward into the main fuel, producing high gain. The efficient assembly of the fuel into this configuration places stringent requirements on the details of the driver coupling, including the time history of the irradiance and the hydrodynamics of the implosion.

In the implosion process, several features are important. The in-flight aspect ratio (IFAR) is defined as the ratio of the shell radius R as it implodes to its thickness ΔR , which is less than the initial thickness because the shell is compressed as it implodes. Hydrodynamic instabilities, similar to the classical Rayleigh-Taylor (RT) fluid instability, impose an upper limit on this ratio, which results in a minimum pressure or absorbed driver irradiance. Control of RT-induced mix of hot and cold fuel is crucial to the successful formation of the central hot spot.

The convergence ratio C_T as defined in Fig. 2.18 is the ratio of the initial outer radius of the ablator to the final compressed radius of the hot spot. Typical convergence ratios to the hot spot for an ignition or

high-gain target design are 30–40. An asymmetric implosion results in enhanced thermal conduction from the hot spot to the cold surrounding fuel and a reduced conversion of the available kinetic energy into compression and heating of the fuel. The tolerable degree of asymmetry depends on the excess of available kinetic energy above the ignition threshold. If we require that this deviation δR be less than $r_h/4$ where r_h is the final compressed radius, and where v is the implosion velocity, we have:

$$\frac{\delta v}{v} < \frac{1}{4(C_r - 1)} .$$

Since $30 \leq C_r \leq 40$ is typical, we require accelerations and velocities that are uniform to about 1%.

2.3.2.2 Direct and Indirect Drive

As shown in Fig. 2.19, two principal approaches are used to generate the energy flux and pressure required to drive an ICF implosion.

In the direct-drive approach, the driver beams are aimed directly at the target which in this case consists of just the fusion capsule. The beam energy is absorbed by electrons in the target's outer corona. With short wavelength lasers, absorption can exceed 80%. Electrons transport that energy to the denser shell material to drive the ablation and the resulting implosion. The most highly developed direct-drive targets use laser drivers although direct-drive targets using ion beams may also be feasible.

In the indirect-drive approach, the driver energy is absorbed and converted to X rays by material inside the hohlraum that surrounds the fusion capsule. The beam and hohlraum geometry are determined by the requirement for X-ray flux uniformity on the capsule. The most highly developed indirect-drive target designs use laser or ion beam drivers. Recent target concepts utilizing z-pinch driven X-ray sources may also prove to be a viable approach to igniting ICF fuel capsules.

Because of the X-ray conversion and transport step, indirect drive is less efficient than direct drive. The fraction of the driver energy absorbed by the fuel capsule varies from about one-tenth to one-third in typical indirect-drive designs. However, ablation driven by electron conduction is in general about a factor of 2 less efficient than ablation driven by X rays. Direct-drive capsules are more hydrodynamically unstable than capsules driven by X rays. Direct-drive targets are very sensitive to intensity variations within individual beams because these variations imprint perturbations on the target that are then amplified by

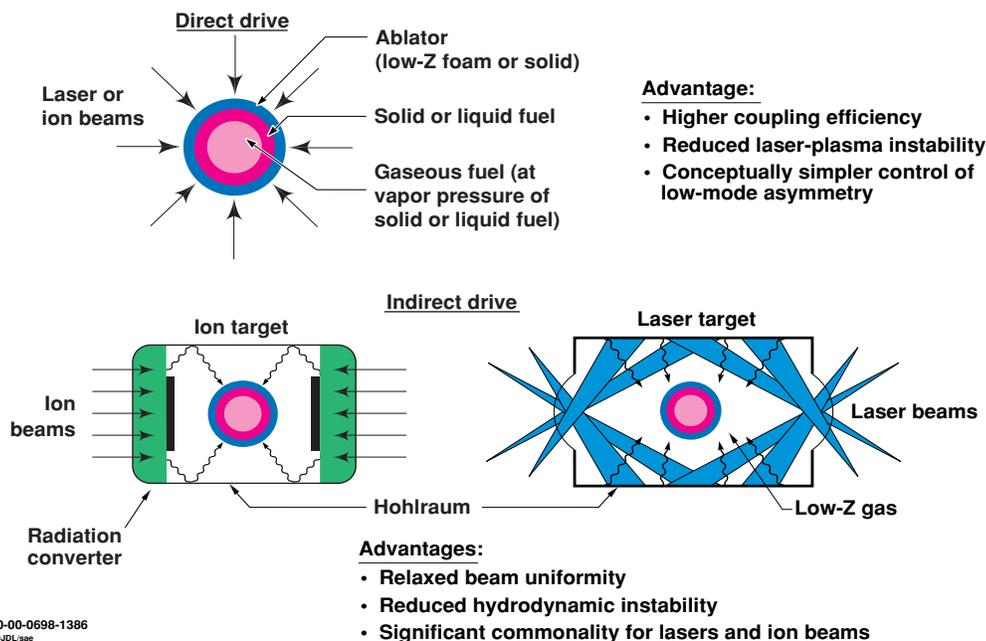


Fig. 2.19. The two principal approaches to ICF are direct drive and indirect drive.

hydrodynamic instability. Measures taken to mitigate hydrodynamic instability in direct-drive targets further offset the efficiency advantage. If adequate beam uniformity can be achieved, calculations for current laser target designs indicate that direct-drive targets have about the same ignition threshold as indirect-drive targets, but that they can have up to a factor of 2–3 higher gain, depending on the level of driver beam imprint and hydrodynamic instability growth that is tolerable. As discussed below, some ion beam driven indirect-drive calculations have higher hohlraum efficiency than laser-driven hohlraums and achieve gain similar to that predicted for laser direct-drive targets.

Reduced coupling efficiency and adverse effects from laser-plasma interaction limit laser-driven direct-drive and indirect-drive targets to $I \sim 10^{15}$ W/cm² for laser wavelengths of 1/4 to 1/2 μm . Because of ion beam emittance limitations, ion-driven targets are also typically limited to $I \sim 10^{14}$ – 10^{15} W/cm².

Exploration of the target physics of inertial fusion has been carried out predominantly by the DOE–DP. Because of the ease with which lasers can achieve the required irradiance, almost all ICF experiments have been carried out with lasers. Preliminary experiments at $\sim 10^{12}$ W/cm² were carried out as part of the DP program in light ion fusion. ICF relevant experiments have begun on z-pinch driven X-ray sources within the past 2 years, because of advances in the intensity achieved. Since the hohlraum wall physics and the capsule physics are essentially the same for any X-ray source, indirect-drive experiments on lasers provide much of the target physics basis for ion-driven targets.

Fast Ignitor. Driver technology advances may make other ICF target concepts, such as the fast ignitor possible. In this approach, the fuel is compressed to high density without a hot spot. A very short separate beam pulse is then used to ignite a spot on the surface of the compressed fuel. Because lower fuel density is required, more fuel can be compressed for a given amount of energy in the fast ignitor approach than in the hot spot ignition approach. If the fuel can be ignited with reasonable efficiency, higher gains and smaller driver requirements would result. Because it is not necessary to produce the central hot spot, this approach may also have somewhat relaxed symmetry and target fabrication finish requirements. To ignite the fuel in this approach, the fast ignitor beam must achieve intensities of 10^{19} – 10^{20} W/cm². The energy must be delivered in a time of about 10 ps into a spot of a few tens of microns in diameter, timed to a few tens of picoseconds with the peak target compression. From a target physics perspective, either lasers or ion beams are potential drivers for this type target. A hybrid scheme, for example using an ion driver for compression and a laser for ignition, is also possible. The development of chirped pulse amplification in laser systems over the past 10 years has opened up worldwide interest in research into this type of ICF target. However, the interactions of lasers with matter at the intensities required for fast ignition are quite complex, and significant work remains before the concept can be evaluated.

2.3.2.3 Experimental Progress

Nova. Since its completion in 1985, the ten-beam Nova laser at LLNL has been the primary U.S. laboratory facility for radiation-driven experiments. Nova can deliver 30 to 40 kJ in 1 ns or over longer periods with a wide variety of temporal pulse shapes at an output wavelength of 0.35 μm .

Nova has been used for a wide variety of experiments on laser-plasma interaction, hohlraum symmetry, hydrodynamic instability, and implosions. Over a 6-year period from 1990–1996, scientists from LLNL and LANL achieved a major fraction of the Nova Technical Contract goals established by the National Academy of Science, sufficient to support proceeding with the National Ignition Facility (NIF). Results from Nova and since 1996 from the Omega laser, approach the NIF requirements for most of the important ignition capsule parameters as shown in Table 2.5.

However, in hohlraums scaled to have a NIF-like ratio of hohlraum size to capsule size, implosions on Nova and Omega have not yet achieved NIF-level convergence with adequate performance. This is a result of the limited number of beams and beam power control on these facilities compared to NIF. NIF targets are designed with an ignition margin so the targets will tolerate the degrading effects of asymmetry and hydrodynamic instability. Without alpha deposition, the performance of an NIF capsule with the maximum acceptable level of asymmetry and instability has about one-half the yield that a 1-D

Table 2.5. The results from Nova and Omega experiments approach the NIF requirements for most of the important ignition capsule parameters

Physical parameter	NIF	Nova (Omega)
Drive temperature (eV)	250–300	>300 eV for 1-ns pulse ~250 eV for shaped pulse in gas-filled hohlraum
Drive symmetry		
— Number of beams	192	10 (60)
— r.m.s. capsule drive asymmetry (all modes)	1%	4% (2%)
— Implosion averaged (P_2)	~1%	~1%
Capsule convergence ratio (C.R.)		
— Capsule hydrotest only	25–45	24
— NIF-like hohlraum/capsule ratio	25–45	10 (17–20)
Hydro-instability e-foldings		
— Acceleration, deceleration	6–7 spherical	4–5 planar 4–5 spherical

implosion would produce. Most of that degradation is due to hydrodynamic instability, which causes a mix region of cold material to penetrate the hot spot. At failure, the mix region has penetrated about one-third of the hot spot radius. The goal of the Nova and Omega implosion experiments has been to test the effects of instability on capsule degradation in the NIF relevant regime for mix penetration. From the point of view of the physics involved, there are no identifiable issues that arise between a convergence of 10 and a convergence of 20–40. However, the higher convergence clearly tests the limits of what can be achieved on any of today’s lasers. On Nova, the effects of asymmetry are sufficiently large, even for convergence 10, that asymmetry is a much larger effect on yield than for NIF. Yields are reduced from 1-D by a factor of 2 to 3 from asymmetry alone. To experimentally see the effects of hydrodynamic instability in this situation requires fairly large capsule perturbations. Although we were able to quantitatively model the yields of these experiments using the 3-D implosion code Hydra, the hydrodynamic instabilities are further into the nonlinear regime and are less sensitive to initial perturbations than NIF capsules. On Omega, which has better symmetry than Nova because of its 60 beams, the experiments at convergence 10 achieve about 80% of the calculated 1-D yield. The calculated degradation was about equal for the effects of asymmetry and instability and is in an NIF relevant regime for the effects of instability and mix. On preliminary Omega experiments at a nominal convergence of 20, the best capsules gave about one-half of the 1-D yield, but there were capsules a factor of 2 to 3 below this. Improvements in laser control and in target fabrication needed for more consistent capsule performance and higher yields at these higher convergences are being pursued for a future experimental series. With the current level of control on Omega, NIF-like cylindrical hohlraum implosion experiments with a convergence of 30–40 are expected to have significantly reduced performance. Experiments on Nova at convergence ratios in this range have been carried out, and the capsules have yields degraded from 1-D by approximately 2 orders of magnitude. Experiments on Omega have also been carried out using spherical hohlraums with tetrahedral symmetry. In principle these hohlraums can have better symmetry than cylindrical hohlraums. However, the Omega laser does not have exact tetrahedral symmetry, and the 3-D computational capability necessary for integrated calculations of these experiments is not yet available. The best tetrahedral symmetry hohlraum experiments demonstrate capsule performance relative to 1-D that is comparable to that seen in cylindrical hohlraums. The capsules in these implosion experiments have been fabricated with 200–300 Å surface finish. This is comparable to the surface finish required for NIF capsules. Since Nova capsules are about one-fourth of the size of the NIF capsules, the effects of hydrodynamic instabilities for capsules with 5 e-foldings of growth are comparable to that for NIF capsules with 6 or more e-foldings of growth

and the same surface finish. The performance of capsules on Nova and Omega with 4–5 e-foldings of RT growth and a surface finish which was varied from 200–300 Å to more than 1 μm is in agreement with 3-D calculations. In addition to the results indicated in Table 2.5, Nova plasmas designed to emulate NIF plasma conditions, using NIF-like smoothing, have absorption of 90–95%, which meets the NIF goal.

The primary modeling tool for indirect drive has been the LASNEX code system. This code is a 2-D integrated model of the physics processes important for ICF. It includes hydrodynamics, electron and ion transport, radiation transport, atomic physics and material properties, thermonuclear burn products, and laser and ion beam transport. LASNEX does not calculate the collective effects that can result from high intensity laser-plasma interaction (LPI). These effects are calculated separately. When the LPI effects are small, as expected for the NIF ignition regime, LASNEX has demonstrated a quantitative predictive capability across a wide range of experiments. Shown as an example in Fig. 2.20 are the experimental and calculated results for a radiation-driven hydrodynamic instability experiment on Nova. In the experiment shown in Fig. 2.20, a foil of plastic material to be accelerated is placed on the side of a hohlraum. X rays to drive the foil are generated inside the hohlraum with 8 of Nova’s 10 beams. A separate Nova beam is used to generate an X-ray backlighter source. These X rays are viewed through the foil. By looking at the foil face-on using a 1-D X-ray streak camera, the growth of sinusoidal grooves can be measured as a change in the X-ray contrast between thick and thin regions as a function of time. By looking at the foil edge-on with an X-ray framing camera, the 2-D shape of the grooves is obtained. The position of the foil versus time is obtained from a 1-D streaked image looking at the foil edge-on. For all three types of data, the calculations are in good agreement with the data.

Over recent years, 3-D codes have been developed as part of the Accelerated Strategic Computing Initiative (ASCI) in DP. These codes, which have also been validated on Nova experiments, are currently being used extensively to model the 3-D effects of RT instability on fusion capsules.

Indirect-drive ICF began as a classified program in DOE–DP. However, since 1993, almost all of the laboratory ICF Program has been unclassified. This has significantly increased the opportunity for international collaboration in IFE.

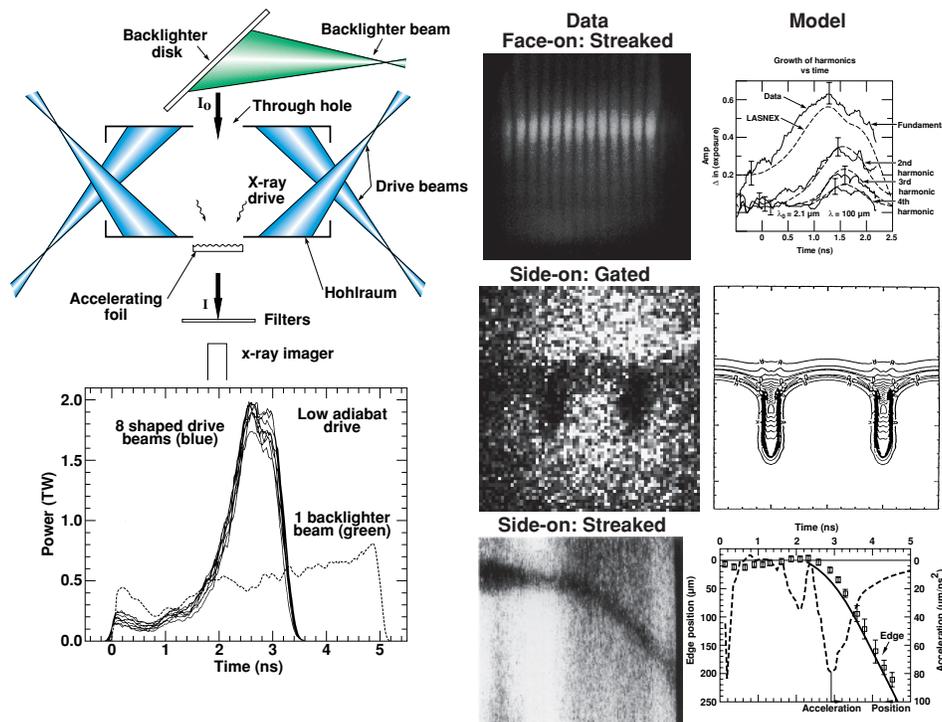


Fig. 2.20. The measured growth of planar hydrodynamic instabilities in ICF is in quantitative agreement with numerical models.

Z-Accelerator. Since 1997, the Z-accelerator at Sandia National Laboratories has made significant progress in X-ray production. The Z-accelerator has produced up to 2 MJ of X rays, which have been used to heat hohlraums to temperatures in excess of 150 eV. Preliminary radiation-driven target designs indicate that a z-pinch driver with about ten times the power and energy of the Z-accelerator could drive high yield ICF targets. Experiments on Z to test the physics basis of these targets have recently been initiated. Initial experiments to examine radiation symmetry in z-pinch driven hohlraums are in agreement with calculations. An assessment of possible repetition-rated Z-pinch concepts is just beginning. However, even in the absence of a z-pinch approach to IFE, experiments on Z and any follow-on machine (such as the ZX or X-1 machines proposed for DP) add to the data base for X-ray driven target concepts. The ignition physics program preparing for the NIF includes experiments on Z to examine shock timing issues.

The **OMEGA laser** at the University of Rochester and the **NIKE laser** at the Naval Research Laboratory (NRL) are the principal direct-drive facilities in the United States.

The direct-drive experiments on the 60-beam, 30-kJ OMEGA laser system ($\lambda = 0.35 \mu\text{m}$) have yielded the highest neutron yields obtained in any laboratory ICF experiments ($\sim 2 \times 10^{14}$ neutrons/shot). Recent OMEGA experiments have investigated various details of laser imprinting and the RT instability in planar and spherical geometries. This work has significantly improved the understanding of these hydrodynamic instabilities, which are crucial for direct-drive ICF. It has also resulted in a better definition of the irradiation requirements for direct drive, culminating in a number of improvements in irradiation uniformity, from smoothing by spectral dispersion in two dimensions (2-D-SSD), to broadband frequency conversion, to polarization smoothing. In addition, OMEGA will be equipped for cryogenic D-T implosion experiments by the end of FY 1999. Previous cryogenic compression experiments on the initial 24-beam, 3-kJ OMEGA laser system demonstrated core densities of ~ 200 times D-T liquid density. Although designed for direct drive, OMEGA has also been used extensively by LLNL and LANL for indirect-drive ICF. These experiments have allowed testing and verification of NIF hohlraum design characteristics such as beam phasing. With its larger number of beams, OMEGA provides improved X-ray drive uniformity compared to Nova. With further improvements to power balance and pulse shaping, experiments on OMEGA capsules may approach NIF convergence ratios with good performance.

NIKE is a krypton fluoride (KrF) laser system with excellent beam uniformity, producing 2–4 kJ of energy at a wavelength of $0.26 \mu\text{m}$. in a 4- to 8-ns pulse onto planar targets. This laser system has been used extensively for the study of hydrodynamic instabilities and laser imprinting of concern to direct-drive laser fusion. NIKE is also used for the study of imprint-resistant target shell designs such as foams and deuterium-wicked foams. In addition, NIKE is being used to determine the EOS of deuterium ice. The data obtained with NIKE are crucial for the proper design of direct-drive NIF targets.

Outside the United States, the Gekko XII laser at ILE in Osaka is the principal ICF experimental facility. This laser is capable of producing 8–10 kJ of energy in 12 beams at either 0.53 or $0.35 \mu\text{m}$. Gekko has been used for both direct-drive and indirect-drive experiments. The Phebus laser at Limeil, the equivalent of two beams of Nova, has been used extensively for indirect-drive experiments. In Russia, the ISKRA-5 laser, an iodine laser operating at a laser wavelength of $1.315 \mu\text{m}$ and 10–15 kJ, at Arzamus-16 has been used for indirect-drive experiments in spherical hohlraums. Other smaller facilities, which have been used for indirect-drive target physics, include the Asterix III laser at Garching and the Shengguang laser facility at Shanghai.

Halite-Centurion. The ICF program has also used data from underground nuclear experiments. The Halite/Centurion (H/C) Program, a joint program between Livermore and Los Alamos, demonstrated excellent performance, putting to rest certain fundamental questions about the feasibility of achieving high gain. This program carried out inertial fusion experiments using nuclear explosives at the Nevada Test Site at higher energies than those available in the laboratory. This is the principle area in which results in the DP activities in ICF remain classified.

2.3.2.4 National Ignition Facility

Results from Nova experiments and modeling, as well as results from the H/C Program provided the technical basis for proceeding with the NIF, under construction at LLNL. The ultimate goal of NIF is to achieve gain in the range of ten, where the gain is defined as the ratio of the thermonuclear yield to the laser energy delivered to the target. The NIF, shown in Fig. 2.21, is a key element in the DOE–DP Stockpile Stewardship Program. It is a \$1.2B project scheduled for completion in 2003. NIF is a 192-beam, frequency-tripled ($\lambda = 0.35 \mu\text{m}$) Nd:glass laser system designed to achieve routine on-target energy of 1.8MJ and power of 500 TW, appropriately pulse-shaped. The NIF laser is being designed to carry out three target shots per day. The 192 beams are delivered to the target chamber in 48 clusters of 4 beams. The technology for NIF was jointly developed with the French CEA which is planning to construct the LMJ, a 240-beam laser with goals very similar to those of NIF.

Indirect-drive targets of the type shown in Fig. 2.22 have been the most thoroughly explored for testing on the NIF. However, the NIF target chamber is being constructed with additional beam ports so that both direct-drive and indirect-drive targets can be tested. NIF will be able to map out the ignition and burn propagation threshold for both target types and begin to map out the ICF gain curves shown in Fig. 2.23. If warranted by results of current research, the NIF could be modified to test fast ignition as well.

The fuel conditions that must be achieved for efficient burn are similar to those of a magnetically confined plasma. In the equation below, ϕ is the fuel burnup fraction, τ is the confinement time, N_0 is the particle number density, ρ is the matter density in the fuel, and r is the compressed fuel radius. In inertial confinement, burn of an ignited fuel mass typically is quenched by hydrodynamic expansion. From the outside of the fuel, a rarefaction wave moves inward at the speed of sound, C_s . By the time this rarefaction has moved a fraction of the radius r , the fuel density in most of the fuel mass has dropped significantly, and the fuel no longer burns efficiently. Because of this, the confinement time is proportional to the compressed fuel radius r .

$$\phi = \frac{\rho r}{\rho r + 6(\text{g/cm}^2)} \approx \frac{N_0 \tau}{N_0 \tau + 5 \times 10^{15} (\text{s/cm}^3)} .$$

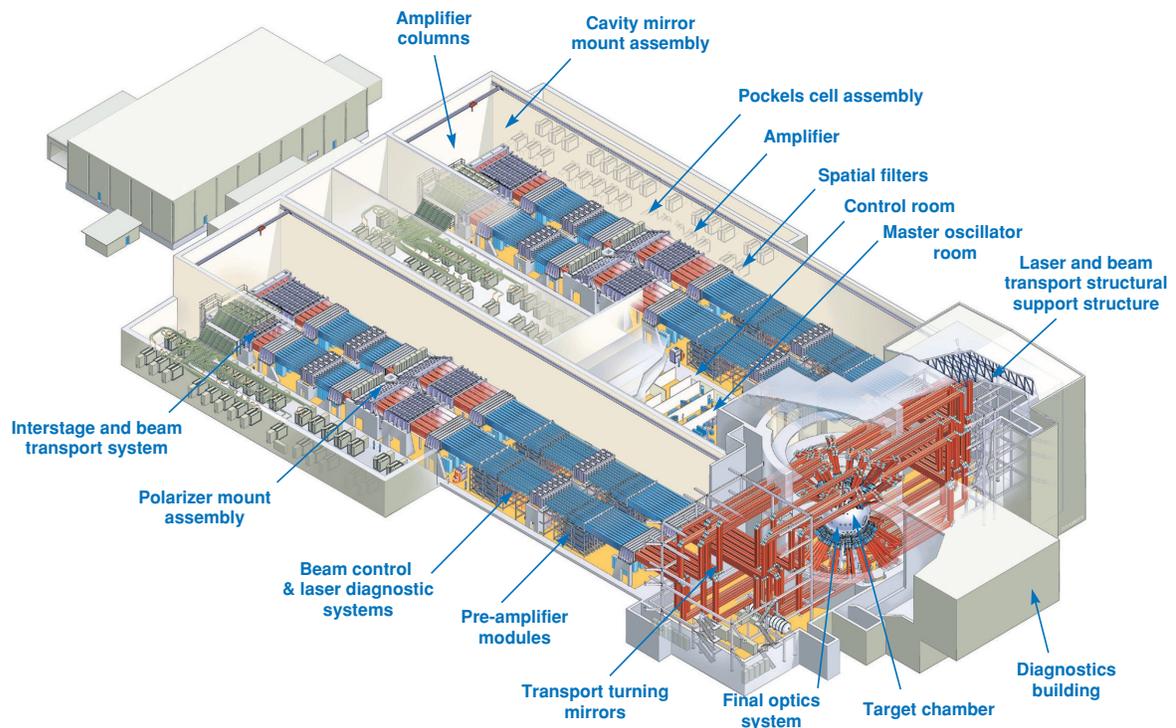


Fig. 2.21. The National Ignition Facility will play a critical role in addressing IFE feasibility.

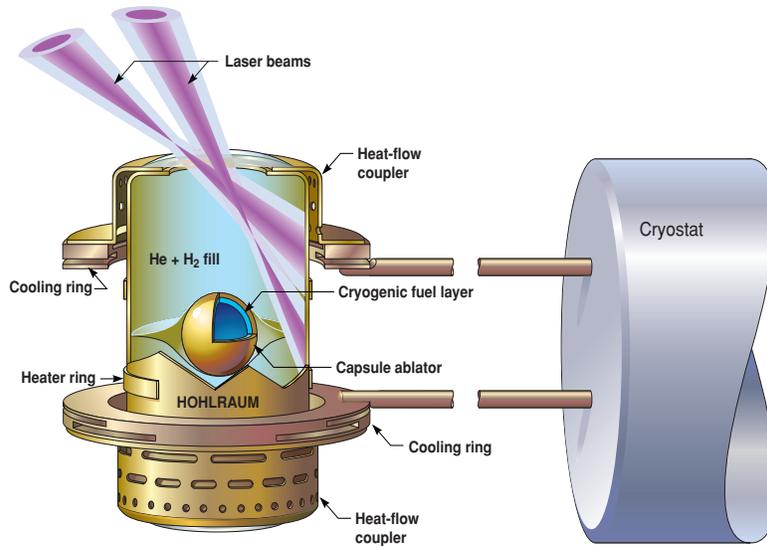
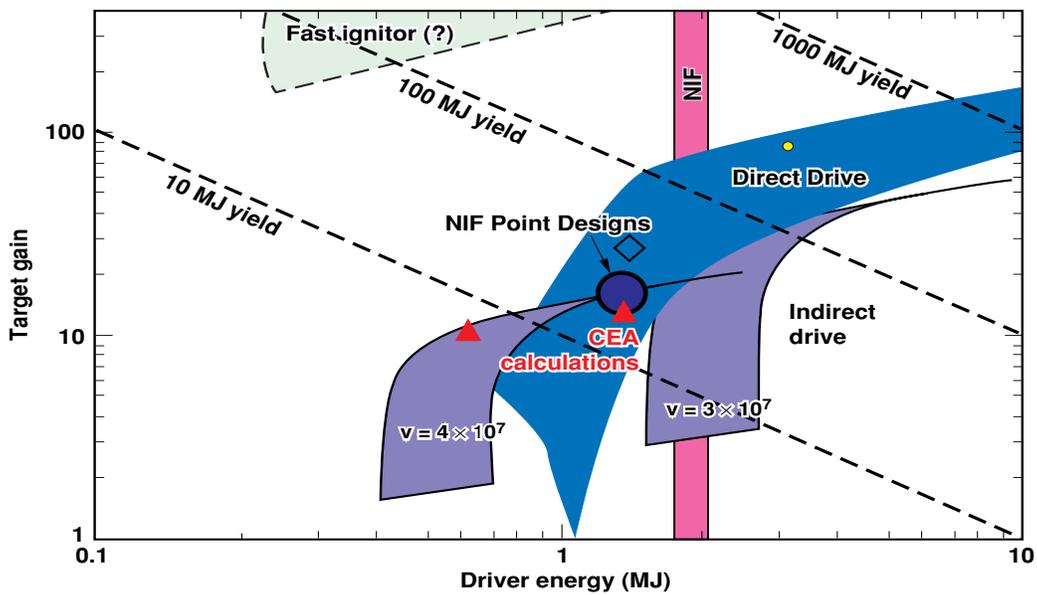


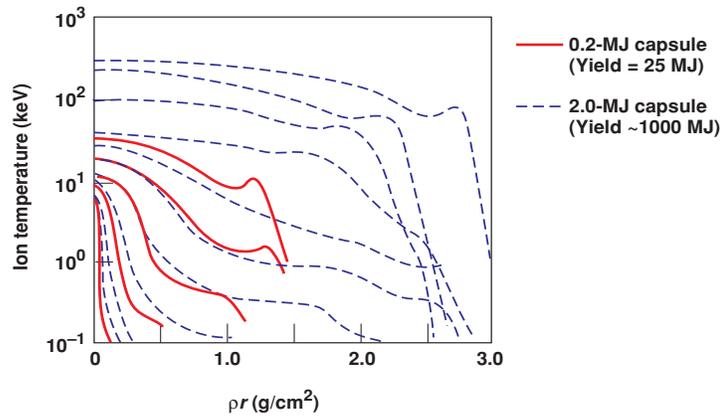
Fig. 2.22. NIF ignition targets utilize precise laser beam placement for implosion symmetry and accurate thermal control for cryogenic fuel layer uniformity. The hohlraum for this target is about 1 cm in length.



● Direct-drive design for KrF by NRL

Fig. 2.23. NIF will map out ignition thresholds and regions of the gain curve for multiple target concepts.

Once the hot central region of the fuel reaches 10 keV with an ρr equal to the range of the alpha particles ($\sim 0.3 \text{ g/cm}^2$ at 10 keV), the burn will propagate into and ignite an indefinite amount of surrounding cold fuel. These ignition and burn propagation conditions are nearly independent of fuel mass over a wide range of sizes. After ignition occurs, the burn wave propagates in ρr and temperature space in a way that is essentially independent of size. NIF fuel capsules are designed to absorb 0.1–0.2 MJ of X rays, while capsules envisioned for energy production typically absorb 1–2 MJ of X rays. Figure 2.24 shows the temperature versus ρr conditions for a 0.2-MJ NIF capsule and a larger 2.0-MJ capsule as the burn wave propagates into the fuel. The two capsules track each other until the smaller capsule starts to decompress.



- Pairs of curves are temperature contours at a series of times as the burn wave propagates through the fuel
- 5% burnup of the initial hot spot is sufficient to propagate the burn into a surrounding 10× denser shell

Fig. 2.24. Burn propagation in NIF capsules tracks that in larger capsules until decompression begins.

Thus information for NIF capsules is widely applicable to capsules with larger yield and can be used to design the higher yield capsules generally appropriate for energy production.

The DOE–DP in ICF includes an ongoing assessment of approaches that could result in higher yields than those that can be obtained with the NIF baseline targets. Direct-drive targets on NIF have the potential for yields on the order of 100 MJ if very high quality beam smoothing and target quality can be achieved. There are exploratory designs for NIF hohlraums that could increase the coupling efficiency by a factor of 2 or more, and capsules in these hohlraums have yields approaching 100 MJ. In addition, there are designs for targets using advanced z-pinchs, such as the proposed X-1 machine, that could have yields of 200–1000 MJ.

2.3.3 An IFE Development Pathway for Lasers and Ion Beams

Recent progress in target physics and target design for high energy gain in the U.S. inertial fusion research programs supports the possibility of developing an environmentally attractive fusion power plant with an acceptable COE, using either laser or ion drivers. Success of the NIF ignition program, expected within the next decade, together with advanced numerical models, will give us confidence that the gains needed in future IFE plants can be achieved.

The NIF is designed to demonstrate the ignition and burn propagation threshold for both indirect-drive and direct-drive targets. The near-term program in IFE can therefore focus on the development of efficient, reliable, and affordable drivers with high pulse rate capability, and associated high pulse rate fusion chambers, target fabrication, injection, and tracking. The IFE program can use a phased approach as shown in Fig. 2.25, with a set of near-term evaluation points, leading to a high average-power Engineering Test Facility (ETF) and a DEMO following the NIF. Figure 2.25 focuses on the ion beam and laser approaches to IFE that are the most developed and have the greatest probability of meeting the IFE requirements in the near term. Because of their relative maturity, the development pathway for these approaches shown in Fig. 2.25 begins at the PoP level. The IFE development pathway also includes some more speculative and less developed concepts in drivers and targets. These concepts provide opportunities for new science and a potentially more attractive ultimate power plants. They are appropriate for CE level research.

Progression through each of the four development steps shown in Fig. 2.25 is determined by meeting criteria for success of the previous step. The criteria start with top level requirements for an attractive, competitive final power plant and work back through each step.

The proposed IFE program in Phase I, at the PoP level, is sufficiently broad that the candidacy of ion accelerators with indirect-drive, and both solid-state and KrF lasers with direct drive, can be adequately assessed for a major Phase II decision, a PE level. This IFE program would leverage the DP's large investments in laser and pulsed power facilities, target design capabilities, and experimental infrastructure including target fabrication and diagnostics. Phase I would involve modest-size ion accelerator experiments and the development of 100-J class high repetition-rate lasers, along with driver scaling studies and further improvements in chamber design, target fabrication, and target design. Concepts which meet the requirements for capital cost, efficiency, durability, chamber wall protection, final optic protection, low-cost target fabrication, target injection, driver propagation through the chamber, and projected target gain could move to Phase II. An approach which fails the Phase II criteria could be considered for further exploratory R&D if appropriate.

The Phase I research must provide the scientific and technical basis for proceeding to an Integrated Research Experiment (IRE) which is the major facility proposed for Phase II. There is also an expectation that progress toward an ignition experiment on NIF will continue as expected in DP.

The IRE objective for the heavy ion approach is a completely integrated ion accelerator from ion source to beam focus in target chamber center. Goals include demonstration of the beam quality required to

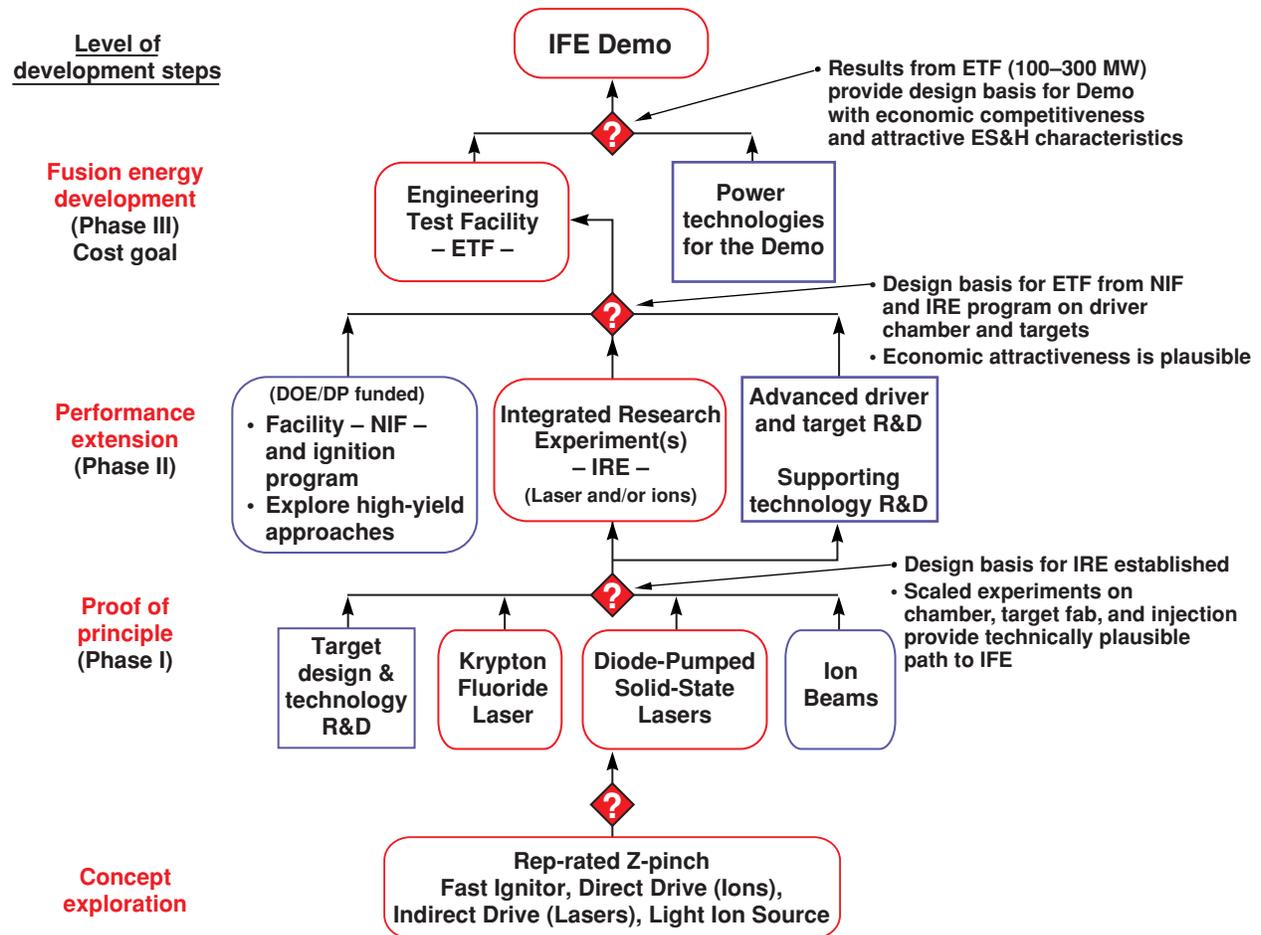


Fig. 2.25. A phased, criteria-driven IFE development pathway.

focus the beams to high intensity and experiments to study beam propagation in the fusion chamber. The purpose of the beam propagation experiments is to validate the physics of the relatively simple ballistic propagation mode and to explore more complex modes of transport in plasma channels. If the channel modes can be made to work, they will lead to improved chambers and reduced cost drivers. Finally, the heavy ion IRE could enable exploration of target physics issues unique to ions, for example, fluid instabilities in ion direct drive. For both lasers and ions, the size of the IRE will be large enough that the cost for the Phase III ETF, a FED level facility, can be accurately projected.

For lasers, the plan is to develop and optimize one complete laser beam line that could in principle be used directly in a power plant. If appropriate, it could then be duplicated in parallel to produce the needed total driver energy for an ETF. While one could also follow this parallel approach with ions, it does not lead to an optimal accelerator in terms of efficiency and cost. If the ion IRE is successful, it would be more appropriate to add acceleration modules in series to produce the needed energy for an ETF. The cost effectiveness of this strategy will depend on how rapidly the technology evolves between the IRE and construction of an ETF.

Results from the IRE(s), together with results from chamber and target fabrication R&D and ignition results from NIF, must be sufficient to justify the ETF, a high average fusion power IFE facility in Phase III. At a total construction cost goal of \$2B–\$3B, the ETF would be capable of testing several candidate fusion chamber approaches to determine which type of chamber and final optics would last sufficiently long for the next step, an IFE DEMO. The goal of Phase III is an integrated demonstration of the driver, targets, and the fusion chamber. Neutron irradiation materials development would have to proceed in parallel on the ETF or on a separate facility, particularly for IFE concepts with unprotected walls.

In the final step DEMO, net electrical power, tritium fuel self-sufficiency, and reliability would be demonstrated at a level sufficient for commercialization to be undertaken by industry. The IFE DEMO would complete the federal investment in IFE fusion energy development. For the DEMO, it may be possible to simply add the “balance of plant” and an appropriate chamber selected from the Phase III project, to the ETF Phase III driver.

IFE Research and Development for Phase I and Phase II

The proposed IFE program in Phase I and II would be distributed in these areas:

- ion and laser driver development,
- target design and optimization,
- IFE fusion chamber R&D including protection for walls and final optics (Sect. 3),
- injection of targets into fusion chambers,
- experiments on methods to mass-manufacture low-cost targets (Sect. 3), and
- safety and environmental R&D (Sect. 3).

In addition, IFE power plant studies would be carried out to explore the compatibility and optimization of all these areas, including definition of appropriate high average fusion power IFE development facilities for Phase III.

2.3.4 IFE Drivers

2.3.4.1 Ion Accelerators

The goal of the Heavy Ion Fusion Program is to apply accelerator technology to inertial fusion power production. As shown in Fig. 2.26, ions with kinetic energies of 10 MeV to 10 GeV, depending on the ion mass, have an ion penetration depth appropriate for inertial fusion targets.

Ion accelerators can readily produce such energies. Since fusion targets require beam powers of 100–1000 TW, the accelerators must deliver 10 kA–100 MA of beam current, depending on the ion energy. At

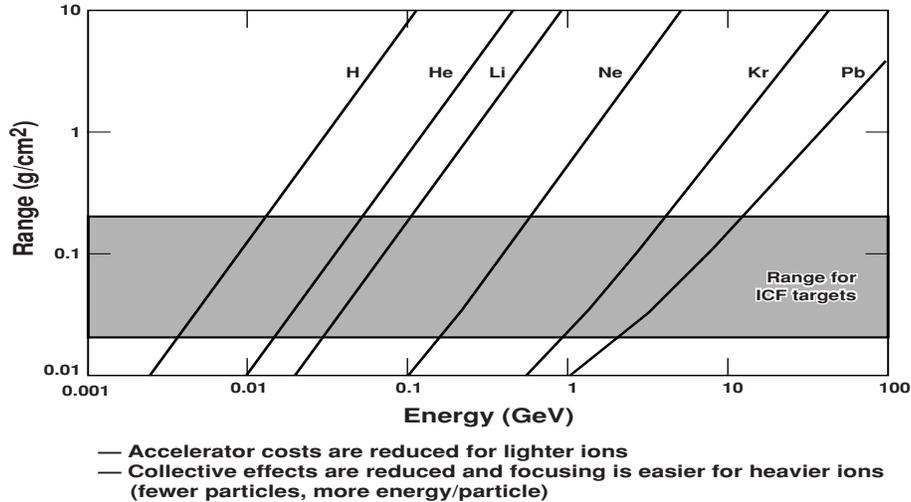


Fig. 2.26. A wide range of ion energies and masses is suitable for ion-driven targets.

the upper end of the ion energy and mass, the currents and space charge effects are small enough that vacuum focusing without charge or current neutralization may be adequate. At the lower energy range, adequate focusing requires a very high degree of current and charge neutralization. Since projected accelerator cost for a given number of joules generally increases for higher particle energies, there is a tradeoff between the difficulty of achieving adequate focusing and accelerator cost. Current expectations are that an intermediate mass ion in a range between K and Cs will prove to be optimal. High energy accelerators are well suited to many of the requirements of inertial fusion power production. They are reliable and durable. They can be efficient, and they can easily produce the needed repetition rates. The beams are focused by magnetic fields. The magnets that produce these fields can be shielded from neutrons, gamma rays, and other fusion products. This possibility provides a plausible solution to the problem of developing optical elements that can survive in the fusion environment.

Several types of accelerators are being developed. Single gap accelerators, using pulsed power, have been developed for accelerating light ions such as H and Li to the required particle energy. For heavier ions, an RF linac (followed by storage rings) is the principal approach being developed in Europe and Japan. The major program outside the United States is at GSI in Garching, Germany. An induction linac (without storage rings) is the main approach being developed in the United States. Existing proton and electron accelerators are comparable to power plant drivers in terms of size, cost, total beam energy, focusing, average beam power, pulse repetition rate, reliability, and durability. High peak power with adequate brightness is the new requirement for inertial fusion. Use of multiple (~ 100) beams and pulse compression after acceleration ($10\times$ or more) implies a power of $\sim 0.1\text{--}1$ TW/beam out of the accelerator. At ~ 3 GeV, this is $30\text{--}300$ A. In addition to the current increase obtained by the increased ion energy, typical driver designs compress the pulse a factor of several during acceleration, so they require injected currents of ~ 1 A/beam or less from a 2-MeV injector. For comparison, the ISR at CERN had a beam power of 1 TW at 30 GeV. Most ion beam experiments for IFE have been scaled, using beams of $10\text{--}20$ mA, but they have tested crucial beam physics in the right dimensionless regime, for example, with driver-relevant dimensionless perveance of up to $\sim 4 \times 10^{-4}$. (Perveance is essentially the squared reciprocal of the distance in beam radii that an unconfined beam can travel before expanding by one beam radius.) Current amplification by a factor of a few has been achieved. Peak beam powers are in the megawatt range.

The heavy ion fusion program evaluating induction linacs has emphasized theory, numerical simulation, and small-scale experiments to address the key issue of focusing high-intensity heavy ion beams. Small-scale experiments which address all beam manipulations and systems required in a full-scale driver have been completed or are near completion. These include a scale focusing experiment that produces millimeter focal spots, an experiment that combines four beams transversely while retaining good beam quality, experiments on beam bending, a target injection experiment that demonstrated adequate accuracy

for indirect drive, and experiments on beam physics and injector physics. The small accelerators for the scaled experiments can operate continuously at repetition rates of the order of 1 Hz, but the beam currents are approximately two orders of magnitude smaller than those required in a full-scale driver. A 3-D numerical simulation capability has been developed, which has been very successful in modeling these experiments, suggesting that it will be possible to achieve adequate focusing for accelerators using the vacuum focusing approach. Preliminary simulations indicate that a low-density plasma can be used to neutralize the beams in the chamber, enabling the use of lower ion kinetic energy and leading to lower accelerator cost. The degree of neutralization and beam plasma instabilities are the issues that will determine the effectiveness of this approach to beam focusing. It is not possible to do definitive experiments on neutralization using existing U.S. heavy ion accelerators. However, intense beams of light ions are available. These beams may offer the best near-term opportunity for experiments in an IFE relevant parameter regime. The heavy ion beams at the GSI nuclear physics research center in Germany may also provide important information.

A development program has the potential to significantly reduce the cost of key accelerator components including insulators, ferromagnetic materials, and pulsers. Initial results from industrial contracts predict that with development, large reductions from current costs are possible for some key components. Using advanced components and recent target design advances, current studies indicate that the direct cost for a DEMO scale driver of <\$0.5B may be feasible.

Goals for Phase I development include the following elements: (a) Complete the present scaled experiments including the study of various possible ion focusing modes. (b) Develop an end-to-end simulation capability. (c) Complete beam physics and injector experiments at driver scale. This means increasing the current in beam physics experiments from the present 10 mA to 1 A and increasing injector currents from the present 1 A to 100 A. (d) Develop inexpensive quadrupole arrays, pulsers, insulators, and ferromagnetic materials for induction cores.

The goal for Phase II is to design and build a multi-kilojoule IRE accelerator facility. Results from this facility in accelerator science, beam focusing, chamber physics, and those aspects of ion target physics that cannot be done on a laser facility, such as the NIF, must be sufficient to justify a high average fusion power IFE facility in Phase III.

At the CE level, source development for light ions could have high leverage. A light ion pulsed power driver would have the lowest cost of proposed ion drivers. The DOE–DP program in light ions achieved a lithium beam intensity $\sim 10^{12}$ W/cm² but was unable to exceed this intensity. Significantly higher intensities might be feasible if an ion source with adequate brightness and beam quality could be developed.

2.3.4.2 Lasers

The KrF laser is an excited dimer (excimer) laser that produces broadband light (2 THz) centered at 0.248 μm . For the high energy systems required for IFE the gas is pumped by large area electron beams. All the large KrF lasers (energies of 1 kJ or greater) are single shot devices developed for the ICF program or for basic science experiments. The NIKE laser at the NRL has been in operation for 3 years and has demonstrated that a KrF laser can be a reliable target shooter (over 600 shots per year). NIKE has the best beam uniformity of any high-power UV laser. It appears to meet the beam uniformity requirements for IFE. The challenge for a KrF laser is to demonstrate that it can meet the fusion energy requirements for repetition rate, reliability, efficiency, and cost. The key issues are (a) the efficiency, durability, and cost of the pulsed power driver; (b) the lifetime of the electron beam emitter; (c) the durability and efficiency of the pressure foil support structure (“hibachi”) in the electron beam pumped amplifiers; (d) the ability to clear the laser gas between pulses without degrading the beam quality, and (e) the lifetime of the amplifier windows and optics in the laser cell. Technologies have been identified that may be able to address these issues. Most have been partially developed elsewhere, but they have been developed separately from each other and not necessarily in a parameter range appropriate for IFE. A leading can-

didate for the KrF pulsed power is based on the Repetitive High Energy Pulsed Power (RHEPP) developed at Sandia. RHEPP II has achieved a broad area electron beam (2 MV, 25 kA, 60 ns, and 1000 cm²) at up to 100 Hz and up to 30-kW average power. NRL is now developing the Electra KrF laser as a step toward developing the capabilities required for IFE.

The KrF goals for Phase I include: (a) Complete design and construction of Electra with a goal of ~400 J/pulse, 5-Hz repetition rate, 10⁵ shots durability (as a first step in the goal of >10⁸ shots), 5% total efficiency. Electra will demonstrate that it is possible to repetitively amplify a laser beam that meets the requirements for bandwidth and beam quality; (b) Develop technology to meet the requirements for pulsed power cost; (c) On NIKE (60-cm amplifier, single shot), demonstrate intrinsic efficiency (laser energy out divided by deposited energy in the gas) and electron beam transmission through the hibachi with a large system that is directly scalable to an IFE laser beam line; (d) In separate, off-line, experiments, develop new window coatings for the amplifiers.

The Phase II goals for KrF follow: (a) Develop a full-scale KrF amplifier that meets all the requirements for IFE. The laser output of this amplifier will be in the range of 30–100 kJ, with an optical aperture ~2 m². The amplifier will run at 5 Hz and would be the prototype for an IFE beam line. (b) Incorporate this prototype into an IRE that is designed to demonstrate the requirements for IFE, including beam propagation under required chamber conditions and an ability to track and hit an injected target. (c) Identify final optic materials and system designs to withstand megaelectron-volt neutrons, gamma rays, and contamination in an IFE power plant.

Solid-State Lasers. Since the earliest days of inertial fusion research, solid-state lasers have served as the main workhorse for unraveling crucial target physics issues. First generation solid-state lasers, initially at the 100-J level in early 1970s based on flashlamp-pumped Nd:glass, will culminate with the 1.8-MJ NIF. To attain the objectives of fusion energy, second generation solid-state lasers will employ diodes in place of flashlamps, Yb-doped crystals instead of Nd:glass, and near-sonic helium cooling of optical elements. A diode-pumped solid-state laser (DPSSL) should have the reliability and efficiency needed for a fusion power plant. The largest risks are believed to be the optical smoothing and the cost of the diodes. Extensive ongoing research in glass lasers will help provide the technical basis for demonstrating that the beam uniformity and overall system design can be improved to meet IFE target requirements. LLNL has concepts that could lead to a large reduction in the current cost of the diodes and achieve the required beam uniformity. LLNL is currently developing the 100-J Mercury DPSSL to demonstrate these and other capabilities. Previous work at LLNL demonstrated a diode-pumped gas-cooled Yb:crystal laser at the joule-level, extracted the stored energy at 70% efficiency with nanosecond pulses in a separate experiment, and developed a laser diode package with 10⁷–10⁸ shot lifetime at 100 W/cm. Yb:S-FAP crystals have continued to progress although instabilities in growth are not yet completely under control.

The DPSSL goals for Phase I follow: (a) Complete design, assembly, and testing of Mercury Laser operating at 1.05 μm; achieve 10% efficiency with 100 J/pulse, 10-Hz repetition rate, 2-ns pulse width, 5× diffraction-limited beam quality, and 10⁸ shots. (b) Perform and validate system-level analysis of achievable beam-smoothness on-target in power plant scenario for solid-state laser. (c) Upgrade Mercury Laser to incorporate average-power frequency-conversion, deformable mirror, and beam smoothing technology. (d) Develop the technology approach for future kilojoule-class DPSSLs.

The DPSSL goals for Phase II follow: (a) Develop technologies to construct ~4-kJ beam line using low-cost diode arrays (\$0.50/peak-W), operating with 10% efficiency and 10⁹ shot lifetime at 0.35-μm wavelength. At least two independent apertures will be integrated to form this beam line, using very high-quality gain media at full size. (b) Incorporate the 4-kJ beam line into an IRE that is designed to demonstrate the requirements for IFE including beam propagation under required chamber conditions and an ability to track and hit an injected target. (c) Identify final optic materials and system designs to withstand megaelectron-volt neutrons, gamma rays, and contamination in an IFE power plant. (d) Define a pathway to achieve a diode pump cost of \$0.07/W-peak or less in a fusion economy.

2.3.5 IFE Fusion Target Concepts and Design

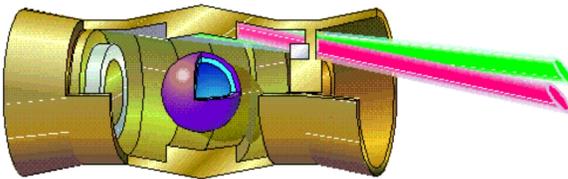
Though NIF will be able to explore ignition and propagating fusion burn, significant additional target design work must be performed to achieve higher gain, consistency of the target design with various IFE driver capabilities, and the illumination requirements of various power plant chamber concepts.

2.3.5.1 Ion-Beam-Driven Targets

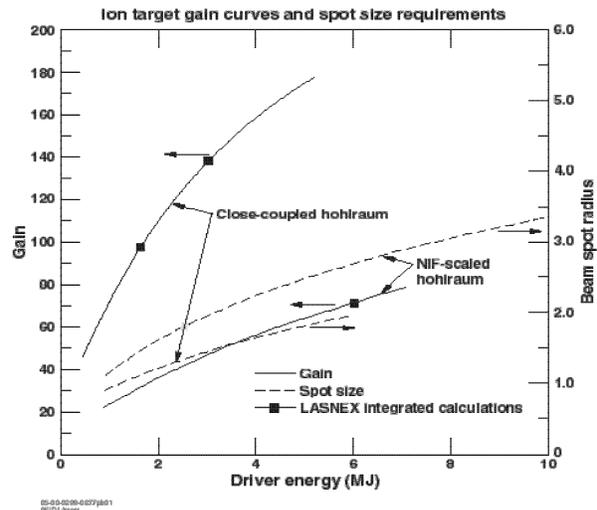
Indirect-drive target designs which meet the gain requirements of fusion energy have been developed at a variety of driver energies (down to as low as 1.7 MJ). Further theoretical and experimental work is needed to validate various aspects of the simulations and to better evaluate the sensitivity of targets to issues such as beam pointing. Also, new designs which relax accelerator phase space requirements and/or lower system costs have high leverage.

The best modeled current target designs rely on radiation which is generated from the ion beams absorbed in a radiator distributed through the hohlraum volume as shown in Fig. 2.27. Implosion symmetry depends on the details of the mass distribution inside the hohlraum, the beam pointing, and the energy loss rate of the ions as they traverse the plasma. Two gain curves are given in Fig. 2.27. One gain curve gives gains for targets which have a ratio of the hohlraum radius to the capsule radius comparable to that of NIF indirect-drive targets shown in Fig. 2.22. By using materials which are in near pressure equilibrium throughout the hohlraum volume in this distributed radiator target, detailed calculations predict that it will be possible to use hohlraums smaller than those needed for laser drivers on the NIF. The higher gain of these close-coupled designs will have to be balanced against the more severe requirements on the beam spot size indicated in Fig. 2.27.

The simulations for the designs of the type in Fig. 2.27 are comparable in complexity to calculations carried out for targets planned for the NIF. Although many aspects of the computational methods used in these calculations have been tested in a wide variety of laser experiments, for ion beam drivers, there is a need for continuing improvement in the physics algorithms and in the detail incorporated in these calculations. Code development to improve the ion deposition models is needed. A true 3-D radiation-hydrodynamics capability including 3-D ion beam ray tracing is needed so that one can do a better job of assessing the effects of pointing errors on symmetry. Although 3-D codes are being developed under the ASCI for DP, modeling ion beam deposition is an energy-specific requirement.



Indirect-drive target for ion beam fusion energy using distributed radiator foam radiation case



Ion target gain and spot size

Fig. 2.27. Distributed radiator ion target design. The fuel capsules in these targets have a radius of 2–3 mm.

To validate the calculations for ion beam targets, some experimental tests are needed beyond those which will be carried out by the DP ICF Program. For the target above, possible experiments are: (1) laser or z-pinch driven hohlraums using very-low-density low-Z foams in pressure balance with low-density hohlraum walls; (2) ion-driven hohlraums, in collaboration with European laboratories, which achieve modest temperatures and pressures and test ion deposition and radiation generation; and (3) radiation-driven RT experiments which include the effect of low-density foams to investigate the stability of closely coupled targets.

Many target designs with potential advantages for energy production with ion beams have not yet been adequately evaluated. Some possibilities include: (1) an ignition target with substantially less than 1 MJ of beam energy; (2) designs with increased coupling efficiency which produce increased yield at fixed input energy; (3) larger spot/lower intensity hohlraums for relaxed accelerator requirements; (4) alternative radiator designs which have reduced pointing and spot size requirements; (5) large spot size targets in which the beam enters through the sides of the hohlraum not the ends; (6) targets which accept beam illumination from one side.

2.3.5.2 Laser-Driven Targets

With our current understanding, the high energy gains required for laser-driven IFE require that the laser beams directly illuminate the target. The DP-sponsored activity in direct drive is currently centered at the University of Rochester and the NRL.

The gain achievable with direct-drive targets is critically sensitive to laser beam smoothing, and a variety of beam smoothing techniques have been developed. The most uniform beams have been produced by a technique called ISI, invented and developed by scientists at NRL. In 1995 NRL completed the NIKE KrF gas laser with ISI and measured an intensity nonuniformity at the focus of each laser beam of only 1%. This nonuniformity was an order of magnitude lower than previous UV lasers and is calculated to meet the IFE requirements. Direct-drive target acceleration experiments throughout the 1990s on Nova, Omega, Gekko XII, and NIKE have mimicked the early-time behavior of a fusion target implosion. The level of agreement between the computer modeling and the experiments provides some confidence that computer modeling can be used to design high gain direct-drive fusion targets for IFE. Using these computer models, scientists from the University of Rochester and NRL have calculated the performance of a variety of direct-drive targets designed for high gain, Rochester results from 1-D calculations, in which pulse shape variations are used to control hydrodynamic instability growth, are indicated by the direct-drive curve band shown in Fig. 2.23. The width of the band is determined by the fuel entropy. Rochester 2-D calculations, including estimates for laser imprinting and capsule fabrication tolerances, indicated that gains in the middle of this band (20–40) may be feasible at NIF-like energies of 1.4–1.8 MJ. NRL scientists have designed direct-drive targets using low-density plastic foam ablaters. An NRL target design with gain 80 at 3.5 MJ is indicated in Fig. 2.23. This gain is estimated from 2-D calculations that include effects of laser beam imprinting and hydrodynamic instability growth. For the direct-drive calculations indicated in Fig. 2.23, the laser was assumed to have a single focal spot throughout the implosion. If the laser focal spot can be zoomed to follow the target as the implosion proceeds, higher gains are possible. For example, the NRL target gain plotted in Fig. 2.23 would exceed 100 with zoom. NRL is also examining target designs that use a high-z overcoat for controlling laser imprinting and hydrodynamic instability growth. In 1-D with zoom, these targets have calculated gains exceeding 100 for laser energies of 1–2 MJ. These designs require additional assessment with 2-D implosion codes to determine if they provide sufficient control of fluid instabilities. Eventually, 3-D calculations will have to be used to correctly model random incoherence that is inherent in the laser beams. Further work will determine the optimal direct-drive capsule design for controlling laser beam imprinting and instability growth as well as for such requirements as injection into a fusion chamber.

In the current National Ignition Plan, direct-drive targets will be tested in NIF in FY 2008 or 2009, following completion of the indirect-drive ignition experiments. Much of the direct-drive target design will be done as part of the DP ICF Program. However, there are aspects of these designs which are unique to

IFE and are included in the IFE development plan. The lasers being proposed for IFE will have different opportunities and limitations compared to the NIF. Direct drive for IFE will also require targets with higher yield and gain than those for NIF, and these will require additional calculations. We require 2-D and 3-D calculations which incorporate the smoothing techniques appropriate for KrF or DPSSL lasers for the different direct-drive target designs. Since high gain is essential for IFE, calculations will examine physics effects which could increase the gain, including a search for new techniques to reduce the RT instability. To accurately assess the achievable gain, improved physics models may be necessary for effects such as X-ray production and transport in the low-density target corona, equation of state for foams, and nonlocal electron transport. Implosion techniques which do not require symmetric illumination would increase the range of chamber options and perhaps open up the possibility of protected wall designs for laser-driven IFE.

Indirect drive with lasers is the best understood ICF target concept. It has received the bulk of the DP ICF funding. However, a gain curve based on the NIF point design is too low for economic energy production unless laser driver efficiency can be increased to about 20%. However, it may be possible to increase the efficiency of laser-driven hohlraums. IFE specific calculations would explore the feasibility of using a variety of techniques, including those developed in the heavy ion design, to substantially raise the gain curve.

Fast Ignitor. Because of their potential for higher gain and reduced driver size, fast ignition targets should be further evaluated. These types of targets are at the Concept Exploration level. An ongoing program of experiments, theory, and numerical calculations will be required to evaluate this potential approach to IFE. Integrated target designs in 2-D and 3-D are needed, which incorporate the results of experiments in coupling and electron transport. For example, it is important to evaluate asymmetric implosions and cone focus geometries which minimize the path length of plasma through which the high-intensity ignition beam must pass. The Fast Ignitor concept requires accurate calculation of relativistic electron currents of about 10^9 A with a return current of approximately equal magnitude. To model these conditions, improved electron transport models will be necessary.

2.4 Technology Opportunities

2.4.1 Overview and Recent Progress

The dramatic progress in fusion science seen in the last few decades has been possible, in part, due to equally dramatic progress in technology in general and plasma technologies in particular. These include the technologies to confine the plasma (magnet coil sets, plasma facing components) and those which are used to manipulate the plasma parameters and their spatial and temporal profiles—plasma heating and current drive, plasma fueling systems, drivers and targets—see Table 2.6 for recent successes.

Experimental advances in plasma performance and progress in theoretical understanding of plasmas places the world fusion program on the threshold of developing systems that generate substantial amounts of fusion energy. Exciting first steps have been taken in TFTR and JET. Next-step devices, for example, the proposed ITER-RC (Sect. 2.2.2.4) and the NIF (Sect. 2.3.2.4), present major challenges in terms of component performance and reliability, fuel handling systems including tritium technology, and maintenance concepts. Moreover, the choice of materials and design concepts for “in-vessel” components (e.g., divertor, first wall, blanket, shield, final optics, and vacuum vessel) will more than anything else determine the safety and environmental characteristics of both magnetic and inertial fusion energy.

Systems design activities, such as those carried out in the ARIES, Prometheus, and SOMBRERO studies, are an important element of the Technology Program because they help to provide the essential framework to construct the overall strategy of the U.S. program. The system studies element provides the “energy” motivation for the future directions of the Fusion Energy Sciences Program: by examining the potential of specific concepts as power and neutron sources; defining R&D needs to support near-term

Table 2.6. Examples of recent technology achievements

MFE

- Participation in the design and analysis of ITER—the most comprehensive effort to date on a fusion power source.
- Construction of the world’s most powerful pulsed superconducting magnet.
- Pellet injection systems with speeds of 2.5 km/s and production of 10-mm tritium pellets.
- Antenna advances for ion cyclotron RF systems such as folded waveguides and comb lines.
- Operation of 1-MW, 110-GHz gyrotrons and development of 170-GHz tubes.
- Demonstration of Be/Cu and W/Cu high heat flux components operating at up to 10 MW/m² and identification of W as a possible erosion-resistant plasma-facing material under detached plasma conditions at the divertor.

MFE & IFE

- Study of helium cooling of high heat flux components and conceptual design of helium-cooled blankets coupled to closed-cycle gas turbine energy conversion systems.
- Study of the thermomechanical behavior of solid breeder blanket concepts.
- Experiments and modeling to verify performance of liquid metal blanket concepts.
- Significant contributions in understanding radiation effects in materials, using molecular dynamic simulations.
- Determination of irradiation effects on the toughness of vanadium and ferritic steel alloys.
- Study of response of basic material properties of low-activation ceramics (e.g., SiC composite) to neutron radiation.
- Understanding of tritium retention characteristics of Be, W, and mixed materials.
- Invention and development of the palladium membrane reactor for efficient tritium recovery.
- Experimental verification, using 14-MeV neutron sources, of shielding, decay heat, and activation nuclear data and codes.
- Development of a database and understanding of the chemical reactivity and volatilization behavior of fusion materials in steam and air at high temperature.
- Demonstration that a D-T burning plasma facility can meet no-evacuation safety criteria.
- Development of attractive tokamak, alternate MFE, heavy-ion and laser-driven IFE concept power-plant conceptual designs.
- Development of physics and engineering solutions to several major design problems for next-generation devices.

IFE

- Integrated testing of full-size induction modules for IFE heavy ion drivers.
- Successful operation of the Nike KrF laser.
- Gas cooling of DPSSLs up to 25 Hz.
- Development of long-optical storage-time crystalline SSLs.
- Annealing of light transmission losses in neutron- and gamma-irradiated fused silica final optics.
- Development of smooth cryo-D-T layers by beta-layering in inertial fusion targets.
- Development of smooth liquid jets for protection of IFE chamber walls.
- Experiments on free surface flows for IFE chamber protection using films and jets.

experimental studies; incorporating plasma and target physics R&D into design methods; analyzing potential pathways to fusion development; carrying out systems analysis of economic and environmental performance; and designing next-step devices.

Near-term emphasis is on developing better tools for the production and control of high-temperature MFE plasmas and, thus, the further development of plasma science. For IFE, research is focused on key feasibility issues that bear on the high-pulse-rate application of candidate drivers for IFE.

The longer term emphasis is on resolving key feasibility issues, including extraction and utilization of heat from fusion reactions, breeding and handling of fuel (tritium) in a self-sufficient system, demonstration of remote maintenance systems and reliable operation, and realization of the safety and environmental potential of fusion energy. The development of reduced-activation materials is particularly important to realize the environmental potential of fusion energy.

2.4.1.1 The Cost of Electricity

The importance of technology to the development of an economically and environmentally attractive fusion energy source and in contributing to the four major challenges described in Sect. 2.2.2.1 can be captured by considering the cost of electricity.

A more attractive reactor embodiment of any MFE concept would obviously result from reducing the capital cost, increasing reliability, reducing in-vessel component failure rates, and/or increasing the net fusion power. Reduced capital costs could be achieved with smaller fusion cores resulting from higher performance plasmas; that is, higher fusion power densities achievable through higher confining magnetic field strengths and high plasma β . Higher field strength superconducting Magnet Technology, RF Heating and Current Drive systems operated in a manner to stabilize MHD activity, and Plasma Facing Component (PFC) technology aimed at facilitating edge transport barriers would be the three principal technology program elements directly applicable to increasing the fusion power density.

The remaining elements of the technology portfolio also play a central role in lowering the cost and increasing the environmental acceptance of fusion energy. For example, net fusion power can be maximized not only by reducing the recirculating power fraction, which implies superconducting magnet technology and more efficient heating and noninductive current drive systems, but also by extracting heat at higher temperature for improved thermodynamic efficiency. The latter is being addressed in the PFC, Fusion Technology, and Materials program elements. Similarly innovative research in the Fusion Technology program aimed at developing thick liquid walls to absorb the bulk of the neutron energy may offer a promising solution to reduce in-vessel component and structural material failure rates (reduced component replacement costs and higher availability). Improved techniques for Remote Handling and Maintenance are also essential for fusion power systems in general and figure heavily in increasing availability. The Tritium System and Fusion Safety elements of the portfolio speak directly to the environmental attractiveness of fusion power in general and licensing issues of next-step burning plasma devices in particular.

2.4.2 The Technology Portfolio

2.4.2.1 Plasma Technologies

Plasma Heating and Current Drive. Significant progress has been made in developing and deploying high-power gyrotrons in the ~ 1 -MW level at 110 GHz (see Fig. 2.28) and the development of 170-GHz prototype units for electron cyclotron heating/current drive (ECH/ECCD) and fast-wave (FW) antenna arrays in the >1 -MW unit size for ion cyclotron heating (ICH) and current drive (via direct electron heating). Progress is also being made in other countries on the development of negative-ion-based, high-power neutral beams (0.5–1.0 MeV). The emphasis of the development of these heating and current drive technologies will concentrate on improving power density (higher voltage limits for ICRF launchers), higher gyrotron unit power (2 to 3 MW), increased efficiency gyrotrons featuring multistage depressed collectors, ICRF tuning and matching systems that are tolerant to rapid load changes, and steady-state gyrotrons and actively cooled ICRF launchers for long-pulse/burning-plasma next-step options.



Fig. 2.28. Prototype 1-MW gyrotron.

Fueling. Fueling is another technology that is essential for achieving fusion-relevant parameters and manipulating plasma parameters to achieve peak performance (peaking of the density profile for higher reactivity and reduction of turbulent transport via turbulence suppression). Recent successes include sustained operation above the gas-fueled density limit in DIII-D, high-field launch with improved density profile peaking and internal transport barrier generation, the development of steady-state pellet injectors operating in the 1.5-km/s speed range, and the demonstration of core fueling in PoP experiments using accelerated compact toroids (CTs). Pellet fueling technology has also been used recently to ameliorate the effects of major disruptions (a potentially serious off-normal event) in tokamaks by delivering massive amounts of low- and high-Z material that rapidly quench the current in vertically unstable plasmas. A critical issue for fueling in next-step device plasma regimes is the degree to which profile peaking is needed (for higher density operation and improved reactivity and confinement) and the technological requirements to meet that need (pellet speed, CT density, and the physics of CT deposition).

Plasma Facing Components and Plasma Materials Interactions. The successful development of high-performance (high heat flux, low-erosion) PFCs and the understanding and the control of the interaction of the plasma material surfaces is important in creating edge plasma conditions that are conducive to developing an edge transport barrier (H-mode) and the development of low-erosion PFCs will have a strong impact on component lifetimes and COE. Recent progress includes (1) the understanding of net divertor erosion pointing to refractory high atomic number materials, (2) mixed materials and co-deposited carbon-tritium films, (3) the development of innovative wall conditioning techniques, and (4) water-cooled PFCs (Be/Cu and W/Cu) with steady-state heat removal rates at the 10- to 30-MW/m² level. A free surface liquid divertor project (ALPS) has recently been initiated to investigate the potential of active heat removal without concern for PFC lifetime limits (see Fig. 2.29). Critical issues that need to be addressed are the development of even higher surface heat flux PFCs (50-MW/m² goal) that do not require periodic maintenance to renew the plasma-facing material (i.e., liquid surfaces or helium-cooled nonsputtering refractory metals). In tokamak experiments, investigations are underway to distribute the heat flux more evenly via radiation without confinement degradation.

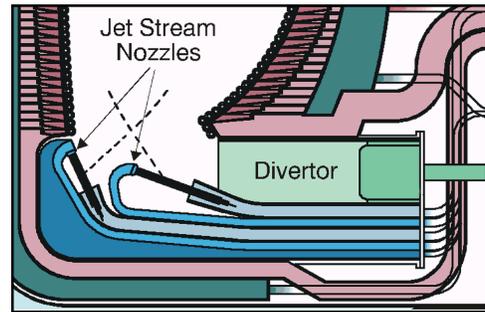


Fig. 2.29. ALPS—Advanced limiter-divertor plasma-facing systems.

Magnet Technology. Superconducting magnet systems which provide the confining magnetic fields represent a major cost element for long-pulse or burning-plasma next-step MFE options. Dramatic progress has been made recently in development of large-scale DC and pulsed Nb₃Sn magnets for ITER at a field strength of up to 13 T (see Fig. 2.30). Further reductions in cost for superconducting magnets could be realized by development of a higher performance (higher current density and increased quench protection capability) superconductor strand, higher strength structural materials, and higher radiation-resistant magnet insulators (which presently limit the life cycle of magnet systems). Dramatic progress has been made with the development of high-temperature superconductors which can be applied to certain fusion problems (e.g., leads for magnets). Quadrupole focusing magnets for heavy ion beam fusion are also a major contributor to the cost of the heavy ion driver. The development of large, warm bore quadrupole arrays has been identified as a key element in developing an affordable next-step Heavy Ion Fusion (HIF) system.



Fig. 2.30. Central solenoid model coil.

2.4.2.2 Nuclear Technologies and Safety

Tritium Processing and Fusion Safety. The safe handling of tritium fuel and tritiated exhaust streams, the minimization of tritium holdup and inventory in in-vessel components, and the understanding (and mitigation) of tritium and activation product mobilization and release are critical to the goal of demonstrating fusion power with attractive safety and environmental characteristics. Significant progress has been made in the development of cryogenic distillation systems for isotope separation and the demonstration of a novel once-through exhaust gas cleanup system (Palladium Membrane Reactor) that efficiently processes tritiated water and has the potential to eliminate tritiated water altogether in fuel processing systems.

From data generated on the mechanisms for mobilization and migration of radiologically hazardous materials and the development of state-of-the-art safety analysis tools, ITER was designed with the confidence that public evacuation would not be required under worst case accident scenarios. Critical/development issues in this area are the minimization or elimination of waste streams (such as tritiated water from fuel cleanup systems) and demonstration of the feasibility of recycle and reuse of fusion materials, minimization and removal of tritium in first-wall materials and co-deposited layers and understanding the interaction between energy sources and the mobilization of tritium and other radiological hazards, and safety R&D and development of techniques for removal of tritium from advanced coolants (i.e., liquid walls) now being considered for future MFE and IFE reactor-class devices.

The accurate estimation of accident consequences requires knowledge of radionuclide release fractions and an understanding of the time-temperature history during an accident. More detailed scenarios need to be developed for power plant concepts and target fabrication facilities.

Remote Handling and Maintenance. In eventual MFE and IFE fusion reactors, all in-vessel maintenance will need to be performed remotely because of activation of materials in the intense radiation environment. Rapid in situ repair operations are important from the perspective of achieving adequate power plant availability levels. Recent successes include remote-handling operations performed on JET, the development of precision in-vessel metrology systems, and demonstration of ITER blanket and divertor remote-handling concepts. Significant additional development will be required to reduce costs, improve reliability and human interfaces, develop dexterous servo manipulation of heavy payloads, and techniques for remote welding and refurbishment of in-vessel components.

Plasma Chamber Technologies. This effort will identify and explore novel, possibly revolutionary, concepts for the in-vessel components that can substantially improve the vision for an attractive fusion energy system. The R&D will focus on concepts that can have high power density, high power conversion efficiency, low failure rates, faster maintenance, and simpler technological and material requirements. This R&D, involving international collaboration, includes theory, modeling, experiments, and analysis in key areas of engineering sciences (e.g., fluid mechanics, MHD, heat transfer, thermomechanics, plasma-material interaction, nuclear physics, and particle transport) and materials, engineering, safety, and other technical disciplines. Also, an assessment will be made of the need for a Plasma-Based Neutron-Producing Facility for testing of heat extraction technology at high power density, data on failure rate, and data on maintainability. The near-term effort on innovative concepts will identify, analyze and evaluate novel, possibly revolutionary, high-performance advanced technology concepts within the APEX program (emphasis on high power density heat removal technology) (see Fig. 2.31).

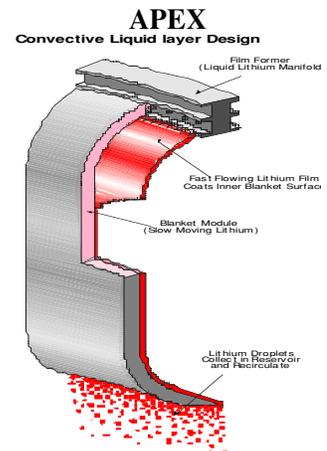


Fig. 2.31. Liquid walls can handle neutron wall loads up to 30 MW/m² with high surface heat loads.

2.4.2.3 Fusion Materials

The long-term goal of the Fusion Materials Program is to develop structural materials that will permit fusion to be a safe, environmentally acceptable, and economically competitive energy source. This will be accomplished through a science-based program of theory, experiments, and modeling that provides an understanding of the behavior of candidate material systems in the fusion environment, and identifies limiting properties and approaches to improve performance, and provides the materials technology required for production, fabrication, and power system design.

Fusion materials for MFE and IFE must operate in a very demanding environment which includes various combinations of high temperatures, chemical interactions, time-dependent thermal and mechanical loads, and intense neutron fluxes. One of the major materials issues to be faced in developing attractive fusion power is the effect of the intense neutron fluxes. The first-wall neutron spectrum from a D-T reacting plasma contains a large 14-MeV component. This not only results in high displacement rates (~ 20 dpa/year at a neutron wall loading of 2 MW/m^2) but also causes higher transmutation rates than are experienced in fission reactors. The transmutation products helium and hydrogen are of particular concern, but other impurities can also be important. The influence of transmutations on property changes has been very well established, the most well-known example being the role of helium in swelling behavior. Thus neutron irradiation is a particularly important issue, due both to its effects on physical and mechanical properties, as well as the production of radioactive materials, and is the most difficult to investigate with currently available facilities.

Three material systems have been judged to have potential for being developed as fusion power system structural materials: SiC composites, vanadium-based alloys, and advanced ferritic steels. High-temperature refractory alloys have been recently added to conceptual design evaluation. Copper alloys, because of their excellent thermal and electrical conductivity, are critically important in near-term applications and will most likely find special applications in fusion power systems including normal conducting coil options.

At present, fission reactors are the primary means to investigate the effects of irradiation on fusion materials. However, the response of materials to a fission radiation field can be significantly different from that due to a fusion neutron spectrum. Various techniques have been used to more nearly reproduce the fusion environment, but an intense source of 14-MeV neutrons will ultimately be needed to develop and qualify fusion materials. The international community has proposed a Point Neutron Source, an accelerator facility based on the D-Li interaction to fill this role. A key programmatic issue which remains to be resolved is the role of such a point neutron source vis-à-vis a VNS which could provide an experience database with a fusion system at moderate availability, as well as component testing and some materials testing capability.

2.4.3 IFE Chamber and Target Technology R&D

2.4.3.1 Chamber Approaches

Many concepts for chamber components have been advanced in design studies during the past 20 years. These include chambers with neutronically-thick layers of liquid or granules which protect the structural wall from neutrons, X rays, and target debris. There have also been chamber designs with first walls that are protected from X rays and target debris by a thin liquid layer, and dry wall chambers which are gas filled to protect the first wall from X rays and target debris. The last two types, the wetted wall and dry wall chambers, have structural first walls that must withstand the neutron flux. The currently favored approaches are (1) heavy ion drivers with indirect-drive targets and neutronically-thick liquid chambers and (2) laser drivers with direct-drive targets and gas-protected, dry-wall chambers.

Generic issues include: (a) wall protection, which involves hydraulics and flow control for liquids and includes ablation damage and lifetime for solids; (b) chamber dynamics and achievable clearing rate

following capsule ignition and burn; (c) injection of targets into the chamber environment; (d) propagation of beams to the target; (e) final-focus shielding and magnet/optics thermal and neutron response; (f) coolant chemistry, corrosion, wetting, and tritium recovery; (g) neutron damage to solid materials; (h) safety and environmental impacts of first wall, hohlraum, and coolant choices.

Neutronically-thick liquid walls. Current designs for neutronically-thick liquid walls, such as the HYLIFE-II chamber (Fig. 2.32) are only compatible with targets which can accept driver beams limited to a narrow range of directions. Other examples of protected wall chamber concepts include thick liquid vortex designs and designs which use a layer of lithium-bearing ceramic granules to reduce structural neutron damage. If successful, some fast ignitor target designs could meet the geometric requirements. In HYLIFE II, the use of a regenerative thick liquid, which recreates the fusion-pocket after disruption by each shot, allows the smallest fusion pocket size permitting the shortest final focus standoff. For heavy-ion drivers, minimizing final-focus magnet standoff reduces the focus spot radius by reducing the chromatic aberration and quadrupole fringe field aberration, reducing the required driver energy.

Prototypical, fully integrated chamber response experiments will not be possible until the ETF driver becomes available in Phase III of the IFE development plan. Efforts to demonstrate the technical viability of HIF fusion chambers, and then to produce fully integrated designs, must therefore use experiments scaled in energy and geometry, coupled with models that integrate phenomena studied in separate-effects experiments. For thick liquid-jet chamber concepts, experiments have approached the scaled conditions required to demonstrate vertical high-velocity, stationary-nozzle rectangular and cylindrical jets without spray generation and with sufficiently smooth surfaces to form the shielding grids. But prototypical parameter ranges in Reynold's number, Weber number, and nozzle contraction must be explored to address issues related to primary droplet ejection, smoothness, bending (horizontal jets), and acceleration (vertical jets) to be confident that these jets behave as required in the sensitive beam line area, while sweeping jets to form an oscillating pocket.

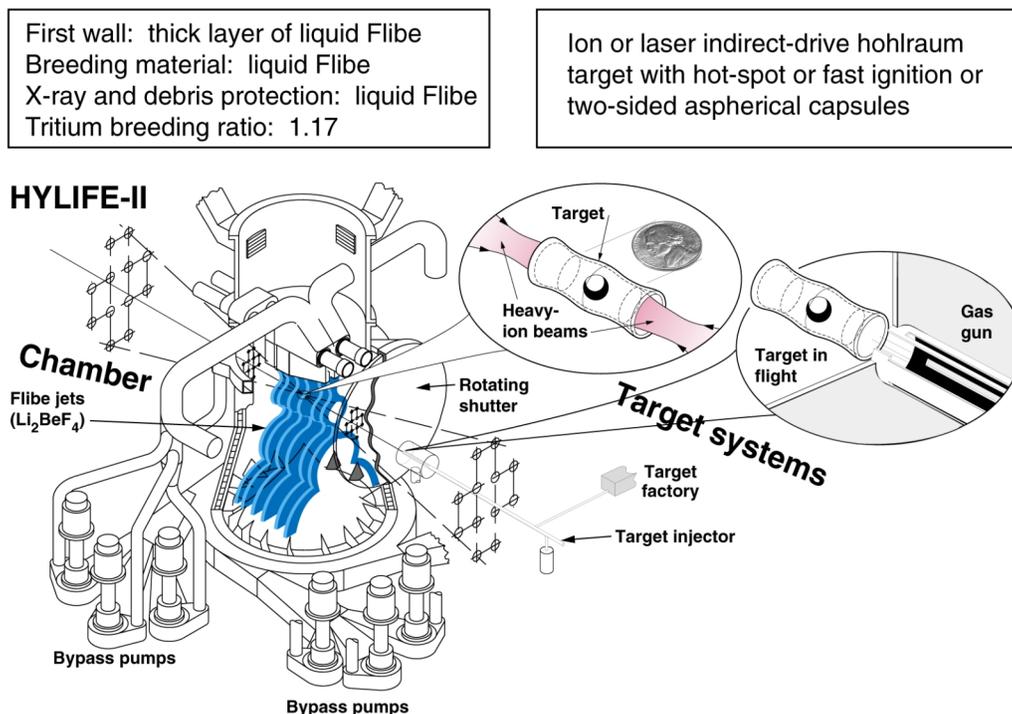


Fig. 2.32. HYLIFE-II liquid-jet protected chamber.

The objective for Phase I in the IFE development path is to assess the feasibility, through computer simulations and scaled experiments, of chamber clearing and vacuum recovery for thick-liquid protected chamber concepts for indirect-drive targets. The goal for Phase II is to test beam chamber transport, debris protection, and vacuum recovery in IRE chambers which are designed to simulate future ETF chambers. Phase II will also involve larger-scale ETF model chambers using NIF and other large X-ray sources such as z-pinch.

Wetted walls. Wetted-wall chamber concepts are relevant for both heavy ion and laser drivers. The thin liquid layers in these concepts provide protection against damage by X rays and target debris, but not neutrons. Several subvariants exist which use rigid or flexible woven substrates in various geometries. Wetted walls offer different technical risks than thick-liquid walls, including the need for low-activation fabric or solid structures that can withstand neutron damage. For heavy ions, current wetted-wall chamber concepts require larger final-focus standoff than liquid wall chambers. The primary hydrodynamics problems involve creating and regenerating a protective target-facing liquid film and estimating the minimum permissible target chamber radius to accommodate stresses from neutron isochoric heating and control vapor evaporation from the liquid film. Even without an available intense neutron source capable of inducing liquid breakup by isochoric heating, important data can be obtained by using a small laser to induce transient rarefaction shocks sufficient to fracture liquids.

Dry walls. Dry-wall chambers are potentially applicable to both ion beam and laser drivers. However, because of their geometric flexibility, larger size, and potential for low debris generation, they are currently the design of choice for lasers using direct-drive targets. Dry-wall chambers typically rely upon a high-Z gas such as xenon or krypton to attenuate X rays and debris, so that the first wall experiences a lower re-radiated flux of X rays over a longer time. For example, the SOMBRERO design uses dry carbon/carbon composite walls with 0.5 torr of xenon in the chamber. Recent NIF design studies suggest that alternative materials for plasma facing components, such as boron carbide, may have even better X-ray response and that louvered first wall configurations may help further control ablation debris mobilization. Some preliminary calculations suggest that it may be possible to use magnetic fields to divert debris from the first wall and optics in both wetted-wall and dry-wall concepts. On the other hand, recent calculations indicate that the erosion rate of irradiated carbon fiber composite, even under the moderated heat pulse conditions in SOMBRERO, may remain a design driver. Further potentially serious issues for the use of a carbon first wall include rapid radiation-induced swelling and buildup of tritium retention.

During target injection the gas and wall apply a heat load to the target through radiation, conduction, and friction with the gas. This is an issue for unprotected direct-drive targets in particular. Again, tests of many of these phenomena can be studied at small scale on a variety of ICF facilities. The Z-machine facility, besides having 2 MJ of X-ray output will have a multi-kilojoule 0.35- μm laser that could be used for propagation studies through simulated fusion chamber environments.

Final laser optics are susceptible to damage by target emissions and chamber gas conditions. Problems requiring solutions include: dust and vapor from the fill gas could stick to the final optics, leading to damage when the depositions are vaporized by the laser beams; radiation from the target explosion is a threat to the final optics, as is the shock generated in the chamber gas fill; heating and swelling from neutron damage to the optics can cause geometrical changes in the surface that can lead to degraded focusing; and sound waves in the chamber gas fill can cause vibrations in the final optics that also lead to defocusing. Gas dynamic windows might fix the issue of debris collection on final optics.

A goal for Phase I is to assess the feasibility, through computer simulations and scaled experiments, of dry-wall chamber first wall and final optic ablation and debris protection for laser drivers. Another goal for Phase I is to assess IFE chamber and final optic materials development requirements and the potential role of a laser-driven micro-neutron source to support the materials science. The goal for Phase II is to test beam chamber transport, as well as ablation and debris protection in IRE chambers which are designed to simulate future average fusion power ETF chambers. Larger-scale ETF model chambers using NIF and other large X-ray sources including z-pinch will also be tested.

2.4.3.2 Targets

Target Injection and Tracking. An IFE chamber requires $1\text{--}2 \times 10^8$ cryogenic targets each year at a rate of up to 10 Hz injected into the center of a target chamber operating at a temperature of 500–1500°C, possibly with liquid walls. The targets must be injected into the target chamber at high speed, optically tracked, and then hit on the fly with the driver beams.

Preliminary design studies of cryogenic target handling, injection, and tracking for both direct-drive and indirect-drive IFE power plants were done as part of all major IFE design studies. The direct-drive SOMBRERO design used a light gas gun to accelerate the cryogenic target capsules enclosed in a protective sabot. The indirect-drive OSIRIS design used a similar gas gun system without a sabot for injection and crossed dipole steering magnets to direct the beams.

A gas gun indirect-drive target injection experiment at LBNL showed that relatively simple gas gun technology could repetitively inject a non-cryogenic simulated indirect-drive target to within about 5 mm of the driver focus point, easily within the range of laser or beam steering mechanisms, but not sufficient to avoid the need for beam steering. Photodiode detector technology is adequate to detect the target position with the $\sim\pm 200\text{-}\mu\text{m}$ accuracy needed for the driver beam positioning. A next logical step in demonstrating the feasibility of IFE target injection and tracking is to couple the indirect-drive target injection and tracking experiment to a beam steering system for an integrated PoP experiment. There is a similar need for direct targets. Finally, these systems must be made to operate reliably on a $\sim 5\text{-}$ to 10-Hz repetition-rated basis.

Target Fabrication. The fabrication techniques used for the DP ICF targets meet exacting product specifications, have maximum flexibility to accommodate changes in target designs and specifications, and provide a thorough characterization “pedigree” for each target, all done “by hand.” A completed target can cost up to \$2000. To keep the target production contribution to the cost of electricity below 1 ¢/kWeh, targets must be produced for less than about \$0.50 for a 5-Hz repetition-rate fusion power plant producing 1 GWe. Even lower target costs are, of course, desirable and appear feasible based on the cost of equipment for the mass manufacture of similar sized complex objects such as those produced in the electronics industry. For example, a target factory costing \$90M amortized at 10%/year with an operating and maintenance budget of \$9M/year could produce 10^8 targets/year for \$0.18/target.

The heart of an inertial fusion target is the spherical capsule that contains the D-T fuel. ICF capsules are currently made using a process which may not extrapolate well to IFE. The microencapsulation process previously used for ICF appears well-suited to IFE target production if sphericity and uniformity can be improved and capsule size increased. Microencapsulation is also well-suited to production of foam shells which may be needed for several IFE target designs. ICF hohlraums are currently made by electroplating the hohlraum material onto a mandrel which is dissolved, leaving the empty hohlraum shell. This technique does not extrapolate to mass production. Stamping, die-casting, and injection molding, however, do hold promise for IFE hohlraum production.

The objective for Phase I in the IFE development path is to assess potential methods for low-cost mass-manufacture of IFE targets, both for indirect and direct drive, including the development of suitable low-density foams for each type of target. In Phase II, the goal is to test performance of candidate indirect- and direct-drive IFE targets in both non-yield experimental chambers (like IRE) and in NIF.

Page Intentionally Blank

3. SCIENTIFIC CONTEXT OF FUSION RESEARCH

3.1 Introduction

The fusion energy quest *demand*s excellent science—it will fail without the nourishment of scientific advance and deep scientific understanding. At the same time, the fusion program has an extraordinary record in *generating* excellent science—bringing crucial insights as well as conceptual innovations to such disciplines as fluid mechanics, astrophysics, and nonlinear dynamics. Most of the scientific advances engendered by fusion research have begun as discoveries about the behavior of that most complex state of matter, plasma.

3.1.1 Plasma Science

A plasma is a gas or fluid in which the two charged atomic constituents—positive nuclei and negative electrons—are not bound together but able to move independently: the atoms have been ionized. (The term plasma was first used by Langmuir in 1928 to describe the ionized state found in a glow discharge.) Because of the strength and long range of the Coulomb interaction between such particles, plasmas exhibit motions of extraordinary force and complexity. Even in the most common “quasineutral” case, where the net charge density nearly vanishes, small, local charge imbalances and local electric currents lead to collective motions of the fluid, including a huge variety of electromagnetic waves, turbulent motions, and nonlinear coherent processes.

Plasma is the stuff of stars as well as interstellar space; it is the cosmic medium. Plasma also provides the earth’s local environment, in the form of the solar wind and the magnetosphere. It is in some sense the natural, untamed state of matter: only in such exceptional environments as the surface of a cool planet can other forms of matter dominate. Moreover, terrestrial plasmas are not hard to find. They occur in, among other places, lightning, fluorescent lights, a variety of laboratory experiments, and a growing array of industrial processes. Thus the glow discharge has become a mainstay of the electronic chip industry. The campaign for fusion power has produced a large number of devices that create, heat, and confine plasma—while bringing enormous gains in plasma understanding.

3.1.2 Conceptual Tools

The central intellectual challenge posed by plasma physics is to find a tractable description of a many-body system, involving long-range interactions, collective processes, and strong departures from equilibrium. This challenge has stimulated a remarkable series of scientific advances, including the concept of collisionless (Landau) damping, the discovery of solitons, and the enrichment of research in chaos. It is deep enough and difficult enough to remain a challenge of the highest level for many years—even with the huge increases in computational power that it has helped to stimulate. It has been addressed so far by a combination of several approaches.

The simplest route to insight is to track the motion of individual charged particles in prescribed magnetic and electric fields. In a non-uniform magnetic field, the orbit of a charged particle consists not only of the basic helical motion around a field line but also of the guiding-center drifts arising from gradients and curvature of the magnetic fields.

Kinetic theory, using an appropriate version of the Boltzmann collision operator, provides a more generally reliable, if not always tractable, approach. In the case of a stable (nonturbulent) plasma, the kinetic approach reduces to a plasma version of collisional transport theory and provides useful expressions for the particle and heat fluxes. If the plasma is magnetized, transport processes perpendicular to the magnetic field are accessible even when the collisional mean-free path is very long.

Fluid descriptions of plasma dynamics make sense in certain circumstances—particularly to describe macroscopic instabilities resulting in fast bulk plasma motion, comparable to the speed of Alfvén waves in the plasma. Sufficiently fast motions of a magnetized plasma are accurately characterized by a relatively simple fluid theory, which does not include dissipative effects, called magnetohydrodynamics or MHD.

Other, slower instabilities, typically of smaller spatial extent, require a more complicated description, essentially because their timescales are comparable to those of the guiding-center drifts, collisions, or other processes omitted by MHD. Drift waves, which can be destabilized by fluid gradients, are a characteristic example; a drift-wave instability driven by temperature gradients is believed to dominate transport in many tokamak experiments. Such phenomena are studied using kinetic theory or by means of fluid-kinetic hybrid descriptions.

3.1.3 Evolution of Fusion Science

The success that has been achieved using these approaches has accelerated in the past few decades. Increasingly, the combination of experiment, theory, and computation has led to scientific understanding with both explanatory and predictive power. Among the plasma phenomena that are now well understood are whistler waves emanating from the ionosphere, Alfvén wave propagation in a magnetized plasma, instabilities in novae, and macroscopic instabilities in magnetized plasmas.

The same broadly ranging fit is even seen in one of the most challenging areas of plasma behavior: turbulent transport. New simulations based strictly on Maxwell's and Newton's laws fit tokamak confinement results with at least qualitative accuracy, again over an impressively wide range of conditions. The effectiveness of velocity shear in controlling this turbulence is similarly reflected in the data. Thus progress has occurred in achieving a predictive understanding of plasma turbulence and its effects. A key to this success, and a theme of several recent confinement physics advances, is close interplay between analytic theory, computation, and experimental physics.

Such interplay is now appreciated as the key to progress in another central area of magnetic fusion physics: the interaction between confined plasma and the structure bounding it. The need to control particle and heat exchange at the boundary apparently requires the use of a divertor, bringing in a host of issues—radiation, atomic processes, transport, and supersonic flow—involving the relatively cool plasma that makes contact with the divertor plate. Divertor research has led to a better understanding of the physics of low-temperature plasmas (some divertor configurations have regions in which the plasma temperature is no more than a few electron volts); one result is enhanced contact with the physics of industrial plasma applications.

Indeed, wider and more fruitful contact with related disciplines increasingly characterizes magnetic fusion research. Some recently proposed confinement schemes, for example, are inspired by astrophysical plasma phenomena. Similarly, advances in understanding plasma turbulence owe much to research in such areas as organized criticality and hydrodynamics. Perhaps most important is the growing community appreciation that improved contact with other areas of physics and science is essential to the continued progress of magnetic confinement research.

3.2 Major Topical Areas in Plasma Science

3.2.1 Hamiltonian Dynamics

Hamiltonian dynamics is defined by the ordinary differential equations $dq/dt = \partial H(p,q,t)/\partial p$ and $dp/dt = -\partial H/\partial q$. The variables are the canonical momentum (p), coordinate (q) and time, and the Hamiltonian (H). Continuum Hamiltonian dynamics is an analogous set of equations but with p and q generalized from being variables to functions of position, momentum, and time. The magnetic fusion program has provided

many widely recognized developments, such as determining the threshold of chaotic dynamics, techniques for removing chaos, non-Hamiltonian dynamical methods, and electromagnetic ray-tracing. Some of the methods developed for toroidal plasmas are now used in astrophysics.

Discrete Hamiltonian dynamics has a number of areas of application in complex magnetic field geometries, including mapping the trajectories of magnetic field lines generated by a set of coils, calculating the trajectories of charged particles in a magnetic field, tracing the propagation of electromagnetic waves, and simulating plasma transport properties by means of Monte Carlo techniques. Rapid progress in the application of Hamiltonian dynamics is being made in each of these areas. Indeed, the applications of Hamiltonian dynamics in plasma physics research have become so pervasive that they define a way of thinking as much as the techniques of calculus define methodology in classroom physics.

3.2.2 Long Mean-Free Path Plasmas

In a plasma (or neutral gas) dominated by collisions, particle motion is randomized on a spatial scale short compared to the scale for change in the temperature or density. As a result, the plasma flows of mass and heat are linearly related to the local pressure and temperature gradients, and a tractable fluid description—exemplified by the equations of Spitzer, Chapman-Cowling, or Braginskii—is possible. Fluid equations have the key advantage of being in three-dimensional (3-D) coordinate space, rather than the six-dimensional phase space of the kinetic equation. On the other hand, for small collisionality or when gradients become very steep (as in the vicinity of a material surface), closure of fluid equations is not obviously possible, and it has been necessary to revert to kinetic theory. Often the goal of long mean-free path research, however, is to find a reduced description that allows some of the simplicity of the fluid description to be retained.

Long mean-free path physics affects many phenomena—such as plasma stability, laser-plasma interaction, and particle edge physics—in which the gradients can become steep. In fusion science it has three overriding goals: understanding the dynamics of the edge region of a confined plasma; characterizing heat transport due to high-energy electrons in laser-irradiated plasmas used in inertial fusion research; and efficiently describing stability, low-collisionality relaxation, and turbulence in the interior region of a magnetically confined plasma, where mean-free paths usually far exceed the parallel connection lengths. Long mean-free path research involves fundamental questions of particle motion and fluid closure that bear significantly on research in other areas, including rarefied gas dynamics, astrophysics, short-pulse laser physics, and space physics (the mean-free path in key regions of the earth’s magnetosphere is approximately the same as the distance from the Earth to Jupiter).

3.2.3 Turbulence

Plasmas provide a versatile platform for research on nearly all manifestations of turbulent phenomena. Probabilistic behaviors in space and time are a consequence of the nonlinearities in plasma dynamics. They produce unique turbulence-driven transport (see Fig. 3.1), anomalous mixing, and a host of other poorly understood important features in astrophysics, geophysics, and fusion energy systems. In turbulent plasmas, a full range of interesting temperatures and densities can be investigated. The investigations of turbulent plasmas can take advantage of plasma diagnostics with their exploitation of a sensitivity to electromagnetic effects as well as the standard diagnostics of gas and liquid flow. Thus, through research on turbulence in plasmas, opportunities are available for the discovery and evolution of new turbulence physics far beyond those found in ordinary fluid dynamics.

In fusion energy sciences, our understanding of turbulence and fluctuations has increased substantially in recent years. There is an improved understanding of the linear and “quasi-to-near nonlinear” RT instabilities in ablatively driven systems. There has been progress in gyrokinetics and gyrofluid modeling of electrostatic turbulence for a limited range of plasma conditions. Turbulence can provide plasma heating by energy transfer, and it can result in the self-organization of a plasma and magnetic field. Under special

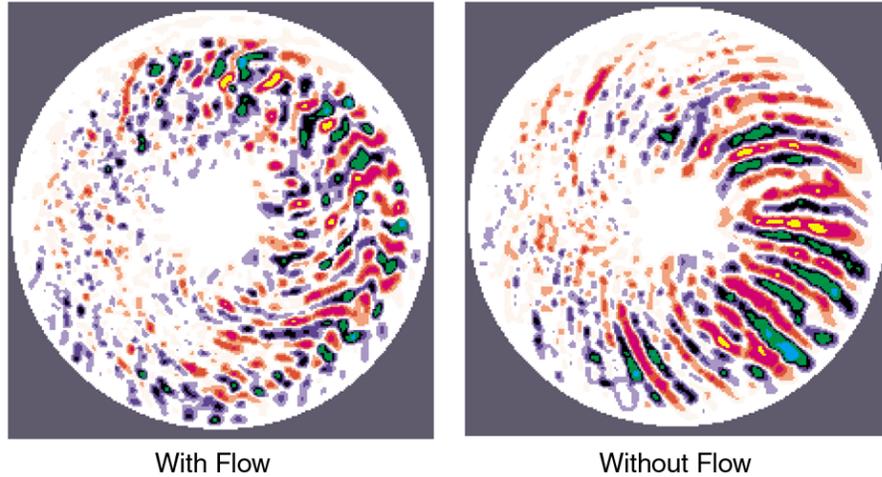


Fig. 3.1. Simulations showing turbulent-like eddies disrupted by strongly sheared plasma flow.

circumstances, turbulence can transfer energy/momentum from small scales to large scales and even lead to a new state with restored symmetry. Examples are the dynamo effect, shear flow amplification, and zonal flows.

Nonetheless, space and time resolution are still issues for further advances. The role of perturbations at plasma-surface interfaces, their sources, and the evolution of the consequences of these nonuniformities remain as challenges in laser-induced implosions. Present-day computers remain challenged by the needs of computational turbulent fluid dynamics for direct numerical simulations. Present-day plasma diagnostics provide unique opportunities to measure turbulence because of the unique electromagnetic properties of plasmas (e.g., internal reflection surfaces for O and X mode waves) but also face challenging experimental environments and strong spatial gradients of plasma and presumably turbulence properties.

3.2.4 Dynamo and Relaxation

The dynamo in a magnetically confined plasma is the process which leads to the generation and sustainment of large-scale magnetic fields, generally by turbulent fluid motions. Dynamo studies originated in astrophysics to explain the existence of some cosmic magnetic fields associated with planets, stars, and galaxies, where the existence of magnetic fields is inconsistent with simple predictions using a resistive Ohm's law. Dynamo processes are believed to play an important role in sustaining the magnetic configurations of some magnetic confinement devices. However, no experiment has yet convincingly demonstrated the existence of kinematic dynamos, with their spontaneous growth of magnetic fields from fluid motions, although a number are currently beginning operation.

A related topic is plasma relaxation, in which the plasma self-organizes into a preferred state. In decaying turbulence, described by 3-D MHD, energy decays relative to magnetic helicity to a static configuration in which the magnetic field and plasma current are aligned. In fusion science, Bryan Taylor predicted the preferred minimum energy state seen in a RFP device. A similar observation of a fluctuation-induced dynamo, seen in a spheromak, also leads to a sustainment of the configuration for longer than a resistive diffusion time. This concept of a "minimum energy state" has had a strong influence on solar, space, and astrophysical plasma research. Other successes include the development of two-fluid generalizations of MHD models and state-of-the-art computer simulations.

3.2.5 Magnetic Reconnection

In plasma systems in which a component of the magnetic field reverses direction, magnetic free energy is liberated by cross connecting the reversed-field components. The reversed magnetic field is effectively

annihilated, converting the released energy to heat and high speed flows. Magnetic reconnection provides the free energy for many phenomena, e.g., solar flares, magnetospheric substorms, and tokamak sawteeth.

Usually the release of energy during magnetic reconnection is nearly explosive, after a slow buildup of the magnetic energy in the system. Research tries to understand the sudden onset and accompanying fast release. The essential problem is that a change in magnetic topology usually requires a dissipative process to break the MHD constraint freezing magnetic flux to the plasma fluid. In the systems of interest, however, the plasma is essentially collisionless. The scientific challenge is therefore to understand how the frozen-in condition is broken in collisionless plasma in a way that will yield the fast rates of magnetic reconnection seen in the observations. Magnetic reconnection affects confinement of fusion plasmas in crucial ways; but it also matters in such areas as space physics, magnetospheric physics (dynamics of the magnetopause and the magnetotail), and in the solar atmosphere (flares and coronal mass ejections).

3.2.6 Wave Propagation

The dynamical evolution of a plasma is often governed by collective phenomena involving exchange of energy and momenta between the plasma constituents and various electromagnetic waves—wave-particle or wave-plasma interactions. Early studies of radio wave propagation in the ionosphere spurred the development of the theory of waves in plasmas. Today, complicated models involving mode conversion, power absorption, and generation of energetic particles are used in magnetospheric physics and astrophysics to describe such phenomena as solar coronal heating, interactions of the solar wind with the magnetosphere, and cyclotron emission observed in the Jovian system.

The application of electromagnetic waves for control of magnetically confined plasmas has been a major part of the fusion program since its inception. External means of plasma heating and noninductive current generation have evolved into tools for increased plasma performance through control and modification of plasma density, temperature, rotation, current, and pressure profiles. The localized nature of wave-particle interactions provides a pathway for the development of optimization and control techniques, allowing long-timescale maintenance of high confinement, stable operating regimes in toroidal magnetic confinement systems. The fundamental models describing these wave-particle interactions are common to all plasmas, both laboratory-based and those that occur naturally throughout the universe (see Fig. 3.2).

3.2.7 Nonneutral Plasmas

Nonneutral plasmas are many-body collections of charged particles in which there is not overall charge neutrality. Such systems are characterized by intense self-electric fields and, in high-current configurations, by intense self-magnetic fields. *Single-species* plasmas are an important class of nonneutral plasmas. Examples include pure ion or pure electron plasmas confined in traps, and charged particle beams in

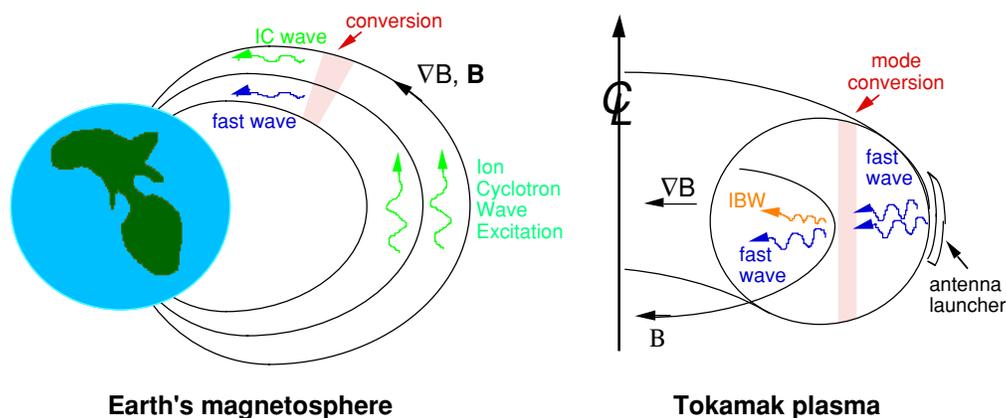


Fig. 3.2. Electromagnetic wave propagation and mode conversion is common to space and laboratory plasmas.

high-intensity accelerators and storage rings. They can be confined for hours or even days, so that controlled departures from thermal equilibrium can be readily studied. Such plasmas are an excellent test-bed for fundamental studies, such as transport induced by like-particle collisions, nonlinear dynamics and stochastic effects, vortex formation and merger, plasma turbulence, and phase transitions to liquid-like and crystalline states in strongly coupled pure ion plasmas.

Applications of single-species plasmas include accumulation and storage of antimatter in traps; development of a new generation of precision atomic clocks using laser-cooled pure ion plasmas; precision mass spectrometry of chemical species using ion cyclotron resonance; coherent electromagnetic wave generation in free electron devices (magnetrons, cyclotron masers, free electron lasers); high-intensity accelerators and storage rings, with applications including heavy ion fusion, spallation neutron sources, tritium production, and nuclear waste treatment; and advanced accelerator concepts for next-generation colliders, to mention a few examples.

3.2.8 Electrostatic Traps

Commercial Inertial Electrostatic Confinement (IEC) neutron generators from electrostatic traps use either a physical grid (Fig. 3.3) or a virtual cathode formed by primary electrons in a Penning-type trap to confine, accelerate, and focus ions toward a focus, usually in a spherical geometry. Ions are formed by glow-discharge operation, by electron impact on neutral fill, or in an external ion source. To produce fusion-relevant conditions, high voltages (>100 kV) and relatively small sizes (few millimeters to few centimeters) are required, making electrical breakdown a critical technology and science issue. The small size and relative simplicity of these systems make them useful as portable sources of D-D or D-T fusion neutrons. A unique fusion reactor concept uses a massively modular array of such sources operating at high Q to solve fusion engineering problems of high wall load, high activation, and tritium production. Traps using a physical grid (usually called IEC machines) have demonstrated useful neutron outputs to the point where assay system and even commercial applications are now underway. Daimler-Benz Aerospace has developed a commercial D-D unit which is virtually ready for market. Virtual cathode machines (usually called Penning Fusion or PF machines) have demonstrated required physics goals of maintaining required nonthermal electron distributions, spherical focussing, and excellent electron energy confinement, and they are poised to attempt to study ion physics.



Fig. 3.3. Commercial IEC neutron generator.

3.2.9 Atomic Physics

Atomic collision and radiation processes critically influence the dynamics of heating, cooling, confinement, particle transport, and stability of high-temperature core plasmas, as well as low-temperature edge and divertor plasmas of magnetically confined fusion devices. In the core plasma, electron collisions with multicharged impurity species determine ionization balance and excited state distributions. Spectroscopic measurement of these parameters provides information on plasma temperature and impurity density. Since power/particle exhaust and plasma diagnostics will be central issues of any reactor design, atomic physics processes have pervasive importance.

Good progress has been made in characterizing atomic-collision cross sections and atomic structure data pertinent to low-density high-temperature core fusion plasmas; this has been achieved by coordination between experiment and theory, while seeking to identify benchmark systems for testing critical theoretical approximations, and discovering useful scalings, and trends along isoelectronic sequences. Efforts are underway to compile comprehensive databases of low-energy elastic scattering, momentum transfer, and viscosity cross sections for interactions involving the various atoms and ions in fusion plasmas.

3.2.10 Opacity in ICE/IFE

The physics of atoms and ions in dense, high-temperature plasmas is interdisciplinary. Its first component consists of atomic structure theory up to very heavy and multiply ionized atoms, for which relativistic and QED effects must be included. Interactions between such ions and the rest of the plasma are important not only for the equations of state (EOSs) and dynamical properties of dense matter, but also for the radiative properties of ICF/IFE (and astrophysical) plasmas. Specific ICF applications are radiation-hydrodynamics of pellets and X-ray hohlraums, spectroscopic diagnostics, and z-pinch X-ray sources.

A central concern of radiation physics is the kinetic modeling of charge-state and level populations. For time-dependent and inhomogeneous plasmas in a non-Planckian radiation field, this modeling requires numerical solutions of large sets of collisional-radiative rate equations coupled with many photon-bins radiative transfer equations and with (magneto-) hydrodynamics equations. The task of atomic physics in this challenging computational physics problem is to provide realistic collisional rate coefficients (cross sections in the case of non-Maxwellian electron distributions), transition energies and probabilities, photon cross sections, and line profiles. For computational reasons, such large kinetic models are normally replaced by reduced models, that is, omission of detailed atomic structure and of highly excited states. At high densities the first simplification may be physically justified by line broadening, and the second by continuum lowering which is closely related to line broadening. An even more desirable replacement is possible if densities are high enough and effects of non-Planckian radiation fields are not too important, such that local thermodynamic equilibrium (LTE) is approached. For such situations, non-equilibrium thermodynamic, linear-response theory can be used to calculate even surprisingly large deviations from LTE with good accuracy.

3.2.11 Plasma Diagnostics

Plasma diagnostics are the instruments used to make measurements in a wide variety of plasma devices. They use electromagnetic radiation, magnetic fields, atomic and subatomic particles, and metallic probes, in both active and passive operation. Diagnostic data are normally integrated with analysis codes to provide the fundamental properties of the plasma. An increasing use of these data is to provide active feedback control of some plasma parameters, including their spatial profiles to improve performance and lifetime of the plasma. Improved theoretical modeling and simulation capability, together with the improved measurement capability, has led to a much better interaction between the experiments and theories.

There has been major progress in the capability of plasma diagnostics over the last few years, for MFE (particularly on tokamaks) and for IFE. New technological developments have allowed many observational sightlines (required because of the presence of steep gradients) and systems with fast time resolution necessary to fully understand turbulence. Such improvements are closely coupled with rapidly improving theoretical modeling. Most of the measurement capability on current tokamaks can be applied to alternate devices, and this application should be a major component of new developments.

3.2.12 Computer Modeling of Plasma Systems

Computational modeling of the plasma and auxiliary systems has been an important component of both the MFE and ICF programs since at least the early 1970s. These codes integrate the plasma physics with external sources and are used to interpret and predict macroscopic plasma behavior. Present MFE transport simulation codes couple MHD equilibria with fluid transport equations for particles, momentum, and energy. In ICF, simulations are usually performed with hydrodynamic codes that incorporate the processes relevant to the target design being investigated. In both MFE and ICF, the modules or algorithms use the best physics understanding that can be supported by available computer systems.

In MFE there are several fluid transport codes in use that differ in the component physics they emphasize and, therefore, in the types of applications they address. Interpretive codes make maximum use of experimental data to deduce confinement properties, while predictive codes make maximum use of models for

experimental validation and design of new experiments (see Figs. 3.4 and 2.20). The simulation codes presently in use for ICF are 1-D and 2-D hydrodynamic codes funded predominately by the DOE–DP. The existing 1-D and 2-D hydrodynamics codes used for ICF research have been heavily checked and validated against experimental data obtained on existing ICF facilities. Transport modeling codes have played a crucial role in interpreting and predicting plasma behavior, even when many aspects of the component models are empirical. The clear success of multidimensional simulation codes has served as a model for other programs to follow.

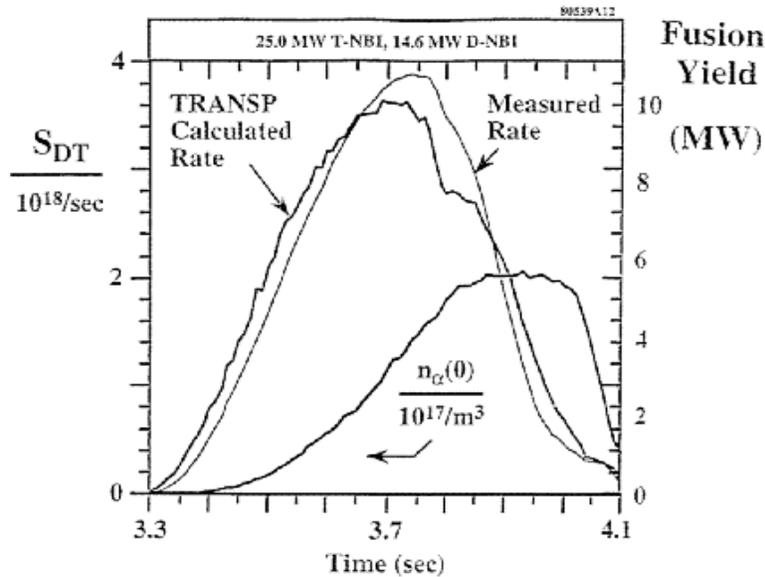


Fig. 3.4. Time evolution of the fusion power, the TRANSP calculation of the fusion power, and the TRANSP calculation of the central alpha density for TFTR plasmas with the highest fusion power.

3.2.13 Advanced Computation

The U.S. fusion community has a history of enthusiastic support for advanced computation and modeling capabilities which can be traced to the establishment of the predecessor to NERSC in the area of network-connected supercomputing, the Magnetic Fusion Energy Computer Center (MFECC) over 20 years ago. This support has been rewarded by impressive advances in simulation and modeling capabilities in the areas of large-scale macroscopic phenomena, fine-scale transport physics, the interaction of the plasma with its surroundings, and the dynamics of intense beams in heavy ion accelerators; e.g., in the turbulent transport area, the full power of the half-teraop SGI/Cray T3E at NERSC has been used to produce fully 3-D general geometry nonlinear particle simulations of turbulence suppression by sheared flows.

The restructured Fusion Energy Sciences program, with its focus on scientific foundations, requires greatly enhanced simulation and modeling capabilities to make optimum use of new national experiments and to leverage large-scale, international facilities. Effectively predicting the properties of these systems depends on the integration of many complex phenomena that cannot be deduced from empirical scaling and extrapolation alone. An enhanced modeling effort, benchmarked against experimental results, will foster rapid, cost-effective exploration and assessment of alternate approaches in both magnetic and inertial confinement and will be the catalyst for a rapid cycle of innovation and scientific understanding. Plasma science shares with other fields the challenge to develop realistic integrated models encompassing physical processes spanning many orders of magnitude in temporal and spatial scales. In general, a computing initiative in plasma science will require the tera- and peta-scale computational capabilities targeted by high-performance computing initiatives such as the new DOE Scientific Simulation Plan (SSP) and the established Accelerated Strategic Computing Initiative (ASCI).

3.2.14 Dense Matter

An accurate EOS for many materials at extreme conditions is vital to any credible ICF/IFE target design. Currently very few materials have their high pressure (greater than a few megabars) EOS experimentally validated and even then, only on the Principal Hugoniot. Compressing matter to extreme densities also provides a testing ground for planetary science and astrophysics, creates new avenues for producing super hard, superconducting or energetic materials, and may lead to new technologies. For planetary and stellar interiors, compression is gravitational and isentropic (ignoring phase separation). On Earth, high densities are achieved with either static compression techniques (i.e., diamond anvil cells) or dynamic compression techniques using large laser facilities, pulse power machines, gas guns, or explosives.

Dynamic compression experiments have made great strides in recreating material states that exist in the outer 25% (in radius) of the Jovian planets, near the core of earth, and at the exterior of low-mass stars. Large laser facilities have recently shown success at producing and characterizing material EOS at significantly higher pressures than gas guns. Both direct and indirect drive have been used to generate well-characterized shocks. Several experimental techniques have also recently been developed. Radiography (to determine opacity), optical conductivity, temperature, displacement, X-ray diffraction, and velocity/displacement sensitive interferometry are some of the diagnostics currently used in laser-generated shock EOS experiments.

3.2.15 High Energy Density Laboratory Astrophysics

Astrophysical models are routinely tested against observational results, rather than against experiments with controlled initial conditions. Creating a surrogate astrophysical environment in the laboratory has heretofore been impossible because of the high energy density required for some topics of investigation. Fusion facilities offer the capability to perform controlled experiments in a realm of plasma temperatures and densities approaching astrophysical regimes in several important parameters. Furthermore the physics of the problems studied may be scaled over many orders of magnitude in spatial scale. Examples include strong shocks in ionized media; high Mach number supersonic jets; material flow in strongly coupled, Fermi degenerate matter; hydrodynamic instabilities in hot, compressible matter with low viscosity; radiation transport dominated by X rays; photoevaporation front, coupled radiative hydrodynamics; and material properties such as EOSs at high pressure. Experiments have been performed on the Nova (U.S., LLNL) and Gekko (Japan, ILE) lasers and reported extensively in the literature. Current experiments are also underway on the Omega (U.S., LLE) laser. Experiments on the Nova, Omega, and Gekko lasers have concentrated on hydrodynamics studies (of turbulence, mixing, material flow, and supersonic jets), material properties at high pressure, and opacity of ionized elements.

3.3 Major Topical Areas in Engineering Science

3.3.1 Bulk Materials Science

There are numerous examples where fusion materials science research has had a positive impact on the broader engineering/materials science fields. For example, a bainitic (Fe-3Cr-W-Ta) steel with superior toughness and strength to existing steels was developed by fusion researchers which has potential applications in numerous commercial systems (e.g., fossil energy). Fusion research has led to improved interphases in SiC/SiC composites which have higher oxidation resistance. This has potential applications in chemical processing systems, as well as defense and aerospace systems. Fundamental research on ceramics by fusion researchers has led to the first known experimental measurements of point defect (interstitial) migration energies in SiC, alumina, and spinel.

Experimental and theoretical analysis of neutron-irradiated metals is providing an improved understanding of the fundamentals of mechanical deformation, which has far-reaching impact on numerous engineering disciplines. For example, it appears possible to obtain the constitutive equations for twinning (which is one of the six possible deformation mechanisms in solids) from an analysis of neutron-irradiated metals. Most of the present-day understanding of fracture mechanics (essential for all advanced

engineering structural applications) is derived from early studies on neutron-irradiated metals. Further fundamental research is needed on the physical mechanisms of flow and fracture of deformed metals and can be readily provided from appropriate analyses of neutron-irradiated materials which are being studied in fusion research. Significant advances in the science of mechanical deformation of refractory metals are being provided by fusion research on vanadium alloys and other refractory metals.

3.3.2 Surface Materials Science and Atomic Physics

Improved control of edge density and impurity influx have been important to the improvement in confinement device performance in the last 15 years. Improved control has been achieved from better understanding of the complex interactions between plasma physics, surface science, and solid state physics that occur at the plasma-to-material interface, through a combination of experiments, and modeling.

The fraction of the plasma large particle flux that is trapped in the surface or bulk material has a strong influence on the density of the edge plasma since recycled particles fuel the plasma. Extensive surface science experiments are being conducted to understand the mechanisms, such as sputtering and evaporation chemical erosion, controlling the release of particles from surfaces. Techniques for cleaning surfaces and coatings that can reduce gas release are being investigated. A combination of chemistry and solid state physics is needed to understand the transport of plasma particles in plasma facing materials. Several techniques have been developed for controlling either the amount of erosion or the transport of the eroded particles back to the main plasma. Further, extensive laboratory measurements of the energy and angle dependence of sputtering process have led to fundamental understanding of the phenomena.

Eroded material may be transported through the edge plasma to the core plasma and cause a reduction of reactivity. Electron or ion impact ionization of the eroded atoms will cause the atoms to follow field lines to a nearby surface. A combination of plasma physics and atomic physics (some lacking) is needed to understand these effects, and there have been extensive laboratory and fusion device studies of these phenomena. Modeling of these effects requires a combination of fluid transport, plasma physics, gas transport and atomic physics. An integrated material surface, plasma edge and plasma core model is the next step and will require science input from many disciplines.

3.3.3 Heat Transfer at Liquid/Vacuum Interfaces

The temperature at the free liquid surface facing the plasma in a liquid wall system is the critical parameter governing the amount of liquid that evaporates into the plasma chamber. The heat transfer at the free surface of a non-conducting liquid wall is dominated by phenomena of rapid surface renewal by turbulent eddy structures generated either near the free surface due to temperature-gradient-driven viscosity variations, or near the back wall or nozzle surfaces by frictional shear stresses. The intensity of these turbulent structures and their effectiveness in cycling energy from the free surface into the bulk flow of the liquid wall will depend heavily on the velocity of the main flow, the stability of the free surface, the distance from back wall and nozzle surfaces, the degree of damping by the magnetic field, and even the magnitude and distribution of the surface heat flux itself. This is a challenging interdisciplinary scientific problem, with relevance to fields such as oceanography, meteorology, metallurgy, and other high heat flux applications like rocket engines.

The picture is different for a liquid metal, which may be fully laminarized by the magnetic field, but is still likely to be highly wavy or possess 2-D turbulence-like structures with vorticity oriented along the field lines. Surface waves and 2-D turbulence increase the area for heat transfer and have motion that helps to convect heat into the bulk flow. Understanding the relative importance of these terms to the dominant conduction and radiation transport effects and judging the effectiveness of using turbulence promoters such as coarse screens are required to assess the feasibility of liquid metal walls from the heat transfer point of view. The complicated hydrodynamics are now heavily coupled to the applied magnetic fields and the motion of the plasma through Ohm's law and Maxwell's equations. The solution to these systems is of similar complexity to the MHD fluid motions in the plasma.

3.3.4 Ablation, Radiation Gas Dynamics, and Condensation

Determination of the inertial fusion chamber environment following a target explosion is another example of complicated, interdisciplinary scientific exploration. The X rays, neutrons, and debris emanating from the exploded target must be absorbed by the chamber, and a reasonably quiescent condition must be reestablished before the next shot can take place. The phenomena that must be understood include photon transport in gases and condensed matter, time-dependent neutron transport, ionized gas dynamics and radiation hydrodynamics, ablation and thermo-physics of rapidly heated surfaces, dynamics of large-scale free liquid flows, and the condensation heat and mass transfer. Simulation tools have been developed or adapted to model these different processes, and work proceeds toward integration into a code that can simulate all relevant physics of the chamber.

3.3.5 Neutron and Photon Transport in Materials

Understanding the physics of neutron and photon interactions with matter is fundamental for many applications of nuclear science. The high-energy neutrons (~14 MeV) emerging from the D-T reaction intercept and penetrate the chamber wall, resulting in several reactions. Photons generated from neutron interactions as well as X rays in inertial fusion undergo various types of interactions with materials. Monte Carlo and deterministic methods are used to determine the fluxes of neutrons and gamma-ray whose accuracy depends on the numerical approximations involved in the underlying transport equation and the adequacy of the nuclear data. Neutron and photon cross section data rely heavily on nuclear science. Models for two-, three-, and N-body reactions are still developing to evaluate accurate representation of the energy and angular distribution of the emerging reaction products. Representation of this information in useable files with format and procedures that are easy to process is still an active area in nuclear data development.

3.3.6 Pebble Bed Thermomechanics

Thermomechanics of materials has been identified as one of the critical issues for solid breeder blanket designs, particularly for materials in the form of pebble beds. Fundamental thermal physical property data have to be quantified accurately, and changes of the packed states through pebble and bed/clad interactions during operation need to be understood because of their dominating effects on performance.

The thermomechanical behavior of a particulate bed made of contacting solid particles material is a complex phenomenon. The existence of the contacts restricts the freedom of motion of the individual particles and, thus, conditions the strength and the rigidity of the bed. This depends on the number and strength of the contact bonds which are themselves a consequence of the size, shape, and roughness of the particles, of the nature of the thermal and/or mechanical interaction between various phases, and of the state of the particle material in question. Such research leads to fundamental thermal-physical-mechanical property data and to advancing the engineering science knowledge base necessary for understanding and extending the thermomechanical performance of particulate bed material systems.

Page Intentionally Blank

4. NEAR-TERM APPLICATIONS

4.1 Introduction

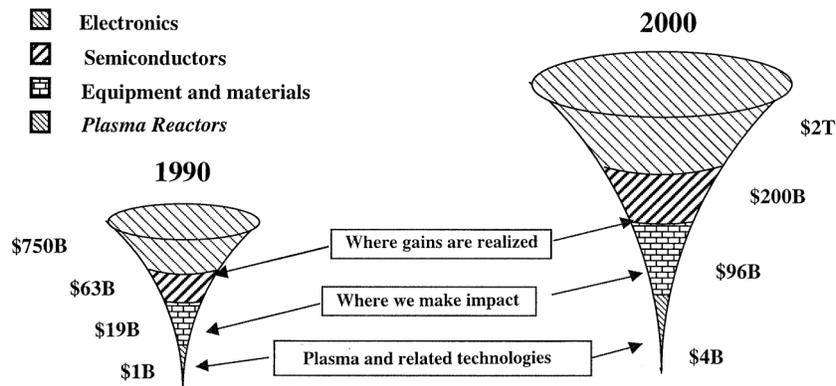
The practical applications of plasmas and associated technologies are of growing importance to the government and national economy.^{*,†,‡} The technologies, theoretical models, and computational tools developed in the fusion program are being used in a variety of market segments including electronics, manufacturing, health care, environmental protection, aerospace, and textiles. This programmatic mixture has led to an effective transfer out of and into the fusion program. There are several high-impact opportunities for applying the plasma expertise developed within the fusion research program to near-term industrial and government needs. In most applications, an interdisciplinary approach is required, where plasma science must be integrated with chemistry, atomic physics, surface and materials science, thermodynamics, mechanical engineering, and economics. OFES and the NSF are major government sponsors of plasma R&D. The NSF has funded near-term application programs. In addition to sponsoring many single-investigator led projects, the NSF has supported three engineering research centers: (1) Advanced Electronic Materials Processing at North Carolina State University; (2) Plasma Aided Manufacturing at the University of Wisconsin; and (3) Environmentally Benign Semiconductor Manufacturing at the University of Arizona.

4.2 Opportunities

4.2.1 Microelectronics and Flat Panel Displays

To date, the highest-impact application opportunity for plasma science is the \$1T microelectronics industry. Plasma technologies are ubiquitous in semiconductor manufacturing (Fig. 4.1). The capital equipment is replaced frequently, consistent with the 18-month performance-doubling period of Moore's Law. At present, plasma technologies are used in 25%–30% of the steps required to process a wafer from bare silicon to a finished integrated circuit. This fraction is projected to increase over the next decade.

The semiconductor tooling market is about \$100B. Plasmas are effective in etching, cleaning, and deposition. Most technologies have been developed by able chemical engineers employed by the private sector outside the fusion program; however, they have benefited from fusion technologies such as rf and



**Plasma Science: From Fundamental Research to Technological Applications*, National Research Council, Washington, D.C., 1995.

†*Plasma Processing of Materials: Scientific Opportunities and Technology Challenges*, National Research Council, Washington, D.C., 1991.

‡S. O. Dean, "Applications of Plasma and Fusion Research," *J. Fusion Energy*, **14** (2), 251–279 (June 1995).

Fig. 4.1. The global electronic food chain (courtesy of R. A. Gottscho, LAM Res. Corp., 1998).

microwaves, plasma and wafer process control diagnostics, beam and laser sources, theoretical models and algorithms. Two companies, each with annual sales exceeding \$100M, have spun out of the fusion program; see Fig. 4.2. Furthermore, there have been many joint cooperative R&D agreements (CRADAs) between the private sector and OFES centers. CRADAs involve activities ranging from fundamental understanding of rf field penetration into the plasma (to optimize throughput and yield), to rf systems, to advanced high-heat flux materials for heat sinks, to an entirely new class of extreme ultraviolet lithography source.

There is a need to pattern 100-nm feature sizes, that places new requirements on lithography, extending beyond the limitations of present day light sources. Plasma sources can meet these requirements. A privately funded, 3-year, \$250M consortium between major manufacturers and three DOE laboratories is underway to develop laser-produced and discharge plasmas to meet this requirement. A complementary DARPA program is examining the efficacy of electron beam lithography.



Fig. 4.2. Plasma source for semiconductor processing from a fusion spin-off company.

New requirements for ion implantation represent another opportunity. Advanced VLSI semiconductor devices will utilize ultra-shallow junctions, often less than 100 nm, produced with low-energy implantation (1 to 10 keV) at high throughput and at low cost. Conventional beamline implanters produce relatively low currents at these energies and may not meet the throughput requirement. Plasma immersion methods are an attractive alternative to beamline processing in semiconductor applications that require a high dose over a large one. The processing timescale is independent of implant area, and the relatively simple plasma immersion equipment can be readily incorporated into the cluster tools currently employed in semiconductor fabrication.

Given the complex interdisciplinary nature of plasma processing, there is a strong need for theoretical modeling and simulation.* Supercomputer Monte Carlo codes containing comprehensive ionization and collision databases, originally developed to study transport and heat flow in tokamak divertors, are being adapted to do similar simulations of plasma tools.

Flat panel displays (FPDs) are enabling a wide range of applications of information technology. World-wide demand for FPDs is projected to approach \$40B in 2000. Applications include computer and vehicle displays, personal digital assistants, video telephones, medical systems, and high-definition, full-motion video. Plasmas are needed to perform etching, cleaning, deposition, and implantation over large surface areas in thin film transistor FPDs. Bright large-scale flat-panel displays are also produced using plasmas to illuminate the pixels themselves.

4.2.2 Materials and Manufacturing

The last decade has witnessed a remarkable growth in the application of plasmas to industrial processing and manufacturing in non-semiconductor markets. Applications include hard coatings for wear and corrosion treatment of tools and components and thin film deposition for optical devices. Processes include plasma spraying, nitriding, polymerization and cross-linking, plasma-enhanced chemical vapor deposition (PECVD), physical vapor deposition (PVD) with magnetron sputtering sources and metal vapor vacuum arc (MEVVA), and ion implantation.

**Database Needs for Modeling and Simulation of Plasma Processing*, National Research Council, Washington, D.C., 1996.

There is a need to refine and improve existing surface engineering techniques and to develop new techniques to serve an explosive growth of applications such as nanoscale devices, high-performance materials for aerospace, medical, traditional large-scale heavy manufacturing, and emerging high-technology manufacturing. The market for surface engineering techniques has been growing rapidly for the last three decades. It has been estimated* that more than \$40B has been collectively invested in surface engineering R&D by North America, Japan, and Western Europe,

Plasma spray technology is a relatively mature technology that is beginning to benefit from fusion science and technology. Spray technology used to generate high-heat flux materials for PFCs has been spun off for advanced coatings for automotive components and manufacturing.

Plasma nitriding, also a relatively mature technology, has its roots in the gas nitriding processes developed by the chemical engineering community. Although up to now there has not been a great deal of interaction between the plasma nitriding and fusion community, industrial nitriders are interested in joint R&D with the fusion community, particularly in the area of PECVD. PECVD enables high deposition rates at reduced processing temperatures. Experimental and modeling techniques developed in fusion science programs are having a large impact on the deposition of diamond and diamondlike carbon (DLC) coatings on manufactured and machine tool components, biomaterials, sensors, heat sinks, X-ray windows, and many other areas.

Plasma Immersion Ion Implantation (PIII) is a nonline-of-sight technique for industrial surface engineering that was developed as a direct outgrowth of fusion technology research. First developed in 1986, this process has spawned more than 55 groups to date worldwide. A related spin-off of the fusion program is the MEVVA technology, which enables high-throughput ion implantation of targets with ion species that include most of the periodic table elements.

4.2.3 Environmental Applications

One of the most pressing concerns of our times is safeguarding the quality of our environment for present and future generations. Past practices have left a legacy of accumulated hazardous waste and pollution that must be remedied. In addition, a critical challenge exists to prevent or reduce generation of waste and pollution. Some examples include the cleanup of DOE nuclear weapons production and EPA superfund sites, reduction of landfills and water pollution from industrial and municipal sources, and reduction or elimination of harmful emissions into the atmosphere.

Plasma science can make a significant contribution to environmental needs. A physics perspective is needed in cleanup efforts, which are currently dominated by chemical engineers. The ultimate development of fusion energy will reduce or eliminate waste streams and pollution currently associated with fossil fuel and nuclear fission power plants. However, in the interim much of the knowledge gained in plasmas and the associated technologies developed to generate, control, and monitor plasmas can have a significant beneficial impact on our environmental needs.

Current R&D opportunities include (1) mixed radioactive waste remediation; (2) faster throughput, reduced emissions, and lower cost waste processing; (3) destruction of hazardous air pollutants (HAPs) such as volatile organic compounds (VOCs); (4) nondestructive decontamination of surfaces; (5) cleaning fine particulate emissions and other HAPs from current thermal processes in industry, power production, and burning of wastes; (6) elimination of SO_x and NO_x emissions from vehicle and stationary sources; (7) sensitive and accurate continuous emission monitors of pollution (e.g., metals, dioxins, furans); and (8) reduction and elimination of CO₂ and other greenhouse gas emissions and research of possible CO₂ sequestering technologies.

**Chem. Eng.*, p. 35 (April 1994).

There are numerous future opportunities. Over the near term, incremental improvements will be made in arc processes, plasma devices, plasma-aided monitoring technologies, and application of advanced diagnostics to environmental processes. Conventional processes will benefit from improved monitoring and control technologies. Over the longer term, advanced applications of plasmas to waste remediation will be developed. Plasmas will be used more to destroy hazardous materials rather than just as a source of heat as in near-term arc processes. New atmospheric plasma generation technologies (Fig. 4.3) will be developed with high throughput and efficient operation. Portable units and in-situ vitrification technologies will be commercialized. More universal plasma waste processing capability will be achieved.

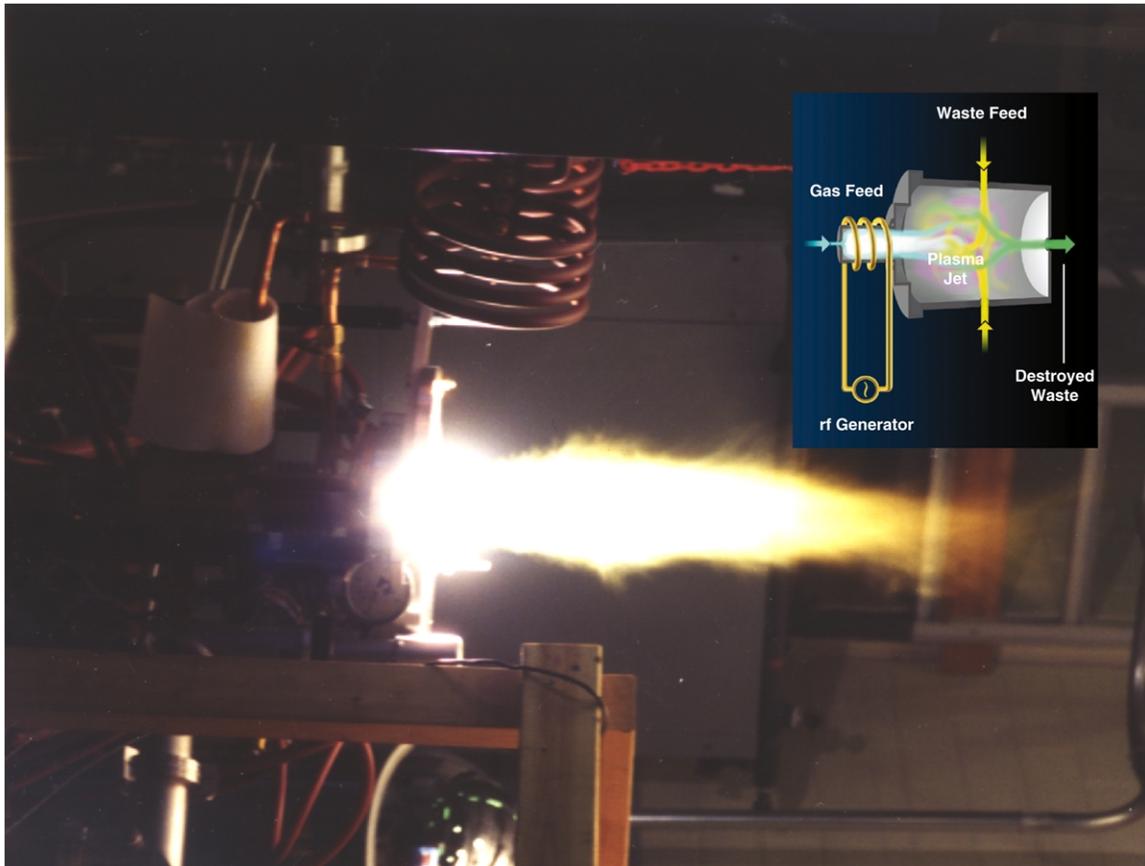


Fig. 4.3. Inductively coupled atmospheric plasma torch for destroying chemical waste.

4.2.4 Biomedical and Food-Safety Applications

Diverse medical diagnosis and treatment applications can trace their origins to magnetic and inertial fusion research. For example, recent advances in magnetic resonance imaging (MRI) magnet technology have taken advantage of superconducting coils developed by the fusion program. A joint venture has been established to supply MRI magnets with present annual sales of the order \$100M. Entirely new medical imaging systems have also been invented. For example, a micro-impulse radar (MIR) diagnostic has been developed enabling one to rapidly and noninvasively detect trauma, strokes, and hematoma. These portable microwave devices transmit electromagnetic pulses that are recorded in time by fast pulse technology developed for the laser fusion. MIR is a complementary alternative to traditional MRI and CT scans in that it allows low-cost prescreening in emergency rooms and ambulances. The anticipated instrument market for MIR has been projected to be about \$200M per year.

Medical treatment methods that have emerged from the fusion program include laser surgery and tissue welding and their associated computerized controls. Ultra-short (~10-ps) lasers developed in the fusion

program offer a breakthrough by enabling precision cuts without damaging surrounding issue. This has made make a difference in dental, spinal, and neurosurgery.

Laser treatment of stroke victims is another area of commercialization. Each year approximately 700,000 strokes occur in the United States, accounting for over \$26B/year for treatment and rehabilitation. A minimally invasive technique called endovascular photo-acoustic recanalization has been developed. Laser light is coupled through an optical fiber and delivered to an occlusion, causing a mechanical disruption of the occlusion and reestablishing blood flow. Cerebral arteries as small as 3 mm in diameter can be treated.

Finally, plasmas are used for sterilization of non-metallic surgical instruments, and techniques are being developed to allow the sterilization of complex, re-entrant structures used for examination of body cavities, by generating plasmas within the small spaces inside of these structures. These same techniques can be used for sterilization of recyclable containers for food. Technologies for generation of radio waves in plasmas are used for cold pasteurization of liquids.

4.2.5 Plasma Propulsion

Plasma-based propulsion systems for spacecraft are receiving increased and considerable interest. Thrust is generated by using electrical energy to accelerate a propellant. The accelerated species is generally ions, with plasma neutralization subsequent to acceleration. Plasma propulsion technologies include arc jets, ion thrusters, Hall thrusters, magneto-plasma-dynamic thrusters, and rf-driven plasma thrusters, with concepts incorporating the possibilities for either pulsed or continuous operation. Plasma propulsion can provide higher specific impulse (thrust/mass flow) than conventional chemical propulsion because of the high speeds attainable by plasma. Applications include “station keeping” for geosynchronous earth orbit (GEO), drag compensation for low earth orbit (LEO), and high-specific-impulse thrust for interplanetary and deep space missions. High specific impulse means that a spacecraft can perform a mission with less propellant mass than with conventional chemical propulsion. The number of satellites is rapidly growing, particularly for LEO, which is driven by the worldwide needs for communication.

There is strong overlap between the fusion and propulsion research such as collisionless plasma flow in crossed electric and magnetic fields. These systems also use many of the same technologies that are required for fusion, with many common diagnostic techniques.

Several plasma-based systems are in use or planned for future missions. The Russian space program has used Hall thrusters for more than 25 years. Hall thrusters are now in limited use on U.S. and European satellites, but it is anticipated that the planned 288-satellite Teledesic commercial satellite system will employ Hall thrusters. In 1998, a Russian-made electron propulsion demonstration module Hall thruster was used to boost the orbit of the U.S. STEX satellite. An ion engine is in use on the Deep Space 1 probe (Fig. 4.4) launched in 1998. A novel rf-driven plasma propulsion system, VASIMR, based on a magnetic mirror configuration, has been proposed for use in human exploration of the solar system.

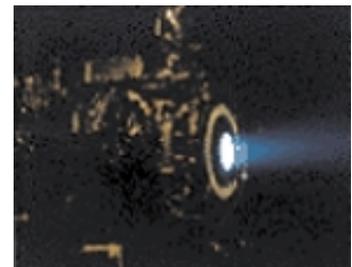


Fig. 4.4. Ion engine for Deep Space 1.

Current opportunities range from basic to applied R&D. Detailed diagnostic measurements and model development for today’s Hall thrusters can lead to optimized designs for immediate applications. Theoretical and experimental work can provide important groundwork for new concepts such as VASIMR. Technology advances are required to provide compact rf power supplies and reduced-weight high-temperature superconducting magnets.

NASA has supported electric propulsion for space applications, but research has been highly performance and mission driven. Compared with OFES programs, there is less emphasis on diagnostics and basic science, which are the building blocks for understanding and innovation.

Page Intentionally Blank

LIST OF ACRONYMS

ARIES	Advanced Reactor Innovation and Evaluation Studies	IFMIF	International Fusion Materials Irradiation Facility
ASCI	Advanced Strategic Computing Initiative	IRE	Integrated Research Experiment
AT	Advanced Tokamak	ITER	International Thermonuclear Experimental Reactor
BPX	Burning Plasma Experiment	JET	Joint European Torus
CDX-U	Current Drive Experiment–Upgrade	KSTAR	tokamak in Korea
CIT	Compact Ignition Tokamak		
COE	cost of electricity		
CRADA	Cooperative Research and Development Agreement		
CT	compact toroid		
CTX	compact toroid experiment		
CW	continuous wave		
D-D	deuterium-deuterium		
DEMO	demonstration reactor		
DIII-D	Doublet III-D tokamak experiment at General Atomics		
DP	Office of Defense Programs (DOE)		
DPSSL	diode-pumped solid-state laser		
D-T	deuterium-tritium		
ECRH	electron cyclotron resonance heating		
ELM	edge-localized mode		
EOS	equation of state		
ET	electric tokamak		
ETF	engineering test facility		
EUV	extreme ultraviolet		
FIRE	Fusion Ignition Research Experiment		
Flibe	fluorine-lithium-beryllium molten salts (Li ₂ BeF ₄)		
FM	frequency modulated		
FRC	field-reversed configuration		
GA	General Atomics		
GDT	Gas Dynamic Trap		
HIF	heavy ion fusion		
HIT	Helicity Injected Torus		
HSX	Helically Symmetric Experiment		
ICF	inertial confinement fusion		
ICH	ion cyclotron heating		
IFE	inertial fusion energy		

LDX	Levitated Dipole Experiment	W7-AS	German Wendelstein 7-AS stellarator experiment
L-H	low-to-high confinement transition in a tokamak	W7-X	Wendelstein 7-X
LHCD	lower hybrid current drive	WFO	Work for Others
LHD	Large Helical Device		
LMJ	laser megajoule		
LSX	Large S Experiment		
MAST	Mega-Amp Spherical Tokamak		
MFE	magnetic fusion energy		
MHD	magnetohydrodynamic		
MST	Madison Symmetric Torus		
MTF	magnetized target fusion		
NBI	neutral beam injection		
NCSX	National Compact Stellarator Experiment		
NIF	National Ignition Facility		
NSTX	National Spherical Torus Experiment		
OFES	Office of Fusion Energy Sciences (DOE)		
PBFA	Plasma Beam Facility-A		
PCAST	President's Committee of Advisors on Science and Technology		
PIC	particle-in-cell		
PoP	proof of principle		
RC	reduced cost (ITER)		
rf	radio frequency		
RFP	reversed-field-pinch		
RFX	(facility in Italy)		
RM	Richtmyer-Meshkov		
RS	reversed shear		
RT	Rayleigh-Taylor		
SSPX	Sustained Spheromak Physics Experiment		
SSTR	steady-state tokamak reactor		
SSX	Swarthmore Spheromak Experiment		
ST	spherical tokamak		
START	Small Tight Aspect Ratio Tokamak		
TEXTOR	Tokamak Experiment for Technology Oriented Research		
TFTR	Tokamak Fusion Test Reactor		
TPE-RX	(facility in Japan)		
TRAP	tokamak refueling by accelerated plasmoids		
TSTA	Tritium Systems Test Assembly		
VNS	volumetric neutron source		

