PPG-821
UCLA-ENG-84-30

FINESSE

A Study of Issues, Experiments and Facilities for Fusion Nuclear Technology Research & Development

Interim Report

Volume I

October 1984

Center for Plasma Physics and Fusion Engineering
University of California, Los Angeles
Los Angeles, California

Printed in the United States of America

Available from:

National Technical Information Service
U.S. Department of Commerce
5285 Port Royal Road
Springfield, Virginia 22161

Price: Printed Copy
Microfiche A01

Distribution Category:
Magnetic Fusion (UC-20)

University of California, Los Angeles
PPG-821
UCLA-ENG-84-30

FINESSE: A Study of the Issues, Experiments and Facilities for Fusion Nuclear Technology Research and Development

Interim Report

Volume I

Contributors

M. Abdou, UCLA
C. Baker, ANL
T. Bauer, ANL
G. Bell, TRW
D. Berwald, TRW
M. Billone, ANL
J. Blanchard, UCLA
R. Burke, HEDL
R. Conn, UCLA
J. Crocker, EG&G
J. Davis, MDAC
G. Deis, EG&G
R. DiMelfi, UCLA
J. Doggett, LLNL
M. Fenstermacher, TRW
J. Garner, TRW
P. Gierszewski, UCLA/CFFTP
J. Gordon, TRW
S. Grotz, UCLA
J. Grover, HEDL
C. Gung, UCLA
G. Hollenberg, HEDL
J. Holmes, HEDL
D. Jassby, PPPL
K. Kleefeldt, UCLA/KfK
T. Kraft, ANL
Y. Liu, ANL
H. Madarame, UCLA/U. of Tokyo
L. Masson, EG&G
L. Miller, EG&G
G. D. Morgan, MDAC
R. Nygren, ANL
E. Opperman, HEDL
G. Orient, UCLA
Y. Oyama, UCLA/JAERI
M. Perrin, MDAC
R. Puigh, HEDL
D. Reuster, MDAC
D. Rose, ANL
Y. Seki, UCLA/JAERI
K. Shin, UCLA/Kyoto U.
W. Steele, TRW
J. Straalsund, HEDL
J. Szabo, UCLA
K. Taghavi, UCLA
K. Thomassen, LLNL
M. Tillack, UCLA
R. Whitley, TRW
M. Youssef, UCLA

October 1984

FINESSE: A Study of the Issues, Experiments and Facilities for Fusion Nuclear Technology Research and Development

Interim Report

Volume I

Task Leaders

M. Abdou, UCLA: Principal Investigator

D. Berwald, TRW: Development Scenarios, Mirror Facilities

M. Billone, ANL: Tritium Extraction Modelling

J. Davis, MDAC: Failure Modes

G. Deis, EG&G: Fission Reactor Utilization

P. Gierszewski, UCLA/CFFTP: Testing Needs, Testing Requirements

G. Hollenberg, HEDL: Solid Breeders

J. Holmes, HEDL: Point Neutron Sources

D. Jassby, PPPL: Tokamak Facilities

G. Morgan, MDAC: Reference Generic Designs

R. Nygren, ANL: Non-Neutron Test Stands

J. Straalsund, HEDL: Materials, Interactive Effects

M. Tillack, UCLA: Issues, Testing Requirements

R. Whitley, TRW: Availability Modelling

VOLUME I

TABLE OF CONTENTS

Page

1. INTRODUCTION 1-1

2. SUMMARY

2.1 Overall Conclusions 2-1

2.2 Fusion Nuclear Issues 2-11

2.2.1 Introduction 2-11

2.2.2 Issues Tables and Summaries 2-11

2.2.3 Statement of Critical Issues 2-13

2.3 Survey of Experimental Needs 2-18

2.3.1 Introduction 2-18

2.3.2 Test Categories 2-19

2.3.3 Testing Needs Summary 2-21

2.4 Requirements of the Experiments 2-24

2.4.1 Introduction 2-24

2.4.2 Blanket Experiment Requirements and Engineering Scaling 2-24

2.4.2.1 Introduction 2-24

2.4.2.2 Liquid Metal Blanket Test Requirements 2-25

2.4.2.3 Solid Breeder Blanket Test Requirements 2-36

2.4.3 Failure Modes 2-43

2.4.4 Fluence Goals 2-49

2.4.5 Neutronics Experiments 2-55

2.4.5.1 Test Device Operating Condition Requirements 2-55

2.4.5.2 Test Module Requirements 2-56

2.4.6 Test Matrix Considerations 2-65

2.5 Non-Fusion Facilities 2-69

2.5.1 Introduction 2-69

2.5.2 Non-Neutron Test Stands 2-69

2.5.3 Point Neutron Sources 2-70

2.5.3.1 Introduction 2-70

2.5.3.2 Point Neutron Sources for Fusion Experiments 2-70

2.5.3.3 Point Source Potential 2-72

2.5.3.4 Conclusion.. 2-73
2.5.4 Fission Reactors.. 2-75
2.6 Fusion Facilities for Nuclear Experiments.. 2-82
2.6.1 Introduction.. 2-82
2.6.2 Tandem Mirror Test Facilities.. 2-83
2.6.2.1 Test Facility Options.. 2-83
2.6.2.2 Resolving the FINESSE Nuclear Issues in a Tandem Mirror Nuclear Test Facility.. 2-89
2.6.3 Tokamak Test Facilities.. 2-95
2.6.3.1 Objectives and Requirements.. 2-95
2.6.3.2 Design Approach and Principal Features.. 2-96
2.6.3.3 Concept Description.. 2-97
2.6.3.4 Risks and Uncertainties of the Reference Approach..2-101
2.6.3.5 Conclusions..2-102
2.6.4 Availability Considerations..2-102
2.6.4.1 Confidence Levels in Component Availability..2-103
2.6.4.2 The Potential Impact of Reliability Development Testing..2-106
2.7 Fusion Research and Development Scenarios..2-111
2.7.1 Planning Considerations..2-111
2.7.2 Fusion Experimental and Test Facilities..2-115
2.7.3 Fusion Development Pathways and Evaluations..2-118
2.7.4 International Implications..2-121
References..2-124

CHAPTER 1

INTRODUCTION

1. INTRODUCTION

Fusion is one of a very limited number of options for a renewable energy source that can sustain an industrial society for a long period of time. Bringing the attractive potential of fusion into realization requires challenging advances in science and technology. Many critical advances are required in the area of fusion nuclear technology.

A fusion energy system consists of: (a) plasma, (b) plasma support components (magnets, vacuum, auxiliary heating), and (c) nuclear components. The primary functions of the nuclear components are: (1) fuel generation and processing, (2) energy extraction and conversion, and (3) radiation protection of personnel and components. The primary nuclear components and other components affected by the nuclear environment are shown on Table 1-1. Most of the world effort on fusion over the past three decades has focused on plasma physics research and plasma confinement experiments. The technical progress to date in plasma confinement has been excellent. Some progress has also been made in plasma supporting technologies as needed for the plasma confinement experiments. In contrast, the resources devoted to fusion nuclear technology research and development (R&D) in the world fusion program have been very limited.

Table 1-1. Nuclear Components and Other Components Affected by the Nuclear Environment

Blanket
Shield
Plasma Interactive and High Heat Flux Subsystems:
First Wall
Impurity Control
RF Antennas, Launchers and Waveguides
Tritium and Vacuum Systems
Instrumentation and Control
Magnets
Remote Maintenance
Heat Transport and Power Conversion

The promise of fusion is so great that a comprehensive and accelerated R&D program to generate a quantitative knowledge and experience base is necessary to permit a quantitative judgement of the potential of fusion as a viable, practical, and attractive energy source. Nuclear technology is a critical element in such a program since it has many of fusion's remaining unresolved issues. These issues relate to: (a) feasibility, a primary acceptance criterion for the scientific and technological communities; (b) economics, a primary acceptance criterion for the utility industry; and (c) safety and environmental impact, a crucial acceptance criterion for the public.

The development of fusion nuclear technology is particularly challenging for several reasons:

(a) The technical complexity of the issues poses a high degree of intellectual challenge requiring advances in several disciplines of science and engineering that are at the forefront of knowledge. These disciplines include materials, chemistry, nuclear physics, thermodynamics, fluid mechanics, electromagnetics, magnetohydrodynamics, nuclear engineering, mechanical engineering, and chemical engineering.

(b) Fusion nuclear development appears to be relatively expensive, primarily because neutrons are required in many key experiments.

(c) Long lead times will be required to perform the necessary experiments and obtain an adequate data base.

(d) New and sophisticated experimental facilities are required. Presently available experimental facilities provide important information, and there is a clear need to continue to use them. However, they are not sufficient to satisfy all the testing needs. In particular, the unique and complex fusion environment can be obtained only in a fusion facility. The characteristics, cost, benefits, and risks of such a facility require careful evaluation as part of the overall scenarios for fusion development.

Because of the importance of fusion nuclear science and technology, the Office of Fusion Energy (OFE) in the Department of Energy (DOE) initiated a new study, called FINESSE, in November of 1983. The general objective of FINESSE is to investigate the technical and programmatic issues in the R&D of fusion nuclear science and technology. The study is led by UCLA and involves major organizations from the U.S.: Argonne National Laboratory (ANL), EG&G Idaho (EG&G), Hanford Engineering Development Laboratory (HEDL), McDonnell Douglas Astronautics Company (MDAC), and TRW Inc. (TRW). Major support is also being provided to the study by the Lawrence Livermore National Laboratory (LLNL) and Princeton Plasma Physics Laboratory (PPPL) as the lead organizations for mirrors and tokamaks, respectively.

An important aspect of FINESSE is significant international participation. A number of experts from major organizations in Canada, Japan, and West Germany, as shown in Table 1-2, have been directly participating in FINESSE. This international participation is particularly important because:

(a) All world fusion programs face the same issues. Therefore, all countries can benefit from investigating issues and approaches to fusion nuclear development.

Table 1-2. Organizations Participating in FINESSE

UNITED STATES
Primary
University of California, Los Angeles
Argonne National Laboratory
EG&G Idaho, Inc.
Hanford Engineering Development Laboratory
McDonnell Douglas Astronautics Company*
TRW, Inc.
Support
Lawrence Livermore National Laboratory
Princeton Plasma Physics Laboratory
CANADA, EUROPE AND JAPAN
Canadian Fusion Fuels Technology Project
Japan Atomic Energy Research Institute
Karlsruhe Nuclear Research Center (KfK)
University of Kyoto, Japan
University of Tokyo, Japan

*MDAC effort supported by the Electric Power Research Institute.

(b) The best prospects for international cooperation are in fusion nuclear technology R&D. Facilities, by their nature, tend to be user-oriented and a diversity of concepts can be tested in the same facility. Therefore, many countries can share the cost and benefits of the facilities without necessarily agreeing on the same design concepts.

An Advisory Committee consisting of senior members of the fusion community has provided an excellent mechanism for community-wide input to FINESSE on a continuing basis. The Advisory Committee membership is shown in Table 1-3.

The FINESSE technical effort has been structured into six major tasks shown in Table 1-4.

Task I is concerned with identifying, characterizing, and prioritizing the key issues in fusion nuclear technology. The focus has been on those issues whose resolution requires new knowledge through experiments. This task has been completed and is the subject of Chapter 3 of this report.

Task II is one of the largest and most important tasks in FINESSE and is concerned with identifying and quantifying testing needs for fusion nuclear technology. In FINESSE, the word "test" is used in a generic sense to refer to a process of obtaining information through physical experiment and measurement, i.e., not through design analysis or computer simulation. In Part A of Task II, the testing needs to resolve the key fusion nuclear issues have been surveyed and characterized as to the type of information needed and overall requirements on the environmental conditions of the experiments. The results of Task II.A are presented in Chapter 4.

Part B of Task II is focused on developing quantitative testing requirements, particularly for multiple interaction and integrated tests. Realistic cost constraints on testing facilities, including fusion devices, dictate that tests be carried out under scaled down conditions (e.g., the power density in test facilities will be much lower than in demonstration and commercial reactors). The evolving technical discipline of developing meaningful act-alike tests at reduced test facility parameters, commonly known as engineering scaling, has received much attention in FINESSE as shown in Chapters 5, 6 and 7. This work is of critical importance to the many tradeoffs between cost and benefit that must be considered with respect to the large number of testing

Table 1-3. FINESSE Advisory Committee

Charles C. Baker, Chairman (ANL)
Everett C. Bloom (ORNL)
John W. Davis (MDAC)
James J. Holmes (HEDL)
Robert A. Krakowski (LANL)
James A. Maniscalco (TRW)
John A. Schmidt (PPPL)
Kenneth R. Schultz (GA)
Thomas E. Shannon (FEDC)
Keith I. Thomassen (LLNL)

Table 1-4. FINESSE Principal Technical Tasks

I. Identification of Issues

II. Investigation of Testing Needs
   A. Survey of Testing Needs
   B. Quantifying Key Testing Requirements
   C. Test Matrix
   D. Summary of Test Requirements and Priorities

III. Evaluation of Experience from Other Technologies
   A. Fission
   B. Aerospace

IV. Survey and Evaluation of Facilities (Emphasis on Neutron Producers)
   A. Non-Fusion Facilities (Point Neutron Sources, Fission Reactors, Test Stands)
   B. Fusion Devices (Mirrors, Tokamaks, Alternate Concepts)

V. Comparative Evaluation of Facilities and Development Scenarios

VI. Recommendations on Fusion Nuclear Technology Development needs and testing facility options. The problems of engineering scaling have proved to be complex and have required a great deal of analysis to deepen our understanding of the testing issues. This analysis is presented in Chapter 5 for solid breeder type blankets and in Chapter 6 for the class of self-cooled liquid metal blankets. Experiments aimed specifically at verification of neutronics methods and data have special scaling issues which are treated in Chapter 7. Although FINESSE is concerned with and has addressed the key issues for all fusion nuclear components, the detailed quantification of the testing requirements has been attempted only for the blanket system. Work on Task II.B will continue into the second year.

Fluence goals represent an important aspect of test requirements which are particularly difficult to quantify but have a substantial impact on the cost of testing. The question of what we learn from testing as a function of neutron fluence is investigated in Chapter 8 as a basis for addressing fluence goals.

The survey of key testing needs and quantitative test requirements provides necessary input to developing a test plan that prioritizes the experimental needs in terms of type, number, time frame, facilities, and other specifics of a development path. Such a test plan will be developed during the second year, but initial test matrix considerations are discussed in Chapter 9. Issues and test requirements associated with instrumentation and control are also addressed in Chapter 9.

While Tasks I and II have been concerned with investigating the fusion nuclear issues and testing needs, Task IV has focused on evaluating facilities in which these tests can be performed. Non-fusion and fusion facilities have been considered. Chapters 10, 11 and 12 provide an initial examination of the capabilities and limitations of non-neutron test stands, accelerator-based neutron sources, and fission reactors.

A key conclusion of the first year of FINESSE effort is that testing in non-fusion facilities, while essential, is not sufficient to satisfy the nuclear technology development requirements. Many critical multiple-interaction and integrated tests require a fusion facility. Possible options for such a fusion test facility are examined within Task IV. The results of an initial examination of mirrors and tokamaks as test facilities are presented in Chapters 13 and 14, respectively.

A key problem identified for these test facilities is the achievable device availability and its impact on the testing program. This is a particularly serious problem for those fusion test facilities that produce such a large amount of fusion power that they require their own tritium breeding blankets. The testing device availability analysis is given in Chapter 15.

Broader and deeper examination of the complex cost/benefit/risk issues in fusion nuclear technology is planned for the second year. Task II.B will continue to provide measures of benefits (usefulness of test information) as functions of testing capabilities, while Task IV will quantify the costs and physics/technology risks of testing facilities as functions of testing capabilities. This information will be utilized in Task V to carry out comparative evaluations of facilities and development scenarios.

A principal objective of FINESSE is the development of recommendations regarding the types and sequences of test facilities that maximize benefits and minimize cost for fusion nuclear technology development. (It should be noted here again that, in FINESSE, the term "test" is used in a generic sense to refer to a spectrum of physical experiments and measurements, e.g. basic property measurements, single- or multiple-effect experiments, rather than a full size component verification.) During the first year of effort, the <u>total</u> testing requirements for fusion nuclear technology have been addressed (Tasks I and II). Task IV evaluated possible options for non-fusion and fusion facilities. To accomplish the above objective of recommending where the various <u>parts</u> of the testing requirements can be optimally performed, it is necessary to explicitly consider possible scenarios, or pathways, for overall fusion research and development. A preliminary screening evaluation of a number of pathways was performed and is summarized in Chapter 16.

Several important supporting studies and analysis performed as part of the FINESSE program are described in the appendices of this report. Task III has focused on evaluation of experience from the fission and aerospace technologies as to lessons learned and approaches to testing. This task has been completed and no further work is planned. The fission experience is summarized in Appendices A and B, while the aerospace experience is reviewed in Appendix C.

In performing the various tasks of FINESSE, it was necessary to define a number of representative design concepts and to perform detailed analyses in many areas to characterize the operating environment of fusion nuclear components in full scale reactors. Such a characterization for key nuclear components is summarized in Appendix D.

An important objective of fusion nuclear experiments will be to obtain information on failure modes and rates. The present understanding of failure modes in fusion nuclear components is extremely limited. An attempt to gain sufficient understanding in this area in order to identify subscale test requirements is presented in Appendix E.

Additional details on experiments in fission breeder test facilities are given in Appendix F. Detailed models of the physics performance of tandem mirror test facilities were developed in support of task IV and are briefly described in Appendix G.

The present phase of FINESSE effort started in November 1983 and is scheduled for completion in October 1985. The work completed during the first year (November 1983 to October 1984) and the effort planned for the second year (November 1984 to October 1985) of the study are shown for each of the six tasks in Table 1-5. This document is an interim report on the results obtained during the first year of the study. The purpose of this interim report is to provide members of the fusion community with sufficient information, results, and conclusions to stimulate feedback to FINESSE at the midpoint of the study. A complete documentation of the two-year effort will be made in a final formal report at the end of the study. The primary purpose of this interim report is to provide a fast mechanism for relaying the essence rather than formal documentation of the first year's findings. As such, no particular attempt has been made to rigorously edit or "word engineer" the content of the report in order to conserve resources and expedite issuance of the report.

Table 1-5. Completed and Planned Technical Efforts in FINESSE

Task	Year 1 (Completed)	Year 2
	November 1983 through October 1984	November 1984 through October 1985
Task I: Issues	Completed identification and characterization of fusion nuclear issues.	Only an update of issues as needed is planned.
Task II: Investigation of Testing Needs	A. Survey of testing needs completed. B. Analysis of component behavior performed. An approach to engineering scaling completed and applied to key cases. C. Approach to test matrix definition developed.	1. Continue investigation and applications of engineering scaling. 2. Scope designs of act-alike test modules to check suitability for providing the needed information on interactive and integrated tests. 3. Scope near-term single and multiple effect experiments. 4. Develop details of test matrix; prioritize testing needs, develop information for test plan.
Task III: Evaluation of Experience from Other Technologies	Completed evaluation of experience from fission and aerospace industries.	No further effort is planned.

Table 1-5. Completed and Planned Technical Efforts in FINESSE (contd.)

Task	Year 1 (Completed)	Year 2
	November 1983 through October 1984	November 1984 through October 1985
Task IV: Evaluation of Facilities (Emphasis on Neutron Producers)	A. Non-Fusion Facilities: Work on point neutron sources and fission reactors completed; work on test stands initiated. B. Fusion Facilities: Brief survey of possible options for mirrors and tokamaks.	1. Complete effort on non-nuclear test stands. 2. Intensive effort on exploring options, innovative ideas, and tradeoffs for fusion test facilities to minimize cost and maximize test benefits consistent with acceptable physics and technology risks.
Task V: Comparative Evaluation of Facilities and Development Scenarios	Task initiated.	1. Compare capabilities, cost, and risk of various non-fusion and fusion facilities. 2. Develop a number of scenarios for fusion development. 3. Develop several possible paths of fusion nuclear technology development. 4. Develop and compare options for testing plan. 5. Compare and evaluate scenarios and paths.
Task VI: Recommendations on Fusion Nuclear Technology Development	No effort.	Translate all results from FINESSE into a set of recommendations on technology development.

CHAPTER 2

SUMMARY

TABLE OF CONTENTS

Page

2.1 Overall Conclusions.. 2-1

2.2 Fusion Nuclear Issues.. 2-11
2.2.1 Introduction.. 2-11
2.2.2 Issues Tables and Summaries.. 2-11
2.2.3 Statement of Critical Issues.. 2-13

2.3 Survey of Experimental Needs.. 2-18
2.3.1 Introduction.. 2-18
2.3.2 Test Categories.. 2-19
2.3.3 Testing Needs Summary.. 2-21

2.4 Requirements of the Experiments.. 2-24
2.4.1 Introduction.. 2-24
2.4.2 Blanket Experiment Requirements and Engineering Scaling..... 2-24
2.4.2.1 Introduction.. 2-24
2.4.2.2 Liquid Metal Blanket Test Requirements............. 2-25
2.4.2.3 Solid Breeder Blanket Test Requirements............ 2-36
2.4.3 Failure Modes.. 2-43
2.4.4 Fluence Goals.. 2-49
2.4.5 Neutronics Experiments.. 2-55
2.4.5.1 Test Device Operating Condition Requirements....... 2-55
2.4.5.2 Test Module Requirements.. 2-56
2.4.6 Test Matrix Considerations.. 2-65

2.5 Non-Fusion Facilities.. 2-69
2.5.1 Introduction.. 2-69
2.5.2 Non-Neutron Test Stands.. 2-69
2.5.3 Point Neutron Sources.. 2-70
2.5.3.1 Introduction.. 2-70
2.5.3.2 Point Neutron Sources for Fusion Experiments....... 2-70
2.5.3.3 Point Source Potential.. 2-72

2.5.3.4 Conclusion.. 2-73
2.5.4 Fission Reactors.. 2-75

2.6 Fusion Facilities for Nuclear Experiments.......................... 2-82
2.6.1 Introduction.. 2-82
2.6.2 Tandem Mirror Test Facilities.. 2-83
2.6.2.1 Test Facility Options.. 2-83
2.6.2.2 Resolving the FINESSE Nuclear Issues in a Tandem Mirror Nuclear Test Facility.. 2-89
2.6.3 Tokamak Test Facilities.. 2-95
2.6.3.1 Objectives and Requirements.. 2-95
2.6.3.2 Design Approach and Principal Features.. 2-96
2.6.3.3 Concept Description.. 2-97
2.6.3.4 Risks and Uncertainties of the Reference Approach.. 2-101
2.6.3.5 Conclusions.. 2-102
2.6.4 Availability Considerations.. 2-102
2.6.4.1 Confidence Levels in Component Availability........ 2-103
2.6.4.2 The Potential Impact of Reliability Development Testing.. 2-106

2.7 Fusion Research and Development Scenarios.......................... 2-111
2.7.1 Planning Considerations.. 2-111
2.7.2 Fusion Experimental and Test Facilities.. 2-115
2.7.3 Fusion Development Pathways and Evaluations.. 2-118
2.7.4 International Implications.. 2-121

References.. 2-124

2. SUMMARY

The purpose of this chapter is to provide a summary of the work of FINESSE performed during the period November 1983 through October 1984. It is anticipated that some readers will not be interested in all the technical details presented in this relatively long report. Therefore, this chapter is kept relatively brief but thorough and self-contained and is intended for all readers as an overview of the key results from the various technical areas of FINESSE. This should provide the reader with a clear context in which specific results in selected sections of the report can be interpreted.

2.1 Overall Conclusions

The development of fusion nuclear technology presents new, unique and challenging questions to many fields of science and engineering that have not been encountered before in the development of other technologies. One reason is that the fusion environment experienced by the nuclear components involves the simultaneous presence of plasma particles, neutrons, photons, magnetic field, surface and bulk heating, tritium and vacuum. A second reason is that many fusion nuclear components perform multiple new and unique functions, e.g., simultaneous tritium production and extraction, and energy conversion and extraction in the blanket; or heat removal and plasma ash removal in the impurity control and exhaust system. A third reason is that the integration of components into a fusion system results in many interactions among components.

The multiple functions that the fusion nuclear components must provide under new and unique conditions result in new phenomena and fundamental changes in the characteristics of previously known phenomena. There is presently neither an adequate data base nor sufficient understanding to characterize these new or changed phenomena. Thus, attempts to select material and design options and to predict the performance of fusion nuclear components suffer from large uncertainties caused by insufficient knowledge.

These large uncertainties result in many critical issues for fusion that relate to: 1) feasibility, a primary acceptance criterion for the scientific and technological communities; 2) economic potential, a primary acceptance criterion for industry and utilities; and 3) safety and environmental impact, a crucial acceptance criterion for the public.

During the first year of FINESSE, the principal nuclear technology issues have been identified, characterized and classified according to their potential impact, level of concern, and importance of environmental conditions (e.g., neutrons, magnetic field). These issues are summarized in Section 2.2 and are documented in detail in Chapter 3.

New knowledge is required to understand and resolve these known and unknown fusion nuclear issues. This new knowledge can be acquired only through new experiments accompanied by intensive theoretical and modelling efforts. FINESSE has assessed the types of experiments, and the environmental conditions that must be provided in these experiments in order to resolve the fusion nuclear issues. In addition, the capabilities and limitations of existing facilities and the needs for new facilities have been evaluated. The relatively large cost, the long lead time, and the complexity of the issues for these experiments require detailed examination of priorities and careful planning of experiments and experimental facilities.

The type of experiments required in fusion nuclear technology development can be classified into: 1) basic, 2) single effect, 3) multiple effect/multiple interaction, 4) partially integrated, and 5) integrated tests. Basic tests are to obtain property data and can be performed in available standard laboratory facilities. Single effect experiments are to explore phenomena and are aimed at a single effect, e.g., electromagnetic response of bonded materials to a transient magnetic field. Some of the required experiments can be performed in present facilities, but others require new facilities. Some irradiation-effects experiments in which a sample is exposed to a neutron field to examine fluence effects fall under the class of single-effect tests. Experiments in fission reactors suffer from spectral differences, and there is a definite need for experiments with 14 MeV neutrons, at least for calibration of results from fission and ion irradiation. There is presently no appropriate facility anywhere in the world. FMIT has been proposed to serve this purpose, but the future of this project is highly uncertain. Without FMIT, there will be a serious gap in irradiation data that must be filled by a new facility. Other than point neutron sources, the only type of 14 MeV neutron-producing facility is a DT fusion device.

Multiple effect/multiple interaction experiments are aimed at exploring the combined effects of two or more environmental conditions and the interactions among two or more physical elements of a component. The fusion environment results in many new multiple effects/multiple interactions that require exploration. For example, corrosion is known in other technologies to depend on temperature and velocity of the fluid, but in the fusion environment a strong dependence of corrosion on the magnetic and neutron fields is also predicted. Thus, reliable data on the corrosion of structural materials by liquid metals in the fusion environment cannot be obtained merely from "classical" corrosion loops but requires new experiments in which magnetic field, heating, velocity and geometry are properly simulated.

Another example is MHD effects in self-cooled liquid metal blankets. Results obtained in FINESSE predict complex interrelations among the magnetic field, fluid flow, heat transfer, bulk heating, surface heating, geometry, pressure drop and stresses. Thus, MHD effects cannot be understood from simple "classical" types of experiments in which the magnetic field is the only imposed environmental condition. Rather, fusion needs experiments in which many or all of the various interactions just mentioned are simulated.

Multiple effect/multiple interaction experiments generally require relatively larger size and are generally much more costly than single-effect experiments. New facilities and upgrades of present facilities are required for these multiple effect experiments.

Some of the multiple effect tests require neutrons as part of the experiment environment. These are experiments in which bulk heating, radiation effects, and/or specific reactions, e.g., Li(n,t), are important. The size of such experiments is relatively large, at least orders of magnitude larger than the size of samples normally used with point neutron sources. The only two types of "bulk" neutron-producing facilities are fission reactors and fusion devices.

The issues related to the benefits and limitations in utilizing fission reactors for fusion nuclear experiments were evaluated. Fission reactors are found useful for a number of experiments, and their utilization should be maximized. They provide the only present means for obtaining neutrons in a significant volume. However, fission reactors have limitations on spectrum, flux level, size of test element, number of test locations and simulation of the non-nuclear aspects of the fusion environment. Hence, fission reactors cannot substitute for fusion testing for many of the interactive tests and are not suitable for integrated tests.

Accelerator-based neutron sources, fission reactors and non-nuclear test stands can satisfy an important part of fusion nuclear technology research and development. However, it appears that a fusion device is required to meet critical needs in R&D for fusion nuclear technology. There are four key reasons why a fusion device is needed for many of the multiple effect/multiple interaction and integral experiments:

1) Many of these experiments require the size of the test article to be 1 m x 1 m x 0.5 m or much larger, e.g., for some of the liquid metal tests. Such a size can be accommodated only in a fusion device.

2) The total volume, surface area and power density for the critical needs in the test matrix correspond to a steady-state neutron source of $> 10^{19}$ $n \cdot s^{-1}$, or > 20 MW of 14 MeV neutron power over ~ 10 m^2 of surface area. This requirement can be satisfied only in a fusion device.

3) Only a fusion device can simultaneously simulate all of the key environmental conditions: neutrons, electromagnetics, plasma particles, tritium and vacuum.

4) A fusion device provides the correct neutron spectrum produced by a 14 MeV neutron source and the complex process of neutron and gamma ray slowing down and backscattering.

A fusion device is necessary for many critical fusion nuclear engineering research and development experiments. However, there are presently a number of issues concerning such a testing device. One issue is the cost. This tends to be relatively high on a single investment basis but not on a per neutron basis. The cost of the device generally increases as the requirements on a number of key parameters increases (e.g., neutron wall load, fluence, test surface area). Reducing these parameters substantially below those typical of full-scale reactors, to keep the cost relatively low, leads to a decrease in the benefits of tests. The problem of obtaining meaningful test data at scaled down environmental conditions has been comprehensively addressed in FINESSE. Table 2.1-1 shows preliminary conclusions on requirements for the primary parameters of a fusion test device that appear at

Table 2.1-1 Preliminary Requirements on Key Parameters of a Fusion Engineering Research Facility

Wall Load
- Minimum: > 1 MW/m^2
- Substantial benefits: 2-3 MW/m^2
- Much higher wall loads can be extremely beneficial and will alter strategy (accelerated testing, more ambitious technology performance goals for fusion, etc.)

Surface Heat Load
- Critical for tests of first wall, solid breeder blankets, liquid metal blankets
- Tokamak blankets: > 20 W/cm^2
  Mirror blankets: < 20 W/cm^2
- Methods to enhance surface heat flux in fusion test facilities, are important

Plasma Burn Cycle
- Pulsing sharply reduces the value of many tests
- Minimum burn time: > 500 s
- Maximum dwell time: < 100 s
- Prefer steady state

Minimum Continuous Time
- Many periods with 100% availability
- Duration of each period:
  Critical: several days
  Important: several weeks

Availability
- Minimum: 20%
- Substantial benefits: 50%

Fluence
- Fluence requirements will depend on whether a point neutron source or other means is available for high fluence material testing
- In general, component tests in the early stages of development are carried out to fluences lower than those for specimen tests
- In all cases, higher fluences are desirable but costly; modest fluences are still extremely valuable
- For component tests:
  Critical: 1-2 MW·yr/m^2
  Very Important: 2-4 MW·yr/m^2
  Important: 4-6 MW·yr/m^2
  Desirable: 6-10 MW·yr/m^2

Minimum Size of Test Assembly
- Interactive tests:
  ~ 0.2 m x 0.2 m x 0.1 m
- Integrated tests:
  1 m x 1 m x 0.5 m
  (Some liquid metal blanket designs tend to require larger size, sector scale)

Test Surface Area
- Critical: > 5 m^2
- Very Important: > 10 m^2
- Important: 15-20 m^2

present to be a reasonable compromise between the increased cost at high performance parameters and the reduction in the benefits at scaled down conditions.

The requirements in Table 2.1-1 indicate the importance of high power density (wall load ~ 2-3 MW/m^2), long plasma burntime (> 500 s) and surface area available for testing (~ 10-15 m^2) in a fusion test device. High fluence (4-10 MW•y/m^2) is important for near end-of-life prediction, but critical information about many interactive effects that are critical to feasibility issues can be learned at lower fluences (~ 1-2 MW•y/m^2).

The option for a fusion facility in which the needs of fusion nuclear technology can be fulfilled has to be considered in the context of scenarios for the overall fusion research and development. Preliminary evaluation screening of a number of scenarios has been performed. Other than cost and schedule, these scenarios seem to have two distinct options with respect to the next major device(s). In the first option, a single fusion device is built to perform both plasma and technology experiments. In the second option, two fusion devices are built; one is aimed at plasma experiments and the other is dedicated to fusion nuclear technology.

The first option has been considered extensively in the world program for the case in which the tokamak is the primary path to commercial reactors. Examples are INTOR, NET, FER and FED. The assessment concludes that high risks and high costs are concerns for this type of device. One specific problem of greatest concern in combining the physics and technology missions in a single device arises because of the inherent characteristics of conventional tokamaks. Plasma physics testing alone requires large fusion power (~ 300-600 MW) to achieve ignition and/or reasonably high wall loads (> 1 MW/m^2), but requires low fluence (< 0.01 MW•y/m^2). On the other hand, nuclear technology experiments require low fusion power (~ 20-50 MW) but high fluence (~ 2-10 MW•y/m^2). The combination of high power and high fluence in a single device leads to high costs and high risks because of several reasons, the most important of which is high tritium consumption (> 100 kg). Since such an amount of tritium is both unacceptably expensive (> $1 billion) and unavailable from external sources, the tokamak facility needs to have its own breeding blanket. Analysis shows that a breeding blanket without prior fusion testing is likely to result in such a low device availability that the risk of the device not achieving its mission during a reasonable operating life is very high.

The second option of two separate facilities for plasma and nuclear technology testing has been considered. The plasma device is aimed at examining long burn and ignition physics. Studying options for a plasma testing device is not within the scope of FINESSE but an example of such a device is the recent TFCX design. Since this device is not burdened with the nuclear technology requirements, it can be optimized to achieve the plasma physics objectives at relatively lower cost and lower risk than an INTOR-type facility. Although the cost of TFCX is predicted to be substantially more than one-half the cost of INTOR, it is possible that further optimization for "plasma testing" only may result in significantly lower cost option.

Possibilities for a fusion device dedicated to nuclear technology testing are being evaluated in FINESSE. A preliminary scoping study of the potential of tokamaks and mirrors as dedicated nuclear technology testing facilities has been completed.

The tokamak effort has attempted to minimize the physical size, fusion power, circulating power, and capital cost while maximizing the wall load and plasma burn time. This effort resulted in a conceptual design for a copper TF coil device that has 185 MW fusion power, ~ 1.15 MW/m^2 neutron wall load, ~ 1000 s pulse length, major radius of 2.55 m and an aspect ratio of 3.4. The device requires an average beta of $\sim 23\%$ and circulating power of ~ 190 MW. While such a device appears considerably more attractive than previous tokamak test facility designs, its captial cost, electrical power and annual tritium consumption requirements are at the higher ends of the acceptable range. Thus, further efforts are required to reduce the physical size and fusion power level while increasing the neutron wall loading. In addition, the physics risk associated with the selected plasma operating parameters needs evaluation.

The most suitable facility for fusion nuclear technology testing is a device in which the power and power density are decoupled. A device that produces 20-50 MW of fusion power at ~ 1-2 MW/m^2 wall load or higher is well suited for this purpose. The plasma can serve only as a neutron producer, and there are no other requirements on the plasma except steady-state or long-burn operation. For example, beam-driven plasmas are acceptable. Mirrors appear to offer an advantage in this area, and a number of possibilities for a mirror device have been evaluated. A facility of the TDF-type appears to satisfy most of the nuclear testing requirements with a capital cost roughly half of that for an INTOR-type facility. While the capital costs for mirror and tokamak devices that are nuclear-dedicated are comparable, the ability to keep the fusion power low in the mirror options results in substantial savings in the annual operating cost and ameliorates the tritium consumption and supply issue.

There is no single unique approach to fusion development that can presently be judged as the best. The approach to fusion development involves a myriad of complex technical, programmatic and financial issues. The FINESSE experience demonstrates that better insight into the merits and disadvantages of various approaches can be gained from identifying and characterizing the R&D requirements and evaluating the capabilities and limitations of the various options for experimental facilities. More effort is needed, e.g., as is planned in the second year of FINESSE, to deepen our understanding of the implications of various options for fusion research and development scenarios. At present, there appears to be some general conclusions from the fusion nuclear technology development viewpoint.

The general framework for fusion nuclear technology R&D, as shown in Table 2.1-2, has two distinct stages. In the first stage (from now to the mid to late 1990s), the fusion nuclear technology R&D program can and should utilize existing facilities (test stands, point neutron sources and fission reactors) to obtain information on materials properties, single effects, and many multiple interaction tests. Full utilization of the technical capabilities of available facilities requires expanding the financial resources allocated to fusion nuclear technology R&D in the world programs. In addition, there is a need to construct a number of new small scale facilities aimed at multiple interaction tests. An example at the upper end of such facilities is one in which many of the fusion environmental conditions, except neutrons, are simulated for a liquid metal blanket and its heat transport loop.

Planning for a second stage (after the mid-1990s) in fusion nuclear technology R&D must start now. In this second stage, experiments in a true fusion environment are required to address many of the feasibility and

Table 2.1-2 General Framework for Fusion Nuclear Technology Development

Now to Mid-1990s	After Mid-1990s
- Utilize existing facilities (test stands, point neutron sources, fission reactors). - Build a number of small-scale experimental facilities. - There may be a need for a partially integrated test facility (PITF), e.g., facility for liquid metal blanket and transport loop experiments in all relevant environmental conditions (vacuum, tritium, magnetic field) except neutrons	- Continue experiments in non-fusion facilities. - Engineering experiments in a fusion facility (possibly dedicated to nuclear technology).

attractiveness issues related to multiple interaction and integrated performance. While we understand the fusion nuclear technology testing requirements on such a facility, identifying the best option for a technically credible and relatively inexpensive fusion device requires further effort. It appears that there are substantial incentives to considering a fusion development scenario in which a fusion device is dedicated to nuclear technology R&D. This scenario is shown schematically in Fig. 2.1-1. The knowledge from the plasma and technology experiments shown in this figure should provide a sufficient data base to quantitatively judge the potential of fusion as an energy source. Successful completion of this milestone can then be followed by an engineering development and demonstration phase.

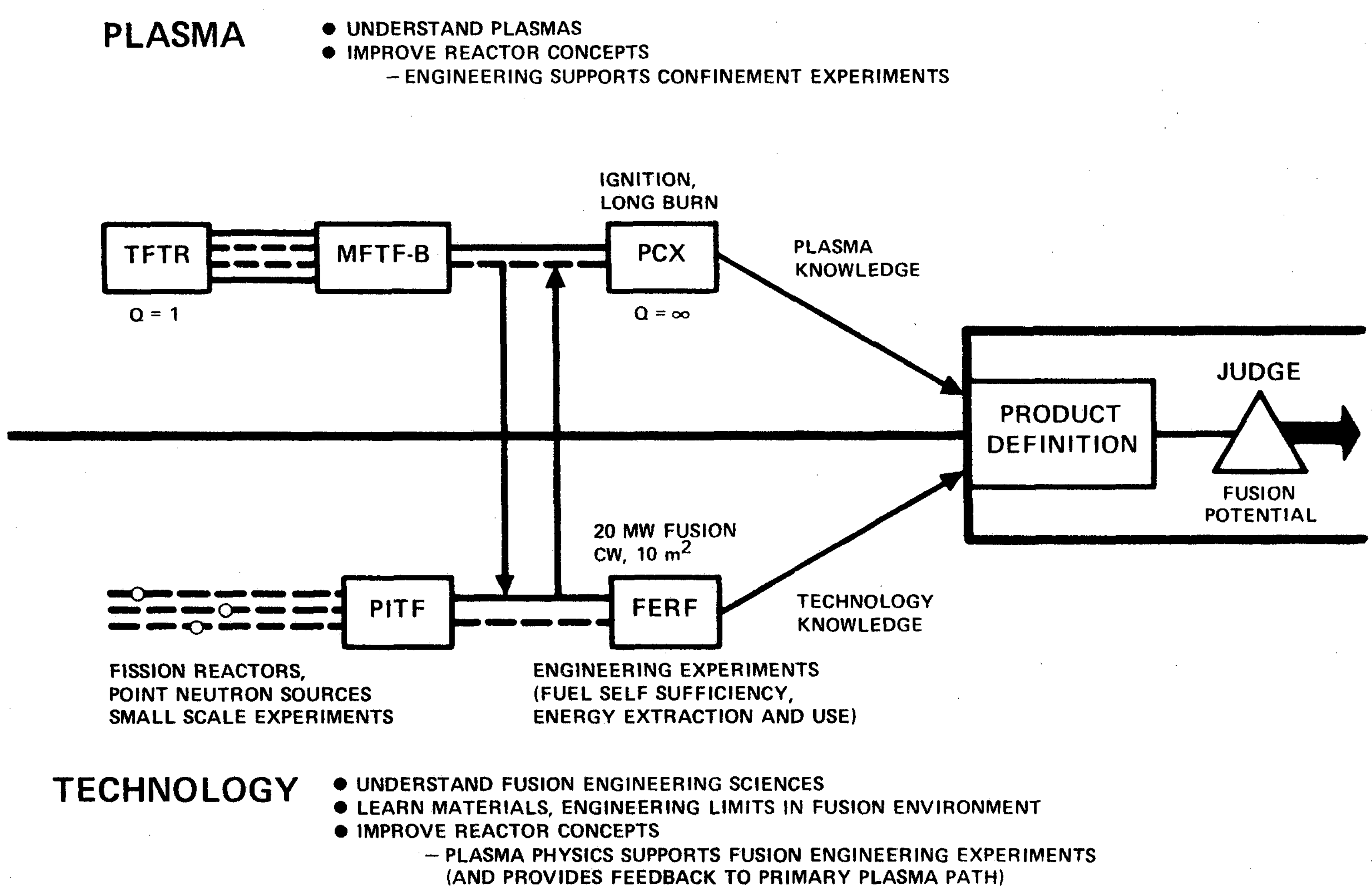


Figure 2.1-1 Key elements of a plausible approach to the next step in fusion reasearch and development.

2.2 Fusion Nuclear Issues

2.2.1 Introduction

A coherent program of engineering testing and nuclear technology development must address the key issues which most seriously impact the feasibility and attractiveness of fusion nuclear components. Identifying and generally characterizing these issues is the first step.

A concise, comprehensive list of testing issues resulted from a program-wide effort involving contributions from experts in materials science, structures, failure modes, thermal hydraulics, MHD, tritium recovery, systems integration, and many other disciplines. In Chapter 3, the issues are described in brief summaries and compiled in a table format which characterizes the issues, their relative importance, and general requirements for testing.

Generic examples of blankets were needed to focus the effort to identify the majority of the requirements on a fusion test facility. The number of blanket options was limited to liquid metal (Li and LiPb) and solid breeder (Li_2O and ternary ceramics) concepts. Inclusion of other concepts (e.g., molten salt) is not likely to substantially change the test requirements for a fusion facility. However, they need to be considered in determining near-term experimental programs.

2.2.2 Issues Tables and Summaries

Issues are defined by the presence of two necessary attributes: uncertainty and negative consequences. Seven potential impacts were defined under two main headings:

Feasibility Issues

- May Close the Design Window
- May Result in Unacceptable Safety Risk
- May Result in Unacceptable Reliability, Availability or Lifetime

Attractiveness Issues

- Reduced System Performance
- Reduced Component Lifetime
- Increased System Cost
- Less Desirable Safety or Environmental Implications

Feasibility issues, which may rule out a design on scientific grounds, are generally felt to be more serious than issues which only threaten to reduce the safety or economic potential of a design. By combining the level of uncertainty, the potential impact, and the degree to which the issue is design specific, a composite index is given for the overall level of concern. The most critical issues have the greatest need for testing; the remaining effort to quantify test requirements and choose test devices concentrates on these issues and how well they can be resolved.

In addition to the potential impact and overall level of concern, each issue has certain important environmental conditions which should be present in order to adequately resolve it through testing. The effect of neutrons is listed separately from the other parameters because the presence of neutrons most strongly affects the cost and other aspects of testing. Neutron effects are classified as: 1) bulk heating, 2) materials damage, or 3) specific reactions (such as tritium and helium production, transmutations, etc.).

The organization of the list strongly influences the statement of the issues. Various possible ways to present the issues include:

- according to technical discipline
- according to blanket function
- according to component affected.

It is believed that the organization according to component and subcomponent is the most appropriate method for a precise statement of testing issues. The reactor components affected by the nuclear environment include: the blanket, plasma interactive components, shield, tritium processing system, magnets, instrumentation and control. Within the blanket, subcomponents are further identified, including structure, coolant, and breeder. In order to address interactive issues, there are categories for coolant-structure interactions, breeder/multiplier interactions, and general blanket phenomena.

The complete list of issues contains approximately 120 specific technical items. An effort was made to keep a somewhat uniform level of detail in the definition of an issue. The adherence to a uniform standard of detail allows a meaningful comparison of the different reactor components. The blanket encompasses approximately half of the testing issues. Plasma interactive components (PIC) account for another 25%, and the other non-blanket components share the remaining 25%. The total number of specific testing issues identified as critical was 21: 15 in the blanket, 6 in PIC, and zero for the other components. Critical issues are defined as those issues which have an impact on component feasibility for a large class of designs. The critical issues from Section 3.2 are collected here and listed in Table 2.2-1.

2.2.3 Statement of Critical Issues

A complementary summary of the critical issues of fusion nuclear technology was compiled using a method of organization which stresses the key functional aspects of the fusion reactor which must be resolved through testing. These are listed in Table 2.2-2 and discussed in more detail here and in Section 3.3.

1. DT Fuel Cycle Self-Sufficiency

One function of the blanket is to breed enough tritium to fuel the plasma, accounting for the various loss mechanisms present. For many reactor concepts the margin in the tritium breeding ratio is not large enough to cover the uncertainties. Uncertainties exist in both the required and the achievable amount of tritium breeding. The required amount of tritium breeding is uncertain due to lack of data and models to reliably predict tritium inventory and behavior throughout the fuel cycle, including the plasma, blanket, and tritium processing systems. The achievable amount of tritium breeding is uncertain due to the variability in design choices and due to the limitations in accuracy of neutronics data and methods.

2. Thermomechanical Loading and Response of Blanket Components under Normal and Off-Normal Operation

Another function of the blanket is to safely and reliably convert nuclear energy to heat in an environment which includes high temperatures, high stresses, high magnetic fields, high radiation fields, etc. Design of a viable and reliable blanket is very difficult and many uncertainties remain to be resolved. The uncertainties involve both the sources of thermomechanical loading (e.g. disruptions, hot spots) and the structural responses (e.g. interaction of primary and secondary stresses, influence of swelling and creep). Liquid metal blankets have additional large uncertainties due to the effects of MHD on fluid flow, heat transfer, corrosion, and thermal and pressure stresses.

Table 2.2-1 Critical Testing Issues

BLANKET

- Uncertainties in Achievable Tritium Breeding Ratio
- Uncertainties in Required Tritium Breeding Ratio
- Changes in Properties and Behavior of Materials
- Effect of First Wall Heat Flux and Cycling on Fatigue or Crack Growth
- MHD Pressure Drop and Pressure Stresses
- MHD Effects on First Wall Cooling and Hot Spots
- MHD and Geometric Effects on Flow Distribution
- Mass Transport Rates and Consequences due to Corrosion
- Intragranular Tritium Diffusivity and Solubility
- Temperature Limits in Solid Breeder Materials
- Clad Corrosion from Li_2O Burnup Products
- Strain Accommodation by Creep and Plastic Flow in Li_2O Solid Breeders
- Swelling Driving Force in Li_2O
- Breeder/Structure Interface Heat Transfer
- Effectiveness of Tritium Permeation Barriers

Plasma Interactive Components

- Erosion and Redeposition
- Thermal Hydraulic Techniques
- Plasma Edge Temperature and Density Control
- Tritium Permeation and Inventory
- RF Launcher Performance Requirements
- RF Window and Feedthrough Performance

Table 2.2-2 Critical Fusion Nuclear Technology Development Issues

1. DT Fuel Cycle Self-Sufficiency
2. Thermomechanical Loading and Response of Blanket Components under Normal and Off-Normal Operation
3. Materials Compatibility
4. Identification and Characterization of Failure Modes and Rates
5. Tritium Inventory and Recovery in the Solid Breeder under Actual Operating Conditions
6. Tritium Permeation and Inventory in the Structure
7. In-Vessel Component Thermomechanical Response and Lifetime
8. Radiation Shielding: Accuracy of Prediction and Quantification of Radiation Protection Requirements
9. Accuracy and Survivability of Instrumentation and Control

3. Materials Compatibility

Materials compatibility of the structure, coolant, breeder, and tritium recovery fluid influences design limits, failure modes, safety and reliability. Limits on the maximum temperature allowed in the blanket are often determined by the allowable corrosion rates. Even within the allowable temperature window, materials interactions limit the maximum lifetime of the blanket by contributing to materials degradation and failure modes. Because of the possibility of mobilizing and transporting radioactive isotopes, materials compatibility is also a serious safety issue. Data needs for materials compatibility issues include basic materials interactions data and information on the interactions among materials in the fusion environment, which includes radiation, high magnetic field and bulk heating.

4. Identification and Characterization of Failure Modes and Rates

Knowledge of failure modes and rates is necessary because of their critical impact on the lifetime, economic potential, and safety of fusion components. Two of the failure modes suspected to be serious concerns include crack growth under irradiation and failure at welds and discontinuities. Experiments are required to examine these suspected failure modes. The most important information from experiments may be the identification of unforeseen failure modes in the unique fusion environment.

5. Tritium Inventory and Recovery in the Solid Breeder under Actual Operating Conditions

Tritium inventory is important because it influences the required breeding ratio and the safety risk of the blanket. Major uncertainties relate to both the fundamental tritium transport mechanisms in the solid breeder and purge, and the effect of the fusion environment, which includes irradiation, mechanical and materials interactions. Tritium transport within the solid breeder is very sensitive to the fabrication techniques and operating conditions, particularly the effect of radiation. The breeder temperature profile is particularly crucial because a relatively narrow window of operation is predicted, based on unreasonably high inventory at low temperatures and sintering and materials properties changes at high temperatures.

6. Tritium Permeation and Inventory in the Structure

Tritium permeation is primarily a safety concern, but the attempt to control it can have a large impact on design and operation. The most serious problem is felt to exist for in-vessel components where tritium passes from the plasma chamber into the coolant streams. The magnitude of permeation depends on plasma edge conditions, on trapping in the structure (which may depend strongly on irradiation), and on the effectiveness of control methods, such as permeation barriers. In the bulk of the blanket, permeation can be significantly altered by the form in which the tritium is released from the solid breeder and the chemistry and kinetics as it travels through the blanket. The form of tritium influences both the release rate and the biological hazard potential.

7. In-Vessel Component Thermomechanical Response and Lifetime

In-vessel components have special problems with thermomechanical performance in addition to those in the blanket. These special problems stem from the very high heat and particle fluxes to which these components are exposed under normal and off-normal conditions. One of the largest uncertainties is erosion and redeposition mechanisms and consequences, which have far-reaching implications on lifetime, failure modes, and design choices. The structural integrity of in-vessel components is also uncertain due to the high thermal stresses and presence of local hot spots. Bonds may be necessary if the surfaces are protected by coatings or composite structures. The structural response of these bonds is a particular concern.

8. Radiation Shielding: Accuracy of Prediction and Quantification of Radiation Protection Requirements

The primary function of radiation shielding is to protect both personnel and sensitive reactor components. The latter is generally more restrictive, including superconducting magnets, some elements of plasma exhaust and heating systems, instrumentation and control. Uncertainties exist in the accuracy of predicting the radiation field and in quantifying the radiation protection requirements for these sensitive components. Although sophisticated neutronics techniques exist, uncertainties remain due to modeling complexities, nuclear data uncertainties, limitations of calculational methods for deep penetration problems, and time-dependent behavior of materials and structures.

9. Accuracy and Survivability of Instrumentation and Control

Failure of instrumentation and control may have a very serious impact on the safety and operation of the reactor. The vulnerability of these components depends to a large extent on radiation shielding as described above. However, because of the added effects of all the environmental conditions present in a fusion reactor (e.g., magnetic field), I&C is considered separately. Instrumentation and control components often contain materials which are sensitive to radiation, electromagnetic effects, and corrosion. It is necessary in a number of key cases to develop new measurement techniques because presently available instruments will not function properly in high fields, with bulk heating, or in corrosive environments. In addition, innovative techniques for measurements related to new phenomena in the fusion environment are needed in order to obtain meaningful information from experiments.

2.3 Survey of Experimental Needs

2.3.1 Introduction

The development of fusion to the commercial reactor stage will require resolving the many known issues, as well as presently unknown ones. The first step is to identify these concerns, the second is to identify the tests that are needed to resolve the concerns, and the third is to implement a test program to perform these tests. This section summarizes the results of the second step, where the fusion nuclear technology testing needs up to the engineering demonstration stage are identified.

For this survey, "test" is used in the generic sense to mean a process of obtaining information through physical experiment and measurement, i.e., not through design analysis or computer simulation. A "testing need" refers to a need for a certain class or type of information that must be obtained through experimental measurements. For example, there is a testing need for irradiated structural material properties, which will require a range of tests such as tensile strength and cyclic fatigue tests applied to thousands of test articles.

The survey relied on experts from many technical disciplines in order to identify the tests that should be performed. All testing needs are addressed, including developing a property data base, understanding underlying phenomena and verifying component performance. It is based largely on a limited number of representative blankets and other components which are expected to indicate most of the needed tests. These tests must address the issues with a minimum of overlap, and with test goals that can be met with measurable and interpretable results under the relevant environmental conditions. The identified tests are presented in a format that is intended to make assumptions and judgements explicit.

The testing needs identified in this survey are organized into two levels: component and type of information. The first level distinguishes between the components, which generally have different functions, different operating conditions, and thus different testing needs. Specific components considered include the blanket, plasma interactive components (e.g., first wall, limiter, divertor, rf antennae), shield, tritium processing system, magnets, instrumentation and control, and balance of plant, as well as interactions among these components. The second level of organization distinguishes between types of tests, such as property measurements versus component verification. This level also provides a rough measure of test complexity and a loose chronological ordering since generally the simpler tests will be performed first.

2.3.2 Test Categories

The test categories adopted here are: Basic, Single Effect, Multiple Effect/Multiple Interaction, Partially Integrated, and Integrated Tests. Table 2.3-1 summarizes the descriptions of these categories. Component tests is another category, but it has not been considered in detail in FINESSE because it represents requirements for a more advanced stage of development than that for which the earlier test categories are needed.

Basic Tests measure basic or intrinsic property data such as thermal conductivity of a solid breeder material. Single Effect Tests are experiments with a single environmental condition aimed at developing an understanding and models of a single phenomena or issue.

At some point, however, additional phenomena and interactions must be added to demonstrate and explore any synergistic effects. These Multiple Effect/Multiple Interaction Tests involve both interactions among the effects of multiple environmental conditions as well as direct interactions among different physical elements.

Partially Integrated Tests attempt to obtain Integrated Test information but without some key environmental condition. This category emphasizes a particular range of tests in the continuum between Multiple Effect/Multiple Interaction and Integrated Tests. It is particularly relevant for fusion where costs may limit complete simulation of all important variables.

Integrated Tests demonstrate that a concept is feasible; they are the "proof-of-principle" experiments. With all key environmental conditions and physical elements present, they specifically indicate any major unanticipated interactions. However, they are often performed under scaled size or environmental conditions. Depending on the degree of scaling, a given test may emphasize one aspect of component performance over another, such as a test that simulates thermomechanical behavior but cannot also simulate full tritium

Table 2.3-1 Test Categories for Single Component Research and Development

---

Basic Test

- **Basic or intrinsic property data**
- Single material specimen
- Examples: thermal conductivity; neutron absorption cross section

Single Effect Test

- **Explore a single effect, a single phenomenon or the interaction of a limited number of phenomena, in order to develop understanding and models**
- Generally a single environmental condition and a "clean" geometry
- Examples: 1) pellet-in-can test of the thermal stress/creep interaction between solid breeder and clad; 2) electromagnetic response of bonded materials to a transient magnetic field; 3) tritium production rate in a slab of heterogeneous materials exposed to a point neutron source

Multiple Effect/Multiple Interaction Test

- **Explore multiple environmental conditions and multiple interactions among physical elements in order to develop understanding and prediction capabilities**
- Includes identifying unknown interactions, and directly measuring specific global parameters that cannot be calculated
- Two or more environmental conditions; more realistic geometry
- Example: testing of an internally cooled first wall section under a steady surface heat load and a time-dependent magnetic field

Partially Integrated Test

- **Partial "integrated test" information, but without some important environmental condition** to permit large cost savings
- All key physical elements of the component; not necessarily full scale
- Example: liquid metal blanket test facility without neutrons

Integrated Test

- **Concept verification** and identification of unknowns
- All key environmental conditions and physical elements, although often not full scale
- Example: blanket module test in a fusion test device

Component Test

- **Design verification and reliability data**
- Full-size component under prototypical operating conditions
- Examples: 1) an isolated blanket module with its own cooling system in a fusion test reactor; 2) a complete integrated blanket in a demonstration power reactor

---

breeding behavior because of the changes in the module needed to accommodate the available test conditions.

In contrast to the above categories of tests where the focus is on basic data, understanding and concept feasibility, Component Tests are performed at a more advanced stage of development to verify that full-sized components operate as expected under complete prototypical conditions. This brings out all interactions and any remaining unknowns, and yields definitive reliability and performance data.

2.3.3 Testing Needs Summary

Each identified testing need is characterized by:

- importance of neutrons;
- importance of fusion neutron energy spectrum;
- other required environmental conditions;
- typical test article size;
- number of test articles (excluding duplicate tests for statistical purposes, off-normal conditions, data at several time intervals for high fluence tests, etc.);
- usefulness and limitations of non-nuclear test stands, point neutron sources and fission reactors as test facilities.

From the test descriptions in Chapter 4, a total of 74 testing needs were identified, with 45% blanket related, 20% plasma interactive components, and 35% for the remainder of the components and component interactions. Three specific tokamak testing needs were identified related to plasma interactive components, while no specific mirror testing needs were defined. Also, there were seven solid breeder, two multiplier and three liquid breeder specific testing needs.

A particularly interesting class of tests are those which require high energy or fusion neutrons, since this is directly related to the question of the ability of non-fusion test facilities to fully develop reliable fusion nuclear components. In these tests, neutrons serve as a source of bulk heating, radiation damage and/or specific reactions. These tests are summarized in Table 2.3-2, along with estimates of test article sizes and the required number of test articles. Such tests include irradiated properties measurements, as well as integrated tests where a fusion environment is believed necessary to confidently address the key issues related to establishing concept feasibility and attractiveness.

Table 2.3-2 Fusion Nuclear Technology Tests Requiring Fusion Neutrons

Tests	Typical Test Article Size (cm x cm x cm)	Number of Test Articles[a]
<u>BASIC TESTS</u>		
Structural material irradiated properties	1 x 1 x 2	20,000
Solid breeder irradiated properties	1 x 1 x 2	1,200
Plasma interactive materials irradiated properties	1 x 1 x 5	900
Radiation damage indicator cross-sections	1 x 1 x 0.5	500
Long-lived isotope activation cross-sections	1 x 1 x 0.1	200
Neutron sputtering rate cross-sections	1 x 1 x 0.1	30
<u>SINGLE EFFECT TESTS</u>		
Structure thermomechanical response experiments	10 x 10 x 10	50
Weld behavior experiments	10 x 10 x 5	50
Shield effectiveness in complex geometries	50 x 50 x 100	50
Optical component radiation effects	2 x 2 x 2	20
<u>MULTIPLE EFFECT/MULTIPLE INTERACTION TESTS</u>		
Submodule thermal and corrosion verification	LB[b]:100 x 100 x 30 SB[b]:10 x 50 x 30	5 5
<u>PARTIALLY INTEGRATED AND INTEGRATED TESTS</u>		
Verification of neutronic predictions - Tritium breeding, nuclear heating during operation, and induced activation	50 x 50 x 100	4
Full module verification - Thermal and corrosion - Module thermomechanical lifetime - Tritium recovery	LB[c]:100 x 100 x 50 SB:100 x 100 x 50	5 5
Instrumentation transducer lifetime	1 x 1 x 2	70
Insulator/substrate seal integrity	1 x 1 x 2	20
Biological dose rate profile verification	DT device	1
Afterheat profile verification	DT device	1
<u>COMPONENT TESTS</u>		
Blanket performance and lifetime verification	SB:30 x 100 x 80 LB:900 x 300 x 80	3 3
Radiation effects on electronic components	1 x 1 x 1	20
Instrumentation performance and lifetime	5 x 5 x 5	100

[a]Test article defined as one physical entity tested at one set of conditions. Duplication of tests for statistical purposes, off-normal conditions, data at several time intervals, for high fluence tests, etc., are <u>not</u> included in the number of test articles.

[b]LB = liquid breeder blankets
SB = solid breeder blankets

[c]Some designs require larger test volume.

2.4 Requirements of the Experiments

2.4.1 Introduction

The previous two sections identified the issues and surveyed the testing needs to resolve these issues. Translating these testing needs into a realistic R&D plan that describes experiments, facilities, schedule and cost requires tradeoffs between two factors: (1) benefits of tests as functions of the environmental conditions provided in the experimental facility, and (2) capabilities, limitations, costs and risks of various options for experimental facilities. This section examines the first factor while the next two sections are focused on facilities. Section 2.4.2 is concerned with quantifying the test requirements and developing engineering scaling relationships for multiple interaction and integrated tests. The quantitative analysis in this area has been limited to date to the blanket subsystem. Section 2.4.2 is focused on operating parameters such as power density, magnetic field intensity, and power pulse length. The special problems related to failure modes and the important considerations of highly time-dependent phenomena that affect fluence goals are the focus of Sections 2.4.3 and 2.4.4, respectively. Experiments aimed specifically at neutronics information (e.g., tritium breeding, nuclear heating) have their own distinct requirements and are discussed in Section 2.4.5. Preliminary evaluation of test matrix considerations, e.g., total surface area and volume for fusion nuclear technology R&D in the neutron environment is the subject of Section 2.4.6.

2.4.2 Blanket Experiment Requirements and Engineering Scaling

2.4.2.1 Introduction

For the class of interactive, partially integrated and integrated tests, it is nearly certain that the parameters of the test device will not all match those of a full-scale fusion reactor because of cost constraints. This will result in changes to the operating conditions in the test module. If nothing is done to correct this situation, the value of the test to resolve the key nuclear testing issues may be severely restricted.

It is possible in many cases for which the phenomena are sufficiently well understood to modify the design or operating parameters of the test module in order to recover the important aspects of the testing issues. This process of developing meaningful tests at reduced device parameters is known as engineering scaling. In some cases, even if the phenomena are well-understood, a reduction of the device parameters beyond certain limits will result in the inability to maintain act-alike blanket behavior. One of the goals of Engineering Scaling is to identify and determine these limits, or test requirements. This procedure is often difficult because blanket behavior usually varies slowly as a function of the device parameters, without an abrupt change of operational performance. Many trade-offs between cost and benefit must be considered with respect to the large number of testing issues and their priorities.

In order to understand and quantify the scaling laws and test requirements, analyses of many technical aspects of blanket operation were performed, including fluid flow, MHD, thermal-hydraulics, tritium recovery, structural mechanics, neutronics, corrosion and materials compatibility. These analyses were based on issues that are believed important, but were limited in scope. Consequently, the conclusions regarding the requirements for useful scaled tests are not complete.

For these analyses, four specific reference blankets were chosen (see also Appendix D); the MARS self-cooled LiPb/HT-9 mirror blanket and three designs representative of blankets considered within BCSS: a self-cooled lithium toroidal/poloidal flow design, a helium-cooled Li_2O design, and a water-cooled $LiAlO_2$ design. These blankets were considered not because they necessarily represent the best possible designs; rather, they serve as tools to identify the problems of scaling plausible blankets. They cover a range of design features of general interest, such as liquid versus solid breeder, and, consequently, their consideration should lead to conclusions on engineering scaling and test requirements that are applicable to a large class of candidate blankets.

2.4.2.2 Liquid Metal Blanket Test Requirements

For the liquid metal blankets, the most critical integrated testing issues are: 1) thermomechanical performance and failure modes, including MHD effects, and 2) materials compatibility. Most of the analysis and the effort to determine test requirements were based on these issues.

Thermomechanical Performance

The uncertainties in thermomechanical performance relate to both the complex loading conditions and the thermal and structural responses to the loading. Integrated testing to verify the thermomechanical performance of the blanket will require testing in the correct geometry and under actual loading conditions.

The sources of structural loading include steady and/or transient thermal stresses, MHD pressure stresses, radiation swelling, and magnetic forces. Probably the most difficult and most important loading condition to simulate is the thermal stresses, which stem from the temperature profiles in the blanket. These profiles depend strongly on MHD velocity profiles which are very design-dependent and poorly understood. The desire to preserve temperature profiles and thermal stresses leads to requirements on the size of the test module, the surface heat flux, the bulk heating, and the MHD velocity profiles.

As the length of the cooling channels is reduced from the reference values, the thermal hydraulic behavior is affected. This is due to the fact that liquid metal flow in magnetic fields can have extremely long entry lengths for the development of velocity, temperature, and dissolved corrosion product profiles. The entire blanket can be in a state of development, in which heat, mass, and momentum transfer coefficients are rapidly varying. Figure 2.4.2-1 illustrates this behavior. It shows, for a typical heated channel perpendicular to the magnetic field, that the heat transfer coefficient drops, corrosion rate increases after an initial saturation, pressures decrease due to MHD effects, and temperatures increase (which in some designs can result in increasing thermal stresses).

As a result of this complex array of loading conditions, shortened channels may not be able to simulate the entire thermal/mechanical state of the blanket. In some blankets, for example the BCSS reference blanket, this may not turn out to be an insurmountable problem. Thermomechanical verification can be performed on a unit cell basis, for which only a small region of the blanket is treated at a time. In addition, the application of scaling, such as reducing the flow velocity or the channel dimensions, may help to recover the entire development region for several of the important loading

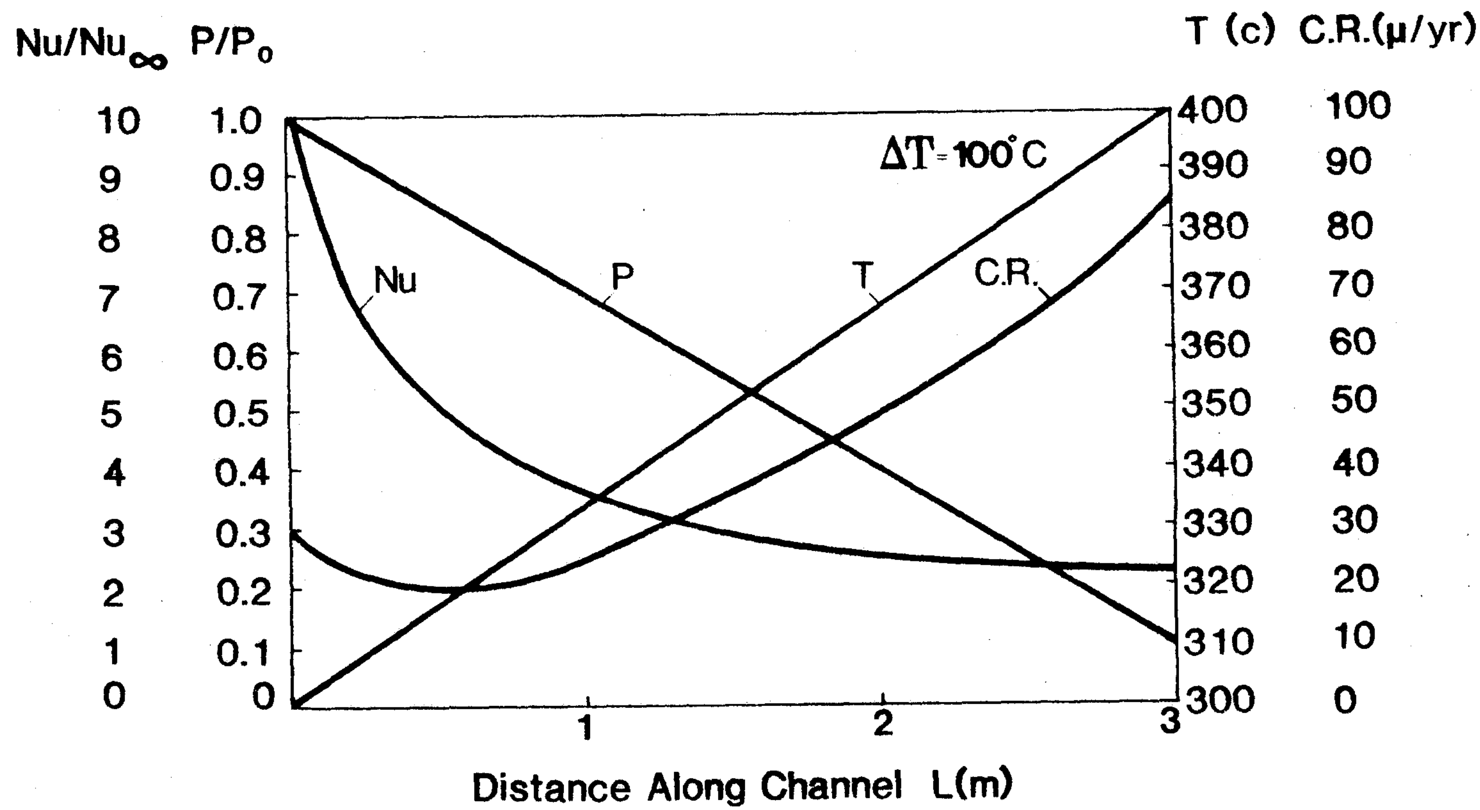


Figure 2.4.2-1 Illustration of the variation of heat transfer (Nu), corrosion (C.R.) and pressure stresses along a heated channel perpendicular to the magnetic field.

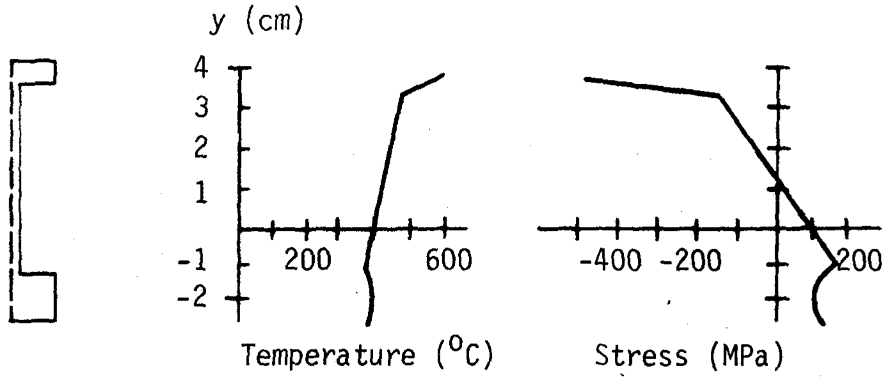
conditions. The fluid residence time is one of the most important scaling parameters for act-alike temperatures and corrosion.

The surface heat flux is the largest contributor to the first wall stress in tokamak blankets. In the BCSS composite first wall structure, the thermal stress depends primarily on the temperature gradient across the first wall and the difference in temperature between the first and second walls. The exact shape of the temperature profiles is less important. Figure 2.4.2-2 demonstrates this effect using an I-beam section from the first wall. Four cases are presented. In Cases 1 and 2, the validity of aspect ratio scaling is demonstrated. In Case 3, the bulk heating is turned off, but the second wall temperatures are maintained (by controlling the coolant temperature) and the stresses are maintained nearly identical. In Case 4, the second wall temperature is allowed to rise, showing a substantial change in stresses. If control of the coolant temperatures is possible, then the radial temperature profile (where "radial" is the direction away from the plasma) can be simulated well without bulk heating.

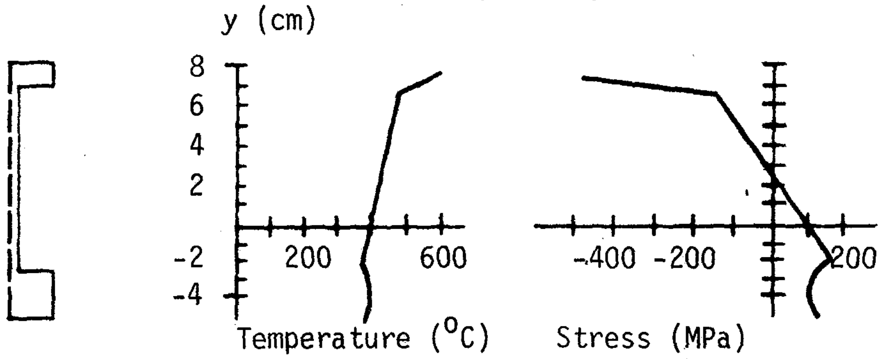
The primary need for neutrons in structural testing of this design is irradiation effects, including swelling, creep, and properties changes. Analysis of the MARS design has demonstrated how much irradiation swelling can alter the structural response. In Figure 2.4.2-3, irradiation creep and a small amount of swelling (0.01%/dpa) were included with the thermal and pressure stresses. The total stress is then plotted at two locations in the blanket as a function of the neutron dose. The end of life stresses are much larger than at the beginning of life. This also illustrates how important a good materials properties data base is, and how difficult scaling can be without it.

These conclusions regarding the importance of bulk heating in the BCSS design assume that the large uncertainties in thermal-hydraulic performance can be resolved separately. Because of the importance of bulk heating in determining the thermal-hydraulic behavior of the blanket, bulk heating may be necessary for thermal-hydraulics experiments. For other designs, such as the MARS reference design, bulk heating is more important in structural testing for two reasons: first, surface heating is a much less dominant contributor to stresses in mirrors (~ 0.1 MW/m^2 vs. ~ 0.5 MW/m^2), and second, the level of coolant temperature control available in the BCSS design is not present in the MARS blanket.

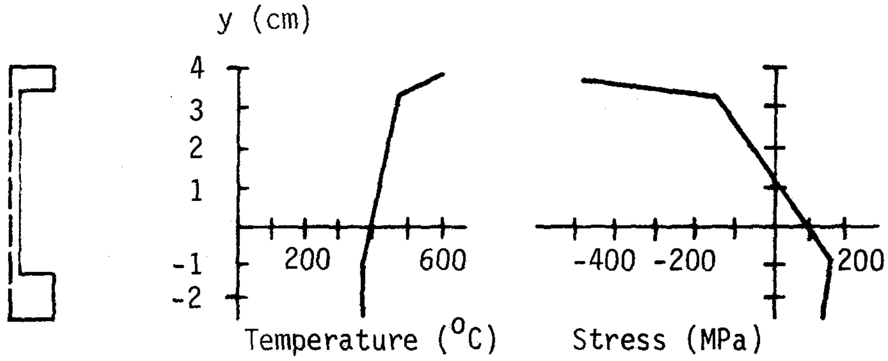
Case 1: BCSS Reference Temperatures and Dimensions



Case 2: All Dimensions Multiplied by 2.3



Case 3: Average Temperatures Preserved with No Bulk Heating



Case 4: Second Wall Temperature Increased (with No Bulk Heating)

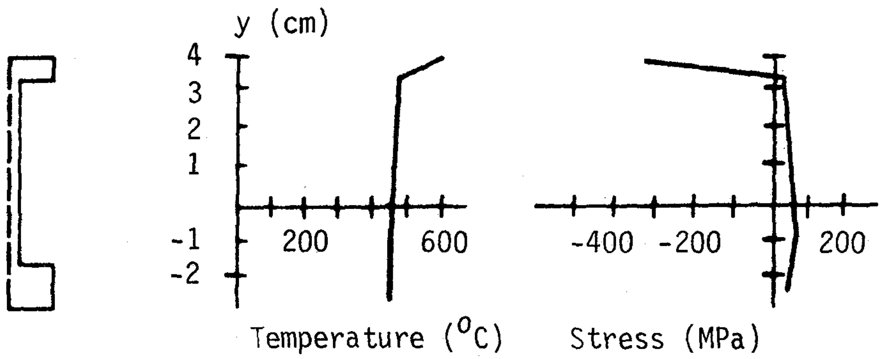


Figure 2.4.2-2 Elastic stress matching using the I-beam model.

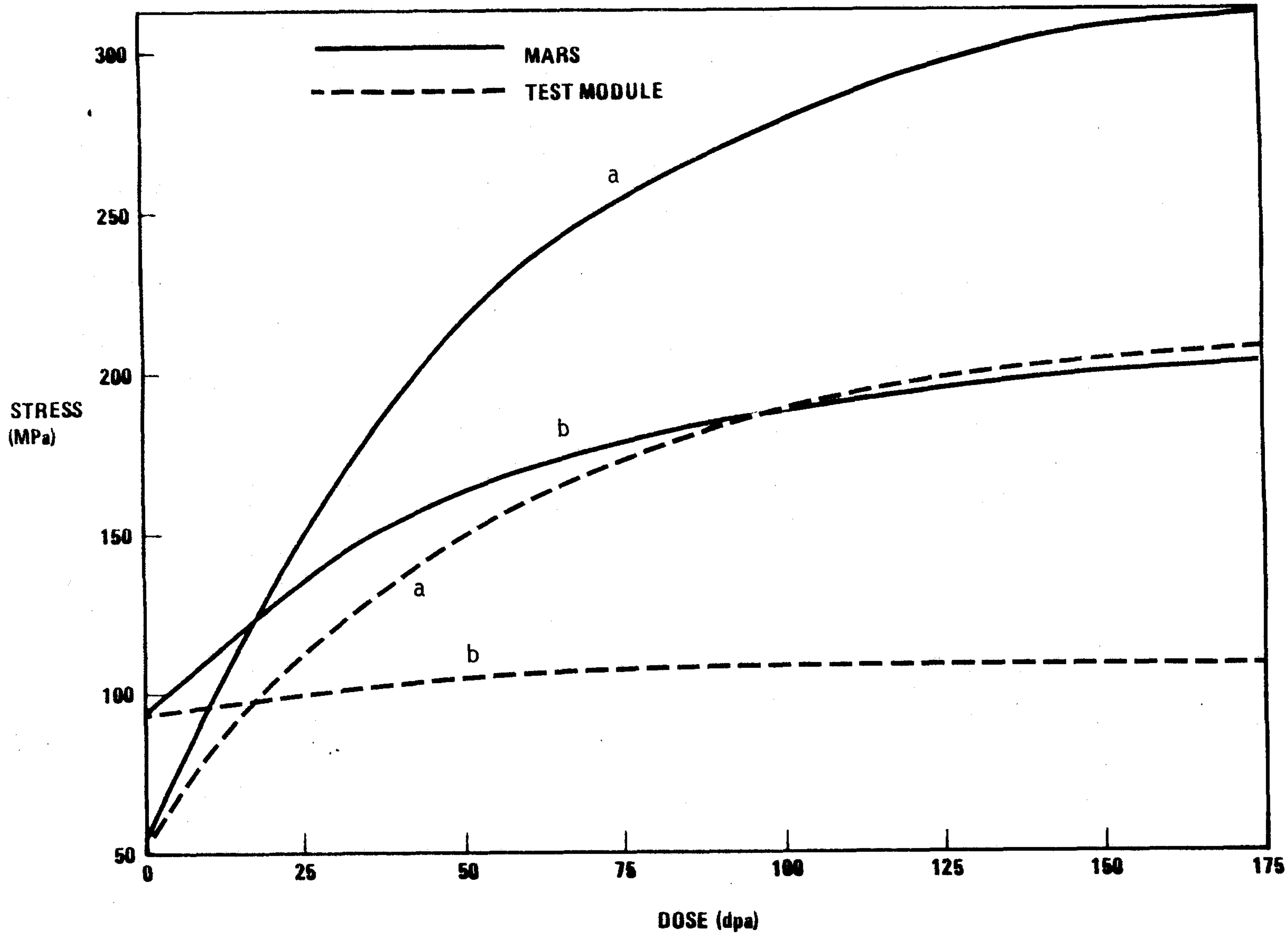


Figure 2.4.2-3 MARS blanket and test module stresses with swelling and creep included: (a) location of maximum end-of-life stress and (b) location of maximum beginning-of-life stress.

The impact of MHD on the thermal hydraulics and thermomechanics of the blanket is pervasive: the velocity profiles have been shown to control the structure temperatures, and the uncertainties in the actual velocity profiles in the reference tokamak blanket are large enough to close its design window. Figure 2.4.2-4 demonstrates the large uncertainties in the first wall temperatures using three sample velocity profiles (slug flow, parabolic flow, and Couette flow). The actual velocity profiles are unknown and depend on the specific design features. The figure also demonstrates the varying degree to which bulk heating affects the structure temperatures, depending on the velocity profile.

Because of their impact on blanket temperatures, it is important to preserve the velocity profiles in a thermomechanics experiment. This is a difficult task because, like the temperatures, the velocity can have large entry lengths, in particular for flow parallel to the magnetic field. In the reference design, global eddy currents affect the velocity distributions, so modelling of the entire blanket is important. This requirement is in contradiction with the unit cell appraoach. However, it is believed that large global eddy currents must be removed through redesign in order to make an attractive tokamak blanket. In this case, the unit cell approach will still be valid for composite first wall blankets such as the BCSS design.

The structural response is governed primarily by geometry. By keeping the structural aspect ratios fixed between the reference blanket and the test module, the response of the two structures will be similar. Aspect ratio scaling is valuable because at reduced surface heat flux, the first wall thickness must be increased to retain the thermal stresses. By increasing all of the blanket dimensions uniformly, most features of the structural response can be maintained. Two cases in which aspect ratio scaling may fail are irradiation effects and failure modes: (1) The radiation damage profiles are difficult to scale because the neutron mean free path is relatively independent of geometry. This problem is illustrated in Figure 2.4.2-3, which compares the MARS response with the response of a smaller test module which has a different damage profile. However, if a method of altering the damage profiles is found, it has been demonstrated that aspect ratio scaling works with inelastic as well as elastic strains. (2) Failure modes, such as crack growth or failure at stress concentrations, may depend on the absolute dimensions of the structure in addition to aspect ratios.

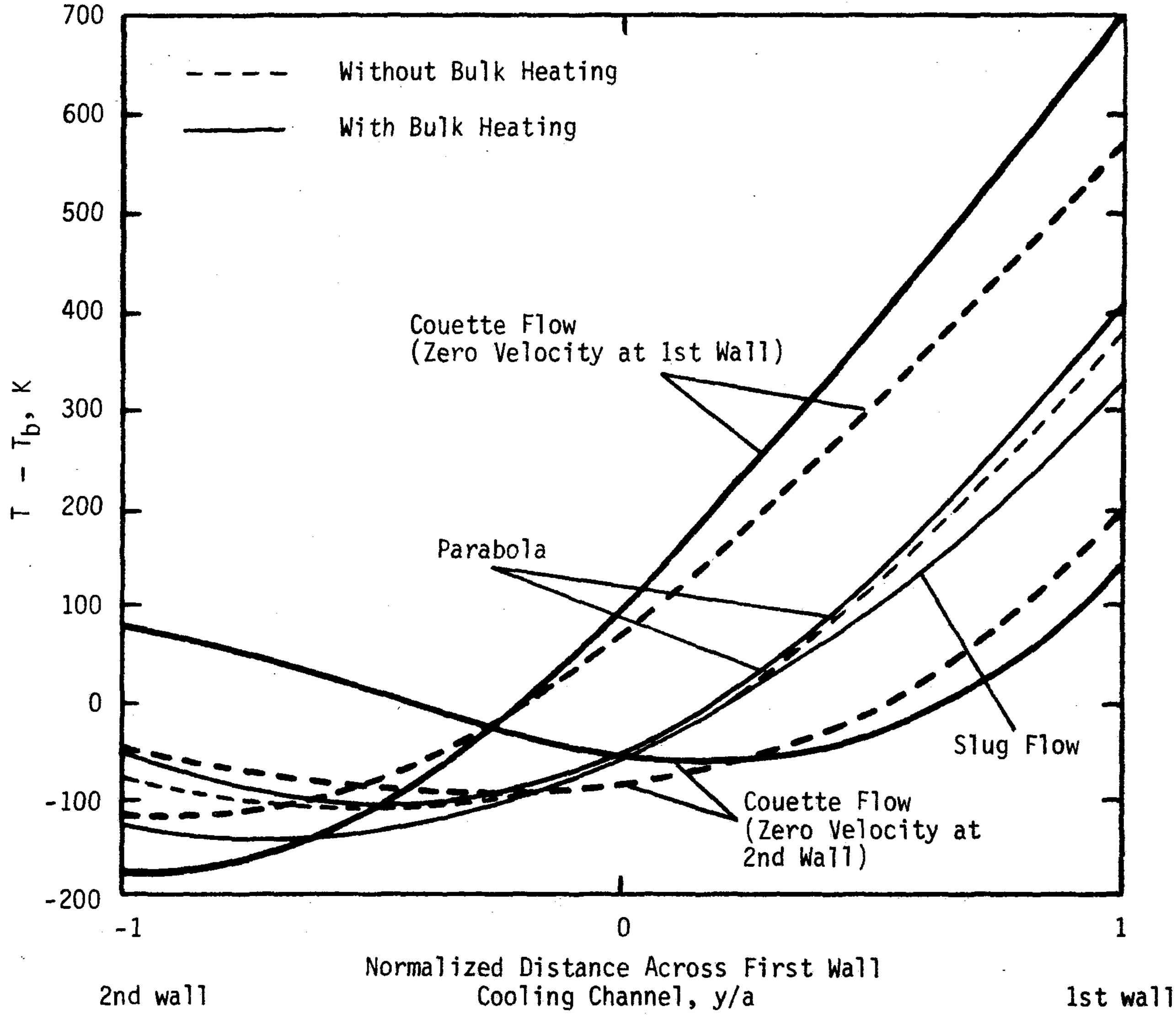


Figure 2.4.2-4 Dependence of coolant temperature profile, first and second wall temperatures on the coolant velocity profile, and bulk heating (temperatures are referenced to the bulk coolant temperature, T_b).

Irradiation also has a strong impact on the structural responses. Results of irradiation creep studies indicate: (1) as shown in Figure 2.4.2-5, thermal stresses relax due to irradiation creep over a period of a few months (5-10 dpa); (2) the rate of deformation in the coolant channels due to irradiation creep driven by primary (pressure) stresses appears to be constant; and (3) preserving aspect ratios may be a feasible method to retain act-alike creep behavior if the damage gradient effect can be overcome.

Materials Compatibility (Corrosion)

Corrosion mass transfer has been studied, with most of the emphasis on the dissolution and convection mechanisms, which are most relevant for stainless and ferritic steel systems. The results show that magnetic field strength, blanket temperatures, and fluid residence time are important parameters to preserve.

The magnetic field has at least two effects on corrosion: laminarization of the flow results in a decrease in corrosion compared to turbulent flow, whereas thinning of the boundary layer (in Hartmann flow) has an enhancing effect. (Streaming profiles, which can occur in certain geometries, may have an even larger impact, but this effect has not yet been examined.) Ignoring for the moment which effect is larger, it is clear that testing without the magnetic field can provide only limited usefulness. In Figure 2.4.2-6, the corrosion rate is plotted as a function of the Hartmann number (which is proportional to the magnetic field) using two different assumptions for the mass diffusion coefficient. It is shown that the regime present in the MARS design (and in fact for most designs) is controlled by the diffusion coefficient rather than the thickness of the boundary layer. This is because the Hartmann velocity boundary layer is so thin as to be transparent to diffusing species. If the magnetic field is dropped by an order of magnitude or the diffusion coefficient is far higher than expected, then the dominant mechanism would change, resulting in a serious loss of information.

The analysis for diffusive transport depends heavily on the solubility of the corrosion products, which is a very temperature-dependent property. Changes in the average temperature as well as the temperature rise along the coolant channels will affect corrosion. In order to observe the initial high corrosion rate in the channel entrance, the saturation mechanism, and the

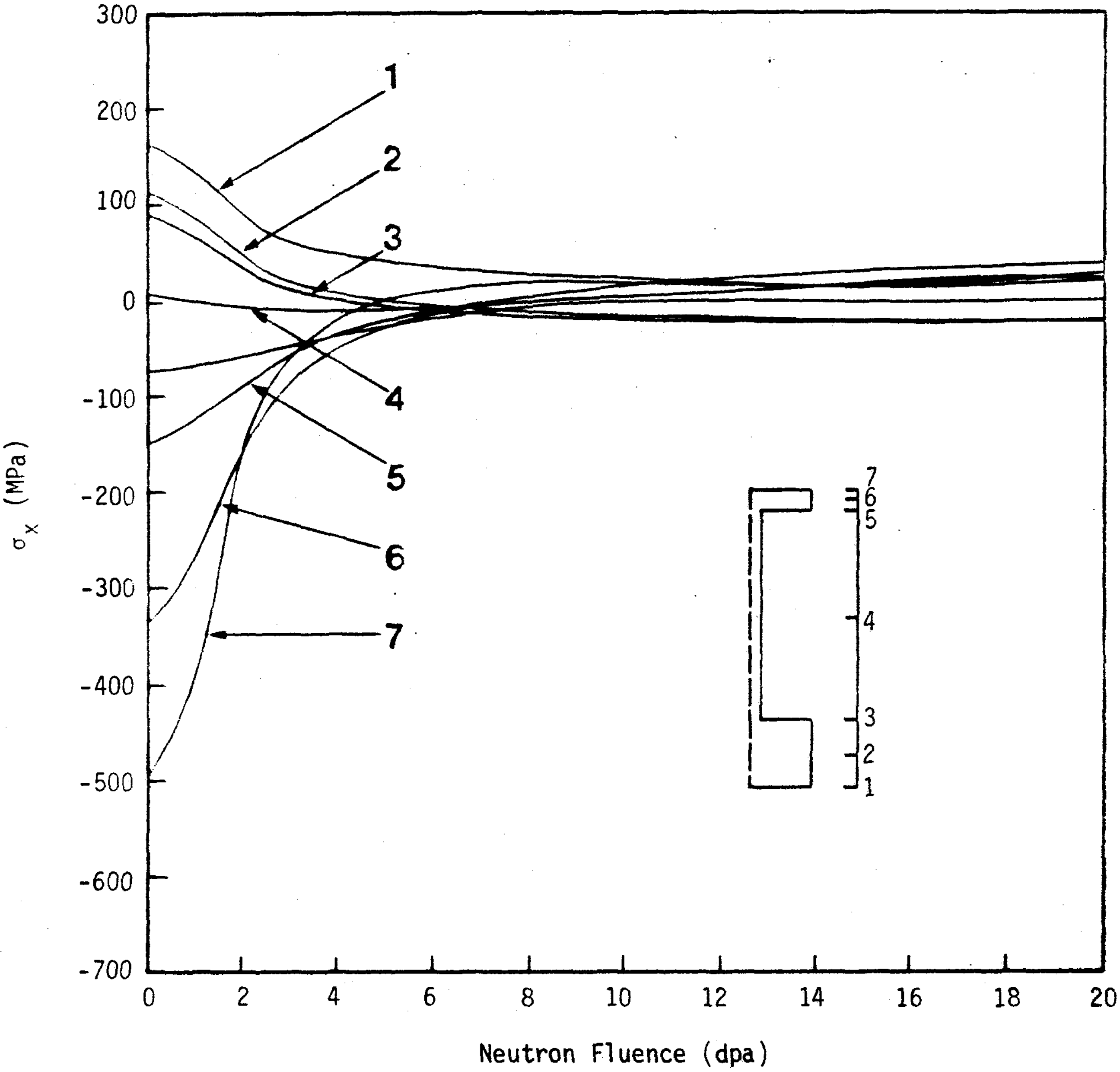


Figure 2.4.2-5 Stress history at several locations in BCSS lithium self-cooled tokamak first wall due to irradiation creep.

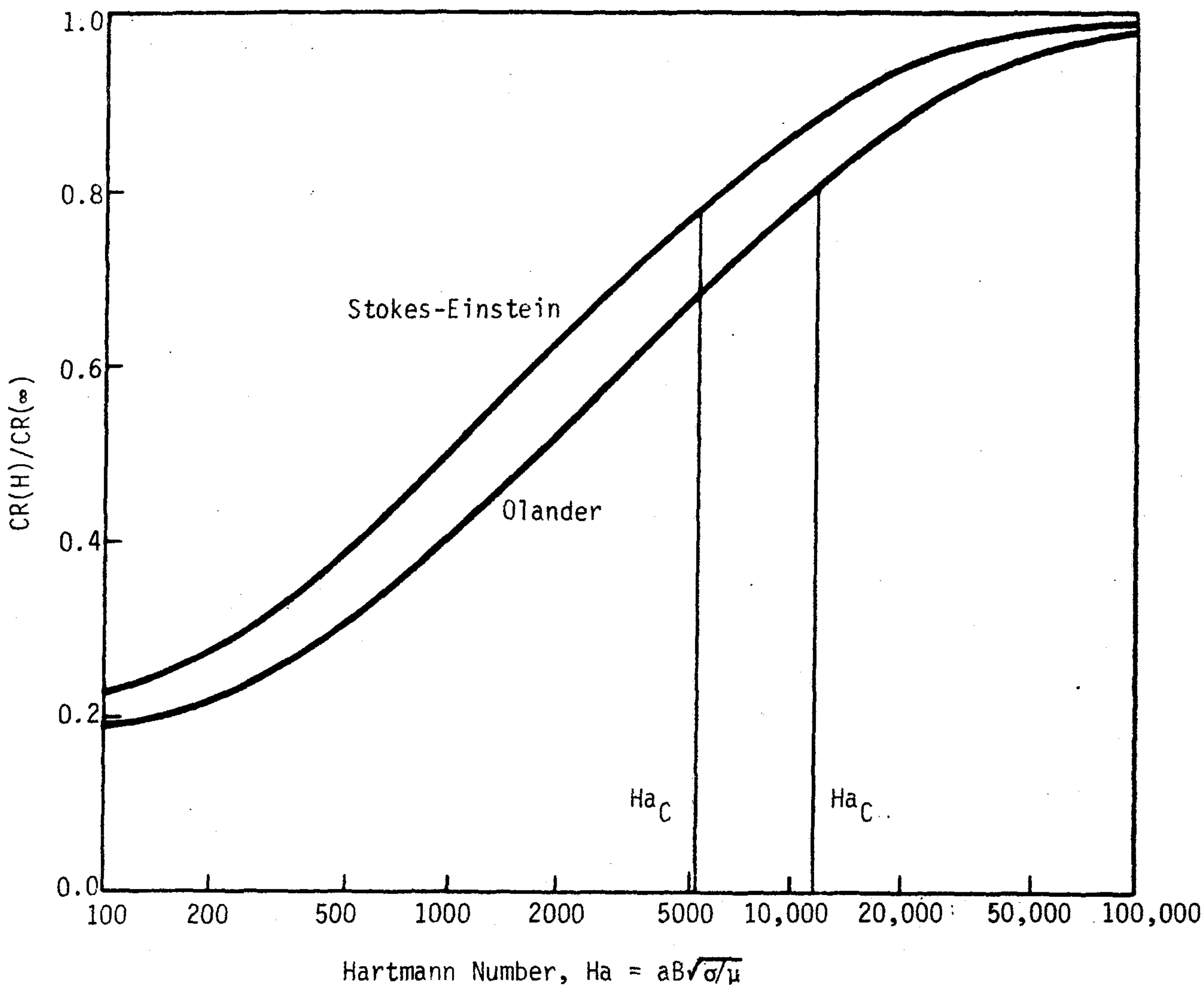


Figure 2.4.2-6 Dependence of corrosion rate on magnetic field due to boundary layer thinning, using two different assumptions for the mass diffusion coefficient.

final dependence on the temperature gradient, the channel length should not be reduced beyond the point where the saturation mechanism has not taken effect. In some cases, these competing factors may exist over half of the channel length.

For preserving the temperature rise and the general corrosion behavior in shortened channels, a useful parameter to maintain is the coolant residence time. It has been shown that by reducing the coolant velocity in proportion to the decrease in channel length, that the same corrosion rate profiles can be obtained.

Although diffusion and convection were emphasized in the work done to date, it is recognized that impurity reactions in the structure and primary cooling system loop interactions may be important contributors to the test requirements. For modelling the effects of the cooling system, including the heat exchanger, it has been suggested that the ratios of surface areas of the hot and cold components in the loop be maintained.

2.4.2.3 Solid Breeder Blanket Test Requirements

The primary integrated testing issues for solid breeder blankets are thermomechanical performance, failure modes and tritium recovery. The analyses summarized here concentrated primarily on the structural, thermal and tritium behavior of the blanket.

The structure may be divided into the first wall and the breeder region. The thermomechanical aspects of the first wall are always a concern, particularly in tokamaks where the high surface heating and wall erosion require special design features such as grooved walls. Major uncertainties in the breeder region relate to the interaction between the solid breeder and its surrounding structure, possibly leading to rupture or deformation. This could, for example, reduce tritium recovery through porosity changes, pressurize and contaminate the purge system, or leak tritium or corrosive lithium compounds into the primary coolant. In both regions, thermal stresses are a major contributor to the overall stress state. Later in life, creep and swelling, as well as material properties changes due to radiation become significant.

Tritium inventory, recovery and permeation from solid breeders are uncertain in the basic trapping and transport processes under fusion environmental conditions, and in the effects of radiation and the breeder/structure interaction. These processes are very dependent on temperature, purge chemistry and breeder microstructure.

The first test requirement for both these issues is to preserve the first wall and breeder region temperature profiles. If the module heat source is reduced (whether based on fusion or auxiliary heating), the most plausible approach is to adjust dimensions and coolant flow conditions to compensate. Since the temperature profile along the perimeter of high-pressure-retaining lobed first walls is important, it is not possible to arbitrarily change the heat source. In particular, both surface and volumetric heating must be changed such that their relative importance to the first wall temperature profile is maintained. This leads to the constraints indicated in Fig. 2.4.2-7, where tokamak modules (dominated by surface heating) have a minimum and mirror modules (dominated by bulk heating) have a maximum surface heating requirement.

If temperatures are preserved, then elastic stress behavior can be preserved if coolant pressure stresses and structural dimension ratios are additionally maintained. Preserving temperatures requires increasing the first wall thickness and channel dimensions as the heat source (surface plus volumetric) is reduced, and so the first wall width (the lobed first wall radius of curvature) must also increase as indicated in Fig. 2.4.2-8. If the width is increased proportional to the first wall thickness, then the elastic stresses are preserved ($M = 1$). However, if there is limited test volume available, then it may not be possible to make the first wall as wide as desired, leading to appreciable changes in the stress profile ($M > 1$). Thus, a limit on test volume implies limits on the heat source changes in order to maintain act-alike stresses.

A further consequence of increasing first wall dimensions is that neutron attenuation eventually becomes significant and leads to an appreciable variation in flux, and consequently creep and swelling rate, across the first wall. For example, an increase in thickness by a factor of four (roughly a factor of five reduction in heat source for a tokamak first wall) leads to a factor of two variation in flux or fluence across a 1 cm first wall. This can

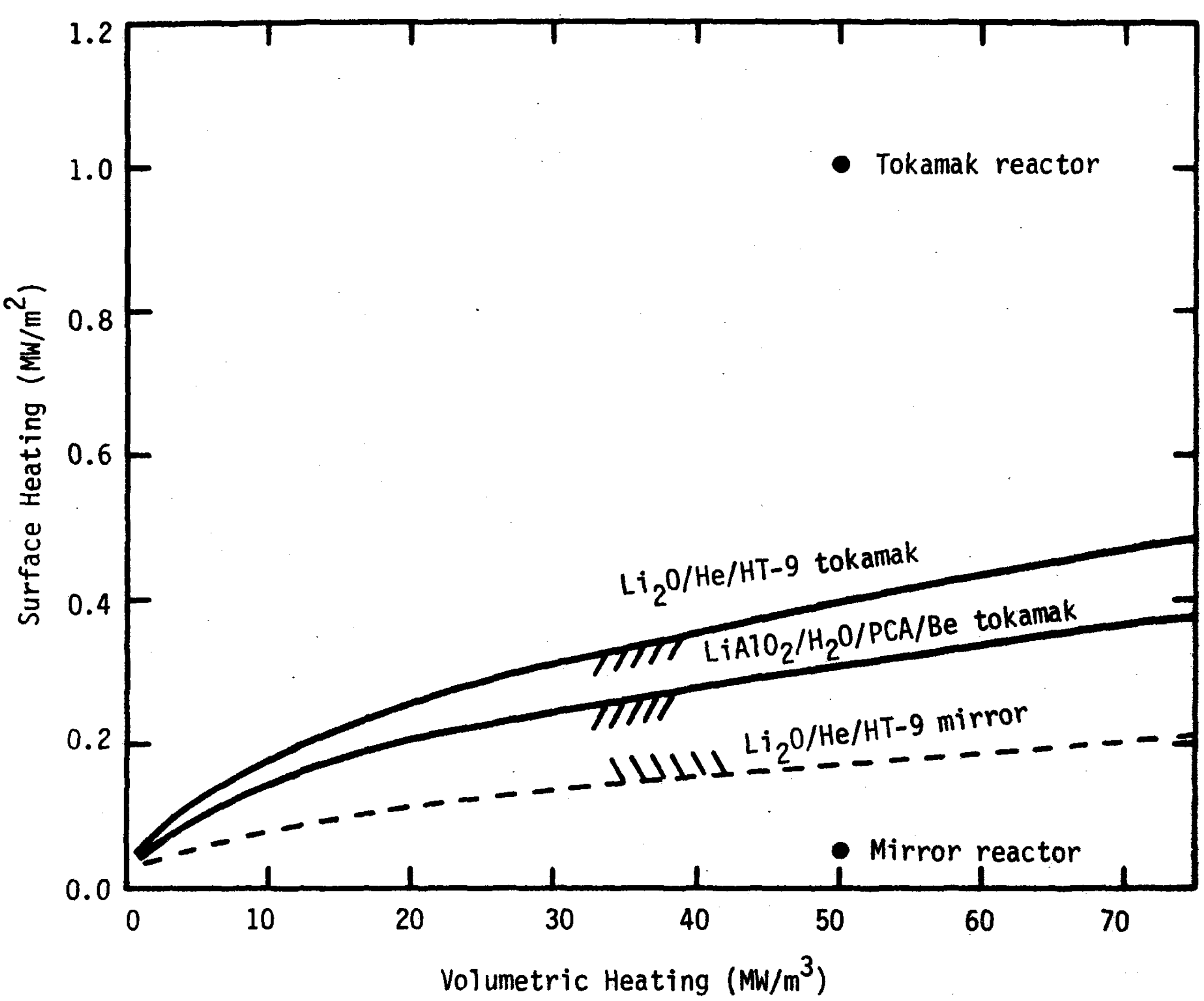


Figure 2.4.2-7 Heat source requirements for preserving first wall temperature profile. The boundaries indicate transition between surface heating and volumetric heating dominated first wall temperatures.

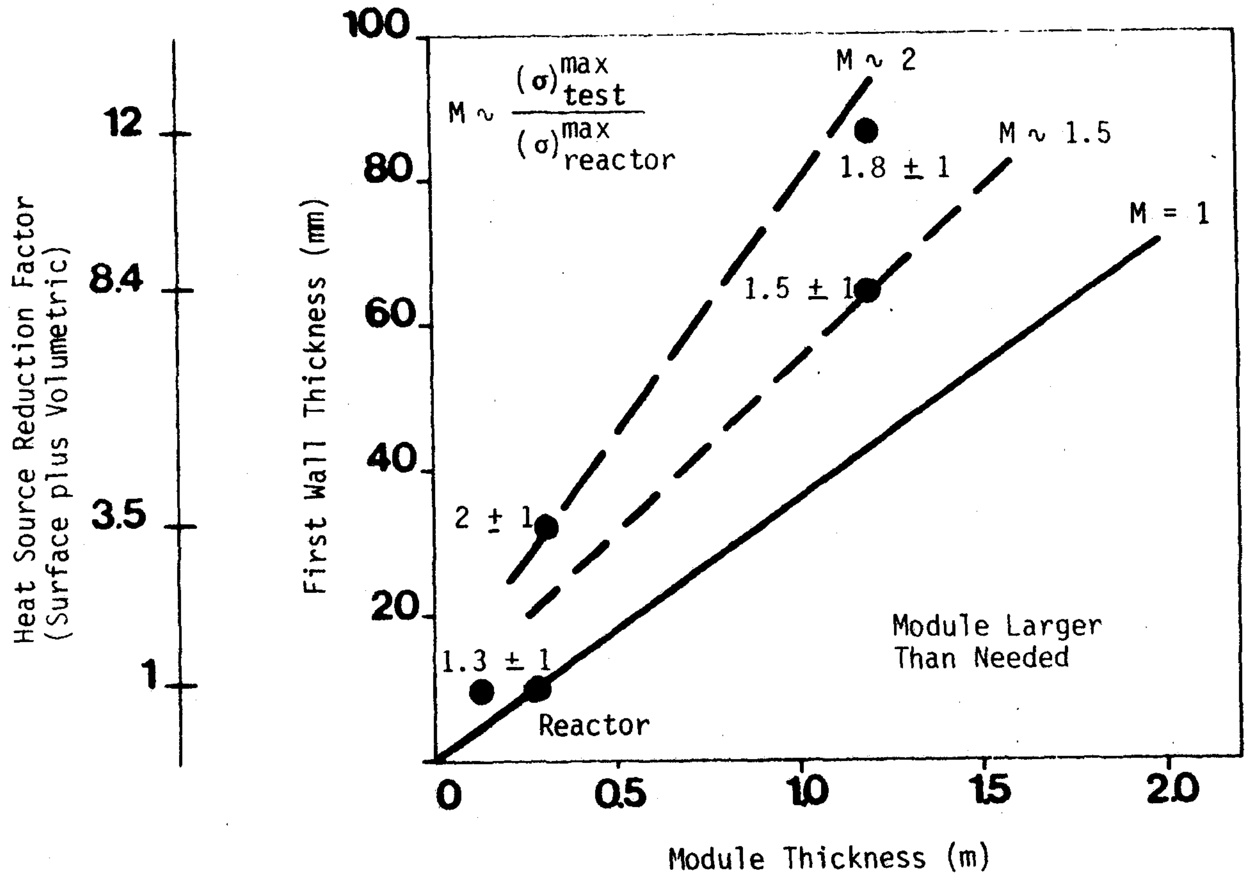


Figure 2.4.2-8 Test module width and device heat source tradeoff for preserving Li_2O/He/HT-9 first wall thermal plus pressure stresses. M is a qualitative measure of the (multiplicative) change in first wall stress profile.

lead to different time-dependent stress behavior than would be observed in a reactor.

A third consequence of increasing dimensions, both in the first wall and in the breeder, is that the thermal and flow time scales increase. This increases the minimum burn time requirements in order to reach thermal equilibrium, but correspondingly increases the tolerable thermal dwell time. Ideally, the test should be operated beyond any significant startup transients and allowed to settle into its equilibrium operating mode. In practice, this might be achievable by single, long pulse burns or a series of pulses maintaining quasi-equilibrium conditions for the needed cumulative operating time. However, cycling is generally undesirable since it can activate processes that are not normally significant such as crack growth, thermal ratchetting or surface barrier degradation. In addition, cycling may introduce uncertainties in interpreting experimental results.

Any alteration in temperature distribution with time is particularly important for the solid breeder, where basic processes are not well understood. Minimizing thermal variations due to pulsing leads to burn and dwell time requirements as a function of neutron wall load (i.e., bulk heating rate) as shown in Fig. 2.4.2-9 for the Li_2O/He/HT-9 layered breeder design. Long dwell times may be less problematic if the breeder is brought to tritium equilibrium during a single pulse.

The verification of tritium behavior is accomplished by monitoring the tritium release rate and final inventory. Generally, attaining 67% of the equilibrium release rate occurs early in the test and can be accurately measured, but 99% recovery or inventory requires substantial operating times (see Fig. 2.4.2-10). Present calculations assuming the addition of hydrogen into the purge stream indicate that intragranular diffusion is the largest contributor to the total inventory. Consequently, the Li_2O and $LiAlO_2$ designs will probably achieve 67% of the equilibrium release rate within about one minute, independent of the neutron wall load. In order to reach inventory equilibrium, however, total operating times of minutes, days and months are needed for Li_2O, hot $LiAlO_2$ (over 510°C), and cold $LiAlO_2$ (over 350°C) breeder designs, respectively. Other processes have time scales on the order of a day (solubility, surface adsorption) or months (fluence effects), which will increase as the neutron wall load and tritium generation rate decrease.

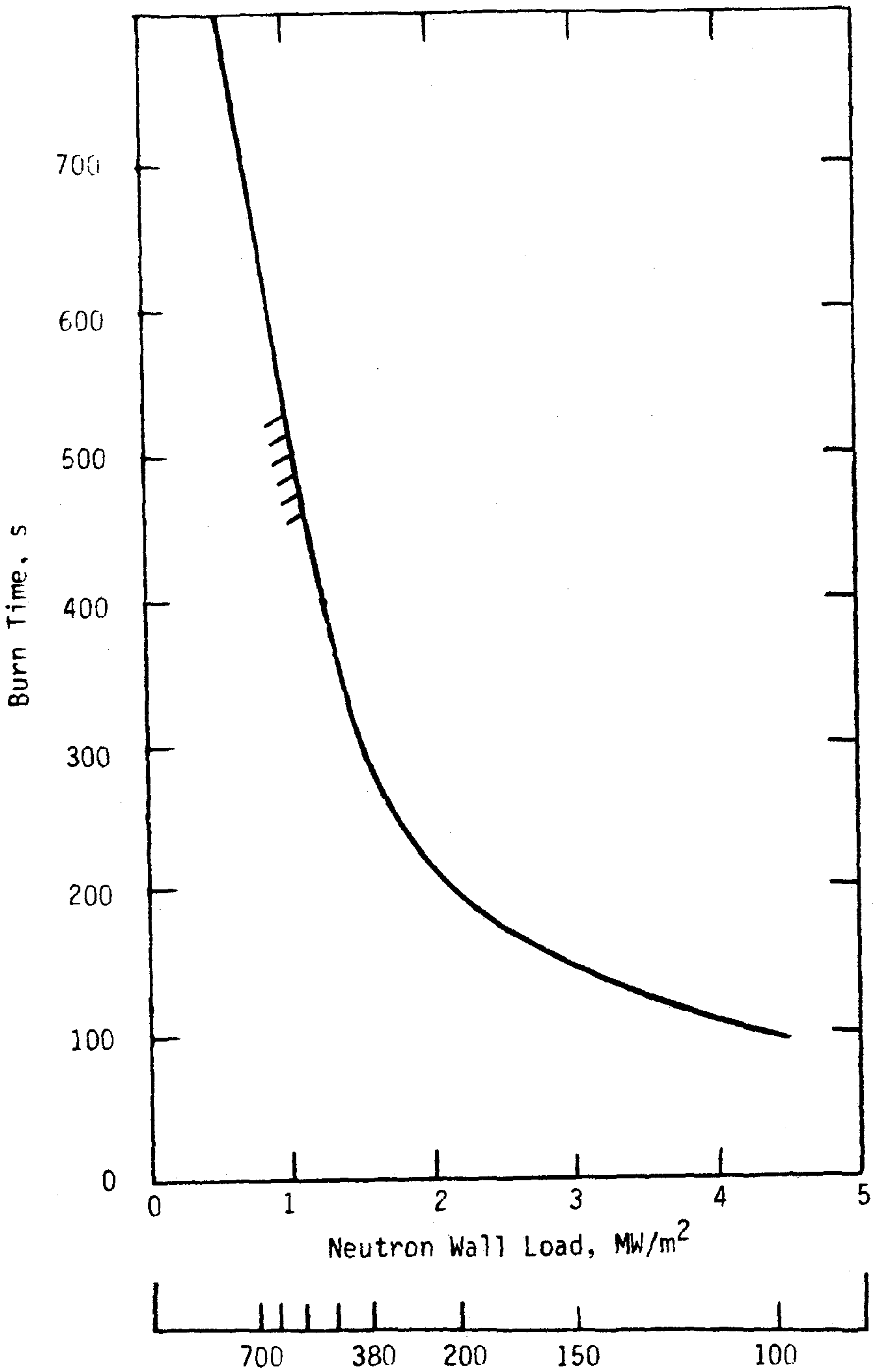


Figure 2.4.2-9 Relation between minimum burn time, maximum dwell time and minimum neutron wall load for breeder thermal equilibrium in the Li_2O/He/HT-9 test module. Breeder dimensions are changed so as to keep the breeder within the reactor temperature limits.

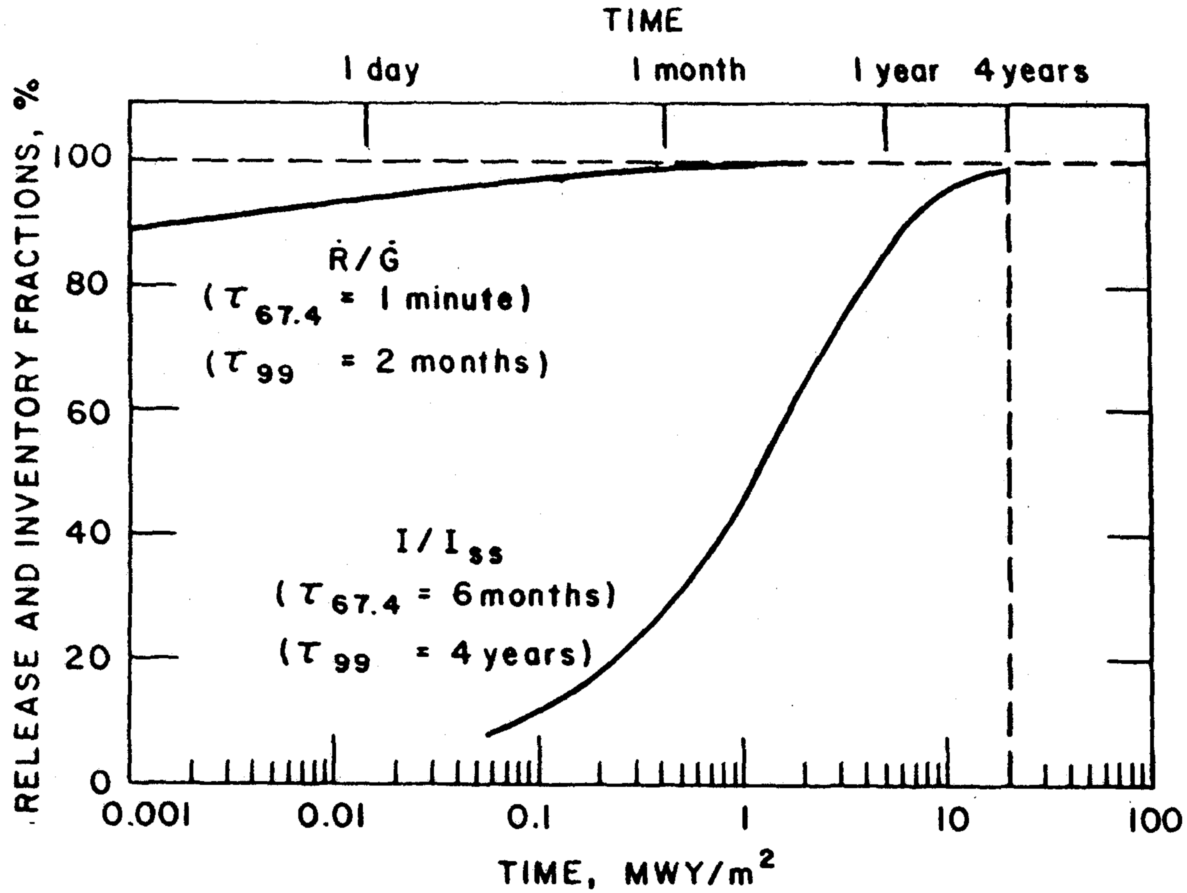


Figure 2.4.2-10 Tritium release and inventory behavior after startup in the reference $LiAlO_2/H_2O$/PCA/Be blanket. Test times to achieve substantial release rates (67% of equilibrium value) are relatively short, but equilibrium inventory requires many months due to the low diffusivity of $LiAlO_2$.

Other scaled test considerations suggest an axial length of 0.5 m is sufficient to simulate flow distribution effects under reactor conditions, increasing in parallel with other dimensions as the heat source is reduced. The module depth must include the 0.15 m reactor first wall depth, plus at least 0.2 m for the breeder in order to breed substantial tritium and to allow any interactions between the high fluence and low fluence breeder regions. No detailed evaluation of magnetic field effects was made, since they are not presently believed to be significant (except possibly for transient forces) in solid breeder blankets. Nonetheless, the energy density of a 5 T magnetic field is about 10 MJ/m^3 and there are possible interactions through structure or corrosion, so this environmental condition should not be entirely neglected.

In summary, it seems that a complete solid breeder module could be tested in under 1 m^3 of test volume. There is a limit to how much the heat source (surface and volumetric) may be reduced before several aspects of act-alike behavior are lost. Furthermore, since there are many basic uncertainties in solid breeder tritium recovery, pulsing should be minimized.

Finally, it should be noted that there are many phenomena and interactions that were not considered. These include specific structural failure modes related to plastic behavior or crack growth in the first wall, the breeder/structure interaction, and fluence effects. Thus, these results should be considered as optimistic with respect to the test requirements.

2.4.3 Failure Modes

First wall/blanket structures are subjected to high temperature, large thermal gradients, coolant pressures, dead weight loads, corrosive environments, and severe radiation. The interaction of thermal stresses, thermal creep, irradiation creep, and swelling will result in complex time, temperature and neutron fluence dependent stress histories in first wall/blanket components. The interaction of these phenomena can cause failure. Cyclic operation, which may be present in some reactors, can also enhance failure. The effects of temperature gradients and neutron spectrum can produce differential swelling and irradiation creep resulting in excessive stresses in the first wall/blanket that can lead to failure.

Swelling can cause stresses in the component by three different methods. First, differences in the swelling rate for different parts of the structure, caused by flux or temperature gradients, can result in increased stress (similar to thermal gradients producing thermal stresses). Second, overall constraint of the component may not allow swelling stresses to be relieved by expansion; if swelling stresses are high, buckling or crippling of the component can occur. Third, stresses can be imposed on the component by differential swelling rates between different materials. If one material swells at a higher rate, it will exert a force on the structure which could produce local crippling or buckling of the component. Irradiation creep will help to relieve swelling stresses resulting from these three sources. However, at end-of-burn or shutdown these stresses will reverse and could cause cracking. On a cyclic machine, this stress reversal may result in a shorter design life.

Distortion of components can result from both primary and secondary stresses. Primary stresses are those stresses which result from mechanical loadings such as coolant pressure, vacuum loading, and electromagnetic loads. Secondary stresses are those stresses which result from thermal gradients, swelling, and creep. The effects of these stresses are enhanced by cyclic operation. Swelling is the most likely source for causing excessive deformation of the component. Excessive deformation can lead to failure of the component or cause problems for removal of the component.

Wall loading will have a large effect on determining the critical failure mode of a first wall/blanket. Based on stress histories and associated fracture mechanics analysis, life at low wall loadings is controlled by stresses during the plasma burn. At higher wall loadings, life is controlled by residual stresses present during the non-burn period. The primary failure mode in either case is likely to be flaw growth to coolant leakage.

The most likely location for a failure in the first wall/blanket is at a point of stress concentration. Stress concentrations will occur at abrupt changes in the cross section, at discontinuities in the material itself, and at cracks that may be present in the structure. The magnitude of the increased localized stress at a point (stress concentration) is dependent on the internal state of the metal, state of stress, stress gradient, temperature, and rate of straining. Often, large stresses due to stress concentrations are developed in only a small portion of a member. In cases where this stress is highly localized, a mathematical analysis is difficult or impractical. Therefore, experimental or numerical methods of stress analysis are used. Because of the difficulty in predicting these stress concentrations, testing will be beneficial in locating areas of stress concentrations and will also build confidence in the design.

Swelling of the first wall/blanket will increase stresses at discontinuities by changing the state of stress and stress gradients. Thermal gradients will produce thermal stresses at the stress concentration. In addition to thermal stresses, thermal gradients will produce differential swelling stresses. Therefore, a fusion environment will most likely increase stresses which, in turn, will decrease life.

In addition to the bulk damage effects (irradiation creep and swelling) resulting from an irradiation environment, the first wall will be subjected to damage from charged particles, neutrons, neutral particles, and electromagnetic radiation. These will cause sputtering of particles from the free surface, blistering by implantation and transmutations. Erosion of the first wall structure will cause increased primary stresses which can lead to unstable failure.

The likelihood of occurrence from these various failure modes was established for a matrix of four reactor types. The four reactor types considered were a steady state tokamak, a steady state mirror machine, a pulsed tokamak with a moderate wall loading, and a pulsed tokamak with a high wall loading. The four blanket concepts considered were representative of those examined by the Blanket Comparison and Selection Study.

The failure modes established for the first wall/blanket are shown in Fig. 2.4.3-1. These failure modes were ranked by likelihood of occurrence for each reactor type and first wall/blanket concept. The failure modes and rankings were reviewed by the fusion community at the "Materials Testing Workshop for the EPRI Assessment of Neutron Requirements and Potential Sources for Fusion Development Project" and by members of FINESSE and the Blanket Comparison and Selection Study.

Based on these failure modes, a list of critical issues relating to failure modes and structural response of the first wall/blanket was

FWB FAILURE MODES[a]	STEADY STATE								PULSED TOROIDAL							
	TOKAMAK				TMR				MOD.WALL LOAD				HIGH WALL LOAD			
	A	B	C	D	A	B	C	D	A	B	C	D	A	B	C	D
Cracking Around a Discontinuity/Weld	H^1	H^2	H^1	H^1	H^1	H^2	H^1	H^1	H^1	H^1	H^1	H^1	H^1	H^1	H^1	H^1
Crack on Shutdown (with cooling)	H^2	L	H^3	H^2	H^2	L	H^3	H^2	H^2	H^2	H^2	H^2	H^1	H^1	H^1	H^1
Breeder Disintegrates/Cracks	H^3	H^3	N/A	N/A	H^3	H^3	N/A	N/A	H^4	H^4	N/A	N/A	H^3	H^3	N/A	N/A
FW/Breeder/Structure Swelling & Creep Leading to Excessive Deformation or FW/Coolant Tube Failure	H^1	H^2	H^3	H^3	M-H	M-H	M-H	M-H	M	M	M	M	M	M	M	M
Crack During Operation (FW/Breeder/Structure)	M	H^{1b}	M	M	M	H^1	M	M	H^3	H^3	H^3	H^3	H^2	H^2	H^2	H^2
Environmentally Assisted Cracking	M	H^{1c}	H^1	H^1	M	H^1	H^1	H^1	M	H^2	H^2	H^2	M	H^2	H^2	H^2
Crack on Start-up (FW/Breeder/Structure)	H^2	L	H^2	H^2	H^2	L	H^2	H^2	H^2	L	H^2	H^2	H^1	L	H^1	H^1
Excessive Tritium Permeation of Coolant Tubes	M	M	N/A	N/A	M	M	N/A	N/A	M	M	N/A	N/A	M	M	N/A	N/A
FW/Breeder/Structure Melting	L	L	L	L	L	M	L	L	L	M	L	L	L	M	L	L
Manifold Tube Breaks	L	L	L	L	L	L	L	M^d	L	L	L	L	L	L	L	L
Insufficient Tritium Diffusion Through Breeder	L	L	N/A	N/A	L	L	N/A	N/A	L	L	N/A	N/A	L	L	N/A	N/A

A - Li_2O/He/HT-9
B - $H_2O/LiAlO_2$/PCA
C - Li/V Self Cooled
D - LiPb/HT-9 Self Cooled

H - Highest Likelihood of Failure
M - Medium Likelihood of Failure
L - Lowest Likelihood of Failure
N/A - Not Applicable Failure Mode for That Blanket Concept

[a] Failure Modes Ranked with Most Likely First
[b] High Pressure
[c] H_2O Corrosion
[d] Based on MARS Constrained Header Design
[1] Superscript indicates relative ranking of H's by column

Figure 2.4.3-1 Likelihood of occurrence for various failure modes.

established. These issues were classified by design specificity and overall level of concern and are shown in Table 2.4.3-1.

The top-ranked issue, "Effect of First Wall Heat Flux and Cycling on Fatigue or Crack Growth Related Failures," has a critical level of concern because as wall loading is increased, thermal stresses will increase and therefore life will decrease. If life is greatly decreased, the design window could be seriously impacted.

Although cracking around a discontinuity/weld is the most likely failure mode, it was ranked as the second most important issue. Failure of a weld has a high level of concern but is not critical because if a failure occurs, the design can be changed to reduce stresses at that location. On the other hand, if the first wall heat flux is higher than expected, a design change may not significantly increase life of the first wall.

The objective of subscale testing of the first wall/blanket is to predict the response of a full-scale component in a radiation environment. From the defined failure modes for the first wall/blanket, subscale component tests can be defined. These tests will be "act-alike" test specimens, rather than "look-alike" test specimens. To avoid design tradeoffs, generic designs will be used to define the subscale tests.

Detailed structural analyses will be performed to identify irradiation levels which will yield useful engineering data. The response of the test specimen to an irradiation environment will be predicted. Through testing and correlation of results to predictions, the method of analyses used to predict the response of the first wall/blanket can be verified.

Subscale testing will also reveal any potential failure modes which were not considered in the design and analysis of the test article. That is, if the test article fails by a different mechanism than predicted, then analysis methods need to be revised to incorporate test results. Therefore, subscale testing will reveal these unpredicted failure modes and, in addition, help to build confidence in the design.

Table 2.4.3-1 Summary of Issues for Failure Modes and Structural Response

Issue	Design Specificity	Level of Concern
1. Effect of first wall heat flux and cycling on fatigue or crack growth related failure	Generic/ Tokamak	Critical
2. Cracking around a weld	Generic	High
3. Hot spots leading to failure	Generic	High
4. Interaction of primary stresses, secondary stresses and deformation	Generic	High
5. Effect of swelling, creep and thermal gradients on stress concentrations - Response of grooved surface concepts	Generic/ Design	Medium
6. Failure due to shutdown residual stress effects	Generic	Medium
7. Interaction between surface effects and first wall features	Generic	Medium
8. Mechanical wear and fatigue from flow-induced vibrations	Generic	Medium
9. Environmentally assisted cracking	Generic	Low
10. Self welding of similar and dissimilar metals	Generic	Low

2.4.4 Fluence Goals

One of the specific objectives of FINESSE is to quantify the testing requirements needed to develop reliable fusion components. One of the major parameters which must be quantified in these testing requirements is neutron fluence. In a typical development scenario for components, tests of the component in both typical and off-normal environments out to, and in some cases, beyond its design goal life would be desirable. However, the very high cost of performing these tests in a fusion-like neutron spectrum forces reconsideration of such an approach for fusion reactor component development.

In considering fluence goals for a testing program, it is necessary to identify what information is learned with increased fluence in nuclear experiments. Also, due to the cost tradeoffs involved in the development of a fusion test facility, the question of what risk must be assumed if nuclear testing were not performed on selected issues associated with a given component must be addressed. The answers to these questions at present must rely upon available materials properties data, present knowledge of neutron fluence effects on materials, and engineering judgement.

The following approach for identifying neutron goal fluences has been applied in FINESSE. The four specific blanket concepts mentioned earlier were considered. The major material properties impacted by neutron fluence were identified and their anticipated behavior ranges with fluences were determined. A method of performing uncertainty projections was then used to assess how the uncertainty for a given material property at the end of life decreased as a function of test neutron fluence. The method of uncertainty projections was then extended to interactive effects experiments assuming that the uncertainty associated with the interaction was directly related to the uncertainties associated with the major properties of materials involved in the interaction. This approach was then applied to the key issues associated with the blanket component's structure in order to assign fluence goals for these key issues.

To assess the importance of neutron fluence on component performance, it is first necessary to understand how the major material properties change with neutron fluence. Our present understanding in this area is based predominantly upon materials testing in fission reactors. Limited data are also available from irradiations in ion sources and the low fluence fusion neutron source, RTNS-II. In general, the fluence dependence for many material properties can be divided into two regimes: 1) a transient regime where the material property is constantly changing with fluence, and 2) a steady-state or saturation regime where the material property is either changing at a constant rate or not changing at all. For example, swelling of the structural materials typically exhibits an incubation period prior to the onset of swelling. After this incubation period, the swelling behavior then undergoes a transient regime until the steady-state swelling rate is achieved. At present, no swelling has been observed in the structural materials being investigated in FINESSE. Based upon limited fission reactor irradiation data, it is anticipated that the incubation fluence for these structural materials will be in the following ranges: 5-8 MW•yr/m^2 for PCA and > 10 MW•yr/m^2 for the HT-9. At present, the swelling in V-15Cr-5Ti is expected to be similar to that of HT-9.

In an attempt to quantify the information gained as a given material property is tested to higher and higher fluences, the concept of uncertainty projections was developed. Given a fluence-dependent model and hypothetical results from a specific materials testing plan, the uncertainty in a given property can be extrapolated to some goal fluence. By requiring these extrapolations to also be consistent with the hypothetical results up to some peak fluence, the reduction in the material property uncertainty at the goal fluence can be quantified as a function of testing fluence.

This approach was applied to the properties of materials being considered in FINESSE. In general, the uncertainties in key properties associated with the strength and fracture behavior of the structural materials are resolved after neutron testing in the fluence range of 3-5 MW•yr/m^2. For creep in the structural materials, the uncertainty projections are dependent upon the goal fluence under consideration. Specifically, if the goal fluence is < 10 MW•yr/m^2, then the swelling behaviors of HT-9 and V-15Cr-5Ti are not an issue, and the uncertainty in creep is significantly reduced after measurements are performed in the neutron exposure range of 1-3 MW•yr/m^2. If the goal fluence under consideration is 20 MW•yr/m^2, then the projected uncertainties for creep and swelling remain large until measurements are performed after the onset of swelling. The method of uncertainty projections assumes that for a given testing program the most information is gained on the material property when testing in the transient regime and that once the steady-state or saturation regime is reached, the importance of further neutron testing is reduced.

It is important to remember that these observations are based upon material behavior in a fission neutron environment for neutron fluences less than those anticipated in a commercial fusion reactor. Limited data suggest that the fusion environment may alter the material behavior with fluence, and these material property changes may be different than those observed or anticipated based upon our fission reactor experience. Also, the concept of saturation in a material property behavior beyond a certain fluence may change as testing is extended to higher fluence.

The major emphasis in FINESSE has been on interactive effects. The general nature of the interaction is dependent upon the material properties responsible for the interaction, and fluence dependence for the interaction is related to the fluence dependence of these material properties. The specific interaction, however, will in general be very design dependent. In considering the fluence dependence of the individual material properties, one can identify the major material properties which are changing with fluence and have a significant impact upon the interaction. For example, consider the mechanical interaction between the solid breeder and the HT-9 cladding in the HT-9/Li_2O/He blanket concept. The changes in several of the key properties with neutron exposure are given in Fig. 2.4.4-1. The swelling in Li_2O is the

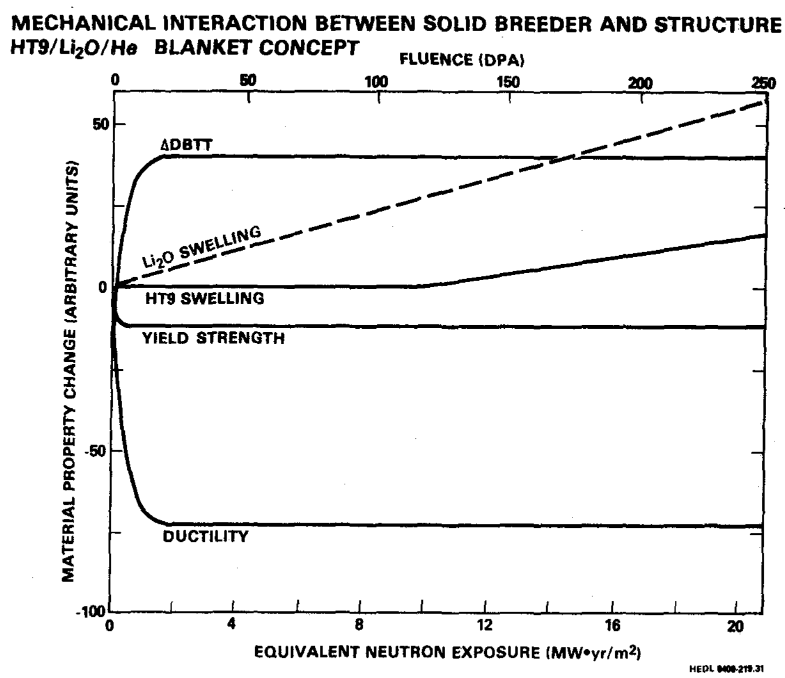


Figure 2.4.4-1 Fluence dependence for several of the key material properties associated with the mechanical interaction between the Li_2O and HT-9 cladding.

major material property responsible for the interaction. Thermal expansion and cracking/redistribution of the Li_2O also plays a role in this interaction, primarily in the early operation of the component (i.e., in the neutron exposure range of 0-0.2 MW•yr/m^2). After a neutron exposure of ~ 2 MW•yr/m^2, most of the strength and fracture property changes have saturated, and the major properties responsible for the interaction are the balance between Li_2O swelling and the creep of Li_2O and HT-9 (not shown in the figure). After a neutron exposure of 10 MW•yr/m^2, HT-9 swelling becomes important. Specific interactive effects tests will not be defined until the second year of FINESSE. However, the key issues defined this year can be considered to assess the general fluence goals required to understand their inherent interactions.

In an attempt to quantify the information gained in interactive testing as a function of testing fluence, the method of uncertainty projections was extended (see Chapter 8). This approach was used to identify fluence goals for the general key issues for the blanket structure which were identified by FINESSE. This approach assumes that the uncertainty associated with an interactive effect is directly related to the fluence uncertainties associated with the major material properties involved in the interaction. As an example of this approach, consider the mechanical interaction between the Li_2O solid breeder material and the HT-9 cladding for the HT-9/Li_2O/He blanket concept. The key material behaviors pertinent to this interaction are Li_2O thermal expansion, cracking/redistribution, creep and swelling, and HT-9 creep and ductility (if the goal fluence is 10 MW•yr/m^2). Uncertainty projections were performed for each of these properties and are combined in Fig. 2.4.4-2. Also shown in Fig. 2.4.4-2 is the absolute value of the derivative for this combined uncertainty. The fluence regions where the combined uncertainties are changing the fastest also represent the fluence regions where the rate of information gained from testing is the fastest. These regions are reflected as relative maxima in plots of the absolute value of the derivative of the combined uncertainties. Specifically for the mechanical interaction between the Li_2O solid breeder and HT-9 cladding, nuclear experiments out to ~ 0.2 MW•yr/m^2 yield most of the information concerning the impact of Li_2O thermal expansion and cracking/redistribution on the interaction. Nuclear testing in the 3-5 MW•yr/m^2 fluence range yields most of the information concerning the remaining facets of the interaction.

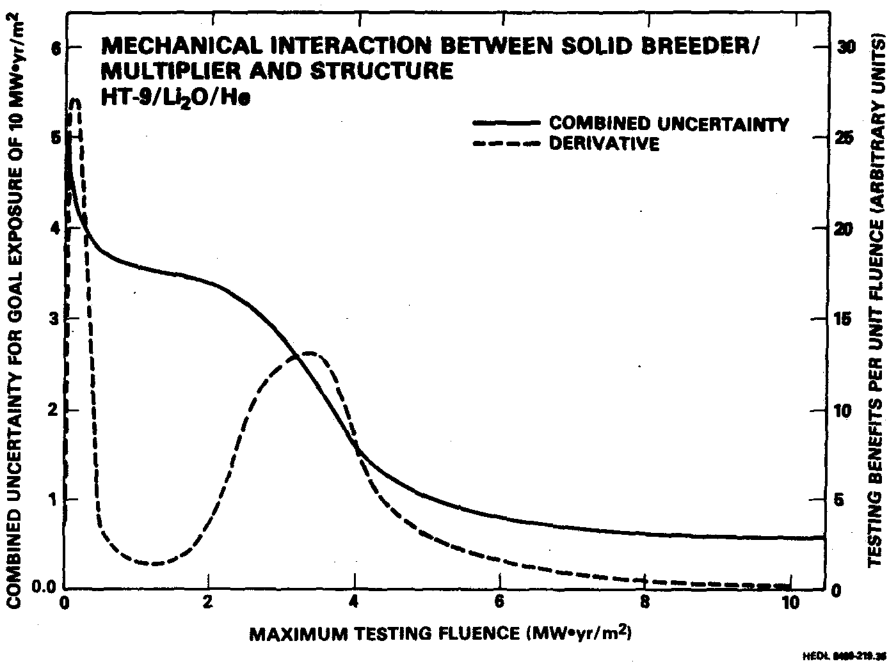


Figure 2.4.4-2 The fluence dependence for the combined uncertainty associated with the material properties associated with the mechanical interaction between the Li_2O and HT-9 cladding. Also shown is the absolute value of the derivative of the combined uncertainty which provides a qualitative measure of the testing benefits per unit of testing fluence.

This approach has been applied to the key issues for the HT-9/Li_2O/He blanket concept, and the results are summarized in Table 2.4.4-1 for those issues relating to the structure. The results of the analysis have suggested that the goal fluences required to resolve these issues are typically in the range of neutron exposures corresponding to 1-5 MW•yr/m^2 if the application of the component is for a goal life of 10 MW•yr/m^2 (i.e., below the projected onset of significant swelling in the structural alloy HT-9). If the final application of the component is for a goal life which is comparable to or beyond the fluence at which swelling in the structural material becomes a major property in the interaction, then the goal fluence for component testing would be ~ 2-4 MW•yr/m^2 beyond the onset of swelling if all uncertainties associated with the interaction are to be addressed.

Table 2.4.4-1 Neutron Fluence Goals for HT-9/Li_2O/He Concept

Issue	Major Material Properties	Fluence Goal ($MW \cdot yr/m^2$)
Changes in properties and behavior of materials	(All material properties pertinent to design)	(See Chapter 8 for specimen fluence goals)
Mechanical interaction within the structure itself	HT-9: thermal differential expansion, fatigue/crack growth, ductility, swelling[a]	3-5; (16-20)[a]
Plasma-structure interaction	—	—
Premature failure of welds in full components	HT-9: thermal differential expansion, fatigue/crack growth, ductility, swelling[a]	3-5; (16-20)[a]
Coolant-structure interaction	—	—
Magnetic field interaction with structure	HT-9: ductility, fracture toughness, fatigue/crack growth, creep	3-5
Mechanical interaction between solid breeder/multiplier and structure	HT-9: ductility differential expansion, fatigue/crack growth, ductility, swelling[a]	3-5; (16-20)[a]

[a]Testing fluence goal required if data to be applied to design with goal life $\geq$ 20 $MW \cdot yr/m^2$.

2.4.5 Neutronics Tests

Integrated tests performed in a fusion test device and aimed specifically at verification of neutronics methods and data require specialized modules. In contrast to issues such as thermomechanical behavior, in which look-alike test modules are least useful under scaled down conditions, neutronics verification tests require that, and are most useful when, the test module is as close to a look-alike as possible. Therefore, neutronics tests have been treated separately from other types of tests. Notice that other types of blanket tests (e.g., thermomechanical, tritium recovery) have their own neutronics considerations concerning simulation of bulk heating, tritium production, etc. to simulate the act-alike behavior that are different from those aimed specifically at neutronics verification tests which are considered in this subsection. The former has been treated in the context of the act-alike tests.

Neutronics testing in a fusion test device will involve several types of measurements such as source neutron yield, tritium production rate, neutron and gamma-ray spectra, heating rates during operation, activation and afterheat. The requirements for neutronics testing fall within two categories: (a) test device operating conditions and (b) test module conditions. The fusion test device conditions include parameters such as the wall load, fluence, and pulse length. The test module conditions are those related to the test module material and configuration, surface area exposed to neutron field, minimum size requirement for optimal testing, and requirements on the test module boundary conditions and geometrical arrangement.

2.4.5.1 Test Device Operating Condition Requirements

From instrumentation considerations, all neutronics parameters except induced activation can be measured in one of two fluence modes: either the low fluence mode ($\sim 1\ MW{\cdot}s/m^2$) or the very low fluence mode ($\sim 1\ W{\cdot}s/m^2$). The low fluence mode can be achieved, for example, with a wall load of 1 MW/m^2 and 1 s plasma burn time or, alternatively, 0.01 MW/m^2 and 100 s. Thus, neutronics tests impose only modest requirements on the product of the wall load and plasma burn time with no stringent requirements on the magnitude of either parameter since the neutronics parameters, except induced activation, vary linearly with both the wall load and operating time. Notice, however, that much larger fluences than those considered here will require the use of different, less accurate, measurement techniques.

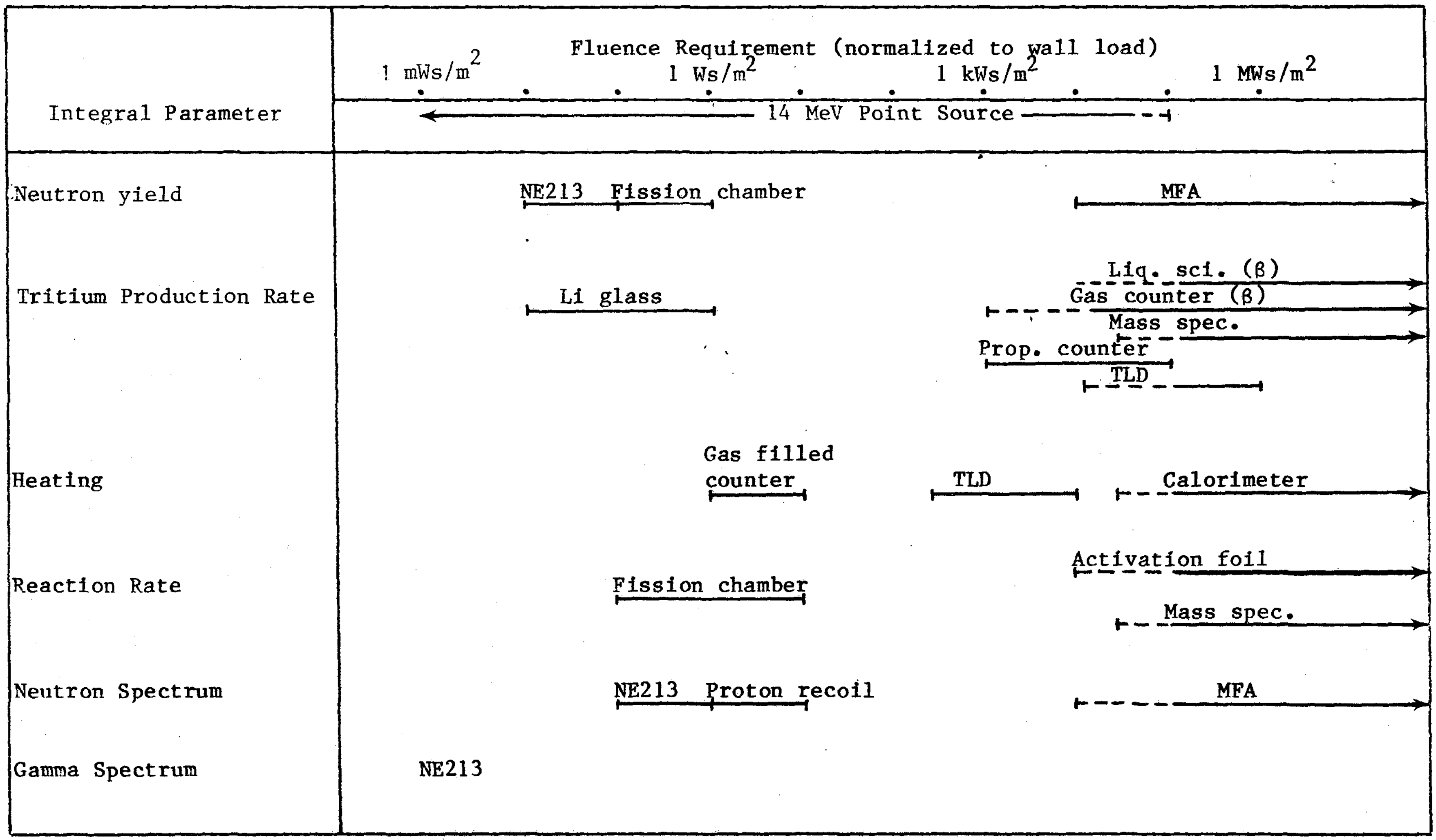
Operating the test device in the very low fluence mode is most suitable for measuring tritium production from ^6Li, gamma-ray heating, neutron and gamma-ray spectra. The main problem associated with the low fluence operating mode is the activation of the test module and device components which may render the test device inaccessible just after shutdown. This will necessitate a long cooldown time to handle the reactor components. On the other hand, the main problem related to the very low fluence operation mode is the poor resolution and instability of measurements. The methods used for measurements in the low fluence mode have better accuracy and spatial resolution as compared to those used in the very low fluence operation mode. For source characterization and neutron yield, which is viewed as a part of the plasma diagnostics, measurements can be undertaken in both operation modes. The methods that can be used to measure various parameters under various operating conditions are summarized in Table 2.4.5-1.

2.4.5.2 Test Module Requirements

The requirements on the test module size and geometry are governed mainly by the objective and the procedure for the particular neutronics test under consideration. If a local measurement of tritium production is intended, for example, then the only useful information that can be obtained from such a neutronics test is to verify the consistency between analytical prediction and experimental measurements. Resolving the question of the adequacy of the nuclear data base can be better achieved in a simple benchmark experiment. However, to improve the analytical prediction and to identify the various sources of uncertainties, one would proceed from a geometrically simple benchmark experiment utilizing a point source to a more complicated one involving a volumetric plasma source in a fusion test device.

On the other hand, if the objective of the neutronics test is to verify an integrated parameter in a given blanket concept such as the tritium breeding ratio (TBR), the test module used in this experimental planning approach should duplicate in great detail the actual blanket module. Verification of achievable breeding ratio requires also that factors that affect the global TBR, such as actual penetrations for heating and fueling, full coverage

Table 2.4.5-1 Fluence Requirements for Various Experimental Techniques



*For counter methods, the measuring time is assumed to be 10 ~ 100 s.

geometrical arrangement, and presence of the impurity control system, be included in the fusion test device. This is an important point of concern since extrapolating the results of measuring the local tritium production rate (TPR) in a partial coverage case to demonstration or commercial reactor TBR involves many uncertainties. These uncertainties arise from: a) uncertainty in specifying the neutron source condition at the first wall of the test module, b) uncertainties in predicting (by calculation or measurements) the energy-dependent angular flux at the test module boundaries, c) uncertainties in extrapolating the effects of penetrations and other configurational features of demonstration or commercial reactors that cannot be easily reproduced in a fusion test device. Since the estimated margin in TBR for candidate blanket concepts is small, very high accuracies in measurements are required, and these sources of uncertainties need to be carefully evaluated.

In a fusion test device, the test area at the first wall is limited by considerations of cost. Hence, a near-full-coverage blanket for neutronics verification tests is obviously not practical. Therefore, the neutronics analysis has focused on examining the usefulness of neutronics test information as a function of the test module size. In addition, an effort to improve the usefulness of test information from a given size test module has been attempted. Variables considered in such an improvement included: 1) the details of material and geometrical arrangement within the test modules, and 2) the conditions at the test module boundaries which are sensitive to factors such as the material and dimensions of the "reflective" region surrounding the test module. Examples of results and conclusions on tritium breeding test modules are summarized below.

The surface area of the test module at the first wall can be characterized by two dimensions in a fusion device which is approximated by a cylinder. The first parameter is the magnitude of the maximum poloidal angle, Θ_m, subtended by the test module. The second is the maximum width, L_m, of the test module in the axial direction of the device. The importance of Θ_m and L_m was examined separately by two 2-D models. The first is an R-Θ geometry, shown in Fig. 2.4.5-1, where R refers to the minor radius of the plasma. The second is an R-Z model, shown in Fig. 2.4.5-2, where Z is the axial direction for the plasma along which L_m is measured.

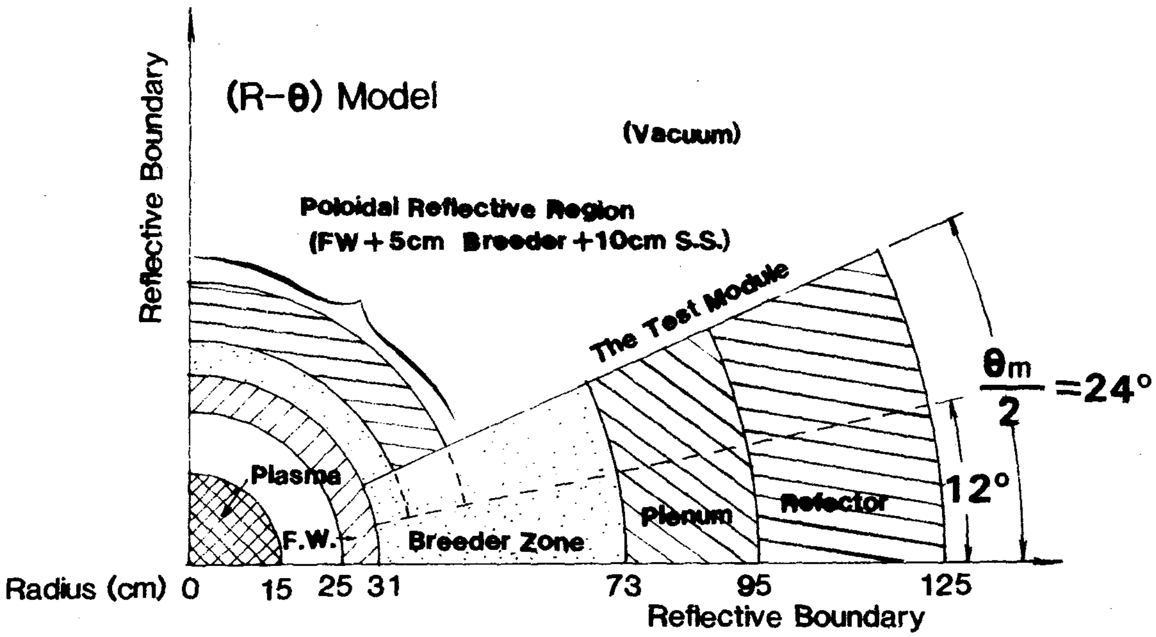


Figure 2.4.5-1 The (R-Θ) geometrical model used to examine the poloidal boundary condition.

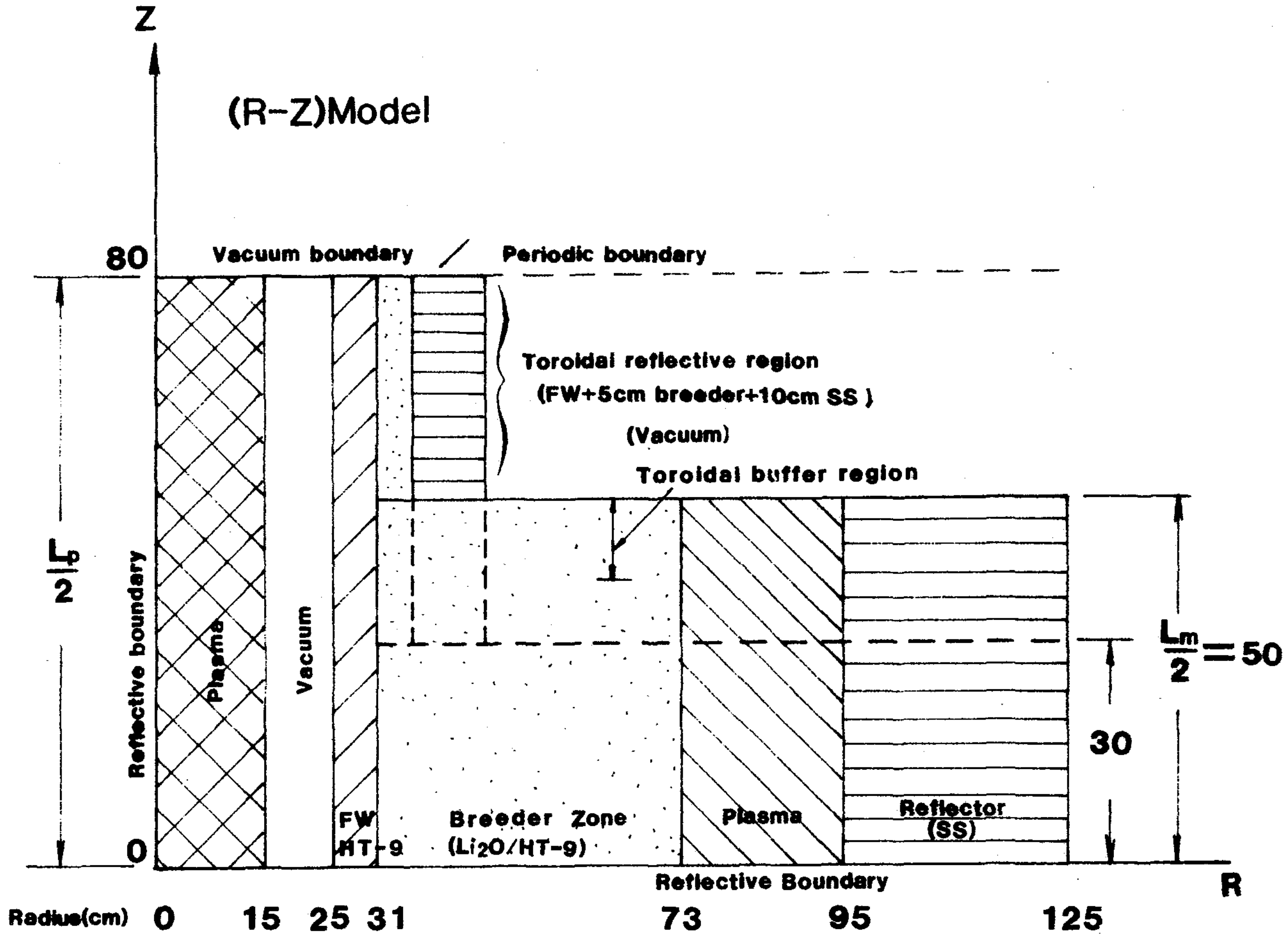


Figure 2.4.5-2 The (R-Z) model used in the 2-D calculation.

Figure 2.4.5-3 shows the variation in the local tritium production rate (sum of T_6 and T_7) as a function of the poloidal angle Θ_m at three locations on the central line of the test module. The values shown are normalized to the corresponding values in the full coverage case. Also shown in this figure is the maximum percentage deviation of the tritium production rate at $\Theta_t/2$ = 5° and at 10° from that at the central line. This deviation depends on the width of the test module, which is characterized by the angle Θ_m, and on the location throughout the test module, e.g., top or middle, as shown in Fig. 2.4.5-3. The information contained in this figure is used to specify the minimum test module width (Θ_m) that is required to obtain a tritium production rate at a given location inside the test module which is within a desired target percentage of the corresponding value in the full coverage case. For example, if the local tritium production rate at the front edge of the test module central line is required to be within 5% of the corresponding value of the full coverage case, the width of the test module should be the one that corresponds to either Θ_m = 22° (-5% deviation) or Θ_m = 48° (+5% deviation). If measurements were to be performed at the back-edge of the test module central line, the corresponding values would be Θ_m = 48° or 71° for the same target accuracy. The situation is different at the middle location where a test module width that corresponds to Θ_m = 55° would give a 5% target accuracy. In addition to this prescribed deviation, it is necessary to add the incremental contribution that comes from performing the measurements within the spatial zone characterized by the angle Θ_t.

Similar curves that specify the minimum test module size in the R-Z geometrical model are shown in Figs. 2.4.5-4 and 2.4.5-5 for the cases where the plasma length is L_p = 160 and 320 cm, respectively. In this geometrical arrangement, the test module width is characterized by the parameter L_m while the test module-central zone where measurements are most likely to be performed is characterized by the parameter L_t. For the case with L_p = 160 cm, the tritium production value at the front-edge of the test module is less than 80% of the corresponding value in the full coverage case and the situation even worsens as the test module width, L_m, increases. However, for the case with a plasma length L_p = 320 cm, the tritium production rate at the front-edge of the test module is within 5% of the corresponding value in the full coverage case provided the test module width is either L_m = 50 cm (+5% deviation) or L_m = 150 cm (-5% deviation). For the former case, one should add ~ 3.4%

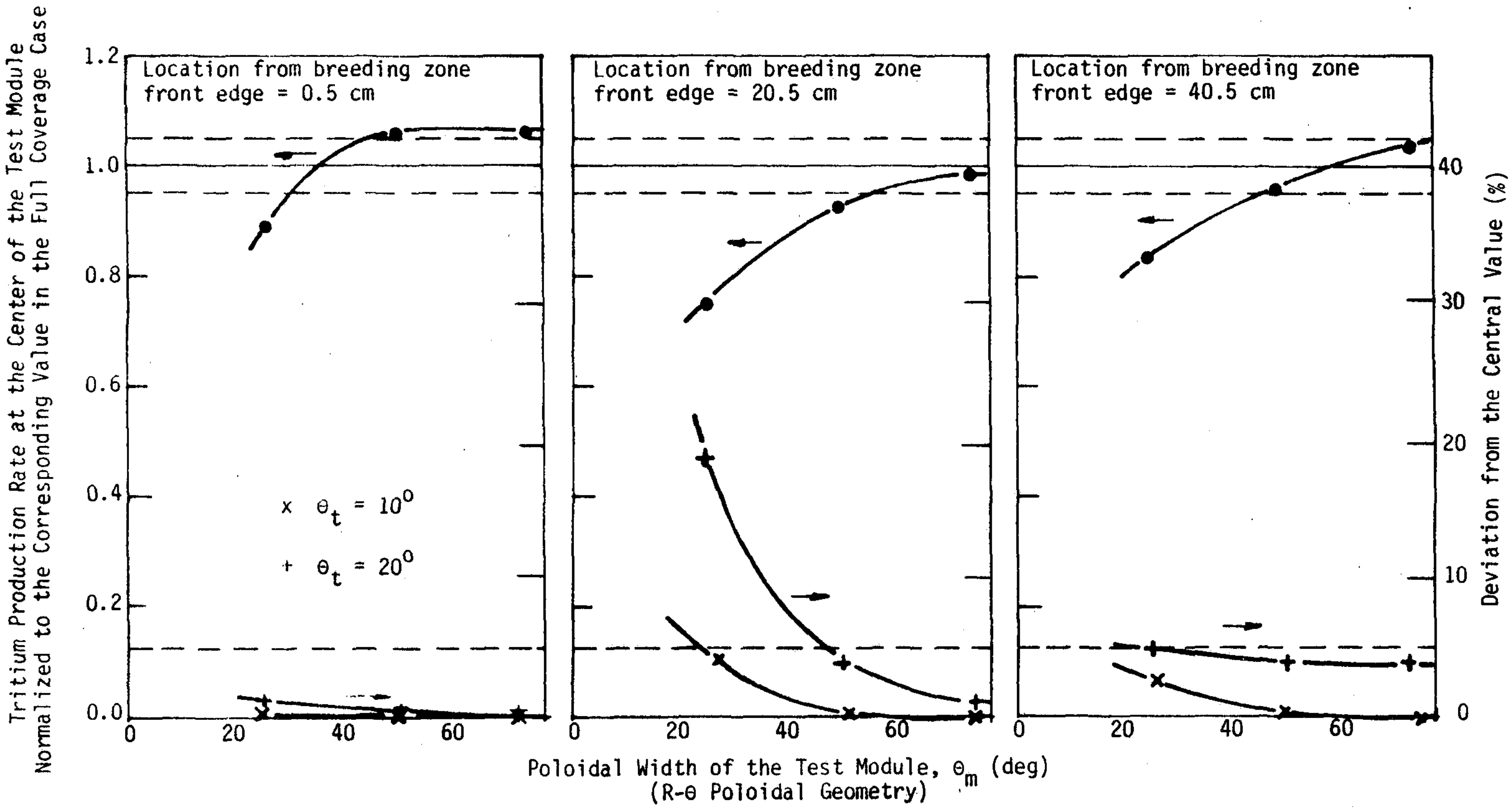


Figure 2.4.5-3 The total tritium production rate ($T_6 + T_7$) as a function of the test module width (Θ_m). Also shown is the deviation (%) from central values as a function of the test module width for two central zone widths, $\Theta_t = 10°$ and $20°$. (R-Θ geometry)

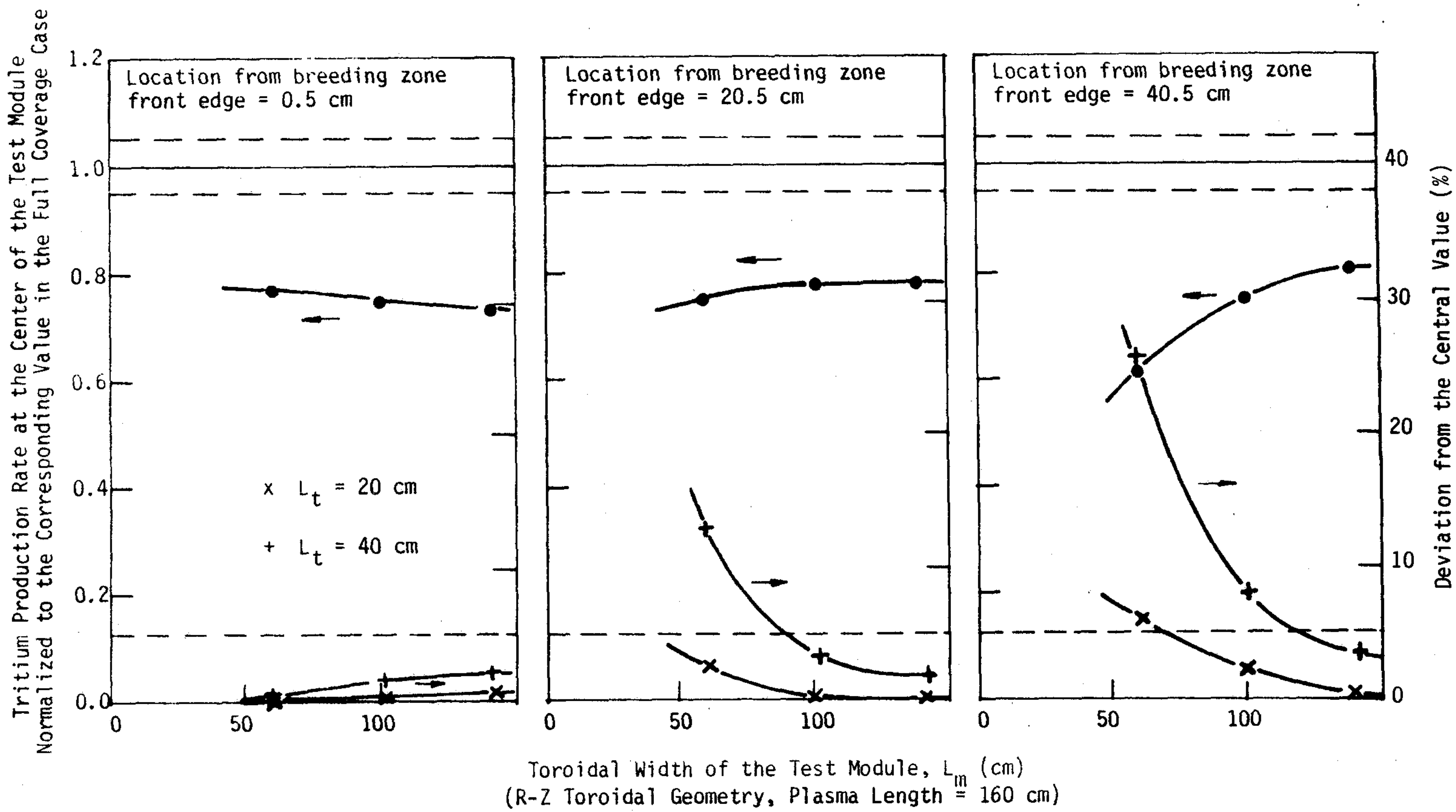


Figure 2.4.5-4 The total tritium production rate ($T_6 + T_7$) as a function of the test module width (L_m). Also shown is the deviation (%) from the central values as a function of the test module width for two central zone widths, L_t = 20 and 40 cm in the case of plasma length L_p = 160 cm.

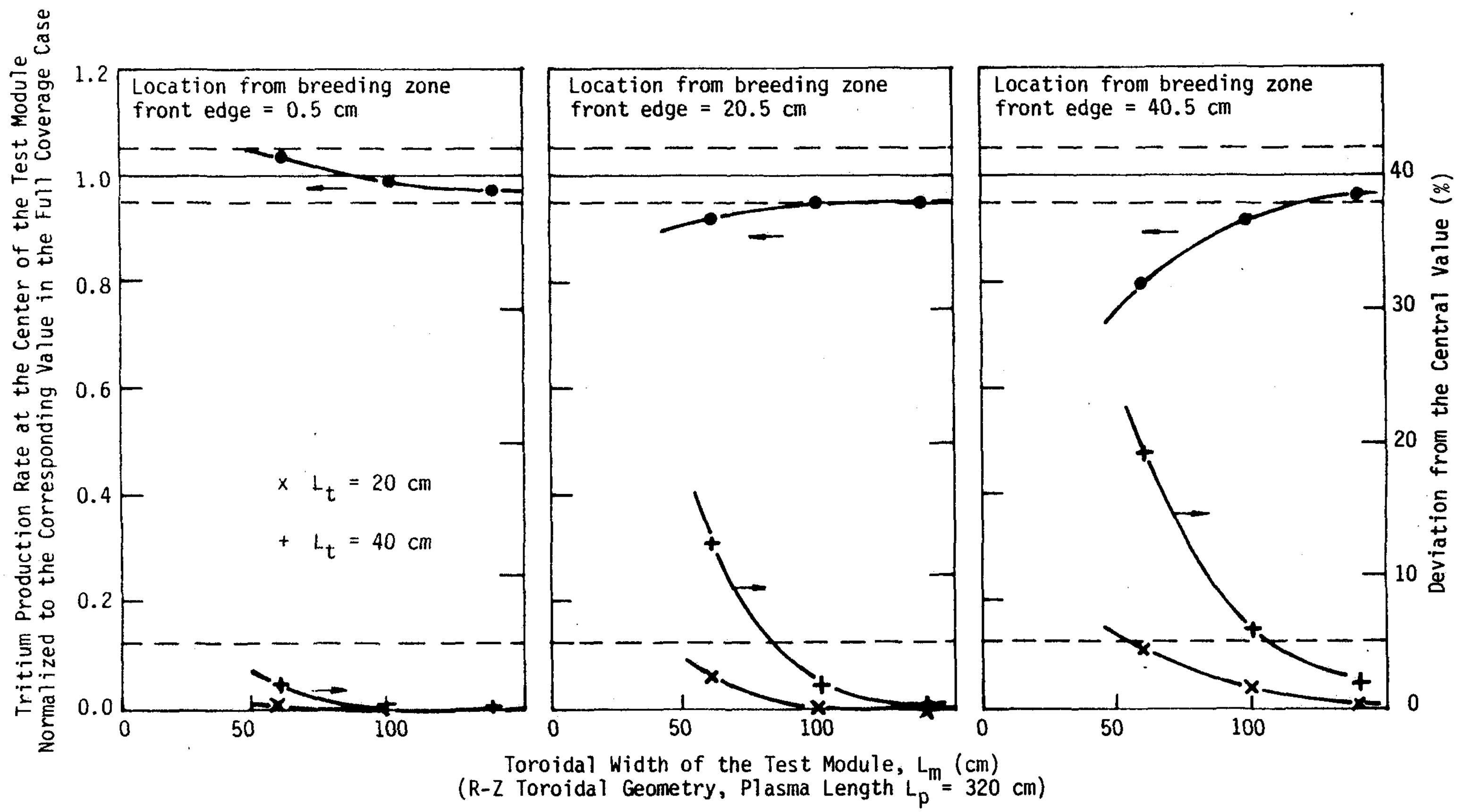


Figure 2.4.5-5 The total tritium production rate ($T_6 + T_7$) as a function of the test module width (L_m). Also shown is the deviation (%) from the central values as a function of the test module width for two central zone widths, L_t = 20 and 40 cm in the case of plasma length L_p = 320 cm.

deviation (total ~ 8.4%) if measurements were to be carried out within the central zone of L_t = 40 cm at that location.

There are several serious problems concerning the usefulness of TBR verification tests in a fusion test device with a test module that partially covers the plasma source. To illustrate some of these problems, Fig. 2.4.5-6 shows the integrated values of tritium production rate in various segments of the test module (characterized by the parameter t_B) and the overall tritium breeding rate as function of the test module width. Curves are shown for both the R-θ and the R-Z arrangements. The values shown in this figure are normalized to the corresponding values in a volume equivalent to the test module volume in the full coverage case. In all the cases shown, the total tritium breeding ratio in the test module is significantly smaller than the corresponding value in the full coverage case. The uncertainties involved in extrapolating the tritium breeding measurements in a test module to an achievable net tritium breeding ratio in a full-scale reactor are greater than presently estimated margins in the tritium breeding potential for candidate blanket concept.

The neutronics analysis leads to two particularly important conclusions. First, blanket neutronics measurements in a test module in any fusion facility, while useful, do not provide the level of accuracy necessary for neutronics verification, particularly resolving the issue of the achievable tritium breeding ratios. Thus, neutronics measurements do not by themselves provide strong justification for a fusion test facility, but such measurements are useful to perform if such a test facility is justified by the other reasons discussed elsewhere in this report. Second, the problem of demonstrating DT fuel sufficiency prior to constructing a full-scale reactor requires further detailed evaluations.

2.4.6 Test Matrix Considerations

The formulation of a detailed test matrix to help define a test plan that includes facility requirements, cost and schedule will take place primarily during the second year of FINESSE. A brief summary of only the trends in the required testing area and volumes is given here. The role that test matrices will play in the development of quantitative facility requirements is described in Chapter 9.

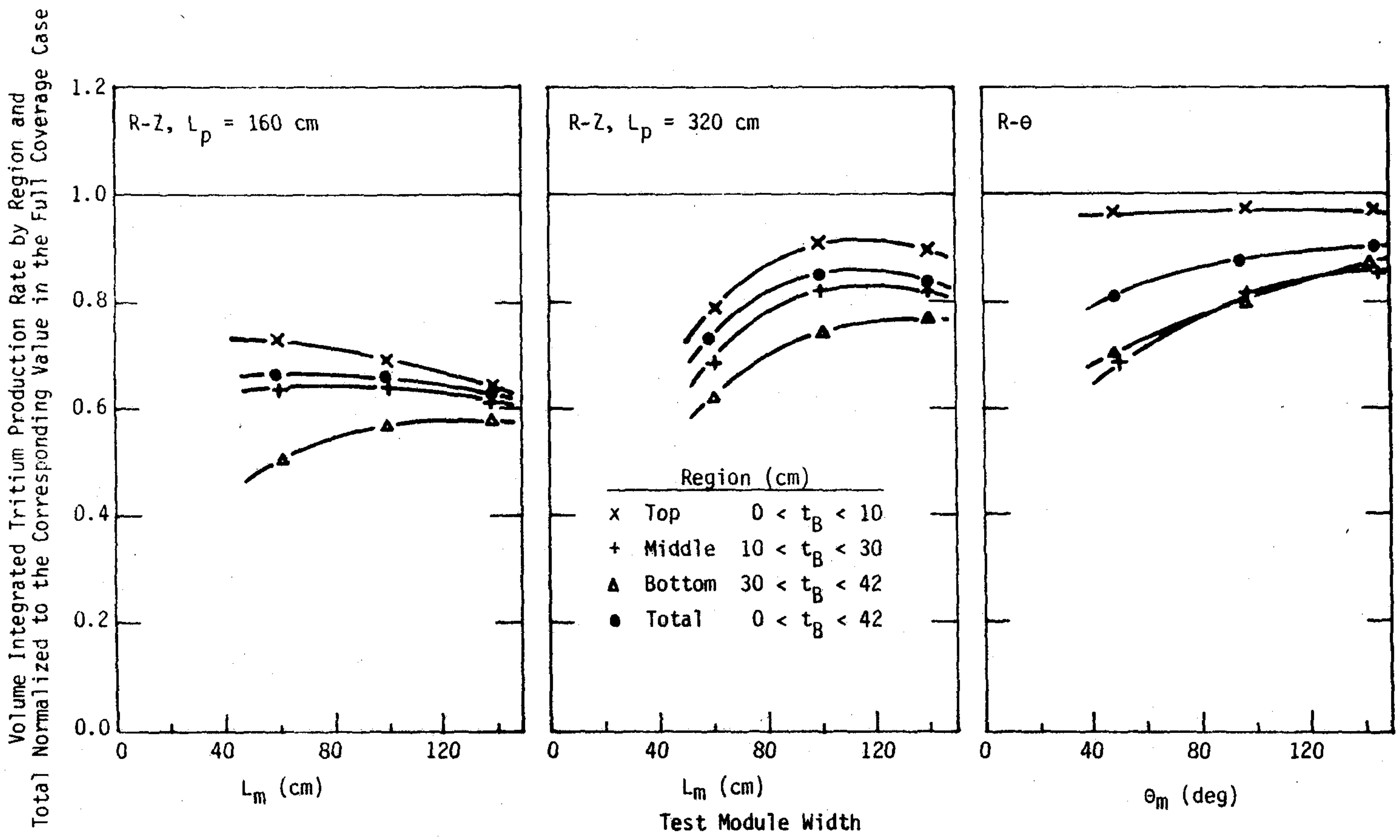


Figure 2.4.5-6 Total tritium production rate integrated over various spatial segments and the overall tritium breeding rate as a function of the test module width (L_m or Θ_m).

Test matrices in general are simply lists of test types or experiments along with information important to the test engineer and/or test facility designer. In FINESSE, the test matrix will perform a slightly different role since the "fusion facility" in which the testing is being considered has not yet been designed. For this study, the test matrix will be developed in parallel with the design of testing hardware and handling scenarios. Since the development of the test handling and operational considerations requires knowledge of the irradiation facility, this will be based on past testing experience from fission reactor, accelerator, and test-stand testing, along with a general understanding of typical mirror and tokamak facility features such as those described in Chapters 13 and 14. Figure 2.4.6-1 illustrates the manner in which test matrices will be utilized in FINESSE.

Test Matrix Trends

The first step in compiling the test matrix has been to estimate the overall irradiation testing area (first wall area) and volume from the data provided in Chapter 4 on testing needs. While this information is preliminary, it provides valuable insights into trends in the testing requirements and into the amount of irradiation space that will be required for fusion research and development. Results of an initial estimate of irradiation testing area and volume are listed in Table 2.4.6-1.

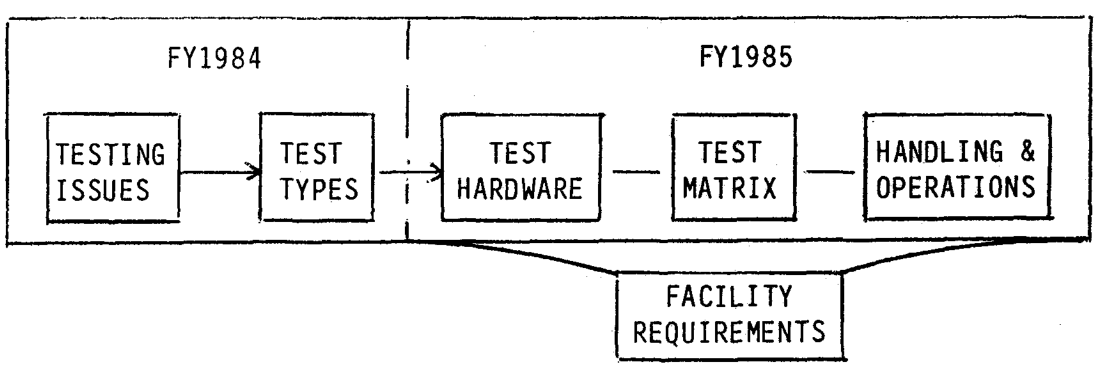


Figure 2.4.6-1 The role of the test matrix in defining fusion facility requirements.

Table 2.4.6-1 Preliminary Summary of Test Area (m^2) and Volume (m^3) Needs[a]

Test	Area	Volume
Blanket	(50-100)[b]	(40-60)[b]
Plasma interactive components	1-2	< 1
Shield	(10-20)[c]	(20-30)[c]
Tritium processing system	1-2	1-2
Magnets	< 1	< 1
Instrumentation and control	< 1	< 1
Balance of plant	—	—
Component interaction	—	—

[a]Table includes only those areas and volumes that have been quantitatively defined to date and do not include duplicates for reasons such as statistics.

[b]Not all these tests have to be performed simultaneously.

[c]Uses the same wall area as that of the blanket.

The area and volume ranges shown in Table 2.4.6-1 were based on the data listed in Chapter 4 along with judgement from breeder reactor and fusion materials irradiation testing experience. Table 2.4.6-1 represents the space that is required for the total, time-integrated testing program. The space requirements are, therefore, not those required in a given reactor at a given time but rather represent the overall space integrated over the test program duration. This is discussed in more detail in Chapter 9.

The conclusion drawn from Table 2.4.6-1, while preliminary, points to the need for a considerable amount of irradiation testing space for fusion research and development.

The utilization of non-fusion irradiation facilities will only satisfy ~ 10% of the area requirements and < 5% of the volume requirements. While these percentages are small in terms of total volume or area, they represent irradiation space that will be used in the near term to test large numbers of small specimens used in basic and single-effects tests.

The irradiation space discussed in this summary is preliminary and will change as the details of testing are developed. During the second year of FINESSE, each experiment type will be studied and appropriate conceptual design features and handling scenarios developed. The priorities of experiments and considerations of test schedule and requirements for duplication will be evaluated. Based upon the information from this effort, the required testing area and volume and other implications on the test facility, e.g., access, will be quantified.

2.5 Non-Fusion Facilities

2.5.1 Introduction

Non-fusion facilities can and should play an important role in fusion nuclear technology research and development. Many suitable facilities are available with a well established operational procedure and at a reasonable testing cost. The use of non-fusion facilities for single-effect and some multiple-interaction tests will provide a cost-effective means for narrowing materials and design concept options. In addition, the information from such non-fusion experiments will be valuable in reducing costs and risks of the more costly and complex integral tests. However, non-fusion testing alone cannot satisfy all the nuclear technology experimental needs. This section summarizes the results of evaluation of the capabilities and limitations of non-fusion test facilities: non-neutron test stands (Section 2.5.2), point neutron sources (Section 2.5.3), and fission reactors (Section 2.5.4).

2.5.2 Non-Neutron Test Stands

During its first year, the effort in FINESSE has focused primarily on neutron-producing non-fusion and fusion facilities. This emphasis is due to the large cost and complex technical and programmatic issues associated with neutron-producing facilities. However, the identified issues and testing needs clearly indicate that non-neutron test facilities can and should serve, now and in the immediate future, an important role in fusion nuclear technology research and development.

The role of non-neutron test stands is in the areas of basic property data, single-effect experiments, and some of the multiple-effect/multiple-interaction tests for which the neutron field is not important. Examples are mechanical properties and corrosion of unirradiated structural materials, sputtering of plasma-side material surfaces, and some liquid metal MHD experiments. The cost of non-neutron experiments and facilities is generally much lower than those involving neutrons. Therefore, information from non-neutron tests are is important for at least two reasons. First, they permit early scoping of some material and design options. Second, they permit better planning of more useful and less risky irradiation experiments.

A survey of the U.S. facilities indicates the availability of some test stands that are potentially useful for some fusion technology experiments. However, multiple-effect test stands appropriate for fusion are not, as might be expected, a forthcoming legacy from other technologies. Thus, there is a definite need for upgrades of existing non-neutron facilities and the construction of new ones. A more quantitative description of the non-neutron test stand needs requires careful examination of their costs as functions of capabilities in simulating multiple-interaction effects. An example of an important issue that must be addressed in this context is whether there are practical methods, other than using neutrons, for providing bulk heating in these experiments. While radiation damage is not critical for some types of experiments, the FINESSE results show that bulk heating is very important in most multiple-interaction experiments.

2.5.3 Point Neutron Sources

2.5.3.1 Introduction

Point neutron sources are attractive for irradiation testing because of their promise of easy access, simplicity, and relatively low cost. In reality, point sources often suffer from limited neutron intensity, inappropriate neutron spectra, and difficulties associated with operating very high technology machines. It should also be noted that as the level of neutron exposure is increased, the costs associated with utilization of point sources increase dramatically. The increased costs are associated primarily with shielding and remote handling.

The following discussion focuses on the various point neutron sources proposed over the recent past for testing fusion materials. As of this writing, no point source with sufficient flux for blanket or first wall experiments has been formally advanced. However, use of an advanced FMIT type machine capable of large scale testing will be discussed as an "upper limit" on point neutron source application.

2.5.3.2 Point Neutron Sources for Fusion Experiments

A number of point neutron sources have been suggested for fusion materials testing. Only the RTNS-II at Livermore is actually in operation. In the mid-70's the INS was pursued at LANL as a somewhat larger test volume than RTNS-II, but it was eventually supplanted by FMIT which had sufficient flux and testing volume to fulfill the materials small specimen testing needs for the U.S. fusion program. Recently, construction of FMIT was postponed indefinitely such that only the RTNS-II will be available for testing in the foreseeable future. It has been suggested that a large volume of relatively high neutron flux will be available at LAMPF in the near future. The potential for fusion testing in LAMPF has been clouded by concerns over the appropriateness of the neutron spectra to blanket and first wall experimental objectives.

Rotating Target Neutron Source (RTNS) at LLNL

RTNS-II generates neutrons by bombarding a tritium-containing target with 400 KeV deuterons. Neutrons are produced by the DT fusion reaction which produces a nearly monoenergetic 14 MeV neutron spectrum. The tritium is held in a target in the form of zirconium hydride which is coated on a water cooled copper disk. The deuteron beam is focused as a 1 cm^2 spot on the target. Rotation is used to reduce heat loads to levels which allow the tritium to be retained on the target. A maximum flux level of about 5×10^{12} n/cm^2-s is attained in a test volume of less than 0.1 cm^3. While this is adequate for low fluence testing of miniature material specimens, RTNS-II is much too small for nuclear experimentation supporting first wall and blanket multiple effects tests.

Intense Neutron Source (INS) at LANL

The intense neutron source (INS) was intended to generate neutrons by injecting a 1.1 amp 300 KeV tritium beam into a gas target consisting of a supersonic deuterium jet. The deuterium density in the jet provides for sufficient (d,t) reaction to yield a neutron source of about 10^{15} n/s. The limiting feature of this arrangement is the ability of the gas target to remove the heat generated by the beam. Since the neutron source is more or less cylindrical in shape, testing is conducted in an annulus around the target. A peak flux of 1×10^{14} n/cm^2-s 14 MeV neutrons was predicted for a few tenths of a centimeter next to the target. Since flux drops off as $1/r^2$, the volume of test space at 10^{13} n/cm^2-s or better was expected to be about 1 cm^3. Again as with RTNS-II, this space would have been useful for the limited material testing but is not large enough for multiple effects experimentation.

LAMPF A-6 Target Station at LANL

The Los Alamos Meson Physics Facility (LAMPF) is an 800 MeV 1 mA proton accelerator. The A-6 target station at LAMPF offers the potential for testing at relatively low neutron flux with test volume approaching levels practical for significant, albeit small, "volume-type" experimentation. At the A-6 target position, spallation neutrons from the LAMPF beam stop provides a flux of about 1×10^{13} n/cm^2-s at a volume of 0.02 m^3. The neutron spectra is much different than for fusion with most having energies below 1 MeV but a significant portion having energies reaching almost to 10^3 MeV. The combination of flux, spectrum and duty factor are sufficient to provide about a displacement level of 1 dpa/year in copper. While of interest to materials science, the flux/spectrum/duty factor combination is probably too small for multiple effects testing for high exposure materials testing.

Fusion Materials Irradiation Test (FMIT) Facility at HEDL

The Fusion Materials Irradiation Test Facility was specifically designed to meet the needs of the U.S. fusion program for a "fusion like" materials neutron irradiation facility. In FMIT, neutrons would be produced by bombarding a flowing lithium target with 35 MeV deuterons. The (d,Li) reaction would provide a broad neutron energy spectra averaging around 14 MeV. Damage calculations indicated that the spectra from 35 MeV deuterons on lithium was adequate for fusion materials evaluation.

FMIT would provide 10 cm^2 of test space with a neutron flux of 1×10^{15} n/cm^2-s or greater. Approximately 500 cm^3 would be available at 1×10^{14} n/cm^2-s or greater. While peak neutron flux levels and volume were expected to be excellent for small specimen testing, the steep flux gradients and source characteristics are inappropriate for multiple effects experiments. It should be noted that FMIT construction has been postponed pending the outcome of negotiations with the EEC and Japan concerning joint sponsorship.

2.5.3.3 Point Source Potential

In an effort to explore the potential of point sources for multiple effects testing, a device for producing a fusion-like neutron environment in a 23 x 23 x 23 cm^3 cube was scoped. The approximately 23-cm-side cube was assumed to be the smallest size of interest to multiple-effect experiments; however, depending on the results of the FINESSE study, a different lower limit may emerge.

The approach taken was to scale-up the FMIT accelerator to the maximum possible using relatively small extrapolations of existing technology. The result yielded a neutron flux of about 5×10^{13} n/cm^2-s at the front face of the 23-cm cube. Gradients from front to back of the cube can be adjusted over a relatively wide range. While it cannot be stated that gradients will match the precise requirements for "act-alike" tests of reduced size, there is considerably more flexibility than in other neutron-generating machines.

For lack of a better designation, this source is called "Super-FMIT." In Super-FMIT, 14 MeV average energy neutrons are produced by interaction of a 70 MeV beam of D_2^+ ions with a flowing lithium target. The accelerator consists of a radio frequency quadrupole of the zero-mode type that supports four accelerating channels, and an Alvarez Linac in which the drift tubes accommodate four beams. The multiple beam lines are contained in a single RF tank. Using the multiple beams and D_2^+ instead of D^+ as in FMIT, a 1 amp beam can be obtained. The beam interacts with a large flowing lithium target which dissipates less energy per unit area than FMIT and therefore should be easier to engineer and operate. Separate aiming of the beams gives flexibility that may be used to achieve the desired flux shaping and gradients.

2.3.5.4 Conclusion

Table 2.5.3-1 summarizes the flux and testing volume capabilities of the point neutron sources surveyed. The primary point source for fusion materials testing in use in the United States is the RTNS-II. Other more powerful sources are needed for materials testing but have been canceled or deferred. The small test space of point sources considered to date precludes their use in multiple-effect and integrated testing; however, a scale-up of the FMIT concept could produce a point source of interest to some fusion nuclear technology experiments.

Table 2.5.3-1 Point Neutron Sources for Fusion Material Irradiation Testing

Facility		Peak Flux	Testing Volume	Applicable to Multiple Effects
RTNS-II	In-Use for Materials Testing	$\sim 5 \times 10^{12}$ n/cm^2-s	~ 0.1 cm^3	No
INS	Conceptual Design Completed - Project Terminated	1×10^{14} n/cm^2-s	~ 1 cm^3	No
LAMPF A-6	Operational	1×10^{13} n/cm^2-s	~ 0.02 m^3	No
FMIT	Design Completed Project Deferred	1×10^{15} n/cm^2-s	~ 10 cm^3	No
SUPER-FMIT	Scoping Study	5×10^{13} n/cm^2-s	~ 0.016 m^3	Yes

2.5.4 Utilization of Fission Reactors

One option for performing the engineering experiments needed for fusion development is to employ fission reactors. For the purposes of FINESSE, it is necessary to determine how large a role fission testing can and should play in the overall fusion nuclear technology research and development program. Although the usefulness of fission testing depends to some extent on the R&D scenario chosen (and vice versa), it is important to examine the technical, and programmatic constraints on fission testing, in order to clarify its advantages and disadvantages. In this study, only "volume-type" experiments (as opposed to small-specimen materials testing) has been considered, and interactive-effect tests have been emphasized.

When considering fusion experiments in fission reactors, eight primary issues are usually cited as important concerns or limitations; these are listed in Table 2.5.4-1. The technical and programmatic aspects of fission testing can be conveniently discussed by considering these issues individually.

Radiation damage simulation is a concern for testing in fission reactors because of the difference between the fusion and fission reactor spectra. Since radiation damage in breeder materials results from lithium burnup, this is discussed in the following paragraph. Fusion radiation damage in structure results mainly from atomic displacements and helium production. Although these physical mechanisms still occur in the fission spectrum, they occur at lower overall rates (for equivalent power densities) and in different relative proportions (the helium to displacement ratio is lower in the fission case) than at a fusion first wall. These results leave uncertainty that materials performance in the fission spectrum can simulate operation in the fusion environment. Although the quality of the simulation may improve for locations deeper in the blanket, radiation damage is of greater concern at the first wall. There are a number of techniques which have been used to artificially increase the helium production in the fission spectrum. One method is to utilize the nuclear characteristics of the normal alloy constituents (perhaps isotopically tailored), for example, by irradiating stainless steel whose nickel content has a large helium production at low neutron energy in thermal fission reactors. Another approach is to add small quantities of ^{58}Ni, B, or Li; although this approach can improve the simulation of first wall damage, it

Table 2.5.4-1. Key Issues for Utilization of Fission Reactors

1. Radiation Damage	a. Types and Rates
2. Power Density	a. Magnitude b. Spatial Profile
3. Lithium Burnup Rate	a. Magnitude b. Spatial Profile
4. Test Volume	a. Size b. Total of Existing Test Locations
5. Non-Nuclear Conditions	a. Magnetic Field b. Surface Heat c. Particle Flux d. Mechanical Forces
6. Reactivity Considerations	
7. Availability for Testing	
8. Cost	

can introduce uncertainties in the material performance due to the effects of the additional elements. Overall, fission testing, therefore, is suited more for beginning-of-life (BOL) testing where radiation damage in the structure is not a fundamental concern. Nevertheless, the capability of fission testing to provide various other important test conditions simultaneously with some materials damage is unique and potentially useful.

Power density and lithium burnup are closely related in fission tests, and are typically felt to be its most outstanding capabilities. No approach other than neutron and gamma heating can provide bulk heating to virtually all materials simultaneously, a capability required in complex engineering experiments. In addition, the capability for simultaneous in-situ tritium production and bulk heating is vital for experiments on solid breeder blanket concepts. Calculations have shown that excellent simulation of power density and tritium production profiles is possible in fission tests with the use of spectrum tailoring techniques. The main concern, however, is in the overall magnitude of the power density and lithium burnup rate, or conversely, in the source flux required for prototypical operation. The calculated fluxes required at the surface of the test assembly for 1 MW/m^2 equivalent power density at the front of the blanket are summarized in Table 2.5.4-2 for both the in-core (submodule), and core-side (slab) test concepts examined. These rather high flux requirements limit the choice of test location, especially when flux depression effects are taken into account.

Table 2.5.4-2. Flux Required at Face of Test Assembly to Simulate Bulk Heating at Front of Fusion Blanket at 1 MW/m^2

Test Type	Blanket Concept	Neutron Filter	Flux[a] ($cm^{-2} \cdot s^{-1}$)
Slab[b]	Li_2O/He/HT-9	—	1.4×10^{15}
		Cd	1.5×10^{15}
Slab	Li/Li/V	—	9.4×10^{14}
		Cd	9.8×10^{14}
Slab	$LiAlO_2$/Be/H_2O/PCA	—	6.2×10^{14}
Slab		Cd	8.7×10^{14}
Submodule[c]	Li_2O/He/HT-9	—	5.1×10^{14}
Submodule	$LiAlO_2$/Be/H_2O/PCA	—	2.1×10^{14}

[a]Water-moderated, plate-fueled test reactor assumed.

[b]Core-side.

[c]In-core.

The total test volume which is available and potentially useful for fusion testing is significant in view of the number and sizes of tests which may be of interest. In addition, it is important to have locations not just of sufficient size, but also at sufficient flux. Summaries of the number of test locations in U.S. and U.S.-plus-foreign reactors are shown in Tables 2.5.4-3 and 2.5.4-4. The numerical entries give the number of locations which could be used for a test, given a test maximum dimension and total flux requirement. In-core experiments will probably be at least 7.5 cm in diameter, considering typical containment requirements, and will require a flux of at least 10^{14} $cm^{-2} \cdot s^{-1}$. A relatively large number (~ 50) of test locations are available at the lower end of the range of requirements for both flux and volume. Unfortunately, no locations exist at the higher requirements for flux and volume. For example, there are no locations with test size > 15 cm. For slab module tests, no locations with an adequate flux could be produced without some modifications to reactor facilities. For instance, an acceptable slab test location could probably be produced at ORR by modifications which would increase the reactor power. Other reactors could perhaps provide slab test locations by conversion of thermal columns.

Table 2.5.4-3 Number of Existing Acceptable In-Core Test Locations in U.S. (U.S. and Foreign) Reactors

Minimum Required Flux ($n/cm^2 \cdot sec$)	Test Assembly Maximum Dimension (cm)				
	5	7.5	10	12.5	15
5×10^{12}	180 (315)	119 (168)	33 (79)	16 (45)	2 (27)
5×10^{13}	167 (292)	106 (145)	30 (66)	15 (44)	1 (26)
5×10^{14}	49 (69)	13 (30)	13 (30)	10 (27)	0 (16)
5×10^{15}	40 (40)	4 (4)	4 (4)	1 (1)	0 (0)

Table 2.5.4-4 Numbers of Existing Acceptable Slab Test Locations in U.S. (U.S. and Foreign) Reactors

Minimum Required Flux ($n/cm^2 \cdot sec$)	Test Assembly Maximum Dimension (cm)				
	25	50	75	100	150
5×10^{13}	7 (11)	1 (4)	0 (2)	0 (1)	0 (1)

It is desirable to have the capability of including non-nuclear conditions such as mechanical forces, surface heating, magnetic field, or particle flux, in fission tests. This is relatively straightforward in the case of mechanical forces, which can be produced by externally-loaded gas cylinders. Although there is no fundamental difficulty with simulating surface heating by using electrical resistance heaters, no test concept with this feature has yet been developed, and the issues of associated volume increase and of interfaces with the test assembly have not been addressed. There will be difficulties in incorporating magnetic fields, due to the large magnets required for high fields and the possible effects of stray field on reactor safety and operation. Finally, no acceptable method of generating particle fluxes at prototypical levels in a fission test has yet been identified, although techniques have been proposed which would produce particle fluxes of lower magnitude.

The effect of a fusion blanket test assembly, which would be a strong neutron absorber, on the reactivity balance of the fission test reactor is also a concern for fission testing. For in-core tests, the negative reactivity effect has been found to be somewhat high, but acceptable. The test assemblies were worth from two to three average control rods for the type of reactor considered; in each case evaluated, however, a critical reactor core configuration without the test assembly was used as the base case, and the larger test assembly required a larger core to remain critical. For the slab module tests which were examined, the effect of the reactor core thickness on the net reactivity effect was found to be large (Fig. 2.5.4-1). As a point of reference for Fig. 2.5.4-1, an average control rod is worth approximately +2.50 $. This implies that small reactor cores (less than 30 cm thick) will have great difficulty in accommodating such tests. Large cores (greater than 50 cm thick) can certainly accommodate them, and medium cores (30-50 cm thick) may require some modifications.

Whether or not reactors will be available for testing when needed is also an issue which can affect program strategy. Most facilities which were contacted during the study will be in operation for the indefinite future; exceptions were EBR-II (with a projected life of 10 more years) and PBF (with a projected life of 1 more year). All indicated that test spaces would be available, given proper programmatic priority. In view of the plans for

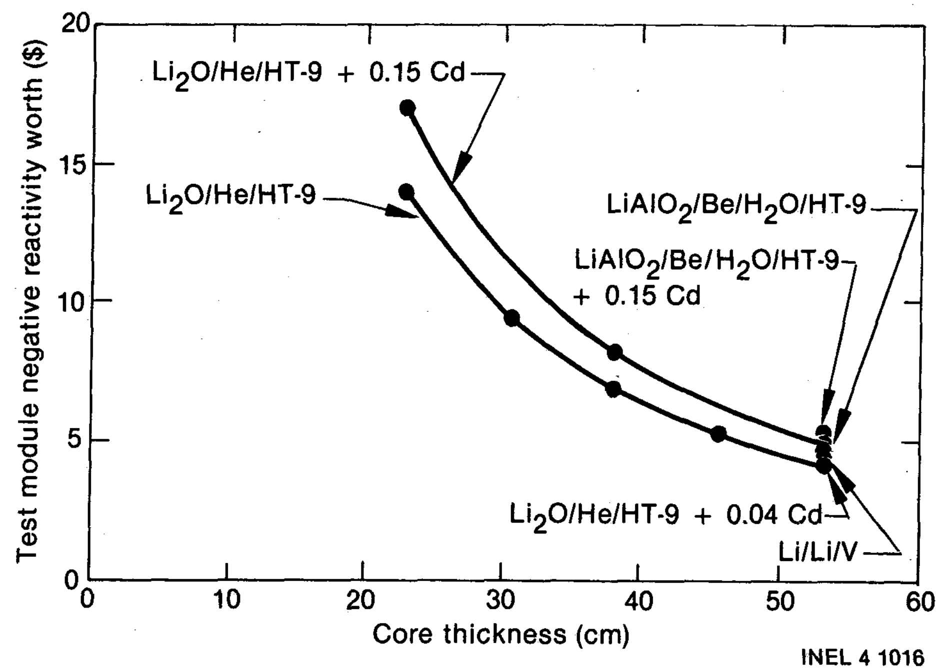


Figure 2.5.4-1 Slab test module negative reactivity worth as a function of core thickness.

EBR-II and PBF, as well as the recent mothballing of ETR, there appears to be a slow but consistent downward trend in the availability of fission test reactors. Presumably this trend could be halted or reversed if a need for additional testing were apparent.

Fission testing will be extremely useful for near-term fusion experiments. In particular, it is well-suited for conducting many multiple-effects tests, but not for complete act-alike performance. Its primary role in engineering testing will probably be in submodule-scale tests, since a number of acceptable test locations exist, and since the simulation requirements are somewhat relaxed for tests of this type. There will be some role for full-module slab tests in BOL performance evaluations and to allow early identification of some radiation-related synergisms. In general, it appears that fission testing will be somewhat more useful for solid breeder blankets than for liquid metal blankets. This is because the most critical issues for solid-breeder concepts (heat transfer and tritium release) match the capabilities of fission testing (bulk heating and in-situ tritium production) better than the most critical issues for liquid metal blankets (MHD and corrosion).

Fission testing is limited in three main areas. First, it is difficult to include all of the non-nuclear conditions which may be of interest. Second, the difference in spectrum between fusion and fission leads to difficulty in simulating structural radiation damage and leaves doubts concerning radiation-related synergisms. Finally, fission testing is currently limited in the total number of acceptable test locations, particularly slab test locations. These limitations apply primarily to integrated testing, and do not seriously reduce the usefulness of fission testing for many multiple-effects tests.

Overall, fission testing can and should be an integral, useful part of the fusion R&D program. Although it cannot completely replace or eliminate the need for fusion testing (except for extremely high-risk development scenarios), it can address many critical testing needs to various degrees. The principal advantages will be timeliness (it is available now) and cost-effectiveness (no new facility construction required). In the final analysis, each fission experiment or fusion development scenario considered must be evaluated on a cost/risk/benefit basis; in this context, fission testing is less costly and lower risk than fusion testing, but also is of less benefit. However, the attractiveness of acquiring considerable data, even though imperfect, on critical fusion engineering issues by testing early and in existing facilities should not be overlooked.

2.6 Fusion Facilities for Nuclear Experiments

2.6.1 Introduction

The identified testing needs for fusion nuclear technology include a number of critical multiple interaction and integrated experiments. These particular experiments have the following characteristics: (1) They require simulating many of the fusion environmental conditions, particularly the neutrons. (2) The size of a typical experiment is large, typically on the order of 1 m^3. (3) The total testing volume requirements for the important needs is large, in the range of 10-20 m^3.

Multiple interaction experiments for which the neutron field is not critical may be performed in non-neutron test stands, even if they require a large size. Although suitable test stands are not readily available, the construction of new ones at a reasonable cost may be justified. One particular problem here that must be considered is that many of these multiple interaction experiments require bulk heating. Although neutrons may not be critical for simulating radiation effects, they may be the only practical source of bulk heating.

Neutrons are needed to simulate radiation effects, to provide bulk heating, and to induce specific nuclear reactions, e.g., Li(n,t). The only presently available source of neutrons for a significant experimental volume is fission reactors. As discussed in the previous section, fission reactors, while useful for some multiple interaction tests, cannot satisfy critical needs for other multiple interaction and integrated tests.

Thus, fusion nuclear technology R&D mandates careful evaluation of fusion devices as test facilities. Section 2.6.2 is a summary of a technical investigation of the potential of tandem mirrors as a Fusion Engineering Research Facility (FERF). Section 2.6.3 presents a summary of an attempt to identify a low-cost tokamak option that can satisfy the nuclear testing requirements. In both the mirror and tokamak evaluations, no physics testing requirements were imposed on the plasma operating mode. For the tokamak option, key differences in the costs and risks between large, high power fusion devices that combine the physics and technology missions and smaller, lower fusion power devices that are dedicated to nuclear testing are expected. A primary difference relating to the impact of additional tritium breeding on the overall availability of a large tokamak is discussed in Section 2.6.4.

2.6.2 Tandem Mirror Test Facilities

2.6.2.1 Test Facility Options

Tandem mirrors offer an excellent unique capability for carrying out nuclear experiments and for demonstrating the operation of nuclear technologies. This capability derives from an ability to produce high fusion power densities by injecting high energy D-T neutral beams into a magnetic mirror test cell which is inserted within the central cell of a tandem mirror. The volume of plasma can be kept arbitrarily small by selecting the length of the cell, with the result that low cost test areas can be designed. The physics of ion confinement in the mirror cell is essentially the physics of single cell mirrors, for which there is a long experimental history. That history shows that well understood classical predictions of ion-ion scattering and ion-electron drag account for losses from the cell. Thus, it appears possible to design the test cell with considerable confidence in the essential physics.

It was recognized more than ten years ago that single cell mirrors might be attractive as nuclear research test facilities. A device, called FERF, was designed[11] for that purpose in 1974. With the invention of the tandem mirror, the idea was adapted to the evolved confinement configuration by beam driving the entire center cell of a tandem mirror as a test cell. Such a facility would operate in the "Kelley mode" with the large majority of ions being magnetically trapped and the minority fraction being electrostatically confined in the tandem mirror end plugs.

To this end, Fowler and Logan[12] proposed a Tandem Mirror Technology Demonstration Facility (TDF) whose primary objective would be to demonstrate the steady-state operation of fusion technologies (e.g., rf heating systems, superconducting magnets, tritium systems) in the nuclear environment and to serve as an integrated development/test facility for tritium breeding blanket modules. TDF would use beam-driven axicell plugs and quadrupole anchors. The end plugs of TDF would be stream-stabilized by the plasma outflow from the central cell, while the anchors were to provide MHD stability through electron conduction between the high density plug and the low density anchors. With 68 MW of 80 kV beams, Fowler and Logan predicted that 15 MW of fusion power would be produced in the 8 m central cell. Because the energetic ion lifetime was assumed to be only one collision time, the physics basis for TDF was believed to be conservative enough that it could be largely verified from operation of the existing TMX facility at LLNL.

A more detailed TDF design was developed by LLNL and other fusion organizations during 1982-1983.[13] This design, shown in Fig. 2.6.2-1, would have a total capital cost in the range of $1-1.5 billion. It would provide two blanket test module ports and a substantial area for neutron damage testing to fluences of 5-10 MW-yr/m^2. To address the problem of trapped particle modes in tandem mirror reactors, the TDF electrostatic plugs were moved outwards from an axisymmetric "axicell" to the quadrupole cell. At the same time, a simple choke coil would replace the axicell and a thermal barrier would be formed in the anchor. The most recent TDF configuration is similar to the MFTF-B configuration.

Two different options for achieving microstability in TDF were considered. With the first, stream stabilization,[14] low electron temperature and confinement times lead to a wall loading of 1.4 MW/m^2 from 20 MW of fusion power (65 MW of 80 kV beams). The second option,[15] based on stability by sloshing ions with the stream gas removed, has potential to provide performance consistent with the engineering parameters shown in Table 2.6.2-1 and the magnetic field, electrostatic potential, and plasma density profiles shown in Fig. 2.6.2-2. In this "beam-fueled" mode, the electron temperature and lifetime doubles, fusion power increases to 35 MW (with 51 MW of 55 kW beams), and the wall loading increases to 2.1 MW/m^2.

Approaches to technology test facilities less expensive than TDF have been proposed recently as novel upgrades to MFTF-B. One such approach, the MFTF-α+T upgrade of MFTF-B,[16] would combine plasma confinement objectives with nuclear experiment and test objectives. The ~ $450 million, 11 MW$_f$ MFTF-α+T upgrade would be a DT-burning machine with significant α-power deposition in the central cell. It would incorporate many recent ideas which are expected to result in tandem mirror concept improvement (i.e., MARS end plugs with additional anchoring, drift pumping, halo pumping) and would also feature several next-step technologies (e.g., 200 keV negative ion beams, 18 T choke coils, direct conversion). A second phase of MFTF-α+T operation would focus upon technology development and low fluence integrated nuclear testing with one or more (at higher cost) beam driven test cell inserts. During the

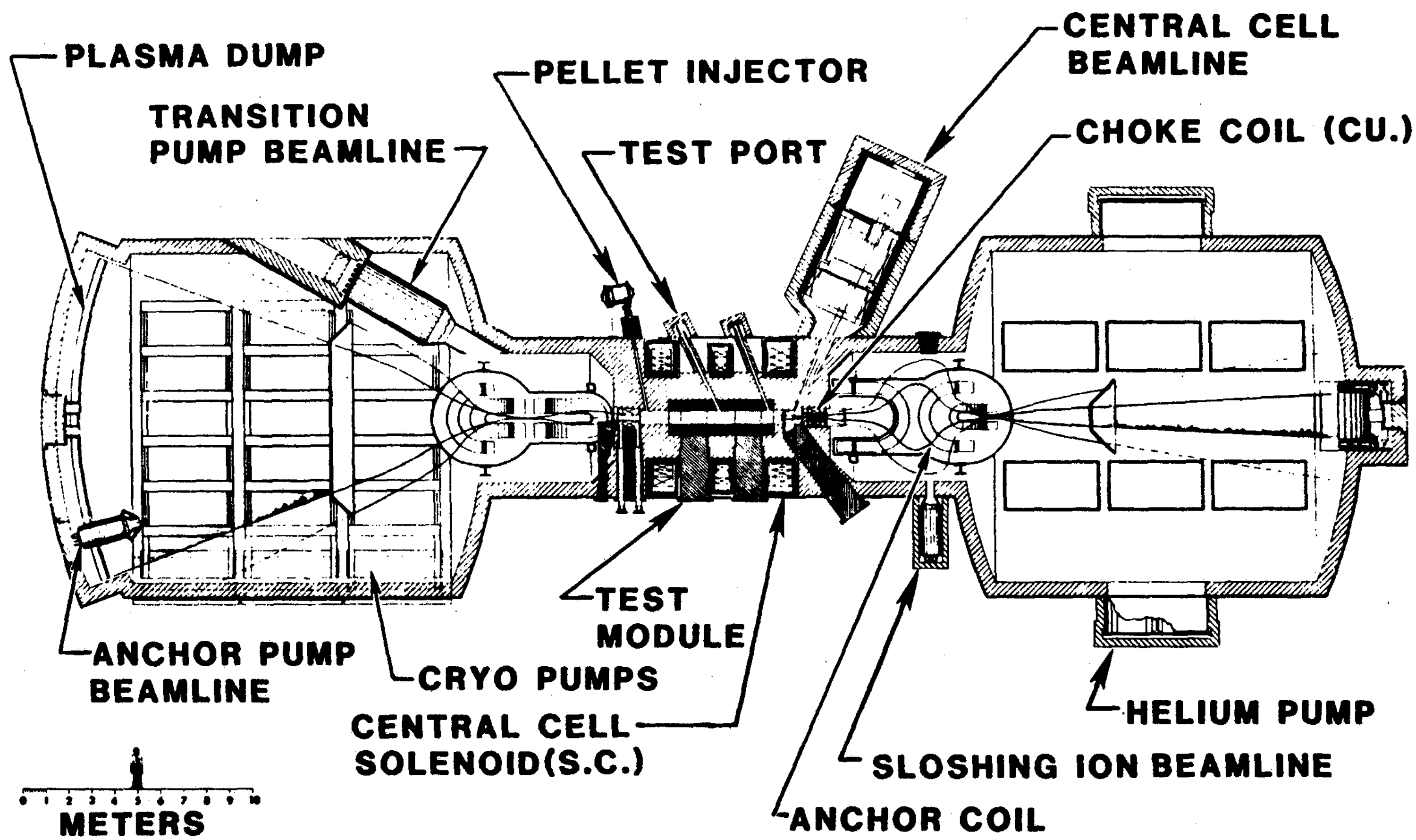


Figure 2.6.2-1 Cross-section of TDF machine. (Some components are shown rotated out of plane for clarity and only one of the eight center cell beamlines is shown.)

Table 2.6.2-1 TDF Engineering Parameters

Parameter	Units	Quantity
<u>OVERALL MACHINE</u>		
Full power run length	h	> 100
Availability	% (life average)	30
Design life	Full power years	5.4
Total capital cost	$ million (1982 dollars)	~ 1000
<u>PLASMA</u>		
Length (central cell)	m	8.0
Radius	m	0.15
Peak beta	%	40
<u>TEST ZONE</u>		
First wall radius	m	0.30
Neutron wall load	MW/m^2	2.1
Test module area	m^2	3.6
Total area	m^2	~ 8
Heat load	W/cm^2 (average)	50
Fusion power	MW	36
<u>TRITIUM</u>		
Consumption rate	g/h	0.23
Inventory	kg	~ 0.3
<u>VACUUM</u>		
Base pressure	Torr	5×10^{-6}
Total pump speed	L/s	1.3×10^{-8}
<u>MAGNETS</u>		
Superconducting:		
Material		Nb-Ti
Peak cond. field	T	8
Peak center field	T	4.5
Resistive:		
Material		Cu alloy
Peak center field	T	15
<u>NEUTRAL BEAMS</u>		
Mode		Continuous
Energy (max)	keV	80
Power:		
Central cell	MW	51
Pumping	MW	7.0
Sloshing	MW	0.8
<u>RADIO FREQUENCY SOURCES</u>		
Type		ECRH
Frequency	GHz	35/60
Power	MW	1.0/0.6

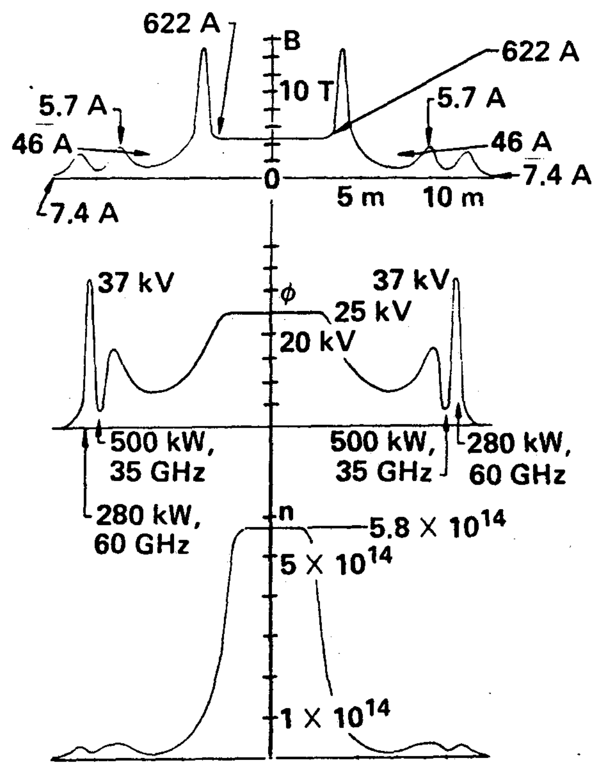


Figure 2.6.2-2 TDF magnetic field, electrostatic potential, and plasma density profiles.

nuclear test phase, the machine would be operated in a low confinement mode, but would provide a relative high (~ 2 MW/m^2) neutron wall loading at an ~ 10% availability for test periods up to ~ 100 hours. As shown in Fig. 2.6.2-3, the $450 million MFTF-α+T facility would be fully shielded and remotely maintained. Unlike in TDF, where the insert would be the entire central cell, in MFTF-α+T the insert would be embedded in the central cell.

Another design study, TASKA-M, was recently completed by KfK in Karlsruhe, West Germany, in conjunction with the University of Wisconsin.[17] The TASKA-M mission was to identify "the smallest and least costly tandem mirror test facility possible, which still retains a considerable degree of reactor relevance." The TASKA-M design was based upon near-term physics assumptions and mid-1980's level of technological capabilities. TASKA-M would produce 6.8 MW of fusion power to provide a peak neutron fluence of ~ 80 dpa in a 0.17 m^3 test area. The facility would also accommodate two small blanket component test modules; however, the ~ 1 MW/m^2 neutron wall loading would vary greatly

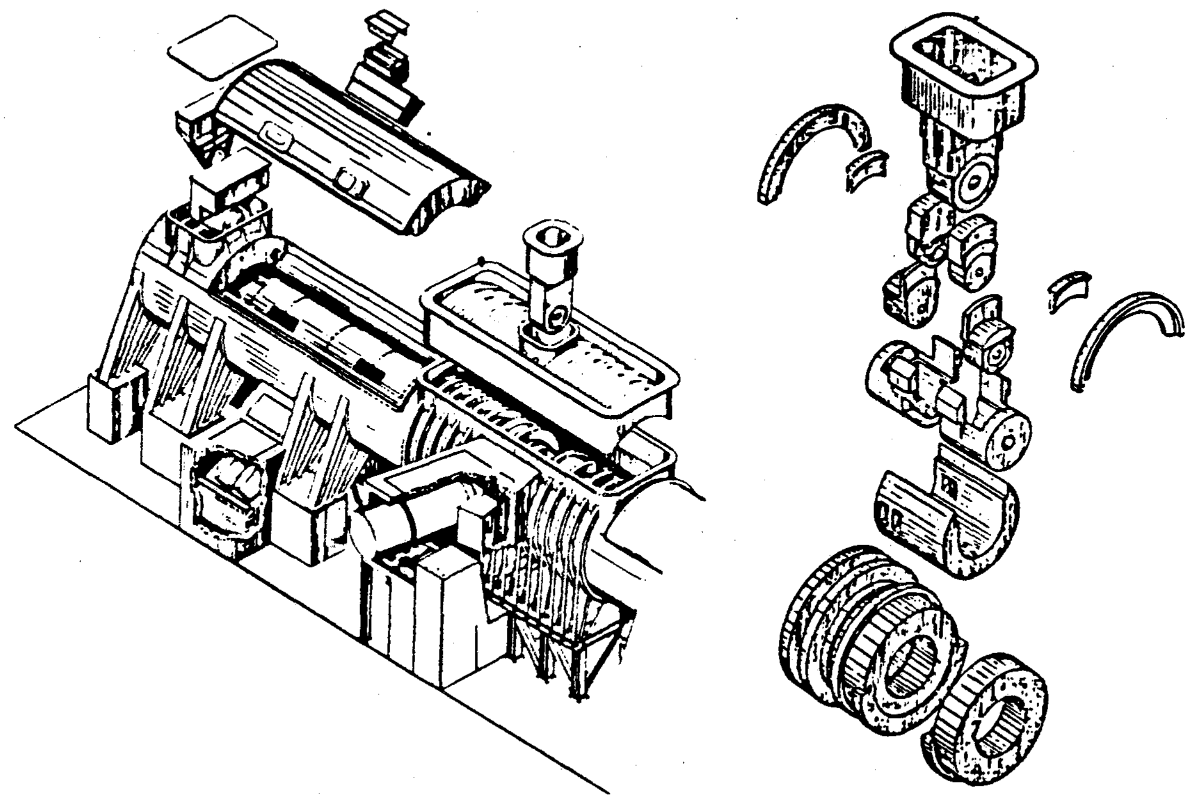


Figure 2.6.2-3 Test station assembly and access ports for the major components of MFTF-α+T.

in the axial direction. Nevertheless, TASKA-M, with a projected direct cost of < $400 million (1983 dollars), serves as an excellent example in defining the bounds for low power but intermediate fluence tandem mirror test facilities.

With TDF, MFTF-α+T and TASKA-M, the design of test facilities has evolved to show the merits of using tandem mirrors with driven test cells to address technology issues. However, further steps in the design of test facilities will surely be taken. For example, the use of octupole end plugs has already been proposed for MHD stabilization in tandem mirrors. Successful use of these octupole designs would permit much shorter end plugs and, therefore, much shorter central cells for a given performance level. With octupole plugs, it is projected that central cell ignition conditions could be achieved in a machine length comparable to the present MFTF-B. Such a device would have wall loadings of ~ 1 MW/m^2, but if operated in a lower confinement mode, with a beam driven test cell inserted, the wall loading could be raised to 2-3 MW/m^2.

2.6.2.2 Resolving the FINESSE Nuclear Issues in a Tandem Mirror Nuclear Test Facility

A preliminary assessment of the ability of an MFTF-α+T or TDF class FERF to resolve the FINESSE nuclear issues was performed. This assessment focused on the ability of tandem mirror FERF options, in conjunction with complementary fusion and non-fusion facilities, to play a role in tokamak development. This type of activity will be expanded in 1985 to cover more cases and to provide a more detailed consideration of the amount of testing required (e.g., number of experiments/tests, required test period, test article size).

The ability of a tandem mirror FERF to provide an act-alike environment for tokamak blanket test modules has been a key concern in FINESSE. Specifically, it was known that the tokamak and tandem mirror geometries, magnetic field profiles, and plasma-side conditions (i.e., heat flux and erosion) were quite different. These issues were addressed, to some extent, during the first year of FINESSE, and it appears that many act-alike aspects of tokamak blanket performance can be achieved in scaled-down tandem mirror test modules, such as the module shown in Fig. 2.6.2-4. In most cases, a radiant heat flux must be applied to the test module first wall to simulate the typically high

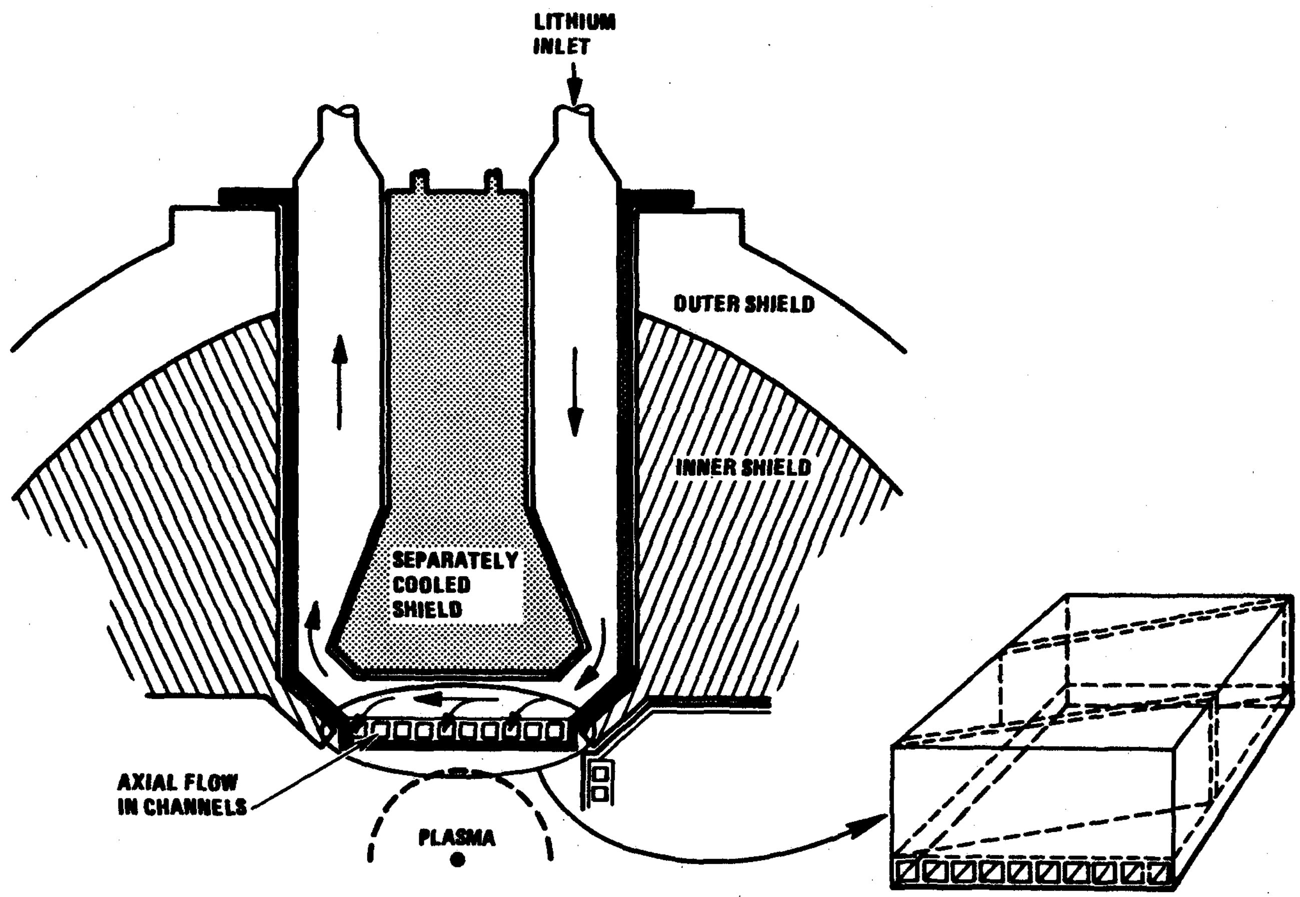


Figure 2.6.2-4 Lithium cooled vanadium tokamak blanket test module design for tandem mirror testing.

surface heat flux in tokamaks. A preliminary design of a tungsten filament resistive heater indicates that such a capability is clearly possible.

Differences between the capabilities of an MFTF-α+T class facility and those of a TDF class facility primarily relate to overall expected availability. Both MFTF-α+T and TDF appear to be attractive candidates to perform thermal-hydraulics and thermal-mechanical testing of act-alike fusion nuclear components. Also, both facilities can and would be used to investigate and resolve early failure modes. However, a TDF class FERF (ultimate fluence of 5-10 MW-yr/m^2) would provide much greater operating time than an MFTF-α+T class FERF (ultimate fluence ~ 1 MW-yr/m^2) and, consequently, is more attractive for tests which involve extended fluence effects (> 4 MW-yr/m^2).

Our preliminary assessment of the contribution to FINESSE issues resolution which can be made by tandem mirrors in conjunction with complementary fusion and non-fusion facilities is encouraging. To perform this assessment, each of ~ 130 FINESSE issues was considered with respect to the contributions which could be made by other experimental facilities (e.g., fission reactors), the adequacy of the tandem mirror test environment, and fluence requirements. Then, one of the five tandem mirror test values shown below was assigned.

Test Value	Tandem Mirror Testing Contribution
a	Sufficiently resolves issue independent of other major test facilities.
b	Sufficiently resolves issue in conjunction with existing fusion and/or potential non-fusion facilities.
c	Confirms performance/reliability goals (developed from non-fusion testing) to mid-life conditions.
d	Establishes an early life performance confirmation/ model benchmark and/or explores early failure modes.
e	No significant benefit due to insufficient or previously existing capability.

The results of the assessment are shown in Table 2.6.2-2. The contribution which can be made by TDF and MFTFα+T class facilities to the resolution

Table 2.6.2-3 Tandem Mirror Fusion Engineering Research Facility Test Value Statistics Summary

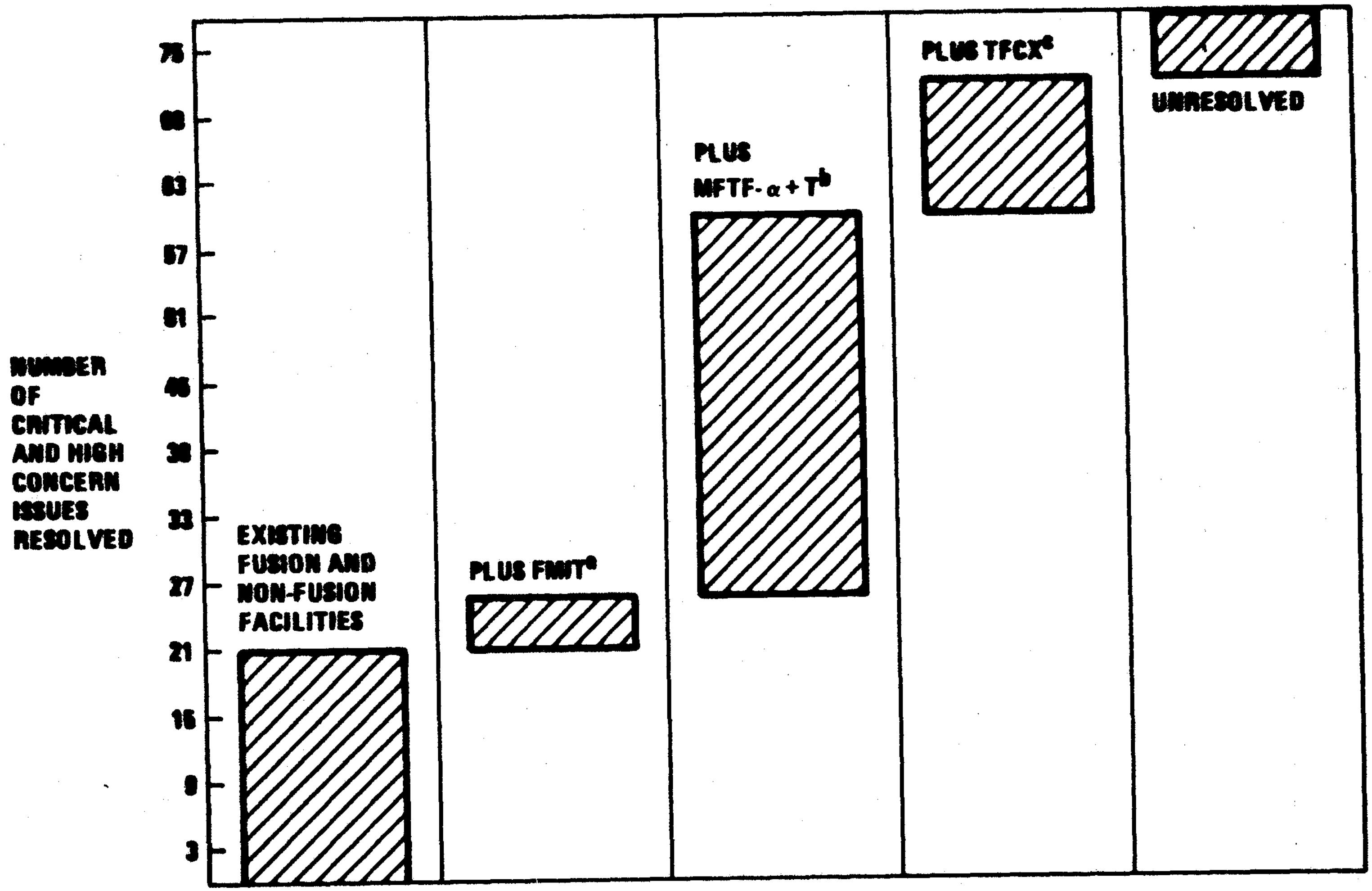
Test Values	Precentage of Critical + High Concern Issues Addressed	
	MFTF-α+T	TDF
a or b	44%	62%
a or b or c	69%	92%
a or b or c or d	92%	97%

of tokamak blanket/first wall nuclear issues is impressive with MFTF-α+T positively impacting the great majority of issues and the higher fluence TDF impacting still more.

As indicated in Fig. 2.6.2-5, an MFTF-α+T facility, with complementary facilities (e.g., existing fusion and non-fusion facilities, FMIT, TFCX) might resolve all but six of the FINESSE nuclear issues associated with tokamaks (i.e., not limited to blanket/first wall issues). The remaining issues which involve fluence effects on large scale components cannot be addressed in MFTF-α+T but can be addressed in TDF.

Given the above perspective on capabilities, it is of interest to compare the performance and cost of MFTF-α+T and TDF with those of recent tokamak test facility designs, the FED-R design[18] and the INTOR design.[19] Such a comparison is presented in Tables 2.6.2-3a and 2.6.2-3b. The reader is advised that the assumptions required to construct this table are uniform and consistent, but do not, in all cases, reflect the published values. Nevertheless, the trends are expected to be relatively valid. Potential improvements in tokamaks as test facilities are summarized in the next section, but a detailed cost evaluation of proposed concepts has not been performed.

As shown, the tandem mirror facilities have potential to produce neutron wall loadings comparable to or in excess of those of the tokamak facilities. This would be done over a much smaller test area at fusion power which is reduced by an order of magnitude or more. The tandem mirror component test area can be increased (Section 13.5.5), but it will, in any case, be small in comparison with that of tokamaks. Although the limited test area is a major concern for tandem mirrors with driven test cells, the virtually steady-state operation of tandem mirrors is believed to be a substantial advantage in the



a) Small size of FMIT box reflects limited breakout of irradiation damage issues
b) Typical low fluence fusion nuclear test facility
c) If normal TF coils, subtract 2 issues and add to unresolved category

Figure 2.6.2-5 FINESSE issue resolution in the context of a multi-facility development plan including MFTF-α+T.

Table 2.6.2-3a Performance Comparisons of Various Fusion Engineering Facility Candidates

	MFTF-α+T	TDF[a]	FED-R(II)	INTOR
Fusion power, MW	17	36	250	620
Neutron wall loading, MW/m^2	2.0	2.1	1.3	1.3
First wall radius, m	0.25	0.30	1.05	1.2
Component test area, m^2	1.6[b]	3.2	60	380
Ultimate availability, %	10	40	40	35
Lifetime at ultimate availability, yr	10	10	10	10
Lifetime fluence, $MW\text{-}yr/m^2$	2.0	8.0	5.2	4.6

[a]Beam-fueled version.

[b]Can be increased to 3.2 m^2.

Table 2.6.2-3b Cost Comparison of Various Fusion Engineering Facility Candidates

	MFTF-α+T	TDF	FED-R(II)	INTOR
Total capital cost, \$M	400	1300	2100	2600
Electrical consumption, MW_e	150	250	600	300
Annual electrical cost, \$M/yr[a]	7	44	105	46
Tritium consumption, Kg/yr	0.10	0.8	5.7	6.2[b]
Annual tritium cost, \$M/yr[c]	2	16	115	124
Annual operating cost, \$M/yr[d]	41	67	105	130
Total annual cost, \$M/yr	50	127	325	300
Total cumulative cost, \$M	~ 1000	~ 2800	~ 5700	~ 6000

[a]At 50 mil/KW_eh

[b]Assumes INTOR TBR and blanket coverage of 50%.

[c]At 20,000 \$/g.

[d]Estimate.

conduct of several types of experiments; most notably, those that involve the dynamics of tritium recovery from solid breeder blankets.

The rough cost comparisons of Table 2.6.2-3b can be used to demonstrate that MFTF-α+T, TDF, and the tokamak alternatives represent distinct cost categories. The MFTF-α+T upgrade would have a low capital cost and an annual operating cost which is expected to be dominated by personnel costs. As a result, the cumulative (i.e., life cycle) cost is expected to be on the order of $1 billion. In comparison, the initial TDF cost is three times higher and the cumulative cost is also three times higher, primarily due to purchased electricity. These differences are compounded by the tokamaks which cost 60-100% more than TDF and require more than double its total cumulative cost despite the fact that the INTOR class facility is assumed to breed half of its own tritium. The FED-R facility is especially costly in electrical consumption because its copper TF coils would require a large electrical input. Thus, the cost differential between MFTF-α+T and a large tokamak engineering test reactor is five- to six-fold and their respective capabilities should be viewed in this light.

2.6.3 Tokamak Test Facilities

2.6.3.1 Objectives and Requirements

The objective of this study was to identify a tokamak device configuration and operational mode that would best utilize the tokamak concept for attaining specific nuclear testing requirements while minimizing capital and operating costs.

The principal test requirements are the following:

Nuclear Performance

1. Test volume at least 0.5 m in depth from frontal area of at least 10 m^2 exposed to the fusion neutron current.
2. Neutron wall loading at least 1 MW/m^2 and preferably 3 MW/m^2.
3. Lifetime fluence capability = 1 to 10 $MW\text{-}yr/m^2$.
4. Surface heat load > 80 W/cm^2.
5. Ease of test module installation and replacement.

Duty Factor

6. Burn time at least several hundred seconds (preferably steady state).
7. Dwell time between pulses < 100 s.
8. Continuous operating period > 1 week.

Cost Constraints

9. Capital cost < $1000 million for the complete facility.
10. Minimum operating cost, i.e., electrical power consumption < 200 MWe and tritium consumption $\leq$ 5 kg/yr.

The last constraint implies a fusion power < 200 MW, assuming a capacity factor near 0.5.

Various device configurations and operational modes that could satisfy the above requirements were examined. These candidates included the numerous toroidal concepts that have been proposed since the mid-1970s as well as new variations made plausible by recent theoretical and experimental plasma physics results. An important guideline was that the reference approach should be a credible one, that could be based on tokamak performance expected to be demonstrated by the mid-1980s.

2.6.3.2 Design Approach and Principal Features

The requirements of small capital cost and small fusion power result in small physical size. An important assumption for minimizing reactor size is that the confinement parameter $n\tau_E$ will not degrade significantly from its value in the ohmic-heated regime; this assumption should be valid if the externally injected power does not significantly exceed the ohmic power.

The following selections were made to insure that the requirements on nuclear performance and duty factor could be achieved in a relatively compact device while minimizing operating cost and risk.

1. Copper TF coils, principally because of the inability to shield superconducting coils in a compact device.
2. Ignited operation, to minimize the electrical power required during the burn. The size penalty for an ignited device was found to be small for the particular class of devices considered here.

3. RF heating (ion cyclotron waves), because of the difficulty of operating neutral-beam injectors in a reactor environment as well as uncertainty in the development of injectors of the required energy and efficiency.

4. Quasi-ohmic heating to ignition, to minimize the auxiliary heating power required.

5. Steady-state current drive was rejected because of its high electrical power cost.

6. Location of the OH coils in the TF coil bore, to maximize the flux swing available for current startup and for driving 1000 s pulses.

7. High-beta operation during the burn ($\langle\beta\rangle$ = plasma pressure/magnetic pressure = 0.23), to minimize TF-coil power loss. This choice requires an elongated bean-shaped plasma.

8. Pumped limiters, to avoid the additional size and complexity of magnetic divertors.

2.6.3.3 Concept Description

Table 2.6.3-1 gives the principal geometric parameters and performance characteristics. Figure 2.6.3-1 shows an elevation view schematic, and Fig. 2.6.3-2 shows how the device sectors are utilized.

Device Operation

To achieve ignition mainly by ohmic heating, very high magnetic field and current are required during startup, but the RF power is relatively low. The startup phase is short compared with the burn phase, so that the time-averaged coil dissipative loss is negligible. The startup plasma is D-shaped. When ignition is attained, the field and current are reduced to moderate values, and beta is increased to 0.23, while the plasma assumes the bean shape needed for high-beta operation. The length of the burn can be of the order of 1000 s, and the duty factor would exceed 0.9.

Magnet Locations

The "pusher coil" required for indenting the bean plasma is located in the inboard blanket/shield region and dissipates about 25 MW of power. The current-driving solenoid is also located inside the TF coils in order to

Table 2.6.3-1 Illustrative Tokamak Nuclear Test Facility

Parameter	Unit	Startup Phase	Burn Phase
<u>Geometry</u>			
Major radius	m	2.55	2.55
Minor radius	m	0.75	0.75
Aspect ratio		3.40	3.40
Plasma shape		D	bean
Elongation		2.0	1.4
Inboard blanket/shield	m	0.50	0.50
Maximum B at coils	T	12	6.0
<u>Plasma</u>			
B at plasma axis	T	5.6	2.8
$\langle\beta\rangle$		.04	.23
<Temperature>	keV	3.0	15
<Density>	$10^{14}/cm^3$	4.5	1.4
Plasma current	MA	7.0	5.4
$n\tau_E$ (neo-Alcator)	$10^{14} s/cm^3$	14	2.1
Z_{eff}		1.2	1.2
Ohmic power	MW	4.9	< 1
RF power	MW	5.0	0
Loop voltage	V	0.65	0.015
Solenoid flux	Wb	32	18
Pulse length	s		> 1000
<u>Magnets</u>			
TF horizontal bore	m	3.5	3.5
TF vertical bore	m	5.3	5.3
TF coil material		Cu	Cu
Maximum J, TF coil	kA/cm^2	.93	.47
TF coil loss	MW	490	122
PF coil loss	MW	60	40
<u>Power Production</u>			
Fusion power	MW	60	185
First wall area	m^2	129	129
<Neutron wall load>	MW/m^2		1.15
at outboard	MW/m^2		1.3
Circulating power	MW	550	190

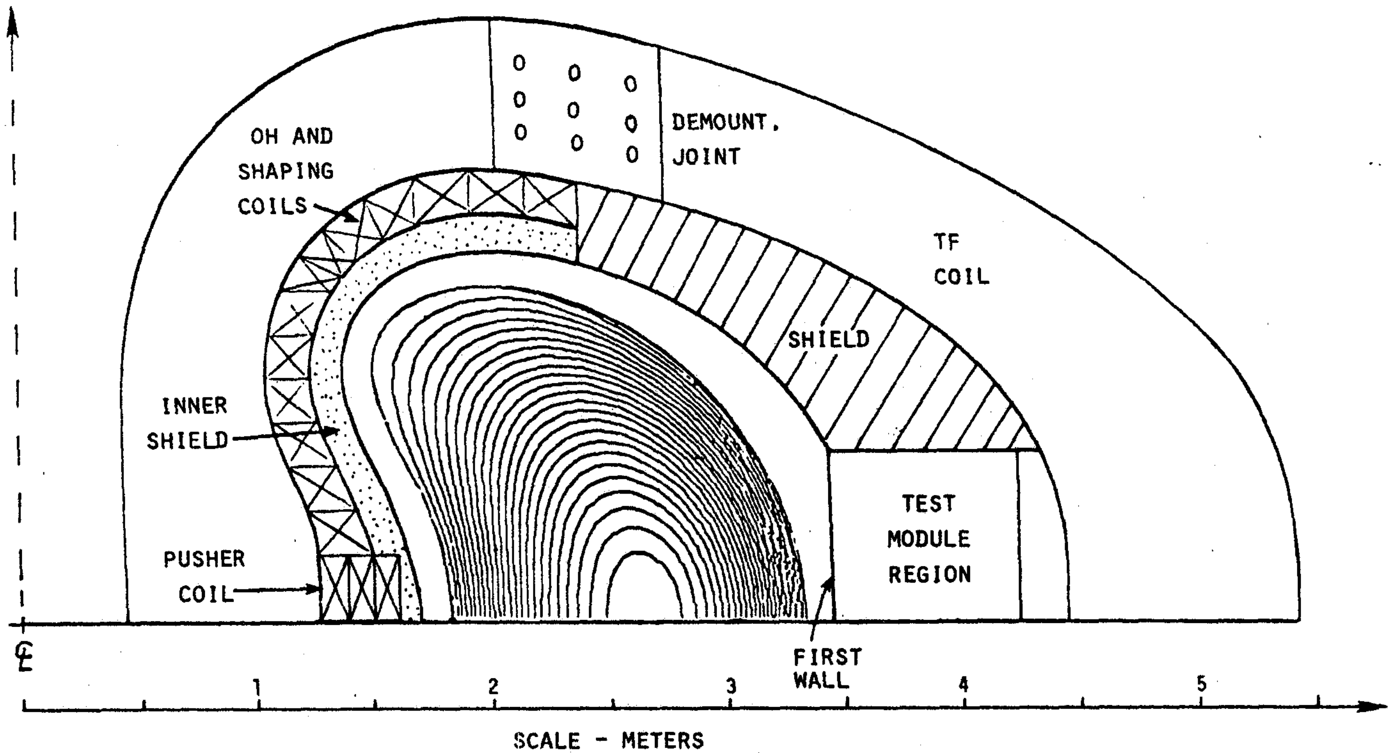
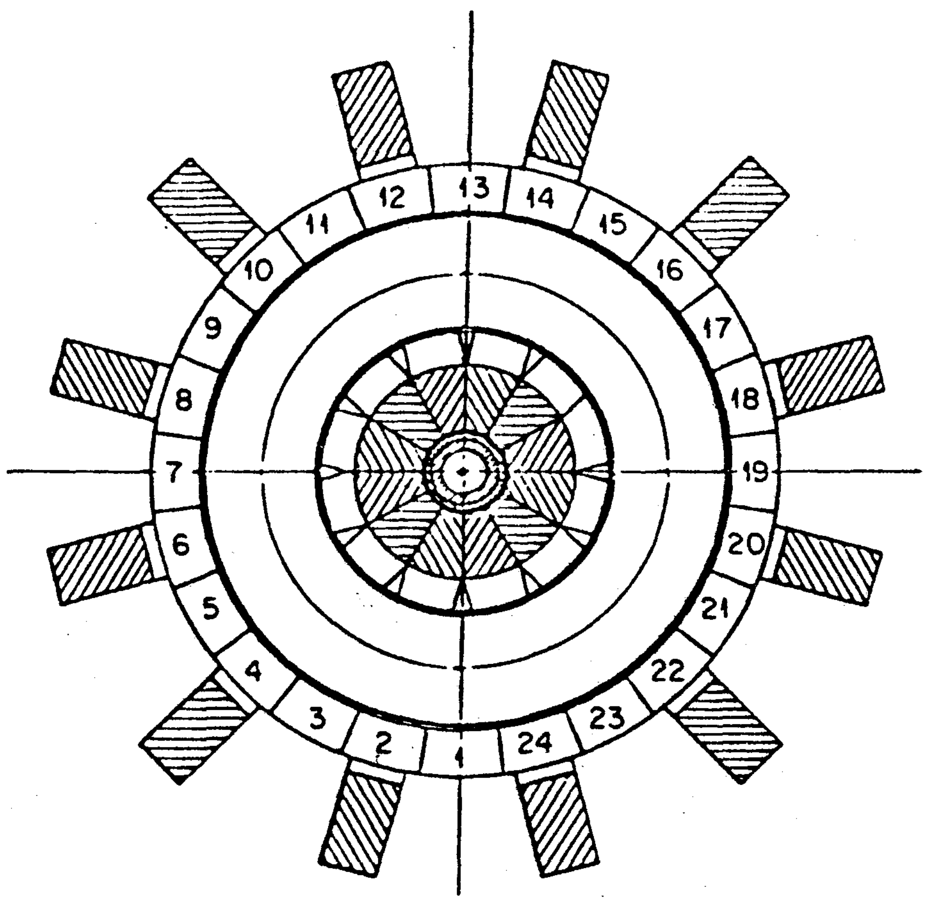


Figure 2.6.3-1 Partial elevation view of tokamak nuclear test facility.



FUNCTION	TOKAMAK NUCLEAR TEST FACILITY SECTORS DEDICATED TO EACH FUNCTION
HEATING	11, 15
FUELING	13, 19
PUMPING	9, 23
DIAGNOSTICS	1, 2, 3
MODULE TESTING	4, 5, 6, 7, 8 AND 17, 18, 20, 21, 22

Figure 2.6.3-2 Utilization of tokamak nuclear test facility sectors.

greatly increase the available flux swing and accommodate the required burn pulse. Minimal neutron shielding is provided for the magnets, all of which use SPINEL insulation. The TF coils are demountable to permit periodic replacement of the inboard TF coil trunk and the in-bore poloidal field coils.

With this configuration, both the fusion power and the circulating electric power are slightly under 200 MW. For a capacity factor of 0.5, the annual operating costs would be approximately $35 million for electricity (at 40 mills/kWh) and $75 million for tritium (at $15,000/g). A significantly different (and still unidentified) design approach would be required to reduce the fusion power and circulating power to the 100 MW level.

2.6.3.4 Risks and Uncertainties of the Reference Approach

The major plasma physics and engineering advances required relative to the expected performances of TFTR, JET and JT-60 are associated with achieving and sustaining ignited, high-beta operation over pulse lengths of the order of 1000 s. On the other hand, reactor-level $n\tau_E$ in the near-ohmic-heated regime, reactor-level temperatures at smaller $n\tau_E$, and the effectiveness of ICRH auxiliary heating. The stability of bean-shaped plasma has been shown at relatively low β values. Within the next few years, existing tokamaks are expected to demonstrate very high beta and pulse lengths of tens of seconds, but no existing machine is likely to achieve ignition.

The following are the physics and technology areas of greatest uncertainty for the proposed approach.

1. The feasibility of achieving and maintaining β = 0.20–0.25.
2. The assumption that the high $n\tau_E$ mode of operation known to be attainable with magnetic divertors can also be realized with pumped limiters.
3. The effectiveness of ceramic insulation in magnet application.
4. Lifetime of the first wall under plasma erosion.
5. Maintenance of the inboard OH and shaping coils is impossible, so that redundancy of these coils is clearly required.
6. Methods for maintaining and replacing tokamak components.

7. Feasibility of the TF coil joints, particularly since the coils must be pulsed to very high field during startup.

8. Cyclic fatigue of the coil systems.

2.6.3.5 Conclusions

The tokamak approach outlined herein can meet the neutron wall loading, fluence and burn cycle requirements for a nuclear test facility, although the wall loading is at the lower end of the range of interest. The electrical power and annual tritium consumption are at the higher ends of the acceptable ranges, and the capital cost would probably exceed $1 billion. Thus, further efforts are needed to reduce the physical size and fusion power level while increasing neutron wall loading.

The proposed approach also has significant technological uncertainties which are likely to be present in any alternative toroidal approach. These issues can be resolved only through extensive development programs.

2.6.4 Availability Considerations for Fusion Engineering Facilities

Studies which can provide a quantitative perspective relating to the reliability/availability aspects of developmental testing in fusion facilities have been completed. These studies were motivated by an observation that many components to be included in the first fusion engineering research and development facilities will have little or no engineering precedence. This will be particularly true of nuclear components which, despite the best efforts in the design, fabrication, and pre-fusion testing phases of development, will not yet have produced a high degree of confidence in their estimated reliabilities. Most likely, early fusion engineering facilities would be used to implement iterative design/test/fix programs aimed at improving the nuclear component reliabilities. However, an apparent paradox will result because those nuclear components which would be targeted in a reliability improvement program depend upon the reliable performance of other nuclear components in the system.

One example of this would be the development of blanket test modules in a high fusion power facility (e.g., INTOR) which must also breed its own tritium in many (typically 60) tritium breeding modules, all of which must function reliably for the overall facility to operate. Although the breeding modules would be designed for high reliability, they would be essentially unproven and necessarily complex. Consequently, they have the potential to negatively influence the blanket test module development program by reducing the overall device availability.

Two studies which address these concerns will be reviewed briefly in this section. First, the total test time required to achieve a given level of statistical confidence in a required component reliability is considered. Second, the integrated time in a test facility which is required to improve component availability from an initial value to a goal value is estimated while accounting for the degradation of operational availability caused by tritium breeding module and test module failures. This work is based on Department of Defense[20,21] guidance relating to the planning and management of reliability improvement programs.

2.6.4.1 Confidence Levels in Component Availability

Unproven component reliabilities or availabilities (considering replacement and repair) can be estimated from the proven performance of components of similar design and application if such designs and applications exist. However, high confidence in component performance in entirely new applications must come from testing in relevant environments. The implementation of an operation/test program to develop high statistical confidence in a reliability data base prior to an engineering demonstration is clearly a desirable goal, but can be very difficult in practice due to the requirement for an extended test period and because such a program would logically follow a relatively long design/test/fix/test sequence (i.e., one should achieve high reliability prior to confirming it). The INTOR critical issues study[21] concluded that the achievement of an 80% statistical confidence level in a given component Mean Time Between Failures (MTBF) in the constant failure rate regime of operation (i.e., random failure probability) would typically require a cumulative test period of 3.5 times the MTBF.

Some components, such as the superconducting toroidal field (TF) coils, are not expected to fail during the lifetime of the facility (a design basis) and have required MTBF periods which are orders of magnitude in excess of the facility lifetime. For example, a tokamak engineering demonstration facility

(EDF) might have ten TF coils and might require a TF coil system reliability of 80% over a ten full power year operational lifetime. This implies that a single coil must have a reliability of $(0.80)^{1/10} = 0.978$, or 97.8%, during the same period. Since the reliability, R, is related to the component MTBF by $R = e^{-\tau/MTBF}$, where τ is the operational period (ten years in this case), it follows that for R = .978, the individual TF coil MTBF must be 450 operating years. For 80% confidence in this MTBF, a 3.5 x 450 = 1575 year test might be required. Although this is reduced ten-fold because there would be ten TF coils, it is clear that such components would not be amenable to the prior development of a high confidence reliability data base.

For blanket modules, the reliability requirements would be somewhat relaxed. With the possible exception of catastrophic failures, such as a severe breach of the primary pressure boundary (resulting in a gross deformation) or a non-routine radioactive spill, blanket removal is expected to be a relatively routine maintenance operation. In this case, a minimal engineering demonstration facility blanket system availability goal might be ~ 60%. Since an EDF might have six blanket modules per TF coil sector (60 total), the required availability for individual components might be $(0.6)^{1/60} = 0.9915$ or 99.15%. Since the component availability is given by the MTBF/(MTBF + MTTR), where the MTTR is the mean time to repair or replace, a typical MTTR of one month results in a required MTBF of about 10 years. This implies a typical test period of 34 years. However, if equal credit can be taken for 60 modules, tested in parallel, the required test period would be reduced to a managable 0.5 full power years.

These results are illustrated in Fig. 2.6.4-1, where the individual blanket module availability is shown as a function of the overall blanket (60 modules) availability. Note that the module availability requirement exceeds 99% for blanket availability goals exceeding 50%. Typical required test times to provide an 80% confidence in the required module availability for different values of the MTTR (which imply different MTBF requirements) are also shown in the figure.

Based upon both of the above analyses, it appears that the goal of developing a reliability data base for blankets in an INTOR class facility would be difficult but not impossible.

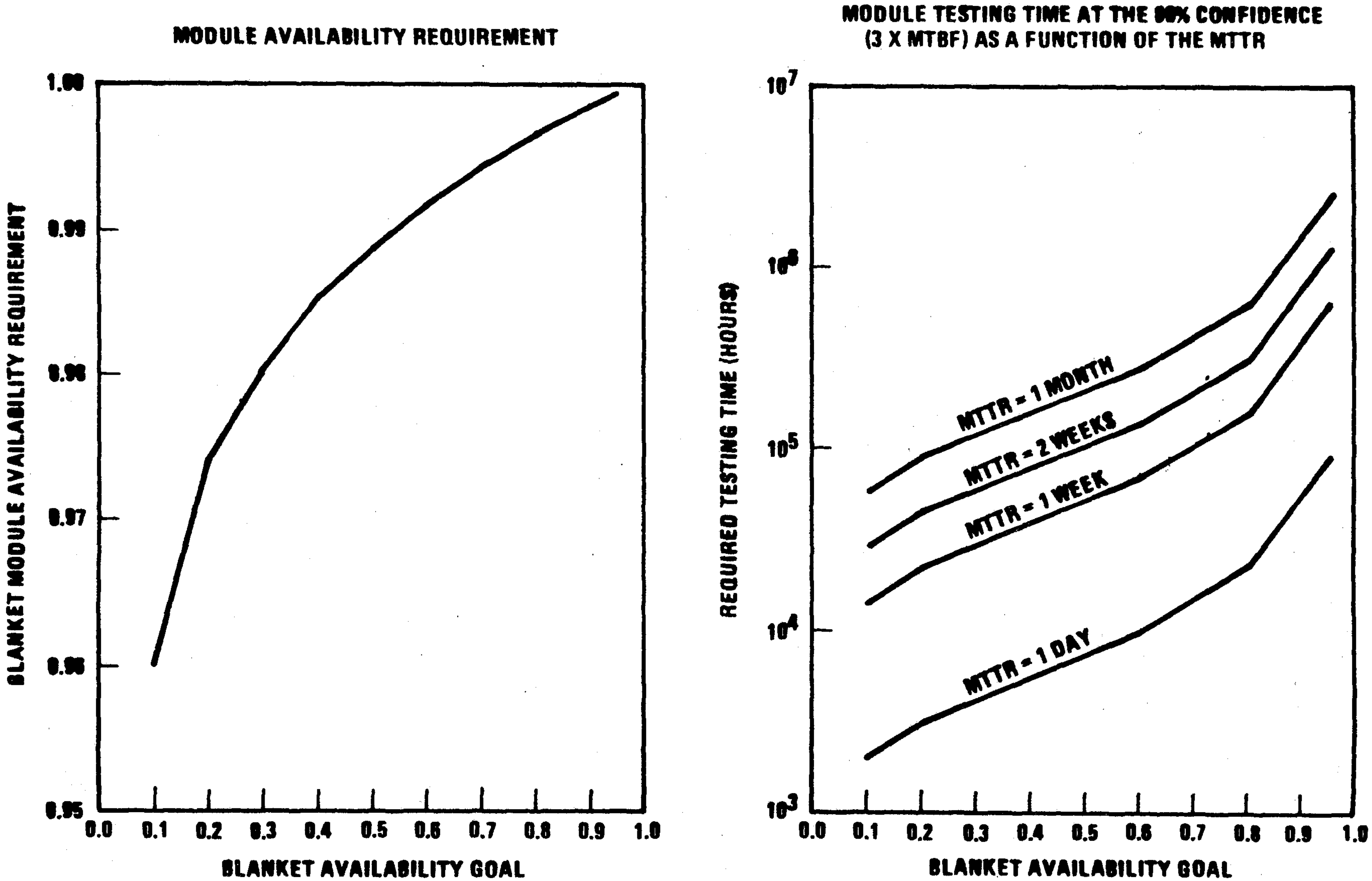


Figure 2.6.4-1 Blanket module availability and typical testing requirements meeting the 80% confidence level (3.5x MTBF).

2.6.4.2 The Potential Impact of Reliability Development Testing

In addition to testing for confidence in an estimated level of reliability, iterative design/test/fix sequences which can result in component reliability improvement are also of interest. Although a reliable predictive capability in this area cannot be obtained (no precedence for fusion nuclear components), Department of Defense systems development documents[20,21] suggest the following general form of a parametric relationship between the component development/test time and the achieved component MTBF:

$$MTBF = CT^m$$

where T is the testing time, m is a testing improvement exponent (typically $0.1 \leq m \leq 0.6$), and C is a constant determined by the initial component MTBF and initial testing time. The above equation presents an MTBF improvement model which achieves a goal MTBF through a more or less aggressive (depending upon m) testing and development program but does not demonstrate the achieved MTBF in the statistical sense described above.

The following analysis assumes that such an improvement schedule can be achieved and goes on to consider the implications on the testing and development of blanket modules for the following typical fusion development pathways (see Section 2.7 for discussion).

1) TFTR/TFTR-U → INTOR → DEMO

2) TFTR → $\left.\begin{matrix}\text{TFCX} \\ \\ \text{TDF}\end{matrix}\right]$ → ETF/EDF

3) TFTR → $\left.\begin{matrix}\text{TFCX} \\ \\ \text{MFTF-}\alpha\text{+T}\end{matrix}\right]$ → ETF/EDF

The initial facilities in these pathways are expected to be oriented primarily towards confinement physics goals, while the intermediate engineering facilities (e.g., INTOR, TDF, and MFTF-α+T) are intended to achieve engineering (e.g., reliability improvement) goals. In the first pathway, an INTOR class facility (~ 600 MW) would be a large tokamak which would breed most of its own tritium in nine of the ten toroidal sectors (six blanket modules per sector). This facility would develop and test more advanced blanket concepts for the DEMO in the tenth blanket sector. Therefore, 54 of 60 modules would be dedicated to tritium breeding, while six of 60 modules would be used for testing and development. In comparison, the TDF class facility would include only the latter six modules and the MFTF-α+T facility would differ from TDF only in its lower overall availability.

In the second and third pathways, the complications caused by relying upon unproven tritium breeding modules are avoided by operating the engineering test facilities at a low enough fusion power to purchase tritium from an external source. In contrast, an INTOR class facility will be required to suffer through any availability reductions which are caused by in-situ tritium breeding. Consequently, it is expected that a TDF class facility will operate at a higher availability and will achieve the MTBF goal for blanket test modules more quickly than the INTOR class facility.

The availability logic for the DEMO and ETF/EDF facilities would be similar to those for INTOR (i.e., 60 blanket modules in series). In the first and second pathways, the blanket test module availability would be improved as a result of iterative design/test/fix sequences in INTOR and TDF, respectively. The result would be the achievement of a goal MTBF sufficient to provide a high initial level of nuclear system availability to support DEMO or ETF/EDF initial availability (after a three-year startup phase) of 30% or 20%, respectively. In the third pathway, MFTF-α+T is expected to begin the design/test/fix program but is not expected to provide sufficient test time to achieve the reliability goal. In this case, the ETF/EDF of the third pathway would complete the reliability growth program.

For each engineering facility, availability models were developed which describe the influence of component failures on facility availability. The facility availability, in turn, determines how much calendar time is required to accomplish testing and development goals. In this analysis, the time required to achieve a relatively low but acceptable blanket module MTBF of 87,600 hours in the three pathways was calculated based on the parameters shown in Table 2.6.4-1. Shorter testing times resulted in lower MTBFs and lower initial availabilities for the DEMO and the ETF/EDF.

Table 2.6.4-1 Key Assumptions in the Availability Analysis

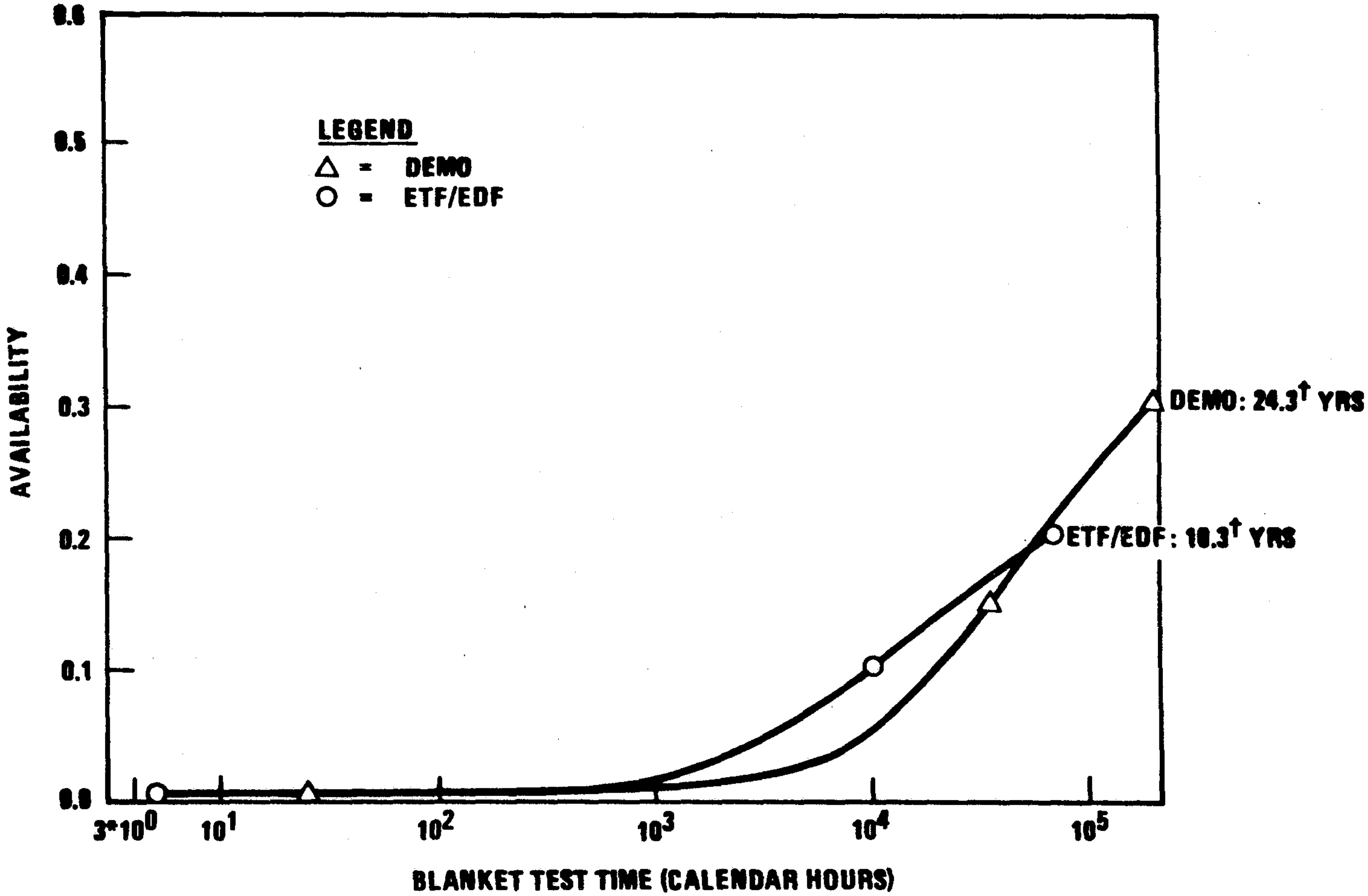
	Blanket Test Modules	Blanket Tritium Breeding Modules
Initial MTBF (hrs)	8760	25263
Initial test experience (hrs)	758	2374
MTTR (hrs)	336	672
Goal MTBF	87600	87600
Test improvement factor	0.50	0.10
Experience factor[a]	0.50	0.50

[a](credit for N modules/credit for 1 module) = N^n, where n is the experience factor, $0 \leq n \leq 1$.

The calendar times required to achieve the nuclear system MTBF levels required to support the above initial facility availabilities are shown in Fig. 2.6.4-2. Pathway 2 achieves the 87,600 blanket module MTBF goal in 10.3 calendar years as compared to the 24.3 calendar years for the same MTBF along Pathway 1. This is because the unproven tritium breeding modules in the INTOR class facility result in slow availability growth such that testing takes a relatively long time. Pathway 3 is not shown in the figure, but it takes about 21 years in MFTF-α+T and the ETF/EDF to achieve an initial EDF availability goal of 30%.

Parametric studies performed over the parameters shown in Table 2.6.4-1 indicate that the relative performance of the three pathways is not expected to change. However, in some cases, the two differences among the three pathways are much longer than indicated here; but, in some cases, the development times become so short or so long that there is little difference among the pathways.

In summary, these results indicate the vulnerability of an INTOR class facility to excessive downtimes from random failures when it is required to do component development and testing and tritium breeding. This concern does not



†INCLUDING 3-YR STARTUP

Figure 2.6.4-2 Follow-on facility availability vs. blanket/test/development time in the precursor facility.

uniquely demonstrate that INTOR class facilities cannot achieve reasonable goals. The results do, however, indicate that the relative difficulty of achieving MTBF goals in such a facility could be high compared to less ambitious facilities which are not required to breed tritium.

2.7 Fusion Research and Development Scenarios

A principal objective of FINESSE will be the development of a recommendation regarding the types and sequences of experimental and test facilities which are expected to be best suited for the R&D of fusion nuclear technologies. As a first step towards fulfilling this objective, a preliminary screening evaluation of many possible technology R&D pathways was performed. A limited number of pathways (perhaps four) will be developed in more detail during the second year of FINESSE.

2.7.1 Planning Considerations

In constructing a "roll-forward" logic for fusion R&D, the planner must strive for a uniformity of assumptions and a consistency of logic to the maximum extent possible. For example, the following five questions can be pivotal in determining the cost, risk, and schedule of fusion development pathways:

- Must every near-term, D-T burning fusion facility operate in a physics mode which is presently perceived to be extrapolatable to a reactor-relevant "strategic goal" of the program? For example, will a non-ignited, beam driven physics mode be acceptable for an early technology facility if it results in a lower cost and/or risk?
- Will early experimental fusion technology facilities be required to provide much or all of the tritium fuel required to sustain their own operation?
- Will the number of blanket structure/coolant/breeder combinations be reduced to one or a few principal options prior to performing experiments in a fusion facility?
- Will the objective of experimental testing in the fusion environment be: a) screening for early failure modes, b) extended testing to achieve a reliability data base with a high statistical confidence, or c) a design/test/fix sequence for reliability improvement? How will these objectives be modified in consideration of available facilities?
- Will a high fluence, 14 MeV, irradiation damage data base be available from point neutron sources prior to the extended operation of structural and breeding materials in a fusion environment? How will the availability (or unavailability) of such a facility affect R&D planning?

In considering the first of these questions, it is important to note that several of the preliminary FINESSE R&D pathway scenarios will not require that early technology facilities operate in a reactor-relevant physics mode. For example, pathways which include Fusion Engineering Research Facilities (FERFs), which do not necessarily feature reactor-relevant fusion plasma physics but operate in parallel with reactor-relevant Plasma Burning Experiments (PBXs), can receive favorable evaluations as FINESSE scenarios.

The issue of tritium production in the first experimental technology facilities is most important when the facility has a high fusion power level. For example, an INTOR class facility,[2] with a 500 MW fusion power level and an ultimate capacity factor of 40% would burn about 12 kg of tritium per year. As shown in Fig. 2.7-1, a steady supply of 2.8 kg/yr tritium from the Canadian nuclear program,[27] starting in 1988, would sustain this level of operation only if an internal tritium breeding ratio (TBR) of about 0.8 could be achieved. If the fusion power were reduced to 200 MW, a 0.5 TBR would still be required. Conversely, even if the tritium were available for purchase (e.g., from a U.S. military stockpile), the cost might be prohibitive. If all of the tritium required to sustain a 500 MW fusion power level were available at a cost of \$10,000-\$20,000/g, the annual operating cost for tritium alone would be in the range of \$120-\$240 million/yr.

Any requirement for in-situ tritium breeding impacts the cost of the facility, but, more importantly, it impacts its operational risk. That is, it is not clear that an operational tritium breeding blanket which is reliable enough to enable a 30-40% overall facility availability can be developed for this application with no prior testing in a fusion environment is available. The apparent paradox of not being able to develop a reliable blanket until such a blanket exists has been explored in some detail (see Section 2.6.4) with the result that it appears prudent to penalize any first generation fusion facility which requires a large in-situ TBR.

The number of combinations of blanket coolants, tritium breeding materials, and structural materials presently being considered by the fusion community is enormous[6] and the cost impact of continuing to carry several options will increase as the program progresses to performing blanket experiments on fusion technology facilities. Each type of blanket will require many experiments and tests and can have unique and far-reaching requirements

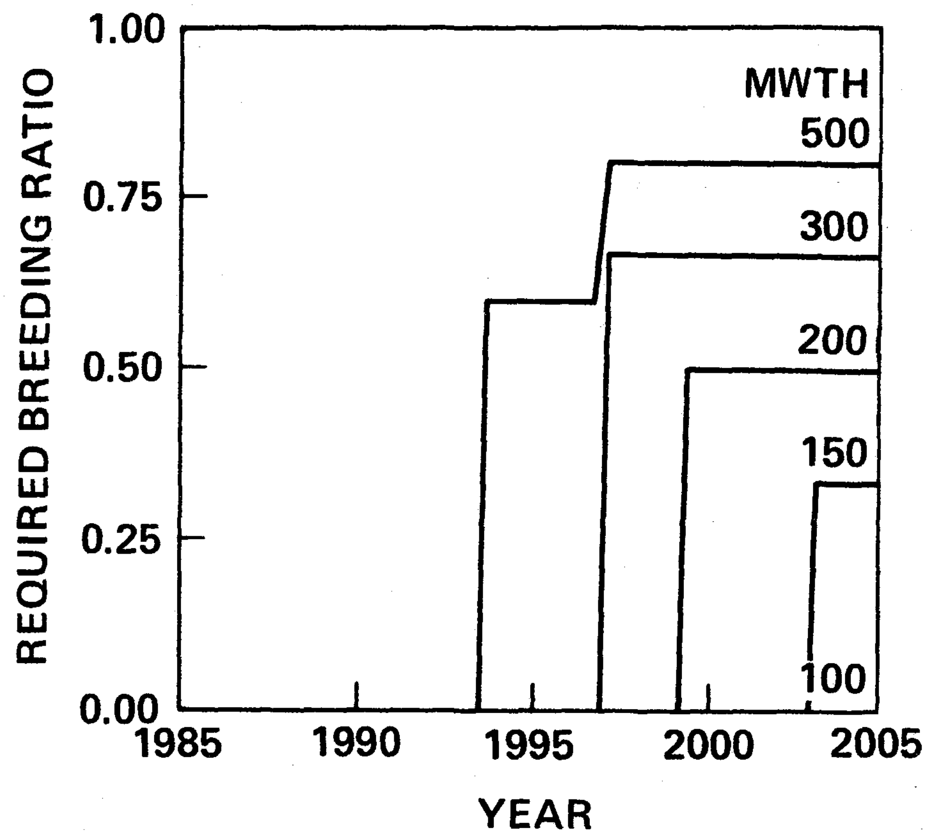


Figure 2.7.1-1 Tritium breeding ratio needs for 2.8 kg/yr tritium supply rate (starting 1988) with startup of fusion device delayed until 1995.

regarding safety, tritium release, power conversion, and remote maintenance systems. It is expected that a dramatic reduction in the number of blanket concepts will be required so that a development program of reasonable scope and cost can be conducted.

Choices for the objectives of experiments to be conducted in fusion facilities determine the requirements on such facilities; therefore, they must be carefully considered in R&D scenarios. Specifically, some possible objectives of the experiments are as follows:

- Obtain required data points by conducting experiments for fixed and predetermined periods of time. Such data would include any information necessary for gaining sufficient understanding of new phenomena such that an evaluation and comparison of the feasibility and attractiveness of candidate concepts can be conducted.

- Identify design flaws and early failure modes by conducting beginning-of-life screening tests.
- Implement reliability improvement goals in addition to the identification of early design flaws and failure modes by conducting iterative design/test/fix/test sequences.
- Generate sufficient data to support a high statistical confidence in the reliability of components by conducting long-term operation/test programs.

Among these, the first type of experiments can apply to any type of test (e.g., basic, single effect, multiple effect, integral and component), while the second is most applicable to multiple-effect and integral experiments, and the latter two objectives are most applicable to component testing in a fully integrated environment. The amount of time required to conduct the second type of experiments depends upon the particular design features, but is typically measured in hours or days. In contrast, the time required to perform the first, third or fourth type of experiments can be much longer.

A key determinant of the size (as opposed to the time duration) of a test program is the number of identical test articles in the test matrix which are required to provide acceptable test statistics. For example, to establish data to a goal fluence, NM test articles could be required, where M would be the number of identical test articles required to achieve adequate test statistical confidence and N is the number of intermediate fluence values at which data is needed from destructive test article examination. Some reasonable observations about the value of M are:

- $M = 1$ is the minimum but can lead to misleading results if spurious materials or manufacturing defects (e.g., bad welds) are present in the test article.
- $M = 2$ or 3 should be sufficient to eliminate some of the above concerns.
- $M > 3$ will provide good test statistics but can result in a large number of test articles.

The required test periods and the required number of test articles have not, as yet, been considered in detail.

A final consideration in the creation of R&D scenarios is the possible future availability of a facility to generate a data base for high fluence irradiation damage due to high energy neutrons. Candidate facilities include accelerator-based neutron source experiments (e.g., FMIT class but possibly more modest) which would be operated prior to or in parallel with fluence testing in fusion technology facilities. In the absence of a high energy, high fluence calibration, there is a significant risk of experiencing unanticipated damage phenomena which could result in a premature and systemmatic failure of the blanket structure and/or other structural components. With a high fluence calibration, it will be possible to consider a program of act-alike, but low fluence, fusion testing as a reasonable compromise to provide a basis for proceeding to the minimum acceptable demonstration facility.

It is important to emphasize that any successful fusion technology development pathway will incorporate the maximum utilization of both existing experimental facilities and any new facilities that can provide a substantial risk reduction or long-range cost savings. To provide the greatest leverage with available funding, it is reasonable that experimentation in such precursor facilities should initially concentrate upon the investigation of critical feasibility issues for which an unfavorable resolution might lead to the unacceptability of a given design option (e.g., MHD flow effects in self-cooled liquid metal blankets might prove to be excessive). A second priority would be the resolution of poorly understood physical phenomena (e.g., tritium recovery from solid breeders). Beyond this, the test program should be planned to develop key engineering data bases for subsystems which will be used in the first generation of fusion facilities.

2.7.2 Fusion Experimental and Test Facilities

As a first step towards the evaluation of fusion technology development scenarios, a set of generic definitions for major fusion facilities was developed. These definitions, summarized in Table 2.7-1, apply to both tokamaks and tandem mirrors (as well as other confinement concepts) and were used as guidelines in the construction of the various R&D scenarios. It is not required that each pathway include each type of facility. For example, several of the candidate pathways which have been included in the preliminary evaluation move directly from a plasma burning experiment (PBX) to an engineering test facility (ETF) followed by an engineering demonstration facility. Other pathways include an early fusion engineering research facility

Table 2.7-1 Descriptions of Major Fusion Development Pathway Facilities

	Plasma Burning Experiment (PBX)	Fusion Engineering Research Facility (FERF)	Engineering Test Facility (ETF)	Engineering Demonstration Facility (EDF)
Mission	Develop understanding of burning plasma operation and optimization.	Investigate operation of fusion components in a high duty cycle nuclear environment.	Optimize and test fusion technologies using a reactor relevant mode of operation.	Provide an engineering demonstration of the technology at reasonable availability.
Description	Configuration relevant to resolve major plasma physics issues.	Configuration relevant to nuclear experimentation. Capability for several experiments.	Fully integrated environment suitable for testing majority of interactive effects.	Nearly all systems prototypical but smaller than full-scale commercial.
<u>Minimum</u> Fluence/Flux Goals	Negligible.	~ 1 $MW\text{-}yr/m^2$ at ~ 1 MW/m^2.	~ 4 $MW\text{-}yr/m^2$ at ~ 1.5 MW/m^2.	~ 8 $MW\text{-}yr/m^2$ at ~ 3 MW/m^2.
<u>Minimum</u> Availability	Negligible.	Tens of runs per year.	Ultimately ~ 30%.	Ultimately ~ 50%.
Risk/Schedule	Risk can be high. Should be first facility in path.	Risk as a neutron provider must be low. Experimental risk can be higher.	Only test articles can be high risk.	Only higher fluence tests ($\gtrsim$ 4 $MW\text{-}yr/m^2$) can be risky.
Facility Examples	TFCX, LITE, MFTF-α+T	MFTF-α+T, TDF, FED-R	INTOR, FPD	STARFIRE/DEMO

(FERF), but eliminate the first generation ETF and replace it with a more advanced engineering test facility which can be upgraded to a first generation engineering demonstration facility (EDF). Such a facility is designated as an ETF/EDF and is considered to be about one-half step beyond the first generation ETF of Table 2.7-1.

Other facilities and facility combinations, which reflect possible upgrades, are defined as follows:

- SFE/SPE-U: Scientific feasibility experiments (SFEs) such as TFTR and MFTF-B might be upgraded for scenarios which proceed directly from the SFE to an ETF. The SFE-U mission would be less ambitious than that of the PBX due to the limitations of existing facilities.
- PBX/FERF: An upgrade of a burning core experiment during a second phase to perform experimentation/testing in a fusion nuclear environment is a reasonable option for consideration, but is expected to substantially increase the initial cost of the PBX.

It is important to emphasize that the primary mission of the FERF is to explore scientific and engineering phenomena involving the operation of fusion components in a fusion environment. The tritium breeding blanket is typical of components which would be a key part of the FERF experimental program, but other components which would be investigated include plasma interactive components, instrumentation and control systems, remote maintenance systems, tritium control systems, safety systems, radiation tolerant magnetic coils, and environmental control systems.

The most important requirements for a FERF include steady-state plasma operation or long burn with a high duty cycle, an ability to simulate act-alike performance for blanket modules (and other components), and the availability of a sufficiently large test cell area. A minimum neutron wall loading of about 1 MW/m^2 is required and a high fluence capability is desirable if the facility capital cost, tritium requirements, electricity costs, and other operating costs are reasonable in an overall budget context. The FERF must have a low physics risk as a neutron provider. An ignited mode of plasma operation is not required. The risk in screening blankets and other components (a primary mission) can be higher than the physics risk.

2.7.3 Fusion Development Pathways and Evaluations

A list of generic fusion development pathways is provided in Table 2.7-2. Typical example cases for tokamak and tandem mirror development (designated T and M) are also shown. It is important to note that both major confinement schemes can contribute to a given development pathway. For example, both tokamak development pathways of generic pathway D (see Table 2.7-2) include FERF class facilities which would be small, neutral beam driven tandem mirror test facilities (i.e., MFTF-α+T and TDF). Such facilities can be very attractive in the tokamak development pathways because they promise a high fusion power density (i.e., wall loading) at a relatively low cost (see Section 2.6.2).

Three pathway milestones which define a uniform set of development pathway goals have been defined. These are designated according to the accumulated neutron fluence and facility availability as follows:

- Engineering Feasibility Milestone: The achievement of a 2 MW-yr/m^2 neutron fluence over several tests and test articles in a FERF or ETF.
- Intermediate Fluence and Availability Milestones: An accumulated neutron fluence of 4 MW-yr/m^2 on a sufficient number of act-alike component tests and/or test modules to resolve key development issues for the operation of an EDF at an average availability of ~ 30%.
- Engineering Demonstration Milestones: An accumulated neutron fluence of 8 MW-yr/m^2 on nuclear components and an average availability of ~ 50% in the near-prototypical EDF.

The first of these, "engineering feasibility," is intended to be analogous to the "scientific feasibility" milestone for fusion plasma physics experiments. The second milestone, "intermediate fluence and availability," would be sufficient to qualify components for intermediate life conditions in an EDF. The "engineering demonstration" milestones correspond to the minimum fluence and availability which might be sufficient to qualify components for end-of-life conditions in the first commercial-scale fusion reactor. The years in which these milestones are achieved depend upon assumptions regarding when the first facilities in each of the respective pathways might operate, the operating availabilities of those and subsequent facilities, their respective neutron wall loadings (to achieve fixed fluence goals), and various interface assumptions.

Table 2.7-2 Generic Fusion Development Pathways and Typical Examples

A) SFE/SFE-U → ETF → EDF

1) T: TFTR/TFTR-U → INTOR → STARFIRE/DEMO

2) M: MFTF-B/MFTF-U → FPD → EDF

B) SFE → PBX → ETF → EDF

1) T: TFTR → TFCX → ETF → EDF

2) M: MFTF-B → MFTF-α → FPD → EDF

C) SFE → PBX/FERF → ETF/EDF

1) T: TFTR → TFCX/TFCX-U → ETF/EDF

2) M: MFTF-B → MFTF-α+T → ETF/EDF

D) $\left.\begin{array}{r}\text{SFE/SFE-U}\\ \text{FERF}\end{array}\right] \rightarrow \text{ETF} \rightarrow \text{EDF}$

1) T: $\left.\begin{array}{r}\text{TFTR/TFTR-U}\\ \text{MFTF-}\alpha\text{+T}\end{array}\right] \rightarrow \text{ETF} \rightarrow \text{EDF}$

2) T: $\left.\begin{array}{r}\text{TFTR/TFTR-U}\\ \text{TDF}\end{array}\right] \rightarrow \text{ETF} \rightarrow \text{EDF}$

3) M: None

E) SFE → $\left.\begin{array}{r}\text{PBX}\\ \text{FERF}\end{array}\right] \rightarrow \text{ETF} \rightarrow \text{EDF}$

1) T: TFTR → $\left.\begin{array}{r}\text{TFCX}\\ \text{MFTF-}\alpha\text{+T}\end{array}\right] \rightarrow \text{ETF} \rightarrow \text{EDF}$

2) T: TFTR → $\left.\begin{array}{r}\text{TFCX}\\ \text{TDF}\end{array}\right] \rightarrow \text{ETF} \rightarrow \text{EDF}$

3) M: None

Table 2.7-2 Generic Fusion Development Pathways and Typical Examples (contd.)

F) $\text{SFE} \rightarrow \left.\begin{array}{r}\text{PBX}\\ \text{FERF}\end{array}\right] \rightarrow \text{ETF/EDF}$

1) $\text{T: TFTR} \rightarrow \left.\begin{array}{r}\text{TFCX}\\ \text{MFTF-}\alpha\text{+T}\end{array}\right] \rightarrow \text{ETF/EDF}$

2) $\text{T: TFTR} \rightarrow \left.\begin{array}{r}\text{TFCX}\\ \text{TDF}\end{array}\right] \rightarrow \text{ETF/EDF}$

3) $\text{T: TFTR} \rightarrow \left.\begin{array}{r}\text{TFCX}\\ \text{FERF (Tokamak)}\end{array}\right] \rightarrow \text{ETF/EDF}$

4) $\text{M: MFTF-B} \rightarrow \left.\begin{array}{r}\text{MFTF-}\alpha\\ \text{TDF}\end{array}\right] \rightarrow \text{ETF/EDF}$

G) $\text{SFE} \rightarrow \left.\begin{array}{r}\text{PBX}\\ \text{FERF}\end{array}\right] \rightarrow \text{EDF}$

1) $\text{T: TFTR} \rightarrow \left.\begin{array}{r}\text{TFCX}\\ \text{FERF (Tokamak)}\end{array}\right] \rightarrow \text{EDF}$

2) $\text{M: MFTF-B} \rightarrow \left.\begin{array}{r}\text{MFTF-}\alpha\\ \text{TDF}\end{array}\right] \rightarrow \text{EDF}$

The following tokamak R&D pathways were subjected to a preliminary evaluation during the first year of FINESSE: A1, B1, D2, E2, and F2. Pathways A1 and B1 are "conventional" pathways which have received a large amount of consideration during the past decade. Pathways D2, E2, and F2 are of particular interest because they attempt to utilize a tandem mirror FERF in a tokamak development pathway. For the preliminary evaluation, it is assumed in all pathways that the issue of irradiation damage calibration is adequately resolved prior to construction of the EDF. Pathways not considered in the preliminary evaluation will be considered during the first part of the second year of FINESSE.

Comparing development pathways A1, B1, D2, E2, and F2, our preliminary ranking would be in reverse order of the above list (i.e., F2 ranks highest). As shown in Table 2.7-3, pathway A1, featuring a scientific feasibility experiment upgrade (SFE-U) followed by an ETF, is expected to result in a high level of risk despite impressive cost and schedule attributes. This level of risk may be unacceptable. Pathway B1 reduces the level of risk by providing a PBX prior to the ETF. Nevertheless, the risk in integrating and testing nuclear components in the ETF remains high.

Pathway D2 removes the PBX but adds a FERF. In this case, the nuclear risk is reduced but the physics risk is increased relative to pathway B1. The time required to achieve the demonstration milestone is extended by about five years, increasing the overall funding requirements.

Pathway E2 features both a PBX and a FERF. In this case, the risk is acceptable but the near-term and overall costs are large due to the large number of major facilities. Some apparent advantage is provided by combining the ETF and the EDF in pathway F2. In the later case, the risk is increased but the overall cost and schedule are improved. Consequently, pathway F2 appears to be the most attractive of those considered in the preliminary evaluation.

Additional pathways which appear to be promising include those which feature a PBX and a low fluence FERF (e.g., MFTF-α+T upgrade of Chapter 13), leading to an ETF or an ETF/EDF. Comparing the latter cases (i.e., pathways E1 and F1) with pathways E2 and F2, the risk would increase somewhat, but the engineering demonstration milestone would be achieved more quickly and the near-term funding requirement would be more moderate. The inclusion of an FMIT class facility in such a pathway would help to ameliorate the risk.

2.7.4 International Implications

Clearly, enhancing fusion technology R&D effort implies a near-term funding requirement which is in excess of current fusion funding levels of any of the large national and multinational programs in the U.S., Europe or Japan. Consequently, international cooperation might prove to be the most effective means of achieving a well balanced and aggressive fusion development program. Although the FINESSE program has made no serious attempt at defining

Table 2.7-3 Summary of Preliminary Evaluations

Pathway	A1	B1	D2	E2	F2
Fusion Facilities	SPE-U ETF EDF	PBX ETF EDF	SFE-U FERF ETF EDF	PBX FERF ETF EDF	PBX FERF ETF/EDF
Overall Operational Risk	High	Mod to High	Mod to High	Low	Low
Nuclear Testing/ Development Risk	High	High	Low to Mod	Low to Mod	Low to Mod
Engineering Feasibility Milestone Date	2005	2008	2002	2002	2002
Engineering Demonstration Milestone Date	2020	2023	2028	2028	2023
Near-Term Funding Required	Low to Mod[a]	Mod	Mod	High	High
Overall Funding[b] Required	Low	Mod	Mod to High	High	Mod
Rank (No. 1 = Best)	5	4	3	2	1

[a]Assuming ETF is not built and, hence, cost not included, in the near term. If ETF is built in the near term (e.g., on the same time frame as PBX and FERF in Pathway F2), the near-term funding changes to high.

[b]Possible increases in cost due to possible failures in the high-risk pathways are not included.

an international strategy which resolves the myriad of institutional difficulties which would be sure to arise, some reasonable observations regarding possible frameworks for such a venture follow.

First, it is observed that many types of experiments and facilities could be involved in an international strategy. These might include (but would not be limited to) non-nuclear test stands, partially integrated test facilities, point accelerator neutronics facilities, fission test reactors, accelerator-based irradiation damage facilities, scientific feasibility experiments for advanced confinement concepts, plasma burning experiments for established confinement concepts, and fusion engineering research facilities. Many opportunities for international cooperation can be available.

Second, it is observed that the time to develop fusion will be long and that no nation is likely to derive an early economic advantage. Consequently, the philosophical basis for a long-range developmental agreement exists.

Third, three cooperative mechanisms are apparent candidates for international participation:

- Several nations could jointly sponsor the same, shared facilities and experimental programs.
- Individual nations could construct and operate separate but complementary facilities.
- Several nations could jointly sponsor one or more "user facilities" (e.g., a FERF) and maintain their own R&D strategies by conducting separate experimental programs.

References for Chapter 2

1. "TFCX Preconceptual Design Report," Revision 1, Princeton Plasma Physics Laboratory, F-AXXX-8407-006/007 (1984).

2. "International Tokamak Reactor, Phase Two, Part 1," International Atomic Energy Agency, Vienna, Austria, STI/PUB/638 (1983).

3. R. Toschi, et al., "NET as the Next Major Device in Europe," Workshop on Fusion Technology Facilities, Karlsruhe, West Germany (1984).

4. K. Tomabechi, et al., "Concept for the Next Tokamak," Fusion Reactor Design and Technology, Vol. I, International Atomic Energy Agency, Vienna, Austria, STI/PUB/616 (1983).

5. "The Fusion Engineering Device," U.S. Department of Energy, DOE/TIC-11600 (1981).

6. C. D. Henning, B. G. Logan, et al., "Mirror Advanced Reactor Study-Final Report," Lawrence Livermore National Laboratory, UCRL-53480 (1984).

7. M. A. Abdou, et al., "Blanket Comparison and Selection Study-Interim Report," Argonne National Laboratory, ANL/FPP-83-1 (1983).

8. M. A. Abdou, et al., "Machine Operation and Test Program," INTOR/TEST/81-3 in "U.S. Contribution to the International Tokamak Reactor Phase 1 Workshop," USA INTOR/81-1 (1981).

9. "TASKA - Tandem Spiegelmachine Karlsruhe, A Tandem Mirror Fusion Engineering Test Facility," Kernforschungszentrum Karlsruhe/University of Wisconsin, KFK 3311/2, UWFDM-500 (1982).

10. E. K. Opperman, "Fusion Materials Irradiation Test Facility-Experimental Capabilities and Test Matrix," Hanford Engineering Development Laboratory, HEDL-TME 81-45 (1982).

11. T. H. Batzer, et al., "Conceptual Design of a Mirror Reactor for a Fusion Engineering Research Facility (FERF)," Lawrence Livermore National Laboratory, UCRL-51617 (1974).

12. T. K. Fowler and B. G. Logan, "Tandem Mirror Technology Demonstration Facility," Lawrence Livermore National Laboratory, UCID-19193 (1981).

13. J. N. Doggett, et al., "A Tandem Mirror Technology Demonstration Facility," Lawrence Livermore National Laboratory, UCID-19328 (1983).

14. K. I. Thomassen and J. N. Doggett, "A Technology Demonstration Facility," J. Fusion Energy, 3, 109 (1983).

15. J. N. Doggett, B. G. Logan, J. E. Osher, K. I. Thomassen, "A Fusion Technology Demonstration Facility (TDF)," Lawrence Livermore National Laboratory, UCRL-90824 (1984).

16. K. I. Thomassen, J. N. Doggett, B. G. Logan, W. D. Nelson, "An Upgrade to MFTF-B for Fusion Technology," Lawrence Livermore National Laboratory, UCRL-90825 (1984).

17. B. Badger, et al., "TASKA-M, A Low Cost, Near Term Tandem Mirror Device for Fusion Technology Testing," Kernforschungszentrum Karlsruhe/Fusion Power Associates/University of Wisconsin, KfK-3680/PFA-83-7/UWFDM-600 (1983).

18. D. L. Jassby and S. S. Kalsi, Eds., "FED-R, A Fusion Engineering Device Utilizing Resistive Magnets," Fusion Engineering Design Center, ORNL/FEDC-82/1 (1983).

19. W. M. Stacey, et al., "U.S. FED-INTOR Activity Critical Issues," Georgia Institute of Technology, FED-INTOR/TEST/82-2 (1982).

20. Reliability Engineering Handbook, Volume II (Management Manual), NAVAIR 01-1A-32 (1977).

21. M. Abdou, et al., "Engineering Testing," Chapter XII, FED-INTOR/TEST/82-4 (1982).

22. Papers presented at the Tokamak Fusion Core Experiment Design Review, Princeton Plasma Physics Laboratory (1984).

23. L. Bromberg, et al., "A Long Pulse Ignited Test Experiment (LITE)," Nucl. Tech./Fusion, 4, 2, 1013 (1983).

24. K. I. Thomassen and J. N. Doggett, Eds., "Options to Upgrade the Mirror Fusion Test Facility," Lawrence Livermore National Laboratory, UCID-19743 (1983).

25. C. D. Henning, et al., "Fusion Power Demonstration: Baseline Report," Lawerence Livermore National Laboratory, UCID-19975 (1984).

26. M. A. Abdou, et al., "A Demonstration Tokamak Power Plant Study (DEMO)," Argonne National Laboratory, ANL/FPP-82-1 (1982).

27. P. J. Gierszewski, CFFTP/UCLA, private communication (1984).

28. Anonymous Experimenters Guide, Rotating Target Neutron Source-II Facility, Lawrence Livermore National Laboratory, M-094 (1978).

29. D. D. Armstrong, et al., "An Intense Neutron Source Facility," Nucl. Inst. and Methods, 145, 127 (1977).

30. M. S. Wechsler, W. F. Sommer, "Spallation Radiation Damage and the Radiation Damage Facility at the LAMPF A-6 Target Position," J. Nucl. Mat., 122, 1078 (1984).

31. A. L. Trego, et al, "Fusion Materials Irradiation Test Facility for Fusion Materials Qualification," Nucl. Tech./Fusion, 4, 695 (1983).

32. A. Schmepp, et al., "Status of the Frankfurt Zero-Mode Proton RFQ," IEEE Trans. on Nucl. Sci., Vol. NS-30, 4, 3536 (1983).