ARGONNE NATIONAL LABORATORY
9700 South Cass Avenue, Argonne, Illinois 60439

ANL/FPP-87-1

ANL/FPP--87-1

DE87 004635

TECHNICAL PLANNING ACTIVITY
FINAL REPORT

Fusion Power Program
Argonne National Laboratory

January 1987

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

prepared for

U.S. DEPARTMENT OF ENERGY
Office of Fusion Energy

MASTER

FOREWORD

In April 1985, the U.S. Department of Energy's (DOE's) Office of Fusion Energy commissioned the Technical Planning Activity (TPA). The purpose of this activity was to develop a technical planning methodology and prepare technical plans in support of the strategic and policy framework of the Magnetic Fusion Program Plan issued by DOE in February 1985.

The planning methodology and comprehensive technical plans described in this TPA report represent a significant accomplishment. This work has been carried out with the broad participation of the magnetic fusion community through the TPA Steering Committee, composed of more than 30 senior technical leaders and experts. In addition, DOE's Magnetic Fusion Advisory Committee has reviewed the progress of the TPA's work.

Although this report represents the views of only the U.S. magnetic fusion community, it is international in scope in the sense that the technical plans contained herein describe the full scope of the tasks that are prerequisites for the commercialization of fusion energy.

The TPA has developed a well-structured methodology that includes detailed definitions of technical issues, definitions of program areas and elements, statements of research and development objectives, identification of key decision points and milestones, and descriptions of facility requirements. Because of the uncertainties inherent in energy development, it is my intention that this be an ongoing, flexible process with continuous updating and modification.

I believe the 18-month effort of the TPA represented in this report has made very important contributions to the future directions of the U.S. magnetic fusion program and to future U.S. participation in international fusion activities.

John F. Clarke
Associate Director
for Fusion Energy
Office of Energy Research
U.S. Department of Energy

CONTENTS

ACKNOWLEDGMENTS	xviii
PREFACE: NATURE OF THE PLANNING ACTIVITY	1
SYNOPSIS: TECHNICAL PLANNING ACTIVITY OVERVIEW	9
S.1 Objectives for Commercial Fusion Applications	9
S.2 Interim Technical Objectives	12
S.2.1 Confinement Systems	13
S.2.2 Burning Plasmas	14
S.2.3 Plasma Technology	16
S.2.4 Nuclear Technology	16
S.2.5 Fusion Materials	17
S.2.6 Systems Design and Analysis	18
S.2.7 Interactions among Program Areas	19
S.3 Overall Program Logic	20
S.4 Major Decision Points	22
S.5 Program Scenarios	24
S.6 Summary of Resource Requirements	28
S.7 Comments on the TPA Results and Recommendations on Future Work	29
S.7.1 Foreign and International Efforts	30
S.7.2 Describing Concept Development	30
S.7.3 Comparisons among Concepts	31
S.7.4 Development of Scenarios	32
S.7.5 Achieving System and Component Reliability	33
S.7.6 Planning for Technology Transfer	33
S.7.7 Recommendations for Future Work	33
1 CONFINEMENT SYSTEMS	49
1.1 Concepts with a Moderately Developed Knowledge Base	58
1.1.1 Tokamaks	60
1.1.2 Tandem Mirror	83
1.1.3 Stellarators	100
1.1.4 Reversed-Field Pinches	108
1.2 Concepts with a Developing Knowledge Base	120
1.2.1 Spheromaks	120
1.2.2 Field-Reversed Configurations	127
1.2.3 Elmo Bumpy Square	134
1.2.4 Dense Z-Pinch	137
2 PHYSICS OF BURNING PLASMAS	143
2.1 Burning Plasma Area for Tokamaks	147
2.2 Burning Plasma Area for Non-Tokamak Concepts	154

CONTENTS (Cont'd)

3	PLASMA TECHNOLOGY	157
3.1	Magnets	160
3.1.1	Technical Issues, Objectives, and Alternatives	160
3.1.2	Program Logic	167
3.2	Heating and Current Drive	168
3.2.1	Technical Issues, Objectives, and Attributes	173
3.2.2	Program Logic	181
3.3	Fueling	186
3.3.1	Technical Issues, Objectives, and Attributes	187
3.3.2	Program Logic	189
3.4	Plasma-Interactive Components	191
3.4.1	Technical Issues, Objectives, and Attributes	191
3.4.2	Program Logic	198
4	FUSION NUCLEAR TECHNOLOGY	297
4.1	Blanket/First Wall Program Elements	219
4.1.1	Technical Issues, Objectives, and Attributes	220
4.1.2	Program Logic and Evaluation/Decision Points, Major Facilities, and Resource Estimates	226
4.2	Radiation Shield	236
4.2.1	Technical Issues, Objectives, and Attributes	237
4.2.2	Program Logic, Decision Points, Facilities, and Resource Estimates	242
4.3	Tritium Processing	245
4.3.1	Technical Issues, Objectives, and Attributes	245
4.3.2	Program Logic, Decision Points, Facilities, and Resource Estimates	248
4.4	Nuclear Elements of Plasma-Interactive Components	253
4.4.1	Technical Issues, Objectives, and Attributes	253
4.4.2	Program Logic, Decision Points, Major Facilities, and Resource Estimates	255
5	FUSION MATERIALS	259
5.1	Structural Materials	276
5.1.1	Technical Issues, Objectives, and Attributes	279
5.1.2	Program Logic	284
5.2	Nonstructural Blanket Materials	290
5.2.1	Technical Issues, Objectives, and Attributes	291
5.2.2	Program Logic	295
5.3	Special Materials	298
5.3.1	Technical Issues, Objectives, and Attributes	299
5.3.2	Program Logic	300

CONTENTS (Cont'd)

6	SYSTEMS DESIGN AND ANALYSIS	303
6.1	Applications, Economics, and Technology Transfer	307
6.3.1	Issues, Objectives, and Attributes	308
6.3.2	Program Logic	308
6.2	Fusion Test Facilities, Critical Issues, and Development Pathways	311
6.2.1	Issues, Objectives, and Attributes	312
6.2.2	Program Logic	313
6.3	Safety, Environment, and Licensing	315
6.3.1	Issues, Objectives, and Attributes	316
6.3.2	Program Logic	317
6.4	Remote Technology	317
6.4.1	Issues, Objectives, and Attributes	323
6.4.2	Program Logic	324
6.5	Alternative Fuel Cycles	326
6.5.1	Issues, Objectives, and Attributes	326
6.5.2	Program Logic	327
7	COMMERCIAL OBJECTIVES AND TECHNOLOGY TRANSFER	333
7.1	Commercial Objectives	333
7.1.1	Electricity Production	333
7.1.2	Fissile-Fuel Production	334
7.1.3	Synthetic-Fuel Production	338
7.1.4	Other Applications and Spinoffs	338
7.2	Technology Transfer	339
7.2.1	Industry as Advisor	340
7.2.2	Industry as Direct Participant	341
7.2.3	Industry as Sponsor	342
7.2.4	Major Opportunities for Industry	343
	APPENDIX A: Technical Planning Activity Mission Statement	345
	APPENDIX B: Technical Planning Activity Participants and Meetings	353
	APPENDIX C: Glossary	361

FIGURES

P.1	Structure of Technical Planning Activity and Its Relationship to Magnetic Fusion Program Plan	4
------------	--	----------

FIGURES (Cont'd)

S.1	Characterization of Tasks for Magnetic Confinement Systems	13
S.2	Characterization of Tasks for Burning Plasmas	15
S.3	Characterization of Tasks for Plasma Technology	16
S.4	Characterization of Tasks for Nuclear Technology	17
S.5	Characterization of Tasks for Fusion Materials	18
S.6	Characterization of Tasks for Systems Design and Analysis	19
S.7	Level 0 Logic Diagram -- Magnetic Fusion Program Plan Overview	21
S.8	Top Level Decision Points in the Magnetic Fusion Program	23
S.9	Examples of Major Decision Branch Points, Based Primarily on the Burning Plasma Issue	25
S.10	Level 0 Logic Diagram Depicting Reference Scenario	27
1.1	Characterization of Magnetic Confinement Systems Tasks	49
1.2	Confinement Systems Development	50
1.3	Level 1 Generic Logic Diagram for Confinement Systems and Burning Plasma Areas	53
1.4	Interrelationships among Concepts for Macroscopic Equilibrium and Dynamics Issues	54
1.5	Schematic Illustration of Alternative Concept-Development Paths	56
1.6	Confinement-Concept Classification	57
1.7	Progress of Confinement Systems Concepts	59
1.8	Level 2 Logic Diagram for Tokamaks	65
1.9	Level 3 Logic Diagram for Tokamak Macroscopic Equilibrium and Dynamics Issue	67
1.10	Level 3 Logic Diagram for Tokamak Transport Issues	68
1.11	Level 3 Logic Diagram for Tokamak Wave-Plasma Interactions	70

FIGURES (Cont'd)

1.12	Level 3 Logic Diagram for Tokamak Particle-Plasma Interactions.....	72
1.13	Level 2 Logic Diagram for Tandem Mirrors	86
1.14	Level 3 Logic Diagram for Wave-Plasma Interactions in Tandem Mirrors	88
1.15	Level 3 Logic Diagram for Configuration Optimization in Tandem Mirrors	90
1.16	Level 3 Logic Diagram for Mirror Technology	91
1.17	Level 2 Logic Diagram for Tandem Mirror for Reactor Evaluation	93
1.18	Level 2 Logic Diagram for Tandem Mirror for Nuclear Technology Testing	97
1.19	Level 2 Logic Diagram for Stellarator.....	104
1.20	Level 2 Logic Diagram for Composite Issues for the Stellarator.....	109
1.21	Level 3 Logic Diagram for Confinement Systems Issues for the Stellarator	110
1.22	Level 2 Logic Diagram for the Reversed-Field Pinch	114
1.23	Level 3 Logic Diagram for Pressure- and Current-Driven Effects in the RFP	115
1.24	Level 3 Logic Diagram for Plasma Technology for the RFP	116
1.25	Level 2 Logic Diagram for the Spheromak	124
1.26	Level 3 Logic Diagram for Spheromak Plasma Technology	128
1.27	Level 2 Logic Diagram for the Field-Reversed Configuration	131
1.28	Level 2 Logic Diagram for Dense Z-Pinch	140
2.1	Characterization of Tasks for Burning Plasmas	143
2.2	Schematic Illustration of Burning Plasma Program Development	145
2.3	Tokamak Burning Plasma Logic Diagram	153

FIGURES (Cont'd)

3.1	Level 1 Logic Diagram for Plasma Technology	159
3.2	Characterization of Tasks for Plasma Technology	160
3.3	Objectives and Attributes for Magnets	168
3.4	Level 2 Logic Diagram for Magnets	169
3.5	Objectives and Attributes for Heating and Current Drive	175
3.6	Level 2 Logic Diagram for Heating and Current Drive	182
3.7	Level 3 Logic Diagram for Ion Cyclotron Heating	184
3.8	Level 3 Logic Diagram for Lower Hybrid Heating	185
3.9	Objectives and Attributes for Fueling	188
3.10	Level 2 Logic Diagram for Fueling	190
3.11	Objectives and Attributes for Plasma-Interactive Components	193
3.12	Level 2 Logic Diagram for Plasma-Interactive Components	199
3.13	Level 3 Logic Diagram for Particle and Impurity Control	200
3.14	Level 3 Logic Diagram for Heat Removal	201
3.15	Level 3 Logic Diagram for Transient Events	202
4.1	Hierarchy of Objectives for Nuclear Technology	213
4.2	Level 1 Logic Diagram for Nuclear Technology	216
4.3	Primary and Secondary Objectives and Attributes for Blanket/First Wall Development	224
4.4	Level 2 Logic Diagram for Blanket/First Wall	228
4.5	Level 2 Logic Diagram for Shield	243
4.6	Level 2 Logic Diagram for Tritium	249
5.1	Program Elements for Fusion-Materials Area	260

FIGURES (Cont'd)

5.2	Objectives for Fusion Materials	265
5.3	Level 1 Logic Diagram for Materials	266
5.4	Level 2 Logic Diagram for Structural Materials	285
5.5	Level 3 Logic Diagram for Structural Materials	288
5.6	Level 2 Logic Diagram for Nonstructural Blanket Materials	296
5.7	Level 2 Logic Diagram for Special Materials	301
6.1	Characterization of Tasks for Systems Design and Analysis	303
6.2	Level 1 Logic Diagram for Systems Design and Analysis	306
6.3	Level 2 Logic Diagram for Applications, Economics, and Technology Transfer	310
6.4	Level 2 Logic Diagram for Fusion Test Facilities, Critical Issues, and Development Pathways	314
6.5	Level 2 Logic Diagram for Safety, Environment, and Licensing	320
6.6	Level 3 Logic Diagram for Safety	321
6.7	Level 3 Logic Diagram for Environment and Licensing	322
6.8	Level 2 Logic Diagram for Remote Technology	325
6.9	Level 2 Logic Diagram for Alternative Fuels	328
7.1	Characteristics of Commercial Fusion Systems	335
7.2	Logic Diagram for Technology-Transfer Plan	344

TABLES

P.1	Major Research Areas and Program Elements under the Technical Planning Activity	6
S.1	Objectives and Planning Targets for Commercial Electric Applications of Fusion Power	10

TABLES (Cont'd)

S.2	Representative Goals for Commercial and Integrated Test Facilities	11
S.3	Confinement-Concept Classification	15
S.4	Decision E_1 on a Short-Pulse Ignition Demonstration	34
S.5	Decision(s) E_2 for Progressing to Integrated Proof-of-Principle Experiments	35
S.6	Decision E_3 on Long-Burn Demonstration	36
S.7	Decision E_4 on Fusion Testing of Nuclear Technology	38
S.8	Decision E_5 on Fusion Testing for Materials	39
S.9	Decision(s) E_6 for Progressing to Reactor-Level-Condition Experiments	40
S.10	Decision Description -- Fusion Assessment and Integrated Fusion Facility	41
S.11	Compact Ignition Tokamak	43
S.12	Proof-of-Principle Test Facility for Confinement Concepts	44
S.13	Engineering Test Reactor	45
S.14	Nuclear Technology Test Facility	46
S.15	Materials Irradiation Test Facility	47
S.16	Reactor-Condition Test Facility for Confinement Concepts	48
1.1	Achievement Objectives for Confinement Systems	55
1.2	Tokamak Program Elements, Subelements, Objectives, and Attributes	61
1.3	Decision E_1 on Short-Pulse Ignition Experiment	74
1.4	Decision <a> on Impurity-Control Option Favored for Future Applications	75
1.5	Decision on Heating and Current-Drive Options Favored for Future Development and Application	77

TABLES (Cont'd)

1.6	Decision <c> on an Advanced-Geometry Tokamak Experiment	80
1.7	Decision <d> on a Steady-State Tokamak Experiment	81
1.8	Decision E_3 on a Long-Burn Facility	82
1.9	Decision <e> on Tokamak Reactor Evaluation	83
1.10	Large Tokamak Facilities	84
1.11	Other Major Tokamak Facilities	85
1.12	Decision E_2 for Progressing to Tandem Mirror Integrated Proof-of-Principle Experiment	94
1.13	Decision E_6 for Progressing to Tandem Mirror Reactor-Level Test	95
1.14	Combined Decisions E_4 and E_5 for Proceeding with a Tandem Mirror Nuclear Technology and Materials Test Facility	98
1.15	Stellarator Program Elements, Subelements, Objectives, and Attributes	101
1.16	Major Stellarator Facilities	106
1.17	Decision <a> on a Large Stellarator Experiment	107
1.18	Decision on a Stellarator Ignition Experiment	108
1.19	Reversed-Field-Pinch Program Elements, Subelements, Objectives, and Attributes	112
1.20	Decision E_2 on the RFP Integrated Proof-of-Principle Test Phase	117
1.21	Decision E_6 on the RFP Reactor-Level Test Phase	118
1.22	Spheromak Program Elements, Subelements, Objectives, and Attributes	121
1.23	Decision for Spheromak Concept Test Experiment	125
1.24	Decision E_2 for Spheromak Integrated Proof-of-Principle Experiment	126
1.25	Decision E_6 for Spheromak Reactor-Level Plasma Experiment	127

TABLES (Cont'd)

1.26	Field-Reversed Configuration Program Elements, Subelements, Objectives, and Attributes	129
1.27	Decision E_2 for Progressing to Integrated Proof-of-Principle Experiment for the FRC	132
1.28	Decision E_6 for Progressing to Reactor-Level Test for the FRC	133
1.29	Program Elements, Subelements, Objectives, and Attributes for the Bumpy Toroid	136
1.30	Program Elements, Subelements, Objectives, and Attributes for the Dense Z-Pinch	138
1.31	Dense Z-Pinch Decision on Proceeding with DD Burn Experiment	141
2.1	Achievement Objectives for Burning Plasmas	146
2.2	Tokamak Burning Plasma Program Elements, Subelements, Objectives, and Attributes	148
3.1	Decisions on Heating System <a> and Current Drive for Long-Burn Demonstration	161
3.2	Decision <c> on Nonwater Coolants for Plasma-Interactive Components	162
3.3	Decision <d> on Pellet Fueling Requirements	163
3.4	Decision <e> on Useful Heat Removal from Plasma-Interactive Components	164
3.5	Decisions <f, g, h> on Plasma Technology Performance for Long-Burn Demonstration	165
3.6	Evaluation <i> of Plasma Technology for an Integrated Fusion Facility	166
3.7	Decision <a> on Magnets for an Integrated Fusion Facility	170
3.8	Revisions to Existing Facilities	171
3.9	Heating and Current-Drive Issues	174
3.10	Fueling Issues	187

TABLES (Cont'd)

3.11	Issues Associated with Plasma-Interactive Components	192
3.12	Important Factors in Constructing Attribute Scales for Plasma-Interactive-Component Objectives	194
3.13	Decision <a> on Plasma-Interactive-Component Selection for the Short-Pulse Ignition Device	204
3.14	Decision on Plasma-Interactive-Component Selection for the Long-Burn Demonstration Experiment	205
4.1	Scope of Fusion Nuclear Technology	208
4.2	Major Technical Issues for Fusion Nuclear Technology	209
4.3	Key Parameters in Constructed Scale for Blanket Attributes	214
4.4	Characterization of Nuclear Technology Tasks	215
4.5	Evaluation Points and Milestones for Nuclear Technology	217
4.6	Nonfusion Facilities for Nuclear Technology	219
4.7	Technical Issues for Blanket/First Wall Components	221
4.8	Attributes and Nominal Goals for the Secondary Objective of Attractiveness	225
4.9	Blanket/First Wall Tasks	227
4.10	Blanket/First Wall Evaluation Points and Milestones	229
4.11	Description of Evaluation Point <a>, Liquid-Breeder Blanket Materials and Design Classes	231
4.12	Description of Evaluation point , Solid-Breeder Blanket Materials and Design Classes	233
4.13	Blanket/First Wall Test Facilities	234
4.14	Summary of Radiation-Shield Issues	238
4.15	Radiation-Shield Development Objectives	241
4.16	Milestones for the Radiation-Shield Technical Planning Activity	244

TABLES (Cont'd)

4.17	Facilities to be Used in Radiation-Shield Development Activities	245
4.18	Summary of Issues Associated with Tritium Processing Systems.....	246
4.19	Decision <c> on Water-Detritionation System	250
4.20	Facilities to be Used in Tritium-Processing Development	252
4.21	Nuclear Aspects of Plasma-Interactive Components.....	253
4.22	Facility for Testing Plasma-Interactive Components with Nonwater Coolants	256
5.1	Major Technical Issues for Fusion Materials	262
5.2	Objectives for Fusion Materials Development	267
5.3	Selection of Primary Candidate First-Wall/Blanket Structural Alloys, <a>	268
5.4	Selection of Primary Candidate Breeder Materials, 	269
5.5	Selection of Primary Candidate Insulator Materials, <c>	270
5.6	Selection of Primary Candidate Plasma-Interactive-Component Structural Alloys, <d>	271
5.7	Selection of Primary Candidate Materials for Special Applications, <e>	272
5.8	Selection of Improved Alloys for High-Fluence Testing, <f>	273
5.9	Selection of Improved PIC Alloys for High-Fluence Testing, <g>	274
5.10	Selection of Reference Nonstructural and Special Materials, <h>	275
5.11	Selection of Reference Structural Alloys, <i>	276
5.12	Additional Materials Test Facilities	277
5.13	Candidate Structural Alloys	278
5.14	Objectives for Structural Materials Development	286
5.15	First-Wall/Blanket Structural Materials	289

TABLES (Cont'd)

5.16	Candidate Nonstructural Blanket Materials	291
5.17	Nonstructural Blanket Materials	297
6.1	Systems Design and Analysis Program Elements and Subelements	305
6.2	Objectives and Attributes for Applications, Economics, and Technology Transfer Program Element	309
6.3	Objectives and Attributes for Fusion Test Facilities, Critical Issues, and Development-Pathways Program Element	312
6.4	Objectives and Attributes for Safety, Environment, and Licensing Program Element	318
6.5	Objectives and Attributes for Remote-Technology Program Element	324
6.6	Objectives and Attributes for Alternative Fuels Program Element	327
7.1	Economic Objectives, Attributes, and Target Values for Electricity Production	336
7.2	Safety, Environment, and Licensing -- Objectives, Attributes, and Target Values	337
7.3	Spinoffs from Fusion Science and Technology Programs	339
7.4	Industrial Roles and Functions	340

ACKNOWLEDGMENTS

This report was prepared by, and represents the contributions of, a large number of people. Overall coordination was provided by the TPA Steering Committee (listed in Appendix B). J.M. Rawls (GA Technologies, Inc. - GA) was the principal author of the Preface. Principal authors of the Synopsis were C.C. Baker and R.E. Nygren (Argonne National Laboratory - ANL).

The Plasma Science Group, led by J.D. Callen and assisted by J.F. Santarius (University of Wisconsin - UW), was responsible for Chapters 1 and 2. Principal additional authors of these chapters included D. Baker (Los Alamos National Laboratory - LANL), D. Baldwin (Lawrence Livermore National Laboratory - LLNL), R. Dory (Oak Ridge National Laboratory - ORNL), R. Hazeltine (University of Texas at Austin), R. Linford (LANL), J. Lyon (ORNL), R. Post (Massachusetts Institute of Technology - MIT), J. Rawls, P. Rutherford (Princeton Plasma Physics Laboratory - PPPL), J. Sheffield (ORNL), R. Siemon (LANL), A. Sherwood (LANL), and B. Wright (LANL).

The Fusion Technology Group, led by M.A. Abdou (University of California at Los Angeles - UCLA) and assisted by R.F. Mattas (ANL), had responsibility for Chapters 3-5. Chapter 3 was coordinated by L.A. Berry (ORNL) and included as additional principal authors W. Gauster (Sandia National Laboratory), C. Henning (LLNL), R. Mattas, D. Schuresko (ORNL), and D. Swain (ORNL). Chapter 4 was coordinated by M.A. Abdou and R.F. Mattas and included as additional principal authors J. Bartlit (LANL), M. Tillack (UCLA), and K. Schultz (GA). Chapter 5 was coordinated by D.L. Smith (ANL) and included as additional principal authors F. Clinard (LANL), D. Doran (Hanford Engineering Development Laboratory - HEDL), C. Johnson (ANL), A. Rowcliffe (ORNL), and H. Yoshikawa (HEDL).

The Fusion Systems Group, led by S.O. Dean (Fusion Power Associates - FPA), was responsible for Chapters 6 and 7. The additional principal authors included D. Cohn (MIT), J. Crocker (EG&G/Idaho), C. Flanagan (Fusion Engineering Design Center - FEDC), J. Gordon (TRW), R. Krakowski (LANL), G. Miley (University of Illinois), K. Schultz, J. Sheffield, and W. Stacey (Georgia Institute of Technology).

Several people made major efforts in the editing, typing, preparation of logic diagrams, and assembling of this report. They include R. Johns (ANL), M. Anthony (ANL), L. Ciarlette (ANL), F. Bennett (ANL), B. Brown (UW), L. Kickels (ANL), J. McGill (ORNL), J. Peerenboom (ANL), T. Purdy (ANL), R. Watkins (FPA), and R. Whitfield (ANL).

The successful development of fusion would make available a safe, economical, and environmentally attractive energy resource based on a secure, essentially unlimited fuel supply. The broad range of potential applications of fusion power includes the production of electricity, fissile fuels, and synthetic fuels, as well as high-grade heat for industrial applications.

This tantalizing prospect has led the major industrialized nations to vigorously pursue programs aimed at harnessing the energy released by fusion reactions. Most of these programs rely upon magnetic fields to confine high-energy plasmas. In addition to making continuous and dramatic progress toward the goal of fusion power, magnetic-fusion programs have been an important source of new scientific insights and technological developments that go far beyond energy applications.

Demonstrating an economically competitive fusion energy source constitutes a challenge necessitating major scientific and technological advances. In seeking to meet this challenge, fusion research has progressed from exploratory studies in the 1950s and 1960s to ever more sophisticated and ambitious undertakings. The significant progress made on all fronts during the past decade justifies confidence that the challenge will be met.

Recent advances, particularly the achievement of plasma conditions approximating those of a fusion power source, provide the technical basis for defining a program plan to advance fusion research. Such a plan will facilitate informed decision making on the viability of commercial fusion systems around the year 2000.

The Magnetic Fusion Program Plan (MFPP), issued by the U.S. Department of Energy (DOE) in February 1985 (Ref. 1), constitutes the planning framework. The MFPP identifies key technical issues and strategic objectives and outlines an overall strategy for fusion research through the end of the century. In order to formulate more detailed technical plans supporting this strategy, DOE established the Technical Planning Activity (TPA). Through a process of broad fusion community participation and review, the TPA has developed the technical planning information presented in this report.

PLANNING ASSUMPTIONS

The planning activity has focused on four technical issues identified in the MFPP:

Magnetic Confinement Systems. Developing a sufficient understanding of the plasma science concepts suitable for commercial applications of fusion energy.

Properties of Burning Plasmas. Understanding the effects introduced when the plasma is internally heated by the fusion reaction.

Fusion Nuclear Technologies. Developing nuclear technologies unique to fusion for the commercial application of fusion energy.

Fusion Materials. Developing materials that will enhance fusion's economic and environmental potential.

Both the technical objectives of the program and the exposition of the plans themselves have been formulated in a

way that is directly traceable to the MFPP through this set of key technical issues. The character of the plans also reflects the three strategic objectives set forth in the MFPP:

Science Objective. Attain a predictive capability for the behavior of plasmas confined in fusion-relevant magnetic configurations.

Technology Objective. Develop unique fusion components that can operate under conditions relevant to fusion energy sources.

Technology-Transfer Objective. Provide a range of options for private-sector investment in the commercial development of fusion.

Together with the MFPP, a mission statement issued in conjunction with the initiation of the TPA (see App. A) provides the basic ground rules for this activity. The most important of these rules is the stated goal of the magnetic-fusion program: to establish the scientific and technological base required to assess the economic and environmental aspects of fusion energy. In structuring a set of technical programs to achieve this goal, a planning horizon of 15-20 years has been adopted, with the greatest attention to detail being devoted to the first five years. The assessment process is expected to culminate in the building and operation of one or more integrated fusion facilities in the post-2000 period.

On the basis of general guidance from DOE, the TPA has made certain assumptions about budget and facility constraints. One overall constraint is that the annual rate of real (inflation-adjusted) U.S. fusion-program expendi-

tures through 1990 will not increase. The effective resources available to the program during this period, however, may increase to the extent that complementary international activity and collaboration takes place. In addition, a measure of program balance is enforced by the requirement that no single project require U.S. annual funding exceeding 20% of the total U.S. program budget.

The TPA has chosen not to undertake an exhaustive treatment of the multitude of program logics encompassed within the framework of this guidance; in any event, technical developments will inevitably require periodic revision of the plans. Instead, the bulk of the TPA's effort has been devoted to a reference case (defined in the Synopsis following this preface) consistent with the TPA mission statement. The TPA has considered alternative scenarios only to the extent of identifying the major branch points in the program logic. Detailed planning information is presented only for the reference case.

In addition to the technical planning information, a principal product of the TPA is a planning methodology that is sufficiently rigorous to be of value to DOE in its planning and implementation of the fusion program, yet sufficiently flexible to accommodate technical advances and changing national priorities.

MECHANISM FOR DEVELOPING PLANS

A vital aspect of the TPA's approach to program planning has been thorough involvement of the fusion community in every stage of the process. The views of all the major participants in the fusion program have been represented through a

Steering Committee composed of about 30 senior technical experts within the fusion community. The Steering Committee has been responsible for documenting the status and plans of the individual program elements. A smaller Executive Committee has directed the day-to-day activities of the TPA and has synthesized the input from the Steering Committee. Both committees are chaired by the leader of the TPA, Dr. C.C. Baker of Argonne National Laboratory. As chairperson, Dr. Baker reports to Dr. J.F. Clarke, Director of the U.S. Magnetic Fusion Program, who has overall responsibility for providing direction to the TPA.

Appendix B lists the membership of these committees and gives a chronology of the formal meetings of the TPA. In addition to these formal meetings, the TPA has provided the community at large with broader exposure to the planning process through briefings at the various laboratories, through presentations to the Magnetic Fusion Advisory Committee (MFAC), and through continuing interaction with the DOE Office of Fusion Energy.

The comprehensive approach undertaken by the TPA has required that a balance be maintained between a near-term perspective, focused primarily on the research tasks required to resolve the four key technical issues, and a long-range perspective, focused on the overall goal of the program. Achieving this balance was a major factor in the organization of the Steering Committee. Three groups were formed to provide the essential elements of this organization:

Plasma Science Group, responsible for roll-forward planning associated with

magnetic confinement systems and properties of burning plasmas (chaired by Prof. J.D. Callen, University of Wisconsin).

Fusion Technology Group, responsible for roll-forward planning associated with fusion nuclear technology and fusion materials (chaired by Prof. M.A. Abdou, University of California, Los Angeles). This group also coordinated the overall planning for plasma technology.

Fusion Systems Group, responsible for roll-back planning starting with the overall program goal and roll-forward planning associated with provision of systems design and analysis support to the fusion program (chaired by Dr. S.O. Dean, Fusion Power Associates).

Figure P.1 shows the relationships between the TPA's structure and the MFPP goal, key technical issues, and strategic objectives.

PLANNING APPROACH

The planning approach adheres to the overall policy and strategy articulated in the MFPP. The TPA broke down the MFPP's four key technical issues into more narrowly defined issues that must be resolved in order to meet the overall program goal. For each issue, a technical objective was specified to define what would constitute resolution of the issue. Attributes were identified to measure progress toward achieving these objectives. The three working groups of the TPA have each issued a report detailing all the issues in their respective areas and their associated objectives and attributes, together with a summary of the technical status for each issue (Refs. 2-4). The groups' principal

FIGURE P.1 Structure of Technical Planning Activity and Its Relationship to Magnetic Fusion Program Plan

findings were assembled into a document that deals with these prerequisites to the planning process, spanning the entire fusion program (Ref. 5).

A set of program elements has been defined, where each element represents a portion of the U.S. Magnetic Fusion Program for which the technical issues and objectives are similar. This report presents the TPA's planning information by means of logic diagrams that show (1) overall sequences of major tasks required to meet the technical objectives, (2) key decision points, and (3) milestones for each program element. Interrelationships among the program elements are also defined.

The principal products of the TPA (beyond the planning process itself) comprise this set of logic diagrams, together with the definition of program elements and the detailed technical issues associated with them, the statement of objectives and their associated attributes, the specification of technical criteria for major program decisions, and the identification of facilities and resource requirements needed to carry out the technical work.

STRUCTURE OF TECHNICAL PLANS

In identifying the program elements that serve as the basic building blocks of the planning activity, the TPA has found it useful to classify the technical work into six major research areas:

1. Confinement Systems
2. Physics of Burning Plasmas
3. Plasma Technology
4. Nuclear Technology
5. Materials
6. Systems Design and Analysis

This classification scheme is the skeletal outline of the comprehensive set of program elements presented in Table P.1. (Four of the areas -- 1,2,4, and 5 -- correspond directly to the key issues in the MFPP.) In this table, the first-order headings (e.g., Confinement Systems) are the major areas of research, the second-order headings (e.g., Macroscopic Equilibrium and Dynamics) are the program elements, and the third-order headings (e.g., Pressure-driven Effects) are program subelements.

The technical strategy for achieving program objectives is summarized throughout this report by means of a hierarchy of logic diagrams. The top-level (Level 0) logic diagram, which deals with the entire magnetic-fusion program, illustrates the major evaluation/decision points and milestones involved in meeting the top-level program objectives. At the next level of planning (Level 1), there is usually one logic diagram provided for each of the six major areas of research. At Level 2, one logic diagram is provided for each program element. In many cases, further planning has been carried out at Level 3 (i.e., at the program-subelement level); however, in order to keep the report to a manageable size, only a few Level 3 diagrams are displayed.

This report discusses the program elements as they are grouped in Table P.1. Chapters 1-6 deal with the six major areas of research. The Synopsis preceding these chapters provides an overview of the program planning information, and the final chapter deals with the roll-back aspects of the program.

TABLE P.1 Major Research Areas and Program Elements under the Technical Planning Activity

<p>1. Confinement Systems</p> <p>1.1 Macroscopic Equilibrium and Dynamics</p> <p>1.1.1 Pressure-driven Effects</p> <p>1.1.2 Current-driven Effects</p> <p>1.1.3 Kinetic Effects</p> <p>1.2 Transport</p> <p>1.2.1 Perpendicular Confinement</p> <p>1.2.2 Parallel Confinement</p> <p>1.3 Wave-Plasma Interactions</p> <p>1.3.1 Wave Heating</p> <p>1.3.2 Non-Maxwellian Effects</p> <p>1.4 Particle-Plasma Interactions</p> <p>1.4.1 Impurity Control</p> <p>1.4.2 Fueling</p> <p>1.4.3 Neutral Beam Injection</p> <p>1.5 Composite Issues</p> <p>1.5.1 Configuration Optimization</p> <p>1.5.2 Profile Optimization</p> <p>1.5.3 Pulse Length Optimization</p>	<p>3.3 Fueling</p> <p>3.3.1 Pellet Fueling</p> <p>3.4 Plasma Interactive Components</p> <p>3.4.1 Particle Control</p> <p>3.4.2 Impurity Control</p> <p>3.4.3 Heat Removal and Recovery</p> <p>4. Nuclear Technology</p> <p>4.1 Blanket/First Wall</p> <p>4.1.1 Liquid Breeder Blankets</p> <p>4.1.2 Solid Breeder Blankets</p> <p>4.1.3 Hybrid Blankets</p> <p>4.1.4 Alternative Power Conversion</p> <p>4.2 Shield</p> <p>4.2.1 Radiation Protection Requirements</p> <p>4.2.2 Code and Data Verification</p> <p>4.2.3 Shield Design</p> <p>4.3 Tritium Processing</p> <p>4.3.1 Plasma Exhaust Processing</p> <p>4.3.2 Vacuum</p> <p>4.3.3 Tritium Permeation and Control</p> <p>4.3.4 Blanket Product Processing</p> <p>4.4 Nuclear Elements of PIC</p>
<p>2. Burning Plasma Physics</p> <p>2.1 Alpha Particle Effects</p> <p>2.1.1 Alpha Particle Containment</p> <p>2.1.2 Alpha Component Effects</p> <p>2.1.3 Alpha Heating</p> <p>2.2 Burn Control and Ash Removal</p> <p>2.2.1 Burn Control</p> <p>2.2.2 Ash Removal</p> <p>3. Plasma Technology</p> <p>3.1 Magnets</p> <p>3.1.1 Copper Magnets</p> <p>3.1.2 Pulsed Magnets</p> <p>3.1.3 Superconducting Magnets</p> <p>3.2 Heating and Current Drive</p> <p>3.2.1 Electron Cyclotron Heating</p> <p>3.2.2 Ion Cyclotron Heating</p> <p>3.2.3 Lower Hybrid Heating</p> <p>3.2.4 Neutral Beams</p> <p>3.2.5 Others</p>	<p>5. Materials</p> <p>5.1 Structural Materials</p> <p>5.1.1 First Wall/Blanket Alloys</p> <p>5.1.2 PIC Alloys</p> <p>5.2 Nonstructural Blanket Materials</p> <p>5.2.1 Solid Breeders</p> <p>5.2.2 Liquid Breeders/Coolants</p> <p>5.2.3 Neutron Multiplier/Moderator</p> <p>5.2.4 Ceramic Insulators</p> <p>5.3 Special Materials</p> <p>5.3.1 Plasma Facing Materials</p> <p>5.3.2 Coatings/Claddings</p> <p>5.3.3 Magnet Materials</p>

TABLE P.1 (Cont'd)

6. Systems Design and Analysis

- 6.1 Applications, Economics, and Technology Transfer
 - 6.1.1 Commercial Reactor Preconceptual Design
 - 6.1.2 Applications Studies
 - 6.1.3 Economics Analysis
 - 6.1.4 Availability Analysis
 - 6.1.5 Technology Transfer Studies
 - 6.2 Fusion Test Facilities, Critical Issues, and Development Pathways
 - 6.2.1 Fusion Test Facilities Preconceptual Design
 - 6.2.2 Critical Issues Analysis
 - 6.2.3 Engineering Data Base Assessment
 - 6.2.4 Development-Pathways Analysis
 - 6.3 Safety, Environment, and Licensing
 - 6.3.1 Safety
 - 6.3.2 Environment
 - 6.3.3 Licensing
 - 6.4 Remote Technology
 - 6.4.1 Program Plan
 - 6.4.2 Concept Design
 - 6.4.3 Component Development
 - 6.4.4 Applications
 - 6.5 Alternative Fuel Cycles
 - 6.5.1 Confinement Systems and Burning Plasmas
 - 6.5.2 Plasma Technology
 - 6.5.3 Nuclear Technology and Materials
 - 6.5.4 Systems Design and Analysis
-

REFERENCES

1. *Magnetic Fusion Program Plan*, U.S. Dept. of Energy, Office of Energy Research, DOE/ER-0214, Feb. 1985.
2. J.D. Callen et al., "TPA Plasma Science Status Report," Univ. of Wisconsin, prepared for U.S. Dept. of Energy, Sept. 1986.
3. M.A. Abdou et al., "TPA Technology Status Report," Univ. of California at Los Angeles, prepared for U.S. Dept of Energy, PPG-908, UCLA-ENG-85-38, Nov. 1985.
4. S.O. Dean et al., "TPA Fusion Systems Status Report," Fusion Power Associates, prepared for U.S. Dept. of Energy, Nov. 1985.
5. C.C. Baker et al., "Summary Report -- Technical Planning Activity," Argonne National Laboratory, Dec. 6, 1985.

This synopsis provides an overview of the results of the Technical Planning Activity (TPA). Section S.1 describes proposed objectives for commercial applications of fusion. This information is used to motivate and guide the near-term research and development (R&D) programs and to set long-range objectives. Section S.2 describes interim objectives for the next 15-20 years and illustrates the general character of the program through descriptions of each of the six TPA program areas.

The essential planning information in this report is contained in a series of logic diagrams that show the sequences of and interrelationships among technical tasks. The logic diagrams display the key technical accomplishments and major decisions necessary to resolve major issues. Section S.3 shows the highest-level (overview) logic diagram.

Section S.4 briefly describes the major decisions on large facilities that will dominate the specific programmatic course of action leading to the key technical accomplishments. Section S.5 describes the reference scenario (a particular sequence of decisions) used in developing the logic diagrams. The same section also introduces other possible decision paths that could lead to alternative scenarios.

Section S.6 summarizes the resource requirements for the fusion program. Section S.7 recommends areas of future work and explains some of the inherent difficulties and problems encountered in the planning activity.

S.1 OBJECTIVES FOR COMMERCIAL FUSION APPLICATIONS

Fusion has a wide spectrum of potential applications. Most studies have focused on the production of electric power, because this has been regarded as the most probable application. Other applications might include breeding of fissile fuels, production of radioisotopes, production of hydrogen and other chemicals, transmutation or "burning" of various nuclear or chemical wastes, radiation processing of materials, food preservation, medical diagnosis and treatment, space power and space propulsion.

Table S.1 lists proposed objectives and planning targets for commercial electric applications. These objectives are grouped under two categories: (1) economics and (2) safety, environment, and licensing. The most important economic objectives are to minimize the cost of products, maximize investment protection, and minimize capital costs. The most important safety, environment, and licensing objectives are to maximize public safety and plant personnel safety, minimize long-term neutron activation, and minimize licensing time. (Further details are presented in Chapters 6 and 7.)

Table S.2 projects magnetic fusion system and plasma parameters for a typical commercial electric-power plant and for an intermediate integrated fusion facility (IFF). (The IFF, a facility based on commercially relevant fusion technology, would be constructed after a

TABLE S.1 Objectives and Planning Targets for Commercial Electric Applications of Fusion Power

Objective	Planning Target ^a
Economics	
Show several attractive fusion applications.	Competitive economic analysis and safety/environmental features are developed for at least <i>three</i> potential applications.
Minimize cost of product: (e.g., electricity).	Cost of electricity is <i>30-40 mills/kWh</i> , evaluated in terms of levelized costing, zero escalation, and inflation in 1985 dollars.
Maximize investment protection.	Cost of recovery from any fusion core component or subsystem failure or accident is less than 5-15% of original direct capital cost. Time required for recovery from any accident is less than 3-9 months.
Minimize capital costs.	Total direct cost to construct plant is about \$1000/kW _e for a nominal 1000-MW _e plant.
Safety, Environment, and Licensing	
Maximize public safety.	Risk to the public from accidents and routine operations are less than <i>0.1%</i> of existing risk from all sources.
Maximize plant personnel safety.	Risk to plant personnel from occupational hazards and accidents is less than <i>10%</i> of the risk from non-occupational hazards.
Minimize long-term neutron activation.	>99% of radioactive waste generated qualifies for near-surface disposal, as defined in 10CFR61 or relevant extensions thereof.
Minimize licensing time.	Licensing process is completed <i>prior to or during</i> construction.

^aChapters 6 and 7 use a slightly different format in which only the numerical value (italicized here) is called the "planning target" and the rest of the description is referred to as the "attribute."

TABLE S.2 Representative Goals for Commercial and Integrated Test Facilities^a

Parameters	Commercial Facility	Integrated Fusion Facility
System Parameters		
Net electric power (MW_e)	600-1,200	---
Fusion power (MW_{th})	1,800-3,600	< 600
Weight of fusion island (tonnes)	6,000-12,000	---
Neutron wall load (MW/m^2)	3-6	1-3
Wall lifetime fluence ($\text{MW}\cdot\text{yr}/\text{m}^2$)	10-20	4-8
Availability (%)	~ 75	20-30
Duty Factor (%)	~ 100	75-100
Tritium breeding ratio	> 1.05	< 1.0
Plasma Parameters		
Average beta, $\langle\beta\rangle$ (%)		
Toroidal devices ^b	8-20	5-20
Tandem mirrors	25-50	25-50
Field-reversed configuration	65-90	65-90
Thermal diffusivity, ^c χ_E (m^2/s)		
Tokamak, superconducting RFP	1.0-1.5	0.2-0.5
Stellarator, resistive RFP, and spheromak	0.4-0.7	0.15-0.25
Tandem mirror	0.10-0.15	0.06-0.08
Field-reversed configuration	0.5-1.2	0.2-0.4

^aThe parameter goals listed are appropriate for commercial production of electricity. Magnetic confinement concepts are defined in more detail in Chapter 1.

^b"Toroidal devices" include tokamaks, stellarators, reversed-field pinches (RFPs), and spheromaks.

^cValues to satisfy $n\tau_E = 3 \times 10^{20} \text{ s}/\text{m}^3$ and $T_i = 10 \text{ keV}$ with the indicated β value.

positive assessment of the commercial prospects for fusion.) The parameter ranges given in the table are representative but are not necessarily an optimized set. The target values are useful in establishing objectives for the R&D programs and in guiding trade-off studies.

Table S.2 also identifies two physics parameters that characterize the fusion core: (1) the average beta, $\langle \beta \rangle$, defined to be the ratio of the average plasma pressure to the average pressure of the confining magnetic fields; and (2) the thermal diffusivity, χ_E , which determines the energy-confinement time of the plasma. These parameters, which serve as key indicators for measuring progress toward meeting confinement system objectives, are discussed further in Chapter 1. The numerical values for various concepts differ. The specific numerical values given in Table S.2 were selected to indicate the level of physics performance required in order for a specific concept to lead to an attractive reactor as indicated by the generic system parameters.

The achievement of commercial fusion applications will require appropriate, timely industrial participation. The roles played by industry in fusion development include activities in three broad categories: (1) as advisors, (2) as direct participants in government-funded projects, and (3) as sponsors of industry-funded activities. The participation of industry will affect how the activities described in the plan are carried out, rather than the nature of the activities required. Commercial objectives and technology transfer issues are discussed more fully in Chapter 7.

S.2 INTERIM TECHNICAL OBJECTIVES (1986-2005)

Interim objectives for the magnetic fusion program, from the present through the year 2005, have been established at the broadest level, based on the four key technical issues identified in the Magnetic Fusion Program Plan (MFPP). These objectives are as follows:

Confinement Systems. Demonstrate reactor-level plasma conditions for commercially attractive confinement concepts and develop a predictive capability sufficient to design and optimize fusion systems for commercial application.

Burning Plasmas. Demonstrate high fusion energy gain with sufficient pulse length and develop the science and technology of burning plasmas.

Nuclear Technology. Develop nuclear technology that leads to commercial fusion applications with attractive economic, safety, and environmental features.

Fusion Materials. Develop improved materials that will enhance the economic, safety, and environmental features for commercially competitive fusion applications.

The TPA has used a timetable that is consistent with the MFPP schedule. The end point of the timetable represents an assessment of fusion's economic and safety/environmental potential, scheduled to occur by approximately 2005. A positive assessment would mark the point of departure for the commercial demonstration of fusion energy and would include a decision to proceed with an

integrated fusion facility. The actual timing for this decision point will depend on the pace of the program (see Sec. S.5). The following pages describe the general progress needed in each of the six major research areas of the program in order to meet the goal of the MFPP. The strong couplings between the program areas are described in Sec. S.2.7.

S.2.1 Confinement Systems

The objectives for the development of magnetic confinement systems are to demonstrate reactor-level plasma conditions for commercially attractive confinement concepts and to develop a sufficient predictive capability to design and optimize fusion systems for commercial applications. This predictive capability will be based both on empirical models derived from experimental data and on theoretical models. Both of the objectives are important in developing attractive concepts and producing an adequate data base for a realistic assessment of fusion. Figure S.1 summarizes the essential characteristics of the magnetic confinement systems tasks.

Sufficiently developed concepts could lead to reactor-level tests that would directly influence the fusion assessment in 2005. Some advanced concepts with a higher reactor potential would probably be explored at more modest parameter levels by about 2005 to demonstrate their potential viability in time for the fusion assessment. These advanced concepts might then proceed to reactor-level tests at a later date.

FIGURE S.1 Characterization of Tasks for Magnetic Confinement Systems

The key technical issue of magnetic confinement systems has been subdivided into the following issues:

Macroscopic equilibrium and dynamics. How can the plasma be contained (in a macroscopic sense)?

Transport. How are heat and particles lost from the plasma when it is in a well-behaved macroscopic state?

Wave-Plasma Interactions. How can externally powered waves be used to produce non-Maxwellian distributions and plasma heating? How can stability be achieved on the fine or velocity-space scale?

Particle-Plasma Interactions. How does the plasma interact with its nonionized environment?

Composite Considerations. How can the magnetic confinement concept be integrated and optimized over all the relevant confinement systems and burning plasma issues?

Planning can be described in terms of these technical issues at the broadest level, but detailed planning of progress toward resolving the overall confinement systems issue necessarily hinges on the specific approaches adopted to confine plasmas. In particular, decisions regarding individual experimental facilities are influenced in large part by the status and prospects of the particular confinement concept involved. Still, the generic, issue-based considerations are important in all such decisions, because the progress made toward issue resolution critically affects both the technical evaluation process and the implications any decision will have for the magnetic confinement program as a whole.

In view of the program's important empirical component, and given the ongoing process of cross-fertilization among confinement concepts, it is not possible to forecast accurately the evolutionary path an individual concept will take as progress is made toward commercialization.

The status of a confinement concept can be characterized in terms of its level of evolution (i.e., exploratory phase, system-concept tests, integrated proof-of-principle, or reactor-plasma conditions), according to the guidelines in Chapter 1. The TPA has documented plans for the eight confinement concepts listed in Table S.3. The four "moderately developed" concepts have significant data bases (primarily at the system concept level), while the four

"developing" concepts have been studied almost exclusively at the exploratory level.

In principle, each of the moderately developed concepts could progress far enough to be regarded as a commercial option at the year-2005 fusion assessment. However, for three of these four (except the tokamak), the limited number of major funded facilities means that the 2005 date could be met only in the case of a largely uninterrupted sequence of funding commitments and experimental successes.

The developing concepts merit attention both for their long-term commercial prospects and for their role in advancing plasma science. Such concepts could achieve reactor-level parameters by 2005 only if progress were rapid. However, if steady progress is assumed, these developing concepts could achieve the integrated proof-of-principle level by 2005, and results at that level could have a significant impact on the fusion assessment.

S.2.2 Burning Plasmas

The burning plasma issue involves possible unknown effects that might be introduced when the heat sustaining the ignited plasma is produced from within by the fusion process itself. The distribution of heat in the plasma with the external heating methods used in present experiments, such as neutral beams and microwaves, differs from that expected with internal heating. The Magnetic Fusion Program Plan strategy calls for rapid progress on this fundamental issue, which must be resolved by observing burning plasmas. Only tokamaks have progressed far enough in

TABLE S.3 Confinement-Concept Classification

Well-Developed Knowledge Base	Moderately Developed Knowledge Base	Developing Knowledge Base
Tokamak	Advanced tokamak	Spheromak
	Tandem mirror	Field-reversed configuration (FRC)
	Stellarator	Elmo bumpy square (EBS)
	Reversed-field pinch (RFP)	Dense-Z pinch

confinement parameters that a burning plasma (ignition) experiment can be undertaken in the near future with confidence of success.

The planned approach is to follow the scientific breakeven experiments of the late 1980s with a short-pulse ignition tokamak, slated for startup around 1992 (see Fig. S.2). A more ambitious long-burn demonstration is anticipated before 2000. Although this long-burn device will provide all the information needed to resolve the burning plasma issue for a tokamak reactor, supplementary information will be required for most other confinement concepts. A mission for the long-burn device that would combine long-burn conditions and operating parameters to make it useful as a component and nuclear technology test bed is being considered. Such a mission would be a major international undertaking, and the mission details and schedule would be matters of international agreements.

FIGURE S.2 Characterization of Tasks for Burning Plasmas

S.2.3 Plasma Technology

The plasma technology area deals with components that directly control the plasma, such as magnets, heating and current drive systems, fueling systems, and components within the vacuum vessel having surfaces in close proximity to the plasma. Research and development in plasma technology provides continuing support to confinement systems devices, as well as critical input to the decisions involving each major facility.

The near-term program for plasma technology focuses on technology development for the short-pulse ignition demonstration and for the long-burn demonstration. Thus, the program schedule is tied closely to the anticipated schedules for construction and operation of the experimental devices (see Fig. S.3).

Near-term experiments, plus those for the short-pulse ignition and long-burn demonstrations, will verify the design and performance of plasma technology components and serve as a test bed for advancing these plasma technology systems to the next stage of development. Where possible, component features compatible with long-term goals will be incorporated; however, developments in plasma technology will be driven primarily by the near-term goals for confinement devices. From this development process, a basis for limited lifetime demonstrations of commercially relevant systems is expected.

Plasma technology development will require new materials and technology to meet long-term goals. For example, an advanced heat-rejection system must be able to survive in the harsh environment adjacent to the plasma in the IFF. These

FIGURE S.3 Characterization of Tasks for Plasma Technology

needs are addressed in Secs. S.2.4 and S.2.5.

S.2.4 Nuclear Technology

Fusion nuclear technology deals with systems that "harvest" neutrons and plasma energy to produce fuels and useful energy. The major systems involved are the blanket and first wall,

the radiation shield, and the tritium-processing system. The nuclear elements of plasma-interactive components are also included.

The objective for the nuclear technology area is to develop fusion nuclear technology leading to commercially competitive applications with attractive economic and safety/environmental features. Fuel self-sufficiency and efficient, reliable, and safe energy conversion must be demonstrated. The 20-yr test plan has four phases, each of approximately 5-yr duration, with the objectives shown in Fig. S.4.

The initial phase focuses on scoping experiments that will explore basic properties and phenomena. (The areas of radiation shielding and tritium handling are exceptions in that more project-specific R&D will be needed in the near future to support a short-pulse ignition experiment.) In subsequent phases, the emphasis shifts gradually to multiple-effects tests (e.g., simultaneous effects of temperature and magnetic field on liquid-metal corrosion) with progressively more sophisticated combinations of effects that simulate component service conditions. Verification of concepts in nonfusion facilities will provide the data base and experience necessary to perform integrated tests in fusion facilities by about the year 2000. An important feature of the plan is the systematic screening of materials, configurations, and design concepts in order to reduce costs associated with the more expensive integrated tests.

S.2.5 Fusion Materials

The overall objective of the fusion materials area is to develop improved

FIGURE S.4 Characterization of Tasks for Nuclear Technology

materials that will enhance the economic, safety, and environmental attractiveness of fusion as a commercially competitive energy source. In order to reduce waste-management requirements and enhance the potential for recycling of material, materials with low, long-term activation are emphasized. The

materials for various applications have been categorized in terms of three program elements: structural materials, nonstructural blanket materials, and special materials for other applications.

The primary focus of the materials program is the development of materials that will provide high performance and long lifetime in the unique environment of high-energy (up to 14-MeV) neutron radiation in a deuterium/tritium (DT) fusion reactor (see Fig. S.5). The development of baseline properties for unirradiated candidate materials is important; materials that exhibit favorable performance in the thermal, chemical, and stress environments of interest will serve as the basis for the radiation testing program.

The strategy for materials development involves an iterative process, evaluating the performance of candidate materials and then developing improved materials by modifying their composition, microstructure, or both. The performance of these improved materials must be measured and their potential for further improvements evaluated. A major difficulty in the materials-irradiation program is the current lack of a high-fluence 14-MeV-neutron irradiation test facility. Therefore, in the early stages, radiation effects in a fusion reactor must be deduced primarily from fission-reactor irradiation data, correlated by means of fundamental radiation studies.

S.2.6 Systems Design and Analysis

The systems design and analysis work supports the major program evaluation and decision tasks and guides fusion R&D

FIGURE S.5 Characterization of Tasks for Fusion Materials

toward practical products. The objectives of the activities in this area are (1) to ensure the development of practical fusion applications; (2) to complete preconceptual designs of major fusion

facilities; (3) to analyze critical issues and optimize development paths, (4) to identify and implement necessary safety, environmental, and licensing features for fusion development; (5) to plan and execute necessary R&D for remote handling equipment; and (6) to evaluate the potential of alternative (non-DT) fuel cycles (see Fig. S.6).

The systems design and analysis activities will provide the fusion program with important tools, data, and perspectives. These include the identification and resolution of critical issues that involve the interaction of plasma physics and technology, the maintenance of an engineering data base, and the setting of subsystem objectives based upon the identification of desired characteristics of commercial fusion applications with regard to economics, safety, and the environment.

S.2.7 Interactions among Program Areas

The preceding sections have characterized the fusion program in terms of six research areas. Of necessity, many of these program areas are closely coupled. Perhaps the most obvious interrelationship is that of the systems design and analysis area with all other program areas. Primarily through conceptual designs and planning studies, the systems design and analysis effort supports the major program evaluations, assists in trade-off assessments, and guides R&D programs toward practical products.

A strong relationship exists between plasma technology and progress in magnetic confinement. The needs of future confinement experiments determine the research objectives for plasma

FIGURE S.6 Characterization of Tasks for Systems Design and Analysis

technology. Conversely, with each decision on a future confinement facility, plasma technology limitations act as a significant constraint in specifying the attainable confinement parameters. The area of plasma-interactive components also couples

strongly with nuclear technology and materials, where the objectives of useful energy recovery and compatibility with other components are addressed.

Developments in nuclear technology are correlated with all other program elements. The design requirements for fusion nuclear systems are interdependent with and strongly affected by the configuration of the confinement concept, by requirements for plasma technology, and by such confinement-related parameters as magnetic field strength and fusion power production. The coupling between materials and nuclear technology is also strong; many of the key materials issues are for materials applications in nuclear components.

There is also a programmatic coupling among the burning plasma, plasma technology, nuclear technology, and materials areas through the long-burn facility. The features of this facility have not yet been established and are the subject of discussion for international collaboration. One option would provide testing for nuclear technology, as well as for confirmation of the physics involved in long-burn operations.

S.3 OVERALL PROGRAM LOGIC

Fusion research and development involves many parallel tasks, and the interrelationships among tasks are extensive. The decisions on facilities, schedules, and detailed objectives in each area typically depend on technical progress and future plans in several other program areas. The TPA has represented these interrelationships by means of a hierarchy of logic diagrams.

At the highest (most general) level in the hierarchy of logic diagrams is the Level 0 logic diagram, which shows all six research areas of the magnetic fusion program (see Fig. S.7). In the logic diagrams, research activities are indicated by horizontal lines. The diamond symbols, $\diamond$, are major decision/evaluation points, and the square symbols, $\square$, are major milestones that represent accumulations of data and understanding critical to the decisions and evaluations. Lower-level diagrams, presented in the later chapters, provide successively more detail on program elements and sub-elements.

The Level 0 logic diagram indicates the basic approach and major functional steps (described in Sec. S.2) that must be carried out to resolve the four MFPP key issues and support activities in plasma technology and systems design and analysis. There are two important points here. First, the hierarchy of logic diagrams shows activities in such a way that they are traceable directly from the detailed tasks on the lower-level diagrams to the resolution of the four key MFPP issues. Second, the activities are functional steps: the objectives to be accomplished and the nature of the required research facilities (but not the specific path of implementation -- facility, site, etc.) have been specified in the TPA. The functional steps may be carried out separately or, where possible, combined to minimize the needed facilities.

The approach to the two key physics issues -- improving confinement systems and understanding burning plasmas -- is an example of separate and parallel functions. Rapid progress can be made

(1) From class of present "moderately-developed" concepts

(2) From class of present "developing" concepts

FIGURE S.7 Level 0 Logic Diagram -- Magnetic Fusion Program Plan Overview

on the burning plasma issue by carrying out a separate first step in the near future with a short-pulse ignition experiment, to be followed later by a long-pulse burn demonstration. Near-term, short-pulse ignition experiments could be carried out only in a tokamak, because alternative concepts will not be ready in the near term. Over the long term, improved concepts for commercial fusion applications are expected from the development of advanced tokamaks and alternative concepts.

The combination of functional steps for the long-burn demonstration and nuclear technology testing is possible if appropriately aggressive goals for availability, fusion power density, and operating schedule are incorporated into a long-burn facility. Furthermore, a consensus has developed worldwide among fusion program leaders concerning the need for a multinational fusion project that combines these functional steps in the mid-1990s.

The strategy indicated in the Level 0 logic diagram retains flexibility for the implementation of programs within the various areas of research, but it also implicitly reflects assumptions about several major decisions in the program. Thus, the set of logic diagrams in the TPA gives a self-consistent picture of the MFPP strategy adopted, but some modifications would be necessary if significant variations in the assumptions about major decisions were to occur (see Secs. S.5 and S.7).

S.4 MAJOR DECISION POINTS

Many different decision sequences are possible, each of which might accomplish the functional steps shown in

the logic diagrams. The character of the program will depend heavily on the facilities selected at certain major decision points. Figure S.8 shows the major decision points associated with the Level 0 logic diagram (Fig. S.7).^{*} These decision points are described in detail in Tables S.4-S.10 at the end of the Synopsis. The major decision points are the following:

E_1 : Decide whether to proceed with a short-pulse ignition test in a compact ignition tokamak (1986-87).

E_2 : Decide which confinement concepts should proceed to integrated proof-of-principle tests (~1990-91).

E_3 : Decide on the approach to be followed regarding a long-burn demonstration (~1990).

E_4 : Decide on the approach to be followed regarding nuclear technology testing in the fusion environment (~1990).

E_5 : Decide on the approach to be followed in obtaining materials lifetime data in the fusion environment (~1990).

E_6 : Decide which confinement concepts should proceed to reactor-condition tests (~1997).

Decisions E_3 - E_5 are to be made at about the same time because the program issues are closely interrelated. A comprehensive assessment of fusion's potential, coupled to a decision to proceed with an integrated fusion facility, will occur around the year 2005.

^{*}In the Level 0 logic diagram, decision points E_2 and E_6 each occur twice to acknowledge the varying paces of development among concepts.

FIGURE S.8 Top Level Decision Points in the Magnetic Fusion Program

Abstracting the key decision points from the overall program logic is essential to the planning process, because the decisions are affected by such factors as overall budget levels and international collaboration, which are influenced but not solely determined by technical progress. One contribution of the TPA is to reveal interrelationships among the technical elements of the program and show how these interrelationships might be affected by various decisions made in its implementation. Although the program logic incorporates some flexibility in the sequence and combinations of functional steps, the logic also contains implicit assumptions concerning the outcomes from key decisions. Interrogating the program logic with various assumed outcomes from key decisions is a useful planning tool.

The large facilities associated with the major decision points are described in Tables S.11-S.16 at the end of the Synopsis. The facilities are:

1. Compact Ignition Tokamak (CIT)
2. Proof-of-principle test facility for confinement concepts
3. Engineering Test Reactor (ETR)
4. Nuclear technology test facility
5. Materials irradiation test facility
6. Reactor-conditions test facility for confinement concepts

Tables S.11-S.16 begin with a purpose/mission statement and list the key features and nature of each facility, together with estimated costs and schedules.

S.5 PROGRAM SCENARIOS

In describing the major program decision points listed in Sec. S.4 (and defined in more detail in Tables S.4-S.10), the TPA has identified more than one possible outcome of these decisions. Depending on the outcome of these decisions, several possible program scenarios can be identified.

It has not been possible for the TPA to examine all of the possible scenarios in detail. Rather, the TPA has adopted a reference scenario that it is believed largely reflects the current position of the U.S. Department of Energy. Furthermore, the technical credibility of this reference scenario is supported by the detailed technical information contained in this report.

The TPA has developed a framework for defining program scenarios that recognizes that two decisions to be made in the near future in the burning plasma program area, one on a short-pulse ignition demonstration (E_1) and the other on a long-burn demonstration (E_3), will involve near-term budgetary commitments significantly larger than those associated with decisions in other program areas. In presenting the program logic, therefore, it is appropriate to ask what impact various possible outcomes from these two important decisions might have on the program's implementation.

Figure S.9 shows one way to define the decision branch points. The figure portrays the decision on the short-pulse burn demonstration (E_1) as a "yes or no" decision and the subsequent decision on the long-burn demonstration (E_3) as positive but paced by differing considerations. The positive E_3 decision is appropriate, because some type of

FIGURE 3.9 Examples of Major Decision Branch Points, Based Primarily on the Burning Plasma Issue

long-burn demonstration must occur for fusion to advance.

A positive decision on a near-term, short-pulse ignition test is basically a strategy to study ignition and long-burn issues at two separate facilities. This permits the study of ignition as soon as possible in a device with the lowest near-term facility costs. The device must be a tokamak, because no other concept has a sufficiently developed data base to undertake an ignition test in the near term. A near-term ignition test would reduce the risk of the next step, the long-burn demonstration (LBD), and permit the LBD to incorporate engineering testing and thus fulfill the role of an Engineering Test Reactor (ETR) (option 1).

Another option, following a short-pulse ignition test, would be to delay the

decision on the LBD/ETR until after further concept improvement, as described in decisions E_2 and E_6 (option 2). The LBD/ETR might be based on an advanced tokamak or alternative concept. This option will result in a delay, compared with option 1.

A negative decision on a near-term ignition experiment implies a strategy to combine ignition and long-burn demonstration into one facility. Again, if this option is to be followed in the near future, the device must be a tokamak. Without benefit of a preceding, separate ignition test, the tokamak would probably have limited engineering testing capability (option 3). On the other hand, one could take a more aggressive approach, with its attendant higher risks, and incorporate an enhanced engineering testing capability to result (again) in an ETR (option 4). Finally, one could again

postpone the decision and await further developments in advanced tokamaks and/or alternative concepts (option 5).

Figure S.10 shows an example of the top path in Fig. S.9 and presents the reference scenario used in the TPA studies. The reference scenario in Fig. S.10 is wholly consistent with the Level 0 logic diagram and, therefore, with the detailed technical planning in the rest of the TPA effort. The reference scenario includes a short-pulse ignition experiment, the Compact Ignition Tokamak, and later a superconducting tokamak, called the Engineering Test Reactor. The ETR, which has sufficient availability and neutron wall loading for nuclear technology testing, can also meet the needs of a long-burn demonstration. The nature and timing of these facilities are consistent with current planning in the U.S. program and with recent international discussions that are an ongoing part of the Economic Summit Process.

It may be useful at this point to elaborate on the use of the terms short-pulse ignition, CIT, long-burn demonstration, ETR, and integrated fusion facility. The terms "short-pulse ignition" and "long-burn demonstration" refer to *functional steps* required to resolve the burning plasma issue. The reference strategy proposes to resolve the burning plasma issue in two separate facilities, the CIT and ETR. These are names given to specific *facility options* for resolving the functional steps. (As noted above, other facility options could possibly result from decisions E_1 and E_3 , such as carrying out the steps of short-pulse ignition and long-burn demonstration in the same facility.) Furthermore, the

ETR represents the particular facility option of combining the functional step of long-burn demonstration under the burning plasma issue with the step of integrated fusion testing under the nuclear technology issue (decisions E_3 and E_4). The decisions identified in Sec. S.4 are written in terms of functional steps, rather than in terms of particular facility options that result from these decisions.

The integrated fusion facility mentioned in the MFPP is also a functional step. The IFF represents the step to be taken after a favorable assessment of fusion's economic, safety, and environmental potential. It is the beginning of the commercialization phase of fusion and may be a single facility (perhaps a demonstration power reactor) or a set of facilities not yet defined in detail.

Confinement systems concept development proceeds through the integrated proof-of-principle (E_2) and reactor-condition (E_6) stages in parallel with resolution of the other three issues. The reference scenario for concept development can be specified in some detail only for the tokamak concept, because other concepts are at an earlier stage of development. The number and timing of concept development tests for nontokamak confinement concepts can only be decided after progress in this research is reviewed in the context of overall program need. The criteria for determining program need will be defined by the rate of tokamak scientific and technical progress and by the continued assessment of the attractiveness of the tokamak-based power system flowing from this progress.

FIGURE S.10 Level 0 Logic Diagram Depicting Reference Scenario

The reference scenario also includes a materials test facility using 14-MeV neutrons. This facility provides sufficient yearly fluence for testing materials properties at a significant fraction of the full projected lifetime. A separate test facility is needed because a high yearly fluence would not be expected from an ETR, which would combine a technology/component-testing mission and a long-burn demonstration for the first time.

The reference scenario is characterized by a number of features:

- Ignition and short-burn physics are explored in the near term.
- Approaches to burning plasma experiments and to confinement concept improvements can be separately optimized.
- CIT input during the ETR's construction phase will provide greater confidence in the operation of the ETR. This should result in cost savings because of reduced learning periods during the initial operation of the ETR.
- Information is provided on 14-MeV-neutron effects on lifetime or performance at neutron fluences beyond the capability of the ETR and simulation/modeling techniques.
- If a concept other than a tokamak (or a similar toroidal system) appears more suitable for commercial fusion applications, at least one more ignition or long-burn experiment could be necessary.

- The higher cost of a device (ETR) designed for both long-burn demonstration and nuclear technology testing is built into this scenario.

S.6 SUMMARY OF RESOURCE REQUIREMENTS

A preliminary estimate has been made of the resources required to carry out the full set of R&D activities described in this report. Although the TPA results provide a framework for estimating the overall schedule and total cost of assessing the commercial potential of fusion, a number of deficiencies contribute to the uncertainties involved in such estimates at this time. First, a single worldwide view of the timescale and pathway for commercial fusion does not exist. Second, the cost projections obtained as part of the TPA do not include input from outside the U.S. fusion community. Third, the ground rules used for these projections need to be applied more uniformly among TPA components. Finally, an iterative review process for this resource estimate has not yet been possible. Nevertheless, the information gathered by the TPA is sufficient in quantity and quality to present a broad view of the resources required for a full assessment of the commercial potential of fusion. This estimate is comprehensive in scope in that all tasks required to achieve the overall MFPP goal are covered, although it is acknowledged that only a portion of the work would be carried out by the U.S.

The outcomes of the various decision points represent multiple parallel paths, so a particular set of outcomes of the

decision points has been adopted in carrying out the resource estimate. The particular scenario used for the resource estimate is the reference scenario discussed in Sec. S.5. Thus, the estimate covers the period from 1987 until the overall assessment and decision on an integrated fusion facility in 2005. The estimate does not cover the design and construction of an IFF. No effort has been made at this time to estimate the resources for any alternative scenario.

The TPA has not considered the trade-offs between the degrees of redundancy and focus appropriate to an international collaborative program. In addition, the overall program costs embody intrinsic uncertainties associated with incomplete definition of major projects (especially the ETR) and with the winnowing, optimization, and/or combination of confinement concepts, a process that does not translate uniquely into costs. Cost estimation was performed at the program area, element, and subelement levels, except for confinement systems; for these, the individual confinement concept research programs served as the basis for accumulating costs. Total program resource requirements have been assembled by accounting for operating/equipment funds and construction funds for each program area and element. Judgments were required regarding the narrowing of options in such areas as confinement concepts, blanket types, and materials.

Although considerable uncertainty exists in the details of these estimates, and further work and integration remains to be done in reviewing and refining individual estimates, some general observations can be made. The overall

resource requirement for the period 1987-2005 has been estimated to be approximately \$20 billion. Although operating costs are nearly constant at about \$800 million annually, the overall funding is estimated to increase to about \$1.5 billion in the mid-1990s because of construction costs associated with the facilities that are most crucial to providing definitive information by 2005.

Throughout the period up until about 2000, confinement systems constitute the largest component of operating costs. In the 2000-2005 period, emphasis on the operating budget for confinement systems is reduced, commensurate with an increased emphasis on tests of burning plasmas, the development of nuclear technology, and the resolution of key materials issues. Indeed, the fraction of the budget devoted to technology issues must increase severalfold to achieve the desired results. This increase in the technology budget should begin in 1988. The overall construction budget is estimated to be about \$6 billion (out of the \$20 billion); the largest item, the ETR, is assumed to cost \$3 billion. This construction cost estimate for the ETR is very preliminary at this time. Work is beginning in the U.S. on ETR approaches that may have a substantially reduced cost.

S.7 COMMENTS ON THE TPA RESULTS AND RECOMMENDATIONS ON FUTURE WORK

The need for the TPA as a continuing activity has been recognized by the senior fusion experts who prepared this report, by expert reviewers charged by DOE's Magnetic Fusion Advisory Committee, and by the Office of Fusion

Energy (which chartered the TPA). Much has been accomplished during the TPA's first 18 months, but important work remains to be done. This section briefly discusses some of the limitations encountered in the planning activity and the perspective gained by the group in formulating the TPA's approach; the section concludes with recommendations for future work.

S.7.1 Foreign and International Efforts

International collaboration in fusion is already extensive and is increasing in importance; there is a commensurate need to place the U.S. program within the larger context of an evolving worldwide program. International interest in the type of planning carried out by the TPA is anticipated, and this report is expected to serve as a useful example to facilitate international discussions on future comprehensive technical planning.

The TPA's mission has been to formulate a comprehensive view of the technical tasks that must be accomplished for fusion development over approximately the next twenty years, without regard to specific roles for the U.S. and other nations. In formulating its approach, the TPA has tried to reflect the evolving world consensus on international goals in fusion. Its approach identifies tasks as functional steps, rather than assigning efforts to specific facilities, sites, or national programs. Future facilities are characterized by technical features, not by location. Where ongoing work is cited, work outside the U.S. has been included. (There are, undoubtedly, omissions and uneven treatment on these

matters, particularly in references to work outside the U.S.) Finally, the program logic is consistent with the mission and schedule for a multinational ETR in the mid-1990s. Such an effort has been discussed by the U.S., the European Community, Japan, and Canada through the Economic Summit Process; with the USSR through the Reagan-Gorbachev Summit; and with the USSR, Japan, and the European Community through the International Atomic Energy Agency (IAEA).

In noting the TPA's efforts to advance a comprehensive view, it seems appropriate to acknowledge its limitations as well. Because the TPA has not yet received contributions or reviews from outside the U.S., this report reflects only a U.S. interpretation of fusion development. The framework for formulating the technical tasks is the U.S. Magnetic Fusion Program Plan; although the MFPP is broad in strategy, it is still a national program plan.

S.7.2 Describing Concept Development

A general consensus has emerged in the TPA community on the following points: (1) an improved confinement concept suitable for an economical, safe, and environmentally attractive fusion reactor can be developed; (2) the pathway for necessary progress will include a significant synthesis of ideas, innovations, and a growth of scientific understanding; and (3) a winnowing of concepts on the basis of technical merit will lead to a limited number of preferred concepts. Finding appropriate language and graphical techniques to incorporate these ideas into the TPA's planning approach, however, has proven difficult.

The TPA participants have faced an inherent dilemma. On the one hand, the strong empirical flavor of plasma physics and the fact that tangible progress occurs in terms of specific confinement concepts favor concept-by-concept planning. On the other hand, an approach to planning that emphasizes only individual concepts is likely to erroneously portray the program as a contest between independent concepts and undervalue the cross-fertilizing syntheses of ideas in magnetic confinement. Such an approach also tends to ignore the constructive, unifying process through which resolution of the technical issues contributes to a predictive capability in plasma physics.

Sufficient progress toward a comprehensive understanding of plasma physics has been achieved that a predictive capability appears to be a reasonable goal. Such a capability will be an extremely powerful tool. To the degree it can be synthesized from the aspects of plasma behavior exhibited in various confinement devices prior to the need for reactor-like demonstration devices, a predictive capability will make the research program more efficient and improve the end product.

The TPA has tried to incorporate both of these aspects of confinement development by using a common, issue-based description for the future development of each concept. More information is presented in Chapter 1, where the introductory text and accompanying figures give historical examples of linkages between concepts and of cross-fertilization (cases where an idea from one concept was applied elsewhere to resolve a technical issue). Despite considerable effort in this area,

the TPA team is still not wholly satisfied with the approach for describing confinement systems development.

S.7.3 Comparisons among Concepts

It has not been the TPA's function to rigorously compare and rank the relative merits and probabilities of success of confinement concepts. Each concept has been considered on an individual basis in identifying its technical issues, as well as its required facilities and resources. However, the detailed issues associated with each particular concept have been described in terms of a set of generic plasma physics issues (e.g., macroscopic equilibrium and dynamics). A common definition of steps in confinement systems development (e.g., integrated proof-of-principle experiments) has been devised to serve as the criterion for decisions on development of future concepts. Thus, the TPA has produced a way to facilitate concept comparison in DOE's future implementation decisions.

When collective comparisons of confinement concepts are undertaken, a difficult and important element in the process will be some method, as yet not developed, for assessing the likelihood of success for each particular confinement concept and development scenario. The relationship between the number of available facilities and the likelihood of significant progress will be important in this comparison. It must be recognized that major differences exist among the facilities available for each concept. These differences are most pronounced when comparing the tokamak with other concepts.

The historical development of the tokamak has led to a worldwide program

based on several large experiments, together with several smaller supporting experiments. Since it is not necessary to rely on only one device to test new ideas and resolve critical issues, the probability of resolving the issues of the tokamak and improving its reactor potential is increased. In their respective logic diagrams, the other confinement concepts generally project development scenarios having fewer steps, far fewer worldwide facilities for each step (sometimes only one), and shorter time periods between steps. Thus, the probability of resolving issues on the indicated timescales must be judged to be higher for the tokamak than for the other concepts.

S.7.4 Development of Scenarios

Planning scenarios like the one considered in Sec. S.5 are very useful for representing program options and the possible consequences of key decisions. The development of planning scenarios is a likely area of continuing effort for the TPA.

The TPA's consideration of planning scenarios emerged late in the first year of work, as a part of efforts to enhance the value of the product as a decision-making tool. The TPA was in general agreement on the value of a set of scenarios that would show the potential impacts of key decisions. In the time available, only the reference scenario presented in Sec. S.5 could be supported by detailed technical planning.

The emphasis, in terms of detailed planning support, given to the reference

scenario does not signify an endorsement of this scenario. The function of the TPA in creating scenarios is to correlate the probable flow of technical progress with key decisions. The product from the technical experts is a set of scenarios that, taken together, show how the program could respond to possible changes imposed by external circumstances, unexpected innovations, or major technical problems.

An important element in the development of a set of scenarios is the consideration of such intangible factors as optimism and programmatic risk. Implicit optimism or the acceptance of high risk can significantly alter the perceived value of a scenario (e.g., tasks and schedules), but these factors are difficult to quantify; the impact of intangibles as separate factors will seldom be obvious. The general sense of the TPA's reflections on this subject is that risk and optimism should not be treated as variables, but should receive even-handed treatment across all scenarios, with some recognized limitations.

Another aspect of scenario development that needs further work is the description of decisions, particularly with respect to criteria and potential outcomes. An effort has been made in this report to include both the programmatic "outcome," in terms of the value of the added capability of the new facility, and the "consequences," in terms of the impacts in other program areas. More work in this area is needed to expand the individual descriptions and to treat the collective information in a more thorough manner.

S.7.5 Achieving System and Component Reliability

Commitment to the goal of high availability in future fusion facilities (e.g., an ETR suitable for nuclear technology testing) implies a significant engineering effort on component reliability and maintenance. The general goal of developing high-reliability components appears in the requirements for certain facilities, particularly an ETR or other devices for nuclear technology testing. Because fusion technology is in an exploratory research phase, engineering development does not yet constitute a significant fraction of the program. This is reflected in the treatment of reliability in the TPA, which is not adequate.

S.7.6 Planning for Technology Transfer

The transfer of technology to industry is one of three strategic objectives stated in the MFPP. The need to plan for technology transfer is certainly recognized; however, the strategy and means for achieving this objective are closely linked with the administrative mechanisms for implementing the program. Nevertheless, the TPA has begun incorporating these issues into the planning process. Technology transfer is discussed further in Chapter 7.

S.7.7 Recommendations for Future Work

Efforts should be continued in three areas discussed earlier in this section: (1) development and analysis of alternative scenarios, (2) improved description of planning for confinement systems, and (3) improved representation of work outside the U.S.

The suggested scope for the development of scenarios should probably include four to six scenarios and related analysis of the program logic, reformulation of alternative logics, and expansion of the decision descriptions (particularly in the areas of "Decision Criteria" and "Outcomes and Consequences"). A good start toward obtaining the necessary information has been made, but a set of scenarios -- rather than simply a reference scenario -- is needed. The formulation of a set of scenarios should provide a useful working basis for expanding the effort on decision criteria and addressing the subject of programmatic risk.

The development of alternative scenarios, and perhaps corresponding alterations in the program logic, may help in the formulation of a better planning description for the confinement systems area. Effort has already been expended in this area, but now a more complete planning basis for the assembled confinement systems planning elements is available, together with the lessons learned from one iteration.

Modification of the planning documents will undoubtedly be necessary as a result of contributions from foreign programs. Use of the TPA results as an input for international discussions is expected. This will require subsequent analysis and reformulation.

There has been general acknowledgment from within and outside the TPA of the value of a continuing effort. The OFE has indicated its intention to continue the activity in FY 1987, and DOE has asked the Magnetic Fusion Advisory Committee to review the TPA and make recommendations on the value and use of this planning information.

TABLE S.4 Decision E₁ On a Short-Pulse Ignition Demonstration

Statement of Decision

To proceed with an ignition experiment, based on the tokamak concept, in which the primary objective is to demonstrate a fusion system that is energetically self-sustaining.

Decision Criteria

Demonstrate nondisruptive moderate- β operation in a tokamak.

Achieve equivalent (non-DT) breakeven and determine transport scaling at these conditions.

Demonstrate high-power heating, fueling, and impurity-control methods.

Demonstrate relevant plasma technologies.

Establish role of ignition experiment in fusion-reactor development.

Project construction and operating cost.

Project total concept development path and cost.

Consider impact on overall fusion physics and technology development cost and schedule.

Examine status of alternative options.

Types and Sources of Information

Existing large tokamak facilities, including equivalent breakeven experiments. Tokamak experimental tests of additional concepts to be utilized in ignition experiment.

Plasma supporting activities.

Plasma technology tests of all needed systems.

Preconceptual designs of the experiment.

Fusion-community assessments of role of ignition experiment in fusion-reactor development.

Outcomes and Consequences of Decision

Favorable assessment and achievement of ignition objective will lead to early insight into the physics of self-sustained, burning plasma. This knowledge would reduce the risk and increase the productivity of the next step, a long-burn demonstration (LBD).

If no short-pulse ignition experiment is built, then most of the burning plasma physics issues will have to be explored for the first time in the LBD, which will likely be a less ambitious step and entail more conservative design margins.

TABLE S.5 Decision(s) E₂ for Progressing to Integrated Proof-of-Principle Experiments

Statement of Decision

To proceed with integrated proof-of-principle experiments in a few alternative concepts.

Decision Criteria

Determine the macroscopic equilibrium and gross stability boundaries.
Explore confinement scaling for $n\tau_E \sim 10^{17}-10^{18} \text{ s/m}^3$.
Develop heating and distribution control methods.
Develop methods for impurity control, fueling, and utilization of injected beams.
Assess need for new plasma technologies.
Determine reactor potential relative to other concepts.
Assess potential contributions to generic plasma science.
Project construction and operating cost.
Project total concept development path and cost.
Examine status of alternative options.

Sources of Information

Confinement system concept tests.
Plasma supporting activities.
Relevant plasma technology tests of critical systems.
Conceptual reactor designs to indicate needed parameter achievements.
Preconceptual designs of the experiment.
Development pathways analyses.

Outcomes and Consequences of Decision

Favorable assessments and achievement of the primary objectives of integrated proof-of-principle experiments will lead to one or more alternative concepts for progression to reactor-level-condition experiments in time for the 2005 fusion assessment.
If no alternative concepts progress to integrated proof-of-principle experiments by 1990-91, then it is not likely that such concepts will be adequately demonstrated to be seriously considered in time for the 2005 fusion assessment.

TABLE S.6 Decision E₃ on Long-Burn Demonstration (LBD)*Statement of Decision*

Choose the optimum approach for demonstrating plasma long-burn/steady-state operation. The approach will include selection of the confinement concept and the combining or separating of the physics and nuclear-technology testing tasks.

Decision Criteria

Plasma physics understanding of alpha effects and impurity control is adequate to provide realistic estimates for long-burn plasma behavior.
Plasma technology status is adequate to provide components for heating, fueling, and impurity control.
Nuclear technology and materials data bases are adequate to provide reliable first wall, shield, high heat flux, tritium processing, and other components.
Project capital cost.
Project operating cost, including maintenance, power supplies, and tritium costs.
Length of plasma burn, magnitude of duty cycle, and capability for steady-state operation.
Usefulness to component reliability and nuclear technology testing (e.g., neutron and surface heat loads, testing area, and availability).
Usefulness to materials testing (neutron fluence).
Impact on time schedule to demonstrate competitiveness of fusion energy.

Types and Sources of Information

Confinement systems provide physics data base and operating experience.
Plasma technology provides data base for heating, fueling, and high heat flux components.
Nuclear technology provides information on the operating requirements, engineering data base, and design of nuclear subsystems. The desired operating conditions and test module designs are also provided.
Materials research provides information on materials properties and projected lifetime. The desired testing conditions for materials development are also provided.
System studies assess the role of the long-burn demonstration step in fusion power development.

Outcomes and Consequences of Decision

The LBD is a steady-state (or long-burn) tokamak with extended capabilities for physics and nuclear-technology testing capabilities, thus eliminating the need for a separate technology facility (see E₄). This option has been identified as an Engineering Test Reactor.

TABLE S.6 (Cont'd)

The LBD is a tokamak that emphasizes physics testing, with little or no technology testing capability. This means that a separate technology facility would be needed (see E₄).

The LBD is based on a confinement concept other than a tokamak, in which case the date of construction will be delayed.

TABLE S.7 Decision E₄ on Fusion Testing of Nuclear Technology

Statement of Decision

Select the optimum approach for testing nuclear technology and demonstrating fuel self-sufficiency and attractive energy conversion in the fusion environment.

Decision Criteria

Achieve physics understanding to provide sustained plasma burn at required power densities.

Develop plasma technology, nuclear technology, and materials data base sufficient to construct and operate reliable components at desired performance.

Conduct nonfusion testing of materials and nuclear components sufficient to construct test modules and determine if attractive options exist.

Project capital cost.

Project operating costs, including those for maintenance, power supplies, and tritium supply.

Assess usefulness to nuclear technology testing (e.g., neutron and surface heat load, reliability and availability, testing-surface area and size).

Assess usefulness to materials testing (test surface area, neutron fluence, flux, and spectrum).

Determine impact on time schedule to demonstrate attractiveness of fusion energy.

Types and Sources of Information

Confinement systems area provides physics data base for reproducible plasma with desired power density.

Plasma technology provides data base and operating experience for fueling, heating, and impurity control.

Nuclear technology provides information and engineering data base to design and construct nuclear subsystems.

Materials research provides information on materials properties, operating limits, and projected lifetime.

Nuclear technology provides information on the testing requirements in terms of major device parameters and design features.

System studies results on reactor needs and development-pathway analysis.

Outcomes and Consequences of Decision

Nuclear-technology fusion testing is performed in a device that combines physics and technology testing, thus eliminating the need for a separate long-burn facility (E₃). This option has been identified as an Engineering Test Reactor.

Nuclear-technology fusion testing is performed in a dedicated fusion device optimized for nuclear testing requirements; plasma does not need to be ignited, and the fusion power can be < 100 MW.

TABLE S.8 Decision E₅ on Fusion Testing for Materials

Statement of Decision

Choose the optimum approach for irradiation testing of advanced materials for fusion.

Decision Criteria

Lifetime goals and performance requirements defined by systems studies and nuclear technology testing.

Information from baseline materials property data, fission reactor irradiations, nuclear technology tests, and systems studies indicates sufficient promise for those materials that are candidates for 14-MeV-neutron irradiations.

Importance of neutron spectrum effects on materials performance.

Ability of small specimen tests to provide required materials property data.

Projected physics, technology, cost, and schedule of candidate 14-MeV-neutron test facilities.

Comparison of alternative methods of achieving 14-MeV-neutron data (e.g., flux, volume, availability).

Sources of Information

Baseline materials property data.

Irradiation tests from fission reactors.

Results of nuclear technology tests and systems studies to identify relative importance of and key problems with candidate materials and range of operating parameters.

Systems studies and nuclear and plasma technology tests to identify desired lifetime goals.

Analysis of alternative pathways.

Outcomes and Consequences of Decision

A separate high-flux, high-availability, small-volume 14-MeV-neutron facility is constructed for irradiation testing of selected materials.

Materials irradiation testing with 14-MeV neutrons is conducted in a moderate flux and fluence fusion nuclear technology facility along with fission reactor irradiations. Additional risk is introduced by lack of high-fluence and lifetime data.

Materials irradiation testing with 14-MeV neutrons is conducted in a long-burn device at low fluences and in fission reactors at high fluences. In this case, the lack of lifetime predictions based on high-fluence 14-MeV data is a deferred risk.

TABLE S.9 Decision(s) E_G for Progressing to Reactor-Level-Condition Experiments

Statement of Decision

To proceed with reactor-level conditions in one or more alternative concepts.

Decision Criteria

Assess β scaling.

Demonstrate satisfactory confinement scaling for $n\tau_E \sim 10^{18}\text{--}10^{19} \text{ s/m}^3$.

Demonstrate high-power-level heating.

Develop efficient impurity control, fueling, and beam methods.

Assess pulse-length regimes.

Develop plasma technologies that are reliable.

Demonstrate reactor attractiveness compared with more developed concepts.

Project construction and operating costs.

Project total concept development path and costs.

Assess status of alternate options.

Sources of Information

Integrated proof-of-principle experiments.

Plasma supporting activities.

Plasma technology tests of all needed systems.

Integrated reactor design studies that indicate needed plasma regimes and technologies for an attractive fusion system.

Outcomes and Consequences of Decision

Favorable assessment(s) and achievement(s) of the primary objectives of reactor-level-condition experiments will lead to one or more viable alternative concept(s) at the time of the 2005 fusion assessment.

If no alternative concepts progress to reactor-level-condition experiments, then no such concepts will have been experimentally demonstrated to be viable options at the time of the 2005 fusion assessment.

TABLE S.10 Decision Description — Fusion Assessment and Integrated Fusion Facility***Statement of Decision***

To make a thorough assessment of the economic, safety, and environmental potential of fusion energy.

To proceed with an integrated fusion facility (IFF), which begins the process-development and commercialization phase of fusion energy.

Decision Criteria

Define at least one competitive commercial application of fusion energy.

Demonstrate at least one confinement concept for a commercially competitive fusion application.

Demonstrate a predictive plasma science capability.

Demonstrate a long-burn, ignited DT plasma that is compatible with the leading confinement concept.

Develop the nuclear technology for commercially competitive fusion applications and, in particular, for the leading confinement concept.

Show that materials can be developed that will enhance fusion's economic and environmental features.

Demonstrate the assured safety features.

Demonstrate that the required plasma technology can be developed for a commercially competitive application of fusion energy.

Define an affordable process-development and commercialization phase for fusion.

Demonstrate how technology transfer to U.S. industry can be accomplished.

Sources of Information

Systems studies will define commercial applications and assess overall economic/safety considerations, future development costs, and technology-transfer status.

The leading confinement device will achieve reactor-level plasma conditions (key attributes: $\langle B \rangle$, n_T , pulse length).

Same device or another will demonstrate long burns at or near ignition conditions (key attributes: Q , pulse length).

Theory and computational techniques will result in a predictive plasma physics capability (key attribute: degree of understanding).

Nuclear technology will be demonstrated in a prototypical fusion environment, with support from nonnuclear and fission test facilities (key attributes: predictability, tritium breeding and recovery, wall loading, reliability, and safety).

Materials will be tested in a 14-MeV-neutron environment, together with substantial fission-reactor testing (key attributes: fluence, lifetime, activation, and predictability).

Plasma technology will be demonstrated in confinement devices, long-burn demonstrations, and supporting facilities (key attributes: impurity control lifetime, magnet and heating system cost).

TABLE S.10 (Cont'd)

Outcomes and Consequences of Decision

A favorable assessment and achievement of major milestones will lead to decision to proceed with IFF.

An unfavorable assessment will result in continued research on key issues responsible for the unfavorable assessment.

TABLE S.11 Compact Ignition Tokamak (CIT)

Purpose/Mission

The mission of CIT is to realize, study, and optimize fully ignited plasma discharges.

Nature of Device

CIT is designed to be sited at a facility with existing equipment -- e.g., adjacent to the present Tokamak Fusion Test Reactor (TFTR) at the Princeton Plasma Physics Laboratory.

Key Features

CIT is a high field (10-12 T), high plasma current (10 MA), compact (major radius 1.3-1.5 m) tokamak. It features all-copper toroidal and poloidal field magnets; it relies on inertial cooling starting from a liquid nitrogen temperature at the beginning of each pulse. Each full parameter pulse is about 3-5 s. A total of 3,000 full parameter DT pulses and 50,000 partial parameter pulses are planned. Auxiliary heating with ICRH is provided (10-20 MW). The total fusion power is 300-500 MW.

Costs and Schedule

The new capital expense (Plant and Capital Equipment -- PACE) associated with the CIT project is \$285 million; this includes \$223 million in new funds for capital equipment and \$62 million as contingency. This estimate assumes the use of a significant amount of equipment and facilities in existence. The escalated total cost over the construction period of 1988-1992 is \$329 million.

A five-year design and construction schedule is assumed for the new PACE funds. Annual operating costs are estimated at \$72 million. The operating period is expected to be about 10 years.

TABLE S.12 Proof-of-Principle Test Facility for Confinement Concepts

Purpose/Mission

Define scaling laws and achieve parameters sufficient for confident extrapolation to reactor-level conditions.

Nature of Device

Integrated proof-of-principle experiments generally focus on testing most of the critical issues simultaneously. More emphasis is placed on parameter achievement than is done for confinement system concept tests.

Key Features^a

Typical features of an integrated proof-of-principle test facility include β values similar to those required for a reactor, $n_i \sim 10^{18}$ - 10^{19} s/m³, plasma temperatures of a few kiloelectron volts, long pulses (if required for a reactor), and effective impurity control.

Costs and Schedule

Total PACE costs for each facility are estimated to be about \$100 million, with an operating budget of \$10-20 million/yr. Construction times are estimated to be three to four years.

^aDetailed parameters will vary from concept to concept.

TABLE S.13 Engineering Test Reactor (ETR)

Purpose/Mission

The mission of the ETR is to demonstrate the design integration and operation of an ignited, long-burn tokamak and to demonstrate the integrated performance of nuclear components, including the ability to achieve fuel self-sufficiency.

Nature of Device

ETR-like designs are under development by the European Community, Japan, the USSR, and the U.S., both individually and jointly in the INTOR Workshop. The ETR will be a new facility and will have the potential to be an internationally sponsored undertaking.

Key Features

Candidate devices based on studies in the U.S. and worldwide suggest that the device would be a tokamak somewhat larger than TFTR or JET (Joint European Torus), would use superconducting toroidal and poloidal field magnets, would operate with long pulses or steady state, and would achieve a final availability of about 30%. The ETR would serve as a test facility for blankets, tritium production, plasma engineering components, and advanced reactor concepts.

Costs and Schedule

No detailed cost estimate for an ETR device has been developed. Estimates have been made by previous study groups. The design and PACE costs are estimated to be about \$3 billion. The estimated design and construction schedule is about eight years. Annual operating costs and duration have not been estimated for the ETR. Based on TFTR experience, it is estimated that operating costs would be about \$150 million. The operating period would be 10-15 years.

TABLE S.14 Nuclear Technology Test Facility

Purpose/Mission

Provide adequate environment for fusion testing of nuclear components.

Nature of Device

Any confinement concept that can provide a prototypical fusion environment with adequate test volume ($5-10 \text{ m}^3$) can serve as the base facility. Options identified include tokamaks and mirror devices.

Key Features

<u>Major Parameters</u>	<u>Parameter Values</u>
Neutron wall load	$1-2 \text{ MW/m}^2$
Surface heat load	$0.2-0.5 \text{ MW/m}^2$
Plasma burn time	1000 s (steady state)
Plasma operating mode	Driven or ignited
Availability	30-50%
Fluence	$2-3 \text{ MW}\cdot\text{yr/m}^2$
Test surface area	$5-10 \text{ m}^2$
Fusion neutron power	< 100 MW

Major Design Features

Access to insert and remove test elements.

Sufficient space external to device to provide ancillary equipment supporting testing elements.

Component design compatible with test program.

Cost and Schedule

Options (based on a limited amount of design effort) have been defined. The PACE cost is on the order of a billion dollars.

Construction time would probably be five to six years.

TABLE S.15 Materials Irradiation Test Facility

Purpose/Mission

Provide capability for testing candidate fusion materials to high fluences (> 100 dpa damage levels) in a high-energy (14-MeV) neutron environment.

Nature of Device

The high-flux materials irradiation test facility must provide a fusion-like radiation environment (14-MeV neutrons) and achieve fluences in a suitable volume (> 10 cm³) sufficient to evaluate the lifetime potential of candidate materials.

Major Design FeaturesRequired Features

Neutron energy	14 MeV (dominant energy)
Minimum test volume/flux	10-100 cm ³ at 5-10 x 10 ¹⁴ /cm ² .s
	Plus
	500-1000 cm ³ at 1-2 x 10 ¹⁴ /cm ² .s
Irradiation condition	Steady state with 50-75% duty factor
Temperature	Measurement/control at 100-800°C
Environment	Controlled chemical environment

Desirable Features

Larger high-flux test volume
 In-situ mechanical testing
 Variable chemical environment
 Time-varying temperature control

Costs and Schedule

The PACE cost for minimum test volumes (point source) is estimated to be \$150 million. The test volume could be increased by a factor of 5-10 for a 50-60% cost increment. Construction is expected to take four years.

TABLE S.16 Reactor-Condition Test Facility for Confinement Concepts

Purpose/Mission

Achieve reactor-level plasma conditions and demonstrate burn control, ash removal, and other necessary techniques.

Nature of Facility^a

Reactor-level plasma experiments will sustain fusion-reactor-relevant parameters for sufficient time to test the plasma physics and technology necessary for advanced concepts. These experiments will function as the transition from a plasma physics to a technology focus. A major question will be whether tritium burning is required.

Costs and Schedule

The PACE cost of each reactor-condition test facility could range from \$100 million to \$300 million or more, depending on the concept. Operating costs might range from \$50 to 100 million/yr. Typical construction times might be four to five years.

^aDetailed parameters will vary from concept to concept.

The development of magnetic confinement systems for commercially attractive fusion reactors is one of the four key technical issues in the MFPP. The overall objectives for the confinement systems issue are as follows:

- Demonstrate reactor-level plasma conditions for commercially attractive confinement concepts.
- Develop a sufficient predictive capability to design and optimize fusion systems for commercial applications. (The desired predictive capability includes both empirical models, derived from experimental data, and theoretical models.)

Neither of these two objectives is sufficient by itself. Both are important in developing attractive concepts backed by an adequate knowledge base for a realistic assessment of fusion.

The confinement systems issue has been divided into various issues and subissues, which are summarized in this chapter. The status of confinement systems is discussed in detail in terms of these issues in a companion report (Ref. 1).

The TPA plasma science strategy (based on the MFPP and the TPA mission statement) is to do the following: (1) use a variety of concepts to address the magnetic confinement systems issue and (2) use the confinement concept that is most highly developed at present (tokamak) to address the burning plasma issue (see Chapter 2). This two-pronged approach is needed because only the tokamak concept can presently be extrapolated with confidence to reactor-level burning plasma conditions, while other confinement concepts, including advanced tokamaks, offer the possibility of more attractive fusion applications.

The different stages of development for magnetic confinement systems being considered for the fusion assessment date of 2005 are characterized in Fig. 1.1. For about the next decade, the emphasis is on concept development and plasma science development in connection with advanced concepts at reduced parameter levels; thereafter, tests will be required at reactor-level plasma conditions. Complementary experiments with modest parameter levels will also be required in the mid-1990s to mid-2000s to develop the necessary predictive plasma science capability.

The development of confinement systems is further elucidated in Fig. 1.2.

FIGURE 1.1 Characterization of Magnetic Confinement Systems Tasks

FIGURE 1.2 Confinement Systems Development

The evaluation points E_2 and E_6 are the same as those in Fig. S.7 and are described in detail in Tables S.5 and S.9. As indicated, concepts for commercial fusion applications are to be developed and demonstrated by about the year 1997. Thereafter, concept development would split into two branches. Some present "moderately developed" concepts would proceed to reactor-level tests in time to directly influence a fusion assessment in 2005. Some concepts, with perhaps a potential for further reactor improvements from the class of present "developing" concepts, might also be explored at lower parameter levels to demonstrate their viability in time to influence a fusion assessment in 2005. These advanced concepts might proceed to reactor-level tests at a later date.

The magnetic confinement systems issue has been subdivided into the following four generic issues and one composite issue:

1. **Macroscopic equilibrium and dynamics (*Macroscopic*):** How can the plasma be contained in a macroscopic sense?
2. **Transport (*Transport*):** How are heat and particles lost from the plasma when it is in a well-behaved macroscopic state?
3. **Wave-plasma interactions (*Wave*):** How can externally powered waves be used to produce non-Maxwellian distributions and plasma heating, and how can plasma stability be maintained on the fine scale in velocity-space?
4. **Particle-plasma interactions (*Particle*):** What is the nature of the interaction of the plasma and its nonionized environment?
5. **Composite system considerations (*Composite*):** How can a magnetic confinement concept be integrated and optimized over all of the relevant confinement system and burning plasma issues (see Chapter 2)?

These five issues have been further divided into subissues, as follows:

<i>Macroscopic</i>	<i>Transport</i>	<i>Wave</i>
Pressure-driven effects	Perpendicular confinement	Wave heating
Current-driven effects	Parallel confinement	Non-Maxwellian effects
Kinetic effects		
<i>Particle</i>	<i>Composite</i>	
Impurity control	Configuration optimization	
Fueling	Profile optimization	
Neutral-beam injection	Pulse-length optimization	

These subissues, together with objectives for resolving them and attributes for gauging progress, are developed in detail in a companion document (Ref. 1).

The four generic fundamental issues (*Macroscopic*, *Transport*, *Wave*, and *Particle*) have as their objective the development of an understanding and predictive capability for their respective parts of plasma science. In addition, together with the composite issue (*Composite*), they have the following plasma parameter achievement objectives:

- *Macroscopic* -- Demonstrate macroscopic control of reactor-level pressure in magnetically confined plasmas ($\langle \beta \rangle \gtrsim 0.1$).
- *Transport* -- Achieve reactor-level magnetic confinement of plasma energy and particles at fusion temperatures ($T_i n \tau_E \gtrsim 3 \times 10^{21} \text{ keV} \cdot \text{s/m}^3$).
- *Wave* -- Heat and control plasma distribution functions to degree necessary for fusion reactor conditions ($T_i > 10 \text{ keV}$).
- *Particle* -- Demonstrate adequate impurity control, fueling, and neutral beam heating for fusion reactor conditions.
- *Composite* -- Develop an attractive fusion system with optimized configuration, profiles, and pulse length.

Figure 1.3 shows the evolution and interrelationship of these five issues, together with the two burning plasma issues (alpha particle effects, burn control and ash removal; see Chapter 2). There is no single critical path line on this diagram, because all the issues must be addressed and their objectives achieved to develop an attractive magnetic confinement system.

Progress toward meeting these objectives has come about through experiments in particular magnetic confinement systems. There is considerable cross-fertilization and interrelationship among the evolutionary paths of the various concepts. The interrelationship among the plasma science issues for the different concepts is often quite close. This point is illustrated for the macroscopic equilibrium and dynamics issue in Fig. 1.4. Here, for example, a magnetic well (average minimum B) was first shown to provide macroscopic stability in mirrors and multipole devices, but this physics has now become an important ingredient in many other types of concepts as well. The cross-fertilization of plasma science understanding among concepts is a very important facet of the development of magnetic confinement systems. It fosters the improvement of concepts in one line through understanding gained by advances in another. Thus, potential contributions to generic plasma science development are an important ingredient in the assessment of the value of particular concept lines for the fusion program.

In addition to their generic contributions, many concepts are potential candidates for an attractive commercial fusion application. As a confinement concept progresses

FIGURE 1.3 Level 1 Generic Logic Diagram for Confinement Systems and Burning Plasma Areas

FIGURE 1.4 Interrelationships among Concepts for Macroscopic Equilibrium and Dynamics Issues

toward this goal, it passes through a number of stages of development -- exploratory studies, concept systems tests, integrated proof-of-principle (POP) experiments, and tests of reactor-plasma conditions. These stages are defined in Table 1.1. The definitions given should be interpreted as measures of performance to be achieved or surpassed in order for a concept to be considered eligible for advancement into the next stage of experiments.

Figure 1.5 shows two possible limiting strategies for concept development: (1) parameter improvements followed by concept improvements and (2) the converse of

TABLE 1.1 Achievement Objectives for Confinement Systems

Issue	Exploratory Studies of Key Issues	Confinement-System Concept Tests	Integrated Proof-of-Principle Experiments	Reactor-Plasma- Condition Experiments
Macroscopic Equilibrium and Dynamics	Establish stable operating regime	Determine equilibrium and stability boundaries	Optimize macroscopic equilibria (β scaling)	Achieve fusion- level β
Transport	Identify loss mechanisms	Explore confinement scaling ($n\tau_E \sim 10^{17} - 10^{18}$, $T \sim 0.2 - 1$) ^a	Establish satisfactory confinement scaling ($n\tau_E \sim 10^{18} - 10^{19}$, $T \sim 1 - 4$) ^a	Achieve fusion level confinement ($n\tau_E > 10^{20}$, $T \gtrsim 10$) ^a
Wave-Plasma Interactions	Determine key wave issues	Develop heating and distribution control methods	Extend to high power levels	Sustain fusion plasma
Particle-Plasma Interactions	Limit neutral/ impurity effects	Develop impurity con- trol, fueling and beam methods	Develop efficient impurity control, fueling, and beam methods	Sustain fusion plasma
Composite	--	--	Assess pulse length requirements	Optimize pulse length

^a $n\tau_E$ in s/m^3 ; T in keV.

FIGURE 1.5 Schematic Illustration of Alternative Concept-Development Paths

this. Realistic strategies are usually a mixture of these two extremes. Also, concepts do not always proceed along a direct line of parameter improvements toward a reactor. Thus, for example, a concept that has advanced to the concept systems test stage may remain at that stage (or even drop back) for the investigation of a concept variant that has greater potential as a reactor.

More specific objectives for the five magnetic confinement concept issues are best elucidated in the context of classes of concepts, distinguished by their current state of development; these states, in turn, reflect the integrated resources applied to the concept classes in the fusion program to date. This classification is shown in Fig. 1.6. In the TPA reference scenario (see Fig. S.10), the burning plasma issue is addressed through the tokamak concept, the only one with a current knowledge base sufficiently well developed for confident extrapolation into a fusion burning plasma regime.

The non-tokamak confinement concepts have benefited from the extensive development of magnetic confinement plasma science in connection with tokamaks and will continue to do so in the future.

Thus, the most promising of the alternative concepts can develop faster and more economically. In addition, the other confinement concepts have historically played a critically important role in broadening the development of plasma science for magnetic confinement, making possible both better understanding and more attractive options for all magnetic confinement concepts, including tokamaks, (e.g., current drive for steady-state tokamaks).

The concepts with a moderately developed or developing knowledge base at present, along with the tokamak, are utilized to address the broader magnetic confinement issues of developing the plasma science base and commercially attractive fusion applications. The concepts listed in Fig. 1.6 are those that have emerged from the evolution of concepts over the history of the fusion program. The present concepts offer unique opportunities for development of attractive fusion applications, either as individual concepts or (often) as contributors of important variants or options for other concepts.

FIGURE 1.6 Confinement-Concept Classification

The anticipated evolution of the moderately developed and developing concepts is shown in Fig. 1.7. The [m] and [d] boxes in Fig. 1.7 indicate the typical times at which particular moderately developed and developing concepts may reach the objectives indicated in Table 1.1. Note that in general the developing concepts continue to lag behind the moderately developed concepts, albeit usually by a shortening time increment. Also, some concepts may move from one category to another by their relative degree of success. The number of concepts of either class is likely to be reduced with time as they progress to larger and presumably more costly phases. Finally, note that some concepts may continue to be carried at the exploratory studies and concept systems test levels because of unique capabilities to contribute to generic plasma science development.

The evaluation/decision points indicated in Fig. 1.7 are the typical times at which concepts in that stage of development can be expected to progress to the next stage. However, a concept will often remain at a given stage for one or more experimental phases (see Fig. 1.5) to improve its reactor potential before proceeding to the next, presumably more costly stage. Also, new concepts may arise with a new invention or as outgrowths of other concepts. Decision descriptions for the evaluation points shown (see Tables S.5 and S.9) follow from the criteria listed in Table 1.1.

The remainder of this chapter lays out the issues, objectives, and attributes for each of the moderately developed and developing concepts. Program logics are laid out in concept-specific programs to facilitate the development of meaningful, detailed objectives and attributes for resolving the various issues involved in the MFPP confinement systems issue.

One should not interpret these concept-specific programs as implying that all the indicated tasks for each concept will be necessarily carried out in parallel with all the other concepts. It is anticipated that only a portion of the indicated research in this chapter will be pursued, depending on the technical progress of each concept, the contributions of the research on each concept to plasma science in general, the relative reactor merit of each concept compared to others, further development in international collaboration and, of course, available resources. While the results of the future selection process cannot be predicted, the TPA has provided the ingredients for the selection process by developing a generic format for describing the technical issues of each concept, by establishing technical criteria for each stage of concept development, and by laying out the future needs for each concept.

1.1 CONCEPTS WITH A MODERATELY DEVELOPED KNOWLEDGE BASE

This section presents the program logics for tokamaks, tandem mirrors, stellarators, and reversed-field pinches. The tokamak subsection (1.1.1) presents a comprehensive program logic that deals with main line issues resulting from using the tokamak to examine burning plasma issues (see Chapter 2), as well as issues to further develop the tokamak as a commercial reactor candidate. Because the tokamak is at a more advanced stage of development than the other concepts, it is possible to define its program logic in greater detail and with a higher level of technical confidence.

FIGURE 1.7 Progress of Confinement Systems Concepts

1.1.1 Tokamaks

In order to achieve definitive experimental results, fusion research must be done in the reactor-plasma regime, ultimately with burning deuterium-tritium (DT) plasmas. On the other hand, to document the existence of a viable fusion-power solution, it is essential to develop innovative features and techniques for enhancing reactor performance. These two requirements can be harmonized in an overall tokamak program consisting of a main line reactor-plasma effort, utilizing tokamaks of relatively conventional design, accompanied by a set of specialized and more innovative supporting experiments. The basic design assumptions of the main line devices can then be held to a level of conservatism appropriate to their missions, while the exploitation of innovative ideas can be paced according to their state of experimental success.

Technical Issues, Objectives, and Attributes

Table 1.2 summarizes the major issues to be addressed in the tokamak confinement program. Issues specific to burning plasma applications are discussed in Chapter 2. A more complete discussion of both sets of issues is to be found in the TPA Plasma Science Final Report (Ref. 1). Tokamak plasma technology issues are discussed in Chapter 3 which, although generic, predominantly discusses technology in terms of the tokamak program needs.

Macroscopic Equilibrium and Dynamics. The maximum beta values achieved in a number of tokamak experiments are consistent with theoretical predictions and range up to $\langle\beta\rangle$ values of 5%. These experiments have all used neutral beam heating, and together they include plasma cross-sections ranging from circular to elongated D-shaped to bean-shaped cross-sections. Although the maximum beta values are accurately described by the theoretically predicted limit, the saturation mechanisms and the roles of particular instabilities are not yet well understood. Promising approaches to the realization of still higher beta values are stronger elongation and shaping, entry into the second stability regime, and low- $q(0)$ operation by sawtooth suppression. Future tokamaks with much higher electron temperatures should be relatively free of resistively produced magnetic transients, since the time scale for magnetic-field reconnection will become correspondingly long. However, the thermal and mechanical stresses associated with even a small number of major "disruptions" constitute a formidable problem in tokamak reactor design.

Transport. By classical criteria, the plasma current needed to confine 3.5-MeV alpha particles in a tokamak is easily sufficient to meet the $n\tau_E$ condition for ignition; thus, tokamak confinement at multimegampere (MA) currents is highly tolerant of moderate enhancements of classical loss processes. However, in confinement experiments -- where it constitutes the main energy loss mechanism -- plasma transport across the magnetic field (especially electron thermal transport) is found to exceed classical predictions substantially. No theoretical explanation for the deviation from classical behavior has been validated, so tokamak transport has generally been described

TABLE 1.2 Tokamak Program Elements, Subelements, Objectives, and Attributes

Program Elements and Subelements	Objectives	Attributes
Macroscopic Equilibrium and Dynamics		
Pressure-Driven Effects	Verify the limits of stability of tokamaks with optimized profiles and cross-sectional shapes against pressure-driven modes (first stability regime) and identify the limiting physical processes.	Predictive capability
	Determine the plasma physics requirements for realizing the second stability regime in tokamaks, consistent with engineering practicality.	Predictive capability
	Achieve reactor-relevant beta value.	< β >
Current-Driven Effects	Understand the processes of resistive magnetic reconnection in tokamaks, in order to suppress or control harmful magnetic transients.	Predictive capability
	Minimize major disruptions.	Disruptivity
Transport		
Perpendicular Confinement	Develop a validated predictive capability for anomalous electron thermal diffusivity in tokamaks that minimizes the required extrapolation to the reactor regime.	Predictive capability
	Develop a validated predictive capability for ion thermal diffusivity in tokamaks, including both neoclassical and anomalous contributions, that minimizes the required extrapolation to the reactor regime.	Predictive capability
	Demonstrate reactor-level perpendicular energy confinement.	< χ >

TABLE 1.2 (Cont'd)

Program Elements and Subelements	Objectives	Attributes
Wave-Plasma Interactions		
Wave Heating	Develop a validated predictive capability for radio-frequency wave coupling, propagation, and absorption in tokamaks, in the ion-cyclotron, lower-hybrid, and electron-cyclotron frequency ranges.	Predictive capability
	Achieve reactor-level ion temperature at high density.	T_i
Maintenance of Non-Maxwellian Distributions	Demonstrate current-drive scenarios that extrapolate to an efficient steady-state or quasi steady-state (e.g., involving non-inductive current ramp-up) mode of tokamak operation.	Efficiency, nRI/P
Particle-Plasma Interactions		
Impurities	Demonstrate limiter and divertor modes of tokamak operation that shield the plasma core from edge-generated impurities.	Z_{eff}
	Develop a validated predictive capability for cross-field impurity (including helium) convection and diffusion in tokamaks, including both neoclassical and anomalous contributions.	Predictive capability
Fueling	Develop a quantitative understanding of the ablation of deeply penetrating pellets in tokamaks and of their effect on plasma transport and stability.	Predictive capability
Neutral-Beam Injection	Characterize reactor neutral beam requirements for heating and current drive.	Predictive capability
	Determine current-drive efficiency, augmented by self-generated currents.	Efficiency, nRI/P

TABLE 1.2 (Cont'd)

Program Elements and Subelements	Objectives	Attributes
Composite Issues		
Profile Optimization	Understand the processes by which radial profiles are established in tokamaks, in order to optimize confinement, heating and impurity control.	Predictive capability
Long Pulse	Validate the predictive capabilities of tokamak codes for high-beta equilibria evolving on a resistive time scale.	Predictive capability
	Demonstrate sustained tokamak equilibria for pulse lengths approaching the resistive skin time, consistent with other desired plasma characteristics.	$\tau_{\text{pulse}} / \tau_{\text{skin}}$

only from an empirical viewpoint. Although the single most important element of classical scaling -- a strongly favorable dependence on plasma size -- appears to be holding firm in empirical scaling, the development of a physical understanding of the underlying transport processes would improve greatly the reliability of extrapolations to the reactor regime. Present empirical scalings are expected to prove adequate for the modest extrapolations required for the short-pulse ignition experiment, provided the design of this device is appropriately conservative in regard to confinement.

Wave-Plasma Interactions. The response of tokamak plasmas to externally generated radio-frequency (rf) waves has been studied extensively, the main objectives being plasma heating and current drive. Ion cyclotron heating (ICH) appears to offer the most promising combination of source feasibility and antenna-plasma coupling for near-term applications, such as the short-pulse ignition experiment, but other rf techniques have potentially superior characteristics for longer-term applications. Radio-frequency waves can also be used to drive a toroidal current noninductively in the tokamak, by wave-particle interactions that introduce relative momentum between electrons and ions. Current drive involving interactions between rf waves and suprathermal electrons appears to offer the best prospect for acceptable efficiency combined with adequate penetration of the rf power to the center of a reactor-like plasma. The efficiency increases with increasing electron energy but decreases as the plasma density increases, resulting in only marginally acceptable theoretical efficiency at reactor-like parameters, unless the reactor is operated in a low-density, high-temperature mode or self-generated (i.e., "bootstrap") plasma currents appear.

Particle-Plasma Interactions. The particle-plasma interaction that is the least understood and has the broadest implications is that of impurity control. Nonhydrogenic ions are undesirable in the plasma core, because they dilute the fusion fuel and (if incompletely stripped) lead to heat losses by line-radiation. Present tokamak experiments can control impurities effectively using mechanical "limiters," usually made of high-grade graphite, to define the plasma edge. In future reactors, the density of heat outflow will have to be endured under thermal steady-state conditions, and the helium "ash" will have to be removed at a rate that exceeds the capability of conventional gas pumping. An appropriate device for such applications is the pumped limiter -- a special type of mechanical limiter with an aperture into which the recycling gas can stream directly. A more powerful alternative approach (divertor) is to define the plasma edge by a magnetic separatrix -- a feature of the magnetic field configuration that tends to occur naturally in tokamaks with strongly shaped cross section. The objectives are the same as for the pumped limiter, but the plasma outflow is channeled more easily, and the first mechanical surface that makes plasma contact can be isolated from the main high-power-density plasma by the pumping action of the outflow along the divertor channels. Confidence in the effectiveness of the magnetic divertor in performing the impurity-control and ash-removal functions for a tokamak reactor is high. However, the special magnetic-field-shaping and space requirements associated with a divertor impose at least moderate additional burdens on tokamak engineering design; it may be inferred that if the pumped-limiter approach proves adequate, its greater simplicity will make it the preferred choice. The particle-plasma interactions involved in plasma fueling and neutral beam injection are relatively well understood; neutral beam heating and current drive -- presently in disfavor -- may become a viable alternative to rf techniques if the technology of negative-ion beams is developed.

Composite Issues. Pulsed operation of a magnetic fusion reactor has a number of potential disadvantages -- especially if the pulses are short. In the tokamak, the principal obstacle to very-long-pulse (or steady-state) operation is the maintenance of the toroidal plasma current. However, because of the possibility of secular accumulation of helium ash and other impurities in the central core of a tokamak, the issue of impurity control is likely to be at least equal in importance to that of current drive in realizing acceptable long-pulse conditions. On timescales approaching the resistive skin time (typically about 10^3 s at reactor parameters), the high-beta tokamak equilibrium and pressure profile will evolve in a manner that can be predicted by theory but has not yet been explored experimentally. The principal "composite issue" for the tokamak will be to maintain a high-beta equilibrium that is satisfactory from the viewpoint of current-drive efficiency and effectiveness of impurity-control techniques. In particular, plasma profiles must be maintained that minimize deleterious plasma-wall interactions.

Tokamak Program Logic

Because less is known at present about the advanced tokamak (very high beta value and/or steady state) than about the main line tokamak, the overall tokamak program plan outlined in Fig. 1.8 envisages a ten-year period before an attempt is made to identify (in 1996-97) an optimized tokamak concept for reactor evaluation. The level

FIGURE 1.8 Level 2 Logic Diagram for Tokamaks

of reactor demonstration that will follow a favorable evaluation of the optimized tokamak concept and favorable performance of the tokamak long-burn facility (evaluation E_g) will depend on the nature of the concept. If the optimized concept represents a substantial departure from the main line tokamaks, a major DT reactor experiment may be needed, beginning in about 2000, to demonstrate the practical potential of the new concept and to help develop a basis for commercialization. If, as seems more likely, the optimized concept represents a straightforward evolution relative to the main line DT tokamak facilities of the late 1990s (as will be the case if high-beta and steady-state capabilities have been introduced into the long-burn facility), a more limited demonstration project will suffice to establish the practical potential of the concept in the reactor context.

Macroscopic Equilibrium and Dynamics. The near-term supporting program in the area of beta-related tokamak innovations (Fig. 1.9) consists of five elements: (1) evaluation of the vertically elongated D-shape, (2) evaluation of the bean shape, (3) investigation of means of access to the second stability regime, (4) investigation of kink (disruption) control using passive stabilizers, and (5) feedback stabilization of resistive kink modes by means of rf heating and current drive.

These exploratory studies will help greatly in assessing the attainability of the moderately high beta values required for the viability of the tokamak as a reactor candidate. During the early 1990s, the most promising tokamak geometry should then be pursued further in an "Advanced-Geometry Tokamak Experiment," with a view to confirming the basis for tokamak reactor viability and contributing to the data base needed for configurational optimization. For the advanced-geometry experiment, a "hydrogen" machine of substantial parameters -- possibly an upgrade of one of the existing facilities -- should be sufficient.

To achieve success in the tokamak DT experiments, the minimum beta requirements are more modest than the level typically envisaged as the goal for reactor viability: beta values of about 2% and 5%, respectively, should suffice for break-even and ignition. The outcome of the supporting high-beta studies will help to determine what part of the main line activity in beta optimization could be performed as an option on the short-pulse ignition experiment, and what part should await construction of the advanced-geometry experiment and the long-burn facility.

Transport. The study of confinement is an important part of all tokamak experiments. As indicated in Fig. 1.10, tokamak confinement will be put to its most critical tests in the main line DT experiments, where the plasma temperature and density will reach the full values characteristic of the reactor regime. Important information also will come from specialized studies on the sensitivity of tokamak transport to profile effects connected with auxiliary heating, plasma edge control (e.g., by a magnetic separatrix), and fueling. One of the most cost-effective ways of achieving high $n_T E_g$ values has been to operate at very high magnetic fields (8-12 T); this is considered to be a principal option for ensuring the success of the short-pulse ignition experiment. Accordingly, high-field confinement studies will need to be extended to an improved facility in the near future.

FIGURE 1.9 Level 3 Logic Diagram for Tokamak Macroscopic Equilibrium and Dynamics Issue

FIGURE 1.10 Level 3 Logic Diagram for Tokamak Transport Issues

The beta-oriented experiments highlighted in the previous section will generate important insights into the behavior of transport near the beta limit. The combination of data from these experiments with confinement results from the short-pulse ignition experiment should serve as a sufficient basis for the documentation of ignition-level confinement in a moderate-sized tokamak reactor.

Wave-Plasma Interactions. While neutral beam heating is still the principal tool for achieving high-temperature experimental plasmas, rf techniques for heating and current drive (Fig. 1.11) have made substantial progress and would lend themselves particularly well to the cost-effective sharing of start-up, heating, current-drive, and perhaps feedback-control functions in a tokamak reactor. The most likely candidate for heating the short-pulse ignition experiment appears to be an ion cyclotron resonance heating (ICRH) system. A substantial increase in the present ICRH data base will be required, including work at ignition-level plasma temperatures, at high beta values, and at high magnetic-field strengths.

The near-term program directed at optimization of rf heating techniques consists of four elements: (1) ion cyclotron heating, (2) ion Bernstein-wave heating, (3) electron cyclotron heating, and (4) lower hybrid heating. Over the longer term, the number of approaches needs to be reduced so that only two techniques are developed as candidates for implementation on the long-burn facility.

Radio-frequency current-drive techniques can provide the key to the very-long-pulse, or even steady-state, tokamak regime -- an important aspect of future tokamak reactor optimization. The general pattern in an efficiently designed tokamak reactor will be that the bulk of the plasma-heating power will come from alpha particles, while rf power is used only for "higher-quality" tasks: initial current ramp-up and plasma heating to ignition, noninductive sustaining of the current, and maintenance of control over detailed features of the temperature and current profiles.

Near-term current-drive studies exploring a number of different techniques (Fig. 1.11) will go a long way toward assessing the attainability of the efficiency required in a tokamak reactor. During the early 1990s, the most promising current-drive technique should then be pursued further in a "Steady-State Tokamak Experiment." The steady-state experiment will confirm the basis for the introduction of steady-state capability into the long-burn facility and will develop the data base needed for optimization of the current-drive technique. The steady-state experiment could be a separate "hydrogen" machine with very-long-pulse technological capabilities, or -- if the current-drive data base from nearer-term studies were sufficiently encouraging and definitive -- it could be provided by introducing a steady-state capability into the long-burn facility itself.

Particle-Plasma Interactions. Since the Q value is a sensitive function of plasma impurity content, the main line DT tokamaks will provide a critical test of impurity control using simple limiter and separatrix techniques. While the need to remove helium "ash" is not a serious obstacle for short-pulse burns, such as those envisaged for the

FIGURE 1.11 Level 3 Logic Diagram for Tokamak Wave-Plasma Interactions

ignition experiment, special experiments can and should be carried out to address helium build-up.

The near-term program in the area of particle and impurity control (Fig. 1.12) will develop a sound technical basis for the choice between divertor and pumped-limiter options for application to the advanced-geometry and steady-state tokamak experiments, as well as for eventual use on the long-burn facility.

Reactor impurity control will involve many phenomena with slow time constants -- such as thermal transients and impurity diffusion through the surfaces of wall components -- which will be addressed by development of appropriate techniques on the steady-state experiment. (The decision on the best magnetic limiter or divertor configuration will be strongly influenced by the outcome of the beta-pushing studies on the advanced-geometry experiment.) Some of the objectives of the steady-state experiment could be addressed by designing a pulsed facility to be operable also in a long-pulse, lower-field mode. The definitive approach, however, will be to construct a specially dedicated facility with steady-state magnet coils and vessel-cooling capabilities. The initial data base from such an experiment, together with initial results from the advanced-geometry experiment, will provide a substantial basis for effective impurity control in the long-burn facility. Full optimization of steady-state fueling and impurity control will be a principal role for the steady-state tokamak experiment in the mid-1990s. Initial results on helium-ash build-up and exhaust will come from the long-burn facility in time for the tokamak reactor evaluation in 2000.

Neutral beam heating and current drive remains a candidate for steady-state applications (Fig. 1.12), provided negative-ion-beam technology is developed and beam-driven currents are augmented by some kind of self-generated (i.e., "bootstrap") plasma current.

Composite Issues. Near-term experiments on all major tokamaks will contribute greatly to our understanding of the processes by which radial profiles are established in tokamaks. The introduction of resistivity into computer codes describing high-beta tokamak equilibria should lead to reliable predictions regarding the long-term evolution of tokamak configurations. Although a definitive experimental demonstration of tokamak equilibria sustained for pulse lengths approaching the resistive skin time must await the "long-burn demonstration," the combined results from the advanced-geometry experiment (high beta), the steady-state experiment (rf current drive and long-pulse impurity control), and the short-pulse ignition experiment (profile optimization for burning plasmas) should serve to provide confidence that the issue of long-pulse equilibrium and profile control can be resolved satisfactorily.

FIGURE 1.12 Level 3 Logic Diagram for Tokamak Particle-Plasma Interactions

Major Tokamak Evaluation/Decision Points

The major evaluation/decision points in the tokamak program are indicated in Figs. 1.8-1.12:

- E_1 : Decision on short-pulse ignition experiment
- E_a : Decision on impurity control option favored for future applications
- E_b : Decision on heating and current-drive options favored for future development and application
- E_c : Decision on an advanced-geometry tokamak experiment
- E_d : Decision on a steady-state tokamak experiment
- E_3 : Decision on a long-burn facility
- E_e : Tokamak reactor evaluation

Tables 1.3-1.9 summarize the criteria, the information needed, and the expected consequences of these decision points. These tables emphasize the plasma science aspects of these decisions. A more general description of decisions E_1 and E_3 is found in the Synopsis (Tables S.4 and S.6).

Major Tokamak Facility Requirements

Existing Facilities. The capabilities of the world's four largest tokamak facilities are indicated in Table 1.10, and the capabilities of other major tokamaks in the world program are indicated in Table 1.11. In each case, the research contributions of principal importance to the overall program have been highlighted.

The initial task of the Tokamak Fusion Test Reactor (TFTR) is to advance the main-line tokamak effort by entering the reactor plasma regime in 1986-87. At this time, critical assessments can begin to be made concerning the outlook for scaling to ignition and the potential for improving confinement and stability by auxiliary techniques. The present plan calls for the TFTR facility to demonstrate $Q \sim 1$ in a DT plasma during 1989, so as to develop a base of DT experience for the short-pulse ignition experiment. An alternative option -- which could become important if the short-pulse ignition step were delayed -- is to modify the TFTR for higher currents and beta values, with a view to achieving even higher Q -value DT operation by the early 1990s. In either case, the introduction of moderate rf capabilities will prove important for maximizing plasma performance.

TABLE 1.3 Decision E₁ on Short-Pulse Ignition Experiment (plasma science aspects)

Statement of Decision

To proceed with an ignition experiment in which the primary objective is to demonstrate a fusion system that is self-sustaining energetically.

Decision Criteria (plasma science)

Success in achieving DT-equivalent breakeven and in identifying a favorable transport scaling in large tokamaks at breakeven-level plasma parameters.
 Effectiveness of plasma elongation for achieving stable, nondisruptive operation at moderate beta values ($\langle \beta \rangle \sim 0.05$).
 Favorable assessment of effect of plasma elongation on confinement.
 Favorable assessment of performance and utility of high-field tokamaks.
 Effectiveness of high-power (> 10 MW) ion cyclotron heating in moderate-beta plasmas with ignition-level parameters.
 Effectiveness of either divertor (separatrix) or limiter techniques for impurity-control at high power density.
 Favorable assessment of pellet fueling techniques for creation and maintenance of preferred density profiles.
 Favorable assessment of the suitability of a high-field tokamak for confining and studying energetic alpha particles.

Sources of Information (plasma science)

DT-equivalent breakeven experiments in large tokamaks (TFTR).
 Other tokamak experiments with capabilities for plasma elongation: Joint European Torus (JET), Big-D Tokamak at GA Technologies (DIII-D); high-power ion-cyclotron heating: JET, Axially Symmetric Divertor Experiment ASDEX), TFTR, DIII-D; and operation with a magnetic separatrix: ASDEX, JT-60, DIII-D, Princeton Beta Experiment (PBX), JET.
 High-field tokamaks (Alcator C and successor programs, FT).
 Basic research, theory and computation, and plasma supporting activities.
 Conceptual designs of short-pulse ignition experiments.

Outcomes and Consequences of Decision

Favorable assessment and subsequent achievement of ignition objective will lead to early insight into the physics of self-sustained fusion burning plasmas. This knowledge will reduce the risk and increase the productivity of the long-burn experiment.
 If no short-pulse ignition experiment is built, then most of the burning plasma physics issues will have to be explored for the first time in the long-burn experiment, which will likely be a less ambitious step and entail more conservative design margins.

TABLE 1.4 Decision <a> on Impurity-Control Option Favored for Future Applications**Statement of Decision**

To choose between divertor (separatrix) and limiter options for application to the short-pulse ignition experiment, and between divertor (with exhaust) and pumped-limiter options for development and subsequent application to advanced-geometry and steady-state experiments and to the long-burn facility.

Decision Criteria

- For the short-pulse ignition experiment, effectiveness in maintaining sufficiently pure plasmas ($Z_{\text{eff}} \leq 1.5$), despite high power densities on the limiter/divertor-plate surface.
- For the advanced-geometry experiment, effectiveness in impurity control and compatibility with shaped, high-beta magnetic configurations.
- For the steady-state experiment, effectiveness in active removal of impurities from the plasma chamber and in maintaining the desired plasma composition and density profile over times that are long compared with particle transport and limiter/divertor-plate thermal time scales.
- For the long-burn facility, effectiveness in removing helium ash while maintaining adequate tritium burn-up.
- For steady-state and long-burn applications, availability of relevant technologies relating to high-heat-flux materials with favorable erosion characteristics.
- For all applications, effectiveness in assuring favorable energy confinement in the plasma and reliability of predictive capabilities in plasma-edge modeling.

Sources of Information

- Tokamak experiments with divertors (ASDEX, JT-60) and with divertor-like capabilities (DIII-D, PBX, JET), especially those with high levels of auxiliary heating power (>10 MW).
- Tokamak experiments with simple limiters and high levels of auxiliary heating power (TFTR, JET).
- Tokamak experiments with pumped limiters: the Tokamak Experiment for Technology Oriented Research (TEXTOR), together with predictive models capable of extrapolating the data base (< 10 MW) to the high-power regime.
- Computational modeling of plasma edge, divertor, and pumped-limiter behavior.
- Plasma technologies, especially high-heat-flux materials development.
- Conceptual design of short-pulse ignition experiment.
- Design studies for a long-burn facility.

TABLE 1.4 (Cont'd)

Outcomes and Consequences of Decision

A technically sound and validated choice between separatrix and limiter options for the short-pulse ignition experiment will improve the cost-effectiveness of the experiment's design and will give added assurance as to its probable success.

Identification of a technically sound impurity and particle control technique for long-pulse tokamaks will improve greatly the prospects for a long-burn facility and for a viable tokamak reactor.

The principal task of the DIII-D is to study the optimization of plasma shape, especially elongation, with a view to achieving high beta values at high plasma temperatures. In pursuit of this task, the DIII-D will also make important contributions to the development of rf techniques -- particularly high-powered electron cyclotron heating (ECH) for profile-shaping. The DIII-D plasma-shaping studies are closely coupled to the development of impurity control using simplified divertor (separatrix) and pumped-limiter techniques. The capabilities of the DIII-D could be enhanced by raising the plasma current and adding rf heating power, especially in the ICRH area.

To explore the high-beta capabilities of the "bean-shape" as early as possible, the PBX device is being reconfigured for definitive second-stability-regime beta-pushing experiments starting in 1987.

The Princeton Large Torus (PLT) facility has proven particularly well suited for development of rf heating and current drive in the areas of ICRH and lower hybrid heating (LHH), as well as for initial exploration of the rf-feedback stabilization idea. The near-term data base required for a short-pulse ignition experiment in the area of rf techniques has been provided by the combined, coordinated efforts of the PLT and Alcator C.

Present budgetary stringencies imply that after 1986, PLT experimental operations will have to be stopped, while Alcator C will be shut down for conversion to a form more directly prototypical of the high-field, shaped-plasma regime of the short-pulse ignition experiment. The transfer of PLT's rf capabilities to TFTR in 1987 will provide valuable opportunities for augmenting the rf-heating data base for the short-pulse ignition experiment.

The overall picture of the world tokamak program presented by Tables 1.10 and 1.11 reveals that there is a significant difference in the relative timing of the U.S. and foreign programs relative to the next steps in the main line DT effort: the existing U.S. experimental facilities will be able to provide a major part of the data base for the short-pulse ignition experiment, whereas the extensive new construction efforts of the foreign programs are timed to yield their peak experimental results in the early 1990s, in support of plans for a technology-oriented long-burn facility.

TABLE 1.5 Decision on Heating and Current-Drive Options Favored for Future Development and Application

A. RF HEATING

Statement of Decision

- (1) To choose among ion cyclotron, ion Bernstein-wave, electron cyclotron, and lower hybrid rf-heating techniques for application to the advanced-geometry and steady-state tokamak experiments; and
- (2) To select two of these techniques for subsequent development and application to the long-burn facility.

Decision Criteria

Demonstrated effectiveness in achieving reactor-level ion temperatures at reactor-relevant plasma densities and beta values.
 Reliability of predictive capabilities regarding rf applications to advanced plasma configurations.
 Availability of relevant rf technologies at high power and long pulse.

Sources of Information

Tokamak experiments with high-power ion cyclotron (JET, ASDEX, TFTR, DIII-D), ion Bernstein-wave (PLT and successor programs, Alcator C and successor programs), electron cyclotron [DIII-D, Large Superconducting Tokamak at the Kurchatov Institute, Moscow, USSR (T-15)], and lower hybrid (JT-60, ASDEX) rf heating capabilities.
 Supporting and innovative experiments on smaller-scale tokamak facilities;
 Theory and computational modeling.
 Radio-frequency technology development.
 Design studies for a long-burn facility.

Outcomes and Consequences of Decision

A technically sound and validated choice among rf heating options for application to future tokamaks, including the long-burn facility, will improve the cost-effectiveness of the designs of these devices and will give increased assurance as to their probable success.

B. RF CURRENT DRIVE

Statement of Decision

To choose among fast-wave, electron cyclotron, lower hybrid, and other possible rf current-drive techniques for application to the steady-state tokamak experiment and for subsequent development and application to the long-burn facility.

TABLE 1.5 (Cont'd)***Decision Criteria***

Efficiency of current drive projecting to values of at least 0.1 A/W at reactor-level plasma densities, temperatures, and beta values.
 Reliability of predictive capabilities regarding effects of non-Maxwellian distributions on plasma behavior.
 Availability of relevant rf technologies at high power and long pulse.

Sources of Information

Tokamak experiments with high-power fast-wave (PLT and successor programs), electron cyclotron (DIII-D, T-15) and lower hybrid (JT-60, ASDEX) rf current-drive capabilities.
 Supporting and innovative experiments on smaller-scale tokamak facilities;
 Theory and computational modeling.
 Radio-frequency technology development.
 Design studies for a long-burn facility.

Outcomes and Consequences of Decision

Identification of an efficient current-drive technique at reactor-level plasma parameters will improve the cost-effectiveness and attainable neutron fluences in a tokamak long-burn facility and will enhance greatly the prospects for an attractive tokamak reactor.

C. NEUTRAL BEAM HEATING AND CURRENT DRIVE***Statement of Decision***

To assess the utility of neutral beam heating and current drive (including negative-ion-based beams) for possible application to the steady-state tokamak experiment and for subsequent development and application to the long-burn facility.

Decision Criteria

Projected efficiency of current drive to values of at least 0.1 A/W at reactor-level plasma densities, temperatures, and beta values.
 Experimental existence and utility of self-generated ("bootstrap") plasma currents.
 Availability of relevant (negative-ion-beam) technologies at high powers and long pulse lengths.
 Assessment of impact on tokamak systems engineering.

TABLE 1.5 (Cont'd)

Sources of Information

Tokamak experiments with high-power neutral beam heating (TFTR, JET, JT-60, DIII-D, ASDEX), especially those with a capability for near-tangential injection.

Neutral beam technology development.

Design studies for a long-burn facility.

Outcomes and Consequences of Decision

A favorable assessment of the utility of neutral beam heating and current drive (using negative-ion beams) for steady-state tokamak applications will enhance the prospects for an attractive tokamak reactor, will have an important impact on the design of the long-burn facility, and will affect the technology development program.

New Facilities. The advanced-geometry experiment will be a "hydrogen" tokamak of substantial parameters, which will simultaneously demonstrate reactor-level beta values ($\beta \sim 0.1$) and reactor-level confinement ($\langle \chi \rangle \sim 1 \text{ m}^2/\text{s}$). The experiment will incorporate advanced rf-heating and impurity-control techniques. The experiment could take the form of an upgrade of the DIII-D facility, particularly if vertical elongation is found to be an optimum route for tokamak optimization. Alternatively, the experiment could be based on the bean-shaping, second-stability-regime approach. The project could be initiated by 1990, in which case the experiment would be operational by the mid-1990s, allowing major input on configurational optimization to the design of the long-burn facility. The capital cost would be in the range of \$70-100 million, spread over the early 1990s.

Ideally, the steady-state experiment will be a "hydrogen" tokamak of moderate parameters, aimed at demonstrating efficient current-drive and long-pulse impurity control in a device with steady-state magnet and power-handling technologies. The experiment will incorporate advanced rf heating techniques. The project could be initiated by 1990, in which case the experiment would be operational by the mid-1990s, allowing major input on steady-state operation to the design of the long-burn facility. The capital cost would be in the range of \$90-120 million, spread over the early 1990s. As an alternative to a separate facility, it may be appropriate to combine the advanced-geometry experiment and the steady-state experiment into a single facility. It also might be possible -- if the early-1990s data base on current drive were particularly definitive -- to implement the steady-state tokamak program on the long-burn facility itself.

The high-field approach to tokamak confinement will be pursued on a modest scale, with an emphasis on demonstrating the compatibility of high fields with advanced plasma-shaping, rf-heating, and impurity-control (divertor) techniques. The possible

TABLE 1.6 Decision <c> on an Advanced-Geometry Tokamak Experiment**Statement of Decision**

To proceed with construction of an advanced-geometry tokamak experiment, for which the primary objectives are (1) to demonstrate the simultaneous achievement of reactor-level beta values ($\langle\beta\rangle \sim 0.1$) and confinement parameters ($\chi \sim 1 \text{ m}^2/\text{s}$) at reactor-like plasma densities and temperatures and (2) to optimize the magnetic configuration for possible further improvements in beta and confinement.

Decision Criteria

Effectiveness of plasma elongation for achieving stable, nondisruptive operation at $\langle\beta\rangle$ values of 0.1.
 Effectiveness of bean shaping for achieving stable, nondisruptive operation at $\langle\beta\rangle$ values of 0.1.
 Experimental existence and utility of the "second stability regime" of tokamak operation.
 Efficacy and practicality of passive-conductor stabilization of external kinks and control of disruptions at low $q(a)$ values.
 Efficacy and practicality of rf techniques for sawtooth suppression at $q(0)$ values substantially below unity.
 Favorable assessment of confinement scaling in large tokamaks at reactor-level plasma parameters.
 Favorable assessment of divertor (with exhaust) or pumped-limiter impurity-control techniques (see evaluation E_a).
 Favorable assessment of rf heating techniques (see evaluation E_b).

Sources of Information

Tokamak experiments with high plasma elongation (DIII-D, JET), bean shaping (PBX), second-stability-regime capability (PBX), passive-conductor stabilization of kinks and disruptions (DIII-D, PBX), and rf capabilities suitable for sawtooth suppression (PLT and successor programs, Alcator C and successor programs, ASDEX, JT-60).
 Supporting and innovative experiments on smaller-scale tokamak facilities;
 Theory of tokamak plasma stability.
 Design studies for an advanced-geometry tokamak experiment and for a long-burn facility.

Outcomes and Consequences of Decision

Favorable assessment and subsequent achievement of beta ($\langle\beta\rangle \sim 0.1$) and confinement ($\chi \sim 1 \text{ m}^2/\text{s}$) objectives will improve the cost-effectiveness of the long-burn facility's design and will further assure its probable success.
 Further improvements in beta and confinement resulting from configurational optimization will greatly enhance the prospects for an attractive tokamak reactor.
 An alternative option would be to implement an experiment aimed at demonstrating advanced geometry on an existing tokamak or the long-burn facility itself.

TABLE 1.7 Decision <d> on a Steady-State Tokamak Experiment

Statement of Decision

To proceed with construction of a steady-state tokamak experiment for which the primary objectives are (1) to demonstrate efficient noninductive current drive (~ 0.1 A/W) at high current (> 1 MA) and (2) to optimize the current-drive technique for possible further improvements in efficiency.

Decision Criteria

Effectiveness of rf heating (see evaluation E_{pA}).
 Effectiveness of rf current drive (see evaluation E_{pB}).
 Effectiveness of neutral beam heating and current drive (see evaluation E_{pC}).
 Experimental existence and utility of self-generated currents in tokamaks.
 Effectiveness of divertor (with exhaust) or pumped-limiter for steady-state impurity and particle control (see evaluation E_a).
 Favorable assessment of pellet fueling techniques for steady-state tokamak applications.
 Availability of relevant technologies.

Sources of Information

Results of evaluations E_a and E_b .
 Tokamak experiments with moderate-pulse capabilities and high levels of auxiliary power (JET, JT-60, TFTR, DIII-D).
 Initial results from long-pulse superconducting tokamaks (Tore Supra, T-15).
 Plasma technologies, especially magnets, high-heat-flux materials, and steady-state rf systems.
 Design studies for long-burn facility.

Outcomes and Consequences of Decision

Favorable assessment and subsequent achievement of long-pulse tokamak operation with efficient noninductive current drive (≥ 0.1 A/W at $I > 1$ MA) will lead to early insight into the physics of steady-state tokamaks and will reduce the risk and increase the performance and productivity of the long-burn facility.
 Further improvements in current-drive efficiency will greatly enhance the prospects for an attractive tokamak reactor.
 An alternative option would be to implement an experiment aimed at demonstrating the stated current-drive efficiency on an existing tokamak facility and to develop the optimized current-drive technique on the long-burn facility itself.

TABLE 1.8 Decision E₃ on a Long-Burn Facility (plasma science aspects)

Statement of Decision

To choose the optimum approach and design for a tokamak long-burn facility.

Decision Criteria (plasma science)

Success in achieving breakeven in DT plasmas and favorable assessment of alpha-particle confinement and stability at breakeven-level DT plasmas.

Effectiveness of the "advanced-geometry tokamak experiment" for achieving stable operation at high beta-values ($\langle \beta \rangle \gtrsim 0.1$) with good confinement ($\chi \leq 1 \text{ m}^2/\text{s}$).

Reliability of predictive capabilities with regard to control of burning plasmas and helium-ash removal.

Effectiveness of techniques for suppression or control of disruption over long pulses.

Effectiveness of selected rf technique for long-pulse plasma heating.

Success in demonstrating efficient noninductive current drive ($\sim 0.1 \text{ A/W}$) at high current ($> 1 \text{ MA}$) in plasma with reactor-level densities and temperatures, either on the "steady-state tokamak experiment" or on existing tokamak facilities.

Sources of Information (plasma science)

DT experiments in large tokamaks (TFTR, JET).

Advanced-geometry tokamak experiment.

Steady-state tokamak experiment or its equivalent.

Tokamak experiments on optimized rf heating.

Tokamak experiments on disruption suppression or control.

Outcomes and Consequences of Decision

The degree to which advanced features such as high-beta and/or steady-state operation can be incorporated into the long-burn facility has important consequences for the performance, cost-effectiveness, and productivity of this facility.

Incorporation of high-beta operation will increase the neutron flux and will enhance the reactor prototypicality of the facility.

Incorporation of steady-state current drive will decrease the cost and increase the availability and neutron fluence, thereby greatly enhancing the usefulness of the facility for nuclear testing.

TABLE 1.9 Decision <e> on Tokamak Reactor Evaluation

Statement of Decision

To evaluate the attractiveness of the tokamak confinement concept for commercial reactor applications.

Decision Criteria

Beta-value and confinement achievable in optimized configuration.
 Current-drive efficiency achievable with optimized technique.
 Level of impurity and particle control achievable in steady-state with optimized technique.
 Success in achieving long-pulse (> 100 s) ignition in tokamak long-burn facility.

Sources of Information

Advanced-geometry tokamak experiment.
 Steady-state tokamak experiment.
 Initial operation of tokamak long-burn facility.
 Tokamak reactor evaluations.
 Plasma and nuclear technology development for the tokamak.

Outcomes and Consequences of Decision

A favorable assessment will lead to the relatively early commercialization of fusion power based on the tokamak.
 An unfavorable assessment will lead to delays in the commercialization of fusion power, for which a superior confinement concept will have to be developed.

application of high-power, pulsed ECH sources to high-field tokamaks is also an area deserving of experimental investigation. Total costs in this area will be in the range of \$60-70 million, spread over the late 1980s and early 1990s.

The short-pulse ignition experiment and the long-burn facility are discussed in Chapter 2 and are described in Tables S.11 and S.13 in the Synopsis.

1.1.2 Tandem Mirror

As illustrated in Fig. 1.13, the tandem-mirror program comprises two lines of endeavor. The first develops the basic physics of axial and radial tandem mirror confinement in quadrupole-stabilized magnetic geometries. The second develops improvements in the magnetic geometry for a more economically attractive reactor.

TABLE 1.10 Large Tokamak Facilities

Characteristic	TFTR	JET	JT-60	T-15
Location	USA	EC/UK	Japan	USSR
Experimental Start	1982	1983	1985	1987
Major Radius (m)	2.5	3.0	3.0	2.4
Minor Radius (m)	0.85	1.25	0.95	0.70
Elongation	1.0	1.6	1.0	1.0
Toroidal Field (T)	5.2	3.5	4.5	5.0
Plasma Current (MA)	3.0	5.0	2.7	2.3
Auxiliary Heating (MW)	30	40	30	30
Heating Pulse (s)	2	10	5	>1
Heating Methods	Neutral Beam (NB) ICRH	ICRH NB	NB LHH ICRH	ECH NB
Working Gas	H,D,DT	H,D,DT	H,D	H
Special Features	Adiabatic compression Tangential NB	D-Shape	Outer divertor	Super-conducting coils
Program Emphasis	Confinement at high nT DT breakeven	Confinement at high β High-power rf Plasma shaping Alpha physics	Confinement at high nT High-power rf Divertor	ECH Plasma control

TABLE 1.11 Other Major Tokamak Facilities

Characteristic	DIII-D	PBX-U ^a	C-Mod ^a	ASDEX-U ^a	Tore-S ^a	FT-U ^a	TEXTOR
Location	USA	USA	USA	FRG	France	Italy	FRG
Experimental Start	1986	1987	---	1988	1987-88	1988	1984
Major Radius (m)	1.7	1.5	0.65	1.6	2.3	0.95	1.8
Minor Radius (m)	0.7	0.3	0.2	0.5	0.7	0.3	0.5
Elongation	1.8	2.2	1.8	1.4	1.0	1.0	1.0
Toroidal Field (T)	2.2	2.2	9.5	3.9	4.5	8.0	2.6
Plasma Current (MA)	5.0	0.6	3.7	2.0	1.7	1.6	0.5
Current Pulse (s)	10	1.5	1.0	6	30	1.5	5
Heating Methods	NB ECH ICRH	NB	ICRH ECH	NB ICRH	NB LHH ICRH	LHH	ICRH
Special Features	High elongation	Bean shaping	Divertor	Divertor	Superconducting		Pumped limiter
Program Emphasis	Beta Confinement	Second stability regime Beta	High field rf	Confinement Impurity	Long pulse	High field rf	Impurity control

- ^aPBX-U - Princeton Beta Experiment Upgrade
C-Mod - Proposed high field tokamak at MIT
ASDEX-U - Axially Symmetric Divertor Experiment Upgrade
Tore-S - Superconducting tokamak at Cadarache, France
FT-U - Frascati Tokamak Upgrade.

FIGURE 1.13 Level 2 Logic Diagram for Tandem Mirrors

Issues And Objectives of the Tandem Mirror Program

Macroscopic Equilibrium and Dynamics. On the basis of the existing Tandem Mirror Experiment Upgrade (TMX-U) and the Mirror Fusion Test Facility (MFTF-B)* at LLNL, and relying on results from the Japanese Gamma-10 facility, tandem mirror physics issues will be explored through a succession of $n\tau$ values up to 10^{19} s/m³. The TMX-U and Gamma-10 are limited to $n\tau < 10^{17}$ s/m³. The MFTF-B is foreseen to operate in a phased manner: first, with 0.5-s pulses and beams capable of plugging potentials of 6-8 keV for $n\tau < 10^{18}$ s/m³; and second, with 30-s pulses and beams capable of the full design potential of 20 keV for $n\tau < 10^{19}$ s/m³. The full test of beta limits must await the full-scale operation of MFTF-B. The calculated central-cell, volume-average beta limit to ballooning modes is <20%, depending on details of the anchor plugs.

Transport. Radial transport is a special issue in these quadrupole-stabilized geometries. Because it competes with axial loss, a given level of radial confinement can be studied only in a device having similar or longer axial confinement. Successive experiments are, therefore, increasingly sensitive to this loss. Even then, in MFTF-B the calculated confinement times of a few tenths of a second can only be achieved using some control of the radial potential profile by means of biased end walls. To form the potentials necessary to limit axial loss under full operation, the MFTF-B will require (as will a reactor) some means for pumping the thermal barriers other than the neutral beams used in current experiments. Drift pumping is the current method of choice.

Particle-Plasma Interactions. Questions concerning startup will be substantially addressed in the early operation of MFTF-B. Fueling by low-energy beams, and possibly by pellets, will come somewhat later.

Wave-Plasma Interactions. Both ICRH and electron cyclotron resonance heating (ECRH) are used in tandem mirrors (see Fig. 1.14). The ICRH mode is used for heating during plasma startup and to build a weakly anisotropic density distribution in the central cell, thereby increasing the central-cell density held by the plugs. The issues are the important details of wave-coupling efficiency, antenna design, power deposition profiles, etc. The ECRH mode is used to form the hot electrons in the thermal barrier and to heat the plug electrons. The issues are the same as those for ICRH, with additional questions concerning the microstability of the resulting non-Maxwellian distributions. In MFTF-B, high-power ICRF and ECRH heating will be necessary to form the large thermal barrier plugging potentials.

In the area of microstability (see Fig. 1.14), in addition to the stability of hot electron distributions, the principal issue will be to continue achieving stable plug-ion

*This facility is currently in a "mothballed" status.

FIGURE 1.14 Level 3 Logic Diagram for Wave-Plasma Interactions in Tandem Mirrors

distributions as higher plasma parameters are obtained. Drift pumping of thermal barriers, or another pumping technique, will be required for a thermal barrier tandem-mirror reactor, although it will not be required in MFTF-B until the time of 30-s operation.

Composite. The physics research efforts discussed above are designed to develop the confinement properties of tandem mirrors having near-reactor parameters. However, system studies have indicated that the quadrupole end cells are not efficient for reactor applications. As a result, the second endeavor of the program is to develop a magnetic geometry in which the ratio of end-cell to central-cell volumes is much reduced below that obtainable with quadrupole-stabilized end cells, while preserving macroscopic stability. The TARA, Phaedrus, and possibly a modified version of the Tandem Mirror Experiment (TMX) will explore several approaches. More details relevant to this area are given in Fig. 1.15.

Plasma Support Technology. Most elements of technology required for mirror fusion are the same as for other approaches. However, there are a few additional requirements (Fig. 1.16) and some changes in emphasis:

Magnets. Tandem mirrors require a choke coil between the center cell and the end plugs. These choke coils are composed of a superconducting coil with a field of 16-20 T, in which a copper hybrid coil is inserted. High-strength copper with radiation-resistant ceramic insulators is needed for the inserted coil.

Neutral Beams. Sloshing ions in the tandem mirror end plugs provide microstability. Because of the high plasma potential, the beams must have voltages up to 500 kV with currents of 5-10 A. Such beams can best be produced with negative ions. Preliminary work has achieved 1-A beams at 80 kV. Next, a current of several amperes at 170 kV should be achieved before using a transverse-field-forcing accelerator to reach the full voltage. Photodetachment neutralizers are expected to be more efficient than gas or plasma cells to strip the extra electron from the ion before injection into the tandem mirror.

Electron Cyclotron Heating. The major method of plasma heating and potential control in a tandem mirror is through ECH. Power in excess of 25 MW requires that present gyrotron tubes of 200-kW power be substantially enlarged or replaced by a free-electron maser.

Direct Conversion. For high energy efficiency, direct conversion of electron and ion losses out the ends of a tandem mirror is desirable. As observed in the TMX, the electrons leak axially, while the ions leak radially to the halo plasma. Such a natural charge separation can lead to cheap power conversion. Recycling of the halo plasma in the direct converter maintains the proper densities for impurity control and acts

FIGURE 1.15 Level 3 Logic Diagram for Configuration Optimization in Tandem Mirrors

FIGURE 1.16 Level 3 Logic Diagram for Mirror Technology

as an efficient vacuum pump. Early versions of the direct converters can be tested on the TMX-U and on the MFTF-B.

Tandem Mirror Program Logic for Reactor Development

Being an open magnetic geometry, the tandem mirror has end-loss plasma plugging as the principal step in its development logic. The thermal barrier end cell appears to be the most efficient means for achieving this plugging. As axial confinement is increased in successive experiments, radial confinement of both particles and energy can be explored on longer time scales. Because of its axial nonuniformity, the tandem mirror is amenable to considerable flexibility in its magnetic design; the optimum flexibility is determined by the criteria of beta limits, transport rates, and plug volume. The interplay between geometry and confinement properties determines the essential tandem mirror developmental issues. Tandem mirror program logic for reactor development is shown in Fig. 1.17. Key decision points for this area are described in Tables 1.12 and 1.13.

Macroscopic Equilibrium and Dynamics. Current experiments, designed to produce early results on thermal barrier end plugging, have used average-minimum-B quadrupole geometries to assure magnetohydrodynamic (MHD) stability. These geometries are sufficient for the near term but are seen to be inadequate in reactor applications. The task in the macroscopic area is to develop a means for stabilizing an axisymmetric magnetic geometry to the beta values of 25-50% required in a tandem mirror reactor. Theory indicates that these beta values will be realizable, provided interchange stability can be assured. Several solutions to the stability issue are being explored.

Transport. The achievable plugging potentials are limited by the energy of the ions that are mirror-trapped in the plugs. Thus, improvements in confinement require increases in neutral beam energy, with the associated conditions for plug-ion adiabaticity. Radial ion confinement has, to date, been consistent with neoclassical confinement, and this confinement is markedly enhanced in an axisymmetric configuration. Long confinement times in the central cell, which lacks magnetic flux surfaces, will likely require symmetric fueling and heating. Whether electron heat transport proves to be anomalous in linear systems, as it has been found to be in toroidal systems, is an important question to be answered in successive experiments as their axial confinement improves.

Wave-Plasma Interactions. Microstability is a particularly important issue for open confinement. Development of stable particle distributions has been one of the principal accomplishments of mirror research, and continued stability will be vital to its success. Externally applied waves in the form of ECRH would be the principal power input to a reactor, so that efficient deposition would directly affect the Q value. Drift pumping of the thermal barriers will require development.

FIGURE 1.17 Level 2 Logic Diagram for Tandem Mirror for Reactor Evaluation

TABLE 1.12 Decision E₂ for Progressing to Tandem Mirror Integrated Proof-of-Principle Experiment

Statement of Decision

To proceed with operation of the MFTF-B facility as an integrated proof-of-principle experiment. Phase I operation would use 0.5-s beams to explore confinement issues with $n\tau_E \lesssim 10^{18}$ s/m³. Phase II operation would use 30-s, pure 80-keV beams to study confinement issues with $n\tau_E \lesssim 10^{19}$ s/m³.

Decision Criteria

Achieve ~15% beta in a configuration suitable for projection to similar performance in MFTF-B.

Achieve $\tau_{\text{perp}} \sim 10\text{-}20$ ms with scaling to $\tau_{\text{perp}} > 100$ ms in MFTF-B.

Demonstrate startup and thermal barrier formation techniques to central cell density $\sim 10^{19}$ m⁻³.

Develop halo formation and fueling techniques sufficient for $n\tau_E \sim 10^{19}$ s/m³. Establish required impurity level.

Sources of Information

Overall performance of TMX-U, TARA, Phaedrus, and Gamma-10 (Japan).

Stability, plugging, and transport comparisons of quadrupole and axisymmetric geometries.

Extrapolation of results to MFTF-B using models validated by proof-of-concept experiments.

Outcomes and Consequences of Decision

A favorable assessment would carry thermal barrier tandem-mirror confinement studies to near-breakeven conditions, with energy confinement times on the scale of 200-500 ms.

If confinement indicates that $n\tau_E$ will scale to $\sim 10^{19}$ s/m³, then MFTF-B could be upgraded for DT operation to provide a neutron source for nuclear technology and materials testing.

Particle-Plasma Interactions. The hot core of a tandem mirror reactor must be protected from charge exchange on ambient gas by a halo of separately maintained cold plasma of sufficient density to absorb low-energy neutral particles. Fueling of the central cell will likely require high-velocity pellets of sufficiently small mass to avoid markedly perturbing the plug-to-central-cell density ratio.

Composite. Reactor attractiveness, particularly in small unit size, is greatly increased in an axisymmetric geometry having low transport and permitting small-volume plug cells. Provided a means for interchange stability can be developed, all criteria point toward an axisymmetric machine as an optimal configuration. Because of

TABLE 1.13 Decision E₆ for Progressing to Tandem Mirror Reactor-Level Test

Statement of Decision

To upgrade MFTF-B to reactor-level performance, possibly including ignition, in a magnetic geometry that can be projected to an attractive reactor.

Decision Criteria

Demonstrate beta limits consistent with theory in MFTF-B and make projections of beta of 25-50% in MFTF-Upgrade.

Demonstrate a total confinement parameter $n\tau_E \sim 10^{19} \text{ s/m}^3$; have demonstrated ion transport times of 0.5-s in MFTF-B and have projections to 2-s in MFTF-Upgrade; have demonstration of $\chi_e < 1 \text{ m}^2/\text{s}$.

Develop ECRH heating techniques under long-pulse operation, with efficiencies that can be projected to attractive reactor performance; develop drift pumping of thermal barriers.

Develop halo and fueling techniques sufficient for reactor-density plasmas.

Develop vacuum and impurity control for thermal barriers in long-pulse operation.

Develop a magnetic geometry that can be projected to an attractive reactor from considerations of plug-volume, stability, and transport. Establish confinement control that will lead to burn control in reactor level experiments.

Sources of Information

Experimental results from 0.5-s MFTF-B operation.

Experimental results from TARA, TMX-U, and their possible modifications.

Experimental results from tandem-mirror facilities in Japan and the USSR, as available.

Evaluation of MFTF-Upgrade and reactor performance based on theoretical models validated by MFTF and other facility operation.

Outcomes and Consequences of Decision

A favorable decision would imply that the tandem mirror is a serious reactor candidate, lacking only a confinement test at or near ignition.

the influence of the magnetic geometry on the confinement characteristics, confinement studies and optimization of the geometry are conducted in parallel.

Tandem Mirror Program Logic for Nuclear Technology Application

The mirror program has focused on the thermal barrier configuration as the best approach to achieve high-Q operation of a tandem mirror. For the lower-Q operation required for neutron production, the complexity of end-loss reduction is considerably

reduced. The reduction of end-losses by raising the end-plug density to equal the central-cell density (the Kelley mode) is employed. This eliminates the velocity-space hole of the large-volume central cell, increasing the overall system confinement over that of a simple mirror. For neutron production, a local mirror is located in the central cell and fueled by high-energy neutral beams. This high-density plasma is stabilized by the background plasma of the Kelley tandem mirror. This program can be carried out in the MFTF-B facility.

In order to build upon the data base of TMX-U and TARA, MFTF-B would first be operated in its 0.5-s Kelley mode. The facility could then be modified to DT operation. Principal advantages of this approach are the low tritium cost and near-term readiness of the mirror neutron-source approach. Tandem mirror program logic for the neutron-source application is shown in Fig. 1.18. The key decision-point description for this area is given in Table 1.14.

Macroscopic Equilibrium and Dynamics. Through the elimination of the thermal barrier, the simplification of the Kelley mode greatly reduces the requirements for potential control. The principal requirement is to obtain $\beta \sim 40\%$ in the central mirror cell, a beta which has been achieved at low magnetic field.

Transport. The transport required is $n\tau_E = 1 \times 10^{19} \text{ s/m}^3$ for the neutron-source application. This will require radial potential control for quadrupole-stabilized machines (as has been documented on the TMX-U), but this should be readily achieved in axisymmetric machines. The higher-density, lower-potential operation of the Kelley mode is favorable for transport.

Wave-Plasma Interactions. Microstability is the major physics issue to be evaluated. At higher density, ion cyclotron instabilities have larger growth rates and will need to be evaluated in 0.5-s operation of the MFTF-B. Startup issues are simplified through the elimination of barrier formation. Thus, rf startup procedures for thermal barriers can be used in a simplified form.

Particle-Plasma Interactions. The Kelley mode has an overall positive potential with no local minimum. Thus, impurities are lost axially and are not expected to accumulate. Vacuum requirements are reduced through high-density operation and the absence of the thermal barrier.

Major Tandem Mirror Facilities

Several techniques for achieving macrostability in more nearly axisymmetric end cells have been proposed and are under experimental investigation. Each involves important physics issues and each represents a physics investigation in its own right. The results from these parallel investigations should be available in advance of an evaluation

FIGURE 1.18 Level 2 Logic Diagram for Tandem Mirror for Nuclear Technology Testing

TABLE 1.14 Combined Decisions E_4 and E_5 for Proceeding with a Tandem Mirror Nuclear Technology and Materials Test Facility

Statement of Decision

To upgrade MFTF-B for DT operation to serve as a moderate-fluence neutron source for nuclear technology and materials testing.

Major Decision Criteria

Demonstrate beta limits of 40% in central-cell mirror.

Demonstrate transport rates of $n\tau_E \sim 10^{19} \text{ s/m}^3$.

Demonstrate start up technique for Kelley-mode operation.

Demonstrate impurity control in long-pulse, high-power operation.

Sources of Information

Overall performance of TMX-U, TARA, Phaedrus, and Gamma-10 (Japan).

Neutron production studies in mirrors (U.S. and Japan).

0.5-s operation of MFTF-B.

Outcomes and Consequences of Decision

Decision to proceed with neutron production in a mirror device requires modification of medium-scale facilities for stability evaluation.

Modification of MFTF-B magnet geometry is required.

Proceed with physics test in MFTF-B, followed by machine modifications for DT operation.

in the early mid-1990s. At that time, the results from these tests will be combined with system studies results and confinement physics data from MFTF-B, selecting the best approach for an improved end-cell design to be employed in the next stage of the program. At present, the mirror program has one operating major facility,* TARA at the Massachusetts Institute of Technology (MIT), and one support experiment, Phaedrus, at the University of Wisconsin. Mirror issues are also addressed on the Constance (MIT) and Lamex (University of California at Los Angeles) mirror experiments. Foreign facilities are the Gamma-10 (Japan) tandem mirror, RFCXX (Japan), and Ambal II (USSR).

The TARA facility at MIT will be able to test several of these approaches. The outboard anchor uses the basic TARA geometry, placing the MHD anchor outside the plugging potential so that none of the central ions see the nonsymmetric fields. Provided this geometry is not adversely affected by the predicted trapped-particle mode, it

*The MFTF-B (LLNL) has been mothballed, and the TXM-U (LLNL) has been recently shut down.

represents the most direct route to an improved reactor. Hot-electron rings employ the same basic mechanism as that used successfully in the Elmo Bumpy Torus (EBT); at issue here is the predicted destabilization of the rings through coupling to Alfvén waves in the central cell. Wall stabilization relies on image currents in the wall driven at finite central cell beta values ($>50\%$); here, the issues are startup and wall resistivity.

The Phaedrus machine at the University of Wisconsin has demonstrated macrostability at low temperature but significant beta (18%) using a radial gradient of ponderomotive ICRF pressure. The issues are the scaling of power required at higher plasma temperature and the degree of ICRF power absorbed by the plasma.

A higher-multipole end plug causes less azimuthal distortion near the axis and, consequently, less transport. An octupole-end-cell experiment has been proposed for the TMX facility. The issues involve the degree of control of ECRH required to generate a high-pressure hot-electron mantle in the end cells.

Existing Facilities. TARA has a 10-m-long, 0.4-m-diameter central cell employing axisymmetric plug cells. Typical partially plugged parameters are $n_c \sim 3.5 \times 10^{18}/\text{m}^3$, $T_{i\perp} = 500$ eV, $T_{i\parallel} = 150$ eV with $n\tau_E \sim 1 \times 10^{16}$ s/m³. Strong plugging experiments have just begun at $n_c \sim 1 \times 10^{17}/\text{m}^3$. This machine is developing magnetic optimization and axisymmetric plugging. Future upgrades could include higher density rf driven operation for $t > 100$ ms, fueling, halo generation, pumping in axisymmetric thermal barriers, and impurity control.

TMX-U has led the development of the thermal barrier concept in the U.S. The device has an 8-m-long, 0.5-m-diameter central cell. Typical plugged parameters are $n_c \sim 2 \times 10^{18}/\text{m}^3$, $T_{i\perp} = 400$ eV, $T_{i\parallel} = 150$ eV with $n\tau_E \sim 1 \times 10^{17}$ s/m³. This device could develop the quadrupole-stabilized tandem mirror to the $n_c \sim 1 \times 10^{19}/\text{m}^3$ plugged central-cell regime. The machine could serve to develop additional physics required for the POP reactor regimes and nuclear-test facility. This includes drift pumping in quadrupole geometry, Kelley-mode operation, fueling, impurity control, potential control, and magnetic optimization in nonaxisymmetric magnetic fields.

The POP facility is the MFTF-B. Its operation could consist of two phases: 0.5-s operation to achieve $n\tau_E \sim 1 \times 10^{18}/\text{m}^3$ and 30-s operation with $n\tau_E \sim 1 \times 10^{19}$ s/m³. Phase I would build on the physics of TARA, TMX-U, and foreign experiments into the higher temperature, density, and confinement regimes where issues of long pulse, microstability, transport, and impurity control will be extended. Phase II would extend the parameters to $n\tau_E \sim 1 \times 10^{19}$ s/m³. New physics required for phase II, in addition to data from Phase I, will be drift pumping and magnetic optimization from the base program.

New Facilities (>\$10 million). Reactor-level tests may be carried out through a modification of the MFTF-B facility. The magnetic optimization program will provide data on magnet-design improvements required to make the facility lead to a desirable reactor product. Alternatively, the upgrade could be for DT operation after phase I of the MFTF-B with the nuclear mission objective, at an estimated cost of \$400 million with site credits.

In the early to mid-1990s, the best of the alternative-geometry approaches would be selected to be combined in an upgrade of MFTF-B. This combination would be designed to exploit the confinement physics developed earlier in MFTF-B, in combination with the benefits of the improved end cell geometry. The nature of this step would depend on the projections of earlier results into the reactor regime, abetted by system study results. If the prognosis were favorable, a device having $Q > 2$ or higher would be called for. With a less favorable prognosis for the thermal barrier reactor, the conventional tandem mirror (without thermal barriers) might offer an attractive approach for modest- Q applications, such as a nuclear technology testing facility, a fusion-fission hybrid, or a neutron source for tritium production.

1.1.3 Stellarators

The stellarator concept is closely related to the tokamak and the reverse field pinch; all three devices are characterized by helical (toroidal plus poloidal) magnetic fields that form toroidally nested, closed magnetic surfaces. Plasma stability can be obtained through both an average magnetic well and magnetic shear. However, stellarators can create the helical field from currents flowing entirely in external conductors, while tokamaks and RFPs require internal plasma currents for confinement. This advantage of the stellarator allows steady-state operation and external control of the magnetic configuration properties (rotational-transform shear, magnetic-well depth, second-stability access, plasma shape, magnetic-axis topology, etc.).

Issues and Objectives of the Stellarator Program

The major issues to be addressed in stellarator research are summarized in Table 1.15. These issues are discussed under the headings of major physics areas below.

Macroscopic Equilibrium and Dynamics. The two key objectives in this area are as follows: (1) demonstration of the ability to maintain a high-beta plasma with good confinement and stability properties in the second stability regime on a long (skin-diffusion) timescale and (2) an understanding of the dependence of the equilibrium and stability boundaries on magnetic-configuration parameters that allows accurate predictions of beta limits. Volume-averaged beta values ($\langle\beta\rangle$) of 2% have been obtained in the first stability regime in Heliotron-E, and $\langle\beta\rangle$ of $\geq 8\%$ are projected in the second stability regime for the Advanced Toroidal Facility (ATF). Understanding of the role of plasma current and the rotational-transform profile will be enhanced by comparison with related tokamak results.

Transport. It is important to understand transport scaling in stellarators at low collisionality and high beta, as well as the role of the ambipolar electric field, to allow extrapolation to reactor conditions. This will require demonstration of reactor-level confinement, control of the ambipolar electric field, and minimization of anomalous transport. Confinement parameters ($n\tau_E$) of 3×10^{18} s/m³ have been obtained in

TABLE 1.15 Stellarator Program Elements, Subelements, Objectives, and Attributes

Program Elements and Subelements	Objectives	Attributes
Macroscopic Equilibrium and Dynamics		
Pressure-Driven Effects	Understand beta limits: second stability region.	Predictive capability within 10%
	Achieve reactor-relevant beta value.	$\langle \beta \rangle > 8\%$
Transport		
Perpendicular Energy Confinement	Understand scaling of trans- port at low collisionality; ambipolar electric field.	Predictive capability within 20%
	Demonstrate reactor-level confinement.	$\chi_E \lesssim 0.5 \text{ m}^2/\text{s}$
Wave-Plasma Interactions		
Radio-frequency Heating	Develop launchers and predic- tive capability.	Predictive capability within 30%
	Achieve reactor-level ion temperature at high density.	$T_i > 10 \text{ keV};$ $\frac{T_i}{n_e} > 10^{20}/\text{m}^3$
Particle-Plasma Interactions		
Impurity Control	Demonstrate adequate impurity- removal rate.	$P_{\text{rad}}/P_{\text{tot}} \lesssim 20\%$
Alpha-Particle Effects		
Alpha-Particle Containment	Predict alpha loss, energy deposition, and effects on plasma.	Predictive capability
Alpha Heating Effects	Achieve significant alpha heating.	$Q > 10$

TABLE 1.15 (Cont'd)

Program Elements and Subelements	Objectives	Attributes
Burn Control and Ash Removal		
Ash Removal	Demonstrate ash removal.	n_α/n_e ; tritium burnup
Composite Issues		
Configuration Optimization	Optimize combination of reactor configuration parameters.	Level of rotational transform, shear, well, helical axis

Heliotron-E, and higher values are projected for ATF and Wendelstein VII-AS. Large electric fields have been inferred from W VII-A data; understanding of the role of the ambipolar electric field and magnetic-field ripple will be enhanced by comparison with related tokamak results.

Wave-Plasma Interactions. Radio-frequency heating (ECH, ICH) is important for currentless plasma formation and subsequent heating to ignition. Attainment of reactor-level plasma temperatures at high-density must be coupled with high-efficiency heating, use of launchers that do not introduce unacceptable impurity influxes, and ability to predict power-deposition profiles. First- and second-harmonic ECH has been used for currentless plasma formation and electron heating to T_e values of 1-2 keV in Heliotron-E and W VII-A, and ICH has produced ion-temperature increases $\Delta T_i \sim 0.5$ keV in Heliotron-E. Studies will be extended from the present 0.2-0.5-MW levels to levels of several megawatts in Heliotron-E, in the stellarator device at the Max Planck Institute of Plasma Physics (W VII-AS), and in ATF.

Particle-Plasma Interactions. A key objective in this area is development of pumped limiters, divertors, and other techniques to prevent impurity accumulation and to minimize Z_{eff} and the ratio of radiated power to total power ($P_{\text{rad}}/P_{\text{tot}}$). This will also require the development of proper materials for the first wall and other plasma-interactive components that minimize impurity effects under steady-state, high-heat flux conditions; steady-state pellet injection and gas fueling for profile control; confinement improvement; and impurity source reduction. Maximum values of density ($\bar{n}_e \sim 1.4 \times 10^{20}/\text{m}^3$) and n_{T_E} are obtained with pellet injection, but impurity-radiation losses remain high.

Alpha-Particle Effects. Long-term needs are to understand alpha-particle confinement and heating in stellarators and the effects of significant alpha-particle densities on the bulk plasma equilibrium, stability, transport, and electric fields. This may require stellarator-specific information beyond that generated in the tokamak program.

Burn Control and Ash Removal. Key issues in this area are the demonstration of an effective temperature-control system and an ash-removal system that can maintain n_{He}/n_e at an adequate level. The natural divertor (separatrix) configuration of stellarators will be exploited in modified pumped-limiter/divertor tests in various stellarators.

Composite Issues. The overall objective is to demonstrate controlled steady-state operation at high beta values in the second stability regime, with good confinement at low collisionality and low P_{rad}/P_{tot} ratio in an optimized magnetic configuration that would have low or zero recirculating power to the plasma in a reactor. This demonstration will require establishing the role of the key parameters (rotational transform, shear, magnetic well, electric field, helical axis, etc.) and their radial profiles in an overall configuration optimization that includes MHD, transport, heating, and impurities. An important element is the understanding of plasma evolution at times long compared with that for resistive diffusion or plasma-wall equilibrium. The ATF, with its flexible helical and poloidal field systems and 1-T steady-state toroidal field capability, should contribute significantly to the overall understanding of toroidal confinement when combined with related long-pulse tokamak studies.

Stellarator Program Logic

The stellarator program logic, shown in Fig. 1.19, follows the ATF approach to lower plasma aspect ratio and relies on input from foreign stellarators at key evaluation points. There are four main steps in the evolution of the stellarator concept: (1) optimization of the stellarator magnetic configuration, (2) steady-state hydrogen (H) operation at reactor-level parameters, (3) short-pulse DT operation at $Q \geq 10$, and (4) steady-state, ignited DT operation.

The main facilities for these studies are as follows: (1) the ATF, a copper-coil torsatron with $R = 2.1$ m and $\bar{a} = 0.3$ m, capable of 2-T operation for 5-s pulses and 1-T steady-state operation; (2) an LSE (large stellarator experiment), a steady-state superconducting-coil experiment with $R \approx 4$, $\bar{a} \approx 0.5$ m, and $B \approx 4$ T; (3) an SIE (stellarator ignition experiment), a copper-coil torsatron with $R \approx 5$ m, $\bar{a} \approx 1$ m, and $B \approx 5$ T, capable of pulsed DT operation ($Q \geq 10$) and steady-state H operation at lower field; and (4) an SITR (stellarator ignition test reactor), an ignited DT, steady-state, superconducting-coil torsatron with $R \approx 10$ m, $\bar{a} \approx 2$ m, and $B \approx 5$ T. The ATF and a complementary experiment (W VII-AS) will begin operation in 1987 and are described in Table 1.16. The two key new facilities -- an LSE of the type analyzed by Japan (Heliotron-H), Germany (W VII-X), and the U.S. (ATF-II); and an SIE, which may be

FIGURE 1.19 Level 2 Logic Diagram for Stellarator

required if our knowledge of alpha physics, burn control, and ash removal from tokamak experiments cannot be extrapolated to stellarators -- are also discussed in Table 1.16. Tables 1.17 and 1.18 list the relevant points for the decisions to proceed with an LSE or SIE device.

In the near term (1987-92), the focus will be on the key plasma science program elements discussed earlier under "Issues and Objectives of the Stellarator Program." The supporting university stellarator program will focus on particular basic physics issues, such as equilibrium (bootstrap and Pfirsch-Schlüter currents), edge physics, divertors, and the role of ambipolar electric fields; wave propagation and ICH; and high-beta equilibrium and stability of linear helical-axis systems.

Figure 1.20 illustrates the critical composite issues for the stellarator program: magnetic-configuration optimization, profile optimization, and steady-state operation. The most important area, configuration optimization, is being addressed through international cooperation using complementary approaches. Japan is following the Heliotron approach, which features high rotational transform and high shear with no vacuum magnetic well. Euratom is following the Wendelstein modular-stellarator approach, which features lower transform and no shear with a magnetic well. Another Euratom approach is the proposed TJ-II flexible heliac, which also has no shear but obtains high transform with a helical magnetic axis. The U.S. approach (ATF) is similar to Japan's, but it features lower plasma aspect ratio and a magnetic well that deepens with beta value to obtain direct access to a higher-beta second stability regime; the ATF, which also uses poloidal field shaping to cover a wide range of magnetic configurations, is supplemented by smaller basic-research facilities. This combination of approaches is sufficiently broad to cover the key plasma science program elements.

The key issues for magnetic-configuration optimization are beta limits and transport. The logic for development of these plasma science program elements is given in Fig. 1.21. The key issues for macroscopic equilibrium and dynamics involve the scaling of beta limits and the plasma behavior in the second stability regime. Foreign stellarators are not optimized for high beta operation, so this information must come from the U.S. program. However, the input from the shearless Euratom experiments and the high-transform, high-shear Heliotron-E experiment will be valuable for understanding the configuration dependence of beta limits. The key issues for transport are low-collisionality transport and the role of the ambipolar electric field. All large, high-power experiments will contribute to this understanding. The unique flexibility of the ATF magnetic configuration and its steady-state capability at 1 T, together with the information from foreign programs, will be essential in developing the optimum stellarator magnetic configuration.

In the intermediate term (1994-98), an LSE would extend the steady-state operating regime to higher beta values ($\langle\beta\rangle = 5-10\%$) and higher ion temperatures ($T_i = 5-10$ keV) with continuous ICH at high power (~ 10 MW) in hydrogen plasma operation. In a possible follow-on experiment (after 2002), an SIE would operate in a pulsed mode in high-Q DT plasma operation to demonstrate reactor-level beta value, rf heating, and impurity control in a near-ignited plasma and evaluate the means for burn control and ash removal in an alpha-dominated plasma.

TABLE 1.16 Major Stellarator Facilities (foreign and U.S.)**Existing Facilities**

- Heliotron-E (Japan) with $R = 2.2$ m, $\bar{a} = 0.2$ m, $B = 2$ T has attained $T_i = 1$ keV, $\langle\beta\rangle = 2\%$, and $n\tau_E = 3 \times 10^{18}$ s/m³; it features high rotational transform and high shear with no vacuum magnetic well.
- Wendelstein VII-AS (Germany) with $R = 2$ m, $\bar{a} = 0.2$ m, $B = 3$ T projects $T_i \sim 2$ keV, $\langle\beta\rangle \sim 2\%$, and $n\tau_E > 3 \times 10^{18}$ s/m³; it features lower rotational transform and no shear with a magnetic well.
- ATF (U.S.) with $R = 2.1$ m, $\bar{a} = 0.3$ m, $B = 2$ T projects $T_i \sim 2$ keV, $\langle\beta\rangle > 5\%$, and $n\tau_E > 3 \times 10^{18}$ s/m³; it has the flexibility to study a wide range of magnetic configurations. The standard configuration with intermediate rotational transform, shear, and a magnetic well should attain high beta values in the second stability regime.

New Facilities**LSB (Large Stellarator Experiment)**

Mission: demonstrate true steady-state hydrogen plasma operation at reactor-relevant parameters of density, temperature, beta value, collisionality, impurity level, etc.

Features: superconducting magnets, continuous-wave (cw) ICH power > 20 MW), steady-state fueling (pellet) and particle-removal (divertor) systems.

Expected values: $R \approx 4$ m, $\bar{a} \approx 0.5$ m, $B \approx 4$ T, $T_i = 5$ -10 keV, $\langle\beta\rangle > 5\%$, $n\tau_E \sim 3 \times 10^{19}$ s/m³.

Construction cost: $\sim$ \$200 million.

Operating cost: $\sim$ \$40 million/yr.

SIE (Stellarator Ignition Experiment)

Mission: demonstrate high $Q \gtrsim 10$) or ignition in long-pulse DT operation with reactor-relevant plasma-interactive components.

Features: copper magnets (no neutron shielding), long-pulse (>30 -s) heating (ICH), fueling (pellets), and particle-removal (divertor) systems.

Expected parameters: $R \approx 5$ m, $\bar{a} \approx 1$ m, $B \approx 5$ T, $T_i > 10$ keV, $\langle\beta\rangle > 5\%$, $n\tau_E > 2 \times 10^{20}$ s/m³.

Construction cost: $\sim$ \$300 million.

Operating cost: $\sim$ \$60 million/yr.

TABLE 1.17 Decision <a> on a Large Stellarator Experiment (LSE)

Statement of Decision

To proceed with construction of a large stellarator experiment for which the primary mission is to demonstrate true steady-state hydrogen plasma operation at reactor-relevant parameters.

Decision Criteria

Demonstrate beta limits $> 3-5\%$ and determine scaling with configuration parameters.
 Demonstrate $n_{T_E} > 3 \times 10^{18} \text{ s/m}^3$, and determine scaling at low collisionality and role of ambipolar electric field.
 Demonstrate effective ECH and ICH.
 Develop methods for particle and impurity control.
 Demonstrate long-pulse operation and determine optimum configuration parameters.
 Assess superconducting coil and ICH antenna development.
 Assess reactor potential.

Sources of Information

Heliotron-E, W VII-AS, ATF experiments.
 Plasma-technology tests of superconducting coils, ICH antenna, pump limiters, materials.
 Conceptual reactor designs based on chosen configuration.

Outcomes and Consequences of Decision

Favorable assessments and achievement of the primary objectives of the Large Stellarator Experiment would lead to decisions on a Stellarator Ignition Experiment and a Stellarator Ignition Test Reactor.
 Unfavorable assessments would lead either to smaller exploratory experiments to address deficiencies or to termination of the program.

Many of the technology requirements for stellarators are similar to those for the related tokamak program. Some examples are pellet fueling, high-power rf (ECH, ICH, and IBWH) modules, superconductor development, divertor targets, and first wall materials and cooling. However, there are also some differences with regard to program timing and unique stellarator features. Stellarators are inherently steady-state devices, with no net toroidal plasma current required, so there is no need for current-drive technology, pulsed fields, or disruption protection. Problems associated with steady-state fueling, heating, and power/particle removal must be addressed at an earlier stage than for tokamaks. The rf launcher geometry will be different because of the more complex magnetic configuration, and the pumped-limiter/divertor geometry must be matched to the stellarator field-line geometry (separatrix). Superconducting coils will have to be nonplanar (distorted TF coils); or, if helical coils are used, they will have joints or be wound as single coils with redundant features to ensure reliability.

TABLE 1.18 Decision on a Stellarator Ignition Experiment (SIE)

Statement of Decision

To proceed with construction of a stellarator ignition experiment for which the main objective is to demonstrate high Q or ignition in long-pulse DT operation.

Decision Criteria

Demonstrate beta limit $> 5\%$ and determine scaling with configuration parameters.

Demonstrate $n\tau_E > 10^{19} \text{ s/m}^3$ and determine low-collisionality scaling.

Demonstrate steady-state operation and determine optimum configuration.

Demonstrate need for a stellarator ignition device after tokamak ignition demonstration.

Assess plasma-interactive components and tritium systems.

Develop attractive reactor concept.

Sources of Information

Tokamak short-pulse ignition demonstration and Large Stellarator Experiment

Plasma-technology tests of plasma-interactive components and tritium systems

Conceptual reactor-design studies.

Consequences of Decision

Favorable assessments and achievement of the primary objectives of the Stellarator Ignition Experiment, combined with the results of the Large Stellarator Experiment, would lead to a decision on the Stellarator Ignition Test Reactor.

Unfavorable assessments would lead either to downgrading the program to an exploratory level or to termination.

1.1.4 Reversed-Field Pinches

The reversed-field pinch (RFP) is an axisymmetric toroidal device having a high-shear field configuration of poloidal and toroidal magnetic fields. The RFP approach leads to compact reactor designs with moderate field strengths and variable (moderate-to-high) aspect ratios. The concept also holds promise for a low-frequency, steady-state current drive and ohmic heating to ignition. In addition, impurity control by high-coverage poloidal pumped limiters or nonlinked TF magnetic divertors is possible.

Favorable $n\tau_E$ scaling and beta values from the ZT-40M and OHTE (U.S.), ETA BETA II (Italy), and TPE-1R (Japan) RFP experiments led to a decision to build the Confinement Physics Research Facility (CPRF) at the Los Alamos National Laboratory (LANL). The primary goal of the first experiment in this new facility will be to test the extension of the present favorable RFP scaling results for $n\tau_E$, beta, and temperature to

FIGURE 1.20 Level 3 Logic Diagram for Composite Issues for the Stellarator

FIGURE 1.21 Level 3 Logic Diagram for Confinement Systems Issues for the Stellarator

the $I_{\phi} = 4$ MA level. Positive results at this intermediate current level would lead directly to the integrated proof-of-principle experiments; these, if successful, would lead in turn to a reactor-level test facility.

The primary issues to be studied for the RFP concept, given in Table 1.19, are discussed briefly below. A more comprehensive table and discussion of the RFP issues, objectives, and attributes can be found in Chapter 1 of Ref. 1.

Technical Issues and Objectives of the RFP Program

Macroscopic Equilibrium and Dynamics. The primary issues in this area are the maintenance of the present reactor-relevant beta as the current and the minor diameter of the plasma are increased and the requirements on the resistive vacuum wall and conducting shell. Minimization of the conducting shell can simplify the reactor designs.

Transport. The chief issue is the maintenance of the favorable scaling of $n\tau_E$ and temperature as the RFP parameters approach reactor values.

Wave-Plasma Interactions. An attractive feature is the possibility of a low-technology steady-state current drive. For the RFP the favored approach, which involves wave-plasma processes, is the low-audio-frequency, oscillating-field technique (F- θ pumping) currently being explored on the ZT-40M. Backup approaches include those being explored for the tokamaks. The results of ongoing experiments will determine whether the RFP reactor will be steady-state or long-pulse. The high ohmic heating in the RFP replaces the wave heating approach being studied in some other concepts.

Particle-Plasma Interactions. Particle-plasma interaction issues, which are important to the continued favorable scaling to a compact and economical reactor, concern the first wall, the fueling, the plasma shell, and the current drive for steady-state operation. The first wall, with its high surface-energy flux, requires a design that resists neutron damage and limits the release of impurities into the plasma to acceptable levels. This requirement leads to the study of the effects of limiters and divertors on the containment and global-stability properties of the RFP configuration. Various techniques of fueling (e.g., gas and pellet injection) are to be studied to determine the optimum method for the RFP. The plasma shell design is important because of its role in controlling field errors, which can lead to local wall-plasma interactions. The possible enhancement of wall-plasma interactions produced by the oscillating fields for current drive is also an important issue.

TABLE 1.19 Reversed-Field-Pinch Program Elements, Subelements, Objectives, and Attributes

Program Elements and Subelements	Objectives	Attributes
Macroscopic Equilibrium and Dynamics		
Global Equilibrium and Stability		
Current- and Pressure-Driven Effects	Maintain reactor-relevant beta as reactor values of the temperature, density, and size are approached. Retain acceptable equilibrium at reactor plasma parameters.	$\langle \beta \rangle$
Transport		
Perpendicular Confinement	Obtain acceptable energy losses at reactor conditions.	$n\tau_E$
Wave-Plasma Interactions		
Current Drive by Helicity Injection	Sustain suitable reactor temperature, plasma density, and confinement time under steady-state current drive.	$T, \bar{n}, n\tau_E,$ steady-state mean current
Particle-Plasma Interactions		
Plasma Fueling	Demonstrate reactor-relevant plasma fueling and adequate profile control.	Sustained $n,$ T_i, T_e profiles
Impurity Control	Demonstrate adequate impurity-control techniques with reactor-relevant first-wall materials.	Z_{eff}
Composite		
Current Drive	Obtain reactor-compatible current drive.	$I_\phi,$ pulse length

RFP Program Logic

The RFP logic diagrams for science and plasma technology are shown in Figs. 1.22-1.24. Two major decision points in the RFP program (labeled E_2 and E_6 at the bottom of the RFP logic diagram) correspond to decision points with the same designation in the Level 0 logic diagram (Fig. S.7). These evaluation points are as follows:

- E_2 : Decision point on the readiness of the RFP 4-MA experiment for an upgrade to make the integrated proof-of-principle tests and to proceed with nuclear technology testing of the components associated with the RFP concept
- E_6 : Decision point on making the transition in the RFP program to the reactor plasma conditions test phase

These evaluation points are summarized in Tables 1.20 and 1.21.

The fundamental logic of the RFP program is to address the RFP issues summarized above and shown schematically in the logic diagrams through a sequence of U.S. and international experiments, categorized as follows:

- The present U.S. devices: ZT-40M and ZT-P (LANL); OHTE and Multi-pinch (GA Technologies); Reversatron (University of Colorado); and MST (University of Wisconsin).
- The present and future non-U.S. devices (listed below).
- A 4-MA experiment.
- An integrated test experiment made as an upgrade of the 4-MA experiment.
- A reactor-level test.

The RFP is currently being studied on the following non-U.S. experiments: HBTX-1B (Culham, U.K.), ETA-BETA II (University of Padua, Italy), TPE-1RM (ETL-Tsukuba, Japan), REPUTE 1 (University of Tokyo, Japan), STP-3M (IPP-Nagoya, Japan), HIT-1 (Hiroshima University, Japan), STE (Kyoto Institute of Technology, Japan), TORIUT-5 (University of Tokyo, Japan), and CN-RFP (Nihon University, Japan). In addition, the following RFP devices are in various stages of construction: RFX (University of Padua) and TPE-1R(M)15 (ETL, Tsukuba). Characteristics of these devices can be found in Chapter 2 of Ref. 1. The first three devices have been in existence long enough to develop extensive data bases and have contributed to the present RFP state of the art. The results from these experiments are particularly valuable in the area of the scaling of the plasma parameters with size. This data base, when combined with that of the U.S. experiments, significantly enhances the physics and technology knowledge needed for future RFP decision points.

FIGURE 1.22 Level 2 Logic Diagram for the Reversed-Field Pinch

FIGURE 1.23 Level 3 Logic Diagram for Pressure- and Current-Driven Effects in the RFP

FIGURE 1.24 Level 3 Logic Diagram for Plasma Technology for the RFP

TABLE 1.20 Decision E₂ on the RFP Integrated Proof-of-Principle Test Phase

Statement of Decision

To proceed with the upgrading of the 4-MA RFP experiment to allow integrated tests at ignition-like conditions, including fueling, current drive, and impurity control; to proceed with nuclear technology testing of components associated with the RFP concept.

Decision Criteria

Achieve $\beta > 10\%$ for $I_{\phi} \sim 4$ MA, $T \sim 4$ keV, and $n\tau_E \sim 2 \times 10^{19}$ s/m³.
 Determine viability of operation without a conducting shell; if the shell is mandatory, determine the optimum shell design.
 Complete studies allowing selection of a viable first-wall design.
 Study fueling techniques and determine the optimum method.
 Complete steady-state current-drive studies and, if feasible, determine the best technique.

Sources of Information

Existing experimental data from all RFP experiments.
 Theoretical understanding of major physics issues.
 System studies.
 Cost projections for the integrated test device.
 Projected reactor attractiveness.

Outcomes and Consequences of Decision

Begin construction of integrated test device.
 Continue program to contribute to physics understanding and improve the RFP concept.
 Terminate program.

The issues of transport, fueling, field errors, high wall loading, the scaling of energy confinement time and plasma temperature with current and size, and the role of the stabilizing shell and conducting liner are all being addressed by the present RFP devices, both in the U.S. and abroad. The 4-MA experiment will extend the parameters into the region of ignition-like conditions. The present experiments will allow upgrading to permit integrated proof-of-principle experiments by adding current drive and divertors. The cost and time scales are materially reduced by confining the integrated experiments to hydrogen and deuterium plasmas. The predicted temperatures are high enough to allow an assessment of important high-energy charged-particle effects using the deuterium-deuterium (DD) reactions. Successful integrated tests will lead to the construction of a new reactor-level test device.

It is envisioned that the reactor level test will be planned initially for deuterium plasma operation but will be located at a site that allows later upgrading to DT ignition

TABLE 1.21 Decision E₆ on the RFP Reactor-Level Test Phase

Statement of Decision

To proceed with an RFP experiment to carry out a reactor-level test of the RFP concept.

Decision Criteria

Achieve $\beta > 10\%$ at ignition-like parameters.
 Suitable projected reactor.
 Determine feasibility of current drive.
 Attain suitable impurity control.

Sources of Information

Experimental data from integrated test and any other existing RFP experiments.
 Theoretical understanding.
 Reactor studies and cost assessments.
 Plasma technology studies.
 Nuclear technology studies.

Outcomes and Consequences of Decision

Begin construction for reactor-level test.
 Modify the concept approach.
 Maintain the experimental facility to expand generic physics and engineering knowledge.
 Terminate program.

and burn experiments, if these are deemed necessary for a fusion reactor assessment of the RFP concept. The decision on the latter option will depend upon the confidence level on the predictions concerning burning plasmas. This confidence level will be influenced by the RFP alpha particle physics learned from the nonignited DD experiments and by the burning plasma physics knowledge available from tokamaks at that time.

Major U.S. RFP Facilities (> \$10 million)

ZT-40M. The ZT-40M experiment at LANL ($R/a = 1.14 \text{ m}/0.2 \text{ m}$) is one of the intermediate-sized RFP experiments; others include OHTE (U.S.), ETA BETA II (Italy), HBTX-1A (U.K.), and REPETE 1 (Japan). The goal of these experiments is to extend the plasma temperatures and confinement times at least an order of magnitude above previous small RFP experiments and to establish the physics base for the next generation of devices. Currently, ZT-40M is being used to explore temperature and $n\tau_E$ scaling, wall interactions, pellet- and/or gas-injection fueling, and low-frequency oscillating-field

current drive. Present plasma parameters are as follows: $T \leq 500$ eV, $\bar{n} = 1 \times 10^{19} \text{ m}^{-3}$ to 10^{20} m^{-3} , and $\tau_E \leq 0.7$ ms.

OHTE. The OHTE experiment at GA Technologies, Inc., ($R/a = 1.24$ m/0.185 m) is used to study RFP issues similar to those of ZT-40M. In addition, the experiment is used to determine the properties of the normal RFP configurations when augmented with a helical field. Present plasma parameters are: $T \sim 400$ eV, $n < 2.4 \times 10^{19} / \text{m}^3$, and $\tau_E \sim 0.4$ ms.

Four-Megampere RFP. A primary purpose of this experiment is to extend the operating regime of the RFP concept to temperatures of 4 keV and $n\tau_E$ to $2 \times 10^{19} \text{ s/m}^3$ for current durations of >200 ms while maintaining $\beta > 10\%$. The new 4-MA experiment ($R/a = 2.25$ m/0.4 m) is to be located at the CPRF installation at LANL. The RFP configuration is produced by an arrangement of ohmic heating, vertical field, and toroidal field coils. The experiment design will allow flexibility in the choice of vacuum-chamber size and materials. The expected parameters for the operation of the 4-MA experiment are as follows: $T \sim 4$ keV, $\bar{n} = (1 \text{ to } 4) \times 10^{18} / \text{m}^3$, wall load = $2\text{--}5 \text{ MW/m}^2$, $\tau_E \sim 85$ ms, and $n\tau_E = 2 \times 10^{19} \text{ s/m}^3$.

The total construction cost is projected to be \$62.5 million distributed over the 4-yr construction period. The annual operating cost is projected to be \$10 million/yr until upgraded for the integrated POP test.

Integrated Proof-of-Principle Experiment. The projected plasma parameters for the 4-MA experiment are in the range of an integrated POP test, so it is envisioned that this phase of experimentation can be carried out by an upgrading of the 4-MA experiment. The primary upgrading will consist of adding divertors for impurity control, current drive, and efficient fuel injectors. The estimated total cost for this upgrading is about \$20 million, distributed over a 2-yr construction period; the expected operating costs are \$12 million/yr over about a 12-yr lifetime.

Reactor-Level Test. The purpose of this test will be to operate with deuterium plasma to assess the RFP characteristics under fusion-reactor plasma parameters. The expected plasma parameters are $T \sim 10$ keV, $n\tau_E \sim 1\text{--}2 \times 10^{19} \text{ s/m}^3$, and $\beta = 10\text{--}20\%$. Precise size and current levels will be projected from the integrated-test experimental results. Present estimates set a $\sim 0.50\text{--}0.60$ m and $I_\phi \sim 8\text{--}12$ MA. The total construction costs of the facility are estimated to be \$200 million, distributed over a 4-5-yr construction time. This figure does not include new buildings; it is assumed that existing shielded experimental sites can be used. Operating costs are estimated at \$25 million/yr for about 5 yr. The experiments would be sited to allow upgrading for DT burn experiments if, at the time, it is deemed mandatory that sustained-burn and ash-removal experiments be performed before the fusion reactor assessment is made.

RFP Experiments (< \$10 million)

ZT-P. ZT-P, a small air-core experiment ($R/a = 0.45 \text{ m}/0.07 \text{ m}$) at LANL, is serving as a prototype for the 4-MA experiment discussed above.

MST. This experiment ($R/a = 1.52 \text{ m}/0.52 \text{ m}$) is under construction at the University of Wisconsin. Its goals are to study shell effects and to do comparative studies of confinement properties of a toroidal plasma configuration as q is varied from < 1 (RFP) to > 1 (tokamak).

Reversatron. This experiment ($R/a = 0.5 \text{ m}/0.08 \text{ m}$), located at the University of Colorado, is currently being used to investigate the effects of removing the stabilizing conducting shell.

1.2 CONCEPTS WITH A DEVELOPING KNOWLEDGE BASE

This subsection presents the program logics for spheromaks, field-reversed configurations, Elmo bumpy squares, and the dense Z-pinch.

1.2.1 Spheromaks

The spheromak concept has unique features that distinguish it from other magnetic confinement configurations. Like the RFP, the spheromak has a toroidally symmetric magnetic-field geometry in which the toroidal and poloidal components of the field are comparable; this field and its structure are determined by strong currents flowing within the plasma. Unlike the RFP, however, the spheromak possesses comparable scale lengths in both the toroidal and poloidal directions and requires no coils or other structure linking the confined plasma torus. The compact geometry, simple topology, and strong internal currents of the spheromak present many potential advantages. These include simplification of the external blanket and coil structures, high engineering beta (plasma pressure compared with external field requirements), high plasma power density and reactor-mass power density, natural divertors at the magnetic separatrix, the options of translation or compression of the configuration between formation and burn, and strong ohmic heating to reduce or eliminate the need for heating by auxiliary means.

Issues and Objectives of the Spheromak Program

The issues and objectives of the spheromak program are summarized in Table 1.22.

TABLE 1.22 Spheromak Program Elements, Subelements, Objectives, and Attributes

Program Elements and Subelements	Objectives	Attributes
Macroscopic Equilibrium and Dynamics		
Macroscopic Stability	Minimize the amount and complexity of external structures (both driven and passive) required to control equilibrium and gross tilt and shift instabilities.	Field-line symmetry and closure
Current- and Pressure-Driven Effects	Obtain q-profiles that reduce kink- and ballooning-mode effects.	$q(\chi)$, $\langle\beta\rangle$
Transport		
Energy Confinement	Control the processes that determine spheromak energy loss.	$n\tau_E$
Wave-Plasma Interactions		
Wave Heating	Apply auxiliary heating or current drive by efficient rf techniques, as required.	Source-to-spheromak efficiency
Particle-Plasma Interactions		
Impurity Control	Reduce impurity effects through combined ohmic-heating burn-through and divertor action of open magnetic flux.	Z_{eff}
Composite		
Pulse-Length Optimization	Develop methods for sustainment against resistive decay, based on helicity injection or current drive.	Efficiency, $\tau_{\text{pulse}}/\tau_R$

Macroscopic Equilibrium and Dynamics. The confining magnetic fields of a spheromak are determined by the distribution of its internal currents, but the equilibrium must be supported by externally driven magnetic fields or (on short time scales) by image currents in conductors placed outside the separatrix. These structures must not only balance the equilibrium pressure, but they must also provide stability against the gross tilting and shifting motions that can occur. Because of the possible effects on edge confinement and the high internal magnetic fields (~ 10 T) anticipated for spheromak reactors, the requirements for external structures should be examined with the aim of increasing their simplicity. Greater understanding of equilibrium control might also lead to improvements in spheromak manipulation, translation, and compression. An important concept in spheromak research is the conjecture that spheromaks (like RFPs) will tend toward equilibria that minimize magnetic energy for fixed magnetic helicity (topological linkage of magnetic flux). This conjecture provides a reference point for the interpretation of observed spheromak equilibria and dynamics, as well as the basis for spheromak sustainment by helicity-injection techniques. Both current-driven (kink) and pressure-driven (ballooning) instabilities can occur inside the spheromak. Kink modes are commonly seen in the less collisional (higher-S) experiments and are adequately modeled by theory. They result from equilibrium magnetic profiles that depart significantly from the minimum-energy state, and in principle they can be eliminated by profile-control measures. Their effect on confinement has not been established. Because of the low shear of conventional spheromak magnetic-field profiles, the threshold for the onset of pressure-driven instabilities is low ($\langle\beta\rangle \sim 1\%$). Although such modes are presumed to exist in the higher- β plasmas observed in some experiments, their effect on confinement has not been quantified.

Transport. With the achievement of high current-to-density ratios in present spheromak experiments, radiation losses from low-Z impurities have been reduced, and work has begun on identifying loss processes inherent to the spheromak concept. Both the diagnostic and theoretical tools applied to these transport studies must have the proper spatial framework (in one or more dimensions) and must be equipped to deal with edge vs. volume effects. As these studies mature, it is essential that the experiments move into a more reactor-relevant regime, with lower collisionality and higher magnetic Reynolds number (S). Relative to the resistive decay rate, the growth rate of pressure-driven modes should increase with S, so that higher-S experiments may more clearly show the effects of ballooning modes on transport.

Wave-Plasma Interactions. Ohmic-heating rates for spheromak reactors are expected to lie in the 10- to 20-MW range. Auxiliary rf heating at comparable power levels is conceivable, but it is not now viewed as a requirement for ignition. However, rf heating could prove valuable for nearer-term confinement studies and as a means for raising the plasma edge temperature to control the current-density profile. If confinement studies establish the need for rf heating (or current drive) in the spheromak program, then the existing body of rf experience from other concepts can be adapted to address that need. Emphasis should be on the efficiency with which rf source power can be converted to spheromak thermal and magnetic energy and on the effects of substantial applied rf power on confinement.

Particle-Plasma Interactions. With regard to reactor fueling, spheromaks are similar to other confinement systems, and pellet injection may be required. For some methods of sustainment based on magnetic helicity injection, spheromak particle losses can be replaced by the injection of plasma along with the requisite linked magnetic fluxes. Impurity control in spheromaks involves unique issues, because of potential plasma-wall interactions with external stabilizing structures and the erosion of electrodes (if used). However, it is possible that the open field lines surrounding the configuration can be developed as a natural divertor to improve impurity control. In addition, high ohmic-heating rates should burn through low-Z impurities and reduce the associated radiation-energy loss.

Composite. The physical principles that determine the equilibrium distribution of spheromak internal currents can be exploited to maintain those currents in a steady state. This process may be viewed as the continual replacement of the spheromak magnetic helicity lost through resistive decay. The basic concept has been demonstrated, and a variety of approaches may be considered for a possible steady-state reactor. Criteria for evaluation include energy efficiency in maintaining plasma currents, nonohmic contributions to plasma heating, effects on impurity levels and confinement, and technical requirements for long-pulse or steady-state operation.

Spheromak Program Logic

Spheromak research to date has exhibited a wide variety of approaches. The principal U.S. spheromaks are the S-1 experiment at the Princeton Plasma Physics Laboratory, the MS experiment (completing construction) at the University of Maryland, and the CTX experiment at Los Alamos National Laboratory. These experiments differ in the techniques adopted for plasma current generation: S-1, inductive flux core; MS, electrode-inductive combination; and CTX, magnetized electrode alone. These differences in formation hardware and procedure extend to important differences in the subsequent time evolution of the plasma current profile, the boundary conditions presented to the plasma by the source, the external structures employed for gross stability, and the techniques being considered for configuration sustainment.

Outside of the U.S., experiments based originally on the Maryland formation technique may be found at Tokyo University and the University of Heidelberg, DBR. Experiments derived from the CTX approach are at Osaka University and at Nihon University (Tokyo); in addition, a CTX-inspired experiment is being started at the University of Manchester, U.K. The foreign spheromaks exhibit substantive differences from their U.S. counterparts, resulting in a greater exploratory coverage of spheromak formation and stabilization issues than would otherwise be possible.

The logic for spheromak development is shown in Fig. 1.25, and decision descriptions are given in Tables 1.23-1.25. During an exploratory phase, conducted with present facilities and their upgrades, a number of spheromak options will be addressed. These include methods for active equilibrium control, the possibility of spheromak translation, the use of external fields as natural divertors, fueling techniques, the

FIGURE 1.25 Level 2 Logic Diagram for the Spheromak

TABLE 1.23 Decision for Spheromak Concept Test Experiment

Statement of Decision

Construct a concept-test experiment addressing all major confinement issues and pursue a parallel effort on the composite issue of steady-state operation at the scale required for subsequent adoption.

Decision Criteria

Determine most favorable formation and stabilization technique. Establish equilibrium control or translation method.
 Identify dominant transport mechanisms at moderate S and demonstrate understanding of controlling factors.
 Evaluate need for auxiliary rf heating or current drive and establish most promising approach.
 Identify principal impurity issues for the chosen formation technique and perform experiments to establish appropriate control measures.
 Determine the most promising sustainment techniques, based on demonstrated capabilities in spheromak experiments.

Sources of Information

Overall performance of medium-scale (< \$10 million) spheromak facilities.
 Specific efforts on those facilities in areas of stabilization, impurity control, rf, sustainment, etc.
 Theoretical understanding of spheromak performance and scaling.

Outcomes and Consequences of Decision

Proceed with design and construction of concept test experiment. (The concept-test design establishes the formation, stabilization, fueling, and heating methods to be considered in all subsequent spheromak research.)
 Continue spheromak research at an appropriate level to assess further options while contributing to key areas of plasma science.
 Terminate spheromak research because of demonstrated fundamental shortcomings of the concept.

requirements for rf heating or current drive, and the choices available for steady-state sustainment. Evaluation of these technical options, along with more basic confinement studies, will provide design criteria for a major concept-test facility. At this point, specific methods for spheromak formation, equilibrium control, heating, and fueling should be established. Continued development of sustainment technology can be conducted on a separate experiment so as not to interfere with the primary mission of the concept-test facility. That mission will be to meet spheromak confinement goals at increased magnetic energy and plasma temperature, with transport occurring in a reactor-relevant collisionality regime.

TABLE 1.24 Decision E_2 for Spheromak Integrated Proof-of-Principle Experiment

Statement of Decision

Construct a proof-of-principle spheromak facility operating at near-reactor temperatures (3-10 keV), with good confinement ($n\tau_E > 10^{19} \text{ s/m}^3$), and with substantial magnetic energy (5-50 MJ). The choice is made between pulsed or steady-state modes of operation.

Design Criteria

Meet confinement goals ($n\tau_E > 10^{18} \text{ s/m}^3$) of the concept-test experiment.
 Demonstrate capability for equilibrium control, heating, fueling, and impurity control as required for integrated test design.
 Establish basis for extrapolation of performance to reactor level.
 Determine whether steady-state operation is a viable option.

Sources of Information

Overall performance of concept-test experiments.
 Evaluation of transport and confinement results from these experiments at reactor-relevant collisionality.
 Results of steady-state spheromak demonstration.

Outcomes and Consequences of Decision

Proceed with design and construction of proof-of-principle experiment that embodies all the major elements required for a spheromak reactor.
 Emphasis is on demonstrating the successful integration of these elements at higher energy, under more technically challenging conditions.
 Work toward correcting the obstacles to proof-of-principle development.
 Terminate spheromak development in the face of serious technical difficulties.

Achievement of the concept-test goals is required to proceed to the next stage, an integrated proof-of-principle test. This experiment will possess the major features and selected options considered most likely for the final reactor design, and it will include sustainment technology (if success has been achieved in that area). The goals of the integrated test are to achieve levels of magnetic energy, beta, temperature, confinement, and overall performance that assure the feasibility of a spheromak reactor and that justify recommending further development toward that end.

The plasma technology program logic for spheromaks is shown in Fig. 1.26. As present experiments on the formation and sustainment of spheromaks proceed, needs are being identified for pulsed or programmable coil systems providing stabilization, heating, helicity injection, etc. These systems will require further technological development in order to progress to the high levels of magnetic energy required of future systems. As the next generation of experiments begins to address the longer-pulse capabilities of spheromaks, additional efforts in heating, fueling, and impurity control will be initiated.

TABLE 1.25 Decision E₆ for Spheromak Reactor-Level Plasma Experiment

Statement of Decision

Construct a spheromak experiment at the reactor level.

Decision Criteria

Achieve reactor-level parameters ($\beta \sim 10\%$, $n\tau_E \sim 2 \times 10^{20} \text{ s/m}^3$, $T \sim 10 \text{ keV}$) in the proof-of-principle experiment or demonstrate technical feasibility of doing so.

Establish desirability of spheromak reactor in comparison with other confinement concepts at equivalent stage of development.

Sources of Information

Performance of the spheromak proof-of-principle experiment.

Systems studies of spheromak reactor designs.

Developmental status of other fusion concepts.

Outcomes and Consequences of Decision

Proceed with design and construction of a reactor-level experiment. With the achievement of reactor-level confinement parameters, emphasis shifts to issues of burning plasmas, reactor technology, and fusion economics.

Continue to develop the spheromak at a lower level as an alternative to other concept(s) being tested.

Terminate spheromak development because of the demonstrated superiority of other concept(s).

These developments will be incorporated into the major experiments to enhance their plasma parameters and capabilities for confinement and sustainment studies. The common goal of the various technology efforts is to demonstrate achievement of the milestones necessary for their incorporation into the integrated POP test.

1.2.2 Field-Reversed Configurations

The field-reversed configuration (FRC) is an elongated, prolate, compact toroid that is formed without a toroidal field. It offers extremely high beta values and the potential for qualitatively different reactor designs based on the possibility of moving the plasma from one region to another. A more complete description is given in Chapter 2 of Ref. 1. The stabilization of macroscopic MHD motions by means of individual particle orbits (kinetic effects) is an important physical process addressed by FRC research.

A parameter $\bar{s}$, the effective number of ion orbits within the plasma, has been identified as being important both to gross FRC stability and to FRC transport.

FIGURE 1.26 Level 3 Logic Diagram for Spheromak Plasma Technology

Table 1.26 summarizes the major issues to be addressed by FRC research in the area of confinement systems. The major research objectives at the present time are to understand FRC behavior with the increased values of $\bar{s}$ required for reactor-like confinement. Program elements in other areas (burning plasmas, plasma and nuclear technology, materials, etc.) are less well defined, in accordance with the status of FRCs as a developing concept.

Technical Issues, Objectives, and Attributes

Macroscopic Equilibrium and Stability. The FRC is an intrinsically high-beta concept, and no problem is anticipated in achieving sufficient beta values for a compact fusion system. However, maintenance of MHD stability represents a major issue in FRC research. The extremely high beta of the FRC equilibrium and the translation-compatible, elongated, prolate shape of the magnetic configuration are beneficial for the development of an economic compact fusion reactor, but these same features are accompanied by unfavorable stability properties as $\bar{s}$ increases, according to theory. Although experimental results have shown no direct evidence for instability to date ($\bar{s} \leq 2$), it may be necessary to provide active means for stabilization as $\bar{s}$ increases to a more reactor-relevant regime ($\bar{s} \sim 30$).

TABLE 1.26 Field-Reversed Configuration Program Elements, Subelements, Objectives, and Attributes

Program Elements and Subelements	Objectives	Attributes
Macroscopic Equilibrium and Dynamics		
MHD Equilibrium and Stability	Maintain stability with increased $\bar{s}$.	Value of $\bar{s}$
Transport		
Energy Confinement	Demonstrate favorable scaling of energy confinement with $\bar{s}$.	$\tau_E(\bar{s})$
Heating	Establish adiabatic compres- sion as viable method.	$\tau_E(\text{Temp.})$
Composite		
Formation	Develop lower-voltage forma- tion method.	τ_f , formation timescale

Transport. Much work remains to be done in the area of transport. A significant increase of τ_E with increased $\bar{s}$ is predicted theoretically, and experimental study of this issue is a key program element. The most attractive method of FRC heating appears to be adiabatic compression by means of an increased magnetic field after FRC formation. An unusual feature of the FRC is the possibility of combining translation and compression to accomplish a simple and efficient heating method. The determination of energy confinement (especially electron thermal conduction) at temperatures higher than those achieved in present devices is a major issue.

Composite. A concern unique to FRC physics is establishment of an attractive technology for FRC formation. Present experiments use high-voltage methods that could prove impractical when extrapolated to reactor operation. Improvements in present methods, or totally new approaches to formation, appear to be needed as part of the FRC concept development.

FRC Program Logic

Figure 1.27 shows the FRC program logic. The bottom of the figure shows two major evaluation points:

- E_2 : Evaluate readiness of the FRC concept to advance to an integrated proof-of-principle experiment.
- E_6 : Evaluate readiness of the FRC concept to advance to a reactor-level test.

Tables 1.27 and 1.28 summarize the criteria, information requirements, and expected consequences of these decision points.

Macrostability and Transport. The issues of stability and transport are the major driving force for experimental efforts planned in the next five years. An increased value of the parameter $\bar{s}$, needed to establish FRC stability and transport in a more reactor-relevant regime, will be provided in the LSX experiment at Spectra Technology (scheduled to begin operation in 1989). Heating by adiabatic compression to allow study of electron thermal conductivity is planned during the same period for the FRX-C facility at Los Alamos National Laboratory. Experiments that address FRC issues of formation, stability, and transport are also being carried out at Osaka University (Japan) and at the Kurchatov Institute (USSR). The various experimental results, combined with the development of a stabilization method (if needed), will allow a transport-scaling experiment to be designed in the early 1990s that would study heating with the parameter $\bar{s}$ in the range of 7-15.

Means of Stabilization. In parallel with the larger experimental programs, a smaller effort -- including the LONGSHOT and MICE experiments at Cornell University

FIGURE 1.27 Level 2 Logic Diagram for the Field-Reversed Configuration

TABLE 1.27 Decision E₂ for Progressing to Integrated Proof-of-Principle Experiment for the FRC

Statement of Decision

To proceed with construction of an integrated proof-of-principle FRC experiment.

Decision Criteria

Determine stability or means for stabilization for $2 < \bar{s} < 10$.

Determine favorable scaling for τ_E vs. $\bar{s}$ and τ_E vs. temperature.

Obtain evidence from slow-source formation studies indicating reasonable prospects for development of FRC source needed in Phase II of proof-of-principle test.

Sources of Information

Results of stability and confinement studies from the LSX experiment with $2 < \bar{s} < 10$.

Results of stability and adiabatic-compression heating studies on the FRX-C experiment at Los Alamos with $2 < \bar{s} < 4$.

Experimental results from Japan and the USSR.

Results of theoretical work (stabilization methods will be particularly important).

Outcomes and Consequences of Decision

Favorable assessment implies a new FRC facility with a cost of approximately \$100 million.

Favorable assessment implies that a qualitatively different fusion reactor is on the horizon, and some rethinking of system engineering will be required.

Unfavorable assessment implies FRC research should be discontinued.

-- is underway to evaluate possible methods of FRC stabilization in the event that theoretically predicted (but not yet experimentally observed) modes of instability are found to be important. One possibility for stabilization involves the maintenance of a non-Maxwellian velocity distribution by the use of rf heating; another possibility is the injection of energetic ion rings. As Fig. 1.27 indicates, a technique will be selected and evaluated before a decision is made and design begins on the transport-scaling experiment in about 1992.

Development of Improved Source. The high-voltage FRC formation methods used at present would pose difficulties in extrapolation to an economic fusion reactor. For this reason, a slow-source experiment has been initiated to develop lower-voltage or equivalent slower-timescale methods of FRC formation. As indicated in Fig. 1.27, a

TABLE 1.28 Decision E₆ for Progressing to Reactor-Level Test for the FRC

Statement of Decision

To proceed with design and construction of an FRC reactor-level test.

Decision Criteria

Establish stable operation at $\bar{s} \sim 30$.

Understand confinement properties well enough to allow confidence in design of reactor system.

Obtain viable solutions to formation technology and magnet design for adiabatic compression.

Develop designs that offer clear advantages over other fusion systems.

Sources of Information

Experimental results from integrated proof-of-principle experiment.

Experimental results from FRC programs in Japan and USSR.

Results from theory and system studies.

Outcomes and Consequences of Decision

Favorable assessment implies that a qualitatively different and improved fusion reactor has been defined and is ready for further development.

Unfavorable assessment with regard to conventional fusion systems may still leave open the possibility of application of FRCs to advanced fuel cycles.

sequence of development steps -- from exploratory studies with formation times of tens of microseconds to a prototype FRC source with a formation time of tens of milliseconds -- is expected to require about ten years of work. Evaluation of this effort will provide important input to the design of an integrated test experiment.

Integrated Test of Concept. The work described above, which mostly involves experiments that are now operating or are under construction, will require approximately five years for evaluation of most issues. At that time, favorable results would lead to the need for an integrated test of concept, which would be carried out in two phases. In the first phase, a facility capable of future expansion would be devoted to studies of transport sealing and stabilization. The goals would be to test heating by adiabatic compression and confinement at the same time in a device with $\bar{s}$ in the range of 7-15, resulting in $n\tau_E$ values in the range of 10^{18} - 10^{19} s/m³ and temperatures in the range of 0.5-3 keV. The stabilization that is likely to be required in this range of $\bar{s}$ would be provided by the method developed in preliminary stabilization studies. Depending on progress in the development of a slow FRC source, the FRC plasma in the first phase would probably be formed with an intermediate-voltage device based on the LSX programmed formation method.

In the second phase, the integrated test-of-concept experiment would have the goal of parameters in the Lawson regime ($\bar{s} = 10\text{--}30$). The increased plasma energy required during formation with such parameters would be provided by the method developed along the slow-source issue line shown at the bottom of Fig. 1.27. Thus, the integrated test would bring together for the first time all elements of the FRC confinement system that are presently judged to be critical -- transport, stability, and slow-source.

Major FRC Facility Requirements

Existing and New Facilities (< \$10 million). In addition to a number of university experiments, the major FRC facilities are the FRX-C experiment at LANL and the TRX-2 experiment at Spectra Technology.

The FRX-C device, originally constructed with a 0.5-m-diameter, 2-m-long coil, has recently been modified to have a 0.7-m-diameter, 2-m-long coil. The plasma typically has $n \sim 2 \times 10^{21}/\text{m}^3$, $T_e \sim 100$ eV, $T_i \sim 400$ eV, $\bar{s} \sim 2$, and $n\tau_E \sim 2 \times 10^{17}$ s/m³. The TRX-2 device has a 0.2-m-diameter, 1-m-long coil and similar plasma parameters.

A new, larger experiment called LSX is under construction at Spectra Technology. It has objectives of $s \sim 8$, $n \sim 2 \times 10^{21}/\text{m}^3$, $T_i \sim 350$ eV, and $n\tau_E \sim 2 \times 10^{18}$ s/m³. It is expected to begin operation in 1989-90.

New Facilities (>\$10 million). The FRC integrated proof-of-principle experiment would be a new facility, with the objectives of testing stabilization and transport scaling. Parameters would be $\bar{s} = 10\text{--}30$, $n\tau_E = 10^{19}\text{--}10^{20}$ s/m³, and $T_i = 3\text{--}15$ keV. The experiment should also establish a formation method that could be extrapolated to a practical fusion system. The formation method would most likely be a modification developed during a second phase of operation. The cost is expected to be approximately \$70 million in the first phase and \$30 million in the second phase, with construction lasting five years. Operating costs are projected to be about \$23 million/yr.

1.2.3 Elmo Bumpy Square

In the "bumpy-toroid" concept, one seeks to eliminate two problems that have plagued fusion devices; end losses inherent in open systems are eliminated by closure into a toroid, while transport and stability problems caused by parallel plasma currents are alleviated by use of magnetic mirrors (bumps) and controlled toroidal curvature. Initial tests in circular geometry with a small number of mirrors (24) were only partially successful, but the cause -- a consequence of excessive toroidal curvature -- is believed to be understood. This has led to a revised bumpy-toroid concept called the Elmo Bumpy Square (EBS). The effects of curvature would be minimized by restricting it to four regions of relatively high magnetic field at the corners of a square platform.

Bumpy-toroid reactors hold the promise of true steady-state operation in magnetic geometries with simple magnetic coils, which have a high degree of access and easily serviceable modular blanket elements.

Issues and Objectives of a Bumpy Toroid Program

The issues and objectives are summarized in Table 1.29.

Macroscopic Equilibrium and Dynamics. Equilibrium for bumpy toroids is well understood theoretically, and experimental results from the Elmo bumpy toroid (EBT) device at Oak Ridge National Laboratory (ORNL) and the NBT at Nagoya University in Japan were in substantial agreement with calculations. Stability in bumpy toroids is provided by effects of energetic (100 keV and higher) electron rings located in the mirrors. Extension of the experimentally observed quiescent behavior to beta values of interest in reactors constitutes the primary long-term research objective for any future program. Elucidation of the hot-electron stabilization mechanisms for possible application to other fusion geometries would be an important subsidiary benefit of bumpy-toroid research.

Transport. Confinement in EBT and NBT systems is limited by processes that have been interpreted as being due to convective cells formed by poor centering of the electric equipotentials; this poor centering, in turn, is caused by having too much toroidal curvature in the containment regions. A key objective of a continued bumpy-toroid program would be to demonstrate control of this problem through the use of toroidal curvature localized in regions of high fields, which would minimize the deleterious effects. The same physics has possible application to control of transport in other nonaxisymmetric systems, such as stellarators, tokamaks with magnetic-field ripple effects, and open-ended devices.

Wave-Plasma Interactions. Bumpy-toroid programs have included the development of the ECH microwave sources now widely used in the fusion program. Continued development of this technology would clearly be necessary to drive the stabilizing energetic hot-electron rings in bumpy toroids.

Particle-Plasma Interactions. Impurity and particle-feed physics in devices with long pulses or steady-state operation have been addressed in the bumpy-toroid program. This program has provided the only information to date on particle balance, impurity dynamics, and wall interaction in an equilibrium plasma having substantial energy density. The bumpy-toroid device could readily be provided with simple divertors in the low-magnetic-field regions between the mirror coils, where access to the plasma edge is good.

TABLE 1.29 Program Elements, Subelements, Objectives, and Attributes for the Bumpy Toroid

Program Elements and Subelements	Objectives	Attributes
Equilibrium and Macroscopic Dynamics		
Control of stability	Demonstrate stability by energetic electrons.	Beta values of 5-10%
Effect of corners	Demonstrate tolerable ballooning modes.	Beta values of 5-10%
Transport		
Effects of electric islands	Demonstrate control and reduction of electric islands.	Determined by curvature distribution
Wave-Plasma Interactions		
High-efficiency production heating	Improve and control by particle-loss reduction.	Performance of rings and main plasma
Particle-Plasma Interactions		
Low impurity contamination	Demonstrate persistent low level of impurities.	Steady-state operation
Fueling	Demonstrate control.	Steady-state operation
Composite		
Field shaping	Define optimum.	Balance stability and transport requirement
Profiles (n and T)	Define optimum.	Balance stability and transport requirement
Ring-physics trades	Control balance of electron cyclotron emission and drag losses.	Electron cyclotron radiation emission

Bumpy Toroid Program Logic and Facilities

The program logic for a bumpy-toroid program has not been worked out in detail, because the program was terminated during FY1986 by the U.S. Department of Energy (DOE). Circular-toroid facilities, including plasma-heating sources, necessary power supplies, and diagnostic apparatus, presently exist at both the Oak Ridge National Laboratory and the Nagoya University Institute of Plasma Physics. The existing ORNL facilities have been estimated to have a value of \$50 million.

Costs for upgrading the ORNL device to provide the key tests of increased confinement (due to improved drift orbits resulting from control of toroidal curvature) were estimated to be about \$6 million. Construction time would be about two years. A research program aimed at resolving the primary issues would require about \$4 million/yr for three years. If a bumpy-toroid program were reactivated in 1987 to proceed with the minor modifications needed to convert the EBT circular facility into an EBS, evaluation of its potential for success and a decision on its future could occur during the period 1990-91. Provided that decision were taken promptly, a concerted design and construction schedule would permit operation of a proof-of-principle experiment to demonstrate operation in nearly reactor-like conditions by 1996 at the earliest. Even with excellent success at that level, the decision could be made no sooner than 2000 whether to proceed to an ignition demonstration in bumpy-toroid geometry.

1.2.4 Dense Z-Pinch

Issues and Objectives of the Dense Z-Pinch Program

The issues and objectives of the dense Z-pinch program are summarized in Table 1.30.

Macroscopic Equilibrium and Dynamics. The present gross size characteristics of the dense Z-pinch are approximately those required for a reactor, but present currents (~ 300 kA) and temperatures (~ 0.3 - 0.5 keV) are below reactor values (1.4 MA and 10 keV). The $\langle \beta \rangle \approx 1$ already meets reactor requirements. Magnetohydrodynamic stability has been demonstrated for tens of Alfvén times at present current levels. The main tasks are to demonstrate size and accretion control, ohmic heating to higher temperatures, and acceptable stability as the current is raised toward the reactor value.

Transport. The confinement parameter $n\tau_E$ is not presently known for the dense Z-pinch. The importance of radial energy losses for the cold boundary (capillary) and vacuum-boundary (solid-fiber) conditions remains to be determined as pinch currents are increased. The estimate that the present length of the pinch (~ 10 cm) is adequate to satisfy the axial thermal-conduction requirements needs to be verified.

TABLE 1.30 Program Elements, Subelements, Objectives, and Attributes for the Dense Z-Pinch

Program Elements and Subelements	Objectives	Attributes
Macroscopic Equilibrium and Dynamics		
Magnetic Transients	Demonstrate stable equilibrium at $I \geq 1.4$ MA.	No gross instability during current rise
Transport		
Energy Confinement	Demonstrate reactor-level confinement.	$n\tau_E$
Particle-Plasma Interactions		
Fueling	Eliminate accretion.	$\dot{N} = 0$ rep rate in Hz
	Reactor-relevant repetition rate.	Repetition rate in Hz
Alpha-Particle Effects	Minimize core plasma heating; minimize exo-column ionization and current diversion; and understand alpha-particle/electrode interaction.	Frequency/mass/cost of electrode replacement
Radiative Collapse of Pinch	Understand dynamics Enhance fuel burning.	DT burnup
Composite	Choose configuration (cold boundary vs. vacuum boundary).	Z_{eff} , τ_E

Wave-Plasma Interactions. Wave heating is not required in the ohmically heated dense Z-pinch. An area where wave-plasma effects might occur for this concept, for the cold-boundary case, would be microturbulence affecting radial transport. At present, the occurrence or absence of such effects is unknown.

Particle-Plasma Interactions. The impurity level is believed to be very low in the solid-fiber-initiated dense Z-pinch. Impurity influx from the electrodes should be insignificant during the short time scale of the present experiments, as well as for a dense Z-pinch reactor. The effect of a quartz corona in the capillary-initiated dense Z-pinch is unknown.

Two methods for controlling the particle-accretion problem are under investigation, the solid fiber in vacuum and the capillary-enclosed gas; continued evaluation of these methods is required. In the case of the solid fiber, stable pressure profiles are required as the pinch current is raised. In the case of the capillary pinch, radial transport and impurity influx from the capillary material are areas requiring investigation.

Alpha-Particle Effects. Alpha particles are not expected to be confined within the dense Z-pinch, but damage to the electrodes could be an issue. Additionally, alpha-particle ionization of the gas external to the pinch could divert the confining current from the main pinch column, in effect causing an undesirable expansion and cooling of the pinch and limiting DT burnup.

Dense Z-Pinch Program Logic

Figure 1.28 shows the logic diagram for the dense Z-pinch approach to fusion. The dense Z-pinch is unique in its simplicity of apparatus and low cost of experiments, allowing a parallel approach to fueling (capillaries and D_2 fibers) through DD burn experiments if necessary. The decision on a DD experiment is summarized in Table 1.31. This parallel approach stems from the need to overcome the present problem of accretion of particles into the pinch from the surrounding medium in a manner consistent with good transport and stability properties. The next generation of experiments requires higher maximum currents and faster current-rise times.

The key short-term issue for the dense Z-pinch concept is stability at reactor-level currents and current-rise times. The key longer-term issue for reactor development is the required repetition rate for fueling and for the pinch-current power supply.

Summary of Major Facility Requirements

The two major dense Z-pinch facilities in the US are the DZP experiment at the Naval Research Laboratory (NRL) and the HDZP at LANL. In addition, there are some smaller devices at Imperial College (U.K.). Other facilities could be quickly

FIGURE 1.28 Level 2 Logic Diagram for Denze Z-Pinch

TABLE 1.31 Dense Z-Pinch Decision on Proceeding with DD Burn Experiment

Statement of Decision

To proceed with the DD burn experiment, in which the primary objective is to obtain equivalent DT $Q > 1$.

Decision Criteria

Obtain stable, static equilibria at the 1.5-MA current level.

Explore confinement scaling for $n\tau_E \sim 10^{17}-10^{18}$ s/m³.

Choose between cold-boundary and vacuum-boundary approaches on the basis of preliminary transport, stability, and impurity-level assessments.

Sources of Information

Results from existing dense Z-pinch experiments.

Preliminary results from dense Z-pinch DD burn experiments.

Plasma supporting activities (principally in Europe).

Outcomes and Consequences of Decision

Favorable assessment and achievement of the objectives of the DD burn experiments would lead to a DT burn experiment and an assessment of the technological possibilities of developing the concept towards a reactor (particularly with respect to the repetition-rate problem).

Undertake further research to resolve the remaining issues.

Terminate the dense Z-pinch program.

commissioned using existing high-voltage, high-current generators, such as the Blackjack III device at Maxwell Laboratories and Imperial College (London).

The DZP experiment operates at about 600 kV and 420 kA with a pulse length of 240 μ s. Recent Z-pinchs in this experiment have been generated in D₂ gas enclosed in quartz capillary tubes. The HDZP experiment has operating parameters of 600 kV, 250 kA, and a pulse length of about 100 μ s. The HDZP uses solid D₂ fibers of about 40 μ m in diameter.

Two new facilities for the DD burn experiments would be required to develop both the fiber and capillary approaches. These facilities would cost about \$1 million each, for a total of \$2 million, and would be capable of $\dot{I} \geq 8$ MA/ μ s and $I \geq 1.5$ MA.

REFERENCE

1. J.D. Callen et al., University of Wisconsin, *TPA Plasma Science Final Report*, Dec. 1986.

OVERVIEW

Development of the physics of fusion burning plasmas is the second key technical issue from the MFPP. The overall objectives are as follows:

- Demonstrate high fusion energy gain with sufficient pulse length.
- Develop the science and technology of burning plasmas. (The desired level of science and technology development includes both empirical models, derived from experimental data, and theoretical models.)

Neither of these two objectives is sufficient by itself. Both are important in developing an adequate knowledge base of burning plasmas for a realistic assessment of fusion.

The burning plasma issue has been divided into two main issues -- alpha-particle effects and burn control and ash removal -- and five subissues, which are summarized in this chapter. The status of various confinement concepts in terms of these issues is discussed in detail in a companion report (Ref. 1, Chapter 1).

As was indicated in Chapter 1, the TPA plasma science strategy is to utilize the most highly developed confinement concept at present (the tokamak) in addressing the burning plasma issues. The tokamak is utilized because only it can presently be extrapolated with confidence to reactor-level burning plasma conditions. Reactor-level burning plasma tests will also be required later on for the advanced concepts, to the degree that their burning plasma issues are different from those in tokamaks (see Sec. 2.2).

The different stages of development for burning plasmas in relation to a 2005 fusion-assessment date are characterized in Fig. 2.1. For the remainder of this decade, the emphasis is on demonstrating the scientific feasibility of fusion energy through achievement of the plasma conditions theoretically required for net energy gain ($Q > 1$). In the early 1990s, the physics of an ignited DT burning plasma, in which the energy from the fusion-product alpha particles is sufficient to energetically sustain the plasma, will be explored for short-burn pulse lengths of about 10 energy confinement times. Thereafter, a long-burn

FIGURE 2.1 Characterization of Tasks for Burning Plasmas

demonstration (LBD) device, perhaps built internationally, would produce and control a long-pulse-length DT fusion burn with high fusion energy gain. Such an LBD device is needed to resolve additional burning plasma physics and technology issues not addressed in the short-pulse ignition experiment.

The development of the burning plasma program is illustrated schematically in Fig. 2.2. The evaluation points E_1 and E_3 (the same as those in Fig. S.7) are described in detail in Tables S.4 and S.6. Figure 2.2 also indicates the flow of information from advanced concepts (primarily advanced tokamaks, but also likely to include stellarators and reversed-field pinches) into the tokamak burning plasma program.

The burning plasma issue has been subdivided into the following two specific issues:

- Alpha-particle effects (*Alphas*): What are the confinement and heating effects of alpha particles on plasmas?
- Burn control and ash removal (*Burn*): How is the fusion burn to be controlled and the fusion ash removed?

These two issues have been further subdivided into five subissues, as follows:

<i>Alphas</i>	<i>Burn</i>
Alpha-particle confinement	Burn control
Alpha component effects	Ash removal
Alpha heating	

These subissues, together with objectives for resolving them and attributes for gauging progress, are developed in detail in a companion document (Ref. 1, Chapter 1).

The two main burning plasma issues have as their objective the development of an understanding of and predictive capability for plasmas. In addition, the following plasma parameter objectives are to be achieved:

- *Alphas* -- Achieve efficient alpha-particle heating with minimal deleterious effects.
- *Burn* -- Demonstrate fusion burn control with high burnup and adequate ash removal.

More specific objectives for the two burning plasma issues and for the five magnetic confinement systems issues are best elucidated in the context of the stages of burning plasma issue development indicated in Figs. 2.1 and 2.2. The relevant objectives for the tokamak concept are shown in Table 2.1. The definitions in the table should be interpreted as measures of performance to be achieved or surpassed for advancement into the next stage of experiments. For each issue, the required achievements become more stringent at each successive stage of development.

FIGURE 2.2 Schematic Illustration of Burning Plasma Program Development

TABLE 2.1 Achievement Objectives for Burning Plasmas

Plasma-Science Issue	Scientific Breakeven Test, 1985-1992	Pulsed Ignition Test ($t_{\text{burn}} \sim 10 \tau_E$), 1992-1998	Long-Burn Demonstration ($t_{\text{burn}} \gg 10 \tau_E$), 1998-2005
Macroscopic Equilibrium and Dynamics	Demonstrate nondisruptive, moderate- β operation	Explore stability limits in a fusion plasma	Operate at reactor-level β ($> 10\%$)
Transport	Achieve breakeven and determine transport scaling ($Q \sim 1$)	Achieve ignition and determine transport scaling ($Q > 5$)	Demonstrate reactor-level plasma confinement ($Tn\tau_E > 3 \times 10^{21}$) ^a
Wave-Plasma Interactions	Demonstrate high-power heating, fueling, and impurity-control methods	Heat plasma to ignition	Demonstrate rf methods for long burn
Particle-Plasma Interactions		Explore pellet fueling and control in an ignited plasma	Demonstrate steady-state impurity control and fueling
Composite Issues	Understand profile effects	Understand pulse-length limiting effects	Demonstrate long-burn operation
Alpha-Particle Effects	Explore single-alpha-particle effects	Demonstrate alpha heating and explore alpha-particle collective effects	Quantify long-burn alpha effects
Burn Control and Ash Removal	Investigate thermal control and stability	Demonstrate burn control and ash removal	Demonstrate burn control

^a $Tn\tau_E$ in $\text{keV}\cdot\text{s}/\text{m}^3$.

The remainder of this chapter lays out the issues, objectives, and attributes for the tokamak concept (Sec. 2.1) and for other concepts (Sec. 2.2). A program logic is laid out only for the tokamak concept to meet these objectives. (In the preceding chapter, the approaches to burning plasma issues for the respective confinement concepts were integrated into the concept specific logic diagrams.)

2.1 BURNING PLASMA AREA FOR TOKAMAKS

Technical Issues, Objectives, and Attributes

Table 2.2 summarizes the major issues to be addressed in the tokamak burning plasma program. A more complete discussion of these issues is to be found in the TPA Plasma Science Final Report (Ref. 1).

Alpha-Particle Effects. For typical plasma profiles, a plasma current of 3 MA is sufficient to confine the orbits of more than 90% of the 3.5-MeV alpha particles in a tokamak. At ignition temperatures, the characteristic time for collisional scattering of energetic alpha particles by background ions is very long compared with the characteristic time for "slowing down" by collisional drag on electrons, so classical and neoclassical transport processes do not pose a significant threat to alpha-particle containment during the heating time. Of considerably more interest than single-particle and neoclassical confinement of alpha particles are the effects of a finite partial pressure of alpha particles on the stability and transport properties of the background plasma: the energetic alpha particles in an ignited plasma may be expected to have partial pressures in the range of 10-20% of the background electron or ion pressure. Energetic ion components with partial pressures in this range are expected to have a substantial impact on plasma stability properties.

Even for an isotropic distribution of energetic ions, as in the case of alpha particles, the toroidally precessing, energetic trapped ions can significantly affect the "kinetic" MHD stability, tending to lower the predicted beta limit. A burning plasma also offers numerous theoretical possibilities for new types of high-frequency microinstabilities. In the steady state, the (isotropic) velocity distribution of alpha particles is monotonically decreasing with velocity, and therefore, absolutely stable to velocity-space modes. Among "finite-medium" modes, which depend on spatial gradients, it appears that the most dangerous microinstability that could be excited by the alpha-particle population is the short-wavelength ion branch of the drift Alfvén mode, which can be made unstable if the alpha-particle density profile peaks sharply near the magnetic axis. The theory of these instabilities in their nonlinear, turbulent regime is likely to be sufficiently complex and unreliable that experimental data from burning plasmas will be needed before the practical significance of the modes can be definitively established. As well as tending to drive the plasma pressure profile to its MHD stability limit, the thermonuclear heat source will play a major role in determining the detailed profiles of the plasma temperature and density -- factors that are known to influence the local plasma transport coefficients, and thereby, the effectiveness of alpha heating.

TABLE 2.2 Tokamak Burning Plasma Program Elements, Subelements, Objectives, and Attributes

Program Elements and Subelements	Objectives	Attributes
Alpha-Particle Effects		
Alpha-Particle Containment	Develop a validated predictive capability for classical alpha-particle containment and energy deposition.	Predictive capability
	Demonstrate "single-particle" confinement of energetic alpha particles in reactor-like plasmas in the presence of macroscopic fluctuations characteristic of those that occur as the beta limit is approached.	$Q \geq 1$
Alpha Component Effects	Determine the effect of a significant alpha-particle pressure on the macrostability (e.g., beta limit) of the tokamak.	Predictive capability
	Investigate whether energetic alpha particles excite new types of microinstabilities, such as drift-Alfven modes, in a tokamak.	Predictive capability
Alpha Heating	Determine the effects of alpha-particle heating on plasma energy confinement.	Predictive capability
	Achieve significant alpha-particle heating.	$Q \gg 1$
Burn Control and Ash Removal		
Burn Control	Determine the transport properties of a tokamak under the competing influences of thermally unstable alpha-particle heating and pressure-driven MHD-like modes.	Predictive capability
	Investigate "sawtooth" behavior in thermally unstable alpha-particle-heated tokamaks, including the effect of resistive reconnection on alpha-particle confinement.	Predictive capability

TABLE 2.2 (Cont'd)

Program Elements and Subelements	Objectives	Attributes
	Demonstrate optimized plasma conditions for short-pulse ignited burn.	$Q \gg 1$
	Demonstrate concepts for control of burning plasmas.	τ_{burn}
Ash Removal	Determine the most effective approach to the removal of helium ash from the edge and scrape-off regions of a tokamak plasma, consistent with tritium burn-up requirements.	Predictive capability
	Demonstrate ash removal.	$n_{\alpha}/n_{\text{Tritium}}$ burnup

Burn Control and Ash Removal. The main new feature concerning transport in a burning plasma will be a tendency toward thermal excursions: the hottest parts of the plasma will react most strongly and will grow still hotter. Such excursions will be curbed, however, by the slowness of the alpha-heating timescale (measured in seconds) and by the plasma pressure limitations imposed by the stability of the confining field's configuration. Even if the ignited state is entered at relatively low plasma beta values (i.e., below the stability limit), there will be a tendency for the temperature to rise through thermal instabilities until either the beta limit is encountered or the fusion reactivity begins to decrease. Thus, the most critical transport issue for burning plasmas will be the nature of the transport processes that will establish themselves under the competing influences of alpha-particle heating and anomalous heat transport due to pressure-driven MHD modes. The understanding and control of resistive relaxation phenomena is also likely to be particularly important in a burning tokamak plasma: energetic alpha particles could be expelled from the plasma as a result of the stochastic fields that arise during "sawtooth" magnetic reconnection.

In any DT fusion reactor, "fusion ash" in the form of helium ions produced in the plasma core could accumulate to the 10% level in the steady state, even without recycling of any of the helium ions transported to the plasma edge. In the case of the tokamak, the density of recycled helium ions can be kept relatively low, because the particle confinement times in the edge region of the plasma are very short. If the edge particle confinement time is 30 times shorter than the central particle confinement time, removal of a minimum of 3% of the recycling helium neutrals will suffice to keep the density of recycled helium ions below that of newly produced helium. Shorter

particle confinement times facilitate ash removal, but they tend to lower the "tritium burn-up fraction" (i.e., the rate at which tritium undergoes fusion reaction, relative to the rate at which tritium must be recycled through external processing systems).

Tokamak Burning Plasma Program Logic

The overall tokamak program consists of a main-line reactor-plasma effort utilizing DT tokamaks, accompanied by a set of specialized and more innovative supporting experiments. A principal objective of the main-line effort is to achieve definitive experimental results in the reactor-plasma regime on the major issues of the magnetic confinement systems program -- macroscopic equilibrium and dynamics, transport, wave-plasma interactions, and particle-plasma interactions. The contributions of the main-line DT facilities toward resolving these issues are described in Sec. 1.1.1. The present discussion concentrates on the D-T-specific issues to be addressed in the tokamak burning plasma program.

During the past ten years, extensive conceptual design work has been carried out in the area of ignited tokamak DT reactors, but thus far, the attainability of the necessary quality of confinement for ignition has been hypothetical. During the past year, solid information on this topic has become available, particularly from high-temperature TFTR and JET experiments. On the basis of these results, the outlook is highly favorable for the TFTR to demonstrate DT equivalent-breakeven conditions in deuterium plasmas in 1987, and reliable extrapolations of the data from these equivalent-breakeven experiments to the ignition region will be possible. The subsequent achievement of actual breakeven in DT plasmas by 1989 would serve to confirm the design point chosen for an ignition experiment and would provide essential information on single-particle alpha containment and energy deposition in tokamaks, as well as the capability to advance fusion technology in such areas as tritium and remote-handling systems.

An obstacle to the early initiation of an ignition project has been the difficulty of gaining a general technical consensus as to the most appropriate mission. A compact device could test the basic viability of the burning plasma regime at a relatively early time and moderate cost. At the same time, a high-performance, technology-oriented device would come much closer to a real fusion reactor and would put the required hardware-development effort to more fruitful use. The MFPP and the TPA have adopted a two-pronged approach: (1) reconsideration of opportunities for innovative design and maximum use of existing facilities, with a view to lowering the cost of a plasma-science-oriented ignition experiment to \$300 million or less; (2) design of an attractive technology-oriented machine in the range of about \$2-3 billion, to be constructed as a multinational project.

These two projects are included in the overall tokamak plan outlined in Fig. 1.8 as the "short-pulse ignition experiment" and the "long-burn demonstration." There are obvious advantages in an overall strategy that combines a science-oriented pilot project with a high-performance technology project developing in parallel on a somewhat longer time scale. International collaboration may well provide the resources to pursue this program in a timely fashion. The smaller project could start by 1988, with facility

completion by 1993 and a meaningful initial contribution to (short-pulse) burning plasma issues by 1995. Construction of the larger project could start in 1992, with facility completion in 1998. Operating in a high-duty-cycle mode like that of a fusion reactor, the latter device could make a major contribution to the resolution of the remaining (long-pulse) burning plasma issues, as well as issues in fusion nuclear technology, by 2005.

Alpha-Particle Effects. The orbits and single-particle confinement of alpha particles have been studied experimentally in non-DT facilities by simulating the alpha particles with MeV-range charged products of DD reactions. In particular, the 1-MeV tritons produced in the proton branch of the DD reactions have about the same Larmor radii as 3.5-MeV alpha particles (and the collisionality parameter of the former is within a factor of three of that of alpha particles); the confinement of the tritons can be determined from the 14-MeV-neutron emission arising from subsequent DT reactions. Such experiments have shown that ions with energies in the MeV range typically are well confined in tokamaks. The reaction rate of 1-MeV tritons falls somewhat short of ideal predictions, probably because of their relatively large Larmor radii and associated orbit losses. When strong MHD-like activity is present, the reaction rates are much reduced, corresponding to significant losses of MeV-range ions.

The TFTR will be the first tokamak in the world to provide significant DT operation near breakeven conditions. This device provides the opportunity to investigate DT plasma issues, including the confinement of energetic alpha particles at reactor-level conditions, in the presence of macroscopic fluctuations (strong "sawteeth," "fishbones," etc.) characteristic of tokamaks subjected to high levels of auxiliary power.

The short-pulse ignition experiment will be designed specifically to address the basic physics of the burning plasma state. The (energetic) alpha-particle pressure will reach equilibrium levels, permitting a study of its effects on macro- and microstability. The short-pulse ignition experiment will also be the first tokamak to operate in the regime where alpha-particle heating is dominant.

Burn Control and Ash Removal. Although the influence of the heat-deposition profile on confinement in tokamaks is becoming better understood empirically as the experimental data base on auxiliary-heated tokamaks becomes more extensive, the specific profile effects and thermal instabilities that will arise because of the temperature dependence of the DT cross section can only be addressed in an ignition experiment.

The short-pulse ignition experiment will provide critical information on the transport properties of a tokamak under the competing influences of thermally unstable alpha-particle heating and pressure-driven MHD-like modes. This experiment will also address the behavior of alpha particles in reconnecting magnetic fields (e.g., sawteeth). The burn times available in the short-pulse ignition experiment are several times the energy confinement time and many times the sawtooth period, so the experiment will be adequate to address these transport phenomena, which are fundamental to an understanding of burn control.

Pulse lengths of at least 20 energy confinement times will be needed to reach an equilibrium level of helium-ash buildup at the center of a tokamak. Thus, the problem of ash buildup will not be an issue for the short-pulse ignition experiment, but it must be addressed in the long-burn facility. The phenomena of helium transport in, and removal from, the edge region of a tokamak plasma have been addressed successfully using computer modeling techniques, but the magnitude and scaling of the relevant cross-field particle diffusivities remain highly uncertain. Moreover, if the central helium particle-confinement time turns out to be unduly long, or if the neoclassically predicted central accumulation of helium ions is found to occur, then the helium ash could build up to intolerable levels even in the presence of effective edge pumping.

The demonstration of active techniques for control of burning plasmas over periods approaching the resistive skin time (typically hundreds of seconds) will be a principal objective of the long-burn facility. Past experience with confinement issues in high-temperature plasmas suggests that direct experimental tests with long-burn plasmas will be needed to unravel the physics of helium transport and to determine the most practical approach to the steady-state removal of helium ash, consistent with acceptable tritium burn-up. Understanding helium transport and helium-ash removal will be a second principal objective of the long-burn facility.

Major Burning Plasma Tokamak Evaluation/Decision Points

The major evaluation/decision points are indicated in the burning plasma logic diagram (Fig. 2.3), which is abstracted from the overall tokamak program logic diagram (Fig. 1.8):

- E_1 : Decision on short-pulse ignition experiment.
- E_3 : Decision on long-burn demonstration facility.

Tables S.4 and S.6 summarize the criteria, needed input, and expected consequences at these decision points.

Major Tokamak Burning Plasma Facility Requirements

Existing Facilities. The capabilities of the world's two DT tokamaks (TFTR and JET) are indicated in Table 1.10. The initial task of the TFTR is to advance the main-line tokamak effort by entering the reactor-level plasma regime in 1986-87. At this point, critical assessments can be made concerning the outlook for ignition and the potential for improving confinement and stability by auxiliary techniques. The present plan calls for the TFTR facility to demonstrate $Q \sim 1$ in a DT plasma during 1989 in order to develop a base of DT experience for the short-pulse ignition experiment. Plans for JET call for DT operation to be introduced in 1991, if results in the DD mode suggest that a substantial central-temperature excursion will be produced by alpha-particle heating.

FIGURE 2.3 Tokamak Burning Plasma Logic Diagram

New Facilities. As presently conceived, the short-pulse ignition experiment will be a modest-sized ($R = 1.3$ m) tokamak with a strong magnetic field (10 T), a high current (10 MA), and a moderate amount of cross-sectional shaping. The beta requirement will be conservative ($\beta \leq 6\%$), and the margin for achieving ignition-quality confinement will be as large as possible. For reasons of economy, the burn time will be kept relatively short ($\tau_{\text{burn}} \geq 10\tau_E \geq 10\tau_{\text{alpha slowing-down}}$; $\tau_{\text{burn}} \geq 10\tau_{\text{sawtooth}}$). The plasma will be heated to ignition by a moderate amount of ICRH power (10 MW); ECH power may also be employed at a later stage. The integrated neutron fluence will be below the threshold for insulator damage (and well below the level of materials-damage interest). Making maximum use of existing facilities, the capital cost of the short-pulse ignition experiment can be held below \$300 million.

International Cooperation on the Long-Burn Facility. Tokamak research has been the principal area of worldwide cooperation in fusion to date. In the past, this cooperation has mainly taken the form of timely information sharing and exchange of technical personnel. Close cooperation was established among the builders of the TFTR, JET, and JT-60 from the outset of the design phase, thus promoting the transfer of useful technical information and minimizing the duplication of experimental research capabilities. Integrated multinational groups have been working together in the design of possible next-step tokamaks, such as the International Tokamak Reactor (INTOR) study.

At present, support is growing for international collaboration in fusion research, and there appears to be an emerging pattern of compatible national strategies. The U.S. magnetic fusion program, which has traditionally been oriented toward confinement concept breadth, is giving primary attention to a modest further step along the DT tokamak line (i.e., the short-pulse ignition experiment). The principal non-U.S. magnetic fusion programs, which already have a much stronger concentration of emphasis in the tokamak confinement area, are primarily aimed at a major technology-oriented facility of the tokamak type.

The overall strategy adopted by the TPA combines a science-oriented short-pulse ignition experiment with a high-performance technology project (the long-burn facility), which is developed in parallel on a somewhat longer time scale. A bilateral or multinational effort would have the resources to pursue this program in a timely fashion; construction of the long-burn project could start around 1992, with facilities completed around 1998. A high-duty-cycle operating mode approximating that of a real fusion reactor could begin around the turn of the century.

2.2 BURNING PLASMA AREA FOR NON-TOKAMAK CONCEPTS

Ignition in a DT fusion plasma requires that cold fuel be heated to fusion temperature by the 3.5-MeV alpha-particle reaction product. The physics issues associated with this self-sustainment include those relating to the alpha particles themselves, to their interaction with the plasma, and to the plasma's behavior under intense alpha heating. This last area includes the plasma energy containment and thermal stability under burn conditions and the removal of alpha-particle ash. The

details of how all of these issues will be manifested in different devices will vary with the device confinement characteristics. Most of the issues are at present conjectural, based in part on experience of intense neutral-beam injection where available; further understanding must await the first ignition experience. As experience with burn physics in one device develops, much of the information gained will be transferable to others, just as it has been for other topics (such as ICRF heating).

Single-Particle Containment. Energetic alpha particles must first have single-particle drift containment. This issue is so fundamental to magnetic containment that it is presumed to have been solved for the fuel ions. The very high energy of alpha particles can, however, give rise to special considerations. Their gyroradii must be small compared with the plasma size, and their drift orbits must be closed within the plasma, leading to conditions on the magnetic field design. In a toroidal system, the poloidal field must exceed a minimum, whether it be created by currents external or internal to the plasma. In a tandem mirror, the central-cell field must be above a minimum set by the radius and, because the plugging potentials are insufficient, the axial mirror ratio must be sufficient to magnetically trap a large fraction of the alpha particles.

Macroscopic Equilibrium and Dynamics. Alpha particles could affect the macroscopic features of a fusion plasma in several ways. Most directly, they add a partial pressure of typically 10-20%, which decreases the allowable pressure of fuel ions. Kinetic effects could, however, be even more important. For example, unstable alpha pressure-gradient modes can arise when the alpha drift frequency (increased owing to their high energy) resonates with normal modes of the plasma. This would likely lead to an enhanced alpha loss, analogous to the "fishbone-oscillation" ejection of neutral-beam ions from tokamaks. If, by this or any other mechanism, the plasma potential were increased by a preferred alpha loss, the associated rotation would affect the equilibrium and might cause instability.

Wave-Plasma Interactions. The slowing-down spectrum of alpha particles is usually monotonic, effectively eliminating energy inversion as a driving mechanism for microinstabilities. An exception is the tandem mirror, in which the alpha distribution must have a magnetic loss cone. Here, microstability requires a central-cell mirror ratio of greater than five for stability. In addition, there are a variety of modes that are carried by the background plasma that can be destabilized by a pressure gradient in the alpha component. If these modes served only to increase the energy-transfer rate from alpha particles to ions, they might actually be beneficial. If they enhanced the alpha loss rate, however, they would be destructive. In the latter case, the avoidance of these modes would be largely a design issue involving proper selection of machine parameters, but this is an important conclusion to be tested by experiment.

Burn Control and Ash Removal. This area is possibly the most important one for burn physics, since it is the least understood and is likely to be the most device-specific. The stability of the burn in a fusion plasma (i.e., whether the temperature

and/or density are stable under burn conditions) depends on the power dependence of the energy and particle confinement. Even for the tokamak, these parameters are known only for conditions well below ignition; for other devices, even less is known. The different confinement processes of the tokamak, RFP/spheromak, FRC, tandem mirror, or stellarator all suggest different answers to the question of stability. At this time, our experience base is too limited to transfer information confidently from one configuration to another.

Related (through the particle lifetime) to the issue of density stability, a second important issue for ignition is that of the removal of alpha-particle ash. The issue for all devices is one of dilution of the fuel by the ash buildup, but this issue is even more serious in the tandem mirror because of potential accumulation in the thermal barrier. The issue is a critical one that probably cannot be fully resolved without long-pulse burn experiments.

REFERENCE

1. J.D. Callen et al., University of Wisconsin, *TPA Plasma Science Final Report*, Dec. 1986.

OVERVIEW

The pace at which performance is improved and understanding is developed on plasma physics devices is frequently determined by the availability of reliable, high-performance, economical components to heat, fuel, and magnetically confine the plasma. Advances in neutral beam technology led to increased temperatures, first on the Oak Ridge Tokamak (ORMAK) (1975) and then on the Princeton Large Torus (PLT) (1978), and provided the basis for the major advances on mirror experiment at LLNL (2XIIB) (1974). Similarly, the use of high-field magnet technology resulted in increased $n\tau$ values on Alcator C. More recently, pellet injection has improved confinement on Alcator C, Doublet III, and the Tokamak Fusion Test Reactor (TFTR).

The further resolution of the Magnetic Fusion Program Plan (MFPP) key issues on plasma confinement and burning plasmas will require continued advances in plasma technologies. In the future, components that control impurities and exhaust spent fuel will be critically important to producing long-pulse to steady-state fusion plasmas. This trend was evident in the importance of improved limiter materials and designs for high-power heating experiments such as PLT, and now TFTR and Joint European Torus (JET). In addition, components must be developed that can be remotely maintained and are compatible with the neutron and plasma environment of a fusion reactor. Thus, plasma technology development will also contribute to the resolution of the fusion nuclear technology issue.

Plasma technology development is also directly tied to the economic objectives of fusion. Estimates suggest that as much as 20% of the direct cost of a typical fusion reactor will be for plasma-related components. In addition, the overall size scale of a facility is influenced by the performance of these same components. Thus, high performance at minimum cost will be a central concern. High reliability will be required in order to minimize operating costs.

The range of components in plasma technologies includes the following: magnets, heating and current-drive systems, fueling systems, and plasma-interactive components. These technologies are characterized by (1) the need for close interaction with confinement programs in order to understand near-term requirements, (2) the need for experiments on plasma devices in order to provide data or develop and validate techniques, (3) the need to follow a development strategy that addresses both physics and technology issues in an iterative fashion, and (4) the need to interact with reactor-systems studies to identify long-range reactor requirements. Because the toroidal program is more extensive than others, and because it is expected that the ignition and long-burn facilities will be based on the tokamak, the individual discussions are oriented toward the tokamak program. As the development of other concepts advances and longer-term needs become better defined, the emphasis in the supporting technology-development programs must shift accordingly. The required changes for concepts like the stellarator and mirror, which have technology requirements similar to those of the tokamak, could be accommodated with relatively minor changes. Different approaches (e.g., a pulsed field batch burn FRC) would lead to a very different plan.

Although the blanket and first-wall considerations will receive most attention, particularly with respect to waste disposal, plasma support components will also have a strong influence in determining the overall environmental characteristics of a fusion power plant. Thus, components must be developed that enhance the safety and environmental attractiveness of fusion systems.

Because of the close coupling between plasma technology and the plasma confinement and burning plasma issues, some aspects of plasma science are discussed in this chapter (see Chapters 1 and 2 for a thorough treatment of these issues). Plasma technology requirements, as well as the attendant issues and objectives, can be quite specific to a given confinement concept. Thus, only a subset of the plasma technologies discussed in this chapter may be required for any particular confinement concept. Where the concept-specificity of an issue is assessed in the issue summary tables (Tables 3.9-3.11), "generic" should be interpreted as generic to those concepts employing that technique or technology. In addition, an attempt is made to clarify each issue as to whether it affects the feasibility or the attractiveness of that particular approach or technology (not necessarily of fusion as a whole).

The overall logic diagram for plasma technology is shown in Fig. 3.1. In the late 1980s, inputs are needed to support the facility that addresses short-pulse ignition physics. Similar support is required in the early 1990s for a facility that would further advance the understanding of burning plasmas and provide a testing capability for fusion nuclear technology. The long-burn facility will involve a larger engineering development activity than the short-pulse facility and will make greater demands on plasma-interactive components (PICs), remote maintenance, and magnet technology. Figure 3.1 also indicates schematically that near-term R&D will provide the basis for narrowing options and resolving such issues as the feasibility of current drive, the need for nonwater-cooled in-vessel components, and the verification of ion cyclotron heating (ICH) as a viable high-power heating system for tokamaks. These stages are characterized in Fig. 3.2, where the movement from concept development to short-pulse, to long-burn, to fusion-environment operation is indicated.

Not illustrated so clearly in Fig. 3.1 is the role of ongoing confinement experiments in providing test data for plasma technology development. Present experiments, as well as the ignition and long-burn facilities, will verify the performance of components and concepts utilized in their operation and serve as test vehicles for advancing those systems to the next stage of development.

The first five decision/evaluation points for plasma technology, <a-e> in Fig. 3.1, are associated with selections of particular technical approaches, which in turn dictate subsequent development tasks. These selections are as follows: choice of heating system <a>, choice of current-drive technique for long-burn facility , evaluation of nonwater-cooled PIC performance <c>, determination of the desired pellet velocity for tokamak fueling <d>, and evaluation of the potential for useful heat from PICs <e>. Those options chosen will specify the development required, both for the long-burn facility itself and for components to be tested in the facility. The second group of evaluation points, <f-h> in Fig. 3.1, involves an analysis of the performance of the components on the long-burn facility and of the need for developing new techniques for improved performance.

FIGURE 3.1 Level 1 Logic Diagram for Plasma Technology (Decision points <a-i> are described in Tables 3.1-3.6.)

Finally, based on this development, the data will be available for evaluating plasma technology, both for the integrated fusion facility (IFF) and for the overall fusion assessment <i>. Decision points are presented in more detail in Tables 3.1-3.6.

3.1 MAGNETS

Magnets that generate strong magnetic fields are an essential requirement for the confinement of fusion plasmas. Magnets may also be used to heat the plasma inductively. Most past experiments have used copper-resistive magnets, because of the pulsed nature of the experimental systems. For example, the TFTR has produced fields of 10 T at the magnet, with durations of a few seconds. However, as plasma confinement times become longer, the hundreds of megawatts of resistive power to the copper magnets becomes difficult and expensive to sustain. Thus, large, steady-state superconducting magnets have been developed for the Mirror Fusion Test Facility (MFTF) and the Large Coil Task (LCT) and are now being tested. Many reactor applications will benefit from efficient, cost-effective superconducting magnets, although it may be necessary to retain pulse capabilities for plasma heating and control. Copper magnets may be appropriate for high-beta devices, which can employ thinner radiation shields.

3.1.1 Technical Issues, Objectives, and Alternatives

The three major subelements in magnet development are resistive magnets, pulsed magnets, and superconducting magnets.

Resistive Magnets

Resistive magnets are appropriate for high-beta reactor systems or for short-pulse plasma facilities. In the near term, copper-magnet development is needed for the

FIGURE 3.2 Characterization of Tasks for Plasma Technology

TABLE 3.1 Decisions on Heating System <a> and Current Drive for Long-Burn Demonstration

Statement of Decision

Specify current drive and heating system for the long-burn facility. These decisions are coupled, because presently proposed noninductive current-drive techniques supply heat to the plasma, as well as drive currents. In addition, the impurity control decision for the long-burn facility may be coupled as well, because of the observation that impurity transport is strongly influenced by neutral beam heating.

Decision Criteria

The chosen system(s) must deposit source energy efficiently ($\geq 70\text{--}80\%$), with the desired radial profiles and a minimum of unfavorable plasma interactions.

Current must be induced with the desired profile with an efficiency of >0.1 A/W.

The radio frequency (rf) or neutral beam source must meet the performance criteria implied by long-pulse burning with sufficient reliability and maintainability.

In general, the specific economic and safety/environmental criteria may represent a compromise between carrying out the immediate long-burn mission as cost-effectively as possible and establishing a technology base for an IFF.

Sources of Information

Confinement experiments that demonstrate heating at significant levels, about 10 MW and/or 1 MA of driven current.

Component-development tasks with plasma data on launcher and source performance.

System studies that indicate the needed technical and economic characteristics of heating and current-drive systems for an attractive reactor.

Outcomes and Consequences of Decision

The determination that an attractive heating/current-drive system can be developed for the long-burn facility will support the decision to initiate construction and provide specifications for component development. It is possible that this technology, while adequate for the immediate need, will give rise to additional issues that must be addressed in the longer term. In this event, additional development of components or the generation of new approaches may be necessary to meet the later requirements. Based on present understanding, it is likely that heating techniques will be satisfactory. Less certain is the existence of an attractive current-drive technique.

TABLE 3.2 Decision <c> on Nonwater Coolants for Plasma-Interactive Components

Statement of Decision

Evaluate the capability of using coolants other than water for in-vessel components, such as limiters and rf launchers.

Decision Criteria

Liquid-metal blanket systems are attractive for commercial applications.

Materials are available that are compatible over long periods with nonwater coolants.

Heat-removal rates under fusion-relevant conditions are adequate to cool PICs with nonwater coolants.

Safety requirements for nonwater coolants are satisfied.

Coolants, materials, and thermal characteristics of components must not adversely affect the potential for tritium breeding and useful heat extraction.

Source of Information

Nuclear technology provides information concerning viability of liquid blankets.

Systems studies provide information on operating conditions and desired features of commercial systems.

Materials research provides information on corrosion, strength, and radiation response of candidate materials.

Plasma confinement concepts provide information on the expected environment for in-vessel components.

Plasma technology and nuclear technology supply information on heat-removal capability for nonwater coolants under fusion-relevant conditions.

Outcomes and Consequences of Decision

Nonwater coolants are unacceptable for removing heat from in-vessel components. This conclusion would adversely affect the use of liquid-metal blankets.

Nonwater coolants can remove heat, but their efficiency is low. Liquid-metal systems can be used, but their attractiveness is reduced.

Nonwater coolants can remove heat at high thermal efficiency. Attractive liquid-metal systems are possible.

TABLE 3.3 Decision <d> on Pellet Fueling Requirements

Statement of Decision

With present technology, pellet injectors can produce pellets with velocities in the range of 1-2 km/s. Even with anticipated improvements, velocities greater than 3-4 km/s will not be available in the early 1990s. The pellets will only penetrate 10-20% of the radius of a reacting plasma. Under these conditions, plasma-density profiles may be too broad for optimum confinement.

Decision Criteria

Determine that the density profiles required for optimum confinement can be maintained with shallow fueling.

Sources of Information

Pellet ablation models, plasma fueling and confinement experiments on JET and TFTR.

Outcomes and Consequences of Decision

- If pellets with $v = 2-4$ km/s produce the needed density profiles in a reacting plasma, then subsequent development is incremental and will focus on such issues as cost, reliability, and experimental flexibility in pellet size, speed, and timing.
- If high velocities are required, then a major effort must be undertaken to develop pellet injectors with $v > 10$ km/s. If such technology cannot be developed, then techniques such as neutral beam injection or plasma injection may be necessary to provide additional density control.
-

short-pulse ignition device. This high-field application is expected to be started in 1988. Initially, the properties of existing copper materials must be measured and characterized for evaluation in 1988 to support the design of the ignition device. On a longer time scale, higher-strength materials (e.g., dispersion hardened copper) would be developed for possible use in construction of the device and for use with concepts that can use resistive magnets in their reactor configurations.

For some applications, demountable, high-current-capacity (100-kA) electrical joints are needed. In addition, some designs require that the joints accommodate relative motion (i.e., sliding joints). Successful joint development would reduce both the cost and construction time of future machines, because it would permit parallel fabrication and preassembly of major items (which could then be more easily field-assembled).

TABLE 3.4 Decision <e> on Useful Heat Removal from Plasma-Interactive Components

Statement of Decision

Evaluate the ability to remove useful heat from in-vessel components. In a DT system, 20% of the energy is deposited on the first wall and/or in-vessel components. Recovery of this heat for power production is highly desirable.

Decision Criteria

Determine if a reasonable amount (20-35%) of the plasma alpha energy can be extracted and converted to electricity.

Sources of Information

In addition to the sources of information described in Table 3.2, additional information regarding the limits on maximum coolant exit temperature imposed by the material's thermal, mechanical, corrosion, and radiation-damage characteristics is needed. These data will be obtained from materials irradiation studies and from tests on corrosion loops and thermal testing facilities.

Outcomes and Consequences of Decision

If useful heat recovery is not possible, then the electric output of a DT fusion reactor will be reduced 10-20%, with a corresponding economic penalty.

Pulsed Magnets

High-field, compact, ohmic-heating (OH) coils in the central cores of tokamaks have been demonstrated to exert a major beneficial effect on the size, performance, and cost of these machines. Reviews of past machine designs -- such as International Tokamak Reactor (INTOR), the MIT proposed Alcator steady state tokamak (DCT) experiment, and such integrated-test-facility designs as Fusion Engineering Reactor (FER), Next European Torus (NET), and U.S. Engineering Test Reactor Design (TIBER) -- have indicated the need for a high-performance, central OH coil. A typical tokamak design is a superconducting solenoid with an outside diameter of 1.8 m, an inside diameter of 1.2 m, a peak field of about 10 T, and a swing rate of about 1 T/s (up to 10 T/s for disruption scenarios). This requires an alloyed Nb₃Sn superconductor operating at a winding-pack current density of 40 A/mm² for compact experiments and reactors that require OH coils.

For some designs of near-term superconducting tokamaks, the use of a small, high-field "pusher" plasma-shaping coil, mounted on the center column of the machine and producing fields of about 15 T, offers benefits in plasma shaping. A demonstration of

TABLE 3.5 Decisions <f, g, h> on Plasma Technology Performance for Long-Burn Demonstration

Statement of Decision

Evaluate performance of plasma technologies on the long-burn facility.

Decision Criteria

Accomplish successful operation by demonstration of the maintenance of burning plasma for long pulses (hundreds to thousands of seconds). Operation in the long-burn facility, compared with test-stand and materials data, should provide the information required to ensure that the requirements of the IFF can be met.

Sources of Information

Plasma data on component performance in a long-pulse burning plasma will be obtained as a direct result of demonstrating the ability to maintain such a plasma.

Component compatibility with the fusion nuclear environment at low-to-moderate fluence will be demonstrated during the later phases of the long-burn program.

Extrapolation to high fluence must be based on materials-irradiation testing and test-stand simulations.

Outcomes and Consequences of Decision

This evaluation will determine whether additional development or new concepts are required in order to meet IFF requirements. In general, remaining development will likely involve establishing economics, including both construction and reliability factors, and ensuring compatibility with an increasingly demanding nuclear environment.

the feasibility of such a coil would represent a significant, yet achievable, accomplishment in superconducting magnet technology. This demonstration can be achieved by building an alloyed Nb₃Sn insert, with an outside diameter of 1.1 m and an inside diameter of 0.6 m, for the multipurpose-coil program.

Economical energy-storage and control systems need to be developed to reduce power-handling costs for pulsed tokamaks. Systems costing less than \$0.10/J are needed for experimental and reactor attractiveness. Inductive energy storage using pulsed magnetic-field coils may meet this need.

Superconducting Magnets

The operation of the MFTF and LCT magnet systems represents a major milestone and provides the technical base for a long-burn demonstration device to begin the

TABLE 3.6 Evaluation <i> of Plasma Technology for an Integrated Fusion Facility

Statement of Decision

Evaluate plasma technologies for the IFF.

Decision Criteria

Confidence must be established that the requirements for the IFF can be met. These requirements involve plasma and engineering performance, safety and environmental attractiveness, and reactor economics.

Sources of Information

Expanded long-burn testing, plasma-performance testing from advanced concepts, and safety, environmental, and economic characteristics of systems must be evaluated.

Outcomes and Consequences of Decision

Failure to meet objectives could result in delays on an IFF, as well as the need for new concepts and more development.

design stage in 1988, construction in 1992, and operation by 1998. Superconducting materials (especially Nb_3Sn) need to be optimized for current-density, strain-tolerance, and neutron-damage limits starting in 1986. Even the very best of today's Nb_3Sn conductors have intrinsic current densities about half that of the best research-laboratory conductors. Further advances are possible if the strong flux-pinning shown by NbTi can be achieved in Nb_3Sn . Basic scientific studies of Nb_3Sn indicate the potential to decrease the cost per ampere-meter of Nb_3Sn conductor by factors of two to five.

An advanced superconductor worth further study, NbN , has critical-field values of 20-40 T, the upper limit being twice that of Nb_3Sn . Additional attractive features of NbN are that the critical current is strain-independent up to rupture, the radiation tolerance is expected to be high, and the compound can be made on high-modulus carbon fibers. Very little is known, however, of the variables controlling the critical currents and field. A conductor fabrication process needs to be developed for this material.

Structural materials and processing technology are needed to develop improved steels that will be stronger (1200-MPa yield strength), with greater ductility and fatigue resistance (200-MPa/ $\text{m}^{\frac{1}{2}}$ toughness), at 4 K. Reviews of recent tokamak designs (such as TIBER and FER), as well as tandem-mirror designs [such as the Mini-Mirror Advanced Reactor Study (MINIMARS)], have indicated that the use of stronger cryogenic structures will materially reduce the mass of "inert" structure and increase the space available for access to the fusion core. Emerging classes of structural alloys include the Fe-Cr-Ni-N and Fe-Mn-Cr-N austenitic steels with 4 K toughness and strength combinations at least 75% better than those of current stainless steels. However, there is a need to

characterize the austenites' mechanical properties when made in large heats, to qualify production and welding processes, and to develop statistically based mechanical-property design allowables.

The development of insulators and conductors that are resistant to radiation damage up to 10^{12} rad and 10^{25} neutrons/m² would allow thinner neutron shields and lead to significant reductions in machine sizes and costs. However, magnets must be designed and demonstrated to operate reliably and safely at these higher neutron fluences and at nuclear heating rates of up to 10^4 W/m³ (10 mW/cm³).

The objectives and attributes associated with the magnet issues are illustrated in Fig. 3.3. For example, the objective of developing pulsed magnets incorporates the lesser objectives of increasing OH coil performance (with the attribute of 10 T/s) and increasing plasma-shaping coil fields (with the attribute of 15 T). These attributes and the others shown would be used in evaluating magnets for an IFF.

3.1.2 Program Logic

Figure 3.4 illustrates the program logic to resolve the magnet issues. Present developments in copper coils would address materials properties and design for a decision on the short-pulse ignition experiment. For more advanced superconducting experiments, a multipurpose-coil program designed to serve both purposes would start in 1987 with design and conductor procurement. A pulsed coil would be tested in Japan at the Japan Atomic Energy Research Institute (JAERI). Meanwhile, a high-field (16-T) insert coil would be built and both coils tested together in the High-Field Test Facility (HFTF) at Lawrence Livermore National Laboratory (LLNL). This activity provides the scalable conductor technology necessary for the subsequent design and development of a long-burn-facility magnet system. Continuing development in advanced conductors and cryogenic materials will lead to a major decision on magnets for the IFF around the year 2005. A description of this decision is given in Table 3.7.

While government laboratories and universities can do research and development efficiently, the incorporation of industry's disciplined engineering approach to the generation and application of suitable design-basis documents and material specifications is essential to begin the process of technology transfer to U.S. industry. Even the manufacture of pilot quantities of production materials and devices is best done by industry. The magnet industry (superconductor suppliers, magnet manufacturers, and producers of structural alloys, copper alloys, and insulators) needs sustained involvement in the fusion program to enhance economic competitiveness and to develop special fusion-related capabilities. Widespread involvement is desirable, so that the costs of future fusion devices are not dictated by single-point industrial sources.

Facilities

No new major facilities are needed for magnet development. Instead, the minor revisions of existing facilities listed in Table 3.8 would be sufficient. The program costs represent total expenditures needed to support the development of the magnet systems for an Engineering Test Reactor (ETR), including prototypes.

FIGURE 3.3 Objectives and Attributes for Magnets

3.2 HEATING AND CURRENT DRIVE

Progress in the development of any particular confinement configuration and in the development of improved configurations is often led by the development of efficient, flexible plasma-heating systems. These can be used to increase plasma temperatures, produce particular plasma components (e.g., hot electrons in a tandem mirror thermal barrier), drive plasma currents (e.g., tokamak current drive), modify plasma potentials (e.g., thermal barriers), or aid in plasma stabilization (e.g., Elmo Bumpy Toroid [EBT] rings or tandem-mirror sloshing ions). A wide variety of techniques have been employed,

FIGURE 3.4 Level 2 Logic Diagram for Magnets (Decision point <a> is described in Table 3.7.)

including (1) slow electromagnetic techniques, such as ohmic heating and adiabatic compression; (2) wave heating, in the range from tens of kilohertz (Alfvén waves) to tens of megahertz (ion cyclotron frequency), to a few gigahertz (lower hybrid frequency), to 10–100 GHz (electron cyclotron frequency); and (3) beam heating with ions and electrons. In general, effective techniques are characterized by reliable and efficient components and by well-understood, beneficial interactions with the plasma. In addition, the long-range potential of any specific technique must be judged in terms of such factors as maintainability, cost, and compatibility with the reactor environment. Specific comments on each of the major techniques are presented below.

TABLE 3.7 Decision <a> on Magnets for an Integrated Fusion Facility

Statement of Decision

Choose the optimum magnet design for an integrated fusion facility.

Decision Criteria

The magnet must be capable of producing fields of at least 12 T with current densities above 2 kA/cm².

The construction approach must be capable of reliable operation and low-cost construction.

Sources of Information

The multipurpose-coil program.

Magnet experience from long-burn-demonstration magnet construction.

Base programs on conductor development and structural materials.

Outcomes and Consequences of Decision

Low performance in field and current density leads to larger reactors.

Unreliable performance expectation would require complicated maintenance procedures and imply low plant availability.

Neutral Beams

Plasma heating by neutral beams is well understood and has been the basis for many of the major advances in toroidal and mirror confinement physics. The systems employed to date have been based on positive-ion beams and have utilized ion sources with energies up to 60 keV/nucleon and with currents up to 75 A. These ion sources, together with related beam-line components (neutralizers, deflection magnets, pumping systems, and ion dumps), have resulted in beam lines with a power of 1-2 MW per ion source and a total power of 6-8 MW. Such systems are the mainstay of present experiments, but they present problems in the long term. As plasma sizes increase, the neutral beam energy required for efficient heating also increases. For toroidal systems, energies in the 70- to 125-keV/nucleon range are required; at these energies positive-ion-based systems become very inefficient. Thus, systems based on negative ions or some form of energy recovery appear necessary.

Perhaps more fundamental is the problem of physical proximity of the complex beam line to the plasma. Reactor studies suggest that maintenance and reliability considerations will make the beam lines both expensive and (perhaps) a dominant maintenance driver. As a result, the emphasis for toroidal systems has shifted (where possible) to the use of radio frequencies (rf) for heating and current drive. For these applications, neutral beams are a backup technology. Tandem mirrors, however, require energetic ions for thermal barrier production. Relatively low-current systems (2-10 A)

TABLE 3.8 Revisions to Existing Facilities

High-Field Test Facility (HFTF)

Purpose	The multipurpose-coil program would be tested in the HFTF 4-m dewar, which is large enough to house a pulsed OH outsert coil for plasma shaping.
Features	4-m-high, 4-m-diameter cryostat to house multipurpose test coil; 10^{-6} Torr vacuum for thermal insulation.
Costs	The facility revision costs will be minimal, but the annual operating costs of the multipurpose-coil program will be \$5 million/yr from 1987 to 1992.
Date Required	1988.

Large-Coil Test Facility (LCTF)

Purpose	Test toroidal field (TF) coils for the long-burn experiment.
Features	10-m-high, 10-m-diameter cryostat to hold TF coil; 10^{-6} Torr vacuum for thermal insulation; 5 kW of flowing helium at 4.2 K to simulate neutron heating.
Costs	The facility revision costs will be about \$5 million, and the annual operating cost from 1993 to 1996 would be \$6 million/yr.
Date Required	1992.

at several hundred keV are required. Negative-ion beams are preferred for mirror systems.

Ion Cyclotron Heating

A number of different modes of propagation and mechanisms for absorption exist for waves in the ion cyclotron range of frequencies. The ion cyclotron frequency ($f_{ci} = 7.6 \text{ MHz/T}$ for deuterium) implies frequencies in the 40- to 80-MHz range for second-harmonic heating (i.e., $f = 2f_{ci}$) in deuterium (for machines with moderate magnetic fields). This technique is potentially very attractive, because power at this frequency can be generated efficiently using technology closely related to that of short-wave broadcast transmitters, and most of the heating components can be located away from the fusion reactor. Several different approaches, based on fast-wave heating at $2\omega_{ci}$, fast-wave minority-species heating, slow-wave heating at ω_{ci} , and ion Bernstein-wave heating, are now under development.

Experiments in tokamaks and stellarators and related technology development are now suggesting that the promise of rf relative to neutral beams can be fulfilled. As a whole, the physics of ICH is not as well understood as the physics of neutral beam heating, and significant advances are required to bring ICH understanding to an equivalent level. Components for rf heating are just now being developed that show promise of being extrapolated to reactor conditions (in terms of power density, maintenance, and reliability).

Lower Hybrid Heating

Lower hybrid heating (LHH) has been studied at high power in several experiments, with mixed results. Although good electron and ion heating efficiencies have been observed in certain cases, there is a lack of reproducibility among tokamak experiments with respect to the efficiency of ion heating. Theoretical understanding of LHH is not as far advanced as that of other rf heating methods; significant uncertainties exist in the parallel spectrum requirements, the transition from electron to ion heating, and the response of the electron distribution function to the applied rf.

Current-drive experiments have been more successful. Lower hybrid waves have been utilized for the startup and flat-top phases of a tokamak discharge. Improved efficiency at high density is central to developing a successful current-drive technique. Regarding the technology, microwave sources in the 2- to 8-GHz range are available, albeit not at the power levels that will likely be required for reactors. There are significant technological issues regarding the lifetime of LHH launchers in a reactor-like environment, because the launchers must be located close to the plasma edge for good coupling efficiency.

Electron Cyclotron Heating

Relative to other rf heating techniques, the basic propagation and absorption physics of electron cyclotron heating (ECH) is well understood. Unlike the case with ICH, the central problem is the availability of high-power sources. During the past ten years, there have been significant advances in sources, with gyrotrons now available at 28 to 60 GHz at powers of 200 kW, and with 30-40% efficiency. Higher power and frequency tubes (140 GHz at a power of 100 kW) are now under development. The development of tubes at frequencies in the 120- to 150-GHz range (needed for reactors) at powers of above 50⁴ kW will require further development. If such tubes were developed, it is likely that they would find ready use either as catalysts for current drive or for profile control, and perhaps even for bulk heating if tubes could be developed at the several-megawatts level. Free electron lasers, currently under development for defense applications, may fill this need.

The discussion here has focused on those techniques that are being developed on a broad basis. A number of other approaches are in an earlier state of development for tokamaks and mirrors or have specialized applications to other concepts: Alfvén-wave heating on tokamaks and stellarators, adiabatic compression on field-reversed configurations, direct current (DC) helical injection on spheromaks, and F- θ pumping on

reversed-field pinches are examples of heating techniques in this category. As the development of specific techniques or configurations becomes relatively more advanced, the issues related to those techniques will need to be incorporated into this plan.

3.2.1 Technical Issues, Objectives, and Attributes

The issues for plasma-heating systems are listed in Table 3.9. These issues are generally grouped by technique, with the last issue referring to the general question of current drive in toroidal devices; a number of different techniques, including LHH and neutral beams, have been proposed to deal with this last issue. These issues are presented in more detail in the following discussion. The objectives and attributes for the plasma-heating program element are shown in Fig. 3.5.

Negative-Ion Beam System Attractiveness

A negative-ion-based neutral beam line has both performance requirements (principally current, voltage, and efficiency) and broader system requirements (such as reliability, ease of maintenance, degree of safety, and moderate cost). The ability to meet these requirements depends strongly on the performance of the major beam-line components (the ion source, accelerator, and neutralizer being the most critical components) and on how well these components are integrated into a beam-line design. The system design will determine how easy components are to maintain, determine the performance requirements of individual components, and control the degree of neutron activation (which, in turn, controls the maintenance strategy). There will likely be a significant difference in requirements for a beam, for example, that must operate in a steady state to maintain a sloshing-ion distribution or current drive, compared with those for a pulsed system used to heat a device to ignition. In addition, the neutron environment may be less severe for a tandem mirror when the beam line is located in a region where the fusion neutron flux is significantly reduced.

Negative-Ion Beam Component Performance

Ion Source. Present estimates indicate that neutral-beam currents in the range of 2 to 10 A are required for tandem mirrors. Beams for current-drive applications are also likely to be in this range because of requirements on overall power efficiency. Negative-ion sources should operate at high gas efficiency in order to minimize stripping before final acceleration and should have a minimum number of electrons. Many elements of such an ion source have been demonstrated.

Accelerator. The accelerators used in present neutral-beam systems are electrostatic Pierce-type systems, which (for each aperture) accelerate up to an ampere. This type of acceleration has been successfully used to produce positive-ion beams of 0.37 A at 600 kV and may be effective for high-current beams in the 200- to

TABLE 3.9 Heating and Current-Drive Issues

Issue	Potential Impact ^a	Design Specificity
Negative-Ion Beam System Attractiveness (neutral beam technique)	F3	Generic
Negative-Ion Beam Component Performance (neutral beam technique)		
Ion Source	A1, A3	Generic
Accelerator	A1, A3	Generic
Neutralizer	A1, A3	Generic
Identification of Attractive Ion Wave Heating Mode (ICH)	F1	Generic ^b
Development of ICH Components		
Launchers	F3	Generic ^b
Feedthroughs	F3	Generic
Matching Systems	A3	Generic
Power Sources	A3	Generic
Understanding of LHH Power Deposition	A1	Closed Systems ^b
LHH Components		
Launcher	F3	Closed Systems
Source	A3	Closed Systems
ECH Tubes	F1	Generic
Identification of Efficient Current-Drive Technique	F1	Tokamak

^aSymbols for potential impact are as follows:

Feasibility Issues

- F1 - May close design window.
- F2 - May result in unacceptable safety risk.
- F3 - May result in unacceptable reliability, availability, or lifetime.

Attractiveness Issues

- A1 - Reduced system performance.
- A2 - Reduced component lifetime.
- A3 - Increased system cost.
- A4 - Less desirable safety or environmental impact.

^bAnswer is concept-specific.

*Word hierarchy to establish attribute that characterizes level of understanding might be:
 theory - exploratory, developing, well developed, established
 experiments - just beginning, some experimental data, good experimental data base, well established.

FIGURE 3.5 Objectives and Attributes for Heating and Current Drive

300-kV range. High-energy systems have longer acceleration paths, and systems that decouple the focusing and acceleration fields are desirable. The most promising candidate for this application uses a transverse-electric-field focusing system with alternating transverse electric fields, providing both acceleration and strong focusing. Electrostatic accelerators are preferred for systems where high efficiency is required, but rf accelerators may be acceptable where lower efficiency is acceptable.

Neutralizer. Three possible approaches to neutralization have been analyzed: gas targets, plasma targets, and photodetachment. Significant development of the lasers for the photodetachment neutralizer is required, but its higher efficiency (100%, in contrast to 60-80% attainable with other approaches) and its elimination of the need to handle gas make it the preferred approach. Gas or plasma neutralizers may be used in the near term if the lasers are not sufficiently advanced.

Identification of Attractive Ion Wave Heating Mode

Radio-frequency heating methods have been under intense development and have been chosen as the prime heating candidates for reactor-size devices. Because of the comparatively simple rf power-generation systems and the projected favorable characteristics of wave penetration into the plasma core, rf heating methods are perceived to have important cost and efficiency advantages over other auxiliary heating methods. Of the various rf heating methods under investigation, the method that has achieved the greatest success to date, in terms of both delivered power and heating efficiency, is the ICH. Over 5 MW of power has been delivered to tokamaks, resulting in full thermalization of the wave energy.

In experiments throughout the worldwide fusion program, various heating mechanisms have been identified as potential candidates for heating in ignition devices. The mode most thoroughly studied experimentally has been heating by minority absorption of fast waves. This mode is effective, but it may require species mixtures that would not be optimal in a reactor. Another mode that has been used successfully is second-harmonic fast-wave heating; this mode appears to be very promising and has a solid experimental and theoretical footing. A third mode for consideration is ion Bernstein-wave heating in the range 1.5-4 times the cyclotron frequency. This mode has been used much less than the others, but it appears to be especially effective in large, hot plasmas. However, a broad experimental data base is lacking. Ion Bernstein-wave heating requires higher frequencies than the other modes (a possible disadvantage), but it could be launched using a waveguide coupler instead of a loop (a possible advantage). Much of the emphasis in present and planned ICH research will be on optimizing efficiency and developing technology and systems configurations compatible with reactor constraints.

Development of Ion Cyclotron Heating Components

Common to all ICH applications are a number of physics and technology questions related to antenna design and the interaction between the antenna and the surface plasma. The optimal wave polarization and excitation wavelength for a given heating situation have not yet been fully determined. Furthermore, appropriate antenna designs that can both excite waves with desired spectral characteristics and satisfy reactor compatibility requirements are yet to be determined. Much of the current attention is focused on the problems associated with the use of a Faraday shield and on the need to simplify the radiating structures as much as possible. An integral part of the antenna-design problem is the heating and transport physics in the near zone of the antenna. In particular, the nature of the electrostatic antenna loading observed in previous experiments needs to be better understood, especially in relation to the proposed Bernstein-wave launching method. Also, better understanding of the possible effects of rf on edge plasma transport and the ensuing wall ion bombardment is of paramount importance for minimizing deleterious effects on the plasma and improving antenna longevity.

Launchers. In addition to providing the electromagnetic conditions required for future ignition devices, the antenna must survive the fusion environment as an integral part of the plasma-vessel interface. Present launcher designs are only for short-pulse, relatively benign plasma edge conditions; a concentrated development effort is required to provide the desired properties while permitting large heat and particle-bombardment loads at the interface.

Three generic types of antenna structures are possible -- loops, cavity-mounted devices, and variants of a waveguide are the most studied. Large loop launchers have been widely employed to date. Some cavity-type experiments have been performed, such as those on the Elmo Bumpy Toroid-Square (EBT-S). No experiment has used a waveguide launcher for ICH waves.

The most significant development required for the ICH launchers is the Faraday shield. The demands of the plasma environments of future ignition devices on the Faraday-shield elements are severe. Plasma heat and particle loads require the use of protective materials (such as graphite) and active cooling. The devices needed for these requirements must also withstand very high tokamak plasma-current-disruption forces. The cavity and waveguide types of launchers offer the best opportunity of meeting future needs, but the large protruding loop permits better control of the launched spectrum. Intensive investigations in these areas, already under way, need added emphasis.

Feedthroughs. The rf feedthrough, the vacuum interface that passes the rf into the plasma, has for years limited the capability of transmitting ICH power to the plasma. This limiting element has been redeveloped and now offers promise of feeding >1 MW cw (continuous wave) of ICH power. Means of handling the high heat loads are under investigation, and the role of multipacting in the unit is being studied. Material differences have recently manifested themselves largely in different breakdown

problems. For the long term, the ability of insulators to function in the fusion-reactor neutron environment must be evaluated. This evaluation may lead to more complex geometries, with bends to attenuate the neutron flux.

Matching Systems. Previous means of matching the antenna/plasma load to the source have concentrated on tuning stubs. These are adequate methods, yet they have some disadvantages, such as size and inconvenience in adjustment. Capacitative matching methods that utilize capacitors at the launching element are being developed and will provide a means for transmitting much greater power to the plasma. Such capacitors must have characteristics developed beyond those now available. As with feedthroughs, neutron damage may become a limiting factor for reactor application.

Power Sources. Power sources for ICH in the frequency range of 40-120 MHz and at power levels up to 2 MW are under development. Future ignition devices will require antennas that launch at 5-10 MW, and concomitant power sources will be needed. The present advance of tube technology will likely not permit a single-output amplifier with 5-MW capability for some years. The power needs can be met with combiners used with 2-MW sources, although this arrangement is more complex and costly. For high magnetic fields, higher-harmonic heating modes that require frequencies above 120 MHz, or both, the power source becomes more expensive and difficult to obtain at the needed power levels.

Understanding of Lower-Hybrid-Heating Power Deposition

Although good coupling of the rf power with multiple waveguide arrays can be achieved, ion heating results tend to be irreproducible. Electron heating in high-density plasmas, such as in Alcator C, tends to be efficient and reproducible. Perhaps the most noteworthy results have been obtained in rf current-generation experiments, in which toroidal currents on the order of hundreds of kiloamperes have been generated by injecting several hundred kilowatts of power via traveling lower hybrid waves.

A decade ago, many small-scale experiments verified the fundamental aspects of wave propagation near and above the lower hybrid frequency; more recently, the greatest interest has been in using lower hybrid waves to heat the plasma and to drive currents in toroidal devices. In the mid-1970s, LHH experiments in tokamaks were carried out at the 100- to 200-kW level, while in recent experiments powers up to 1 MW have been injected into tokamaks. Earlier lower hybrid experiments concentrated on ion heating ($\omega \approx \omega_{LH}$). In more recent experiments, electron heating and current drive ($\omega > 2\omega_{LH}$) have been emphasized. This change has occurred because bulk ion heating near the mode-conversion layer appears to be less reproducible and more difficult to achieve than electron heating with subsequent collisional bulk ion heating. The reason for this is not well understood, but it is likely that as the wave frequency gets closer to the lower hybrid frequency, the shorter-wavelength waves may be more effectively absorbed and/or scattered near the plasma surface by nonlinear effects. Toroidal effects may further complicate wave propagation to the plasma center. In addition, toroidal effects

-- combined with collisional dissipation in the outer, cold plasma layers -- become more pronounced at shorter wavelengths and further aggravate the wave-penetration problem.

Current interest in using lower hybrid waves to heat fusion plasmas stems partly from the technological attractiveness of the wave-launching structures; arrays of waveguides that may be mounted flush with the wall, or protrude only slightly beyond the wall (and stay in the shadow of the limiter), may be utilized as antenna structures. The frequencies are on the order of a few gigahertz, where cw high-power and high-gain klystrons have already been developed.

Lower-Hybrid-Heating Components

Launcher. One of the important aspects of any rf heating scheme is the coupling of rf power by the antenna to the waves in the plasma. It is necessary to match the plasma impedance to the transmission line in order to reduce the reflected power. At the same time, it is desirable to maximize the power coupled to the wave. One of the fortunate aspects of lower hybrid wave launching is that the plasma impedance can be matched to the antenna even in the absence of tuning elements. Good matching is achieved by positioning the array radially to the correct density. Hence, the waveguide array must be radially movable by bellows. A phased array of waveguides is used as an antenna, because it is necessary to slow down the wave phase velocity (as compared with c , the velocity of light) for accessibility. This can be achieved by placing several waveguides with their narrow dimension side by side, fed into the TE_{10} mode, so that the electric field is in the toroidal direction and E varies between adjacent waveguides. For reasonable plasma parameters (density gradients), theory indicates that 70-90% of the power can be coupled to the plasma; most of this power is coupled to the slow wave.

The most significant concern about the LHH launcher is that its grill must be maintained only 2 cm from the plasma. The intense heat and particle load on the grill dictates that it be actively cooled and have protective coatings. The potential problems that will exist with the plasma inside the grill at areas of large electric field are yet to be solved.

Source. The LHH rf power sources used to date in tokamak fusion devices have been klystrons. Future ignition devices and other experiments will also require frequencies in the klystron range for both heating and current-drive applications. Such klystrons are available for short-pulse and low-power applications as a result of radar development. However, heating and current-drive applications demand cw sources, and these are not yet available. The Japan Atomic Energy Research Institute is developing a 2-GHz, 1-MW klystron with a pulse length of 10 s. Higher frequencies are needed (perhaps to 5 GHz), primarily for current drive.

Electron Cyclotron Heating Tubes

Potential uses of ECH require total powers of 5-10 MW at 50-100 GHz for thermal barrier production and up to 25-50 MW at 120-140 GHz for use in toroidal-system bulk heating. System studies suggest that this power should cost on the order of \$2/W for bulk heating, with a somewhat higher figure allowable for thermal barrier production. Present systems, even at 28-60 GHz, cost about \$5-10/W and operate with 30-40% efficiency. In order to reduce this cost, efficiency must be raised to the 40-60% level and the power per tube increased to the 0.5-MW level.

The development of higher-power tubes is likely to be a very difficult task. Present systems are based on gyrotrons operating in the TE_{02} or TE_{03} modes. Power density in the tubes is very high, because the microwave cavity in which the power is generated is typically 1 cm in diameter and 2-3 cm long. Peak thermal loads in the tubes already reach several kilowatts per square centimeter and are near the limits for water cooling. As a result, higher power based on present technology is likely to be limited to 200 kW or less. Power densities can be reduced by going to higher-order modes (TE_{04} or higher) or, for example, by use of so-called "whispering-gallery" modes. The use of higher-order modes, although it does lead to reduced power density, complicates the electromagnetic design of a gyrotron: cavity resonances are closely spaced in frequency, but clean oscillation in a single mode has been demonstrated in experiments with output power exceeding 500 kW.

Identification of an Efficient Current-Drive Technique

Neutral beams, LHH, ECH, and ICH (among others) have all been proposed as a means for establishing or maintaining the plasma current in a tokamak. Ideally, such a system would, as will be discussed later, have an efficiency of greater than 0.5-1.0 A/W (~ 0.2 A/W for second stability regime) and would not have any deleterious effects on the plasma or introduce any significant cost or reliability concerns. Significant progress has been made, particularly using lower hybrid waves. Efficiencies are observed to follow the scaling $P/I \sim 0.1 T_e/nR$ (where n is in units of $10^{20}/m^3$ and T_e , a measure of the electron temperature relative to the wave spectrum, is associated with the energy of the electrons actually carrying the plasma current). If the energy of the electrons can be made sufficiently high, then efficiencies may be reasonable. The conflicting requirements posed by accessibility, which requires a parallel index of refraction greater than some critical value, and avoidance of the lower hybrid resonance, which requires a parallel index of refraction less than a critical value, may be a more serious issue. The analysis and experiments to date have been based on use of the slow-wave branch of the lower hybrid wave. The fast-wave branch appears to offer significant improvements over the slow-wave branch if the problems associated with requiring a higher density near the launcher to avoid an evanescent region can be overcome.

If the technology of steady-state, high-energy beams and their associated reactor-maintenance problems can be solved, then neutral beam current drive may prove to be an alternative technique, because of its stronger physics base and the absence of critical in-vessel components.

In discussions of needed current-drive efficiency, such as the above, amperes per watt is typically used as the appropriate figure of merit. In practice this must be judged in terms of overall reactor economics. In this context the total amount of driven current, the associated power for driving this current, and the cost of this power must be considered. In the following examples we assume that the goal for competitive fusion power is about \$1000/kW direct cost (as suggested in Ref. 1) and that the current-drive system should not account for more than 10-20% of this total. For a 1000-MW plant with a plasma current of less than 10 MA and with a cost of \$1/W for the current-drive power, an efficiency of 0.1 A/W, or even less, would be acceptable. If a plasma current of over 15 MA is required and the cost is \$3-4/W, then an efficiency of 0.4 A/W or more would be desired. For the present discussion, the near-term goal of 0.1 A/W has been adopted.

3.2.2 Program Logic

The heating and current-drive program elements are divided into four subelements: electron cyclotron heating, ion cyclotron heating, lower hybrid heating, and neutral beam injection. Figure 3.6 shows the Level 2 logic diagram for this section.

Work during the initial five-year period provides the basis for two major decisions. First, the physics and technology of various ICH heating modes will be sufficiently developed to allow selection of the specific ICH heating mode for the short-pulse ignition device. In parallel, the key issues of ECH, LHH, and neutral-beam heating will be explored in order to provide the basis for selecting the optimum mix of heating techniques for the long-burn demonstration facility. Following these selections, components for these facilities will be developed. The physics and technology data will then provide the basis for an overall evaluation in the 2000-2005 timeframe.

A parallel but later process will be followed for the development of an attractive current-drive technique. Use of current drive on the short-pulse ignition device is unlikely, but it is a possibility on the long-burn demonstration device.

The last subelement is related to the FRC, RFP, and dense Z-pinch concepts. At the present time, the necessary technology for these concepts is being developed within the physics programs and will not be discussed further in this section. It may be appropriate at a later date to include specific technology activities for these concepts.

Electron Cyclotron Heating

Electron cyclotron heating has been demonstrated to be an efficient heating technique for plasmas. The main issue that must be resolved for ECH is the development of low-cost, high-power, efficient rf sources in the 120-140-GHz frequency range. A five-year development program on high-power gyrotrons or other rf sources should be carried out with the objective of evaluating and selecting the most promising ECH sources in 1991. Given that such a source is developed, the objective by 1995 would be to develop a 1-MW, 150-GHz, steady-state ECH heating source. Following that successful development, ECH could be used to do a substantial heating experiment in an ignition-type device.

FIGURE 3.6 Level 2 Logic Diagram for Heating and Current Drive

Ion Cyclotron Heating

There are two primary near-term questions for ion cyclotron heating:

1. Which heating mode is the best for heating hot reactor-grade plasmas?
2. How should the components for these systems be developed to provide the required efficiency, dependability, and cost?

As shown in Fig. 3.7, the main path for ICH development is to perform high-power experiments in Doublet-III (D-III) and TFTR-class devices, with the goal of 10 MW of ion cyclotron radio frequency (ICRF) heating in 1989. A parallel component-development effort will be required. At the same time, different ICH modes should be studied with the goal of evaluating the relative effectiveness of different modes and selecting the most applicable mode for an ignition device by 1990. These same data will support the broader testing evaluation required for the long-burn demonstration.

Lower Hybrid Heating

Lower hybrid heating will be studied primarily for its possible use as a current-drive mechanism in toroidal devices. Although less developed, both LHH and neutral beams also have significant potential. Figure 3.8 indicates that in 1991 the various current-drive techniques should be evaluated. If attractive, one or more techniques should then be tested in the early 1990s. The objective should be to use current drive to sustain a greater-than-1-MA plasma current with a mean density of $n_e = 10^{20}/\text{m}^3$ in a hot, fusion-relevant tokamak plasma with an efficiency of about 0.1 A/W.

Neutral Beams

The present state of development of neutral beams is adequate to support near-term confinement experiments. The long-term requirements will be determined by the need for high-energy beams for specialized applications, such as tandem mirrors or tokamak current drive. Neutral beams are also considered as a backup for bulk heating in tokamak plasmas. The mirror and current-drive uses would require high-energy, negative-ion-based neutral beam systems. The energy requirements for bulk heating depend on heating-profile effects now under study; positive-ion-based systems remain a possibility. While the long-term needs for neutral beams are being evaluated, subsystem development of critical negative-ion beam components will be carried out. Based on these studies and on long-term needs, a decision can be made in 1990 on the requirements for neutral beams.

Major Facility Descriptions

The only area in which a major new facility for heating and/or current-drive technology is required is for neutral beams. Reactor applications of neutral beams will

FIGURE 3.7 Level 3 Logic Diagram for Ion Cyclotron Heating

FIGURE 3.8 Level 3 Logic Diagram for Lower Hybrid Heating

require development of negative-ion-based neutral beam systems capable of producing tens of amperes of neutral, steady-state beams at energies of 500 keV to 1 MeV. Prior to the decision point in 1990 for choosing heating and current-drive methods, the neutral-beam development program will concentrate on development and qualification of sources capable of producing multiple amperes of deuterium ions and on dc technology for accelerating these ions to the MeV range. If neutral beams are selected as an option for further development in 1990, it will be necessary to construct a Component Test Facility (CTF) to demonstrate total-system performance. This facility will resemble an actual beam line and will be capable of testing components operating at reactor-level power densities and pulse lengths. Nominal specifications for such a facility are 10 A (D^-) and 500 keV, with a pulse length of hours, and with neutron shielding; the cost is estimated to be \$20 million over 3.5 years, beginning in FY 1990. The operations budget is estimated to be about \$3 million/yr.

Facilities

No major new facilities are required for heating and current-drive technology, other than that described above for neutral beams.

3.3 FUELING

Hydrogen fuel must be supplied to a fusion plasma in order to replace particles that have been burned (in the case of a reactor) or that have been lost from the plasma core as a result of transport (the exception being those devices -- for example, the FRC -- that might employ a batch burn approach). Most confinement experiments to date have been fueled by gas from the plasma edge. The central plasma is fueled, in turn, by inward transport and/or convection, by the ionization of energetic (Franck-Condon) neutral particles produced from hydrogen molecular ions, or by charge-exchange neutral cascades. Analyses of experiments on tokamaks and theoretical calculations indicate that strong gas puffing can lead to flat profiles and enhanced beam losses by charge exchange, resulting in confinement saturation at high densities; this deterioration will be even worse for the larger reactor-size plasmas. In addition, there is theoretical analysis that suggests a class of instabilities, called n_i modes, which are driven by density profiles.

In mirror experiments and low-density tokamaks, fueling has been provided by ionization of neutral beams. However, the power required to operate neutral beams at levels suitable for fueling is excessive. In addition, neutral injection in tokamaks adversely affects plasma energy confinement, and the possibility that alpha heating might allow better confinement could not be realized if beams were required for fueling.

Thus, based on our present knowledge, the preferred plasma fueling method is the injection of high-velocity, solid hydrogen pellets. For reactors, these pellets would be about 0.5-1.0 cm in diameter and should have velocities in the range of 2-20 km/s. To date, 4-mm-diameter pellets with velocities approaching 2 km/s have been produced. Tokamak experiments have not only shown that the basic approach is feasible, but have also demonstrated that confinement can be significantly improved as a result. This could

be due to the centrally packed profiles that result from the more central deposition of new fuel.

3.3.1 Technical Issues, Objectives, and Attributes

Fueling issues are presented in Table 3.10 and described in detail below. Design specificity is generic for all three issues. The objectives and attributes for the plasma-fueling program element are outlined in Fig. 3.9.

Specification of Optimum Fueling Profiles

The desired fueling profile will depend on plasma stability and transport characteristics and can vary from concept to concept. For example, devices with strong poloidal connection, such as tokamaks and stellarators, are likely to be insensitive to poloidal variations in fueling, whereas asymmetric fueling in a tandem mirror may perturb the confining potential and lead to enhanced losses. Thus, both radial profiles and the degree of poloidal symmetry must be specified. In tokamaks, fueling at the plasma center may not be required, because (for reasons not yet understood) both gas and

TABLE 3.10 Fueling Issues

Issue	Potential Impact ^a
Specification of Optimum Fueling Profiles	A1
Pellet-Ablation Modeling	A3
Pellet-Injector Performance (velocity)	F1

^aSymbols for potential impact are as follows:

Feasibility Issues

- F1 - May close design window.
- F2 - May result in unacceptable safety risk.
- F3 - May result in unacceptable reliability, availability, or lifetime.

Attractiveness Issues

- A1 - Reduced system performance.
- A2 - Reduced component lifetime.
- A3 - Increased system cost.
- A4 - Less desirable safety or environmental impact.

*Word hierarchy to establish attribute that characterizes level of understanding might be:
 theory - exploratory, developing, well developed, established
 experiments - just beginning, some experimental data, good experimental data base, well established.

FIGURE 3.9 Objectives and Attributes for Fueling

pellet fueling fuel the center much better than present models indicate they should. Whether or not this effect applies to stellarators has not yet been determined. Central fueling may be required for tandem mirrors, because the effectiveness of inward diffusion of new fuel may be strongly limited by axial losses. In terms of technology, the use of large pellets for increased penetration is far easier than the use of high velocities. Thus, the maximum allowable density perturbation from a single pellet must be determined.

Pellet-Ablation Modeling

Once desired fueling profiles are known, the required pellet parameters must be determined from an ablation model. Early estimates of pellet-ablation rates indicated that pellets of extremely high velocity would be required. These models, however, neglected the effects of the high-density neutral gas cloud that, in effect, provides self-shielding. Models that incorporate these effects have been developed, and experimental results are in general agreement with these models over a limited range of plasma conditions. Suprathermal particles and thermal particles on the tail of the distribution function determine ablation rates; the current models have not been adequately tested in plasmas with such levels of energetic particles as will exist in reactor-like plasmas. The combination of profile effects on n and T_e and the likely existence of an even more strongly peaked energetic alpha-particle distribution may lead to ablation rates that increase very rapidly at small radii and may make central fueling prohibitively difficult.

Pellet Injector Performance

Pellets with diameters up to 4 mm and velocities up to 2 km/s at repetition rates of 5-40/s have been produced using mechanical and pneumatic acceleration. The performance limits of these approaches are not precisely known, but it is unlikely that they will exceed velocities of 2-3 km/s. Making pellets of various sizes does not appear to be a particular problem. If deep fueling is required and the allowable density perturbation is small (~10% or less), then high-velocity pellets will be required. Although concepts exist for higher-velocity injectors (such as rail guns or arc-boosted pneumatic guns), the mechanical properties of hydrogen may lead to fundamental limitations, and such injectors may not be feasible. It should be noted that at pellet velocities in the range 50-200 km/s, impact fusion becomes possible.

3.3.2 Program Logic

Pellet fueling efforts include pellet-injector development, pellet-ablation modeling, and pellet-injection experiments, as shown in Fig. 3.10. At present, pellet-injector development is centered on pneumatic and centrifugal injector concepts for use on the present generation of confinement experiments. These injectors have characteristic velocities in the 1-2 km/s range, which can perhaps be increased to 2-3 km/s. Present needs for tokamaks are judged to be in the range of 2-10 km/s; other concepts, which are thought to be more sensitive to profiles or to require deeper fueling,

FIGURE 3.10 Level 2 Logic Diagram for Fueling

may need velocities on the order of 50-100 km/s. The development of injectors for the higher range of velocities will likely be time-consuming and very costly. As a result, the program has adopted the strategy of aiming the development program toward the 5-10 km/s objective. While this is being done, plasma-physics experiments will be carried out on tokamaks and other configurations to determine the need for velocities in excess of 10 km/s. This evaluation will be made around 1990, after the experiments with pellets in the 1-2 km/s range are completed on TFTR and JET.

Development of advanced injection systems for higher velocities is beginning, as is work to support continuous pellet fueling and tritium operations. These development efforts should lead to advanced concepts for D_2 and T_2 systems, and perhaps to a single advanced system suitable for the long-burn demonstration. If injectors with $v > 10$ km/s are required, the development program will have to be significantly larger than the one presented in this discussion.

Pellet-ablation modeling efforts and the anticipated experimental studies will support technology development. Major milestones in the ablation modeling effort include (1) benchmarking ablation models, which include electron and fast-ion effects, with data obtained from pellet injection into a $T_e > 10$ keV, neutral-beam-heated plasma; and (2) evaluating fast-alpha effects on pellet ablation. These data could be obtained from the DT breakeven demonstration experiment (TFTR), from one of the long-pulse toroidal experiments (Doublet-III - D configuration [DIII-D], JET, Tore Supra), or from the short-pulse ignition experiment. In turn, the technology-development efforts and ablation studies will support both the short- and long-burn ignition experiments.

Major Facility Description

One new facility, a High-Velocity Injector-Development Facility (HVIDF), is required. The purpose of the HVIDF would be to test and evaluate new concepts for attaining high-velocity pellet injection (e.g., plasma gun, MHD railgun, ablation-driven accelerator). Facility features would include a generic accelerator frame, generic control and data-acquisition capabilities, cryogenic capabilities, and high-energy pulsed power system (100 kJ over 1-5 ms with a repetition rate of 5-10/s for several seconds). The construction cost is estimated at \$20 million over three years, beginning in 1988, with operating costs of \$2 million/yr for five years.

3.4 PLASMA-INTERACTIVE COMPONENTS

3.4.1 Technical Issues, Objectives, and Attributes

The key issues associated with PICs are listed in Table 3.11 and are discussed below. The objectives and attributes for these issues are indicated in Fig. 3.11 and Table 3.12.

TABLE 3.11 Issues Associated with Plasma-Interactive Components

Issue	Potential Impact ^a	Design Specificity
Particle Exhaust	F1	Closed Systems
Recycling	F3	Generic
Surface Conditioning, Cleaning	A1	Generic
Erosion, Redeposition	A1	Generic
Thermal Hydraulics	A1, A3	Generic
Thermomechanical Effects	A1, A3	Pulsed Systems
Disruptions	F3, A2	Tokamaks
Tritium Inventory and Permeation	F2, A3	Generic
Fabrication	A3	Generic

^aSymbols for potential impact are as follows:

Feasibility Issues

- F1 - May close design window.
- F2 - May result in unacceptable safety risk.
- F3 - May result in unacceptable reliability, availability, or lifetime.

Attractiveness Issues

- A1 - Reduced system performance.
- A2 - Reduced component lifetime.
- A3 - Increased system cost.
- A4 - Less desirable safety or environmental impact.

Particle Exhaust

The issue of particle exhaust deals with configuration dependence, plasma-parameter dependence, and the connection to discharge performance. The design of a particle-exhaust system is specific to the particular type of fusion device. It is easier, in principle, to provide particle-exhaust schemes for open configurations, where exhaust is "automatic," than for closed ones. For this reason, and also because closed systems (such as the tokamak) have been the first to encounter the need for active impurity-control systems, most systematic work has been done for the tokamak configuration.

Recycling

"Recycling" refers to the processes by which plasma ions leave the discharge, impinge on a material's surface, become neutralized, and reenter the plasma. Recycling

FIGURE 3.11 Objectives and Attributes for Plasma-Interactive Components

TABLE 3.12 Important Factors in Constructing Attribute Scales for Plasma-Interactive-Component Objectives

Secondary Objectives	Important Factors for Attributes
Maximize Particle Control	Effective Z Particle-removal efficiency He concentration in plasma
Maximize Lifetime	Erosion rate Stress Radiation-damage effects Number of thermal cycles Corrosion rates
Maximize Energy Removal and Utilization	Peak heat loads Critical heat flux Coolant outlet temperature Pumping power
Maximize Power Transmission	Power density Transmission efficiency
Maximize Design Integration	Degree of integration (constructed scale)
Maximize Fault Tolerance	Disruption erosion rate Disruption stresses Degree of damage during loss of coolant accident (LOCAs) and loss of flow (LOFs) (constructed scale)
Minimize Tritium Inventory and Permeation	Tritium-inventory level Permeation rate to coolant
Minimize Stored Energy	Stored energy
Minimize Activation and Waste	Activation level Mass of waste
Maximize Understanding of Surface-Related Phenomena	Data-base needs (%)
Maximize Understanding of Plasma Edge Phenomena	Data-base needs (%)
Maximize Understanding of Neutron-Damage-Related Phenomena	Data-base needs (%)

TABLE 3.12 (Cont'd)

Secondary Objectives	Important Factors for Attributes
Maximize Bulk-Materials Properties Data Base	Data-base needs (%)
Maximize Erosion/Redeposition Data Base Maximize Fabrication Data Base	Data-base needs (%) Data-base needs (%)

occurs at the vacuum-vessel wall or limiter front surface and can also occur in the throat of a divertor or pumped limiter. To the extent that the plasma ions are recycled at the wall or limiter, less fueling is required for operation. However, impurity ions may also be recycled, aggravating the contamination problem. Energetic ions may result in a recycling fraction greater than unity. This would make density control difficult, and it might also change the DT ratio in an uncontrollable way. Not much is known about controlling recycling at the walls. Past experiments have used titanium and chromium gettering to effect such control; however, these techniques are not applicable to either a near-term or a long-term burning fusion device, because the pumping effect saturates within a few seconds.

The important questions related to recycling are as follows:

1. What are the recycling coefficients of the materials of interest?
2. How does the recycling coefficient scale with surface conditions, plasma edge conditions, and integrated particle flux?
3. How is the recycling coefficient changed by radiation damage?

Surface Conditioning and Cleaning

Surface conditioning is necessary to minimize impurity influx from thermal and particle-induced outgassing and to stabilize the hydrogen-recycling properties of the surface. In addition, conditioning minimizes the tendency for unipolar arc formation. Conditioning is also required for neutral-beam and rf heating systems to minimize high-voltage breakdown. The conditioning of PICs involves careful pretreatment (cleaning and vacuum baking) of the material to ensure good vacuum qualities (i.e., low outgassing surfaces, and bulk material free of large voids, defects, and cracks that can serve as leak paths). In addition, in-situ conditioning, involving baking and extensive interaction with hydrogen plasmas or atomic hydrogen, is necessary to minimize plasma-wall interactions. Successful wall conditioning has been achieved when low-Z impurity introduction from first-wall and limiter surfaces is negligible, the hydrogen-recycling

properties are well characterized and static, and the tendency to unipolar arc formation is minimal.

Erosion and Redeposition

Energetic plasma edge particles or charge-exchange neutral particles will strike the surfaces of PICs and the wall and cause erosion by physical and chemical sputtering. These processes are a major source of impurity introduction into the plasma. This problem is addressed by passive (limiter materials selection) and active (gettering, use of magnetic divertor) impurity-control methods. Sputtered particles can enter the scrape-off region of the plasma, undergo recycling, and be redeposited on material surfaces. The redeposited material may have properties different from those of the base material. Because of the short pulses and low duty factors of current and near-term machines, erosion and redeposition are not important factors in the lifetime of near-term PICs. They are crucially important, however, for reactor applications.

Thermomechanical Effects

Tokamaks and stellarators are now expected to have peak (normal-operation) heat fluxes at limiters and divertors of 5 MW/m^2 , with lower values (1 MW/m^2) over large surface areas (such as the first wall). Mirrors, because of rapid axial particle and energy flows in the outer regions, have low heat fluxes (0.1 MW/m^2) on the first wall. Due to the potentially large-volume end cells, peak heat fluxes need not be high. In practice, components are not made larger than necessary, leading to similar 2- to 5-MW/m^2 heat fluxes on direct convertor or halo scraper designs. Reversed-field pinches and other alternative concepts may operate as higher-power-density devices. Consequently, larger fractions of their surface area must operate at high heat fluxes (e.g., first wall at 5 MW/m^2). In all cases, large areas ($1\text{-}1000 \text{ m}^2$) must be cooled.

Water is the preferred near-term coolant because of its excellent heat-transfer capabilities and established data base. Some of the major concerns in the area of thermal hydraulics are: heat-transfer limits (normal and augmented); flow distribution and stability; channel erosion; and heat-source profile. Heat-transfer limits arise from a transition to a less efficient heat-transfer mechanism (e.g., critical heat flux); limits on increasing flow velocity (e.g., pressure drop, choked flow, channel erosion, flow vibrations); or temperature limits (e.g., coolant boiling, thermal decomposition, solidification, energy-recovery efficiency, structural mechanical strength, and chemical compatibility).

Thermal stresses are caused when temperature changes induce thermal expansion in a constrained material. These stresses can be very large in PICs, which are exposed to high surface heat fluxes and large temperature gradients. Thermal fatigue becomes a concern when the heat flux is cycled, as is the case for current and near-term pulsed tokamak fusion devices. Damage caused by thermal fatigue occurs in two stages: crack initiation and crack growth. For the next generation of fusion devices, it will be necessary to bond armor tiles to the heat-sink structure of actively (or, in near-term experiments, inertially) cooled PICs. Two examples are graphite bonded to molybdenum

cooling tubes and beryllium tiles bonded to a copper heat sink. The long-term structural integrity of the bond is a critical issue for PICs, because debonding would cause rapid overheating of the armor material and eventual failure of the system.

Disruptions

Disruptions can be described as rapid reductions in the plasma current, accompanied by localized deposition of much of the plasma energy on interior surfaces. Disruptions are observed in all tokamaks; the phenomenon is poorly understood. Consequently, elimination of disruptions for future near-term machines cannot be assured, and PICs must be designed to withstand the induced forces and high heat fluxes that accompany these events.

From the point of view of damage to in-vessel components, disruptions can be divided into two phases: the thermal-quench phase, which produces localized high heat fluxes, and the current-decay phase, which induces large electromagnetic forces on the vacuum vessel and on in-vessel components. Heat fluxes as high as 10 kW/m^2 can cause surface vaporization and melt-layer formation. The interaction between the vaporized surface material and the incident plasma is not understood. Considerable effort should be expended on testing and modeling in this area, because the outgoing vapor cloud has the potential to shield the surface and prevent more extensive damage. The difficulty in analyzing the stability of the surface melt layer is compounded by the induced eddy-current forces (generated during the current-decay phase) that act on it. This, too, is an area requiring further analysis and testing. The relative time dependence of the two phases of the disruption is not known.

Tritium Inventory and Permeation

The repeated exchange of hydrogen isotopes between the plasma and surrounding walls is a dominant factor in plasma refueling in present-day devices, and it will control the critically important issues of tritium inventory and permeation in DT machines. Tritium inventory is a near-term safety and environmental problem that must be addressed for the tritium operation of TFTR and JET and for the design of next-generation experiments. A considerable data base has been generated for hydrogen isotope interaction with structural first-wall materials, such as austenitic stainless steels and Inconel alloys. However, little is known about the interaction of low-Z limiter materials, such as graphite and beryllium, with tritium. The devices TFTR and JET use graphite exclusively for PICs; the Compact Ignition Tokamak (CIT) design may also use graphite as the baseline material. However, graphite is a porous material with an affinity for hydrogen.

Much controversy exists in the literature over fundamental tritium properties, such as the surface and transgranular diffusivity, bulk solubility, and trap concentration and binding energies. A detailed, reliable model of tritium uptake and retention by graphite in a fusion-reactor environment is not currently available. The data base for and understanding of tritium interaction with other candidate low-atomic-number materials (beryllium, TiC, SiC, etc.) are as bad as or worse than the case with graphite.

Fabrication

Fabrication of PICs involves the bonding of a plasma-side material to a structural heat sink. Both the plasma side and heat sink may consist of special materials for which there is little or no fabrication data base. Fabrication of the base materials and of the assembled PICs needs to be demonstrated. Of particular interest is the interface bond; in many cases, a considerable mismatch in the physical properties of the materials makes attachment especially difficult. The successful fabrication of materials and bondings over lengths approaching 1 m presents a significant challenge for fusion.

3.4.2 Program Logic

Logic Diagrams and Decision Points

The development of PICs and related research is outlined in Fig. 3.12 in terms of three principal activities -- particle and impurity control, heat removal and recovery, and transient events. Interactions also take place within the areas of confinement systems, nuclear technology, and materials.

The development of PICs requires testing in a plasma environment; hence, there are strong ties to confinement activities. Engineering data and basic information in individual areas are provided from laboratory studies, tests in off-line engineering facilities, and existing confinement devices. This information is combined to form the basis for design and construction of PICs in the main-line confinement devices. A set of evaluation points at about 1990 will provide input for the construction of PICs for the short-pulse ignition device. Operating experience from the ignition device and information from additional laboratory studies and tests in off-line facilities are to be utilized in the evaluation and design of PICs for the long-burn demonstration device. Additional information concerning performance-related issues, as well as the interactions with nuclear components, is supplied from materials and nuclear technology. Materials information includes baseline property data and irradiation-performance data. This leads to another set of evaluation points at about 1995 to assess PICs for the long-burn device. Information on the capability of nonwater coolants to remove the high surface heat loads from PICs is supplied to nuclear technology prior to 1995 for the assessment of liquid-metal blankets.

The device that will demonstrate long-burn operation is a fully integrated test facility to evaluate the performance potential of PICs for reactor applications. Tests in this device around 2000-2005 will provide the major input to the final assessment of PICs. A demonstration of useful heat removal from reactor-compatible components is also planned in the long-burn device.

Level 3 diagrams for particle and impurity control, heat removal and recovery, and transient events are shown in Figs. 3.13-3.15, respectively. As shown in Fig. 3.13, the primary areas of particle and impurity control are erosion and redeposition, particle and impurity exhaust, recycling, and surface conditioning. The focus of the development work is the interaction of the plasma edge and scrape-off region with materials

FIGURE 3.12 Level 2 Logic Diagram for Plasma-Interactive Components (Decision points <a> and are described in Tables 3.13 and 3.14.)

FIGURE 3.13 Level 3 Logic Diagram for Particle and Impurity Control

FIGURE 3.14 Level 3 Logic Diagram for Heat Removal

FIGURE 3.15 Level 3 Logic Diagram for Transient Events

surfaces. Information is provided from laboratory studies, tests in off-line facilities, and tests in confinement devices. Data from the program elements for tritium permeation and control, plasma transport, burn control and ash removal, fueling, and special materials also provide key inputs to this activity.

The subelements of heat removal and recovery are thermal hydraulics and thermomechanical effects. This area deals with providing adequate heat removal for PICs and demonstrating structural integrity for the heat-flux levels expected in fusion devices. In the long term, the useful heat recovery must also be demonstrated for reactor applications. Most testing is expected to be conducted in off-line facilities to acquire the necessary engineering data for the design of PICs.

The subelements of transient events are surface ablation, electromagnetic forces, and characterization and control. There are close interactions with materials and confinement. Included here are the demands on structures and materials posed by startup and shutdown conditions, runaway electrons, disruptions, and similar phenomena. Development will take place on two major fronts: The first is to develop structures and materials that can best withstand transient events, and the second is to develop methods to minimize the occurrence and severity of transient events. Progress in the characterization and control of these events will greatly influence the development of the research program. Near-term testing of transient events is difficult because, in many cases, the events have not been well characterized.

The most important decisions for plasma-interactive components involve the selection of components for the short-pulse ignition experiment and the long-burn demonstration experiment. These decisions are described in Tables 3.13 and 3.14, respectively. In both cases, the performance of PICs will have a direct bearing on the overall level of performance of these large facilities.

Facility Descriptions

Much of the data needed for the development of PICs will be obtained from tests in various plasma-confinement devices, such as CIT or ETR; these devices are described in the Synopsis. Other test needs can be supplied by existing devices, such as the Fusion Electromagnetic Induction Experiment (FELIX) (for studying disruption forces) and the Plasma Materials Test Facility (PMTF) (for studying effects of high heat loads). Laboratory equipment can also be used to investigate basic phenomena, such as sputtering and redeposition. It is desirable, however, to study such issues as erosion and redeposition in a facility dedicated to plasma/materials interactions. Such a facility would closely simulate the plasma scrape-off conditions expected in fusion reactors.

The purpose of the off-line plasma-simulation facility would be to simulate plasma-edge/scrape-off conditions (e.g. plasma density, particle energies, heat loads, magnetic fields, and recycling) in order to examine critical surface-related phenomena,

TABLE 3.13 Decision <a> on Plasma-Interactive-Component Selection for the Short-Pulse Ignition Device

Statement of Decision

Make final selection of and supply plasma-interactive components for the short-pulse ignition experiment.

Decision Criteria

Operating goals and conditions for ignition experiment have been identified.
 Operation of PICs has been demonstrated in other tokamak experiments and separate test facilities.
 Predictive capability for plasma/materials interactions and thermomechanical behavior has been established.
 Fabrication of materials has been demonstrated.
 Materials data base for PIC materials has been established.

Sources of Information

Information on operating conditions, as well as performance of present PICs, is supplied by confinement experiments.
 Performance limits and engineering data are supplied by plasma-technology program.
 Materials experiments provide materials data.

Outcomes and Consequences of Decision

Development of high-performance PICs will make it possible to operate the ignition experiment at high heat-flux levels and possibly for extended burn periods.
 If only reduced-performance PICs are available, then operating goals for the ignition experiment may have to be scaled downward.

such as redeposition. The facility would be used to test samples ($0.1\text{--}0.2\text{ m}^2$) in a plasma simulating the following scrape-off conditions:

Particle energies (ions and electrons)	10-200 eV
Sheath potential	2-3 T_e
Magnetic field	4 T
Peak heat flux perpendicular to field lines	10 MW/m ²
rf power input to plasma	1 MW

In addition, the facility would be highly instrumented and would operate in a steady state mode. Facility costs are estimated to be about \$7 million, with about a 3-yr construction period. Operating costs would be about \$1 million/yr.

TABLE 3.14 Decision on Plasma-Interactive-Component Selection for the Long-Burn Demonstration Experiment

Statement of Decision

Make final selection of and supply plasma-interactive components for the long-burn experiment.

Decision Criteria

Operating goals and conditions for the long-burn experiment have been identified.

Operation of PICs has been demonstrated in other tokamak experiments, including the short-pulse ignition device, and in other test facilities. Adequate impurity-removal capability has been established for impurity-control systems.

Predictive capability for plasma/materials interactions and thermomechanical behavior has been established.

Fabrication of prototypical components has been demonstrated.

Materials data base, particularly the effects of using fusion neutrons to moderate (10-30 dpa) fluences, has been established.

Sources of Information

Information on operating conditions, as well as performance of present PICs, is supplied by confinement and burning-plasma experiments.

Performance limits and engineering data are supplied by plasma-technology program.

Lifetime limits and materials data are supplied by materials program.

Information on design integration with other nuclear components and the rest of the reactor is supplied by nuclear technology and systems studies.

Outcomes and Consequences of Decision

Development of high-performance, reactor-relevant PICs will make it possible to run the long-burn experiment at high heat-flux levels and for extended periods of time.

If only reduced-performance PICs are available, then the operating goals for the long-burn experiment may have to be scaled downward.

REFERENCE

1. J. Sheffield et al., *Cost Assessment of a Generic Magnetic Fusion Reactor*, Oak Ridge National Laboratory Report, ORNL/TM-9311, March 1986.

OVERVIEW

Nuclear technology is an essential element for the evolution of fusion toward energy and other applications. Fusion nuclear technology (FNT) encompasses those components concerned with fuel production and processing, energy extraction and use, and radiation protection of components and personnel. The performance characteristics of these nuclear components will be a vital factor in determining the economics, safety, and environmental impact -- and hence, the ultimate attractiveness -- of fusion.

The fusion environment experienced by nuclear components involves the simultaneous presence of energetic plasma particles, 14-MeV neutrons, magnetic fields, surface and bulk heating, tritium, and vacuum. Furthermore, many fusion nuclear components are required to perform multiple functions (e.g., simultaneous tritium production/extraction and energy conversion/extraction in the blanket, heat removal and plasma-ash removal in the in-vessel components). The unique fusion environment and the multiple functions of the nuclear components produce new phenomena or substantially modify the characteristics of presently identified phenomena. These phenomena are associated with critical feasibility and attractiveness issues for fusion. Resolving these issues requires knowledge that can be acquired only through additional experiments, theories, and models. This knowledge is critical to fusion and is important to science and technology outside fusion. Identifying an optimum R&D strategy to acquire this knowledge, leading to successful and timely development of nuclear components, is a vital task in the area of FNT research.

The program elements included within the FNT area are blanket/first wall, radiation-shield, tritium-processing, and nuclear elements of plasma-interactive components. The scope of FNT is summarized in Table 4.1. The results of the TPA effort indicate that the issues, objectives, and performance associated with these program elements are strongly interrelated. However, the blanket/first-wall program element poses the most complex issues and development problems of all the nuclear components. The TPA results also indicate a strong coupling between the nuclear-technology area and the other planning areas -- confinement systems, burning plasma, plasma technology, materials, and systems design and analysis. The interrelations among program elements and planning areas have been considered in the TPA efforts.

The TPA has addressed the technical issues, objectives, test logic, and facilities for nuclear technology. The planning effort was carried out by senior experts from the fusion community. The TPA has attempted to maximize the use of results from previous studies. In particular, results from FINESSE on issues, experiments, facilities, and costs have provided an important basis for the TPA effort on FNT (Refs. 1 and 2).

TABLE 4.1 Scope of Fusion Nuclear Technology

Functions

Fuel Production and Processing
 Energy Conversion and Utilization
 Radiation Protection of Personnel and Components

Subsystems

Blanket/First Wall
 Radiation Shield
 Tritium Processing Systems
 Nuclear Elements of Plasma-Interactive Components

Experiments and Facilities

Basic Tests (nonfusion facilities)
 Property Data
 Multiple-Effect Tests (nonfusion facilities)
 Exploration of Phenomena
 Integrated Tests (fusion facilities)
 Verification of Concepts
 Component Development (fusion systems)
 Growth in Reliability

Theory, Analytical and Computer Modeling

Technical Issues

The technical issues in all four FNT program elements are related to the following:

- Fuel Self-Sufficiency
- Energy Conversion and Use
- Radiation Protection of Personnel and Components

The technical issues were examined by the TPA and classified according to design specificity; potential impact on feasibility, safety, and environment; and overall level of concern for fusion development. These issues are discussed in Secs. 4.1-4.4; further information is given in Refs. 3 and 4. The major issues for the four program elements are listed in Table 4.2 and are summarized below.

TABLE 4.2 Major Technical Issues for Fusion Nuclear Technology

Blanket/First Wall

Tritium and/or Fissile-Fuel Breeding
 Solid-Breeder and Neutron-Multiplier Tritium Issues^a
 Tritium Permeation and Recovery^a
 Compatibility of Coolant, Structure, and Breeder^a
 Liquid-Metal MHD Flow Effects
 Structural Response and Component Lifetime^a
 Reliability, Failure Modes, and Accident Consequences

Radiation Shield

Specification of Radiation Protection Requirements
 Verification of Shield Effectiveness

Tritium Processing

Tritium Monitoring and Accountability
 Impurity Removal from DT Fuels
 Tritium Removal from Water-Coolant Loops and Room Atmosphere
 Tritium-Processing System Integration
 Cryopump and Valve Lifetime

Nuclear Aspects of Plasma-Interactive Components

Particle Pumping and Impurity Control
 Surface Erosion and Redeposition
 Thermomechanical and Thermalhydraulic Performance
 Irradiation Effects on Component Lifetime
 Tritium Inventory and Permeation
 Fabrication Techniques
 Development of Alternate Coolants

^aMaterials research activities for these issues are also discussed in Chapter 5.

Blanket/First Wall. The nuclear system most commonly identified with FNT development in the MFPP and elsewhere is the blanket/first wall system. During the past several years, a wide variety of potentially attractive fusion-electric, fusion breeder (hybrid) and fusion synfuel-production blanket/first wall concepts have been proposed. However, there has been very little experimental work in this area to date. Each of the proposed blanket concepts is associated with one or more feasibility or attractiveness issues. An important element in the blanket/first wall development strategy will be to address these feasibility and attractiveness issues so that a selection among the various candidate technologies can be made.

The blanket/first wall technologies can be divided into two classes: blankets that would utilize a solid breeder (e.g., Li_2O or LiAlO_2) and blankets that would utilize a liquid breeder [e.g., lithium, lead-lithium, and molten salts such as LiF-BeF_2 (Flibe)]. The solid breeder blankets would be cooled using helium or water, while the liquid breeder blankets would be "self-cooled" or cooled using a separate fluid, such as helium. Ferritic steels and vanadium alloys are typically considered as structural materials.

In nearly all U.S. designs, the first wall and blanket would form one integrated structure with one coolant. However, separate first wall systems may be required for high-power-density configurations ($\geq 5 \text{ MW/m}^2$). The Europeans and Soviets have considered separate first wall systems for moderate-power-density applications ($3\text{--}5 \text{ MW/m}^2$) as well.

The issues for blanket/first wall systems can be categorized in terms of nine major groups:

- Tritium and/or fissile fuel breeding issues
- Solid-breeder and neutron multiplier tritium issues
- Tritium permeation and recovery
- Compatibility of coolant/structure/breeder materials
- Liquid-metal MHD flow effects
- Structural response and component lifetime (including irradiation)
- Reliability, failure mode, and accident consequence issues
- Fusion-breeder fuel-cycle issues (e.g., reprocessing cost)

The materials aspects of some of these issues are discussed in Chapter 5. Several of the above issues (e.g., liquid-metal MHD effects) are not generic to all blankets, but the issues implied by the alternative design solutions (e.g., solid-breeder tritium inventory, instead of liquid-metal MHD) are equally serious.

Radiation Shielding. Radiation shielding will be required for early DT burning experiments, as well as for later facilities. In the short-pulse experiment, the emphasis will be on limiting the levels of direct radiation and indirect radiation (i.e., via activation) to plant personnel, diagnostic equipment, and especially sensitive components (such as cryopanel and copper magnets). For the long-burn demonstration device, requirements related to limiting the radiation damage levels to all components will become more important.

Radiation shielding presents no feasibility problems, but uncertainties in the requirements for and effectiveness of shielding can lead to inefficient use, increased

cost, and/or reduced lifetime for the shielded components. Two shielding issues have been identified:

- Specification of radiation protection requirements (e.g., allowable dose for magnets)
- Verification of shield effectiveness

The first of these issues is closely coupled to fusion materials and component development, because advanced materials will allow higher doses. The second issue is critical to small-size devices and has particular relevance to near-term shields.

Tritium Processing Issues. Tritium processing is a highly developed technology in comparison with the first wall/blanket or other fusion nuclear subsystems. Nevertheless, the immediate need for highly reliable systems (e.g., an ignition experiment) can elevate the priority of tritium-system-related issues. These issues include the following:

- Tritium monitoring and accountability
- Impurity removal from DT fuels
- Tritium removal from water coolant loops and room atmosphere
- Tritium processing system integration (including safety and reliability)
- Cryopump and valve lifetime

These issues are generic to all fusion applications, but the required levels of tritium throughput vary greatly between near-term ignition systems (< 10-s pulse length) and follow-on facilities (i.e., an integrated test facility with ~ 500-s pulse length, nearly continuous operation, and much higher fusion power).

Nuclear Issues for In-Vessel Plasma-Interactive Components. The nuclear issues for the in-vessel PICs, such as pumped limiters, magnetic divertors, heating systems, and the plasma side of the first wall, emphasize the reliability, safety, and lifetime of those components. These issues supplement the near-term performance-related PIC issues discussed in Chapter 3. Resolution of the PIC issues can be highly dependent on the magnetic confinement scheme, because each confinement option provides different geometries, heat loadings, particle loadings, and plasma edge characteristics. Nevertheless, several common issues can be identified:

- Particle pumping and impurity control
- Surface erosion and redeposition (see also Chapter 3)

- Thermomechanical and thermal hydraulic performance
- Irradiation effects on component lifetime (see also Chapter 5)
- Tritium inventory and permeation to the coolant
- Fabrication techniques (see also Chapter 5)
- Development of alternative coolants (non-H₂O)

Among these common issues, the erosion/redeposition rates can drive the other issues (by determining the required surface thickness) and are also the least certain. Pumped-limiter/magnetic-divertor gross erosion rates are estimated to be up to tenths of meters per year at modest power densities ($\sim 5 \text{ MW/m}^2$). Also, if liquid-metal-cooled (e.g., lithium-cooled) blankets are to be retained as an option, nonwater-cooled PIC components must be developed.

Objectives and Attributes

The overall objective of the FNT effort is to develop technology that leads to commercial fusion applications with attractive economic, safety, and environmental features. Achieving this objective will require that the following "functional" tasks be accomplished for the nuclear components:

- Demonstrate fuel self-sufficiency.
- Demonstrate efficient, reliable, economical, and safe energy conversion and use.

An important aspect of the FNT R&D objective is to provide a predictive capability that can be used to assess the performance of fusion nuclear technologies. Specific objectives for the four program elements are given in Fig. 4.1. Lower-level objectives, together with attributes to be used in measuring progress toward achieving objectives and in prioritizing R&D tasks, were developed in Ref. 3. Most of the attributes are based on constructed scales that involve parameters related to performance, economics, safety, environmental impact, and predictive capability. An example of these parameters is given for the blanket in Table 4.3.

Program Logic

A program strategy has been developed to achieve the MFPP goal and the nuclear technology objectives discussed above. The R&D program consists of experimental and modeling efforts; the test plan for the next 20 years is based on four 5-yr phases. The characteristics of R&D tasks for the nuclear technology program elements are shown for each of the four phases in Table 4.4. The initial phase focuses on conducting scoping experiments to explore basic properties and separate-effect

*Also shown under Plasma Technology in Fig. 3.11.

FIGURE 4.1 Hierarchy of Objectives for Nuclear Technology

TABLE 4.3 Key Parameters in Constructed Scale for Blanket Attributes

Economics/Performance	Safety/Environment	Prediction/Understanding
Neutron Wall Load	Chemical Reactivity	Magnetohydrodynamics (MHD)
Surface Heat Flux	Response to Loss of Coolant	fluid velocity profile
Tritium Breeding		pressure drop
Thermal Efficiency	Vulnerable Tritium Inventory	heat transfer
Energy Multiplication	Long-Term Activation	corrosion
Blanket Thickness	Afterheat	Tritium Inventory
Reliability	Routine Radioactivity Release	solubility
Lifetime	Others	diffusion
Sector MTBF/MTTR ^a		adsorption
Blanket/Transport-Loop Cost		trapping
		permeation
		Materials Interactions
		breeder/structure
		coolant/structure
		purge/breeder

^aMTBF = mean time between failures; MTTR = mean time to repair.

phenomena. The emphasis in subsequent phases shifts gradually to interactive-effect tests with an increased level of integration. Verification of concepts in nonfusion facilities would provide the data base and experience necessary to perform integrated tests in fusion facilities by about the year 2000. An important feature of the plan is the systematic screening of materials, configurations, and design concepts in order to reduce costs associated with the more expensive integrated tests. The plan also provides opportunities for innovation and concept improvement.

The Level 1 logic diagram for nuclear technology (Fig. 4.2) indicates the strong interrelations between the nuclear technology and materials areas, as well as the interrelations among nuclear technology and confinement systems, burning plasma physics, and systems design and analysis. The major evaluation points (indicated by diamonds) and milestones (indicated by squares) for the four program elements are described in Table 4.5.

Some decision points are concerned with selection of design options for further R&D effort, while others are concerned with identifying the optimal approach for obtaining desired information, particularly as it pertains to major facilities. Fusion nuclear technology R&D requires testing in nonfusion facilities and fusion devices.

TABLE 4.4 Characterization of Nuclear Technology Tasks

Component	Late 1980s	Early 1990s	Late 1990s	Early 2000s
Blanket/First Wall	Perform separate-effect tests and obtain scoping data.	Perform multiple-interaction experiments to explore and characterize phenomena.	Perform integrated tests in nonfusion facilities for concept verification.	Operate blanket experimental modules in fusion test facility.
Tritium	Perform permeation and plasma exhaust-processing experiments. Test cryo-pump module.	Demonstrate plasma exhaust-processing technology. Test tritium-extraction techniques on laboratory scale.	Demonstrate tritium extraction. Operate vacuum test stand.	Operate tritium systems in fusion test facility.
Shield	Perform point-source shield tests. Obtain data on component radiation-protection criteria.	Design and test shields in tritium-burning devices, effectiveness, and predictive capability.	Verify shield.	Verify shield performance in fusion test facility.
Plasma-Interactive Components	Perform separate-effect tests. Develop predictive capability for plasma edge and recycling.	Demonstrate energy-removal techniques for PIC systems.	Design and test PIC systems for long-pulse device. Verify predictive capability.	Operate PIC system in fusion test facility.

FIGURE 4.2 Level 1 Logic Diagram for Nuclear Technology (Decision points <a-j> and milestones are described in Table 4.5.)

TABLE 4.5 Evaluation Points and Milestones for Nuclear Technology^a

a	Select leading materials combinations for liquid breeders.
b	Select leading materials combinations for solid breeders.
c	Compare results for solid and liquid breeders. Select one breeder if possible. Select leading configurations.
d	Select primary designs for experimental blanket test modules in fusion test.
e	Select tritium-extraction and -control methods.
f	Select method for fuel clean-up.
g	Assess feasibility of nonwater cooling of PIC.
h	Assess heat-removal techniques for PIC.
i	Evaluate system impact on selection of blanket concepts.
j	Evaluate system impact on selection of blanket test modules.
M ₁	Verify blanket concepts in nonfusion facilities.
M ₂	Demonstrate tritium system operation.
M ₃	Achieve moderate fluence [30 displacements per atom (dpa)] for candidate alloys in nonfusion facilities.
M ₄	Achieve moderate fluence (30 dpa) for reference alloys and high fluence (100 dpa) for candidate alloys in nonfusion facilities.
M ₅	Achieve high fluence (100 dpa) for reference alloys in nonfusion facilities.
M ₆	Verify data and methods for shield design.
M ₇	Verify shield-design effectiveness. Obtain data on component radiation protection criteria.

^aSee Fig. 4.2.

Selection of nonfusion facilities is influenced most by the specific design options selected. In contrast, decisions on fusion facilities are influenced most by progress on the confinement systems and plasma-burn issues. A top-level decision point identified in the Synopsis is the selection of the approach for fusion testing of nuclear technology (E_4). This decision may result in identification of one fusion facility to demonstrate plasma long-burn and to perform FNT testing. Another possible outcome is identification of separate fusion facilities for long-burn demonstration and nuclear testing. Table S.7 in the Synopsis indicates the criteria, sources of information, and consequences of this decision.

Most of the decision points in Fig. 4.2 involve selection of technology options and nonfusion facilities. For the blanket, both solid- and liquid-breeder options will be pursued in the early phases of R&D, and the primary candidate materials for these options will be selected around 1990, <a> and . The two classes of blanket options will be compared around 1995, <c>. This evaluation will be based primarily on results from multiple-interaction tests for the blanket, but it will also require important input from other areas, such as (1) experimental information on the feasibility of nonwater cooling of plasma interactive components, <g>; (2) moderate-fluence data on structural materials and electrical insulators for the blanket region; (3) data on tritium extraction techniques and permeation, <e>; (4) efficiency of tritium clean-up and impact on required tritium breeding ratio, <f>; (5) key parameters for promising reactor concepts, as they evolve from confinement systems and burning-plasma experiments; and (6) impact of system design on component attractiveness and vice versa, <i>. Selection of only one of the two classes (liquid and solid breeders) of blanket options may not be possible prior to fusion testing. Narrowing of materials combinations and configurations for each class of blankets is an important aspect of decision point <c>.

The outcome of the comparison of blanket concepts at <c> affects not only the subsequent blanket experiments, but also the selection of options and experiments on tritium processing, materials, and PICs. The integrated tests in nonfusion facilities beyond 1995 are aimed at concept verification [M_1], to be completed by about the year 2000.

Experiments on fuel cleanup at the Tritium System Test Assembly (TSTA) and on tritium extraction and control in existing facilities will lead to demonstration of the tritium systems [M_2] in about 1995. Radiation shielding will play a key role in near-term, as well as future, tritium-burning devices. Measurements in such devices will contribute to verification of data and methods [M_6] and of shield-design effectiveness [M_7].

The data base obtained by about the year 2005 should be sufficient to perform (1) quantitative assessment of the potential of fusion energy for commercial application and (2) fusion testing (test modules for the blanket and full components for the shield and tritium system). Fusion testing prior to the year 2000 would be possible only if the pace of the effort were accelerated.

Facility Descriptions

The facilities for FNT include nonfusion facilities and fusion devices. The nonfusion facilities can be classified as fission reactors, point neutron sources, and nonneutron test stands. The nuclear technology requirements for fission reactors and point neutron sources can be satisfied by existing facilities in the U.S. and other countries. Important nonneutron facilities exist and will continue to be utilized; however, a number of new nonneutron test facilities need to be constructed. Table 4.6 summarizes the nonfusion facilities.

Testing of nuclear components in the fusion environment is critical for concept verification in the early phases and for component development and reliability growth in the later phases. The nuclear technology testing requirements of the fusion facility are indicated in Table S.14 in the Synopsis.

4.1 BLANKET/FIRST WALL PROGRAM ELEMENTS

The blanket/first wall program element includes some of the most complex issues and development problems of all the nuclear components. This section describes the technical issues, objectives, test logic, facilities, and resource requirements for the blanket/first wall. The material presented here makes extensive use of results obtained in the FINESSE studies (Refs. 1 and 2).

TABLE 4.6 Nonfusion Facilities for Nuclear Technology

Fission Reactors (existing)

Solid-Breeder Tritium-Recovery Experiments
Solid-Breeder Nuclear Submodule
Test Elements for Various Experiments

Point Neutron Sources (existing and upgrade)

Blanket Neutronics: Tritium Breeding, Heating, Activation
Shielding: Integral Experiments for Bulk and Penetration Shields

Nonneutron Facilities

Liquid-Metal Flow Facility (momentum and heat transfer)
Liquid-Metal Corrosion/Mass-Transfer Loops
Liquid-Metal MHD Mass Transfer
Liquid-Metal Thermomechanics Integration
Solid-Breeder Thermomechanics
High-Heat-Flux Thermomechanical Test Facility for Nonwater Coolants
Tritium-Processing, Tritium-Permeation, and Blanket Tritium-Extraction Facilities

4.1.1 Technical Issues, Objectives, and Attributes

Technical Issues

A diverse set of blanket/first wall designs currently exists, including solid-breeder and liquid-breeder designs for fusion electric applications and similar classes of designs for hybrid (fission/fusion) applications. Important uncertainties concerning these designs could restrict their ability to operate reliably under the range of conditions anticipated. Some of the technical issues -- particularly those involving the behavior of the structural elements of the designs -- overlap among different categories. In general, however, each class of blankets involves a number of distinct uncertainties regarding their behavior under fusion conditions. The issues, summarized in Table 4.7, are briefly reviewed here. More detailed information appears in Refs. 1-3.

Issues common to most blankets include irradiation effects on material properties, structural response in the fusion environment, and failure modes. The mechanical response of liquid-metal blankets is complicated by the complex loading conditions (MHD pressure stresses, thermal stresses, etc.). In all cases, radiation swelling and creep, changes in ductility, helium embrittlement, and other irradiation effects are important in the primary candidate structural alloys. These currently include austenitic and ferritic steels and refractory alloys. Some modes of failure have been identified, but the determination of failure rates and probabilities (which must be examined in order to quantify and improve blanket lifetime and safety) will require integrated testing.

Most of the blanket issues are specific to the particular class of blanket design. The most critical issues often change, even among the individual designs within a given class. In the following, the issues are discussed generically for each class of blanket.

Solid-Breeder Issues. The major issues specific to solid breeders involve tritium self-sufficiency; breeder/multiplier tritium inventory and recovery; mechanical interactions among breeder, multiplier, and structure; breeder corrosion and mass transfer; and tritium permeation and processing from the blanket.

To achieve tritium self-sufficiency, the tritium production rate must exceed the consumption rate. However, uncertainties exist in determining both the achievable tritium breeding rate (e.g., neutronics calculation accuracy) and the required tritium breeding rate (e.g., blanket tritium holdup), as well as in some issues beyond the scope of the blanket system (e.g., tritium inventory in the fuel system).

Resolving the issue of breeder/multiplier tritium inventory and recovery requires that the retention and transport of tritium within the breeder and multiplier be understood. Uncertainties range from basic properties, such as tritium intragranular diffusivity in nonirradiated breeder material, to the effects of the complex physical environment of the breeder.

TABLE 4.7 Technical Issues for Blanket/First Wall Components

Issue	Potential Impact ^a	Design Specificity
Tritium and/or Fissile-Fuel Breeding Tritium breeding Fissile-fuel breeding	A1, A3	Specific issues for all designs
Solid-Breeder and Neutron Multiplier Tritium Issues Tritium inventory and transport Solid-breeder form and mechanical interactions	F2, A4	Solid-breeder blankets
Tritium Permeation and Recovery Permeation and recovery in helium systems Tritium control and recovery in liquid-breeder systems	F1, A4	Specific issues for all designs
Compatibility of Coolant, Structure, and Breeder Helium/vanadium compatibility Solid-breeder compatibility and mass transfer Liquid-breeder compatibility and mass transfer Pebble flow for fusion breeder	F3, A1	Specific issues for all designs
Liquid-Metal (LM) MHD Flow Effects MHD pressure drop, heat transfer, and fluid flow Electrical insulators for flowing LM systems	F1, A1	Liquid-metal blankets
Structural Response and Component Lifetime Structural response and lifetime Development of ceramics technologies	F3, A1	Specific issues for all designs
Reliability, Failure Modes, and Accident Consequences Failure modes and reliability Radioactive-release pathways and consequences	F2, A4	Specific issues for all designs
Fusion Breeder Fuel-Reprocessing Costs	A1, A3	Fusion breeder hybrid

^aImpact codes are as follows:

- F1 - may close design window.
- F2 - may result in unacceptable safety risk.
- F3 - may result in unacceptable reliability, availability, or lifetime.
- A1 - may reduce system performance.
- A2 - may reduce component lifetime.
- A3 - may increase system cost.
- A4 - may have less desirable safety or environmental implication.

Mechanical interactions among the breeder, multiplier, and structure occur at the interface between the breeder and multiplier and the structural cladding. This can be an area where stress and heat-transfer limits occur. Uncertainties include the mutual loads and responses imposed by the elements on each other (e.g., swelling of cladding and breeder or multiplier) and their influence on the thermal behavior of the breeder (e.g., gap conductance).

Materials interactions lead to uncertainties in such areas as breeder mass transfer (within the internal porosity or out of the blanket entirely) and chemical reactions at the structure/breeder interface. These uncertainties affect both the configuration and operating-temperature limits of the blanket. In general, the issue of material compatibility is less serious in solid-breeder blankets than in liquid-breeder blankets.

The permeation of tritium outside of the breeder zone is important in defining the coolant detritiation requirements, but the nature and effects of the environment and surface conditions are uncertain. Uncertainties in the recovery of tritium from the primary breeder extraction stream include the tritium form and the inventory and efficiency of the recovery process.

Liquid-Breeder Issues. The major liquid-breeder-specific issues involve MHD effects, materials interactions, tritium control, and tritium self-sufficiency. The presence of a magnetic field adds further uncertainty to many aspects of blanket operation. Because of the dominant influence of the magnetic field on the fluid flow characteristics, both heat transfer and mass transfer (and, consequently, thermal stresses, pressure stresses, and failure modes) are strongly affected. The critical MHD uncertainties involve pressure drop, fluid flow, and heat transfer in complex geometries.

Material compatibility is a special problem for liquid-breeder blankets, both because data are generally lacking and because the design window can be strongly affected. Experimental data on lithium, ^{17}Li -83Pb (LiPb), and Flibe are not adequate to predict acceptable operating limits for the blanket system. The dissolution of steel at relatively low temperature (450-550°C) motivates the exploration of advanced high-temperature alloys, such as vanadium alloy, as well as innovative techniques for impurity control.

Materials interactions are a serious safety concern for liquid-breeder blankets. The potential for chemical reactions between either Li or LiPb and water, as well as for lithium/air fires, can significantly alter design approaches or (in some cases) rule out potential materials combinations.

The difficulty of tritium inventory control in liquid-breeder systems strongly depends on the choice of breeder. The principal problems involve efficient extraction from the breeder and adequate permeation control. Tritium control in lithium-cooled blankets is not considered a feasibility issue. However, for LiPb and Flibe, the very low tritium solubility leads to very high required extraction efficiency in order to limit loss rates through the heat exchanger. For Flibe, tritium permeation may be reduced by chemical means (such as increasing the fluorine concentration), but chemical means to reduce tritium permeation often lead to an unacceptable increase in corrosion. The effectiveness of tritium permeation barriers, in the form of either coatings or double-walled piping, are both important issues for all coolant/structural-material systems.

Tritium self-sufficiency is also an issue for liquid-breeder blankets but is not generally as critical as for solid breeders. The margin for tritium breeding is fairly high in self-cooled liquid-metal blankets, but it could be reduced in helium-cooled or Flibe designs, becoming critical if partial coverage were mandated by blanket-cooling constraints.

Hybrid-Blanket Issues. Hybrid blankets involve many of the same uncertainties as fusion electric blankets. The tritium-related issues for the helium-cooled molten-salt and liquid-lithium-cooled hybrid blanket designs are similar to those for the

corresponding liquid-breeder blanket. The issue of material compatibility for the molten-salt hybrid blanket is also similar to that for the molten-salt liquid-breeder blanket. Beryllium-related issues, such as creep and swelling, are similar to those for other beryllium-multiplied blankets. However, the flowing pebble nature of the beryllium in recent hybrid designs raises new issues, such as beryllium self welding. The MHD issue for the hybrid blanket differs from that for the liquid-metal breeder blanket, in that the liquid metal flows through a packed bed in some hybrid designs.

Hybrid-specific issues include fuel handling and reprocessing, neutron economy, and potentially new failure modes and effects. The major reprocessing R&D need is associated with removal of uranium by fluorination. The process is fairly well understood, but it requires demonstration at a moderate scale. The neutron-economy issue requires better calculations, experimental verification of nuclear performance, and emphasis on designs that minimize structural material. (The breeding performance of the molten-salt hybrid design is very sensitive to parasitic absorption in structural material, principally iron. Alternatives to steel should be considered.)

Objectives and Attributes

Blanket/first wall technology has two major subobjectives: predictability and attractiveness. In particular, it is necessary to gain sufficient understanding to assess the performance of blanket/first wall components. Attractive blanket/first wall technologies must also be established to provide fuel and energy production and recovery. The objectives and attributes are indicated in Fig. 4.3. Rough estimates are made of the reactor goals for each attribute. These estimates are intended only as guidelines, based on our current concept of an attractive reactor. References 1, 2, and 4 provide more detailed explanations of the objectives and attributes than can be presented here.

The predictability objective seeks to specify a level of confidence in the predictions of component behavior in actual reactor usage. There is a need to understand and characterize the basic phenomena and to develop the capability to predict the performance of subsystems. Such understanding and predictive capability must be obtained through a combination of theoretical and analytical modeling and experiments performed under testing conditions relevant to the fusion environment. The first aspect of predictability requires the development of an understanding of the important fundamental phenomena (e.g., liquid-metal flow in a magnetic field) that define component performance; the second aspect requires the definition of engineering parameters (e.g., neutron wall load) that limit the design conditions.

The attractiveness objective requires a demonstration that technologies exist that can perform the key functions of fuel and energy production and recovery while providing attractive performance, cost, safety, and environmental features for all candidate confinement concepts and applications. Material combinations, design options, and configurations selected must have excellent potential for maximizing the attractiveness of fusion reactors. This secondary objective also has two parts: performance/economics and safety/environment. Specific quantitative goals are listed in Table 4.8. For performance/economics, for example, higher allowable wall loads would

FIGURE 4.3 Primary and Secondary Objectives and Attributes for Blanket/First Wall Development

TABLE 4.8 Attributes and Nominal Goals for the Secondary Objective of Attractiveness

Attribute	Nominal Goal
Neutron Wall Load (MW/m^2)	5
Surface Heat Flux (MW/m^2)	1.25
Tritium Breeding Ratio	1.05 minimum (3-D)
Product of Blanket Energy Multiplication and Net Thermal Efficiency	0.5
Blanket/First Wall Shield Power-Conversion System (% of plant cost)	20
Lifetime Fluence ($\text{MW}\cdot\text{yr}/\text{m}^2$)	15
Inboard Blanket/First Wall Shield Thickness (m)	1.0
Blanket-Sector MTBF (h)	10^5
Blanket-Sector MTTR (h)	730
Routine Radioactivity Release (mrem/yr)	1^a
Response to Loss of Coolant Flow	Passive safety
Short-Lived Activation (km^3 BHP)	10^8 (10^4 AFT maximum)
Vulnerable Tritium Inventory in One Component (g)	15
Long-Term Disposal Hazard	Near-surface burial

^a1 mrem/yr = 10 Ci/d tritium.

provide better reactor performance in terms of mass utilization and power density. Safety/environment attributes include the potential source terms (e.g., tritium inventory per component) and the nature and likelihood of release mechanisms.

Complex trade-offs between these two secondary objectives may be necessary. For example, a lower-power-density system may be more expensive because of the lower mass utilization. However, it is likely to be more tolerant of accidents, and consequently simpler, with fewer safety-related systems and potentially higher availability.

4.1.2 Program Logic and Evaluation/Decision Points, Major Facilities, and Resource Estimates

Program Logic

The blanket/first wall test plan is based on four phases, described in the chapter overview. The near-term objectives of the test program during the 20-yr period of the plan are listed in Table 4.9 for each major task area. After approximately 10 years, the individual tasks are combined, as experiments contribute data to a large number of tasks simultaneously.

Figure 4.4 indicates the test logic and key relationships among tasks. Relationships to the tritium processing and materials tasks are also shown. Table 4.10 provides an explanation of the milestones and evaluation points in the figure.

The major evaluation points clearly divide the plan into the four phases described earlier. The end of Phase 1 is defined by evaluation points <a-d>, which involve early selection of candidate materials. Early selection is important in order to reduce the breadth, and consequently the cost, of the program. Tables 4.11 and 4.12 provide details on <a> and .

Evaluation points <e-g> define the end of Phase 2, which emphasizes multiple-effects testing. By approximately 1995, concept verification experiments must be defined for solid-breeder, liquid-breeder, and hybrid (fission/fusion) blankets. In addition, a comparison will be made between solid- and liquid-breeder blankets, potentially resulting in the termination of research in one area and focusing of resources, <h>. The final evaluation point, <i>, defines the end of Phase 3 and the initiation of fusion integrated testing. Leading blanket designs must be selected and test modules designed for fusion testing. It is in this time frame that a quantitative assessment of fusion will be made.

More detailed analysis of the test plan for solid-breeder, liquid-breeder, and hybrid blankets has been performed (see Refs. 1, 2, and 5).

TABLE 4.9 Blanket/First Wall Tasks

Issue	Late 1980s	Early 1990s	Late 1990s	Early 2000s
MHD (including insulators)	Explore MHD phenomena in simple geometry tests.	Explore MHD phenomena in complex tests. Perform scoping experiments for macroscopic effects.		
Material Compatibility	Explore basic materials interactions.	Perform further loop testing to determine design limits and impurity-control techniques.		
(Tritium recovery and permeation)	Explore tritium-recovery techniques in small-scale experiments. Measure basic permeation properties and rates. Develop tritium design goals.	Verify tritium-recovery techniques in small-scale experiments. Operate transport loops to demonstrate tritium control.	Perform nonfusion concept verification testing for selected liquid-and/or solid-breeder material combinations and concepts.	
Solid-Breeder Tritium/Thermal Behavior	Measure basic properties of solid breeders.	Perform advanced in-situ tritium-recovery experiment on selected material combinations.		Perform fusion testing for selected concepts.
Neutronics	Perform simple geometry experiments.	Perform engineering mockup experiments.		
(Materials)	Generate moderate-fluence data for initial alloys in fission reactors. Measure basic properties.	Generate moderate-fluence data for improved alloys in fission reactors. Develop fabrication and recycling techniques.	Generate high-fluence data for improved alloys in fission reactors.

FIGURE 4.4 Level 2 Logic Diagram for Blanket/First Wall (Decision points <a-i> are described in Table 4.10.)

TABLE 4.10 Blanket/First Wall Evaluation Points and Milestones^a

a	Select liquid-breeder blanket materials and design classes.
b	Select solid-breeder blanket materials and design classes.
c	Assess tritium breeding potential of Li_2O .
d	Select fuel form and fuel-processing methods.
e	Select material combinations and configurations for nonfusion concept-verification testing of liquid-breeder blankets.
f	Select material combinations and configurations for nonfusion concept-verification testing of solid-breeder blankets.
g	Select material combinations and configurations for nonfusion concept-verification testing of hybrid blankets.
h	Compare solid-breeder and liquid-breeder concepts. Select a small number of concepts from one or both categories.
i	Select and design blanket modules for fusion testing.
M ₁	Complete characterization of Li_2O ceramic breeders.
M ₂	Complete characterization of ternary ceramic breeders.
M ₃	Achieve moderate fluence for initial candidate alloys and select advanced reference alloys.
M ₄	Complete MHD experiments. Establish feasibility of self-cooling and optimize cooling methods. Determine MHD design limits.
M ₅	Complete basic materials interaction experiments. Determine operating limits and demonstrate adequate impurity-control techniques.
M ₆	Demonstrate tritium-extraction system for Li and/or LiPb.
M ₇	Complete advanced in-situ tests.
M ₈	Complete neutronics engineering mockup tests. Demonstrate margin for achievable tritium-breeding ratio.
M ₉	Achieve moderate-fluence nonfusion irradiation for reference alloys.
M ₁₀	Complete nonfusion concept verification for liquid-breeder blankets.
M ₁₁	Complete nonfusion concept verification for solid-breeder blankets.

TABLE 4.10 (Cont'd)

M_{12}	Complete nonfusion concept verification for hybrid blankets.
M_{13}	Achieve high-fluence nonfusion irradiation for reference alloys.

^aSee Fig. 4.4.

Major Facilities

The blanket/first wall test program is currently at a very early stage. Consequently a number of relatively inexpensive experiments ($\leq \$5$ million each) will be critically important; the larger experiments that will be required in the future are difficult to specify in detail, because they will rely on results of ongoing experiments and concept screening. This section briefly describes ongoing experiments and facilities, as well as the major new facilities.

Some experiments are already underway, both in the U.S. and around the world. The most significant active experiment for solid-breeder blankets in the U.S. is FUBR-1B/BEATRIX, in which solid-breeder specimens are subjected to closed-capsule irradiation in the EBR-II fission reactor, with lithium burnups of up to 9% (see also Sec. 5.2). These tests will provide data on tritium and helium retention (and swelling), as well as information on breeder mechanical behavior, thermal stability, mass transfer, and temperature limits. Other current U.S. activities include the fabrication of solid-breeder material as part of the BEATRIX international materials-exchange program through the International Energy Agency and smaller materials preparation and -characterization programs.

A number of existing neutron sources are potentially available for additional testing. Fission reactors are primarily useful for testing tritium-recovery and thermomechanical issues for solid-breeder blankets. Point neutron sources are useful primarily for tritium-breeding and neutronics experiments related to both solid- and liquid-breeder blankets.

Several nonneutron experiments and facilities also exist, in technical disciplines primarily relevant to liquid-metal blanket issues. A key element in the current U.S. MHD program is the ALEX (Argonne Liquid Metal Experiment) facility, which is expected to provide information on single-channel pressure drops and velocity profiles in simple geometries (Ref. 6). In the area of material compatibility, several corrosion loops are in operation (see also Sec. 5.2). These loops provide valuable information for identifying compatible material combinations, but large uncertainties remain in defining accurate temperature limits, the effects of impurities, and methods of controlling corrosion.

The nuclear technology programs worldwide are at a comparable level to that of the U.S. The experimental efforts generally include solid-breeder fission-reactor

TABLE 4.11 Description of Evaluation Point <a>, Liquid-Breeder Blanket Materials and Design Classes^a

Statement of Decision

Narrow coolant breeder and structural-materials options for liquid-breeder blankets.

Decision Criteria

Thermomechanical limits on reactor parameters due to MHD effects and material compatibility.

Ability to predict temperatures and pressure drops in relatively complex geometries.

Safety and environmental characteristics, including activated corrosion-product transport, tritium release rates, consequences of accidents, and activation inventory.

Sources of Information

MHD fluid-flow/pressure-drop experiments and models; insulator development.

Tritium breeding experiments and models.

Corrosion-loop experiments, including bimetallic loop to determine the feasibility of refractory-alloy/steel loops.

Demonstration of breeder tritium-extraction and -control techniques from tritium task.

Results from chemical-reaction tests (for safety evaluation).

Input from the materials program to define structural materials and insulators possessing acceptable properties and radiation limits.

Input from systems studies to define reactor parameters, such as magnetic field, surface heat flux, etc.

Input from plasma experiments to define plasma edge conditions.

Outcomes and Consequences of Decision

Structural materials choices may reduce the test matrix for structural-materials irradiation. In particular, the perceived attractiveness of refractory alloys may be increased or decreased relative to steels.

Breeder-materials choices will limit research in tritium extraction and materials interactions. A smaller number of tritium-extraction and impurity-control systems will require demonstration.

If elemental lithium remains a primary breeder candidate material, then research on nonwater-cooled components should be increased.

A choice between self-cooled and separately cooled designs will affect several key issues for liquid-breeder blankets, including the importance of tritium breeding and MHD effects, and corrosion criteria.

TABLE 4.11 (Cont'd)

Demonstration of fabricability and reliability of MHD insulators will significantly alter the future of the test program and the relative attractiveness of solid vs. liquid breeders and self-cooled vs. separately cooled blankets. (If self-cooled LM blankets look attractive, there will be greater emphasis on MHD insulators, etc. If self-cooled LM blankets look unattractive, there will be greater emphasis on separately cooled designs.)

^aSee Table 4.12.

irradiation tests, as well as some nonneutron experiments for liquid-metal MHD and corrosion. In some cases, the approach of international programs has emphasized early interactive-effects testing. An example is the Malice loop at MOL, where materials interactions are studied together with magnetic-field effects (Ref. 7).

Existing experiments will not provide sufficient data to completely satisfy the objectives of the testing program. In order to explore and resolve the important issues, a range of new experiments and facilities will be required. These new experiments span various levels of integration, from basic properties, to exploration of phenomena in separate and multiple-effects tests, to concept verification in integrated fusion tests.

The most significant phenomena for solid-breeder blankets require tests that correctly simulate tritium generation and heating rates. By appropriate ⁶Li enrichment in the breeder material, it is possible to simulate fusion tritium generation and heating rates in fission reactors. Reactors with vented-test capabilities can also provide direct simulation of the purge environment. Overall, nuclear testing in existing fission reactors is an important resource for solid-breeder blankets. For liquid-breeder blankets, many of the critical issues involve magnetic-field effects and material interactions. Large uncertainties exist even in the absence of irradiation, so that nonneutron test stands can provide valuable information. Many of the experiments required for hybrid-blanket development will be performed naturally in conjunction with the solid- or liquid-breeder experimental programs. A small number of additional, unique facilities will be needed to carry out the fuel processing, fuel-handling, and beryllium-technology tasks.

A variety of experiments and facilities will be required in the future to develop blanket/first-wall concepts. Most of these experiments, such as corrosion testing for liquid-metal systems or irradiation behavior of solid breeders in fission reactors, are relatively inexpensive and will not require large new facilities. (See Ref. 1 for detailed information on the test and facilities needs for blankets.) However, a number of large facilities have been identified; brief descriptions are provided in Table 4.13.

Two types of facilities have been defined for the purpose of acquiring concept-verification information for liquid-breeder blankets under nonneutron conditions. Because their operation would occur only after 5-10 years of more fundamental testing,

TABLE 4.12 Description of Evaluation Point , Solid-Breeder Blanket Materials and Design Classes^a

Statement of Decision

Narrow coolant, breeder, multiplier, and structural materials options for solid-breeder blankets.

Decision Criteria

Tritium inventory in solid breeder and multiplier.
 Achievable tritium breeding.
 Stability of materials and material combinations at moderate neutron fluence.
 Permeability of structural materials and effect of surface conditions.
 System safety under off-normal conditions.

Sources of Information

Basic properties from blanket and materials tasks.
 Tritium recovery and thermomechanics data from the solid breeder-blanket task.
 Tritium-breeding potential and uncertainties, as determined from neutronics.
 Tritium permeation, containment, and processing characteristics from the tritium systems task.
 Input from systems studies to define reactor parameters, such as neutron wall load, surface heat flux, etc.
 Input from plasma experiments to define first-wall erosion rates, tritium permeation rates, etc.

Outcomes and Consequences of Decision

Materials choices will reduce the test matrix for irradiation testing and will help define the achievable TBR and the required accuracy of predictions.
 Coolant choice (He vs. H₂O) will determine tritium-extraction and containment requirements.
 Breeder choice will determine need for and form of multiplier.
 If Li₂O looks attractive, there will be less emphasis on ternary oxides with multipliers.

^aSee Table 4.11.

only qualitative features can be specified at this time. A Thermomechanical Integration Facility (TMIF) is one potentially attractive concept. It would test the combined influence of heat-, mass-, and momentum-transport issues, as well as some structural issues. This testing would help develop empirical relations for describing the global behavior of the blanket. The TMIF would be larger than earlier MHD experiments, and it would include more prototypical geometries and a number of key subsystems with strong blanket interfaces (such as impurity-control and tritium-extraction systems).

TABLE 4.13 Blanket/First Wall Test Facilities

Facility Characteristics	Description
Liquid-Metal Flow Facility	
Purpose	Validate MHD fluid flow and heat transfer; provide design data for configuration screening and for input information to design the thermomechanical integration facility; explore techniques to reduce pressure drop and enhance heat transfer.
Key Features	Basic elements of relevant geometries; large, high-field magnets (4-6 T, 1-2 m ³).
Cost Estimates	Construction: \$7-10 million over 1-2 yr. Operation: \$1.0 million/yr over 4-6 yr.
MHD Mass-Transfer Facility	
Purpose	Provide design data on MHD heat and mass transfer; develop and verify techniques to reduce corrosion and corrosion effects; test instrumentation in relevant environment.
Key Features	Basic elements of relevant geometries, relevant material combinations, complete transport loop; long operating time; relevant thermal hydraulics and chemical environment; strong magnetic field (4-6 T, 1-2 m ³).
Cost Estimates	Construction: \$8-12 million over 2 yr. Operation: \$1.5 million/yr over 6-8 yr.
Liquid-Metal Thermomechanics Integration Facility	
Purpose	Validate and refine design configurations; generate design data for blanket test module (to predict operating characteristics and operating limits).
Key Features	Actual materials and geometry; transport loop and blanket subsystems included; relevant environmental and operating conditions (4-6 T magnetic field, 1-2 m ³ field volume, 10-40 MW heat input).

TABLE 4.13 (Cont'd)

Facility Characteristics	Description
Cost Estimates	Construction: \$20-40 million over 2-3 yr. Operation: \$2-3 million/yr over 8-10 yr.
Blanket Neutronics Engineering Mockup	
Purpose	Reduce uncertainties in predicting achievable tritium breeding; reduce uncertainties in predicting nuclear heat-generation rates and profiles.
Key Features	Well-calibrated 14-MeV-neutron source; low fluence; relevant blanket geometries; measurement of neutron flux, energy spectra, and reaction rates (including tritium heating and transmutations).
Cost Estimates	Construction: \$5-10 million over 2 yr. Operation: \$2-3 million/yr over 5 yr.
Advanced In-Situ Tritium-Recovery Assembly	
Purpose	Explore tritium recovery with local reactor-relevant conditions; develop ability to predict tritium inventory and transport in solid-breeder materials.
Key Features	One or more instrumented and purged subassemblies in fission reactors; includes the following conditions, which can be varied from canister to canister: (1) moderate to high burnup; (2) temperature gradient; (3) purge flow; (4) breeder cladding mechanical and chemical interaction; and (5) time-dependent effects.
Cost Estimates	Construction: \$3-5 million each over 2 yr per assembly. Operation: \$1.5 million/yr each over 3 yr.
Solid-Breeder Nuclear Submodule	
Purpose	Verify overall behavior of blanket submodule in neutron environment with conditions simulating middle and end of life.

TABLE 4.13 (Cont'd)

Facility Characteristics	Description
Key Features	Complete coolant and purge flow systems and instrumentation; reproduction of blanket geometry, material combinations, and all environmental conditions achievable in a fission reactor.
Cost Estimates	Construction: \$5-10 million each over 3 yr (per submodule). Operation: \$2-3 million/yr each over 3 yr.

A second type of facility, called the Partially Integrated Test Facility (PITF), would be a full- or near-full-scale test that would simulate all environmental conditions except neutrons. (The omission of neutrons results in large cost savings, while many of the key issues can still be addressed.) For many important operating parameters, such as surface heat flux and magnetic field, partially integrated experiments could provide a good simulation of the operating characteristics of a power reactor. These experiments should provide useful information on failure modes and component reliability.

Future tests for solid-breeder blankets will require integrated neutron testing. The neutron sources would be either existing fission reactors or point neutron sources. The primary facility needs are the design and construction of blanket modules and submodules for testing in a neutron environment. The facilities that have been identified are a blanket neutronics engineering mockup, an in-situ tritium recovery assembly, and a solid-breeder nuclear submodule (see Table 4.13).

4.2 RADIATION SHIELD

The radiation shield of a fusion reactor protects the various reactor systems from excessive activation, limits nuclear heating and radiation damage, and minimizes radiation exposure to plant workers and the public. The radiation shield is a "generic" component; the designs are functionally similar for all confinement schemes, although the mechanical configuration must fit the specific machine and the required radiation attenuation will depend on whether normal or superconducting magnets are used.

A large number of conceptual designs of fusion radiation shields have been produced, and several detailed designs have been completed. However, only very simple homogeneous concrete shields for 14-MeV point source experiments, such as RTNS-II, have been built. Some small-scale experiments on more sophisticated shields have been done at ORNL and in Japan, but at present very little is being done on radiation-shield development in the fusion programs of the U.S., the European Community (EC), or

Japan. Radiation-shield development will be needed in order to acquire a firm basis for the successful design, construction, operation, and repair of the fusion ignition device and of subsequent fusion reactors. Reliable shield design and analysis are needed for near-term fusion experiments. More sophisticated, optimized shield designs are needed for fusion reactor applications.

4.2.1 Technical Issues, Objectives, and Attributes

Radiation-Shield Development Issues

There appear to be no special technological difficulties associated with the construction and thermomechanical operation of the fusion reactor radiation shields. There are critical issues related to the radiation-attenuation capability of the shield; these are not fundamental to the feasibility of fusion, but they could strongly affect the design of fusion reactors. Attenuation problems could have significant impact on the successful operation of a fusion experiment, in that it may be very difficult (or impossible) to add more shielding after the experiment has been built.

Obtaining good accuracy in predicting the local effects and deep penetrations of concern for radiation shields will be much more difficult than predicting the integral effects and shallow penetration associated with first-wall and blanket analysis. Fortunately, precise accuracy is not required, because the addition of more shielding during the design stage can provide an additional attenuation design margin. The addition of extra shielding, however, may require extra space in reactor designs that are already extremely space-limited. This uncertainty could prove extremely expensive. It is even more important to avoid the "Mutsu phenomenon," the belated discovery of a major flaw in the performance of the radiation shield that would render the reactor inoperable.

The effort required to address the radiation shield issues is not large, but it must be made. The radiation shield development issues are summarized in Table 4.14.

Specification of Radiation-Protection Guidelines. Specification of the radiation source terms and the design criteria for the shield are needed to adequately protect components and personnel:

Radiation Source Definition. There is presently a large uncertainty associated with our ability to predict the magnitude and the spatial distribution of the fusion neutron source from the plasma. The predictions of radiation transport and response can be no better than the knowledge of the source. This issue must be pursued primarily by the plasma physics task area. The TFTR may (and the ignition device should) give much more confidence in our ability to predict the plasma source of fusion neutrons.

TABLE 4.14 Summary of Radiation-Shield Issues

Issue	Potential Impact ^a	Design Specificity (reactor/application)
Specification of Radiation Protection Requirements		
Radiation-source definition	F3	Generic
Radiation-limit design criteria	A1, A3	Specific issues for all designs
Corrosion, transport, and deposition	A3, A4	Specific issues for all designs
Verification of Shield Effectiveness		
Effectiveness of bulk shield	F3, A3	Generic
Radiation response	F3	Generic
Penetration shielding and streaming	F3, A1, A3	Generic
Shield design method improvements	A3	Generic

^aFeasibility Issues

F1 - May close design window.

F2 - May result in unacceptable safety risk.

F3 - May result in unacceptable reliability, availability, or lifetime.

Attractiveness Issues

A1 - Reduced system performance.

A2 - Reduced component lifetime.

A3 - Increased system cost.

A4 - Less desirable safety or environmental impact.

Radiation-Limit Design Criteria. The design criteria must be specified for the radiation dose and dose-rate limits of the components the shield is to protect. Preliminary limits exist, but many are uncertain by an order of magnitude or more. Limits must be established for superconducting-magnet materials at liquid-helium temperatures, for vacuum systems, for rf and other heating systems, and for diagnostic components. These limits, for the most part, should be established by the groups responsible for the component materials and designs.

Corrosion, Transport, and Deposition. A major source of radiation dose to operating personnel can come from materials that become activated inside the radiation shield but are transported outside that shield by various fluid streams. Corrosion, erosion, and sputtering of blanket and shield materials, transport in coolant streams, and deposition in pipes or heat exchangers outside the shield must be included in shield and maintenance design considerations. Most of the required data will have to be provided by the fusion material research tasks.

Verification of Shield Effectiveness. The key critical issue for radiation shielding is verification that the shield will satisfy its design requirements:

Effectiveness of Bulk Shield. The radiation-attenuation effectiveness of the bulk shield must be verified by integrated experiments. At present, the calculational capability exists to predict the radiation attenuation of the bulk shield, but the accuracy of these calculations is not yet known. Integrated verification experiments are needed to test the completeness and accuracy of the nuclear data, the accuracy of the calculational methods, and the adequacy of the shield designs. This is particularly important for sophisticated, multilayer shields. Simple point source benchmark experiments (1986-90), using existing point neutron source facilities, are needed; these experiments would be followed by point source mockup experiments (1990-95) and prototype experiments on the fusion ignition machine (1995-2000).

Radiation Response. The accurate prediction of radiation transport and attenuation is only part of the job of radiation-shield analysis. Another important part is the accurate prediction of the response to this radiation. Radiation response includes activation and the resulting secondary radiation source, heating, materials damage, and personnel dose. Calculational capability exists to predict radiation response; the issue is the verification of the accuracy of this calculational capability. Integrated experiments, probably coupled to or simultaneous with those for verification of radiation transport, are needed to test the understanding of radiation response (including the production of activation products, secondary gamma rays, neutron and gamma heating deposition, and radiation damage). Existing point source facilities with sufficient strength can be used (e.g., RTNS-2 and FNS). Activation measurements should be made in the TFTR in 1990.

Penetration Shielding and Streaming. A major concern is streaming of primary and secondary radiation through component ducts, diagnostic-port penetrations, coolant lines, and engineering-assembly gaps in the radiation shield. Monte Carlo methods are probably adequate for the analysis of even complicated penetrations, but the calculations can be so computationally intensive that they may be impractical to use as a routine design tool. Approximate design methods appear inadequate, and a compilation of typical penetrations and gaps ("designers'

handbook") would be useful. Verification testing of important penetration configurations is needed. Experiments are needed to verify the analysis of penetration streaming and shielding and to generate simplified methods of penetration-shield design. Benchmark experiments (1985-90), penetration mock-up experiments (1990-95), and prototype experiments (1995-2000) using existing point source facilities (integrated shield test facility, RTNS-2, and FNS) are needed, as are measurements in the lower penetrations of TFTR in 1990. Prototype measurements can be done in 1995-2000 using the fusion ignition experiment.

Shield Design-Method Improvements. The computational methods and data needed to design adequate radiation shields appear to be available. Certain improvements, however, should be made to facilitate design analysis. These include (1) the extension of activation-analysis codes to include the capability to explicitly follow the time-dependent operation schedule of an experiment or reactor; (2) the addition of variance-reduction methods for deep-penetration transport analysis and penetration streaming analysis, in order to make these analyses more computationally tractable; and (3) deterministic methods for analysis of void streaming.

Objectives and Attributes

In order to develop a firm basis for designing, building, operating, and repairing the radiation shields for fusion reactors and experiments, development work on radiation-shield systems should be directed toward the following objectives:

Radiation Protection Requirements. Verify DT source specifications from fusion experiments and establish limits and uncertainties for allowable radiation heating, dose and dose rate for key reactor components, including magnet insulation, superconducting magnet materials, diagnostics, and other fusion components. Attributes of this objective are the allowable limits that must be established and their uncertainties: magnet nuclear heating, dose to insulator materials, dose to diagnostic components, heating in vacuum pumping systems, etc.

Shield-Effectiveness Verification. Validate radiation-shield analysis codes and data for radiation transport, streaming, activation, heating and dose calculation, and verify the adequacy of shield designs. Attributes of the "verification" objective are the degree of agreement for the key shield characteristics: neutron and gamma fluxes, nuclear heating rates, activation levels, and materials doses.

These objectives are further defined in Table 4.15.

TABLE 4.15 Radiation-Shield Development Objectives

Objective	Late 1980s	Early 1990s	Late 1990s	Early 2000s
Radiation-Protection Requirements	Verify DT source specification.			
	Specify radiation limits for magnet insulations and diagnostics.	Specify radiation limits for S/C coils and vacuum pumps. Estimate corrosion, transport, and deposition in piping and heat exchangers.		
Verification of Shield Effectiveness	Perform bulk-shield benchmark experiments.	Perform bulk-shield mockup experiments.	Perform bulk-shield prototype experiments.	Verification and demonstration of performance in fusion facilities.
	Verify activation response.	Verify materials damage response.		
	Perform penetration benchmark experiments.	Perform penetration mockup experiments.	Perform penetration prototype experiments.	
	Improve design methods: activation, streaming, and sensitivity.	Develop optimized shield designs.	Develop fully integrated shield designs.	

Radiation-Shield Experiments

Some of the radiation-shield development issues must be resolved by means of shielding experiments. These issues include verification of bulk shield attenuation, validation of activation and other response calculations, and measurement of streaming effects. It appears that present facilities -- such as FNS and Octavian in Japan, the ORNL 14-MeV source and RTNS-II in the U.S., and LOTUS in Switzerland -- could be used effectively to do these experiments. Some limited benefit may be gained from TFTR and JET, and the ignition device could provide a significant testing facility.

The tasks required to resolve the radiation-shield issues described above are modest tasks and need not consume a great deal of time or money.

4.2.2 Program Logic, Decision Points, Facilities, and Resource Estimates

The program logic for the development of radiation shields for fusion experiments and reactors is quite straightforward. While shielding information is needed and shield experiments must be done, no major program branch points are foreseen.

Program Logic

The logic diagram for radiation shield development is shown in Fig. 4.5, and the milestones are described in Table 4.16. The logic flow is straightforward; each of the activities proceeds linearly to obtain the needed information, which is then passed on to activities in other fusion task areas. The information obtained from determination of radiation-protection requirements, shield design-method improvement, and activation measurements will play a key role in the final design of the radiation-shield design effort for the short-pulse ignition machine (~1990). The results from investigation of superconducting-magnet-materials radiation limits, activation and materials-damage experiments, and bulk-shield and penetration-shield effectiveness experiments will be important for the various design choices that must be made and may have a significant impact on the design activity for the long-burn demonstration (LBD) facility (~1995). The final results of all the radiation-protection-requirement tasks and shield-effectiveness-verification tasks will be used in evaluating the decision on whether or not to proceed with the design, construction, and operation of the IFF (2000).

Evaluation/Decision Points

There are no explicit, high-level decision points within the radiation-shield activity itself. The data measured and methods improved by the shield activity will result in design definition and changes, however, and these will be important for higher-level evaluation/decision points for the short-pulse ignition experiment, the LBD machine, and the IFF.

TABLE 4.16 Milestones for the Radiation-Shield Technical Planning Activity^a

Milestones	Description
M_1	Specify radiation dose limits for diagnostic equipment and coil insulation for the short-pulse ignition experiment.
M_2	Verify neutron activation for materials for the ignition device to within a factor of two.
M_3	Complete shield design-method improvements to include time-dependent activation, variance reduction, and 3-D sensitivity analysis.
M_4	Verify activation and materials damage in materials for the LBD device to within a factor of two.
M_5	Verify bulk-shield effectiveness for the LBD device to within a factor of two.
M_6	Verify analysis of penetration shielding to within a factor of two.
M_7	Specify radiation dose limits for the LBD device, including superconducting-magnet materials.
M_8	Verify effectiveness of bulk shields for fusion reactors for input to the decision on the IFF.
M_9	Verify analysis of penetration shielding for fusion reactors for input to the decision on the IFF.

^aSee Fig. 4.5.

Current Activities and Facility Descriptions

At present, virtually no shield-development activities are in progress in the U.S., Japan, or the European Community. Some work is under way on activation response to 14-MeV neutrons using the FNS facility at JAERI (Japan). Radiation shield development activities may be accomplished with careful utilization of existing facilities (point neutron sources, TFTR, or JET) and experiments done on the short-pulse ignition experiment. Facilities that could be used for shield development are listed in Table 4.17.

TABLE 4.17 Facilities to be Used in Radiation-Shield Development Activities

Facility	Developer	Source Strength
14-MeV-Neutron Point Sources (existing)		
Rotating Target Neutron Source II (RTNS-II)	LLNL (U.S.)	5×10^{12} n/s
Integral Shield Test Facility (ISTF)	ORNL (U.S.)	10×10^{12} n/s
Fusion Neutron Source (FNS)	JAERI (Japan)	5×10^{12} n/s
Octavian Facility (Octavian)	Osaka Univ. (Japan)	10^{12} n/s
LOTUS Neutron Source (LOTUS)	EPFL (Switzerland)	10^{12} n/s
Fusion Neutron Sources		
Tokamak Fusion Test Reactor (TFTR, existing)	PPPL (U.S.)	0.1 MW/m^2 , 1 s
Joint European Torus (JET, existing)	European Community	$\sim 0.1 \text{ MW/m}^2$, 10 s
Compact Ignition Experiment (CIT, planned)	U.S.	$\sim 3 \text{ MW/m}^2$, 5 s

4.3 TRITIUM PROCESSING

Issues considered in this section are those dealing with plasma exhaust processing, room and water detritiation (including permeation issues), blanket product processing (from fluid outside the blanket), and vacuum pumping.

4.3.1 Technical Issues, Objectives, and Attributes

Technical issues in tritium processing are summarized in Table 4.18. The issues may be characterized generally as issues of efficiency and reliability rather than feasibility, although the public visibility of tritium and safety tend to elevate the priorities of tritium issues.

Among the subissues in monitoring and accountability is the development of practical tritium monitors that indicate tritium concentrations accurately in the

TABLE 4.18 Summary of Issues Associated with Tritium Processing Systems^a

Issue	Impact ^b
Tritium monitoring and accountability needs	A3, A4
Impurity removal from DT fuels	A1, A3
Tritium in water-coolant loops	A3, F2
Tritium-processing-system integration	F3, A4
Room-atmosphere detritiation	A3, A4
Cryopump and valve lifetime	F3, A2
Maintenance of plasma-chamber vacuum	F1, A1

^aSpecificity is "generic" for all issues listed.

^bF1 = may close design window.

F2 = may result in unacceptable safety risk.

F3 = may result in unacceptable reliability, availability, or lifetime.

A1 = may reduce system performance.

A2 = may reduce component lifetime.

A3 = may increase system cost.

A4 = may have less desirable safety or environmental implication.

presence of neutron and gamma fluxes. Substantial progress has already been made toward resolving this subissue. Another subissue involves defining the regulatory requirements for tritium accounting. The existing requirements, developed for a different scale of use, are not achievable in a fusion machine. Related to accounting is the subissue of developing methods of on-line (or "near-real-time") accounting, such as have been developed for fission-fuel processing.

Impurity removal from DT fuels incorporates such subissues as defining the impurity species and amounts present in the plasma exhaust gases under various operating conditions and in the gaseous product from the breeder blanket. The resolution of these two subissues could determine the design of tritium-processing systems but would not be likely to affect the feasibility of such systems. A third subissue, of greater concern, is the question of how much tritium is lost or "hung up" in the purification processes. Conceivably, tritium losses or inventory in processing could be large enough to affect the required tritium-breeding ratio, and therefore, the blanket design.

Resolution of the issue of tritium in water-coolant loops requires resolution of a variety of subissues. Coolants other than water are discussed in Secs. 4.1 and 4.5. Key questions involving tritium in water coolant loops include: (1) magnitude of tritium permeation through the first wall, in-vessel components and to the water coolant; (2) leakage out of the blanket purge gas and from coolant loops; (3) tolerable levels of

tritium permeation and leakage; (4) methods to reduce tritium permeation; and (5) required level and cost of water detritiation.

Tritium-processing-system integration, covering the safety and reliability of integrated tritium-processing systems, includes the subissues of interfacing between subsystems and reliability of subsystems individually and taken together, including both transient and off-normal conditions. In part because of the experience and data base in tritium inherited from national security programs, the remaining tritium issues tend to be those of interfacing and integration. Although the feasibility of the interfaces (with a few possible exceptions) is not in question, the configuration of these interfaces (and fueler interfaces) that will give maximum overall efficiency, safety, and reliability remains to be established.

Room-atmosphere detritiation again includes policy issues, as well as many purely technical questions. The policy aspects concern the necessity of room-atmosphere detritiation systems (RADS) and their characteristics. Examples of technical issues include: (1) the rate of tritium oxidation under realistic conditions following a room release of molecular tritium and the speed of absorption and adsorption on the walls and surfaces; (2) reliability of RADS; and (3) the degree to which tritium can be confined during maintenance.

Cryopump and valve lifetime encompasses the subissues of the operating lifetime of the plasma-chamber vacuum valves and the plasma exhaust cryopumps. The vacuum valves for this service must be large (approximately 1-m diameter) and tritium-compatible (all metal). Such valves, capable of maintaining their performance after repeated cycling, do not now exist. The possibility of particulate matter in the plasma exhaust raises further questions about valve development. Questions about the compound cryopumps under development for exhausting the plasma chamber include the effect of tritium and the number and type of regeneration cycles on the adsorbents (molecular sieve or charcoal) and adsorbent binders used in these pumps. Plasma interactions with first-wall materials may be important in determining the particulate loading seen by the vacuum pumps and valves. Although the issue must be resolved, the level of concern is moderate, because various potential resolutions are available.

Objectives and Attributes

The overall objective is to provide proven, safe systems that efficiently handle and process the tritium-containing streams outside the plasma chamber and blanket. This overall objective may be expanded into the following primary items:

- Demonstrate the fuel-processing cycle for fusion power reactors;
- Develop and test environmental and personnel protective systems;
- Develop, test, and qualify equipment for tritium service in the fusion energy program;

- Provide subsystems that can be used for demonstration and that can be copied for advanced fusion machines;
- Demonstrate long-term reliability of components and safe handling of tritium;
- Define system response to off-normal and emergency operations.

The secondary objectives associated with these primary objectives are as follows:

- Fuel cleanup -- Meet fuel purity goals. The attributes are the nonhydrogen impurity after cleanup (goal value <10 ppm) and the tritium impurity level in the neutral beam feed (goal value <100 ppm).
- Availability -- Maximize availability. The attribute is the tritium system availability (goal value >90%).
- Tritium release -- Minimize tritium releases to air and water. The attribute is the tritium release to atmosphere and surface waters (goal value <1,000 Ci/yr).
- Tritium loss -- Minimize tritium losses. The attribute is tritium bound in molecular sieves or waste compounds (goal value ≤ 10 g/yr).
- Tritium inventory -- Minimize tritium inventory. The attribute is the tritium in any individual fluid processing unit (goal value <300 g).

4.3.2 Program Logic, Decision Points, Facilities, and Resource Estimates

Program Logic

The logic and sequencing of R&D work in the tritium area can be summarized on a logic diagram (Fig. 4.6) along four main lines of work -- plasma-exhaust processing, vacuum, tritium permeation and control, and blanket-product processing (which includes only the recovery and processing of tritium from a fluid carrier outside the blanket). Important interfaces also exist (shown in Fig. 4.6) between the tritium experiments and those in plasma technology and blanket development.

The overall tritium R&D strategy centers on the Tritium Systems Test Assembly (TSTA), an existing facility for the development and demonstration of the integrated plasma-exhaust-processing systems and associated safety systems. Later use of this facility ([M₃] on Fig. 4.6) for integration of vacuum components and blanket-product processing systems, after initial development elsewhere, is the logical course of the R&D lines in vacuum and blanket tritium. The permeation activity is done in smaller,

FIGURE 4.6 Level 2 Logic Diagram for Tritium (Decision point <c> is described in Table 4.19.)

dispersed facilities and requires no central effort in systems integration. A summary of the five-year milestones is given in Table 4.19.

Development of integrated plasma-exhaust processing is proceeding at TSTA. Issues under study include impurity removal, safety systems (especially room-air detritiation), tritium monitoring and accountability, tritium-processing losses, and integrated system reliability. Not included at present are water detritiation (under way in nonfusion programs) and blanket-product processing. Work on alternative cleanup components (e.g., palladium diffuser) is under way in several foreign programs, including programs in Japan, Federal Republic of Germany, and Canada. This work will lead to the selection, , and demonstration of an optimized system, [M₂], in the early 1990s that will be available for an ignition machine or IFF. The TSTA also provides expertise and training in tritium handling to TFTR.

TABLE 4.19 Decision <c> on Water-Detritiation System

Statement of Decision

Decide the system requirements (duties) and optimum process for water detritiation.

Decision Criteria

Rate of tritium permeation into coolant water (from first wall, from blanket).
Rate of tritium oxidation to form water (from room and glovebox atmosphere detritiation systems).

Form and concentration of tritium extracted from blanket.

Allowable tritium concentration in coolant loop (dependent on permeation from coolant loop and inventory and breeding-ratio considerations).

Cost of water detritiation.

Sources of Information

Permeation studies (ion-beam implantation, tritium plasma facility, permeability studies, etc.).

Blanket-development experiments.

Tritium self-sufficiency analyses.

Experience with water-detritiation systems (CANDU reactor at Darlington, Toronto; Mound Facility; Chalk River Nuclear Laboratories).

Fusion reactor design studies.

Outcomes and Consequences of Decision

Consequences (cost and safety) affect attractiveness. The water-detritiation system may become a feasibility question if leaks or removal costs become very high. This may also affect the in-process tritium inventory and the required tritium-breeding ratio.

^aSee Fig. 4.6.

The TSTA includes a single cryopump module to be tritium-tested. Beyond the mid-1990s, tests will also be needed to verify the success of development work on a large (1-m-diameter), tritium-compatible vacuum valve operable in the environment of activated particulate matter of the plasma chamber.

Tritium permeation (both conventional pressure-driven and plasma-driven permeation) must be understood for a range of conditions and materials to permit the design of safe steam-cycle detritiation systems, reactor maintenance procedures, and blanket systems of acceptable tritium inventory and breeding capacity. Current work is under way in both ion-beam and tritium-plasma facilities (TPX). Materials property measurements (diffusivity, solubility, heat of transport) are needed. Because of the high importance to permeation rate of oxidation chemistry (both for the tritium and the permeation barrier), experiments are needed to understand this, both as an integral part of blanket work and for permeation studies. Current industrial work on detritiating the heavy-water moderator in CANDU reactors will provide most of the data needed, at <c>, for detritiating water reactor coolant or steam-cycle working fluid.

Experiments in blanket extraction (outside the blanket) fall into three general (and sequential) activities, which must maintain close interactions with progress in the blanket-development programs. The first activity is testing of the extractors, getters, diffusers, etc. that are proposed, at <a>, for extracting tritium from the fluid carrier streams from liquid-metal breeding blankets. The second and subsequent activity is the semiscale testing, at [M₁], of one or (at most) two extraction processes for leading candidate blankets. Third, beyond 1995, would come the integration, at [M₃], of the blanket extraction system with the other tritium-processing systems at TSTA. This step might require the addition of one cryogenic distillation column to the four existing ones. The above work would lead to a decision, <e>, soon after 2000 on selected options for the integrated fusion facility.

Decision Points

The most complex evaluation point (most interactive with other technology areas) is <c>. This evaluation point is described in Table 4.19.

Facilities

No major new facilities are needed for tritium-processing development. Instead, the minor facilities or minor revisions to existing facilities indicated in Table 4.20 would be sufficient.

TABLE 4.20 Facilities to be Used in Tritium-Processing Development

**Tritium Systems
Test Assembly (TSTA)
with Blanket-
Development Facility**

Purpose	Blanket-product processing integration.
Features	Required processing technology (i.e., extraction processes, perhaps additional isotope separation). Comparable in scale to TSTA. This interface could be added either at TSTA or at blanket facility.
Cost	Revision of about \$5 million; the annual operating cost for the multipurpose process-development program would be about \$3 million/yr for perhaps an added 2.5 years.
Date Required	1997

Vacuum Test Stand

Purpose	Test 1-m-diameter vacuum valve and test cryopump life.
Features	Tritium handling capability for intermediate levels of tritium (few grams) and capability to simulate plasma-generated particulate matter.
Cost	\$2-5 million for construction. Operating costs will be \$0.5 million/yr for two years during mid-1990s.
Date Required	1997

**Semiscale Blanket-
Product Extraction
Facility**

Purpose	Develop and demonstrate technology for extracting blanket product from a fluid stream outside the blanket.
Features	Tritium handling capability.
Cost	Construction cost would be \$2-5 million, with operating costs of \$1 million/yr for three years.
Date Required	1993

4.4 NUCLEAR ELEMENTS OF PLASMA-INTERACTIVE COMPONENTS

4.4.1 Technical Issues, Objectives, and Attributes

The technical issues associated with PICs are listed in Table 4.21. In contrast to PIC development for plasma technology, the issues considered here emphasize commercial applications and long-term lifetime; and thermomechanical considerations have additional importance.

TABLE 4.21 Nuclear Aspects of Plasma-Interactive Components

Issue	Potential Impact ^a	Design Specificity
Particle pumping and impurity control	F1, A1	Closed systems
Erosion and redeposition	F3, A2	Specific issues for all confinement concepts
Thermomechanical and thermohydraulic performance	A1, A2	Generic
Irradiation lifetime	F3, A1	Specific issues for all designs
Tritium inventory and permeation	A3, A4	Generic
Fabrication techniques	A1, A3	Specific issues for all designs
Development of alternative coolants	F1, A1	Liquid-metal designs

^aSymbols for potential impact are as follows:

Feasibility Issues

- F1 - May close design window.
- F2 - May result in unacceptable safety risk.
- F3 - May result in unacceptable reliability, availability, or lifetime.

Attractiveness Issues

- A1 - Reduced system performance.
- A2 - Reduced component lifetime.
- A3 - Increased system cost.
- A4 - Less desirable safety or environmental impact.

Particle exhaust is required in a burning fusion device in order to avoid the build-up of helium ash, which would otherwise quench the fusion burn. In a steady state, the rate at which alpha particles are removed must be equal to their rate of production by the fusion reactions, with the level of helium in the plasma kept below about 5%. The helium-ash removal should be accomplished with minimal tritium removal to minimize reprocessing.

Erosion and redeposition of materials is a consequence of interactions with particles in the plasma edge and scrape-off regions. Energetic plasma edge particles will strike the surface of plasma-interactive components and cause sputtering erosion. It is expected that the sputtered particles will enter the scrape-off region (and possibly the main plasma), will be recycled, and eventually will be redeposited on other exposed surfaces. Calculations predict that the redeposition process will be important in reducing the impurity level in the plasma and extending the erosion lifetime of plasma-interactive components. At present, data on erosion and redeposition effects are largely unavailable, because of the short pulses and low availability of present experimental devices. In addition, the plasma scrape-off conditions in present devices are quite different from those expected in an ignition device or a reactor.

Redeposited materials may exhibit properties that make them inferior to the base materials in performance. The actual properties are unknown, because materials of sufficient thickness for testing have not yet been produced.

Heat transfer and thermomechanical response are key issues, because plasma-interactive components must withstand very high heat loads. Plasma-interactive components provide the particle and energy interface between the plasma and the surrounding reactor, and these functions require energy-removal capabilities. Furthermore, because the total energy transferred can be 10-20% of the fusion power, it is also desirable to be able to recover as much energy as possible at high temperatures. Energy removal is a near-term requirement, while energy recovery is an additional long-term goal. The issues and potential solutions are the same for both, except that energy recovery requires high-temperature coolant and provides even less margin for heat-transfer uncertainties.

Long-term options for coolants include water and such other coolants as helium, liquid metals, organic compounds, or molten salts for safety, efficiency, or design-simplicity reasons. This choice is closely related to the primary-blanket-coolant choice.

Thermal fatigue and bond integrity are two problem areas of long-range concern. In plasma-interactive components, the growth of thermal fatigue cracks can cause debonding of armor tiles, coolant leakage, or catastrophic fracturing of pressurized components. Prevention of each of the three failure modes will require extensive testing and improved models for failure analysis.

Radiation damage due to 14-MeV neutrons will affect most of the thermal and mechanical properties of materials. In the case of structural alloys, the primary concerns are the effects on mechanical properties, including ductility (impact, tensile, and creep), strength, fatigue and crack-growth behavior, and dimensional instability resulting from void or gas-bubble swelling and radiation-induced creep (see Secs. 5.1 and

5.2.). These effects are of lesser concern for the plasma-side materials, because the plasma-side materials do not carry primary loads. The integrity of bonds is expected to be degraded by radiation damage. In the case of nonmetals and insulators, the thermophysical properties, such as thermal conductivity and electrical resistance, will also be degraded. In addition, activation of materials is an important concern.

Radiation damage will affect fusion systems by affecting the lifetime and operating limits of PIC systems. Radiation-damage effects will be more important in high-fluence reactor systems, but low-fluence effects (particularly for nonmetals and insulators) will be important for near-term devices. In many cases, radiation damage will limit the choices of materials for plasma-interactive components (particularly if other life-limiting effects, such as erosion, are reduced).

In commercial reactor devices, tritium permeation through the surface of first-wall and impurity-control components into the coolant may contaminate the coolant to unacceptable levels. Tritium inventory in the advanced devices may be a less serious issue, because much larger inventories might exist in other components, such as breeding blankets. Safety issues associated with large on-site tritium inventories and provisions for handling tritium-contaminated components must be addressed, regardless of the inventory in first-wall and impurity-control components. The three materials options available to reduce tritium permeation in a power reactor are: (1) selection of lower-permeability materials, (2) coating the plasma surface with a high-tritium-recycling material, and (3) coating the coolant side with a low-permeability material.

The successful fabrication of plasma-side components is required for the demonstration of fusion. Fabrication does not present a fundamental feasibility issue, but a significant effort is needed, because the present data base is very small. Fabrication development issues differ for near-term and long-term devices, because the materials and component designs are different. The specific materials combinations are determined by many of the issues discussed above, such as energy removal and recovery, thermomechanical effects, and radiation effects.

The objectives and attributes are generally the same as those presented on PICs for plasma technology (Sec. 3.4).

4.4.2 Program Logic, Decision Points, Major Facilities, and Resource Estimates

Information on program logic and decision points is incorporated into Sec. 3.4.2.

Much of the data needed for the development of PICs will be obtained from tests in plasma-confinement devices, such as CIT or ETR. These devices are described in Chapters 1 and 2. Other test needs can be supplied by existing devices, such as FELIX for studying disruption forces and PMTF for studying effects of high heat loads. Laboratory equipment can also be used to investigate such basic phenomena as sputtering and redeposition. One facility specific to the nuclear elements of PICs has been identified; the details are presented in Table 4.22.

TABLE 4.22 Facility for Testing Plasma-Interactive Components with Nonwater Coolants

Name

Thermomechanical test facility using liquid-metal and helium coolants.

Purpose and Mission

Determine heat-removal capability and operating limits of PICs using different coolants.

Key Features and Physical Parameters

Facility will test components up to $\sim 1 \text{ m}^2$ to peak heat loads of 10 MW/m^2 . It will include a liquid-metal cooling loop capable of operating at temperatures $>300^\circ\text{C}$. The components for liquid-metal testing will operate in a magnetic field $>2 \text{ T}$.

Cost Estimates

Construction: \$8-10 million over 2 yr.

Operation: \$1 million/yr for 5 yr.

REFERENCES

1. M.A. Abdou et al., *Technical Issues and Requirements of Experiments and Facilities for Fusion Nuclear Technology*, University of California, Los Angeles, PPG-909, UCLA/ENG-85-39, Dec. 1985.
2. M.A. Abdou et al., *FINESSE: A Study of the Issues, Experiments, and Facilities for Fusion Nuclear Technology Research and Development*, Interim Report, University of California, Los Angeles, PPG-821, UCLA-ENG-84-30, Oct. 1984.
3. M.A. Abdou et al., *TPA Fusion Technology Status Report*, PPG-908, UCLA-ENG-85-38, Nov. 1985.
4. C.C. Baker et al., *Summary Report - Technical Planning Activity (TPA) Status*, Argonne National Laboratory, Dec. 6, 1985.
5. C.C. Baker et al., *Technical Planning Activity Interim Report*, Argonne National Laboratory, Feb. 1986.
6. C.B. Reed, B.F. Picologlou, and P.V. Dauzvardis, *Experimental Facility for Studying MHD Effects in Liquid-Metal-Cooled Blankets*, *Fusion Technology* 8(1), Part 2A:257, July 1985.
7. *Characteristics of Experimental Facilities for Research and Development in Lithium and Lithium-Lead Technology*, Studiecentrum Voor Kernenergie, Centre d' Etude de l' Energie Nucleaire, FT/MOL/B4-7/T85-002, Jan. 1985.

OVERVIEW

Fusion-materials development is one of the four key technical issues identified in the MFPP. The ultimate economics and acceptability of fusion energy, as with most other energy sources, will depend to a large extent on the limitations of materials for the various components. In addition to the thermal, chemical, and mechanical environments associated with most energy systems, fusion-reactor materials will be exposed to intense radiation generated by the high-energy (14-MeV) neutrons produced by the DT fusion reactions. Primary concerns are related to (1) the degradation of materials properties in the fusion-reactor environment, which will affect the functional performance of components and reduce their lifetimes, and (2) the production of radioactive transmutants, which will affect the safety and environmental aspects of fusion energy. Studies have shown that the development of improved materials is important for improving the economics of fusion power and reducing the need for long-term waste disposal. In addition to the importance of improved materials performance to the development of an attractive fusion-energy option, the high priority placed on the issue of fusion materials derives from the relatively long lead time required to provide an adequate materials-irradiation data base to evaluate the potential of fusion energy.

The objective of the fusion materials area is the development of new or improved materials that will enhance the economic and environmental attractiveness of fusion as an energy source. This area primarily involves the development of a fundamental understanding of the effects of the unique fusion-reactor environment on the properties and performance characteristics of materials under projected operating conditions and includes the development of baseline data on nonirradiated materials to support irradiation experiments. Although the effects of radiation on the properties of materials are the major concern, the effects of the thermal, chemical, and radiation environments must be studied simultaneously, in closely correlated experiments, because the effects of radiation cannot be understood without knowledge of thermally driven phenomena and chemical changes produced by interactions with the environment.

This area also includes the rationale and strategy for near-term (next 10 years) irradiation testing utilizing fission reactors and charged-particle accelerators and, in the long term, the rationale and strategy for utilization of a high-flux, high-energy-neutron (14-MeV) materials test facility. An important aspect of this issue involves development of a predictive capability for relating irradiation data from available test facilities to expected performance at extended lifetimes in a fusion-reactor environment. Baseline and fission-reactor irradiation data, plus information from theoretical and fundamental studies, will provide the basis for defining the allowable operating-parameter space (temperature and stress limits, fabrication and environmental constraints, etc.) for candidate materials at low-fluence/low-damage levels. Although fission-reactor and charged-particle irradiation data will provide a basis for scoping high-damage-level irradiation effects, high-fluence 14-MeV-neutron irradiation data are required to predict adequately the lifetime potential of materials, which must be known for the economic evaluation of fusion.

The scope of the materials area includes (1) development of new or improved materials for components of those subsystems for which materials performance poses a primary constraint to the development of an improved fusion-energy system and (2) fundamental research on selected materials to provide a basis for determining the feasibility and performance characteristics and for evaluating the safety and environmental features of fusion energy. The materials issues are addressed in terms of three program elements: structural materials, nonstructural blanket materials, and special materials (see Fig. 5.1).

FIGURE 5.1 Program Elements for Fusion-Materials Area

Technical Issues, Objectives, and Attributes

The technical issues associated with the performance of the materials in all three program elements can be categorized into the following eight areas:

- Mechanical performance (strength, ductility, etc.)
- Dimensional stability (creep, swelling, etc.)
- Thermophysical performance (thermal conductivity, thermal expansion, etc.)
- Thermochemical performance (compatibility, chemical stability, etc.)
- Neutronic performance (parasitic absorption, transmutations, etc.)
- Preparation and fabrication (welding, fabricability, etc.)
- Costs and resources (available resources, processing costs, etc.)
- Safety/environmental performance (activation, waste management, etc.)

The relative importance and potential impact of each technical issue will differ for the various materials functions, as indicated in Table 5.1. Mechanical performance, which is probably the most important issue for the structural materials, incorporates the combined radiation, thermal, and chemical effects on strength properties, residual ductility, fracture toughness, rupture life, and fatigue properties. Dimensional stability, which affects lifetime, includes radiation-induced swelling, phase stability, and creep. Thermophysical performance, which is of greater importance for the ceramics (structural, insulator, and solid-breeder), includes radiation effects on thermal conductivity and thermal expansion. Thermochemical performance, which involves most classes of materials, includes tritium (hydrogen) solubility and transport properties as they relate to tritium inventory and/or containment, chemical compatibility and stability, and compositional effects. The safety/environmental issue primarily concerns activity release, afterheat, waste management, and chemical reactivity. The nuclear performance issue incorporates parasitic absorption, transmutations, and nuclear heating. The preparation and fabrication issue includes fabricability, welding and joining, and reprocessing considerations as they relate to performance in the fusion-reactor environment. The cost and resource issue for all materials will affect the economics, and possibly the environmental impact, of fusion energy.

The emphasis is currently placed on the first-wall/blanket (FW/B) structural alloys and the tritium-breeding materials, because the performance requirements and the primary candidate materials for fusion-reactor applications are better defined. In general, the need, performance requirements, and candidate materials for the other materials functions are not as well established. However, the feasibility of fusion will

TABLE 5.1 Major Technical Issues for Fusion Materials

Materials Type	Properties
Structural Materials	
First-Wall/Blanket and PIC Alloys	Tensile and creep strength, embrittlement and loss of fracture toughness, as they affect the mechanical performance.
	Radiation-induced swelling and creep that affect dimensional stability.
	Property changes that result from compositional or phase changes caused by corrosion or chemical instabilities.
	Baseline properties of developmental materials required for selection.
	Fabrication/joining issues that relate to feasibility or performance in a fusion environment.
Nonstructural Blanket Materials	
Solid Breeders	Tritium inventory, as it is affected by tritium solubility and transport, and microstructure.
	Swelling and creep that affect dimensional stability or mechanical integrity.
	Changes in thermal conductivity and chemical stability at high fluence/Li burnup that affect thermal performance.
	Optimization of material microstructure and configuration for improved performance.
Liquid Breeders and Coolants	Radiolytic decomposition of molten salts and organic coolants.
	Solubility and stability of phases that affect thermochemical performance.
Ceramic Insulators	Fracture strength and thermal-shock resistance to withstand thermal transients.

TABLE 5.1 (Cont'd)

Materials Type	Properties
Ceramic Insulators (Cont'd)	Degradation of electrical properties by radiation.
	Swelling and creep that affect dimensional stability.
	Fabrication/joining for reactor performance.
Special Materials	
Plasma-Facing Materials	Fracture strength and toughness as they affect mechanical performance.
	Radiation-induced swelling that affects dimensional stability and/or mechanical integrity.
	Thermal conductivity and thermal expansion that affect thermophysical performance.
	Tritium solubility and transport that affect tritium inventory and thermochemical performance.
	Fabrication/joining issues that affect feasibility or performance.
Coatings and Claddings	Chemical stability/compatibility and tritium-transport issues that affect thermochemical performance.
	Fabrication/joining issues that affect feasibility or performance.
	Baseline properties that affect performance.
	Radiation-induced swelling and creep that affect mechanical performance.
Magnet Materials	Radiation-induced properties changes that affect thermophysical performance.
	Strength properties and embrittlement that affect mechanical performance.

also depend on satisfactory performance of these materials in the fusion-reactor environment. As in the case of the structural materials, a substantial development period will be required to provide a sufficient data base on irradiated materials to evaluate their performance characteristics for fusion applications. Therefore, it is essential to include these materials in the current fusion-materials program plan.

The overall objective in the fusion-materials area is the development of new or improved materials that will enhance the economic and environmental attractiveness of fusion as an energy source. This objective can be achieved by resolving the following four secondary subobjectives: (1) optimize the materials performance under the projected operating conditions, (2) maximize the capability for predicting materials performance under relevant conditions, (3) maximize the safety and minimize the environmental impact, and (4) minimize materials cost and resource requirements. Secondary and lower-level objectives and attributes are listed in Fig. 5.2.

Program Logic

The program logic for achieving the objectives defined for the fusion materials issue is based on the following factors:

- **Strategy for Fusion Development (defined by the MFPP).** Establish the scientific and technological base required to carry out an assessment of the economic and environmental aspects of fusion.
- **Overall Materials Objective.** Develop improved materials that will enhance the economic and environmental attractiveness of fusion energy.
- **Performance Goals.** Define goals on the basis of the roll-back assessment of the systems design and analysis area.
- **Strategy for Materials Development.** Provide baseline and high-fluence fission-reactor and 14-MeV irradiation data on low-activation materials with existing and projected test facilities.

Figure 5.3 indicates the programmatic logic for materials development for the three program elements. Table 5.2 provides additional explanation of the logic diagram. Tables 5.3-5.11 provide a detailed definition of the evaluation points of the materials logic diagram. In general, the first phase (approximately five years) of the program will provide baseline and low-fluence irradiation data (fission reactors) on selected low-activation materials. The second phase (5-yr period) will provide reference baseline data and moderate-fluence fission-reactor irradiation data on primary candidate materials. The third phase (5-yr period) will provide high-fluence (to 100 dpa) fission-reactor data with simulated transmutation effects on primary candidate materials. The fourth phase (5-yr period) will provide high-fluence, 14-MeV irradiation data on reference materials using a materials-irradiation facility designed and constructed during the second and third phases.

FIGURE 5.2 Objectives for Fusion Materials

FIGURE 5.3 Level 1 Logic Diagram for Materials

TABLE 5.2 Objectives for Fusion Materials Development (Level 1)

Materials Type	Late 1980s	Early 1990s	Late 1990s	Early 2000s
Structural Materials	Develop baseline and low-fluence data on low-activation FW/B materials.	Develop reference data and moderate-fluence property data with simulated He effects on primary candidate alloys.	Provide high-fluence (100 dpa) property data on primary candidate alloys with improved compositions and microstructures.	Evaluate effects of high-fluence, 14-MeV-neutron irradiation on properties of selected FW/B reference alloys.
	Improve radiation resistance through composition and thermo-mechanical treatment modifications.	Provide data on conventional alloys at high damage levels (100 dpa) and high helium concentrations.		
	Develop helium-simulation techniques.			
	Develop data on selected PIC alloys.	Develop selected PIC alloys with improved composition and microstructure.	Provide high-fluence (up to 100 dpa) property data on primary candidate alloys.	Evaluate 14-MeV radiation effects on reference PIC alloys.
Nonstructural Blanket Materials		Design high-fluence data on selected breeder materials.	Construct high-fluence, 14-MeV-neutron materials test facility.	Test irradiation effects in 14-MeV neutron facility.
	Develop baseline and low-fluence data on candidate breeder materials.	Provide moderate-fluence data on selected breeder materials.	Obtain high-fluence/burnup data on primary candidate breeder materials.	
	Develop insulator materials with improved composition and microstructure.	Provide low-fluence data on improved ceramic insulator materials.	Provide high-fluence data on primary candidate materials.	Provide 14-MeV radiation data on selected reference materials.
Special Materials		Provide baseline and low-fluence data on selected materials.	Define material operating limits based on fission-reactor irradiation data.	Evaluate 14-MeV-neutron radiation effects.

An assessment of the economic and environmental aspects of fusion requires that a sufficient materials data base be developed to adequately predict (1) the allowable operating-parameter space (temperature, strength, properties, etc.) at limited fluences for primary candidate materials and (2) the lifetime potential of these materials in a fusion-reactor environment. Fission-reactor irradiation data, in addition to baseline property measurements, should provide a reasonable assessment of the first requirement. Although high-damage-level neutron and charged-particle irradiations and theoretical and fundamental irradiation studies will enable advances to be made in materials performance, high-fluence, 14-MeV irradiation data are necessary to adequately predict the lifetime potential of materials for the economic assessment.

TABLE 5.3 Selection of Primary Candidate First-Wall/Blanket Structural Alloys, <a>

Statement of Decision

Select primary candidate FW/B structural alloys for further development.

Decision Criteria

Scoping tests indicate that alloys are readily fabricable and that baseline properties offer potential for attractive performance.

Scoping tests indicate that chemical stability/compatibility considerations will not excessively alter performance.

Low-fluence fission-reactor irradiation does not excessively alter performance.

Fundamental radiation studies indicate potential for long lifetime.

Low-activation compositions potentially provide attractive performance characteristics.

Potentially attractive first-wall/blanket concepts are defined.

Sources of Information

Materials tests provide baseline, mechanical, thermochemical, and radiation-effects data.

Systems studies and nuclear technology provide information on desired materials properties, operating conditions, and lifetime goals.

Outcomes and Consequences of Decision

Tests indicate that both vanadium alloys and ferritic steels offer potential for attractive performance: Continue development of both alloy classes.

Critical flaw identified in one alloy system: Continue development of other alloy system; focus other effort on identified critical flaw.

Tests indicate critical flaws in both alloy systems: Focus effort on resolution of critical flaws; increase effort on austenitic steel or other alternative materials.

On the basis of the overall objective of the materials area, emphasis is placed on development of materials with low long-term activation characteristics in order to reduce the waste-management requirements and to enhance the potential for materials recycle. The primary focus of the materials program area is the development of materials that exhibit high-performance characteristics and long lifetime in the high-energy-neutron (to 14-MeV) environment of a DT-fueled fusion reactor. However, an important aspect of the program is the measurement and improvement of baseline properties for candidate materials. Those materials that exhibit favorable performance characteristics in the thermal, chemical, and stress environments of interest will serve as a basis for the radiation test program.

**TABLE 5.4 Selection of Primary Candidate Breeder Materials, **

Statement of Decision

Select primary candidate tritium-breeding materials for further development and testing.

Decision Criteria

Scoping tests indicate that selected materials exhibit acceptable fabricability and baseline properties (physical and mechanical).

Scoping tests indicate acceptable thermodynamic performance.

Low-fluence/burnup tests indicate potentially acceptable irradiation performance.

Potentially attractive blanket concepts are identified.

Sources of Information

Materials tests provide baseline thermochemical and radiation-effects data.

Systems studies and nuclear technology provide information on desired materials properties, operating conditions, and lifetime goals.

Outcomes and Consequences of Decision

Scoping tests indicate that both Li_2O and several ternary oxides will provide attractive performance: Continue development of Li_2O and one or two ternary oxides.

Scoping tests indicate that selected ternary oxides are superior to Li_2O : Focus solid-breeder development on ternary oxides.

Liquid-breeder blanket concepts provide substantially better performance than solid-breeder concepts: Focus program on liquid-breeder development.

Solid-breeder blanket concepts provide substantially better performance than liquid-breeder concepts: Focus program on solid-breeder development.

Typical systems-parameter goals that are relevant to materials development for commercial reactor applications are a minimum integrated neutron wall loading of $10\text{-}20 \text{ MW}\cdot\text{yr}/\text{m}^2$, a minimum first wall lifetime of 2-5 yr, and a minimum neutron wall loading of $3\text{-}6 \text{ MW}/\text{m}^2$.

The strategy for materials development involves an iterative process that consists of evaluation of the performance characteristics of candidate materials, followed by development of materials with improved performance characteristics through compositional and/or microstructural modifications. The performance characteristics of these improved materials must be determined and the potential for further improvements evaluated. A major difficulty in the materials-irradiation program is the current lack of a high-fluence, 14-MeV-neutron irradiation test facility. Therefore, radiation effects in a fusion reactor must be deduced primarily from fission-reactor irradiation data, with displacement damage and transmutations, particularly helium effects, correlated by fundamental radiation studies.

TABLE 5.5 Selection of Primary Candidate Insulator Materials, <c>

Statement of Decision

Select primary candidate ceramic-insulator materials for further development.

Decision Criteria

Ceramic insulators are required in leading first-wall, blanket, and/or PIC concepts.

Scoping tests indicate that selected materials exhibit acceptable fabricability and baseline properties.

Scoping tests indicate that selected materials exhibit acceptable electrical/mechanical performance.

Moderate-fluence fission-reactor irradiation tests indicate that performance is not excessively degraded.

Sources of Information

Materials tests provide baseline, electrical, mechanical, and radiation-effects data.

Systems studies and nuclear technology provide information on desired materials properties, operating conditions, and lifetime goals.

Outcomes and Consequences of Decision

Scoping tests indicate that several candidate insulators will potentially provide adequate performance: Select best materials for high-fluence irradiation testing.

Scoping tests indicate that candidate materials will provide marginal/unacceptable performance: Investigate advanced/improved materials.

Electrical insulators are not required for high-flux regions: Terminate high-fluence irradiation testing.

Because materials performance influences the operating limitations of all components, the materials program must maintain a close interface with the other program areas, particularly systems design and analysis, nuclear technology, and plasma technology. The data generated provide a basis for establishing the operating limits, such as radiation dose levels, temperature, and stress, for the nuclear components. This information has important consequences for the design, performance, and lifetime of the first wall, blanket, and plasma-interactive components. Similarly, input from nuclear technology and plasma technology is used to establish the desired materials properties and serves to help guide the materials-development programs.

Structural Materials. The structural-materials program element consists of two subelements: FW/B structural alloys and PIC structural alloys. The FW/B alloys subelement focuses primarily on development of low-activation compositions of 9-12%

TABLE 5.6 Selection of Primary Candidate Plasma-Interactive-Component Structural Alloys, <d>

Statement of Decision

Select primary candidate PIC structural alloys for further development and testing.

Decision Criteria

Scoping tests indicate that alloys are readily fabricable and that baseline properties offer potential for attractive performance.

Scoping tests indicate that chemical stability/compatibility considerations will not excessively alter performance.

Low-fluence fission-reactor irradiation does not excessively alter performance.

Fundamental radiation studies indicate potential for long lifetime.

Low-activation compositions potentially provide attractive performance characteristics.

Potentially attractive PIC components are defined.

Sources of Information

Materials tests provide baseline, mechanical, thermochemical, and radiation-effects data.

Systems studies and nuclear technology provide information on desired materials properties, operating conditions, and lifetime goals.

Outcomes and Consequences of Decision

Scoping tests indicate that low-activation refractory metal alloys offer potential for attractive performance: Select best alloys for further development.

Scoping tests indicate that copper alloys offer potential for superior performance: Select limited number of copper alloys for further development.

Neither candidate refractory metal alloys nor copper alloys exhibit attractive performance characteristics: Investigate other innovative/advanced alloys/materials.

TABLE 5.7 Selection of Primary Candidate Materials for Special Applications, <e>

Statement of Decision

Select primary candidate special-purpose materials for further development.

Decision Criteria

Scoping tests indicate that selected materials exhibit acceptable fabricability and baseline properties.

Scoping tests indicate that selected materials exhibit acceptable thermochemical performance.

Moderate-fluence fission-reactor irradiation tests indicate that performance is not excessively degraded.

Sources of Information

Materials tests provide baseline, thermochemical, and radiation-effects data. Systems studies and nuclear technology provide information on desired materials properties, operating conditions, and lifetime goals.

Outcomes and Consequences of Decisions

Scoping tests indicate that several candidate materials will potentially provide adequate performance: Select best materials for high-fluence irradiation testing.

Scoping tests indicate that candidate materials will provide marginal to unacceptable performance: Investigate advanced and improved materials.

chromium ferritic/martensitic steels and selected vanadium base alloys, but this subelement also includes further development of conventional steels, such as HT-9 and PCA, as well as possible metallic and nonmetallic innovative materials. The program subelement on structural materials for plasma impurity control components (PIC alloys) focuses on high-strength copper alloys and selected refractory metal alloys, such as tantalum, tungsten, molybdenum, and vanadium.

Nonstructural Blanket Materials. This program element includes four types of materials: materials for solid breeders, materials for liquid breeders and coolants, multiplier/moderator materials, and ceramic insulators. Materials that provide the focus for the nonstructural-blanket-materials program include the following:

- **Solid breeders.** Lithium ceramics, such as Li_2O , LiAlO_2 , and Li_4SiO_4 .
- **Liquid breeders.** Liquid metals; lithium and Pb-17 at/o Li eutectic alloy.

TABLE 5.8 Selection of Improved Alloys for High-Fluence Testing, <f>

Statement of Decision

Select primary candidate alloy compositions/microstructures for high-fluence fission-reactor irradiation testing.

Decision Criteria

Baseline properties data and moderate-fluence fission-reactor irradiation data indicate that improved performance of selected primary candidate alloys is achieved by compositional/microstructural modifications.

Candidate alloys exhibit acceptable thermal and chemical compatibility characteristics under projected operating conditions.

Fundamental radiation studies indicate potential for long lifetime.

Sources of Information

Materials tests provide baseline, thermochemical, and radiation-effects data. Systems studies and nuclear technology provide information on desired materials properties, operating conditions, and lifetime goal.

Outcomes and Consequences of Decision

Materials data base, nuclear technology results, and systems studies indicate that one alloy system offers significantly better performance than the other system: Focus high-fluence irradiation effort on optimum alloy system.

- **Multipliers/moderators.** Beryllium and SiC.
- **Ceramic insulators.** Ceramics, such as MgAl_2O_4 , CaO, MgO, Si_3N_4 , and SiC.

The overall strategy for this program element is similar to that for the structural alloys: feasibility testing and selection of primary candidate materials from baseline and low-fluence irradiation data and lifetime evaluation based primarily on high-fluence irradiation testing.

Fission-reactor irradiation data may be adequate for the evaluation of solid-breeder performance. The necessary information on the liquid-breeder and coolant materials can be obtained primarily from nonirradiation tests. However, high-fluence 14-MeV irradiation data are necessary for evaluation of ceramic insulator and multiplier/moderator materials performance.

TABLE 5.9 Selection of Improved PIC Alloys for High-Fluence Testing, <g>

Statement of Decision

Select reference alloy compositions/microstructures for high-fluence fission-reactor irradiation testing.

Decision Criteria

Baseline properties data and moderate-fluence fission-reactor irradiation data indicate that improved performance of selected primary candidate alloys is achieved by compositional/microstructural modifications.

Candidate alloys exhibit acceptable thermal and chemical compatibility characteristics under projected operating conditions.

Fundamental radiation studies indicate potential for long lifetime.

Sources of Information

Materials tests provide baseline, thermochemical, and radiation-effects data. Systems studies, plasma technologies, and nuclear technology provide information on desired materials properties, operating conditions, and lifetime goals.

Outcomes and Consequences of Decision

Materials data base, nuclear technology results, and systems studies indicate that one alloy system offers significantly better performance: Focus high-fluence irradiation effort on optimum alloy system.

Special Materials. Materials functions that are not part of the first two program elements are included in the special-materials program element. Subelements of this program include plasma-facing materials, coatings and claddings, and magnet materials. In general, the candidate materials and the operating requirements are less well defined for these materials. As indicated in the logic diagram, effort in this area will be increased near the end of Phase I, when requirements have been better defined. The plasma-facing materials will most likely include low-Z materials, such as beryllium and graphite, and/or high-Z composites. Coatings and claddings may be used as tritium barriers, for compatibility purposes, etc. Oxide films are potential candidates. Advanced magnet materials, such as NbN, are included in this program element. In general, the early phases of the program will provide scoping baseline and low-fluence irradiation data. Primary candidates for various applications will be identified in Phase II, based on nonirradiated and moderate-fluence irradiation data. Reference materials will be selected during Phase IV.

TABLE 5.10 Selection of Reference Nonstructural and Special Materials, <h>

Statement of Decision

Select reference nonstructural blanket and special-purpose materials for high-fluence/lifetime testing in 14-MeV-neutron environment.

Decision Criteria

Reference materials that exhibit attractive performance characteristics after high-fluence fission-reactor irradiations are identified.

Reference materials offer potential for acceptable performance in integrated systems identified in systems studies, nuclear technologies, and/or plasma technologies.

Testing in a 14-MeV-neutron environment is necessary to provide reliable data for lifetime and/or performance evaluation.

Sources of Information

Materials tests provide baseline, thermochemical, and radiation-effects data. Systems studies, plasma technologies, and nuclear technology provide information on desired materials properties, operating conditions, and lifetime goals.

Outcomes and Consequences of Decision

Testing to high fluences in a 14-MeV-neutron environment is necessary to provide reliable data for lifetime and/or performance evaluation of reference materials: Test reference materials in high-flux 14-MeV test facility.

Lifetime and/or performance characteristics of reference materials can be adequately predicted from baseline and fission-reactor irradiation data: Test reference materials to high fluences in fission-reactors.

Testing at low-moderate fluences in a fusion test facility plus high-fluence fission-reactor tests will provide sufficient basis for lifetime and/or performance evaluation of reference materials: Test reference materials in ETR; test reference materials to high fluences in fission reactors.

Facility Descriptions

The facilities required to resolve the fusion-materials issues include a high-flux, 14-MeV-neutron materials test facility and several small (< \$10 million) materials test facilities.

A high-flux, 14-MeV-neutron test facility is required to provide necessary data to evaluate the lifetime potential of most materials. As indicated in the evaluation of the development of lifetime irradiation data, different possible approaches have been considered. These include (1) an accelerator-based facility that utilizes the D-Li reaction to produce high-energy neutrons, (2) a fusion-based test facility that could also

TABLE 5.11 Selection of Reference Structural Alloys, <i>

Statement of Decision

Select reference structural alloys for high-fluence irradiation testing in high-flux 14-MeV-neutron test facility.

Decision Criteria

Attractive reference structural alloys have been developed through comprehensive baseline testing programs, fission-reactor irradiations, fundamental irradiation studies, and moderate-fluence 14-MeV-neutron irradiations.

Attractive FW/B and PIC systems have been developed as part of the systems studies and nuclear and plasma technology programs.

Sources of Information

Materials tests provide baseline, thermochemical, and radiation-effects data. Systems studies, plasma technologies, and nuclear technology provide information on desired materials properties, operating conditions, and lifetime goals.

Outcomes and Consequences of Decision

Testing to high fluences in a 14-MeV-neutron environment is necessary to provide reliable data for lifetime and/or performance evaluation of reference materials: Test reference materials in high-flux 14-MeV test facility.

Test materials to moderate fluences in a fusion test facility and high fluences in fission reactors for lifetime and/or performance evaluation of reference materials: Test reference materials in ETR; test reference materials in fission reactors.

serve as a nuclear technology test facility, and (3) an ETR that could be used for materials testing. Since a high priority is placed on high-fluence (≥ 100 dpa) irradiation data, the accelerator-based source appears to be the most reasonable approach at this time. Table S.15 in the Synopsis presents both the minimum requirements and the desired characteristics of the materials-irradiation test facility.

Several minor test facilities costing less than \$10 million each will be required for the materials program. These test facilities and the estimated costs for each are listed in Table 5.12.

5.1 STRUCTURAL MATERIALS

Development of improved structural materials for use in the combined radiation, chemical, thermal, mechanical, and electromagnetic environment of an operating fusion

TABLE 5.12 Additional Materials Test Facilities

Facility Type	Cost (\$10 ⁶)
In-Pile Fission-Reactor Modules	10
Irradiation test vehicles for materials testing in fast-breeder and mixed-spectrum fission reactors	
Post-Irradiation Test Facilities	8
Remote hot-cell facilities for irradiation experiment disassembly, reencapsulation, mechanical testing, and examination.	
Materials-Fabrication Facilities	4
Facilities for fabrication, joining, and bonding of experimental materials	
Mechanical-Properties Test Facilities	4
Facilities for baseline mechanical-properties testing of structural and nonstructural materials	
Environmental Test Facilities	5
Facilities for determining the effects of various chemical environments on the properties of materials	

reactor is one of the important materials issues. Candidate alloys currently being considered for each subelement are listed in Table 5.13. The materials program is currently dominated by the first-wall/ blanket-structure alloy-development program, not only because of its importance, but also because the requirements are better defined. This subelement is focused primarily on development of low-activation compositions of 9-12% chromium ferritic/martensitic steels and selected vanadium-base alloys, but it also includes further development of conventional steels, such as the ferritic/martensitic alloy HT-9 and the PCA austenitic steel, as well as possible innovative metallic and nonmetallic materials.

The ferritic/martensitic steels are selected primarily because of their resistance to radiation-induced swelling, which could allow for long lifetime. Primary concerns are reduction in tensile strength at higher temperatures (>500°C) and possible low-temperature (25-300°C) embrittlement. The ferromagnetic properties of this steel and the difficult weld requirements are also of concern.

TABLE 5.13 Candidate Structural Alloys

First-Wall/Blanket Structural Alloys**Ferritic and Martensitic Steels**

HT-9, Fe-9Cr-1Mo

Fe-Cr-W, Fe-Cr-V

Vanadium-Base Alloys

V-Cr-Ti

V-Ti-Si

V-Ti

Austenitic Steels

Fe-Cr-Ni-Mo-Ti

Mn-Steels

**Plasma-Interactive-Component
Structural Alloys****Refractory Metal Alloys**

Tantalum Alloys

Vanadium Alloys

Molybdenum Alloys

Copper Alloys

Cu-Be Alloys

Dispersion-Strengthened Alloys**Precipitation-Strengthened Alloys (e.g., AMZIRC)**

Selected vanadium-base alloys are of interest for the FW/B structure because of their superior tensile properties at higher temperatures (up to 750°C) and their resistance to radiation-induced swelling, which may also allow for long lifetime. Dominant concerns relate to chemical reactivity with oxygen and air, cost, and possible radiation-induced embrittlement.

Other materials (referred to as innovative materials), which may include structural ceramics as well as different alloy classes, may be identified as candidate FW/B structural materials as the program progresses. The program plan should provide for the possible development of these types of materials later in the program.

Many aspects of the PIC structural requirements are similar to those for the first wall/blanket. The major differences are the higher heat fluxes associated with the plasma-interactive components and the general perception that the acceptable lifetime may be less than that for the FW/B structure. Because of the limited heat flux capabilities of ferrous alloys, which result from characteristic physical properties,

austenitic and ferritic steels are generally not proposed for the PIC structure. Copper alloys or selected refractory metal alloys (V, Ta, etc.) are currently being considered as candidate materials. In either case, the data base is very limited. Therefore, extensive tests on various alloys will be required to identify primary candidate alloys for development. This effort must be closely coordinated with the nuclear and plasma technology areas.

5.1.1 Technical Issues, Objectives, and Attributes

Technical Issues

The technical issues associated with the structural materials can be categorized in terms of the eight issues listed below:

1. Mechanical performance as it affects operating limits (e.g., temperature, allowable stress, etc.), measured in terms of tensile strength, ductility, fracture toughness, and fatigue/crack growth.
2. Dimensional stability as it affects lifetime, measured in terms of radiation-induced swelling, phase stability, and creep.
3. Thermophysical performance as it affects operating performance/limits and/or safety, including melting/vaporization temperatures, thermal conductivity, and thermal expansion.
4. Thermochemical performance as it affects operating constraints/limits and/or safety, including chemical corrosion/compatibility, liquid-metal and/or hydrogen embrittlement, chemical stability, and tritium inventory.
5. Safety and environmental aspects associated with radioactivity generation and release, nuclear afterheat, waste management, and chemical reactivity.
6. Nuclear performance related to parasitic absorption, nuclear heating, and transmutations that affect chemical composition.
7. Fabrication/joining characteristics related to ease and reliability of welding, fabricability, and reprocessing.
8. Materials costs and resource considerations as they affect the economics and environmental impact in relation to resource availability and processing.

For the candidate structural alloys, the mechanical performance related to loss of ductility or fracture toughness in a radiation environment (including those aspects

associated with synergistic thermochemical effects and welding/joining) is regarded as the most important feasibility issue. The other technical issues primarily affect the economic, safety, and/or environmental attractiveness of fusion by affecting the lifetime, operating limits, inherent safety, waste-management requirements, or cost.

In addition to the radiation-embrittlement issue, dominant issues for the ferritic steels involve development of desirable low-activation compositions with acceptable creep strength, liquid-metal corrosion, weld procedures, and weldment reliability. Other dominant issues for the vanadium-base alloys involve chemical compatibility associated with hydrogen and other nonmetallic-element interactions, processing and fabrication costs, weldment reliability, and tritium inventory. Dominant issues for the candidate refractory metal PIC alloys are similar to those for vanadium-base alloys. Additional issues for copper alloys primarily involve mechanical performance at elevated temperature, effects of alloying and radiation on thermal conductivity, safety and waste management, and properties of weldments. An important aspect of the materials area is the development of low-activation alloys. Detailed aspects of each technical issue are described below.

Mechanical Performance. The mechanical performance of the structural alloys is measured in terms of the tensile strength, residual ductility, fracture toughness, fatigue properties, and creep-rupture properties.

Tensile Strength. The yield strength and ultimate tensile strength are the basic parameters used to define the time-independent allowable stresses in structural design. These properties vary strongly with alloy composition and thermomechanical treatment and are strongly temperature-dependent. For example, cold-working increases the yield strength of Type 316 stainless steel by approximately a factor of four. Also, the creep strength of HT-9 ferritic steel at 600°C is less than one-third that at 500°C. Neutron irradiation may increase or decrease strength properties, depending upon initial microstructural condition and service temperature. Since changes in strength typically tend to saturate with increasing fluence, important information can be obtained at modest fluences.

Ductility/Fracture Toughness. These properties both indicate the capacity of the material to resist crack propagation. Ductility is measured by the uniform and total elongations in a tensile test or by the total elongation before failure in a creep test. These parameters are used to define the allowable strains and to assess response of the structure to off-normal events. Ductility parameters are strongly dependent on alloy composition, impurity levels, and microstructure. For example, the uniform elongation of solution-annealed austenitic steels are typically three to four times the values for 20%-cold-worked material. Irradiation generally leads to a reduction in ductility as a result of radiation hardening or changes in grain boundary properties (resulting from segregation of solid or gaseous transmutants and precipitation of intermediate phases).

Fracture-toughness testing involves the measurement of the stress-intensity factor to initiate fracture and is a direct measurement of the energy necessary to cause fracture. Fracture toughness is strongly dependent on alloy composition, impurity level, and microstructure and tends to be lowered under irradiation conditions that produce substantial hardening. The bcc alloys (e.g., ferritic steels and vanadium) exhibit a ductile-to-brittle transition temperature (DBTT) below which the material behaves in a brittle fashion characterized by a cleavage failure mode. The DBTT is increased by irradiation, with the increase tending to saturate at modest neutron exposures (~10-30 dpa) similar to those producing strength saturation.

Fatigue/Crack Growth. Fatigue properties are concerned with the capacity of a material to sustain repeated cyclic strains in both the elastic and plastic range. At the modest neutron dose levels examined so far (10-30 dpa), fatigue life appears to be more sensitive to the testing environment than to radiation damage.

Rupture Life. Rupture life refers to the time to failure of a material subjected to a constant stress (creep test); it is strongly dependent on stress, temperature, chemical environment, alloy composition, and microstructure. The minimum stress needed to cause creep rupture in a given time is a parameter that is frequently used in the determination of allowable stresses. Irradiation affects rupture life through microstructural changes that alter the creep rate, through grain-boundary segregation, and through the generation of gaseous and solid transmutants, which migrate to and affect the properties of grain boundaries. Rupture life during irradiation can be measured with pressurized tubes.

Dimensional Stability. Dimensional stability is measured in terms of swelling and phase stability and creep properties.

Swelling and Phase Stability. Swelling occurs in all alloys subjected to displacement damage at temperatures around 0.3 to 0.55 T_m , as a result of the bias-driven growth of gas-stabilized vacancy clusters. Void nucleation and growth are strongly temperature- and fluence-dependent, with a weaker dependence on stress and damage rate. Swelling is highly sensitive to alloy composition, gas content, and the dislocation and precipitate microstructure. Swelling-induced volume changes on the order of tens of percent are frequently observed at moderate damage levels (50-100 dpa); however, some alloys, such as HT-9 and V-15Cr-5Ti, show almost no swelling at similar or higher damage levels.

Extensive changes in alloy phase stability can occur during irradiation through the combined effects of solute segregation, enhanced diffusion,

and recoil dissolution. Precipitation phenomena can result in volumetric changes on the order of 1%. The nature and distribution of phases can have a major effect on the magnitude of void swelling and can substantially affect mechanical properties.

Creep. Creep is the phenomenon of plastic flow under constant stress. It is strongly dependent on stress, temperature, alloy composition, and microstructure. At temperatures greater than $0.5 T_m$, creep is dominated by thermal processes. Creep parameters, such as the stress required to produce 1% strain in a given time or to cause the onset of tertiary creep in a given time, are utilized in calculating allowable stresses. Creep phenomena are strongly affected by neutron irradiation. At temperatures lower than $0.4 T_m$, the phenomenon of irradiation creep becomes important and, under certain conditions, can become the major source of dimensional changes. Irradiation creep also plays an important role in the relief of secondary stresses. The dominant irradiation-creep mechanisms involve the stress-oriented nucleation of interstitial loops or the climb-controlled glide of dislocations. Irradiation creep is strongly coupled to void swelling and is, therefore, sensitive to alloy composition and microstructure.

Thermophysical Performance. The thermophysical performance involves those effects and properties associated with melting temperatures, vaporization pressures, thermal conductivity, and thermal expansion. These properties are characteristic of the alloy system and are relatively unaffected by small compositional variations or radiation damage. Safety aspects associated with radioactive release at high temperature and surface heat-flux limits related to thermally induced stresses are strongly influenced by the thermophysical properties.

Thermochemical Performance. The thermochemical performance includes those effects for which compositional changes caused by the chemical environment can have a dominant influence on the structural performance. Specific examples include interstitial-element or liquid-metal embrittlement that is enhanced by radiation hardening. Also, radiation-induced swelling and phase stability can be significantly affected by compositional changes caused by the chemical environment. It is important to distinguish those changes that are dominated by chemically driven phenomena, those that are dominated by displacement damage, and those chemical effects that are simply enhanced by the radiation. The nature of this issue is further complicated by compositional changes that result in part from radiation-induced transmutations and in part from the chemical environment.

Safety/Environmental Performance. Safety and environmental aspects related to the first-wall/blanket structural alloys include radioactivity/ afterheat, waste-management, and radioactivity-release characteristics. Because most of the important materials properties related to these issues are determined by materials selection, these

issues provide the incentive for development of special alloy compositions with low activation advantages. Radioisotopes with half-lives longer than five years are of concern in determining the long-term waste disposal of an irradiated reactor component. Achieving only waste that meets "Class C" or better classification under current regulations mainly requires control of Cu, Mo, N, Nb, and Ni.

Nuclear Performance. The dominant concern associated with nuclear properties of the structural materials involves transmutations following neutron capture, which change the alloy compositions. These transmutations can result in changes in the properties of alloys, and in some cases they may affect performance. The most important transmutation in most cases is the production of helium in (n, α) reactions, since the helium can affect swelling and mechanical properties. The effects of other transmutation products are less well documented, but it can be anticipated that changes in the alloy content of some critical elements may affect such properties as phase stability, chemical compatibility, and other physical or mechanical properties. In the special case of copper alloys used because of their high conductivity, the production of Ni and Zn will progressively reduce the conductivity as time in service increases. Transmutation reaction yields and projected effects on properties must be considered in the selection of material compositions.

Fabrication and Joining. Structural alloys must be readily fabricated and reliably joined. A major factor in the elimination of molybdenum as a candidate structural alloy is the difficulty associated with welding. Difficulties associated with welding are considered important technical issues for both ferritic/martensitic steels and vanadium-base alloys. The ferritic/martensitic steels require high-temperature postweld heat treatments to obtain reliable weldments, whereas vanadium-base alloys require controlled environments to avoid contamination during welding.

Materials Costs and Resources. Materials cost and resource considerations can substantially affect the economics of fusion. Costs of the base metal and certain alloying elements must be considered. The viability of tantalum alloys for PIC applications and vanadium alloys for FW/B structures may depend on materials and processing costs.

Objectives and Attributes

The objective of the structural-materials issue is the development of improved materials that will provide safe, environmentally attractive, and economically acceptable performance characteristics in a fusion reactor. This objective can be achieved by resolving the following four secondary objectives:

- Optimize the materials performance, through alloy development, for the projected operating conditions of a commercial power reactor.

- Maximize the capability for predicting materials performance under relevant operating and possible accident conditions.
- Maximize the safety and minimize the environmental impact through selection and development of materials.
- Minimize cost and resource impact through proper selection and development of improved materials.

Figure 5.2 indicates the second-level objectives and attributes by which attainment of the objectives is measured. The materials performance is optimized for the specific structural application through an iterative development process of testing, with subsequent compositional and/or microstructural modifications. Maximizing the capability to predict materials performance under projected operating conditions will provide for improved component performance and more reliable operation. An important aspect is the selection of materials that will maximize safety during reactor operation and minimize waste-management requirements.

5.1.2 Program Logic

Figure 5.4 presents the Level 2 logic diagram for the structural materials development program, which includes two subelements — first-wall/blanket structure and plasma-interactive-component structure. Table 5.14 provides additional explanation of the logic diagram. Included in this subelement is the design and construction of a major facility: a high-flux, 14-MeV-neutron materials-irradiation test facility, which is needed to verify materials performance characteristics at high fluences where significant nuclear transmutations occur. The near-term strategy for irradiation testing utilizes fast fission-reactor and charged-particle irradiations to evaluate high damage effects and mixed-spectrum reactors to obtain data with simulated helium-transmutation effects in alloys that contain nickel. Tritium-decay schemes (to He) with fission-reactor irradiations and dual-ion charged-particle irradiations are also used to evaluate helium effects. Results obtained in early phases of the program are used to develop improved alloys for further testing and development.

First-Wall/Blanket Alloys

Figure 5.5 presents a more detailed logic diagram for development of the first-wall/blanket structural alloys; Table 5.15 provides supporting information. The dominant issue in this plan is the development of new alloy compositions that provide for reduced long-term radioactivity. The strategy for development of new alloys involves the following two steps: (1) determination of operating feasibility by defining allowable operating parameters based on short-term operation, and (2) optimization of lifetime performance in the unique environment of an operating fusion reactor. The first step requires development of baseline, thermochemical, and low-to-moderate-fluence irradiation data. The second step requires generation of high-fluence irradiation data, which will necessitate 14-MeV-neutron irradiation to verify lifetime.

FIGURE 5.4 Level 2 Logic Diagram for Structural Materials

TABLE 5.14 Objectives for Structural Materials Development (Level 2)

Materials Type	Late 1980s	Early 1990s	Late 1990s	Early 2000s
PW/B Structure				
Reduced-Activation Ferritic Steels and Vanadium Alloys	Develop scoping baseline data and fabrication requirements for selected low-activation alloys.	Develop reference baseline and fabrication data on primary candidate alloys.		
	Evaluate effects of chemical and microstructural variations on properties of selected low-activation alloys.	Determine effects of chemical and thermo-mechanical treatments (TMT) on properties of primary candidate alloys.		
	Determine effects of low fission neutron fluences on embrittlement of selected low-activation alloys.	Determine moderate-fluence radiation on properties (including simulated helium effects).	Develop high-fluence fission neutron data on alloys with improved compositions/microstructures.	Develop high-fluence (fission) mechanical-property data on reference alloys.
	Develop simulation techniques for evaluating helium effects.	Provide helium and damage correlations.	Evaluate helium effects on primary candidate alloys.	
Conventional Alloys (HT-9, PCA)	Develop improved composition and microstructures of conventional alloys.	Design high-fluence 14-MeV materials test facility.	Construct high-fluence 14-MeV materials test facility.	Test selected reference alloys in 14-MeV materials test facility.
	Evaluate effects of moderate-fluence neutron irradiations on properties of conventional alloys.	Evaluate helium effects at high fluence (100 dpa) on conventional alloys.	Assess performance to 100 dpa (with helium), refining alloy design in iterative process.	
Innovative Materials		Develop scoping baseline and radiation data on selected innovative materials.	Develop moderate-fluence data on selected innovative materials.	
PIC Structure	Develop scoping baseline data on selected copper and refractory metal alloys.	Develop baseline data on selected refractory metal alloys.	Develop moderate-fluence data with simulated transmutation effects on improved alloys.	
	Develop low-fluence scoping data on copper and refractory metal alloys.	Develop moderate-fluence radiation data on selected copper alloys.	Select primary candidate alloys for 14-MeV tests.	Test selected reference alloys in 14-MeV materials test facility.

The program on reduced-activation materials includes four areas of R&D: baseline properties, thermochemical performance, radiation effects, and fundamental radiation studies. The baseline properties are used to scope the alloy systems. The thermochemical performance involves evaluation of changes in properties that result from unique chemical environments. Radiation effects on material performance are determined primarily from fast-breeder and mixed-spectrum reactor irradiations. Since neither fission reactor adequately simulates the fusion neutron spectrum, theoretical investigations, charged-particle irradiations, and fundamental radiation studies are conducted to correlate fission-reactor data and to predict materials performance in the fusion environment. The dominant issue involves differences in nuclear transmutation rates, particularly for such gaseous species as helium and hydrogen.

In Phase I of the plan (first five years), data are generated on selected ferritic steels and vanadium alloys. Evaluation of these data and additional data from tests on conventional alloys will lead to the identification of primary candidate alloys (compositions and microstructures) for investigation during Phase II (~ 1990-95). Reference baseline and chemical-performance data and moderate-fluence radiation data on primary candidate alloys are generated during Phase II. Fundamental radiation studies provide additional data on helium effects, damage correlations, and lifetime potential that are used to select improved alloy compositions and microstructures for high-fluence fission-reactor testing in Phase III. Reference alloys are selected for testing during Phase IV in the 14-MeV-neutron materials-irradiation test facility that was designed and constructed during Phases II and III. The operating-performance limits for a limited number of reference alloys will be defined during Phase IV. Radiation lifetimes to ~100 dpa in the 14-MeV-neutron environment will be achieved by the end of Phase IV. This dose is necessary to provide reasonable predictions of materials performance at the minimum lifetimes considered for fusion applications. Additional testing will be required to verify the desired higher-radiation lifetimes that may be attainable.

Limited effort on development of conventional alloys for near-term applications (e.g., HT-9 ferritic steel and PCA austenitic steel) will continue. The larger data base that exists for these alloys, particularly high-fluence data, will provide valuable input to the development of low-activation alloys. The primary thrust of this effort is on higher-fluence irradiation data, particularly with simulated helium effects. As indicated in Fig. 5.4, incorporation of innovative materials is accommodated later in the program.

Plasma-Interactive-Component Alloys

Development of PIC structural alloys will generally be similar to that for the FW/B alloys; however, development will lag behind that of the FW/B alloys by a few years. This later development results primarily because of the less-well-defined requirements for this application, due to uncertainties in plasma-surface-related issues. Scoping data on the two classes of alloys being considered -- copper and refractory metal alloys -- will be generated during the latter part of Phase I and early in Phase II. Midway through Phase II, on the basis of the scoping data and input from the plasma technology program, candidate alloys will be selected for further study. Sufficient baseline and radiation-effects data will be generated during Phase III to select primary candidate

TABLE 5.15 First-Wall/Blanket Structural Materials (Level 3)

**Reduced-Activation Alloys
(Ferritic Steels and Vanadium Alloys)****Baseline Properties**

Determine effects of fabrication and weld procedures and thermomechanical treatment on baseline properties, particularly DBTT, including effects on postweld heat treatment.

Determine baseline mechanical performance -- in particular, DBTT and creep properties.

Thermochemical Performance

Determine effects of chemical environment, including interstitial-element interactions (such as H, C, N, and O) and thermal aging effects, on materials performance.

Radiation Effects

Determine swelling characteristics of fission-neutron irradiated (FBR/HFIR) alloys as a function of composition and TMT.

Determine effects of low-fluence (< 30 dpa), low-temperature ($< 400^{\circ}\text{C}$) fission-neutron irradiation on embrittlement. Later, evaluate high-fluence irradiation hardening.

In the longer term, evaluate radiation-creep characteristics.

Fundamental Radiation Studies

Develop helium-simulation techniques and displacement-damage correlations for fission-reactor irradiations.

Evaluate potential for long lifetime by high-fluence FBR and charged-particle irradiations.

14-MeV-Neutron Irradiation Test Facility

Design and construct a high-fluence, 14-MeV-neutron facility for materials testing.

TABLE 5.15 (Cont'd)

Conventional Alloys (HT-9, PCA)**Radiation Effects**

Use expanded radiation data base to develop better understanding of radiation effects on materials and for more rapid development of new low-activation alloys.

Baseline/Chemical Effects

Develop improved composition/microstructures as reference for developing improved low-activation alloys.

alloys for high-fluence testing during Phase IV. Selected reference alloys will be tested in the 14-MeV-neutron test facility.

5.2 NONSTRUCTURAL BLANKET MATERIALS

Nonstructural materials that are exposed to the combined radiation, chemical, thermal, mechanical, and electromagnetic environments characteristic of the blanket region of an operating fusion reactor present important materials issues. The nonstructural-blanket-materials program element consists of four subelements:

- Solid-tritium-breeder materials
- Liquid breeders/coolants
- Multiplier/moderator materials
- Ceramic insulators

Proposed DT fusion-reactor designs will require a tritium-breeding blanket to ensure an adequate supply of tritium. Lithium and lithium-containing ceramics and alloys are being considered. In addition, neutron/energy multipliers or moderators may be required to enhance the nuclear performance of the blanket, and ceramic insulators may be required for adequate performance of certain components. Each class of materials presents a distinct set of issues that must be addressed in order to evaluate the potential of each material. Candidate materials currently being considered for each subelement are listed in Table 5.16.

TABLE 5.16 Candidate Nonstructural Blanket Materials

Solid-Breeder Materials	Liquid-Breeder/Coolant Materials	Multiplier/Moderator Materials	Ceramic Insulators
Li ₂ O	Li	Be	MgAl ₂ O ₄
LiAlO ₂	Pb-17% Li	Pb	CaO
Li ₂ SiO ₃		Pb-compounds	MgO
Li ₄ SiO ₄		--	Si ₃ N ₄
Li ₂ ZrO ₃		Graphite	SiC
		SiC	
		CaO	

The effort in this program element is currently dominated by solid-breeder and ceramic-insulator development. Corrosion/compatibility issues, which are dominant issues for the liquid breeders and coolants, are included in the structural materials section and in the nuclear technology section. The multiplier/moderator development is focused predominantly on beryllium.

5.2.1 Technical Issues, Objectives, and Attributes

Technical Issues

The technical issues associated with the nonstructural blanket materials can be categorized in terms of the same eight generic issues that were given for the structural alloys. These include the following:

- Mechanical performance
- Dimensional stability
- Thermophysical performance
- Thermochemical performance
- Safety and environmental characteristics
- Nuclear performance

- Preparation/fabrication issues
- Materials cost and resources

However, the relative importance of these issues, which varies between the different subelements, also differs from that for the structural materials.

Solid-Tritium-Breeder Materials. The main feasibility issue associated with solid-breeder blankets involves acceptable tritium release and recovery. The tritium inventory, which is dependent on the tritium release/recovery characteristics, is both a safety and an economic issue for solid breeders. The operating constraints imposed by tritium release/recovery considerations are affected by all eight technical issues listed above. In most cases, close interrelationships exist between two or more of these technical issues. For example, interconnected porosity of the ceramic structure, which tends to enhance the tritium release, leads to reductions in thermal conductivity of the solid, and hence, to higher temperature gradients in the solid and larger thermal stresses.

The tritium inventory is strongly influenced by the thermochemical performance of the ceramic. The thermochemical performance depends on such properties as tritium diffusivity, surface desorption of tritium, radiation-induced and/or gas trapping of tritium, and chemical stability of the ceramic. Selection and development of optimized materials will require an understanding of the relative importance and parametric dependence (e.g., temperature dependence) of these properties.

Preparation and fabrication, as applied to the breeder material alone, are feasibility and economic issues. Aspects involved in preparation and fabrication include (1) the development and implementation of the chemical processes needed to produce the required quantities of the particular chemical compound with the required purity and (2) the development and implementation of processes for conversion of this material to the desired forms (e.g., powders, spheres, rods, plates, hollow cylinders, etc.). This area comes early in the overall development process, but subsequently it has to receive feedback from the thermochemical and mechanical performance areas for information about needed chemical and physical changes required to optimize overall performance.

The nuclear performance is measured in terms of tritium breeding and other nuclear transmutations, nuclear heating, and parasitic absorption. These aspects are dominated primarily by materials selection and control of certain impurities. The safety and environmental aspects of solid breeders, which are dominated by nuclear characteristics, include radioactivity release, waste management, nuclear afterheat, and chemical reactivity.

The thermophysical performance primarily involves thermal-expansion properties and the effects of radiation on thermal conductivity. The thermal conductivity of many ceramics is reduced significantly (more than a factor of two) by modest neutron fluences (equivalent to a few $\text{MW}\cdot\text{yr}/\text{m}^2$). The conductivity of ceramic breeders is also dependent on microstructure and porosity.

The mechanical performance issue deals with the retained strength and form of the breeder material under operating conditions. Such properties as resistance to fracture, tensile strength, ductility, creep, and swelling all come into play and are influenced by radiation effects. Mechanical performance aspects involve how well the integrity of the breeder is maintained in the optimized configuration and how changes will affect its structural containment. Also (as mentioned under tritium inventory), grain size bears on diffusive inventory, so that a variable that affects mechanical performance will also be related to tritium inventory. Economics of construction and maintenance are affected by mechanical performance. If deleterious effects are severe and sudden, safety becomes an issue.

Liquid Breeders/Coolants. For the liquid metals and alloys, molten salts, and organic coolants, the relevant issues include thermal stability, safety, and nuclear response. Again, materials that have low vapor pressure, low resistance to decomposition, minimal change in viscosity, etc. are preferred. Characteristically, trade-offs must be made among these properties when selecting a breeder material. Many blanket designs use water as the coolant. Because liquid metals are excellent thermal conductors, they are generally considered to function as both coolant and breeder.

Radiation is not believed to affect the performance of liquid metals significantly. However, radiolytic decomposition of molten salts may be important, particularly as it affects corrosion, while radiolytic decomposition is generally regarded as the most important feasibility issue for organic coolants. Chemical reactivity and tritium solubility and recovery are also important issues.

Ceramic Insulators. An important issue for fusion is whether ceramic insulators can be developed that will perform satisfactorily in a fusion-reactor environment and that will exhibit lifetimes commensurate with the specified applications. Applications for ceramic insulators can generally be divided into two broad categories: (1) insulators for high-flux, high-temperature regions, such as the blanket and plasma-heating systems; and (2) insulators for lower-temperature, lower-flux applications, such as rf windows and magnet applications. Ceramic insulators may be in the form of bulk material or as coatings on structural materials. The technical issues associated with ceramic insulators involve mechanical performance, dimensional stability, thermal performance, electrical performance, thermochemical performance, and fabrication and joining.

The two dominant issues related to mechanical performance of ceramic insulators are strength and thermal-shock resistance. Development of high-toughness ceramics with acceptable electrical performance characteristics appears essential. Excessive degradation of fracture strength and thermal-shock resistance in a radiation environment is a feasibility issue. The data base for conditions of interest is very limited. The problem is further exacerbated by limitations imposed by low activation and safety considerations on materials selection.

Radiation-induced swelling and creep may lead to excessive deformation, which -- when associated with the brittle characteristic of ceramics -- can result in limited lifetimes. Differential swelling caused by larger temperature and/or microstructural variations could lead to premature failure. The lifetime of the insulators should be commensurate with that of the component in which it is used.

Radiation affects both the thermal and chemical properties of ceramics. The thermal conductivities of most ceramics vary strongly with temperature and are significantly reduced by radiation. For high-heat-flux regions, reductions in thermal conductivity by radiation can lead to higher temperatures, which in turn may further reduce the conductivity, and hence, the performance limits. Radiation-induced transmutations can lead to changes in stoichiometry that can affect properties and performance. The extent to which these effects occur must be established to define lifetimes in reactors.

The primary purpose of ceramic insulators is to provide regions of low electrical conductivity in certain components. Radiation can affect the electrical resistivity, dielectric strength, and rf transmissivity, all of which will affect insulator performance. The extent to which these properties are degraded is a feasibility issue for certain applications.

Fabrication and joining are considered to be feasibility issues for most ceramic-insulator applications. Development of optimized microstructures to obtain desired properties, high-integrity joining techniques, and possibly coating techniques will be required to provide acceptable electrical insulators.

Neutron Multipliers/Moderators. Beryllium is the leading candidate for neutron multiplier applications, while graphite and silicon carbide are candidates for neutron moderators or reflectors. In most cases, these materials will be contained in a structure; therefore, structural integrity is not of primary importance. The two dominant technical issues for a beryllium neutron multiplier involve (1) radiation-induced swelling, caused primarily by the high helium-generation rates; and (2) thermochemical performance associated with material compatibility. Development of microstructures that minimize swelling and enhance gas release are required. Surface oxidation and compatibility with other blanket materials are important issues.

The same two issues are of primary importance for graphite and SiC moderators/reflectors. In this case, the thermochemical performance associated with tritium trapping and retention is important. Tritium transport into graphite and SiC is predicted from thermodynamic considerations. However, the rate will be controlled by kinetic considerations, which include radiation-enhanced diffusion and defect trapping. Radiation-induced swelling at high fluences is also a concern. The need for neutron multipliers and moderators will depend on the blanket concept.

Objectives and Attributes

The objective of the nonstructural-blanket-materials issue is the development of improved materials that will provide safe, environmentally attractive, and economically competitive performance characteristics in a fusion reactor. This objective can be achieved by resolving the following four secondary objectives:

- Optimize the materials performance by developing improved materials for specific functions.
- Maximize the capability for predicting materials performance under relevant operating and possible accident conditions.
- Maximize the safety and minimize the environmental impact through selection and development of improved materials.
- Minimize cost and resource impact through proper selection and development of materials.

Figure 5.2 indicates the second-level objectives and attributes by which the attainment of the objectives is measured. The materials performance is optimized for the specific nonstructural blanket application through an iterative process of testing, with subsequent compositional and/or microstructural modifications. Maximizing the capability to predict materials performance under a range of projected operating conditions will provide for improved component performance and more reliable operation. An important aspect of this objective is the selection of materials that will maximize safety during reactor operation and minimize waste-management requirements.

5.2.2 Program Logic

Figure 5.6 presents the Level 2 logic diagram for development of non-structural blanket materials; Table 5.17 provides additional information. This program element includes four types of materials: solid breeders, liquid breeders/coolants, multiplier/moderator materials, and ceramic insulators. The overall strategy for this program element is similar to that for the structural alloys -- feasibility testing and selection of primary candidate materials from baseline, thermochemical, and low-fluence irradiation data and lifetime evaluation based primarily on high-fluence irradiation testing.

Solid Breeders. The solid-breeder-development program, which is focused on Li_2O , LiAlO_2 , Li_4SiO_4 , etc., includes generation of baseline data, evaluation of thermochemical performance, and evaluation of radiation performance. Scoping data are generated on selected materials in all three areas during Phase I. Improved compositions and microstructures are selected for more detailed evaluation during Phase II. Primary candidate materials are selected for high-fluence testing during Phase III. The lifetime is evaluated for reference materials during Phase IV.

FIGURE 5.6 Level 2 Logic Diagram for Nonstructural Blanket Materials

TABLE 5.17 Nonstructural Blanket Materials

Materials Type	Late 1980s	Early 1990s	Late 1990s	Early 2000s
Solid Breeders	Develop baseline properties data and fabrication methods.	Develop reference data and fabrication methods.	Evaluate commercial fabrication methods.	
	Collect thermochemical and transport data for candidate materials.	Evaluate effects of oxidizing environment on material properties.	Complete properties data on prime materials.	
	Determine effects of low-fluence irradiation on materials behavior.	Develop moderate-fluence data on selected materials.	Develop high-fluence data on prime candidate materials.	Perform dynamic in-situ tritium recovery test of prime material in relevant neutron energy environment.
Liquid Breeders/ Coolants	Develop baseline physical properties data.	Evaluate performance of primary candidate materials.		
	Develop baseline thermochemical properties data.			
Multiplier/ Moderator Materials	Establish fabrication requirements.	Develop improved microstructures and fabrication requirements.	Evaluate commercial fabrication methods, including recycling of beryllium.	
	Evaluate thermochemical performance.	Determine tritium thermodynamic and transport properties.	Develop high-fluence data on primary candidates.	Qualify materials for fusion applications.
	Evaluate effects of low-fluence irradiation requirements.	Develop moderate-fluence data on selected materials.		
Ceramic Insulators	Establish baseline properties and fabrication requirements.	Develop improved forms of candidate materials.	Evaluate selected materials in detailed test matrix.	Qualify materials for fusion applications.
	Collect baseline electrical- and mechanical-properties data.	Develop optimized microstructures.		Test selected reference materials in 4-MeV neutron facility.
	Evaluate effects of low-fluence irradiation on properties data.	Evaluate moderate-fluence irradiation effects on materials properties.	Develop high-fluence data on primary candidates.	

Liquid Breeders/Coolants. Only limited testing of liquid breeders and coolants is projected at this time. Fundamental baseline data are required in the near term to evaluate the performance of certain candidates, and hence, the feasibility of selected blanket concepts. In general, liquid metals are not significantly affected by radiation; therefore, extensive radiation testing is not identified as a high priority for liquid metals.

Multiplier/Moderator Materials. Multiplier/moderator materials currently identified as of primary interest include beryllium and silicon carbide. Since beryllium appears essential to the feasibility of several blanket concepts, evaluation of its performance characteristics in the near term is given a high priority. This subelement must be closely coordinated with the solid-breeder materials program and the blanket technology program.

Ceramic Insulators. The strategy for the ceramic insulator development is also similar to that for the structural alloys. Although the operating requirements for ceramic insulators are less well defined, it appears that use of insulators in high-flux regions will be necessary. Acceptable performance is generally considered to be a feasibility issue. The strategy involves generation of scoping baseline and low-fluence radiation effects data in the early phases, improving compositions and microstructure during Phases II and III, and finally lifetime testing during Phase IV.

5.3 SPECIAL MATERIALS

Materials that must serve special functions in an operating fusion reactor are included in this program element. The special-materials program element consists of three subelements:

- Plasma-facing materials
- Coatings and claddings
- Advanced magnet materials

The effort in this area is dominated by the plasma-facing materials. Plasma-facing materials include low-Z materials, such as beryllium and graphite, and selected high-Z, low-sputtering materials, such as tungsten and tantalum. Composites of low-Z and high-Z materials that incorporate the favorable characteristics of both may also be considered. Although these materials will be selected primarily for their response to the edge plasma, they must also exhibit adequate mechanical integrity, radiation-damage resistance, and chemical stability. The plasma-materials interaction aspects of the plasma-facing materials are also included in the plasma technology area. Other engineering aspects are included in the nuclear technology area.

Coatings and claddings may be used in several applications, such as a tritium barrier for the first wall or blanket structure or as an electrically insulating coating on

the blanket structure to reduce MHD effects. In general, the requirements for these applications are not well characterized, compared with those for the structural-materials applications.

Magnet materials include the conductors, insulators, and structures used in the magnets. The two major categories of magnet materials are materials for normal conducting magnets and materials for superconducting magnets. Normal magnets being considered for fusion make use of high-conductivity copper alloys, which act as both the conductor and structure, and insulators, which can be organic or inorganic. Normal magnets are often designed to withstand much higher levels of radiation damage than superconducting magnets. Only advanced superconducting magnet materials are included in this subelement. Conventional superconducting magnet materials, such as NbTi and Nb₃Sn, are included in the plasma technology area.

5.3.1 Technical Issues, Objectives, and Attributes

Technical Issues

The most important issues for the plasma-facing and coating/cladding materials are identification of the most promising candidates, the establishment of baseline properties, and determination of the response of the materials in the fusion environment. In particular, these materials will be exposed to a high 14-MeV neutron fluence, so they must have a resistance to radiation damage comparable with that of other blanket materials. Neutron-radiation-effects data for many candidate materials are almost nonexistent. The special materials also include a unique compatibility issue, since they are usually bonded to other materials. Therefore, it is important to test the bonded structures and not just the individual materials.

The major issue for normal magnet materials involves the effects of radiation damage on the electrical and mechanical properties. The copper alloys could experience a compositional change due to nuclear transmutations that would increase the electrical resistivity. The electrical insulating properties of organic materials are adversely affected by radiation. Inorganic insulators are believed to be capable of withstanding higher neutron fluences. The major issue for inorganic materials is their long-term mechanical integrity.

Radiation damage is also the primary concern for superconducting magnets. Peak fields and currents in the superconductor will be adversely affected by radiation, and the electrical and mechanical properties of organic insulators may be degraded. The test needs are different for the superconducting magnet insulators, because of the ~4 K operating temperature; room-temperature irradiation tests conducted for normal conducting insulators are not suitable for superconducting applications. Finally, there are special fabrication issues for superconducting magnets. The use of the superconductor is often limited by its fabricability.

Objectives and Attributes

The objective of the nonstructural-blanket-materials issue is the development of improved materials that will provide safe, environmentally attractive, and economically competitive performance characteristics in a fusion reactor. This objective can be achieved by resolving the following secondary objectives:

- Optimize the materials performance by developing improved materials for specific functions.
- Maximize the capability for predicting materials performance under relevant operating and possible accident conditions.
- Maximize the safety and minimize the environmental impact through selection and development of improved materials.
- Minimize cost and resource impact through proper selection and development of materials.

Figure 5.2 indicates the second-level objectives and attributes by which the attainment of the objectives is measured. The materials performance is optimized for the specific nonstructural blanket application through an iterative process of testing with subsequent compositional and/or microstructural modifications. Maximizing the capability to predict materials performance under a range of projected operating conditions will provide for improved component performance and more reliable operation. An important aspect of the objective is the selection of materials that will maximize safety during reactor operation and minimize waste-management requirements.

5.3.2 Program Logic

Figure 5.7 presents the Level 2 logic diagram for the special-materials program element. Materials functions that are not part of the first two program elements are included in this program element (viz., plasma-facing materials, coatings and claddings, and magnet materials). Since the requirements for these materials currently are not well-defined, effort in this area must be closely coordinated with the other plasma and technology areas.

The strategy for this program element involves development of data in the early phases of the program for use in evaluating the feasibility of various concepts, followed by development of reference data later in the program to provide a basis for establishing operating and lifetime limits. The plasma-facing materials will most likely include low-Z materials, such as beryllium and graphite, and/or high-Z composites of tungsten or other refractory metals. Related information for near-term physics devices will be provided by the plasma technology area. The materials area will emphasize development of a data base on bulk-materials properties required for reactor applications.

Coatings and/or claddings may be needed for tritium barriers, material compatibility, etc. Oxide films are potential candidates. Materials development in this

FIGURE 5.7 Level 2 Logic Diagram for Special Materials

area must be closely coordinated with the nuclear technology area to define needs and requirements.

Advanced magnet materials, such as NbN, are included in this program element. Conventional normal and superconducting magnet development is conducted as part of the plasma technology area.

In general, the early phases of the program will provide baseline and low-fluence irradiation data. Primary candidate materials for various applications will be identified in Phase II on the basis of nonirradiated baseline data and moderate-fluence fission-reactor irradiation data. Reference materials will be selected early in Phase IV and, where appropriate, 14-MeV-neutron irradiation data will be provided to verify operating and lifetime limits.

OVERVIEW

The systems design and analysis area supports the major program evaluation and decision points and guides fusion research and development toward practical products. The objectives of the activities in this area are (1) to ensure the development of practical fusion applications; (2) to complete preconceptual designs of major fusion facilities; (3) to analyze critical issues and optimize development paths; (4) to identify and implement necessary safety, environmental, and licensing features for fusion development; (5) to plan and execute necessary research and development for remote technology equipment, and (6) to evaluate the potential of alternative (non-DT) fuel cycles. The major systems activities are characterized in Fig. 6.1.

Systems design and analysis activities provide the fusion program with important tools, data, and perspective. Activities include the identification and resolution of critical issues that involve the interaction of plasma physics and technology, the maintenance of an engineering data base, and the setting of subsystem objectives based on identification of desired economic and safety/environmental characteristics of commercial fusion applications.

A recent accomplishment of these activities is the preconceptual design of the proposed short-pulse ignition experiment. The integrated physics and engineering effort involved in that high-field design showed that a low-cost, short-pulse ignition experiment was possible. This conclusion had not been widely accepted by the fusion community a few years earlier. The systems activities have also provided a key basis for international collaboration through the International Tokamak Reactor (INTOR) program. Guidance from systems studies also has had a major impact on programmatic directions in such areas as steady-state current drive and high-beta tokamak operation. Design studies of reactors based on advanced fusion concepts have also guided research programs for these concepts.

Commercial fusion facility designs are important, particularly at this early stage of fusion development, to help identify necessary R&D program goals. The design studies

FIGURE 6.1 Characterization of Tasks for Systems Design and Analysis

are essential for guiding fusion R&D, and they provide a focus for the fusion program -- namely, the development of useful products. The designs highlight the importance of good safety and environmental features, combined with acceptable costs.

Through design studies, system requirements are identified and the research and development needed for the fruition of fusion applications is forecast. For any given confinement scheme, the design activities ensure that all subsystems can be integrated within the constraints imposed by materials, technology, and physics to produce a system that is economically attractive and technologically feasible, while simultaneously maximizing safety and minimizing environmental effects. Depending on the confinement scheme being considered, these studies range from simple scoping analyses to detailed, multiyear preconceptual designs using sophisticated models.

Systems studies will provide important programmatic guidance in the preconceptual design of the device to produce a long-burn demonstration. The long-burn demonstration links broad national and international interests in fusion development. There is a spectrum of possibilities for long-pulse ignited devices, with a substantial variation in cost and technology requirements. The fusion program must find the most attractive design concept.

Another important function of systems activities is to search for development paths that have test-facility requirements that minimize the cost and risk of fusion development and compress the schedule. Test facilities should be capable of relatively rapid construction and very reliable operation. This is a very difficult issue, requiring contributions from people who have special expertise in global systems analysis.

Systems design and analysis covers a broad array of conceptual studies and facility designs, defines and maintains a listing of subsystems and component objectives for commercial and integrated test fusion reactors, relates these objectives to the objectives of specific science and technology programs, and assists in the optimization of program-implementation strategies. Systems design and analysis also treats the programs required (1) for developing remote technology equipment and (2) for developing fusion concepts based upon non-DT fuel cycles.

The systems design and analysis area includes the following program elements:

- Applications, Economics, and Technology Transfer
- Fusion Test Facilities, Critical Issues, and Development Pathways
- Safety, Environment, and Licensing
- Remote Technology
- Alternative Fuel Cycles

A further breakdown of these program elements (discussed below) into subelements is given in Table 6.1. The Level 1 logic diagram is shown in Fig. 6.2.

TABLE 6.1 Systems Design and Analysis Program Elements and Subelements

Program Elements	Subelements
Applications, Economics, and Technology Transfer	Commercial-Reactor Preconceptual Design Applications Studies Economics Analysis Availability Analysis Technology-Transfer Studies
Fusion Test Facilities, Critical Issues, and Development Pathways	Fusion Test Facilities Preconceptual Design Critical-Issues Analysis Engineering-Data-Base Assessment Development-Pathways Analysis
Safety, Environment, and Licensing	Safety Environment Licensing
Remote Technology	Program Plan Concepts Equipment Development Applications
Alternative Fuel Cycles	Confinement Systems and Burning Plasmas Plasma Technology Nuclear Technology and Materials Systems Design and Analysis

Applications, Economics, and Technology Transfer. This program element includes activities in commercial-reactor preconceptual design, development of a range of fusion applications, development and application of methods to analyze the economic potential of fusion applications, and studies of factors affecting availability of facilities. It also provides for studies to identify the appropriate roles and timing for industrial participation in fusion R&D activities and the process of transferring the technology to industry. The Level 1 milestones for this element are to use the methods and data from these studies in reaching major program decisions.

Fusion Test Facilities, Critical Issues, and Development Pathways. This program element includes, in part, preconceptual designs of fusion test facilities, such as the short-pulse ignition experiment, the long-burn demonstration, and the integrated fusion facility. Milestones are established to provide the data to support those program decisions (see Fig. 6.2). This program element also provides for ongoing analysis of critical issues, including the assessment of systems issues arising from physics/technology interfaces identified in design studies. This element also provides for

FIGURE 6.2 Level 1 Logic Diagram for Systems Design and Analysis

establishment and maintenance of an engineering data base for component fabrication and design standards and includes development-pathways analysis, to develop and apply methodologies for estimating the time, risk, and cost impact of alternative technical options for fusion power development.

Safety, Environment, and Licensing. This program element is focused on the identification of critical fusion safety and environmental issues and on providing (1) experimentally verified methodologies for analysis, assessment, and resolution of these issues; (2) a technical basis for safety and environmental improvements in fusion reactor designs; and (3) a technical foundation and recommended strategies for licensing of commercial fusion reactors. The major milestones are timed to provide information to be used in making major program decisions.

Remote Technology. This program element includes activities aimed at developing the necessary design inputs, equipment, and procedures to support availability goals for a sequence of more ambitious test facilities, leading eventually to commercial plants. While substantial advances in remote technology can be anticipated independently of the fusion program, many aspects will be unique to fusion. The major milestones in this area are to provide the necessary remote-technology readiness needed for the decisions to build major fusion facilities (such as the short-pulse ignition experiment, the long-burn demonstration, and the integrated fusion facility) and to provide special-purpose equipment for these facilities.

Alternative Fuel Cycles. This program element includes the analysis of the potential of fusion fuel cycles other than the primary deuterium-tritium (DT) option. Operation with a fuel cycle other than the DT one could potentially reduce significantly the constraints on fusion-reactor design by eliminating the requirement for a tritium-breeding blanket. However, substantially improved values of beta and density-confinement-time product are necessary, compared with those required for DT operation. In addition, devices of larger size, stronger magnetic field, or both may be required. A variety of R&D activities important to the assessment of alternative fuel cycles is required. The results of these activities are not projected to influence a Level 0 major program decision until the late 1990s.

6.1 APPLICATIONS, ECONOMICS, AND TECHNOLOGY TRANSFER

This program element consists of the following five subelements:

- Commercial Facilities Preconceptual Design
- Applications Studies
- Economic Analysis

- Availability Analysis
- Technology-Transfer Studies

6.1.1 Issues, Objectives, and Attributes

The issues associated with the program element for applications, economics, and technology transfer are as follows:

- The full range of potential commercial applications for fusion science and technology must be identified. The most likely application and the one that has received the most study is the generation of electricity; however, a number of other potential applications must be considered.
- Preconceptual designs of potential commercial facilities must be performed now (and updated frequently), so that systems-related issues important to the design of attractive end products and likely to affect near-term R&D programs will be appropriately identified.
- Fusion research and development is carried out primarily in national laboratories and universities at present, but the skills required to commercialize fusion must eventually be based in industry.
- Perspective is required on the potential economics of fusion applications, because fusion must compete with other technologies in the commercial marketplace. Plant availability is a critical factor in economic analysis.
- Data, approaches, and methodologies are needed to establish a basis to achieve acceptable availability in fusion facilities.

The objectives, associated attributes, and proposed planning targets required to resolve the above issues are listed in Table 6.2.

6.1.2 Program Logic

The Level 2 logic diagram for this program element is shown in Fig. 6.3. The program logic is discussed below.

Commercial-Reactor Preconceptual Design. This design effort includes conceptualization of commercial fusion facilities. Design activities are carried out for all fusion confinement concepts. The insights gained from these design activities are used to guide the science and technology programs and to assist with major program evaluations. Studies may be performed at varying levels of detail, as appropriate. For

TABLE 6.2 Objectives and Attributes for Applications, Economics, and Technology Transfer Program Element

Objective	Attribute	Planning Target
Maximize number of fusion applications.	Number of potential applications with competitive economic and safety/environmental features	At least 3
Maximize attractiveness of commercial fusion applications.	Number of designs that meet economic and safety/environmental targets for each application	At least 3
Optimize industrial participation in fusion program.	Preparation of technology-transfer plan	Complete plan.
Develop skill in projecting fusion economics.	Preparation and standardization of economic models	Complete models.
Maximize plant availability.	Development of model and data base to analyze plant availability	Complete model and data base.

example, concepts with no direct experimental basis would be limited to scoping studies to assess their potential; preconceptual designs based on physics scaling and general reactor modeling would be carried out for concepts with some small-scale experimental verification, and concepts having a major experimental base would undergo detailed preconceptual engineering design.

Fusion Applications Studies. These studies are carried out in three areas: (1) continued assessment of the supply, demand, and cost of electricity from fusion; (2) investigation of fissile-fuel and nuclear-materials breeding; and (3) exploration of other nonelectric fusion applications.

The study of fissile-materials production would provide a basis for the technical evaluation of using fusion as a source of neutrons for such applications. An updated evaluation of fast fission hybrids should be made that incorporates the new safety and fuel-cycle ideas developed in recent years as part of the fission-suppressed hybrid designs. These ideas could lead to a significant improvement in the attractiveness of fast fission designs. Low-Q fusion reactors should be included in hybrid-system designs. These evaluations would contribute to the decision on a reference fusion breeder design. In parallel with the design of the reference fusion breeder, fuel cycle and reprocessing

studies and deployment and development studies should be performed. The results of these studies would provide the basis for an evaluation of the technical, economic, safety, and environmental characteristics of the fusion breeder for input into the integrated fusion facility decision and the overall assessment of fusion.

The other potential applications of fusion should be studied to provide a thorough assessment of the capability of fusion to produce a wide variety of products other than electricity and nuclear fuel. Fusion has the potential to produce hydrogen and other nuclear materials and chemical products, to "burn" nuclear and chemical wastes, to produce useful radioisotopes for food preservation and medical applications, to perform radiation processing of materials, and to provide space power and space propulsion. Still other applications may be possible. These studies will provide a preliminary assessment of the many potential applications of fusion and identify potential new ideas. These new ideas will be compared with applications that have already been evaluated. The results of small-scale experiments to verify the nuclear and chemical processes required for some applications will be considered in these studies. This task will culminate in an assessment of the potential of fusion to produce useful products in addition to electricity and fissile fuel. This assessment will produce valuable information for the overall assessment of fusion and for the decision whether to proceed with an integrated fusion facility.

Economics Analysis. This task consists of (1) developing and applying methodologies for estimating the costs of fusion products, (2) maintaining a cost data base, and (3) assessing the impact of development costs on fusion economics.

Availability Analysis. This task consists of (1) developing and applying methodologies to predict the availability of commercial fusion facilities and fusion test facilities, (2) establishing and maintaining a component reliability/maintainability data base, and (3) recommending design and/or operational modifications to improve facility availability.

Technology Transfer. This activity consists of studies to analyze and apply procedures for identifying appropriate industrial roles in fusion R&D activities. Results of these studies will be available as input to major program decisions and for incorporation into agreements on international collaboration.

6.2 FUSION TEST FACILITIES, CRITICAL ISSUES, AND DEVELOPMENT PATHWAYS

This program element consists of the following four subelements:

- Fusion Test Facilities Preconceptual Design
- Critical-Issues Analysis
- Engineering-Data-Base Assessment
- Development-Pathways Analysis

6.2.1 Issues, Objectives, and Attributes

The issues associated with this program element are as follows:

- Timely preconceptual design studies of required fusion test facilities must be available to support major program decisions and discussions on international collaboration. These studies are also critical to development-pathways analysis.
- Many critical technical issues involve the interaction of plasma physics and one or more technology components and, thus, would not necessarily be addressed by the component-development or science programs.
- A continuing formal effort is required to optimize the technical program, because a large number of optional technical pathways exist, each with its associated cost, schedule, and risk.

The objectives and associated attributes required to resolve these issues are listed in Table 6.3.

TABLE 6.3 Objectives and Attributes for Fusion Test Facilities, Critical Issues, and Development-Pathways Program Element

Objective	Attribute	Planning Target
Design fusion test facilities.	Number of designs completed or under way	At least 4
Minimize cost of fusion test facilities.	Capital cost of any individual test facility, expressed as percent of annual magnetic-fusion budget	Less than 20%
Maximize resolution of systems-based critical issues.	Number of design studies formally reviewed and critical issues analyzed	All
Minimize cost, schedule, and risk of fusion-energy development.	Preparation of methodology for performing development-pathways analysis	Complete model.
Maximize excellence of engineering in fusion facilities.	Establishment and maintaining of engineering data-base from fusion and fusion-related experience	Establish engineering-data-base center.

6.2.2 Program Logic

The Level 2 logic diagram for this program element is shown in Fig. 6.4; program subelements are discussed below.

Fusion Test Facilities Preconceptual Design. Preconceptual design activities are carried out for test facilities having a fusion plasma core. The activities include support for the short-pulse ignition experiment and a long-burn demonstration, as well as possible engineering or materials test reactors or other integrated fusion facilities.

Preconceptual design studies typically are performed to identify promising embodiments of a given confinement concept to satisfy the fusion test facilities missions and objectives. Each study evaluates candidate options at a scoping level, makes a choice among the options that becomes a baseline design, and then develops the physics and component engineering of all the major systems and subsystems to a depth sufficient to establish feasibility, R&D needs, and performance. Companion activities are performed that include establishment of design and construction schedules, safety and environmental evaluations, siting evaluations, conventional facilities needs, and preparations of comprehensive cost estimates.

Completion of preconceptual design studies for major fusion test facilities is a significant systems design and analysis activity. Such studies require an integrated project team with physics and technological expertise and project-oriented tasks and milestones. The output of such studies provides the program with the basis for decisions to launch major construction projects.

Critical-Issues Analysis. This task consists of reviews of all conceptual-design reports and other systems studies to identify critical technical issues that involve the interaction of several aspects of plasma physics and fusion technology and the analyses of these issues. The activity investigates innovative solutions and identifies required R&D. Some technical issues, such as impurity control and transient electromagnetics, can be properly addressed only in a systems context. Basic information is developed in the physics and technology R&D programs, but the synthesis of this information into workable solutions requires systems analysis.

The study of critical issues may be broken down into two categories: (1) feasibility analysis and optimization and (2) innovative-solutions studies. Feasibility analysis and optimization consists of in-depth analyses of leading candidate systems (e.g., the poloidal divertor for impurity control). This activity consists of model development and verification, analyses to determine performance limits, and detailed analyses to optimize design parameters. Such studies should produce an understanding of how the system works, verification of calculational tools, and guidelines for design optimization. These results are important for the preconceptual design activities of both test facilities and commercial facilities.

Innovative-solutions studies are intended to find better or simpler solutions to critical systems issues. For example, a scheme for cooling the plasma edge could allow a

FIGURE 6.4 Level 2 Logic Diagram for Fusion Test Facilities, Critical Issues, and Development Pathways

simpler limiter to replace the divertor for impurity control. The results of such studies provide input to the feasibility analysis and optimization studies and also provide guidance for innovative conceptual-design studies. These results will also be integrated into the science and technology R&D programs.

Engineering-Data-Base Assessment. This activity involves the compilation and evaluation of engineering information that will aid the design, construction, and operation of future fusion facilities. As fusion research moves toward the development of more engineering-oriented fusion devices and ultimate commercialization, an ongoing program will be needed to develop and maintain a data base of engineering practices, experiences, and needs. This compilation should be based on knowledge and understanding obtained from the design, construction, and operation of previous and existing fusion devices; on a compilation of perceived engineering-related needs for future fusion devices; and on engineering advances made in advanced technologies related to fusion. Development and maintenance of such an engineering data base will enhance the ability of the fusion program to incorporate the best components, systems, and engineering practices into future fusion devices.

Development-Pathways Analysis. This activity consists of developing and applying methodologies for assessing the cost, risk, and schedule impacts of differing approaches to fusion development. The methodologies incorporate such factors as technical uncertainties and the size, cost, and number of needed test facilities. The technique will incorporate standardized methods for comparing different concepts and different potential applications. An important objective is to identify pathways that lead to useful commercial products while minimizing development times and costs.

6.3 SAFETY, ENVIRONMENT, AND LICENSING

The safety, environment, and licensing program element consists of the following three subelements:

- Safety
- Environment
- Licensing

This program element is aimed at performing the experiments and analyses required to develop a quantitative understanding of fusion safety and environmental issues and to provide the needed safety, environment, and licensing input. This input will affect (1) the selection and design of the short-pulse ignition experiment, (2) the long-burn demonstration, (3) fusion-technology separate effects and integrated testing, (4) fusion technology and materials testing in a fusion environment, and (5) the overall evaluation of the potential for commercial fusion and the decision on an integrated fusion facility.

6.3.1 Issues, Objectives, and Attributes

The primary issues in the safety, environment, and licensing program element are associated with the radioactive inventories that will result from the operation of fusion reactors. These inventories can vary widely, depending on the fusion fuel cycle (e.g., DT, DD, or D³He) and reactor materials chosen. For example, a DT-burning fusion reactor with a stainless steel structure will contain approximately 10^9 Ci of activation products per gigawatt thermal (10^9 Ci/GW) and approximately 10^8 Ci of tritium. Regardless of the choices of fuel cycle and materials, protection of the health and safety of the general public and plant operating personnel must be ensured during normal operation and during accident conditions. Fusion plants must also have acceptable environmental features, and the licensing of fusion plants must be accomplished in an efficient manner. The primary issues are as follows:

- Protection of the general public and plant operating personnel must be provided during all normal and accident conditions that could occur at a fusion plant. Whenever possible, this protection should be provided by having plants that are inherently safe (having passive safety features rather than extensive engineered safeguards). Research to resolve this issue should focus on the presence and potential release of radioactivity in the fusion plant.
- Safety-analysis methodologies must be developed for analyzing the potential safety and environmental impacts of fusion plants. These methodologies must be adequately verified by comparison with data from separate-effects and integral systems tests. Results obtained by use of these methodologies must be realistic and have quantifiable uncertainties.
- Fusion plants must be environmentally benign and comply with environmental criteria acceptable to regulatory agencies and the general public. For example, the U.S. Environmental Protection Agency's standards for air and water quality must be followed. In terms of waste management, the fusion community should adopt the goal that fusion radioactive wastes be amenable to disposal by shallow land burial, as specified in 10CFR61. Also, materials-utilization strategies must be developed, including recycling, so that implementation of a fusion economy does not strain the natural resources available for fusion-plant deployment.
- A rational, efficient licensing system must be developed for commercial plants. Ideally, licensing should not constitute a significant cost of installing a fusion plant, and the licensing activity should be easily accomplished within the time required to physically construct the plant. At the present time, a licensing approach based on risk-based safety goals and the risk-assessment methodology appears to be the most rational approach. With such an approach, the risks from fusion can be placed in context with other societal risks.

The objectives and their associated attributes needed for the resolution of these issues are listed in Table 6.4.

6.3.2 Program Logic

The Level 2 logic diagram for this subelement is shown in Fig. 6.5. Figures 6.6 and 6.7 show the Level 3 diagrams for the safety subelement and combined environment and licensing subelements, respectively. The implications of these figures are discussed below.

The primary activities involved in the safety subelement are (1) to develop and apply methodologies for assessing accident consequences and (2) to develop and collect the data base necessary for the verification of these methodologies. Research is focused on the safety concerns of tritium and activation products and on potential mechanisms for their release; these concerns include lithium fires, magnet accidents, plasma disruptions, and coolant-system failures. Safety-related data to be used in the activities will be generated by safety and fusion-technology experiments, by the materials-research program, and by such fusion facilities as TSTA, TFTR, the short-pulse ignition experiment, and the long-burn demonstration.

The primary activities in the environment subelement are (1) to develop methodologies for analyzing and resolving waste-management issues; (2) to prepare (or assist in preparation of) environmental reports for major fusion-research facilities; and (3) to analyze and develop strategies for utilization and recycling of fusion materials, especially those with near-term supply limitations. The primary generators of radioactive-waste data for this activity include TSTA, TFTR, the short-pulse ignition experiment, and the long-burn demonstration. The output of this activity will be used to prepare environmental reports for future facilities. Ongoing commercial-reactor design studies will be used to assess resource-utilization issues and the need for developing resource-utilization/recycling strategies.

The actual need for a fusion licensing system will not arise until after the beginning of the next century; however, the data requirements for such a system must be anticipated so that relevant data can be generated and collected by ongoing programs. Also, because of the impact that a licensing system can have on fusion economics and on the acceptability of the technology to utilities and the general public, a technically well-founded and streamlined system must be established. The activities in this subelement provide for safety-approval strategies for the major fusion experimental facilities and development of the risk-assessment methodology that will form the basis for a fusion licensing system. This latter activity requires establishment of risk-based safety goals and collection of a fusion-relevant, failure-rate data base to allow the probabilities of various radioactivity-release scenarios to be calculated.

6.4 REMOTE TECHNOLOGY

The primary purpose of the remote-technology program element is to develop the necessary equipment and procedures for design, operation, and maintenance of all future

TABLE 6.4 Objectives and Attributes for Safety, Environment, and Licensing Program Element

Objective	Attribute	Planning Target
Maximize safety of general public during normal operation and during accidents.	Risk to general public, expressed as an incremental increase in existing risk from all routine and accidental sources	Less than 0.1% per individual
Maximize plant personnel safety.	Risk to plant personnel, expressed as a percent of risk from non-occupational hazards	Less than 10%
Maximize quantitative understanding of safety aspects of fusion systems.	Methodologies developed for assessing safety consequences of fusion power	Develop and verify models, with quantifiable uncertainties in calculated results.
Maximize inherent safety of fusion.	Number of prompt fatalities in general public, calculated as a result of severe but credible accidents, with passive safety features	Zero
Maximize understanding of fusion radioactive wastes produced.	Methodologies developed for calculating quantity of radioactive waste to be handled for each radioisotope	Develop and verify models, with quantifiable uncertainties in calculated results.
Minimize high-level radioactive wastes from fusion systems.	Percentage of radioactive wastes from fusion plants that can qualify for near-surface burial, as defined in 10CFR61	>99%
Maximize use of abundant or easily recyclable materials.	For materials with near-term supply limitations, percentage of wastage per recycle	Less than 5%
Minimize impact of licensing activities on cost of fusion power.	Percentage of cost of fusion power that can be attributed to licensing activities and delays	Less than 5%
Minimize licensing time for fusion plants.	Time frame for completion of licensing process	Within time required to physically construct plant

TABLE 6.4 (Cont'd)

Objective	Attribute	Planning Target
Maximize use of probabilistic risk-assessment techniques in fusion licensing.	Capability developed to implement probabilistic risk assessment for fusion systems	Develop methodologies gather appropriate data base, and obtain approval of safety goals.

fusion devices, ultimately including commercial fusion plants. All fusion devices to date have been designed, operated, and maintained with the capability of full access by personnel and equipment when adjustments for operation, maintenance, or replacement of components have been necessary. Operation and maintenance in a highly activated environment have, until now, not been required. All future fusion devices will operate and require service in a highly activated environment. Providing for the operation and maintenance of such fusion devices will require substantial integration of remote-technology equipment and practices into the basic design of the device.

Substantial advances in remote technology can be anticipated in fields outside the fusion program. However, many aspects of the fusion program will be unique in the application of this technology. From the requirements unique to fusion will come the guidelines for the design of fusion-system components and the remote-technology equipment necessary to handle and maintain these components. At present, the fusion program relies on each major project to incorporate the remote technology needed in that project. To date, remote-technology needs have been modest, and it has not been necessary to develop a separate remote-technology program within the overall fusion program. Future, major fusion devices will require sufficient attention to remote-technology needs to warrant a base program, in addition to major efforts within large projects.

Planning the elements of the remote-technology base program, as well as defining the large project development needs, should be the first task performed in this program activity. Such planning should involve experts in the field of remote technology; few such experts are found within the fusion program. The two dominant areas for which detailed planning is required are development of components and subsystems compatible with remote-technology applications and development of remote-technology equipment and procedures for future fusion needs.

The hardware associated with this program is embodied in a series of mock-up systems -- one for the ignition experiment, one for the long-burn experiment, and one leading to the integrated fusion facility. An alternative approach would involve combining these systems into a dedicated Remote-Technology Development Facility that

FIGURE 6.5 Level 2 Logic Diagram for Safety, Environment, and Licensing

FIGURE 6.6 Level 3 Logic Diagram for Safety

FIGURE 6.7 Level 3 Logic Diagram for Environment and Licensing

would support each future reactor project. The fundamental objectives of the remote-technology program are envisioned to be as follows:

- Establish the reliability of remote-technology equipment for use in the projected operating environment.
- Modify existing equipment and develop new remote-technology equipment for anticipated operations and maintenance requirements; implement these on prototype reactor components. Existing equipment consists of tele-operated manipulator systems and transport systems, along with the support equipment to accomplish remote operations (i.e., viewing systems, cutting and welding equipment, and other end-effector tools). New developments will focus on robotic equipment that uses a greater degree of artificial intelligence.
- Investigate the use of standardized interfaces on vacuum joints, coolant connections, electrical connections, and structural joints. Investigate interfaces with mechanical attachments and those that require cutting and welding.
- Establish requirements for standard hardware and procedures.
- Develop a remote-technology design and user manual that covers hardware, design practices, and applications principles.
- Establish a data base for the mean time to repair (MTTR) based on prototype operations and experience from existing operating devices and equipment.
- Establish a data base for mean time between failures (MTBF) of reactor equipment based on existing and ongoing experience with device operations.

6.4.1 Issues, Objectives, and Attributes

The issues associated with the remote-technology program element are as follows:

- Fusion facilities will require design approaches and practices compatible with the need to operate and maintain them remotely. Both the design and subsequent operation and maintenance must be performed in a cost-effective manner.
- Fusion facilities will require complex remote-technology equipment, much of which is beyond the current state of the art.

The objectives and associated attributes for the issues are listed in Table 6.5.

TABLE 6.5 Objectives and Attributes for Remote-Technology Program Element

Objective	Attribute	Planning Target
Maximize reactor availability.	Minimization of MTBF and MTTR	Availability greater than 75%
Develop and use cost-effective remote maintenance equipment.	Simple, modular subsystems and components; compatible remote-maintenance equipment	Equipment costs less than 20% of total capital costs
Minimize exposure of personnel to radiation.	Application of remote-handling technology	Less than 25% of exposure permitted by federal regulations

6.4.2 Program Logic

The Level 2 logic diagram for this program element, shown as Fig. 6.8, is discussed below.

Develop, Issue, and Update Plan. This task includes the formulation of a comprehensive base program for remote technology within the fusion program. This program will define the context, tasks, coordination, schedule, and resources required to achieve remote-technology objectives in support of the overall program.

Develop, Issue, and Update Guideline Document. This task includes the preparation and maintenance of a remote-technology handbook of design practices, principles, typical examples, equipment, and related information necessary for the cognizant design engineers to assure all components and systems are designed to be compatible with the need for operations and maintenance by remote means.

Achieving high availability with remote technology will require the development and maintenance of a data base of relevant information essential to determination and minimization of the MTTR and the MTBF of equipment in fusion devices. This data base will be essential to the development of reliable equipment for fusion applications. Determination of the remote-technology equipment needed to maintain and operate future fusion devices is required. It includes specification of remote-technology practices essential to the effective use of such equipment. The equipment and practices

FIGURE 6.8 Level 2 Logic Diagram for Remote Technology

for the design, installation, maintenance, and replacement of components and systems in fusion devices are included, as are all mock-up developments and associated test equipment. Incorporated into this effort are the necessary training and education of personnel essential to these efforts.

6.5 ALTERNATIVE FUEL CYCLES

This program element covers the complete range of science and technology programs required for a non-DT fuel cycle and specifies the required programs that would not otherwise be carried out in support of the DT fuel-cycle applications. Consequently, the following program subelements have been chosen:

- Confinement Systems and Burning Plasmas
- Plasma Technology
- Nuclear Technology and Materials
- Systems Design and Analysis

6.5.1 Issues, Objectives, and Attributes

The development of an alternative (nontritium) fuel cycle represents a potentially attractive long-range goal for fusion. A base research program and continued assessment of nontritium fuel cycles will allow directions in DT fusion development to be identified that are consistent with potential evolution into an attractive alternative fusion fuel. Operation of fusion systems with a fuel cycle other than DT could significantly reduce certain constraints on reactor design by eliminating the requirement for a tritium-breeding blanket. Eliminating this requirement could allow a much wider range of structural and thermal hydraulic designs, resulting in blankets with higher thermal efficiency. Improved overall designs for reactor assembly and repair may be possible, and greater reliability may also be attainable. However, use of a nontritium fuel cycle means that substantially improved values of beta and density--confinement-time product are necessary relative to those required for DT operation. In addition, devices of larger size and/or stronger magnetic fields may be required in order to use such alternative fuels.

The issues associated with the alternative-fuels program element are as follows:

- The plasma-confinement conditions required for the alternative fuel cycles must be achievable.
- The technologies unique to alternative-fuel-cycle applications must be developed.
- The concepts for alternative fuel cycles must meet the systems requirements for commercial applications.

The objectives and associated attributes for the resolution of these issues are listed in Table 6.6.

6.5.2 Program Logic

The Level 2 logic diagram for this program element is shown in Fig. 6.9; the program logic is discussed below.

Confinement Systems and Burning Plasmas. This task involves theoretical and experimental physics activities to establish the feasibility of concepts based on fuel cycles other than deuterium and tritium. There are two basic alternative cycles: deuterium-based (D-based) and proton-based (p-based) cycles. The present plan is directed mainly at D-based cycles, because these cycles indicate high energy gain can be achieved with ambitious, but possible, plasma parameters. The possibility of igniting p-based cycles, however, remains questionable. Two aspects of D-based cycles should be stressed: DHe^3 is an attractive cycle, offering a significant reduction in neutron flux with only a modest increase in ignition requirements. Other D-based cycles could lead to attractive hybrid (fusion-fission) and synfuel systems.

A key limitation of the DHe^3 cycle is the issue of where to obtain the He^3 . An important recent development is the recognition that mining the lunar surface could

TABLE 6.6 Objectives and Attributes for Alternative Fuels Program Element

Objectives	Attributes
Minimize production and handling of tritium.	Cost of tritium-handling subsystem, expressed as percent of total plant cost
Minimize production of neutrons.	Fraction of total fusion energy carried by neutrons, expressed as percent
Maximize potential for nonthermal energy conversion.	Overall plant efficiency, in percent
Maximize capability to achieve the higher beta and confinement times necessary for alternative-fuel systems.	Predictive capability of plasma theory to verify experiment

FIGURE 6.9 Level 2 Logic Diagram for Alternative Fuels

provide a plentiful, economic supply of He^3 . This might make alternative fuels a competitive route to fusion power.

Another possibility is DD/DT operation, where the required tritium-breeding ratio is less than one. Breeding requirements are reduced, relative to those for DT operation, but the plasma-physics requirements are not as severe as those for DD operation. The physics requirements for operation of hybrid fusion-fission reactors with DD and DD/DT fuel cycles could be substantially less demanding than for the operation of pure fusion DD reactors.

Issues associated with confinement systems and burning plasmas are as follows:

- Requirement for very high beta places emphasis on alternative confinement concepts with high beta.
- Requirement for high-temperature plasmas would emphasize possible extension of DT ignition experiment(s) to stress heating and burn dynamics at higher temperature.
- Confinement concepts using alternative, nontritium fuels require high beta and high-temperature plasma; such concepts must be developed.
- The potential for direct energy conversion is a desired feature that must be developed.
- High-temperature, high-density operation leads to increased importance of and sensitivity to plasma-wall interactions, with emphasis needed on understanding impurity-generation mechanisms, transport, and control.
- Fusion cross-section data for alternative fuels must be obtained.

Plasma Technology. This area involves the development of technologies unique to handling the plasmas associated with nontritium fuels. Alternative-fuel fusion involves two key characteristics: (1) a high beta or high field to compensate for the relatively low plasma power density and (2) a high plasma temperature to achieve ignition. Corresponding engineering constraints require the ability to handle a relatively high first-wall heat load and development of methods to capitalize on the large charged-particle fusion yield (e.g., direct energy conversion). Issues include the following:

- Efficient methods are needed to achieve higher ignition temperatures (e.g., bootstrap heating in combination with normal methods).
- Heating methods are needed that operate at higher temperatures (e.g., negative ion beams and wave heating).

- Fueling technology is needed for He^3 .
- Methods of impurity control must be developed.
- A reliable fuel source must be established, especially for He^3 (evaluate lunar mining, satellite approaches, etc.)
- Advanced and direct energy-conversion methods are needed.
- High-field magnet technology is needed.
- Plasma-control systems are needed to prevent rapid plasma quenches.

Nuclear Technology and Materials. This element consists of development activities associated with the nuclear technologies and materials required for alternative-fuel fusion systems. One objective of alternative fuel cycles is to minimize the production of neutrons, so the nuclear technology requirement for alternative fuel-cycle systems is expected to be adequately covered by R&D for DT systems, and no special requirements are projected. Materials needs are expected to be mostly in the area of handling high heat flux.

Systems Design and Analysis. This area consists of studies of alternative fuel systems. Studies are required to achieve the following:

- Evaluate the potential of alternative fuels on a basis consistent with DT-based studies.
- Establish attractive systems-integrated approaches to high beta value, improved confinement concepts, direct energy conversion, and unique blanket designs.
- Identify needs and build on the emerging DT data base.
- Develop test-facility designs and commercial-design concepts.
- Identify attractive alternative applications that offer unique advantages (e.g., hybrid and synfuels plants).

The key activities are as follows:

- Select the most promising high-beta confinement concepts for burning alternative fuels. This selection will require studies of available theoretical and experimental data for such concepts as high-field tokamaks, compact tori, reversed-field configurations, tandem mirrors, etc.

- Identify possible reactivity-enhancement approaches. Enhancement is not essential for D-based fuels, but if p-based cycles are to be considered, this is mandatory.
- Identify cross-section needs. Again, this is essential for p-based cycles.
- Identify experiments to study ways to handle the high heat flux on the first wall and identify technology developments (e.g., direct energy conversion) that would provide a high leverage in the utilization of alternative fuels.

OVERVIEW

Fusion is an essentially inexhaustible source of energy that has the potential for economically attractive commercial applications with excellent safety and environmental characteristics. The primary focus for the fusion-energy development program is the generation of central-station electricity. Fusion has the potential, however, for many other applications. The fact that a large fraction of the energy released in a DT fusion reaction is carried by high-energy neutrons suggests potentially unique applications. These include breeding of fissile fuels; production of hydrogen and other chemical products; transmutation or "burning" of various nuclear or chemical wastes; radiation processing of materials; production of radioisotopes; food preservation; medical diagnosis and medical treatment; and space power and space propulsion. In addition, fusion R&D will lead to new products and new markets.

Each fusion application must meet certain standards of economic and safety/environmental attractiveness. For this reason, economics and safety/environment/licensing have been chosen as the two criteria for setting long-range commercial fusion objectives. These objectives are discussed in Sec. 7.1. A major function of the systems activities described in Chapter 6 is to evaluate the potential of fusion against these objectives and to help guide the fusion R&D program toward practical applications.

The transfer of fusion technology and skills from the national laboratories and universities to industry is the key to achieving the long-range objective of commercial fusion applications. This process is described in Sec. 7.2.

7.1 COMMERCIAL OBJECTIVES

7.1.1 Electricity Production

The application of fusion that has received the most study is the production of electricity in a central-station power plant. Commercial objectives for fusion electricity production have been developed with the following aims:

- Make fusion economically competitive with other forms of central-station power for the 21st century.
- Exploit the safety and environmental advantages of fusion in plants that offer a very low risk to the public and to plant workers, as well as providing a very low risk of losing plant investment costs.
- Make the R&D costs for fusion an acceptable fraction of the potential benefit.

The key aspects of economic performance are low capital cost, short construction and licensing time, high availability, and low operating costs. These aspects are directly related to requirements on component performance, lifetime, repair time, and safety characteristics that affect licensing. Stringent requirements must be set in all of these areas. The most important areas involve component failure rates and repair times, because high availability must be maintained in fusion plants, which are capital-intensive. Another key objective is to provide a range of unit electrical power ratings in economically attractive plants.

The safety and environmental objectives stress inherent safety under all credible accident conditions. Inherent safety offers many potential benefits, including ease of licensing; elimination of high-cost, engineered safety systems; reduction in backfitting; lower cost for the balance of plant; and public acceptance. Inherent safety must be accompanied by very low normal emissions of hazardous materials.

The commercial objectives are set for a hierarchy of characteristics, as shown in Fig. 7.1. These objectives are given, along with attributes and target values, in Tables 7.1 and 7.2. A set of parameters that might characterize a commercial electric-power plant and an intermediate integrated fusion facility was given in Table S.2 in the Synopsis.

7.1.2 Fissile-Fuel Production

An application of fusion that may have economic potential is the cogeneration of both electricity and fissile fuels for later consumption in fission converter reactors (e.g., LWR or HTGR) that produce electricity or process heat. Fissile-fuel production has the following aims:

- Make fusion-bred fuels economically superior to mined and enriched uranium in the early decades of the 21st century.
- Provide a source of affordable fissile fuel to support the domestic and international demand for nuclear power for the indefinite future.
- Provide the technological basis and operating experience required to advance fusion technology to a level such that fusion electric-power generation will become an attractive alternative to conventional nuclear power.

The key aspects of economic performance are low capital costs, high fissile-fuel breeding, high availability, low operating costs, and low fuel-cycle costs. These aspects are directly related to requirements on component performance, lifetime, repair time, and safety characteristics. The most difficult problem areas are component failure rates and repair times. High availability is required in fusion plants because they are capital-intensive. For fissile-fuel breeding, the requirements on fusion performance (relative to fusion-electric applications) are relaxed, because the fusion energy is multiplied

FIGURE 7.1 Characteristics of Commercial Fusion Systems

TABLE 7.1 Economic Objectives, Attributes, and Target Values for Electricity Production

Program Element and Subelement	Objective	Attribute	Planning Target
Economics	Minimize cost of product.	Cost of electricity (mills/kWh), evaluated by levelized costing, zero escalation and inflation, in 1985 dollars	30-40
	Maximize investment protection.	Cost of recovery from any fusion-core component or subsystem failure or accident, expressed as percent of original direct capital cost	5-15
		Time required for recovery from any accident (months)	3-9
Capital Costs	Minimize capital costs.	Total direct cost to construct plant ($\$/kW_e$) for a nominal 1000-MW _e plant	~ 1000
Operating Costs	Minimize operating costs.	Cost to operate plant, expressed as percent of cost of electricity	< 5
Availability	Maximize availability.	Percent of total time plant is available for full-power operations	~ 85
Development Costs	Minimize development costs.	Total projected development cost before first commercial order ($\$10^9$)	~ 20

TABLE 7.2 Safety, Environment, and Licensing — Objectives, Attributes, and Target Values

Program Element and Subelement	Objective	Attribute	Planning Target
Public Safety	Maximize public safety.	Risk to public from accidents, expressed as percent of existing risk from all accidental sources	< 0.1
		Risk to public from routine operations, expressed as percent of existing risk from all routine sources	< 0.1
Plant-Personnel Safety	Maximize plant-personnel safety.	Risk to plant personnel from occupational hazards and accidents, expressed as percent of risk from non-occupational hazards	< 10
Materials	Maximize use of domestically available, abundant, or recyclable materials.	For those elements for which fusion is the driver in domestic demand, recycle, expressed as percent of wastage per cycle	1-5, depending on element
		Procurement of no more than stated percent from non-domestic sources	< 20
Environment	Minimize thermal effluent from facility.	Waste thermal effluent from facility, expressed as percent of gross thermal power	< 70
	Minimize long-term activation.	Percentage of radioactive waste generated that qualified for near-surface disposal, as defined in 10CFR61 or relevant extensions thereof	>99%
		Dilution of used material to meet standards, expressed as factor increase in volume of disposed radioactive materials	< 10
Licensing	Minimize licensing time.	Time frame during which licensing process is completed	Prior to or during construction

severalfold in the blanket and because an additional product, fissile fuel, is produced. However, safety/environment/licensing issues may be similar to those encountered for fission plants.

7.1.3 Synthetic-Fuel Production

Fusion energy could be a source of electricity and high-temperature process heat for the production of synthetic fuels. Hydrogen production by thermochemical water splitting or by high-temperature electrolysis has received limited study. The hydrogen produced can be used as a feedstock material to produce other fuels, such as methanol.

The advantage of fusion as the heat source for synthetic-fuel (synfuel) technologies stems from the deep penetration of the 14-MeV neutrons produced in the DT reaction. This penetration allows thermal decoupling of the high-temperature blanket from the fusion core. Such decoupling is not possible with combustion or fission heat sources.

The economic performance required of a fusion reactor for synfuel production is equal to or better than that required for central-station electricity production. Thus, most of the objectives defined for electricity are equally applicable to synfuels. The most important environmental objective is to keep the product (hydrogen or organic fuel) free of tritium contamination. The high mobility of tritium and its affinity to replace hydrogen in organic compounds will make achieving the required product cleanliness very difficult.

7.1.4 Other Applications and Spinoffs

The three fusion applications described above are those that have been most extensively analyzed. Plausible conceptual designs have been developed for fusion systems to produce these products, and preliminary economic evaluations indicate that fusion has potential to compete with alternative sources of these products in the planning time frame for commercial fusion applications. Other potential applications include transmutation of high-level wastes produced in fission reactors, production of radionuclides for commercial and medical applications, and production of special nuclear materials for military applications.

Preliminary analyses have been made of the use of fusion to transmute or "burn" actinide wastes from fission reactors in order to reduce the amount of waste and its long-term toxicity. The prime concerns are economic feasibility and technical complexity.

An application area where fusion may have a significant advantage over alternative sources is the production of radionuclides. Fusion produces about five times more net neutrons per unit of thermal power than does fission, and these neutrons are of much higher energy (14 MeV, compared with 3 MeV for fission). This means that fusion can be a more prolific source of transmutation products than fission. Preliminary

analysis of the production of cobalt-60 and other isotopes by fusion indicates that fusion reactors could easily satisfy future market requirements.

Finally, initial studies indicate that fusion reactors can potentially produce tritium and plutonium for military applications much more efficiently than fission reactors. Many other potential fusion applications exist that have not yet been evaluated.

An important benefit from fusion is the "spinoff" of knowledge and technology to other fields. As the space program has amply demonstrated, when high-technology research and development is undertaken, products and applications result that are of significant benefit outside the immediate program. Fusion has already produced a positive benefit from spinoffs (see Table 7.3). Additional spinoff benefits from fusion science and technology programs are expected.

7.2 TECHNOLOGY TRANSFER

The DOE Magnetic Fusion Program Plan contains the following technology-transfer objective:

The technology transfer objective is to provide a range of options for private sector investment and commercial development of fusion. The establishment of the scientific and technological base for fusion requires industrial participants both to provide expertise in conventional components and to gain experience with the unique aspects of fusion science and technology. The necessary degree of industrial experience is best gained through the technical participation of industrial personnel in the state-of-the-art developments produced

TABLE 7.3 Spinoffs from Fusion Science and Technology Programs

Field	Spinoff
Computers	Cray timesharing-system software
Metal forming	Magneform system
Isotope separation	Plasma-separation process
Defense	Neutral- and charged-particle beams; high-power, high-frequency microwave sources and microwave transmission components
Welding	Refractory-armor materials and tiles, homopolar resistance welding
Magnets	Superconducting magnets for energy storage, materials processing, and medical applications

by the fusion program. A noteworthy aspect of this objective is that, through it, the fusion program will also serve as a stimulus to United States technological growth and the further training of scientists and engineers in industry.

This section describes how this DOE objective can be achieved, in the context of the detailed technical plans previously described. Industry has played and will play a variety of roles in the development of fusion energy. The more useful roles fulfilled by industry can be separated into three main categories. These categories, along with the principal functions performed in each category, are listed in Table 7.4. Each of these roles and the corresponding potential functions will be described in detail below.

7.2.1 Industry as Advisor

The advisory role is filled frequently by corporate officials, who are asked to help assess various stages of program development and may serve on management boards of development projects. The principal benefit expected from the advisory role is the development of appropriate program goals. This role is discussed below.

Support Services. Industry acts as a support-services contractor when it assigns individuals or small groups of individuals to work in direct support of a manager at DOE or at a national laboratory. In such an arrangement, the customer benefits by acquiring immediately needed skills and having a long-term personnel commitment. Industry benefits by acquiring knowledge, contacts, and income.

TABLE 7.4 Industrial Roles and Functions

Roles	Functions
Advisor	Support-Services Contractor Advisory Committees
Direct Participant	Research and Development Materials Supplier Component Supplier and Manufacturer Subsystems Contractor Prime Contractor, Project Manager Facilities Operator Customer
Sponsor	Research and Development

Advisory Committees. Utilities, as well as industry, can provide members of their technical staffs to serve on technical committees assigned to carry out specific tasks. This participation serves to facilitate development of program goals and provides useful feedback regarding program direction.

7.2.2 Industry as Direct Participant

A number of studies have been conducted to examine the characteristics of research programs that led to the successful commercialization of new technologies. A common characteristic among those technologies was the early involvement of the ultimate user. Therefore, it is important both to the user and to the national program to include early participation of the user at all phases of the program.

To become major participants in the fusion program, industrial executives must understand and identify with near-term program objectives. The program objectives must be developed in conjunction with the ultimate users, if they are expected to become involved early. One near-term goal, essential to the success of fusion, is the resolution of environmental and safety issues, which have proven to be a major stumbling block for fission.

Direct participation requires a corporate commitment of financial and manpower resources to perform fusion work. This type of participation can take a variety of forms; the more notable functions, listed in Table 7.4, are discussed below.

Research and Development. Private companies with unique ideas should be able to compete for funding when, after peer review, such support is technically warranted. The fusion program should make full use of novel ideas and approaches conceived in all sectors of the society, including individuals, national laboratories, universities, and private companies.

Materials Supplier. Materials used for commercial application will require a well-documented history for quality-assurance/quality-control (QA/QC) purposes. As the program enters the commercialization phase of fusion development, detailed engineering specifications will be needed.

Component Supplier and Manufacturer. Industry acts as component supplier or manufacturer when it supplies an off-the-shelf component or when the customer has build-to-print requirements. Industry also frequently designs and manufactures components to customer-supplied specifications. Increasingly, manufacturers will have to take on the job of fabricating fusion-specific components, subsystems, and (eventually) complete reactor systems.

Subsystems Contractor. Industry acts as a subsystems contractor in situations where the customer has defined a scope of work and has assigned to a company the responsibility for performance. This is a most desirable form of industrial participation. The customer benefits by having corporate commitment to the project, and industry benefits by being able to fully exercise its managerial and technical skills through a task assignment where it can bring to bear its background and experience.

Prime Contractor, Project Manager. Industry acts as prime contractor or project manager when it is directly responsible to a customer (e.g., DOE, a national laboratory, or eventually, an electric utility) for defined aspects of management, engineering, fabrication, and installation of a product, such as a fusion device or power reactor. An architect-engineer usually represents the client for engineering, procurement, and construction. However, the manufacture of the steam-supply system is usually carried out by separate companies. The industry roles that will emerge for fusion can only be developed by having experienced companies participate in the program. Preparing companies for these roles should be an important component in the fusion program.

Facilities Operator. Many companies are in the business of operating manufacturing plants, chemical plants, communications networks, and other sophisticated operational activities. To compete in the marketplace, these companies must also mobilize and manage the personnel logistics and training required for facilities operation. Industrial companies can operate fusion equipment that they or others fabricate. This can be an important learning experience and can help prepare the companies for more important roles in the future and for direct participation in the development of fusion power. This is particularly true for those fusion experiments that involve technology development.

Customer. It is important for potential customers to interact with the developers of fusion to ensure that the final product is one that is acceptable for commercial use. Customers must be fully knowledgeable about the scientific and technological questions to be addressed in determining design trade-offs. Companies that deal with customers must be prepared to stand behind their products and services with performance guarantees.

7.2.3 Industry as Sponsor

Sponsorship includes contributions of direct funds, labor, or both. This form of participation has existed since the 1950s, albeit on a small scale, and is continuing now. Industrial sponsorship of fusion R&D includes the direct support of university research by utilities. Indirect utility support has also been provided through the Electric Power Research Institute (EPRI).

7.2.4 Major Opportunities for Industry

The major opportunities for industrial participation in the fusion program are coupled to the design and construction of major fusion facilities and associated support activities for these facilities. A schedule for these anticipated opportunities is shown in Fig. 7.2.

FIGURE 7.2 Logic Diagram for Technology-Transfer Plan

APPENDIX A:

TECHNICAL PLANNING ACTIVITY MISSION STATEMENT

PURPOSE

The purpose of the Technical Planning Activity (TPA) is to assist the Office of Fusion Energy (OFE) by developing a methodology for planning the national magnetic fusion program and by preparing technical planning documents in support of the strategic and policy framework of the Magnetic Fusion Program Plan (MFPP).

SCOPE

The TPA will define program elements and subelements that provide a basis for developing program plans that solve the four key MFPP technical issues (Confinement Systems, Burning Plasmas, Materials, and Nuclear Technology). The TPA will also prepare program plans for completing the program in such a way that the three MFPP strategic objectives (Science Development, Technology Development, and Technology Transfer) are fully satisfied. The following items describe the scope of this activity.*

- Identify technical issues that must be resolved to fully resolve the four key technical issues.
- Identify program elements and subelements to resolve the technical issues.
- Prepare statements of objectives for program elements and subelements and attributes that measure progress toward achieving those objectives.
- Develop technical plans for each program element that include research activities, experiments, and test facilities and identify decision points and milestones. These plans should:
 - Consider existing programs and facilities in the U.S. and abroad and possible modifications.
 - Identify necessary and/or desired characteristics of new facilities.
- Develop logic networks to show interdependence among the various program elements or subelements.
- Identify resource requirements for technical plans.

*Definitions of selected terminology are provided at the end of this appendix.

- Stimulate the development of community consensus and recommendations for priorities within program elements.

PLANNING ASSUMPTIONS

The MFPP provides the policy and overall framework for the TPA. Several points in the MFPP have been further elucidated as follows:

Planning Horizon

- The MFPP goal is to establish the scientific and technological base required to carry out an assessment of the economic and environmental aspects of fusion energy. As a culmination of this assessment process, one or more integrated fusion facilities would be built and operated in the post 2000 period.
- TPA should consider all possible applications of fusion energy. Emphasis is expected to be on electricity and fissile fuel production. The main emphasis should be on deuterium-tritium fueled reactor concepts.
- The planning horizon for the TPA is 15-20 years. Detailed planning should emphasize the next five years and major facilities required in the 1990s.

Budget Assumptions

- Individual projects (or contributions to international projects) should not require annual funding more than about 20% of the Magnetic Fusion Energy (MFE) budget in any given year. This is approximately equivalent to a limitation on the total cost (or total contribution to an international project) to about one annual budget for the MFE program.
- A total budget for the U.S. fusion program no greater than flat (constant dollars) funding through 1990 should be assumed. However, for planning purposes, it is not necessary to constrain each part of the program to its current funding level.
- Substantial complementary international activity and collaboration should be assumed, so that the total resource requirements for the activities in the plan could exceed the U.S. budget by specifically accounting for complementary work or resources to be made available to the U.S. from Japan and the EC. (The MFPP states

that progress in the 1990s will be determined, in part, by the degree of international collaboration on major test facilities.)

Major Facilities

- Construction by either EC or Japan of a facility similar to the proposed NET and FER with operation in the late 1990s, and construction of a U.S.-based ignition experiment with operation in the early 1990s, should be assumed.
- Study of burning plasmas is to be pursued in parallel with confinement system development and need not be on the direct path of concept improvement. The tokamak is the only viable candidate for a near-term ignition experiment.

REPORTING

The TPA will proceed through three overlapping phases. In Phase I the structure must ensure comprehensive derivation of technical issues, development of objectives, and identification of program elements and subelements. In Phases II and III the product must show how each program element contributes to the resolution of the four key technical issues.

A report will be prepared for each phase of the program.

Phase I: An informal status report, due in November 1985, will identify technical issues and program elements and subelements and will include:

- TPA mission, methodology, and organization.
- Summary of objectives for most program elements and subelements of the MFE program.
- Summary of technical issues.
- Consideration of various applications of fusion energy and identification of characteristics of potential commercial fusion production.
- Interpretation of the role of "integrated fusion system" concept.

Phase II: An interim report, due February 1986, is intended as a working supplement to the final report. The Interim Report will combine logic networks with the information developed in Phase I and will include:

- Summary of November 1985 status report.
- Update on technical issues, program elements, and objectives.
- Initial logic networks for technical plans.

Phase III: A final report, due in July 1986, will contain two levels of reporting: (1) a summary document suitable for general technical audience outside the fusion program, and (2) documents that contain detailed technical and planning information. The Final Report will include:

- Information from interim and status reports.
- Resource and facility requirements.
- Logic networks for technical plans.

INTERFACE BETWEEN DOE/OFE AND TPA

- J.F. Clarke (in his absence, N.A. Davies) is responsible for direction of the TPA.
- The leader of the TPA activity (C.C. Baker) reports to J.F. Clarke.
- OFE staff may monitor progress of the TPA by attending working meetings and quarterly briefings and through distribution of status reports and other information.
- R.E. Nygren coordinates activities within OFE related to TPA and is responsible for distribution of information to OFE staff as well as solicitation and consolidation of feedback from OFE.
- OFE staff recommendations for redirection of TPA activities will be routed through J.F. Clarke.
- The TPA will keep the OFE staff informed of its activity, including notification of TPA meetings and events.
- The TPA will conduct periodic (at least every three months) briefing sessions for the OFE staff.

DEFINITIONS OF TERMS USED IN TPA MISSION STATEMENT

GOAL:

The term "goal" refers to the goal of the magnetic fusion program, which is to establish the scientific and technological base required for fusion energy.

STRATEGIC OBJECTIVE:

The term "strategic objective" refers to the three strategic objectives stated in the MFPP: (1) science development, (2) technology development, and (3) technology transfer.

OBJECTIVE:

An objective identifies what must be achieved and a direction for achievement.

ATTRIBUTE:

An attribute indicates the degree to which its associated objective is met. A special measurement scale must be established for each attribute.

KEY TECHNICAL ISSUES:

There are four key technical issues stated in the MFPP: (1) magnetic confinement systems, (2) properties of burning plasmas, (3) fusion materials, and (4) fusion nuclear technology.

TECHNICAL ISSUE:

A technical issue is a problem that needs to be solved to help resolve one or more of the key technical issues.

PROGRAM ELEMENT:

A portion of the MFE program having reasonably common technical issues and objectives.

SUBELEMENT:

A subelement is a subset of a program element or higher subelement.

LOGIC NETWORK:

A diagram and supporting text that describe the overall sequence of research and development tasks within a program element required to meet objectives. The logic network should also show the interdependencies among related program elements and subelements, as well as indicating major decision points.

PLANNING METHODOLOGY:

Planning methodology refers to the approach and organization of the TPA required to carry out its mission. Central features of this methodology include reliance on the MFPP for overall policy, strategy, and involvement of a broad segment of the magnetic fusion community in the planning process.

MILESTONE:

Particular level of achievement of an objective, tied to an approximate date.

APPENDIX B:
TECHNICAL PLANNING ACTIVITY PARTICIPANTS AND MEETINGS

TPA STEERING COMMITTEE
Charles Baker, Chairperson
Argonne National Laboratory

PLASMA SCIENCE GROUP

<u>Name</u>	<u>Affiliation</u>
James Callen, Leader	The University of Wisconsin (UW)
John Santarius	The University of Wisconsin
(Asst. to Leader)	
David Baldwin	Lawrence Livermore National Laboratory (LLNL)
Richard Hazeltine	The University of Texas at Austin
Rulon Linford	Los Alamos National Laboratory (LANL)
James Lyon	Oak Ridge National Laboratory (ORNL)
Richard Post	Massachusetts Institute of Technology (MIT)
John Rawls	GA Technologies, Inc.
Paul Rutherford	Princeton Plasma Physics Laboratory (PPPL)
John Sheffield	Oak Ridge National Laboratory

TPA STEERING COMMITTEE (Cont'd)

FUSION TECHNOLOGY GROUP

<u>Name</u>	<u>Affiliation</u>
Mohamed Abdou, Leader	The University of California at Los Angeles (UCLA)
Richard Mattas	Argonne National Laboratory (ANL)
(Asst. to Leader)	
Lee Berry, Plasma	Oak Ridge National Laboratory
Technology Coordinator	
Dale Smith, Fusion	Argonne National Laboratory
Materials Coordinator	
John Bartlit	Los Alamos National Laboratory
Jim Crocker	EG&G Idaho, Inc.
Wilhelm Gauster	Sandia National Laboratories (SNL)
Carl Henning	Lawrence Livermore National Laboratory
John Schmidt	Princeton Plasma Physics Laboratory
Kenneth Schultz	GA Technologies, Inc.
Herbert Woodson	The University of Texas at Austin
Herbert Yoshikawa	Westinghouse Hanford Company

TPA STEERING COMMITTEE (Cont'd)

FUSION SYSTEMS GROUP

<u>Name</u>	<u>Affiliation</u>
Stephen Dean, Leader	Fusion Power Associates
Daniel Cohn	Massachusetts Institute of Technology
Jim Crocker	EG&G Idaho, Inc.
Charles Flanagan	Fusion Engineering Design Center
James Gordon	TRW, Inc.
Robert Krakowski	Los Alamos National Laboratory
George Miley	The University of Illinois
Kenneth Schultz	GA Technologies, Inc.
John Sheffield	Oak Ridge National Laboratory

TPA EXECUTIVE COMMITTEE

<u>Name</u>	<u>Affiliation</u>
C.C. Baker, Chairperson	Argonne National Laboratory
M.A. Abdou	The University of California at Los Angeles
J.D. Callen	The University of Wisconsin
S.O. Dean	Fusion Power Associates
R.A. Krakowski	Los Alamos National Laboratory
R.F. Mattas	Argonne National Laboratory
J.M. Rawls	GA Technologies, Inc.
J.F. Santarius	The University of Wisconsin
W.M. Stacey	Georgia Institute of Technology (GIT)

TPA Staff

R.E. Nygren*	Argonne National Laboratory
J.P. Peerenboom	Argonne National Laboratory
R.W. Shaw	Arete Corporation
R.G. Whitfield	Argonne National Laboratory

*Responsible for liaison with DOE's Office of Fusion Energy.

TPA MEETINGS

Steering Committee

June 4-5, 1985	ANL
August 15-16, 1985	MIT
November 14-15, 1985	ANL
February 3-5, 1986	GA
April 10-11, 1986	GIT
June 25-26, 1986	ANL

DOE Magnetic Fusion Advisory Committee Briefings

May 8-9, 1985	UCLA
August 14-15, 1985	MIT
November 20-21, 1985	LANL
February 19-20, 1986	LLNL
May 13-14, 1986 (Panel XV)	New York University (NYU)
October 30, 1986 (Panel XV)	NYU

DOE Office of Fusion Energy Briefings

February 5, 1985
August 8, 1985
January 30, 1986

Industrial Advisory Committee Meetings

January 15-16, 1986	San Francisco
March 13, 1986	San Diego, Calif.

Laboratory Visits

-- GA Technologies, Inc.
April 3, 1985
July 2, 1986

-- Los Alamos National Laboratory
June 7, 1985
July 1, 1986

TPA MEETINGS (Cont'd)

Laboratory Visits (Cont'd)

- Lawrence Livermore National Laboratory
March 7, 1985
July 10, 1986
- Massachusetts Institute of Technology
April 22, 1985
July 30, 1986
- Oak Ridge National Laboratory
May 13, 1985
July 18, 1986
- Princeton Plasma Physics Laboratory
April 11, 1985
July 18, 1986

Senior Fusion DOE and Laboratory Directors' Meeting

- June 17-19, 1985 Annapolis, Md.

APPENDIX C:
GLOSSARY

ALEX	Argonne Liquid Metal Experiment
ANL	Argonne National Laboratory
ASDEX	Axially Symmetric Divertor Experiment
ASDEX-U	Axially Symmetric Divertor Experiment Upgrade
ATF	Advanced Toroidal Facility
BEATRIX	Breeder Exchange Matrix
BHP	Biological hazard potential
CANDU	Canadian heavy water (deuterium) reactor
CIT	Compact Ignition Tokamak
C-Mod	Proposed high-field tokamak at MIT
CPRF	Confinement Physics Research Facility
CTF	Component Test Facility
cw	Continuous wave
DBTT	Ductile-to-brittle transition temperature
DCT	Steady-state tokamak proposed by MIT
dpa	Displacements per atom
DIII-D	Doublet-III D Configuration
DD	Deuterium-deuterium
DOE	U.S. Department of Energy
DT	Deuterium-tritium
EA	Environmental assessment
EBS	Elmo Bumpy Square
EBT	Elmo Bumpy Torus

GLOSSARY (Cont'd)

EBT-S	Elmo Bumpy Toroid-Square
EC	European Community
ECH	Electron cyclotron heating
ECRH	Electron cyclotron resonance heating
EIS	Environmental impact statement
EPRI	Electric Power Research Institute
ETR	Engineering Test Reactor
FEDC	Fusion Engineering Design Center
FELIX	Fusion Electromagnetic Induction Experiment
FER	Fusion Engineering Reactor
FINESSE	Study of Issues, Experiments, and Facilities for Fusion Technology Research and Development
FNS	Fusion Neutron Source at JAERI
FNT	Fusion nuclear technology
FPA	Fusion Power Associates
FRC	Field-reversed configuration
FSAR	Final safety analysis report
FT-U	Frascati Tokamak Upgrade
FUBR-1B	Fusion Breeders
GA	GA Technologies, Inc.
GEIS	Generic environmental impact statement
GIT	Georgia Institute of Technology
HEDL	Hanford Engineering Development Laboratory

GLOSSARY (Cont'd)

HFTF	High-Field Test Facility
HTGR	High-temperature gas-cooled reactor
HVIDF	High-Velocity Injector-Development Facility
IBWH	Ion Bernstein-wave heating
ICH	Ion cyclotron heating
ICRF	Ion cyclotron radio frequency
ICRH	Ion cyclotron resonance heating
IFF	Integrated fusion facility
INTOR	International Tokamak Reactor
JAERI	Japan Atomic Energy Research Institute
JET	Joint European Torus
LANL	Los Alamos National Laboratory
LBD	Long-burn demonstration
LCP	Large Coil Project
LCT	Large Coil Task
LCTF	Large Coil Test Facility
LHH	Lower hybrid heating
LLNL	Lawrence Livermore National Laboratory
LM	Liquid metal
LOCA	Loss of coolant accident
LOF	Loss of flow
LSE	Large Stellarator Experiment
LWR	Light water reactor

GLOSSARY (Cont'd)

MFE	Magnetic Fusion Energy
MHD	Magnetohydrodynamics
MFPP	Magnetic Fusion Program Plan
MFTF-B	Mirror Fusion Test Facility
MINIMARS	Mini-Mirror Advanced Reactor Study
MIT	Massachusetts Institute of Technology
MTBF	Mean time between failures
MTR	Materials Test Reactor
MTTR	Mean time to repair
NBT	Nagoya Bumpy Torus
NET	Next European Torus
NYU	New York University
OFE	Office of Fusion Energy (DOE)
OH	Ohmic heating
ORMAK	Oak Ridge Tokamak
ORNL	Oak Ridge National Laboratory
PBX	Princeton Beta Experiment
PBX-U	Princeton Beta Experiment Upgrade
PIC	Plasma-interactive component
PITF	Partially Integrated Test Facility
PLT	Princeton Large Torus
PMI	Particle material interactions
PMTF	Plasma Materials Test Facility

GLOSSARY (Cont'd)

POP	Proof-of-principle
PPPL	Princeton Plasma Physics Laboratory
PT	Plasma technology
QA	Quality assurance
QC	Quality control
RAD	Radiation absorbed dose
RADS	Room-atmosphere detritiation systems
rf	Radio frequency
RFP	Reversed-field pinch
RFTF	Radio-Frequency Test Facility
SNL	Sandia National Laboratory
SPI	Short-pulse ignition
TARA	Tandem Mirror Device at MIT
TIBER	U.S. Engineering Test Reactor design by LLNL
TEXTOR	Tokamak Experiment for Technology Oriented Research
TFTR	Tokamak Fusion Test Reactor
TF	Toroidal field
TMIF	Thermochemical Integration Facility
TMX	Tandem Mirror Experiment
Tore-S	Superconducting tokamak at Cadarache, France
TPA	Technical Planning Activity
TPX	Tritium Plasma Experiment
TSTA	Tritium System Test Assembly

GLOSSARY (Cont'd)

UCLA	The University of California at Los Angeles
UW	The University of Wisconsin
2XIIB	Mirror Experiment at LLNL

DISTRIBUTION FOR ANL/FPP-87-1Internal:

C. Baker	R. Leonard	D. Smith
E. Bennett	L. LeSage	R. Smither
M. Billone	R. Lewis	M. Steindler
J. Brooks	Y. Liu	H. Stevens
Y. Cha	B. Loomis	D. Sze
O. Chopra	S. Majumdar	S. Tam
D. Diercks	J. Marchaterre	C. Till
D. Ehst	R. Mattas	L. Turner
K. Evans	M. Mendelsohn	R. Valentin
P. Finn	B. Misra	E. VanDeventer
A. Fischer	R. Nygren	S. Vogler
Y. Gohar	J. Peerenboom	R. Weeks
L. Greenwood	M. Petrick	R. Whitfield
D. Gruen	E. Pewitt	T. Yule
A. Hassanein	B. Picologlou	FPP Files (289)
C. Johnson	R. Poeppel	ANL Contract File
A. Krauss	K. Porges	ANL Libraries
R. Kustom	C. Reed	ANL Patent Dept.
R. Lee	A. Schriesheim	TIS Files (6)

External:

DOE-TIC, for distribution per UC-20 (106)
 Manager, Chicago Operations Office, DOE
 M. Abdou, University of California-Los Angeles
 W. Ard, McDonnell Douglas Astronautics Company
 C. Ashworth, Pacific Gas and Electric Company
 D. Baker, Los Alamos National Laboratory
 D. Baldwin, Lawrence Livermore National Laboratory
 J. Bartlit, Los Alamos National Laboratory
 W. Bauer, Sandia National Laboratory-Livermore
 P. Beament, Virginia Power
 D. Beard, Department of Energy/Office of Fusion Energy
 S. Berk, Department of Energy/Office of Fusion Energy
 L. Berry, Oak Ridge National Laboratory
 R. Blanken, Department of Energy/Office of Fusion Energy
 W. Bohke, Stone & Webster
 C. Bolta, Combustion Engineering
 C. Bolton, Department of Energy/Office of Fusion Energy
 R. Botwin, Grumman
 J. Callen, University of Wisconsin
 B. Carreras, Oak Ridge National Laboratory
 J. Cecchi, Princeton Plasma Physics Laboratory
 J. Clarke, Department of Energy/Office of Fusion Energy
 F. Clinard, Los Alamos National Laboratory
 F. Coensgen, Lawrence Livermore National Laboratory
 M. Cohen, Department of Energy/Office of Fusion Energy
 D. Cohn, Massachusetts Institute of Technology
 R. Conn, University of California-Los Angeles

D. Crandall, Department of Energy/Office of Fusion Energy
 J. Crocker, EG&G Idaho
 R. Davidson, Massachusetts Institute of Technology
 A. Davies, Department of Energy/Office of Fusion Energy
 J. Davis, McDonnell Douglas Astronautics Company
 J. Dawson, University of California-Los Angeles
 S. Dean, Fusion Power Associates
 V. Der, Department of Energy/Office of Fusion Energy
 D. Dobrott, Science Applications International
 D. Doran, Hanford Engineering Development Laboratory
 R. Dory, Oak Ridge National Laboratory
 W. Dove, Department of Energy/Office of Fusion Energy
 R. Dowling, Department of Energy/Office of Fusion Energy
 H. Dreicer, Los Alamos National Laboratory
 W. Drummond, University of Texas
 G. Ducat, Southern California Edison
 S. Eckstrand, Department of Energy/Office of Fusion Energy
 A. Favale, Grumman Corporation
 C. Finfgeld, Department of Energy/Office of Fusion Energy
 T. Finn, Department of Energy/Office of Fusion Energy
 C. Flanagan, Fusion Engineering Design Center
 H. Forsen, Bechtel
 K. Fowler, Lawrence Livermore National Laboratory
 J. Freidberg, Massachusetts Institute of Technology
 H. Furth, Princeton Plasma Physics Laboratory
 R. Gariboldi, Department of Energy, Chicago Operations Office
 W. Gauster, Sandia National Laboratory-Albuquerque
 K. Gentle, University of Texas
 T. George, Department of Energy/Office of Fusion Energy
 J. Gilleland, GA Technologies
 F. Goldston, Princeton Plasma Physics Laboratory
 J. Gordon, TRW
 M. Gottlieb, Grumman
 H. Griem, University of Maryland
 R. Gross, University of Columbia
 J. Grover, Hanford Engineering Development Laboratory
 G. Haas, Department of Energy/Office of Fusion Energy
 R. Haselton, Oak Ridge National Laboratory
 R. Hawryluk, Princeton Plasma Physics Laboratory
 R. Hazeltine, University of Texas
 C. Henning, Lawrence Livermore National Laboratory
 N. Hershkowitz, University of Wisconsin
 D. Hitchcock, Department of Energy/Office of Fusion Energy
 A. Hoffman, Spectra Technology
 D. Hoffman, Oak Ridge National Laboratory
 T. James, Department of Energy/Office of Fusion Energy
 M. Johnson, Department of Energy/Princeton Operations Office
 A. Katz, Department of Energy/Office of Fusion Energy
 R. Krakowski, Los Alamos National Laboratory
 G. Kulcinski, University of Wisconsin
 J. Leiss
 P. Liewer, California Institute of Technology
 R. Linford, Los Alamos National Laboratory
 J. Lyon, Oak Ridge National Laboratory

B. Logan, Lawrence Livermore National Laboratory
 K. McGuire, Princeton Plasma Physics Laboratory
 J. Maniscalco, TRW
 D. Markevich, Department of Energy/Office of Fusion Energy
 W. Marton, Department of Energy/Office of Fusion Energy
 K. Matson, PSE&G Research Corporation
 D. Meade, Princeton Plasma Physics Laboratory
 G. Miley, University of Illinois
 R. Moir, Lawrence Livermore National Laboratory
 O. Morgan, Oak Ridge National Laboratory
 P. Murray, Westinghouse
 G. Nardella, Department of Energy/Office of Fusion Energy
 G. Navratil, Columbia University
 D. Nelson, Department of Energy/Office of Fusion Energy
 T. Ohkawa, GA Technologies
 R. Parker, Massachusetts Institute of Technology
 Y. Peng, Oak Ridge National Laboratory
 L. Perkins, Lawrence Livermore National Laboratory
 S. Piet, EG&G Idaho
 M. Porkolab, Massachusetts Institute of Technology
 R. Post, Lawrence Livermore National Laboratory
 R. Post, Massachusetts Institute of Technology
 S. Prager, University of Wisconsin
 R. Price, Department of Energy/Office of Fusion Energy
 D. Priester, Department of Energy/Office of Fusion Energy
 L. Price, Oak Ridge National Laboratory
 W. Rasin, Duke Power
 J. Rawls, GA Technologies
 T. Reuther, Department of Energy/Office of Fusion Energy
 M. Roberts, Department of Energy/Office of Fusion Energy
 F. Ribe, University of Washington
 M. Rosenbluth, University of Texas
 M. Rosenthal, Oak Ridge National Laboratory
 A. Rowcliffe, Oak Ridge National Laboratory
 P. Rutherford, Princeton Plasma Physics Laboratory
 W. Sadowski, Department of Energy/Office of Fusion Energy
 M. Saltmarsh, Oak Ridge National Laboratory
 J. Santarius, University of Wisconsin
 N. Sauthoff, Princeton Plasma Physics Laboratory
 J. Schmidt, Princeton Plasma Physics Laboratory
 K. Schultz, GA Technologies
 D. Schuresko, Oak Ridge National Laboratory
 M. Scott, Department of Energy/Chicago Operations Office
 C. Sege, Department of Energy/Chicago Operations Office
 T. Shannon, Fusion Engineering Design Center
 R. Shaw, Arête
 J. Sheffield, Oak Ridge National Laboratory
 A. Sherwood, Los Alamos National Laboratory
 J. Shohet, University of Wisconsin
 R. Siemon, Los Alamos National Laboratory
 D. Sigmar, Massachusetts Institute of Technology
 T. Simonen, Lawrence Livermore National Laboratory
 W. Stacey, Georgia Institute of Technology
 H. Staten, Department of Energy/Office of Fusion Energy

L. Steinhauer, Spectra Technology
P. Stone, Department of Energy/Office of Fusion Energy
D. Swain, Oak Ridge National Laboratory
R. Taylor, University of California-Los Angeles
M. Tillack, University of California-Los Angeles
A. Todd, Grumman Corporation
J. Turner, Department of Energy/Office of Fusion Energy
H. Weitzner, New York University
H. Williams, Los Alamos National Laboratory
J. Willis, Department of Energy/Office of Fusion Energy
K. Wilson, Sandia National Laboratory-Livermore
H. Woodson, University of Texas
B. Wright, Los Alamos National Laboratory
H. Yoshikawa, Hanford Engineering Development Laboratory