# A STUDY OF THE ISSUES AND EXPERIMENTS FOR FUSION NUCLEAR TECHNOLOGY

OVERVIEW

BLANKET ENGINEERING

M. A. ABDOU, P. J. GIERSZEWSKI,* M. S. TILLACK, K. TAGHAVI, K. KLEEFELDT,+ G. BELL, H. MADARAME,† and Y. OYAMA‡
*University of California, Los Angeles, School of Engineering and Applied Science*
*Department of Mechanical, Aerospace, and Nuclear Engineering*
*Los Angeles, California 90024*

D. H. BERWALD, J. K. GARNER, and R. WHITLEY
*TRW, Inc., One Space Park, R1/2128, Redondo Beach, California 90278*

J. STRAALSUND, R. BURKE, J. GROVER, E. OPPERMAN, and R. PUIGH *Hanford Engineering Development Laboratory*
*Richland, Washington 99352*

J. W. DAVIS and G. D. MORGAN
*McDonnell Douglas Astronautics Company, P.O. Box 516*
*St. Louis, Missouri 63166*

G. DEIS *EG&G Idaho, P.O. Box 1625, Idaho Falls, Idaho 83401*

M. C. BILLONE *Argonne National Laboratory*
*9700 South Cass Avenue, Argonne, Illinois 60439*

K. I. THOMASSEN *Lawrence Livermore National Laboratory*
*Livermore, California 90278*

D. L. JASSBY *Princeton Plasma Physics Laboratory*
*Princeton, New Jersey 08544*

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*The operating environment to be experienced by the nuclear components of a fusion reactor is unique and leads to a number of new phenomena and effects. New experimental knowledge is necessary to resolve many of fusion's remaining issues. Investigation of the required experiments reveals the importance of simulating multiple interactions among physical elements of components and combined effects of a number of operating environmental conditions. Some experiments require neutrons not only as a source of radiation damage effects but as a practical economical means for bulk heating and producing specific nuclear reactions.*

*The evaluation of required facilities suggests important conclusions. Present fission reactors and accelerator-based neutron sources are useful and their use should be maximized worldwide, but they have serious limitations. Obtaining adequate data for fusion nuclear technology over the next 15 years requires a number of new nonneutron test facilities in addition to the use of fission reactors. Experiments in the fusion environment will then be required for integrated tests and concept verification. The key nuclear needs for a fusion facility are 20 MW of deuterium-tritium fusion neutron power over 10 $m^2$ of experimental surface area with long (<1000 s) plasma burn and 2 to*

*Permanent address: Canadian Fusion Fuels Technology Project.
+Permanent address: Karlsruhe Nuclear Research Center, Federal Republic of Germany.
†Permanent address: University of Tokyo, Japan.
‡Permanent address: Japan Atomic Energy Research Institute.

*10 MW·yr/m² fluence capability. Fusion test devices with fusion power >100 MW are shown to be undesirable because of high cost and high risk. The analysis favors fusion devices that are able to operate at low total power and high power density. For fusion devices with large minimum power, e.g., conventional tokamaks, results indicate strong incentives for two separate test devices: one for plasma physics experiments and the other for fusion engineering research experiments.*

## I. 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 plasma, plasma support components (magnets, vacuum, auxiliary heating), and nuclear components. The primary functions of the nuclear components are

1. fuel generation and processing
2. energy extraction and conversion
3. radiation protection of personnel and components.

The primary nuclear components and other components affected by the nuclear environment are shown on Table I. 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.

The promise of fusion is so great that a comprehensive and accelerated R&D program is necessary to permit a quantitative judgment 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.

1. 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 science, chemistry, nuclear physics, thermodynamics, fluid mechanics, electromagnetics, magnetohydrodynamics (MHD), nuclear engineering, mechanical engineering, and chemical engineering.

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

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

4. 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 plan for fusion development.

Because of the importance of fusion nuclear science and technology, the U.S. Department of Energy/Office of Fusion Energy (DOE/OFE) initiated a new study[1] called FINESSE in November 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 the University of California, Los Angeles and

TABLE I

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 |

TABLE II

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, GmbH |
| University of Kyoto, Japan |
| University of Tokyo, Japan |

involves major organizations within the United States, and from Canada, Japan, and Federal Republic of Germany (FRG) (Table II). This international participation is particularly important for the following reasons:

1. All world fusion programs face the same issues. Therefore, all countries can benefit from investigating issues and approaches to fusion nuclear development.
2. 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.

### I.A. Overview

This paper contains three main parts: a description of the issues and experimental needs of fusion nuclear technology, a survey and evaluation of facilities for performing the needed experiments, and finally, an examination of possible scenarios for fusion R&D.

In Sec. II, the key issues are identified with special emphasis on the most critical feasibility issues. The focus is on those issues whose resolution requires new knowledge through experiments.

In Sec. III, the testing needed to resolve these key fusion nuclear issues is surveyed. The word "test" is used in this paper 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. The survey of testing needs 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 is under development.

In addition to identifying the testing needs, it is important to develop quantitative testing requirements, particularly for multiple interaction and integrated tests. Realistic cost constraints on testing facilities, including fusion devices, dictate that integrated 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, is discussed in Sec. IV. This work is of critical importance to the many trade-offs between cost and benefit that must be considered with respect to the large number of testing needs and test facility options.

The problems of engineering scaling are complex and require a great deal of analysis to deepen our understanding of the testing issues. Although we are concerned with and have addressed the key issues for all fusion nuclear components, at this time the detailed quantification of the testing requirements has been attempted only for the blanket system. Some of this analysis is summarized in Secs. IV.B and IV.C for liquid-metal and solid breeder blankets, respectively. Experiments aimed specifically at verification of neutronics methods and data have special scaling issues, which are treated in Sec. IV.D.

Fluence goals represent an important aspect of test requirements that are particularly difficult to quantify but have a substantial impact on the cost of testing. The amount we learn from testing as a function of neutron fluence is investigated in Sec. IV.E as a basis for addressing fluence goals.

Using insight from the investigation of the fusion nuclear issues and testing needs, it is possible to evaluate facilities in which these tests can be performed. Both fusion and nonfusion facilities have been considered. Section V provides an initial examination of the capabilities and limitations of nonfusion facilities, including nonneutron test stands, accelerator-based neutron sources, and fission reactors.

A key conclusion of the first year of the FINESSE effort is that testing in nonfusion facilities, while essential, is not sufficient to satisfy all 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 presented in Sec. VI, with an initial examination of tandem mirrors and tokamaks in Secs. VI.B and VI.C, 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 high-power fusion test facilities that consume a large amount of tritium and require their own tritium breeding blankets. The testing device availability analysis is given in Sec. VI.D.

Broader and deeper examination of the complex cost/benefit/risk issues in fusion nuclear technology is planned for the future effort of the FINESSE study. We will continue to provide measures of benefits (usefulness of test information) as functions of testing capabilities, and to quantify the costs and physics/technology risks of testing facilities as functions of testing capabilities. This information will be utilized 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. This testing includes a complete spectrum of physical experiments and measurements, i.e., basic property measurements, single- and multiple-effect experiments, and integrated experiments. During the first year of effort, the *total* testing requirements for fusion nuclear technology have been addressed and possible options for nonfusion and fusion facilities evaluated. To accomplish the above objective of recommending where the various *parts* of the testing requirements can be optimally performed, it is necessary to explicitly consider possible scenarios, or pathways, for overall fusion R&D. A preliminary screening evaluation of a number of pathways was performed and is summarized in Sec. VII. This work utilized many of the results and concepts reported in earlier studies.[2-8]

Overall conclusions emphasizing future technical direction for fusion nuclear technology are summarized in Sec. VIII. Limitations on space mandate that the technical details in this paper be brief in many cases. Supporting details are contained in Ref. 1.

## II. FUSION NUCLEAR ISSUES

### II.A. Introduction

A coherent program of engineering testing and nuclear technology development must address the key issues that 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 was generated by considering the behaviors of the nuclear components involving materials science, structural mechanics, failure modes, thermal hydraulics, MHD, tritium recovery, systems integration, and many other disciplines. In the FINESSE interim report,[1] the issues are described in brief summaries that characterize the issues, their relative importance, and general requirements for testing. In addition to the known issues, it is acknowledged that unforeseen effects are likely to be observed throughout the test program. These additional issues provide added importance to thorough, reactor-relevant testing.

Generic examples of components were needed to focus the effort to identify the issues. The blanket options were limited to liquid-metal (lithium and Li-Pb) 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. They need to be considered, however, in determining near-term experimental programs.

### II.B. Issue Descriptions

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

1. Feasibility issues
   - a. may close the design window
   - b. may result in unacceptable safety risk
   - c. may result in unacceptable reliability, availability, or lifetime.
2. Attractiveness issues
   - a. reduced system performance
   - b. reduced component lifetime
   - c. increased system cost
   - d. 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 that 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 can be determined for the overall level of concern. The most critical issues have the greatest need for testing; the effort to quantify test requirements and choose test devices concentrates on these issues and how well they can be resolved.

The complete list of issues contains ~120 specific technical items, as listed in Table III. An effort was made to keep a somewhat uniform level of detail in the definition of an issue, thereby allowing a meaningful comparison among the different reactor components. The blanket encompasses approximately half of the testing issues. Plasma interactive components

TABLE III

List of Fusion Nuclear Testing Issues

(An asterisk identifies critical issues.)

I. Blanket/first-wall issues

A. Structure
- 1.* Changes in properties and behavior of materials
- 2. Deformation and/or breach of components
  - a.* Effect of first-wall heat flux and cycling on fatigue or crack growth-related failure
  - b. Magnetic forces within the structure (including disruptions)
  - c. Premature failure at welds and discontinuities
  - d. Failures due to hot spots
  - e. Interaction of primary and secondary stresses and deformation
  - f. Effect of swelling, creep, and thermal gradients on stress concentrations (e.g., in grooved surfaces)
  - g. Failure due to shutdown residual stress effects
  - h. Interaction between surface effects and first-wall failures
  - i. Self-welding of similar and dissimilar metals
- 3. Tritium permeation through the structure
  - a.* Effectiveness of tritium permeation barriers
  - b. Effect of radiation on tritium permeation
- 4. Structural activation product inventory

B. Coolant
- 1.* MHD pressure drop and pressure stresses
- 2.* MHD and geometric effects on flow distributions
- 3. Coolant flow stability
- 4. Stability/kinetics of tritium oxidation in the coolant
- 5. Helium bubble formation leading to hot spots
- 6. Salt coolant stability and decomposition
- 7. Coolant/purge stream containment and leakage

C. Breeder and purge
- 1. Tritium recovery and inventory in solid breeder materials
  - a.* Intragranular tritium diffusivity and solubility
  - b. Tritium surface migration and desorption
  - c. Porosity, purge flow distribution, and tritium transport
- 2. Liquid breeder tritium extraction
- 3. Temperature limits and variability in solid breeder materials
  - a.* Temperature limits
  - b. Thermal conductivity changes under irradiation
  - c. Effect of cracking
  - d. Effect of LiOT mass transfer
- 4. Tritium release form from solid breeder

D. Coolant/structure interactions
- 1. Mechanical and materials interactions
  - a.* Corrosion mass transport rates and consequences
  - b. Mechanical wear and fatigue from flow-induced vibrations
  - c. Failure of coolant wall due to stress corrosion cracking
  - d. Failure of coolant wall due to liquid-metal embrittlement
- 2. Thermal interactions
  - a.* MHD effects on first-wall cooling and hot spots
  - b. Response to cooling system transients
  - c. Flow sensitivity to dimensional changes

E. Solid breeder/multiplier/structure interactions
- 1. Solid breeder mechanical and materials interactions
  - a.* Clad corrosion from breeder burnup products
  - b.* Strain accommodation by creep and plastic flow
  - c.* Swelling driving force
  - d. Stress concentrations at cracks and discontinuities
  - e. Thermal expansion driving force
- 2. Neutron multiplier mechanical interactions
  - a. Swelling driving force in beryllium
  - b. Strain accommodation by creep in beryllium
  - c. Mechanical integrity of unclad beryllium
- 3. Thermal interactions
  - a.* Breeder/structure interface heat transfer (gap conductance)

F. General blanket
- 1. D-T fuel self-sufficiency
  - a.* Uncertainties in achievable breeding ratio
  - b.* Uncertainties in required breeding ratio
- 2. Tritium permeation
  - a. Permeation from breeder to blanket coolant
  - b. Permeation from beryllium to coolant
  - c. Permeation characteristics at low pressure
- 3. Chemical reactions
- 4. Tritium inventory behavior during transients
- 5. Uncertainties in failure modes and frequencies
- 6. Nuclear heating rate predictions
- 7. Time constant of magnetic field penetration for plasma control
- 8. Blanket response to near blanket failures
- 9. Assembly and fabrication of blankets
- 10. Recycling of irradiated lithium and beryllium
- 11. Prediction and control of normal effluents associated with fluid radioactivity
- 12. Liquid-metal blanket insulator fabrication, effectiveness, and lifetime

(Continued)

TABLE III (Continued)

II. Plasma interactive components

A. General concerns
1. Materials data base development
2. HHFC surface damage mechanisms
   a.* Physical erosion and redeposition
   b. Arcing and related erosion
   c. Chemical erosion
   d. Surface damage due to helium implantation (blistering)
   e. Disruption-induced surface melting and erosion
3. HHFC thermomechanical response
   a.* Thermal-hydraulic techniques
   b. Leading edge design
   c. HHFC structural integrity
   d. Heat sink bond fabrication and failure
   e. First-wall hot spots due to plasma spatial distribution
   f. Disruption loads
4. Plasma edge conditions and exhaust
   a.* Plasma edge temperature and density control
   b. Helium and impurity exhaust
   c. Plasma exhaust stream pressure and composition
   d. Surface conditioning effectiveness
5. Safety
   a.* Tritium permeation and inventory
   b. Tritium inventory behavior during maintenance
   c. Eroded activation product behavior in the vacuum chamber

B. Limiter/divertor specific issues
1. Maintenance and replacement
2. Choice of limiter versus divertor
3. Alignment

C. Vacuum systems
1. Compound cryopump helium pumping and regeneration lifetime
2. Vacuum chamber outgassing and leak rates
3. Large-diameter vacuum valve reliability
4. Vacuum pump operation under thermal/pressure transients

D. rf components (auxiliary heating systems)
1.* rf launcher performance requirements
2.* Window and feedthrough performance
3. rf transmission system performance requirements

III. Shield
1. Shield effectiveness
   a. Protection of sensitive components
   b. Biological dose during operation and maintenance
   c. Analytical techniques and data base
2. Shield compatibility with blanket including assembly and disassembly
3. Time constant of magnetic field penetration for plasma control
4. Shield compatibility with the vacuum boundary and one turn resistance

IV. Tritium processing system
1. Impurity removal in the fuel cleanup process
2. Tritium monitoring and accountability
3. Tritium losses in solid waste
4. Tritium extraction from water coolant
5. Tritium processing system integration
6. Atmospheric cleanup process
7. Breeder tritium extraction stream characteristics

V. Magnets
1. Structural overloading and quenching due to plasma disruptions
2. Internal cooling requirements and cryostability
3. Maximum levels of radiation damage and recovery process for magnet components
   a. Critical current reduction
   b. Electrical resistivity of the stabilizer
   c. Insulator structural degradation
   d. Annealing of radiation damage
   e. Consequences of magnet failures

VI. Instrumentation and control
1. Definition of transducer lifetimes and hardening requirements
2. Breakdown of insulation resistance
3. Decalibration of transducer through transmutation
4. Ceramic insulator/substrate seal integrity
5. rf transmission losses in horns, antennae, wave guides, and windows
6. Optical window, lens, and prism darkening and distortion
7. Shielding of instrument penetrations
8. Radiation effects on electrical components beyond the first wall and shield
9. Cable noise from rf, magnetic fields, charged particles, etc.

(PICs) account for another 25%, and the other non-blanket components share the remainder. Critical issues, defined as those issues that have an impact on component feasibility for a large class of designs, are identified in Table III.

### II.C. Critical Issues Summary

A complementary summary of the critical issues of fusion nuclear technology was compiled that stresses the key functional aspects of the fusion reactor that must be resolved through testing. These are listed in Table IV and discussed in more detail below.

1. *Deuterium-tritium (D-T) 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 (TBR) 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 that 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, radiation swelling) and the structural responses (e.g., interaction of primary and secondary stresses, influence of 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.

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, 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.

TABLE IV

Critical Fusion Nuclear Technology Development Issues

| |
|---|
| 1. D-T 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 I&C |

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 mass transfer 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 the 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 (I&C).* Failure of I&C 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. Because of the added effects of all the environmental conditions present in a fusion reactor (e.g., magnetic field), however, I&C is considered separately. The I&C components often contain materials that 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.

## III. SURVEY OF EXPERIMENTAL NEEDS

### III.A. Introduction

The development of fusion to the commercial reactor stage requires 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 perform the 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 that 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 that 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 testing needs are distinguished by the relevant component and by the level of integration of the test. Different components generally have different functions, different operating conditions, and thus different testing needs. Specific components considered include the blanket, PICs [e.g., first wall, limiter, divertor, radio-frequency (rf) antennae], shield, tritium processing system, magnets, I&C and balance of plant, as well as interactions among these components. For each component there is a set of tests ranging from property measurements to component verification. The level of integration also provides a rough measure of test complexity and a loose chronological ordering since the simpler tests are generally performed first.

The test categories adopted here are: basic, single effect, multiple effect/multiple interaction, partially integrated, integrated, and component tests. Table V summarizes the descriptions of these categories. Component tests were not examined in detail because they represent a more advanced stage of development than is presently being considered.

TABLE V

Test Categories for Single Component R&D

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: (a) pellet-in-can test of the thermal stress/creep interaction between solid breeder and clad; (b) electromagnetic response of bonded materials to a transient magnetic field; (c) 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: (a) an isolated blanket module with its own cooling system in a fusion test reactor; (b) a complete integrated blanket in a demonstration power reactor

### III.B. Testing Needs Summary

Each identified testing need is characterized by

1. importance of neutrons
2. importance of fusion neutron energy spectrum
3. typical test article size
4. number of test articles (*excluding* duplicate tests for statistical purposes, off-normal conditions, data at several time intervals for high fluence tests, etc.).

More detailed characterization can be found in Ref. 1, including the required environmental conditions and the usefulness and limitations of nonneutron test stands, point neutron sources, and fission reactors as test facilities.

A total of 74 testing needs were identified,[1] with 45% blanket, 20% PICs, and 35% for the remainder of the components and component interactions. Three specific tokamak testing needs were identified relating to PICs, 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 that require high-energy or fusion neutrons, since this is directly related to the question of the ability of nonfusion 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 VI, along with estimates of test article size and number.

### III.C. Test Plan Considerations

The next step in developing a test program is to define the experiments surveyed here in enough detail to formulate a test plan that includes facility requirements, cost, and schedule. Previous studies have

TABLE VI

Fusion Nuclear Technology Tests Requiring Fusion Neutrons

| Tests | Typical Test Article Size (cm) | Number of Test Articles[a] |
|---|---|---|
| Basic tests | | |
| Structural material irradiated properties | 1 × 1 × 2 | 20 000 |
| Solid breeder irradiated properties | 1 × 1 × 2 | 1 200 |
| Plasma interactive materials irradiated properties | 1 × 1 × 5 | 900 |
| Radiation damage indicator cross sections | 1 × 1 × 0.5 | 500 |
| Long-lived isotope activation cross sections | 1 × 1 × 0.1 | 200 |
| Neutron sputtering rate cross sections | 1 × 1 × 0.1 | 30 |
| Single-effect tests | | |
| Structure thermomechanical response experiments | 10 × 10 × 10 | 50 |
| Weld behavior experiments | 10 × 10 × 5 | 50 |
| Shield effectiveness in complex geometries | 50 × 50 × 100 | 50 |
| Optical component radiation effects | 2 × 2 × 2 | 20 |
| Multiple-effect/multiple interaction tests | | |
| Submodule thermal and corrosion verification | LB[b]: 100 × 100 × 30 | 5 |
| | SB[b]: 10 × 50 × 30 | 5 |
| Partially integrated and integrated tests | | |
| Verification of neutronic predictions | 50 × 50 × 100 | 4 |
| Tritium breeding, nuclear heating during operation, and induced activation | | |
| Full module verification | LB[c]: 100 × 100 × 50 | 5 |
| Thermal and corrosion | SB: 100 × 100 × 50 | 5 |
| Module thermochemical lifetime | | |
| Tritium recovery | | |
| Instrumentation transducer lifetime | 1 × 1 × 2 | 70 |
| Insulator/substrate seal integrity | 1 × 1 × 2 | 20 |
| Biological dose rate profile verification | D-T device | 1 |
| Afterheat profile verification | D-T device | 1 |
| Component tests | | |
| Blanket performance and lifetime verification | SB: 30 × 100 × 80 | 3 |
| | LB: 900 × 300 × 80 | 3 |
| Radiation effects on electronic components | 1 × 1 × 1 | 20 |
| Instrumentation performance and lifetime | 5 × 5 × 5 | 100 |

[a]A test article is 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 *not* included in the number of test articles.
[b]LB = liquid breeder blankets; SB = solid breeder blankets.
[c]Some designs require a larger test volume.

defined test matrices that specify the number, type, conditions, and size of specimens needed for structural and breeder materials testing,[1,3,9,10] but the more complex tests also indicated in this survey were not quantified. Such tests are obviously important, and Sec. IV focuses on their requirements.

One preliminary requirement that can be estimated from the information in this survey is the overall irradiation testing area (first-wall area) and volume. Based on Table VI for tests requiring significant fusion (or at least high-energy) neutrons, the irradiation testing area and volume are listed in Table VII. The space requirements are not needed in a given reactor at a given time, but rather represent the overall space integrated over the test program duration. While tentative, these numbers point to the need for a considerable amount of irradiation testing space for fusion R&D.

## IV. EXPERIMENT REQUIREMENTS

### IV.A. Introduction

In Secs. II and III, the issues were identified and the testing needs to resolve these issues were surveyed.

TABLE VII

Preliminary Summary of Test Area ($m^2$) and Volume ($m^3$) Needs for Tests Requiring Fusion or High-Energy Neutrons*

| Test | Area | Volume |
|---|---|---|
| Blanket | 10 to 20[a] | 5 to 8[a] |
| PICs | 1 to 2 | <1 |
| Shield | 10[b] | 10 |
| Other | <1 | <1 |

*This table includes only those areas and volumes that have been quantitatively defined to date and does *not* include duplicates for reasons such as statistics.
[a]*Not* all these tests have to be performed simultaneously. Variation relates to number of blanket concepts tested.
[b]Does not need exposure to the plasma.

Translating the testing needs into a realistic R&D plan that describes experiments, facilities, schedule, and cost requires trade-offs between two factors: (a) benefits of tests as functions of the environmental conditions provided in the experimental facilities, and (b) capabilities, limitations, costs, and risks of various options for experimental facilities. This section examines the test benefits.

In general, there is a range of needed experiments for both blanket and nonblanket components, from simple property and single-effect tests to more complex interactive and integrated tests. The simpler experiments require a limited set of environmental or "state" conditions to be similar to those in an operating component—average temperature and neutron fluence, for example, to measure breeder thermal conductivity. These tests have been characterized in several recent studies.[1,11,12] The more integrated tests have also been surveyed,[1] but raise the important question of how close the experiment should simulate operating conditions. For example, does testing a breeder module at a power density of 10 $MW/m^3$ provide a reasonable simulation of reactor operation at 60 $MW/m^3$? Can the experiment be scaled such that it does?

This section is primarily concerned with developing engineering scaling relationships and quantifying the test requirements for highly interactive tests. Most of the effort focuses on blankets since they have the majority of unresolved issues. The analyses are generally performed in the context of a fusion device as a desirable test facility; however, the results are more universally applicable. For example, an understanding of the first-wall thermal-hydraulic behavior under scaled conditions also applies to other high heat flux components (HHFCs) such as limiters. Furthermore, an identified requirement on surface heat flux should apply regardless of the particular test facility. Future work will explicitly consider nonblanket components and experimental requirements for nonfusion test facilities.

The device parameters with the most influence on component operation are summarized in Table VIII. For highly interactive experiments that require most of these parameters (and associated effects) to be present, it is nearly certain that the parameters in the test device will not match those of a full-scale fusion reactor because of cost constraints. This could restrict the test's ability to resolve the key nuclear testing issues.

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 continuously 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.

To understand and quantify the scaling laws and test requirements, analyses of technical aspects of blanket operation were performed, including MHD, thermal hydraulics, tritium recovery, structural mechanics, neutronics, and materials compatibility. These

TABLE VIII

Device Parameters with Major Influence on Operation and Testing

| |
|---|
| Neutron radiation |
| Fluence |
| Spectrum |
| Heat source |
| Surface |
| Nuclear (power density) |
| Operating time |
| Burn/dwell time |
| Continuous operating time |
| Magnetic field |
| Intensity |
| Geometry |
| Test port surface area |
| Test port depth |

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 the analysis, four specific reference blankets were chosen; they are the Mirror Advanced Reactor Study[6] (MARS) self-cooled Li-Pb/HT-9 tandem mirror reactor (TMR) blanket and three designs representative of blankets considered within the Blanket Comparison and Selection Study[7] (BCSS): a self-cooled Li/V toroidal/poloidal flow design, a helium-cooled $Li_2O$/HT-9 design, and a water-cooled $LiAlO_2$/primary candidate alloy (PCA) design. These blankets were considered not because they necessarily represent the best possible designs for all possible reactor concepts; 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.

Sections IV.B and IV.C address test requirements and engineering scaling for liquid-metal and solid breeder blankets. Experiments aimed specifically at neutronics information (e.g., tritium breeding, nuclear heating) have their own distinct requirements and are discussed in Sec. IV.D. The blanket fluence requirements are considered separately in Sec. IV.E.

### IV.B. Liquid-Metal Blanket Experiment Requirements

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

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 stress distribution, which stems 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 in a test module 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 1 illustrates this behavior. For a typical heated channel perpendicular to the magnetic field, it shows that along the length of the channel the heat transfer coefficient drops, the corrosion rate increases after an initial saturation, pressures decrease due to MHD effects, and temperatures

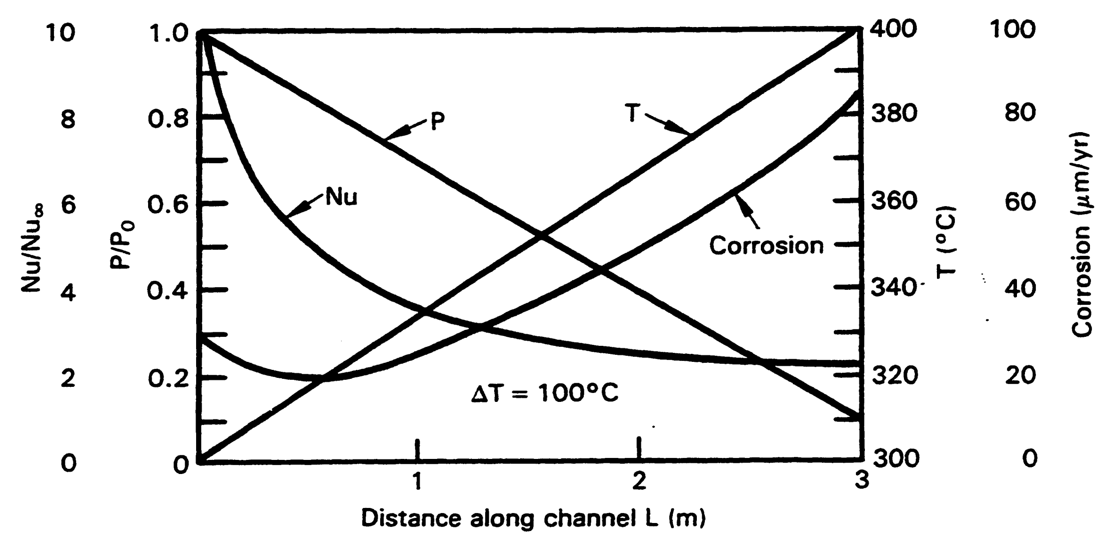



Fig. 1. Illustration of the variation of heat transfer (*Nu*), corrosion, and pressure stresses along a heated channel perpendicular to the magnetic field.

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 thermomechanical state of the blanket. In some blankets, for example, the BCSS reference blanket, this may not be an insurmountable problem. In the BCSS toroidal/poloidal flow design, the local stresses are dominated by the local loading conditions; therefore, 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 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 demonstrates this effect using an I-beam section from the first wall (shown in Fig. 3). Three cases are presented. The first case shows the temperature and stress distributions for the reference case, which models the actual blanket operating conditions. In case 2, the bulk heating is turned off, but the second-wall temperatures are maintained by controlling the coolant temperature. In this case, the stresses are maintained nearly identical. In case 3, 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) may be simulated well without bulk heating. Other calculations demonstrate that the dimensions of the structure can be changed without affecting the thermal and pressure stress distributions, provided the structure aspect ratios (ratios of dimensions) and the temperature distribution are maintained unchanged.

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 Fig. 4, irradiation creep and a small amount of swelling [0.01%/displacements per atom (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 (EOL) stresses are much larger than at the beginning of life (BOL). This also illustrates how important a good materials properties data base is, and how difficult engineering scaling can be without it.

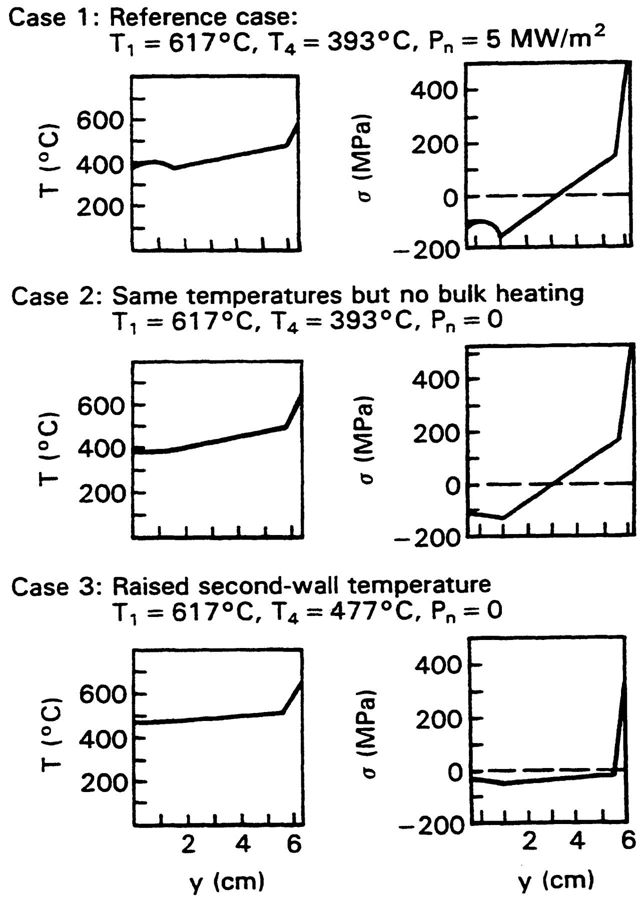



Fig. 2. Elastic stress matching using the I-beam model.

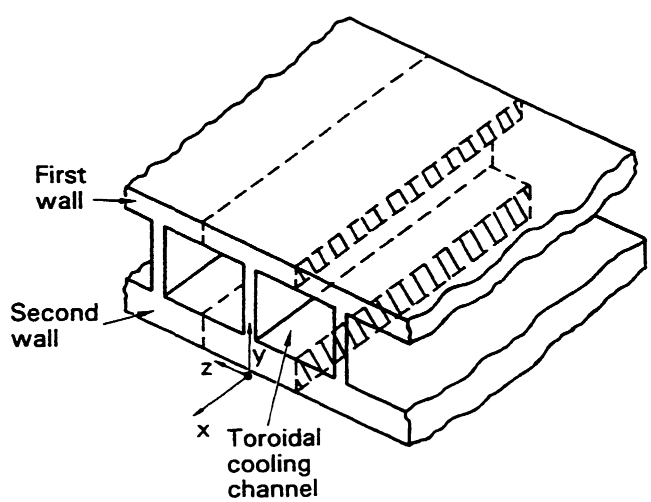



Fig. 3. I-beam model of the composite first-wall structure of the reference design.

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

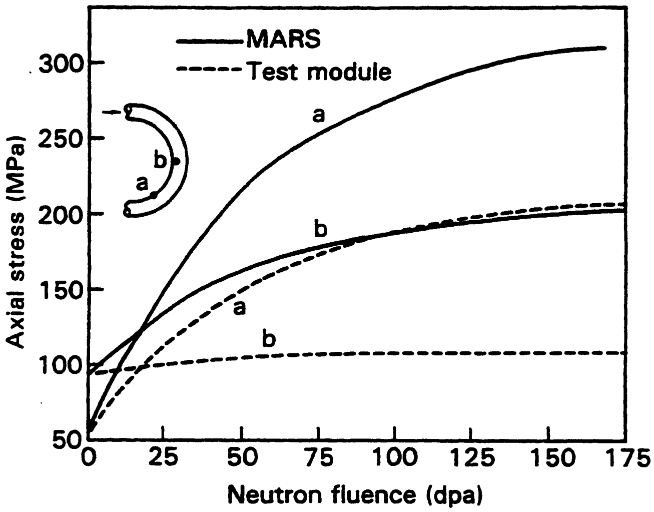



Fig. 4. MARS blanket and test module stresses with swelling and creep included: (a) location of maximum EOL stress and (b) location of maximum BOL stress.

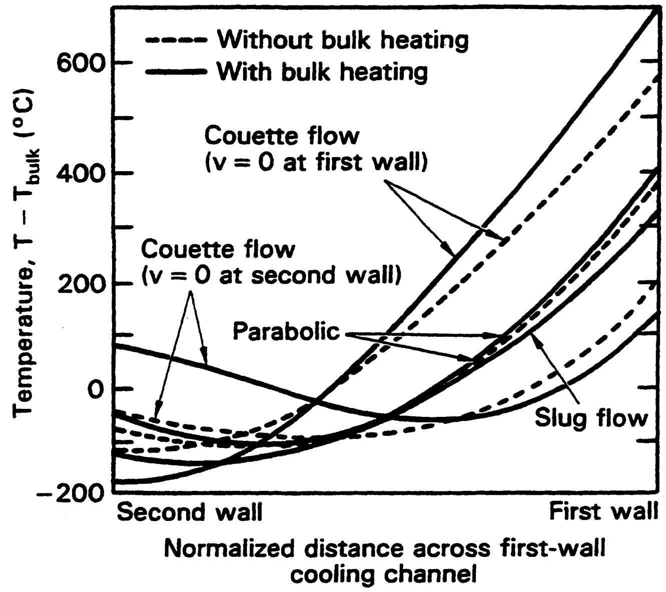



Fig. 5. Dependence of coolant temperature profile, first- and second-wall temperatures on the coolant velocity profile and bulk heating (temperatures are referenced to the bulk coolant temperature $T_b$).

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 TMRs ($q \sim 0.1$ versus $\sim 0.5$ MW/m$^2$); second, the level of coolant temperature control available in the BCSS design is not present in the MARS blanket.

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 5 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 in the toroidal first-wall cooling channels 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, particularly for flow parallel to the magnetic field. In the reference design, global eddy currents affect the velocity distributions, so modeling of the entire blanket is important. This requirement is in contradiction with the unit cell approach. It is believed, however, 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 blanket 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 is expected to 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.

The radiation damage profiles are difficult to scale because the neutron mean-free-path is relatively independent of geometry. The impact of altered damage profiles on testing is illustrated in Fig. 4, which compares the MARS response due to irradiation swelling with the response of a smaller test module. The responses are quite different; 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.

Irradiation creep also has a strong impact on the structural responses. Results of irradiation creep studies indicate the following:

1. As shown in Fig. 6, irradiation creep relaxes thermal stresses over a period of a few months (5 to 10 dpa).

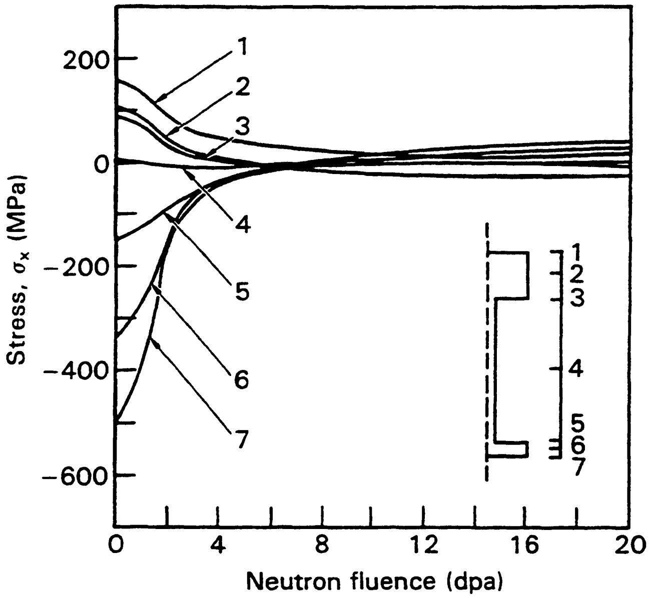



Fig. 6. Stress history at several locations in BCSS lithium self-cooled tokamak first wall due to irradiation creep.

2. The rate of deformation in the coolant channels due to irradiation creep driven by primary (pressure) stresses appears to be constant.

3. As with swelling, preserving aspect ratios may be a feasible method to retain act-alike creep behavior if the damage gradient effect can be overcome.

In addition to radiation damage effects, failure modes may depend on both the absolute dimensions of the structure and the aspect ratios. Failure modes are an important concern, owing to the complex and severe environment in which the first wall/blanket operates, including high temperatures, large thermal gradients, pressures, corrosive materials, radiation effects, and the influence of cycling. The most likely failure modes are given in Table IX. These are ranked according to the likelihood of occurrence for four reactor types and four blanket concepts.

Corrosion mass transfer has also 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 Fig. 7, 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. To observe the initial corrosion rate in the channel entrance, the saturation as corrosion product builds up in the coolant, and the final dependence on the temperature gradient, a minimum channel length is necessary. 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.

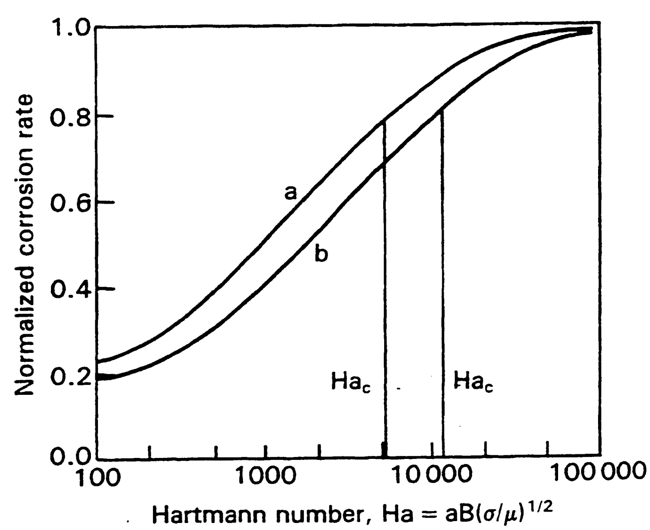



Fig. 7. Dependence of corrosion rate on magnetic field due to boundary layer thinning, using mass diffusion coefficients given by (a) classical Stokes-Einstein theory and (b) an empirical expression given by Olander. The corrosion rate is normalized by the value at Ha = ∞.

TABLE IX

Likelihood of Occurrence for Various Failure Modes*

| First-Wall/Blanket Failure Modes[a] | Steady State | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| | Tokamak | | | | TMR | | | |
| | A | B | C | D | A | B | C | D |
| Cracking around a discontinuity/weld | $H^1$ | $H^2$ | $H^1$ | $H^1$ | $H^1$ | $H^2$ | $H^1$ | $H^1$ |
| Crack on shutdown (with cooling) | $H^2$ | L | $H^3$ | $H^2$ | $H^2$ | L | $H^3$ | $H^2$ |
| Breeder disintegrates/cracks | $H^3$ | $H^3$ | NA | NA | $H^3$ | $H^3$ | NA | NA |
| First-wall/breeder/structure swelling and creep leading to excessive deformation or first-wall/coolant tube failure | $H^1$ | $H^2$ | $H^3$ | $H^3$ | M-H | M-H | M-H | M-H |
| Crack during operation (first wall/breeder/structure) | M | $H^1$[b] | M | M | M | $H^1$ | M | M |
| Environmentally assisted cracking | M | $H^1$[c] | $H^1$ | $H^1$ | M | $H^1$ | $H^1$ | $H^1$ |
| Crack on startup (first wall/breeder/structure) | $H^2$ | L | $H^2$ | $H^2$ | $H^2$ | L | $H^2$ | $H^2$ |
| Excessive tritium permeation of coolant tubes | M | M | NA | NA | M | M | NA | NA |
| First-wall/breeder/structure melting | L | L | L | L | L | M | L | L |
| Manifold tube breaks | L | L | L | L | L | L | L | M[d] |
| Insufficient tritium diffusion through breeder | L | L | NA | NA | L | L | NA | NA |

| First-Wall/Blanket Failure Modes[a] | Pulsed Toroidal | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| | Moderate Wall Load | | | | High-Wall Load | | | |
| | A | B | C | D | A | B | C | D |
| Cracking around a discontinuity/weld | $H^1$ | $H^1$ | $H^1$ | $H^1$ | $H^1$ | $H^1$ | $H^1$ | $H^1$ |
| Crack on shutdown (with cooling) | $H^2$ | $H^2$ | $H^2$ | $H^2$ | $H^1$ | $H^1$ | $H^1$ | $H^1$ |
| Breeder disintegrates/cracks | $H^4$ | $H^4$ | NA | NA | $H^3$ | $H^3$ | NA | NA |
| First-wall/breeder/structure swelling and creep leading to excessive deformation or first-wall/coolant tube failure | M | M | M | M | M | M | M | M |
| Crack during operation (first wall/breeder/structure) | $H^3$ | $H^3$ | $H^3$ | $H^3$ | $H^2$ | $H^2$ | $H^2$ | $H^2$ |
| Environmentally assisted cracking | M | $H^2$ | $H^2$ | $H^2$ | M | $H^2$ | $H^2$ | $H^2$ |
| Crack on startup (first wall/breeder/structure) | $H^2$ | L | $H^2$ | $H^2$ | $H^1$ | L | $H^1$ | $H^1$ |
| Excessive tritium permeation of coolant tubes | M | M | NA | NA | M | M | NA | NA |
| First-wall/breeder/structure melting | L | M | L | L | L | M | L | L |
| Manifold tube breaks | L | L | L | L | L | L | L | L |
| Insufficient tritium diffusion through breeder | L | L | NA | NA | L | L | NA | NA |

*In this table, A = $Li_2O$/He/HT-9; B = $H_2O/LiAlO_2$/PCA; C = Li/V self-cooled; D = Li-Pb/HT-9 self-cooled; H = highest likelihood of failure; M = medium likelihood of failure; L = lowest likelihood of failure; NA = not applicable failure mode for that blanket concept. The superscript numbers indicate relative ranking of values of H by column.

[a]Failure modes ranked with most likely first.

[b]High pressure.

[c]Water corrosion.

[d]Based on MARS constrained header design.

It has been shown that, by reducing the coolant velocity in proportion to the decrease in channel length, the same corrosion rate profiles can be obtained. This conclusion assumes that the controlling mechanism is diffusion through the coolant and not transport through the structure, which can be invalid at very high velocities.

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 modeling the effects of the cooling system, including the heat exchanger, it is suggested that the ratios of surface areas of the hot and cold components in the loop be maintained.

### IV.C. Solid Breeder Blanket Experiment Requirements

The primary integrated testing issues for solid breeder blankets are (a) thermomechanical performance and failure modes, and (b) 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. Some failure modes are indicated in Table IX.

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. The temperature profile along the perimeter of high-pressure retaining lobed first walls has a different dependence on the surface and volumetric heating, however, so it is not possible to arbitrarily change the heat source. In particular, both surface and volumetric heating must be changed such that their relative contribution to the first-wall temperature rise is maintained. This leads to the constraints indicated in Fig. 8, where tokamak first walls (dominated by surface heating) have a minimum and TMR first walls (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. 9. If the width is increased proportionally to the first-wall thickness, then the elastic stresses are preserved ($M = 1$). If there is limited test volume available, however, 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 4 (roughly a factor of 5 reduction in heat source for a tokamak first wall) leads to a factor of 2 variation in flux or fluence across a 1-cm composite first wall. This can 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

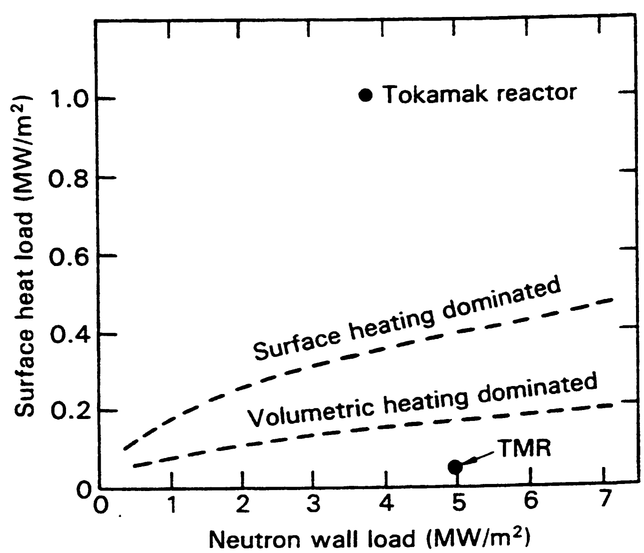



Fig. 8. Heat source requirements for preserving first-wall temperature profile. The boundaries indicate transition between surface heating and volumetric heating dominated first-wall temperatures.

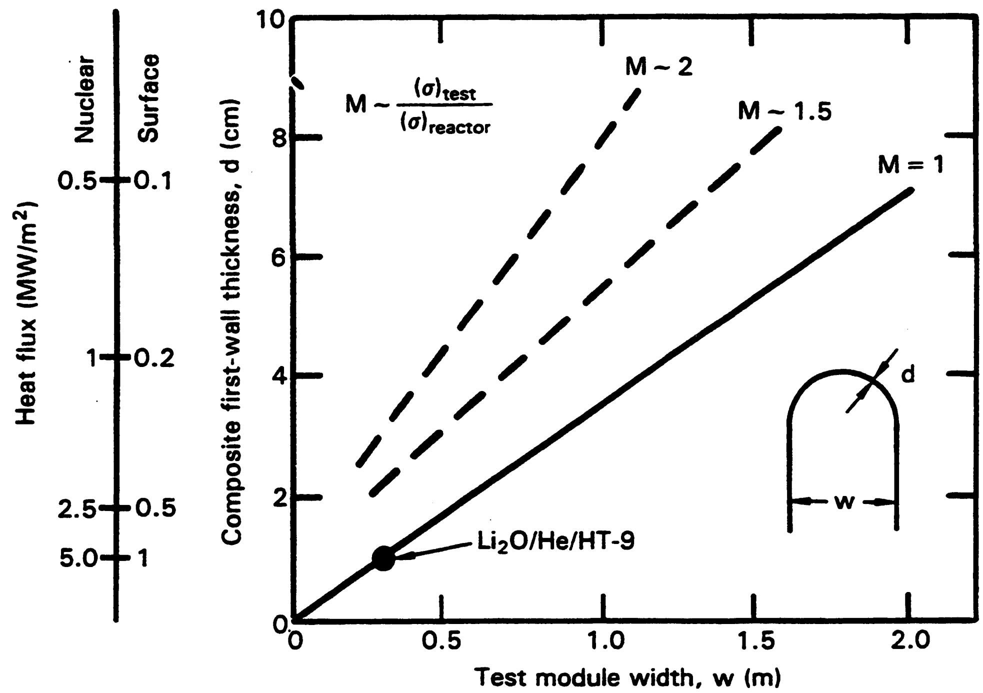



Fig. 9. Test module width and device heat source trade-off for preserving $Li_2O$/He/HT-9 tokamak first-wall thermal plus pressure stresses. Here, *M* is a qualitative measure of the (multiplicative) change in first-wall stress profile.

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. 10 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 (Fig. 11). Present calculations that assume 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 ~1 min, independent of the neutron wall load. To reach inventory equilibrium, however, total operating times of minutes, days, and months are needed for $Li_2O$, hot $LiAlO_2$ (>510°C), and cold $LiAlO_2$ (>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.

Other scaled test considerations suggest that an axial length of 0.2 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 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 field is ~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 0.5 $m^3$ 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.

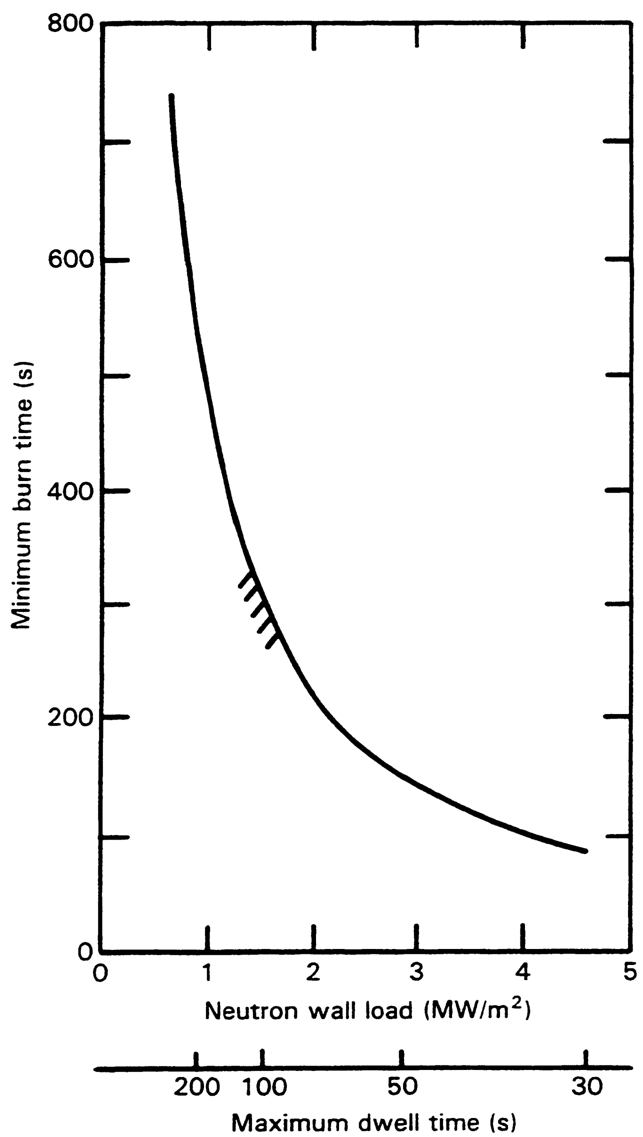



Fig. 10. Relation between minimum burn time, maximum dwell time, and neutron wall load for breeder thermal equilibrium in the $Li_2O$/He/HT-9 test module. Breeder dimensions are changed to keep the breeder within the reactor temperature limits.

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.

### IV.D. Special Requirements for Blanket Neutronics Experiments

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

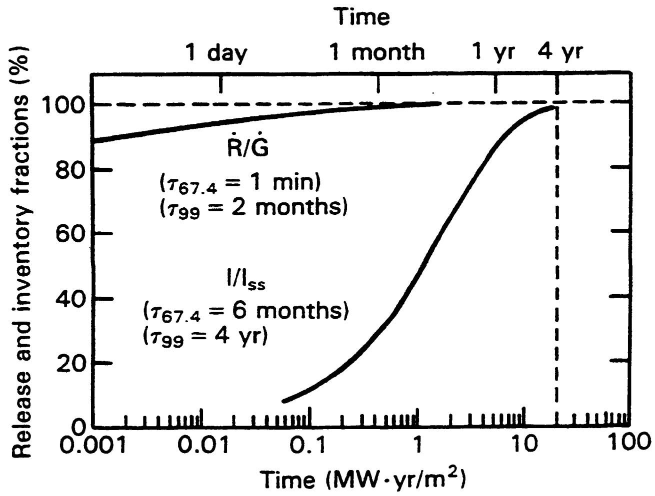



Fig. 11. Tritium release and inventory behavior after startup in the reference $LiAlO_2/H_2O$/PCA/Be blanket. Test times to achieve substantial release rates (67% of equilibrium value) are relatively short, but equilibrium inventory requires many months due to the low diffusivity of $LiAlO_2$.

which look-alike test modules are least useful under scaled-down conditions, neutronics verification tests require that the test module be as close to a look-alike as possible. Therefore, neutronics tests have been treated separately from other types of tests. In blanket tests other than neutronics tests (e.g., thermomechanical, tritium recovery), the simulation of bulk heating, tritium production, etc., for act-alike behavior involves neutronics considerations that are different from those aimed specifically at neutronics verification.

Neutronics testing in a fusion test device involves several types of measurements such as source neutron yield, tritium production rate (TPR), neutron and gamma-ray spectra, heating rates during operation, activation, and afterheat. The requirements for neutronics testing fall within two categories: operating conditions and geometry constraints. 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.

From instrumentation considerations, all neutronics parameters except induced activation can be measured in one of two fluence modes: either the low fluence mode (~1 MW·s/m$^2$) or the very low fluence mode (~1 W·s/m$^2$). The low fluence mode can be achieved, for example, with a wall load of 1 MW/m$^2$ and a 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. Note, 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, and neutron and gamma-ray spectra. The main problem is the poor resolution and instability of measurements. On the other hand, the methods used for measurements in the low fluence mode have better accuracy and spatial resolution. The main problem here is the activation of the test module and device components that may render the test device inaccessible just after shutdown. 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 X.

The requirements on the test module size and geometry are governed mainly by the objective and 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. To improve the analytical prediction and to identify the various sources of uncertainties, however, 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 TBR, the test module used in this experimental planning approach should

TABLE X

Fluence Requirements for Various Experimental Techniques*

| Integral Parameter | Fluence Requirement (Normalized to Wall Load) — 1 mW·s/m², 1 W·s/m², 1 kW·s/m², 1 MW·s/m²; 14-MeV Point Source |
|---|---|
| Neutron yield | NE-213 fission chamber; Multifoil activation (MFA) |
| Tritium production rate | Lithium glass; Liquid scintillator ($\beta$); Gas counter ($\beta$); Mass spectrometer; Proportional counter |
| Nuclear heating | Gas-filled counter; TLD; Thermoluminescent dosimeter (TLD); Calorimeter |
| Nuclear reaction rate | Fission chamber; Activation foil; Mass spectrometry |
| Neutron spectrum | NE-213 proton recoil; MFA |
| Gamma spectrum | NE-213 |

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

duplicate the actual blanket module in great detail. Verification of achievable TBR also requires that factors affecting the global TBR, such as actual penetrations for heating and fueling, full coverage 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 TPR in a partial coverage case to demonstration or commercial reactor TBR involves many uncertainties. These uncertainties arise from

1. uncertainty in specifying the neutron source condition at the first wall of the test module
2. uncertainties in predicting (by calculation or measurements) the energy-dependent angular flux at the test module boundaries
3. 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 difficult. 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 (a) the details of material and geometrical arrangement within the test modules, and (b) the conditions at the test module boundaries that 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 $\theta_m$, 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 $\theta_m$ and $L_m$ was examined separately by two two-dimensional models. The first is an $R$-$\theta$ geometry, shown in Fig. 12, where $R$ refers to the minor radius of the plasma. In this model, full coverage is obtained at $\theta_m = 180$ deg due to the presence of a symmetry boundary condition at $\theta = 90$ deg. That is, the model assumes a second identical test module on the other side of the test device. The second is an $R$-$Z$ model, shown in Fig. 13, where $Z$ is the axial direction for the plasma along which $L_m$ is measured.

Figure 14 shows the variation in the local TPR as a function of the poloidal angular width $\theta_m$ at three

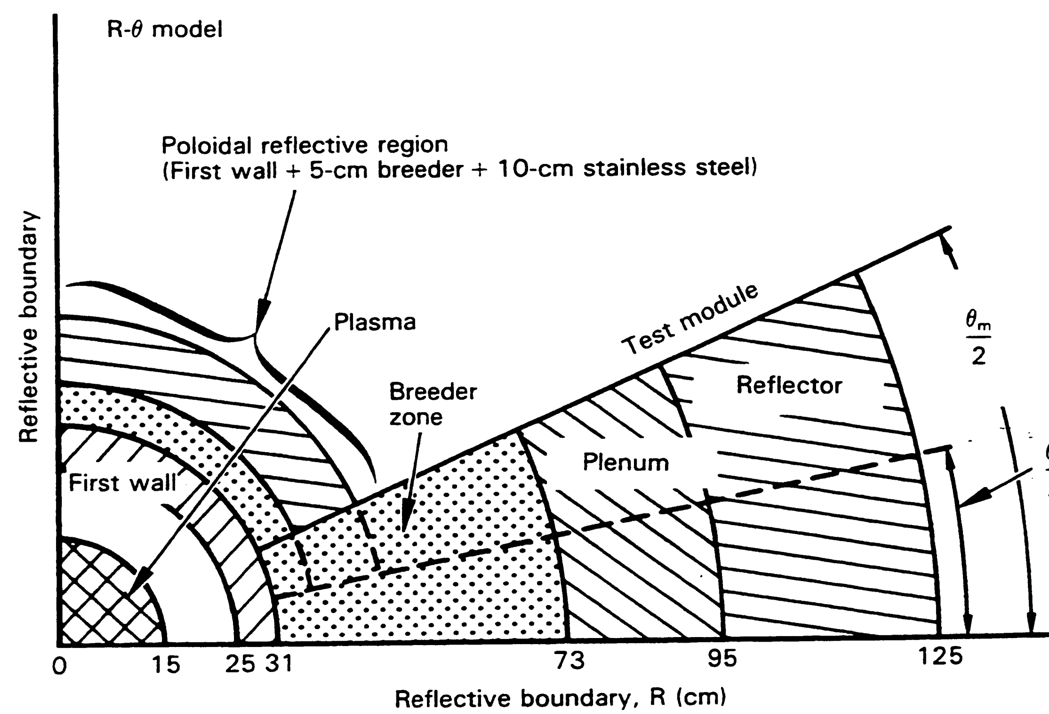



Fig. 12. The $R$-$\theta$ geometrical model used to examine the effects of poloidal boundary conditions on the test module TPR.

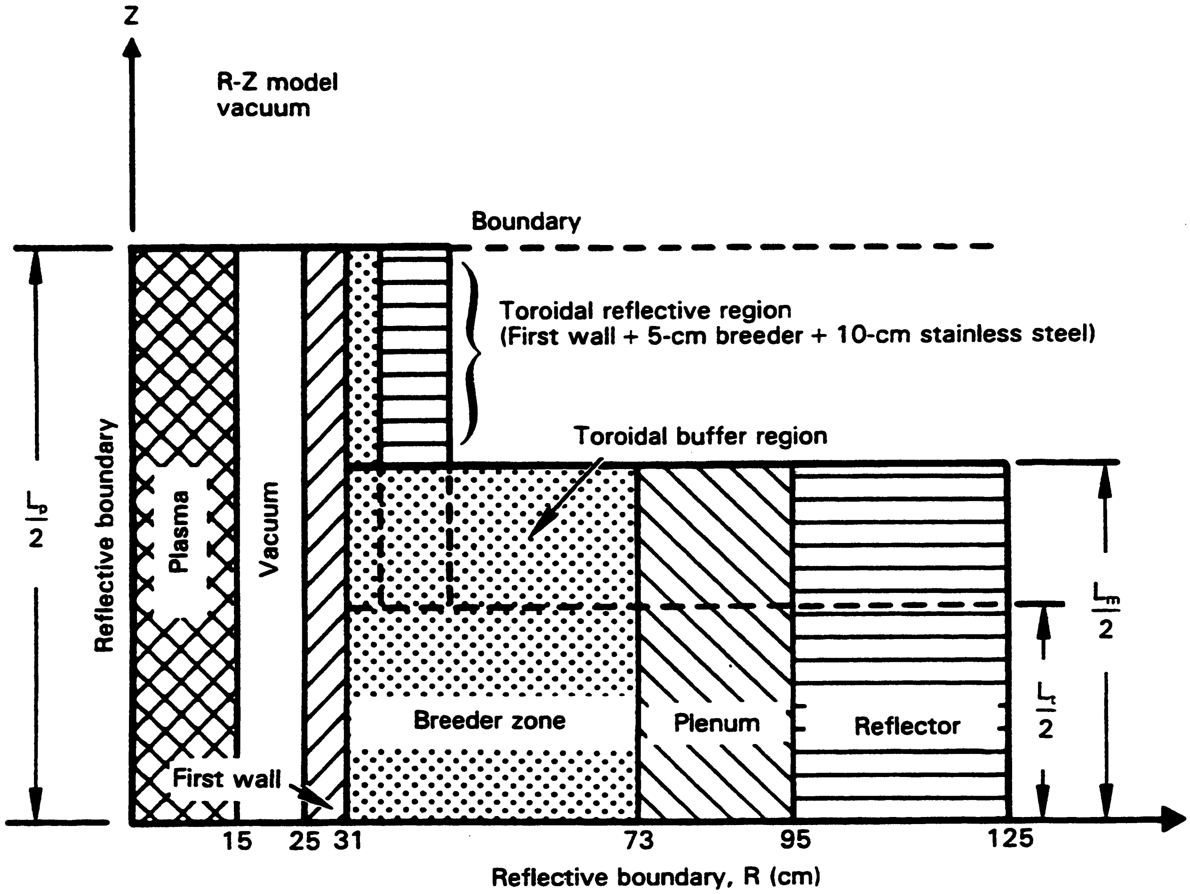



Fig. 13. The *R-Z* model used to examine the effects of axial boundary conditions on the test module TPR.

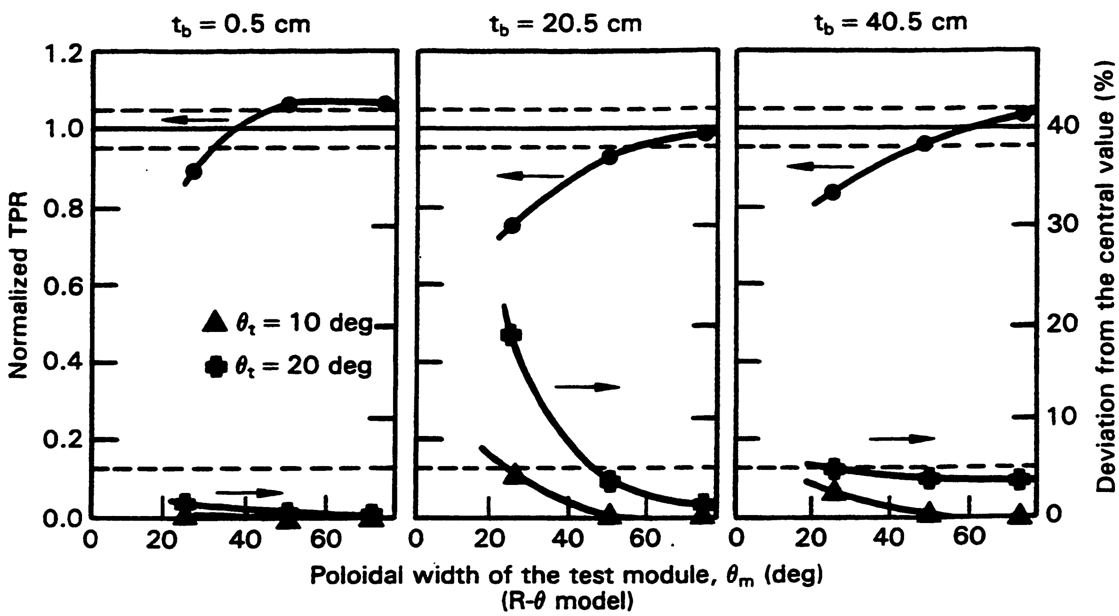



Fig. 14. The total TPR at the test module center as a function of the test module width $\theta_m$, normalized to the corresponding value in the full coverage case. Also shown is the deviation of the TPR from the central value at the locations $\theta_t = 10$ and 20 deg (*R-θ* geometry).

locations (characterized by the distance from the first wall $t_b$) 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 TPR at $\theta_t = 10$ and 20 deg, where $\theta_t/2$ is the angle where the measurement is taken relative to the central line. This deviation depends on the width of the test module, which is characterized by the angle $\theta_m$, and on the location throughout the test module, e.g., front or middle, as shown in Fig. 14.

This figure can be used to determine requirements on the test module width, which maintain the TPR within certain bounds. If the local TPR 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 between $\theta_m = 22$ deg (−5% deviation) and 48 deg (+5% deviation). If measurements were to be performed at the back edge of the test module central line, the corresponding values would be $\theta_m = 48$ to 71 deg for the same target accuracy.

Similar curves that specify the minimum test module size in the *R-Z* geometrical model are shown in Fig. 15 for the cases where the plasma length is $L_p = 160$ and 320 cm. In this geometrical arrangement, the test module length is characterized by the parameter $L_m$. The test module central zone, where measurements are most likely to be performed, is characterized by the parameter $L_t$. Full coverage in the *R-Z* geometry corresponds to the case where $L_p$ and $L_m$ both approach infinity. For the case with $L_p = 160$ cm, the tritium production value at the front edge of the test module is <80% of the corresponding value in the full coverage case. This situation worsens as the test module length $L_m$ increases (at fixed plasma length). For the case with a plasma length $L_p = 320$ cm, however, the TPR at the front edge of the test module is within 5% of the corresponding value in the full coverage case provided the test module length is between $L_m = 50$ and 150 cm (+5 and −5% deviation respectively).

There are several serious problems concerning the usefulness of TBR verification tests in a fusion test device with a test module that covers the plasma source only partially. To illustrate some of these problems, Fig. 16 shows the integrated values of TPR in various segments of the test module and the overall tritium breeding rate as function of the test module size. 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 TBR 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

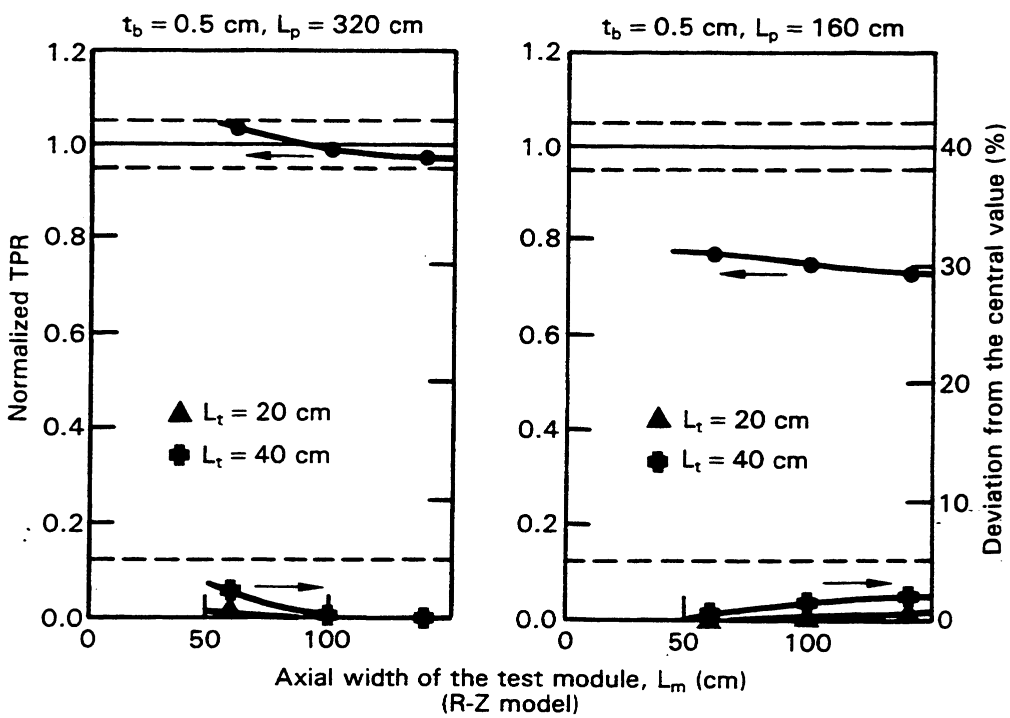



Fig. 15. The total TPR at the test module center as a function of the test module width $L_m$, normalized to the corresponding value in the full coverage case. Also shown is the deviation of the TPR from the central value for central zone widths, $L_t = 20$ and 40 cm (*R-Z* geometry).

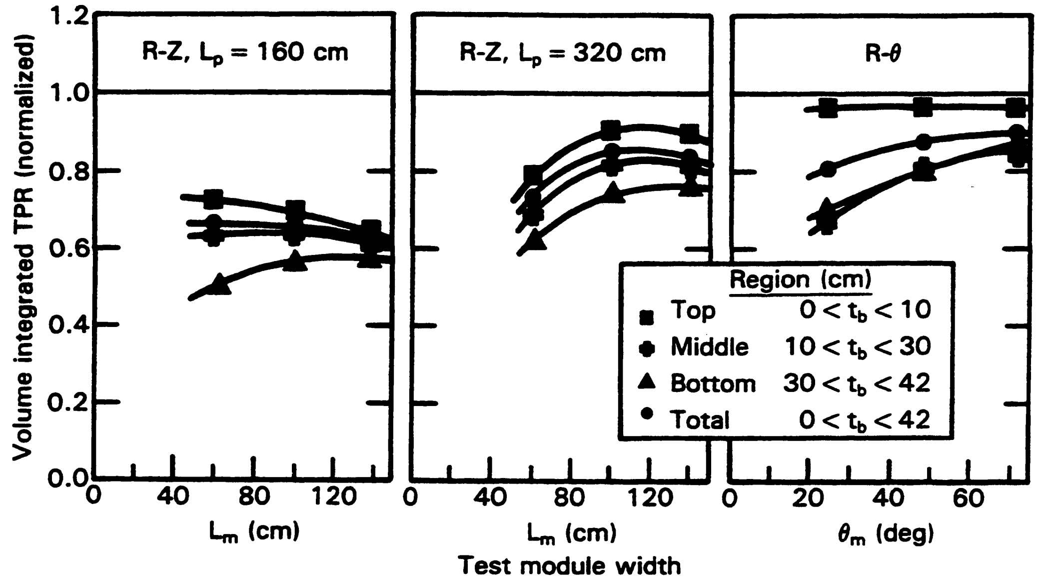



Fig. 16. Total TPR integrated over various spatial segments, normalized to the full coverage case, as a function of the test module width ($L_m$ or $\theta_m$).

TBR in a full-scale reactor are greater than presently estimated margins in the tritium breeding potential for candidate blanket concept.

The neutronics analysis demonstrates that 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 for resolving the issue of the achievable TBRs.

### IV.E. Blanket Fluence Goals

In a typical development scenario for components, tests in both normal and off-normal environments out to, and possibly beyond, the 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 the information learned as fluence increases in nuclear experiments. Our predictions must rely on available materials properties data, present knowledge of neutron fluence effects on materials, and engineering judgment.

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 on materials testing in fission reactors. Limited data are also available from irradiations in ion sources and the low fluence fusion neutron source, rotating target neutron source-II (RTNS-II). In general, the fluence dependence for many material properties can be divided into two regimes: a transient regime where the material property is constantly changing with fluence, and a steady-state or saturation regime where the material property is either changing at a constant rate or not changing at all. For example, structural materials typically exhibit an incubation period of low (or zero) swelling prior to the onset of breakaway swelling, then undergo a transient regime until the steady-state swelling rate is achieved. Based on very limited data and theoretical considerations, it is anticipated that the incubation fluence will be 5 to 8 $MW \cdot yr/m^2$ for PCA, and over 10 $MW \cdot yr/m^2$ for HT-9 and V-15 Cr-5 Ti.

In an attempt to quantify the information gained as a given material property is tested to 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 has been applied to the fluence behavior properties of key structural materials. 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 to 5 $MW \cdot yr/m^2$. For creep in the structural materials, the uncertainty projections are dependent on the goal fluence under consideration. Specifically, if the goal fluence is $<10$ $MW \cdot yr/m^2$, then the swelling behavior of HT-9 and V-15 Cr-5 Ti is not an issue, and the uncertainty in creep is significantly reduced after measurements are performed in the neutron exposure range of 1 to 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 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 on 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 on 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 this study has been on interactive effects. The general nature of an interaction is dependent on the contributing material properties; the 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 that 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. 17. The swelling in $Li_2O$ is the major material property responsible for the interaction. Thermal expansion and cracking/redistribution of the $Li_2O$ also play a role in this interaction, primarily in the early operation of the component (i.e., in the neutron exposure range of 0 to 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 have not yet been defined. However, the key issues 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. This approach was used to identify fluence goals for the key issues for the blanket structure. 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. 18. Also shown 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 largest. 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 to 5 MW·yr/m$^2$ fluence range yields most of the information concerning the remaining facets of the interaction.

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 XI for those issues relating to the structure. The results of the analysis have suggested that the goal fluences required to

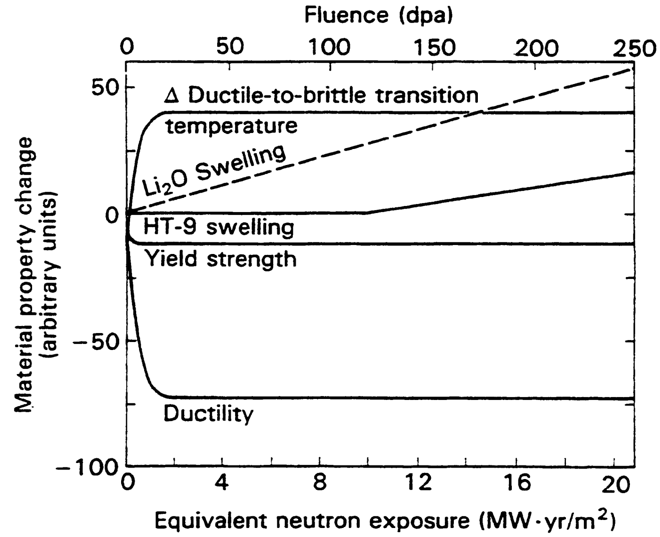



Fig. 17. Fluence dependence for several of the key material properties associated with the mechanical interaction between the $Li_2O$ and HT-9 cladding.

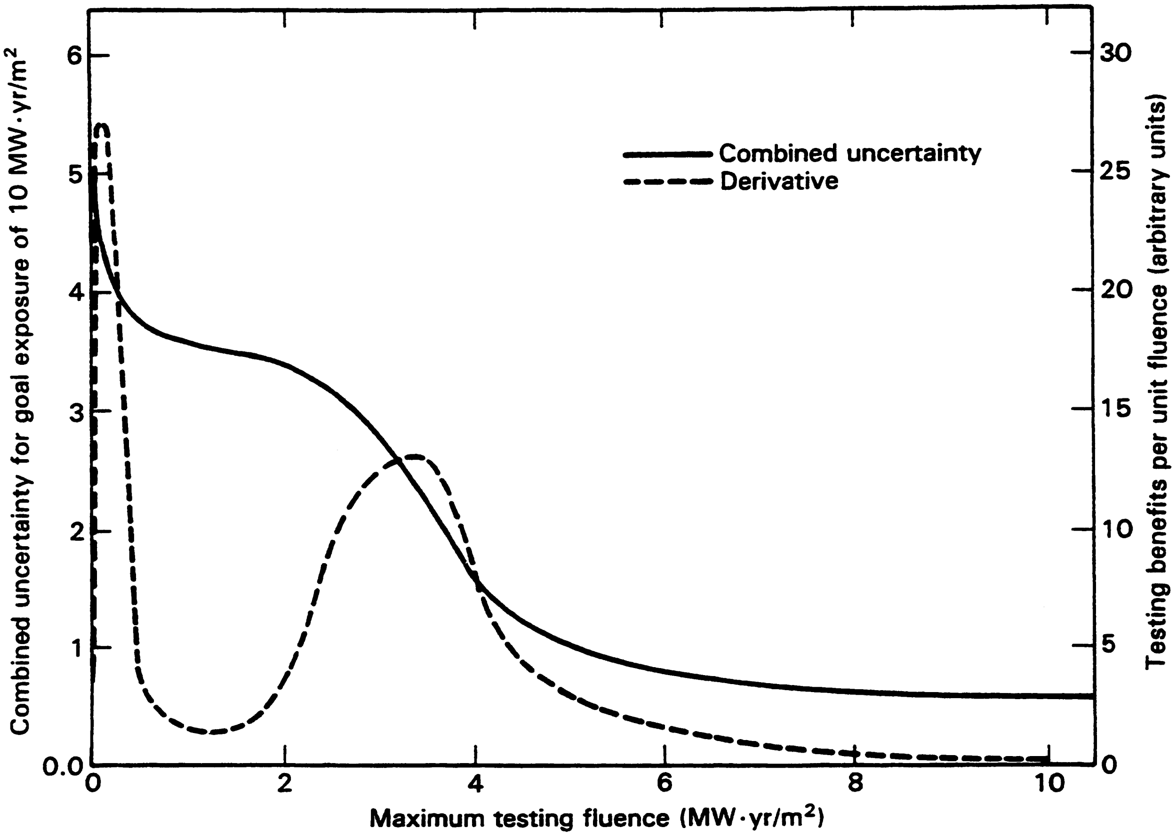



Fig. 18. The fluence dependence for the combined uncertainty of the material properties associated with the mechanical interaction between the $Li_2O$ and HT-9 cladding. Also shown is the absolute value of the derivative of the combined uncertainty, which provides a qualitative measure of the testing benefits per unit of testing fluence.

TABLE XI

Neutron Fluence Goals for HT-9/$Li_2O$/He Blanket 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 Chap. 8 of Ref. 1 |
| Mechanical interaction within the structure itself | HT-9: thermal differential expansion, fatigue/crack growth, ductility, swelling[a] | 3 to 5; (16 to 20)[a] |
| Plasma/structure interaction | --- | --- |
| Premature failure of welds in full components | HT-9: thermal differential expansion, fatigue/crack growth, ductility, swelling[a] | 3 to 5; (16 to 20)[a] |
| Coolant/structure interaction | --- | --- |
| Magnetic field interaction with structure | HT-9: ductility, fracture toughness, fatigue/crack growth, creep | 3 to 5 |
| Mechanical interaction between solid breeder/multiplier and structure | HT-9: ductility differential expansion, fatigue/crack growth, ductility, swelling[a] | 3 to 5; (16 to 20)[a] |

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

resolve these issues are typically in the range of neutron exposures corresponding to 1 to 5 $MW \cdot yr/m^2$ if the application of the component is for a goal life of 10 $MW \cdot yr/m^2$ (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 that 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 to 4 $MW \cdot yr/m^2$ beyond the onset of swelling if all uncertainties associated with the interaction are to be addressed.

## V. NONFUSION FACILITIES

### V.A. Introduction

Nonfusion facilities can and should play an important role in fusion nuclear technology R&D. Many suitable facilities are available with a well-established operational procedure and at a reasonable testing cost. The use of nonfusion 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 nonfusion experiments will be valuable in reducing costs and risks of the more costly and complex integral tests. Nonfusion testing alone, however, cannot satisfy all the nuclear technology experimental needs. This section summarizes the results of evaluation of the capabilities and limitations of nonfusion test facilities: nonneutron test stands (Sec. V.B), point neutron sources (Sec. V.C), and fission reactors (Sec. V.D).

### V.B. Nonneutron Test Stands

The identified issues and testing needs clearly indicate that nonneutron test facilities can and should serve an important role in fusion nuclear technology R&D. The role of nonneutron 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 nonneutron experiments and facilities is generally much lower than those involving neutrons. Therefore, information from nonneutron tests is important for at least two reasons. First, they permit early scoping of some material and design options. Second, they permit 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, available from other technologies. Thus, there is a definite need to upgrade existing nonneutron facilities and to build new ones. A more quantitative description of the nonneutron 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 for providing bulk heating in these experiments without neutrons. While radiation damage is not critical for some types of experiments, our results show that bulk heating is very important in many multiple interaction experiments.

### V.C. Point Neutron Sources

Point neutron sources are attractive for irradiation testing because of their 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. These 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. Use of an advanced fusion materials irradiation test (FMIT)-type machine capable of large-scale testing, however, will be discussed as an "upper limit" on point neutron source application.

#### *V.C.1. RTNS at Lawrence Livermore National Laboratory (LLNL)*

RTNS-II (Ref. 13) generates neutrons by bombarding a tritium-containing target with 400-keV deuterons. Neutrons are produced by the D-T fusion reaction, which produces a nearly monoenergetic 14-MeV neutron spectrum. The tritium is held in the 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 that allow the tritium to be retained on the target. A maximum flux level of $\sim 5 \times 10^{12}$ $n/cm^2 \cdot s$ is attained in a test volume of <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.

#### *V.C.2. Intense Neutron Source (INS) at Los Alamos National Laboratory (LANL)*

The INS (Refs. 1 and 14) was intended to generate neutrons by injecting a 300-keV, 1.1-A 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 $\sim 10^{15}$ 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 $1 \times 10^{14}$ n/cm$^2$·s 14-MeV neutrons was predicted for a few tenths of a centimetre next to the target. The volume of test space at $10^{13}$ n/cm$^2$·s or better was expected to be $\sim 1$ 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.

### *V.C.3. LAMPF A-6 Target Station at LANL*

The Los Alamos Meson Physics Facility[15] (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 provide a flux of $\sim 1 \times 10^{13}$ n/cm$^2$·s at a volume of 0.02 m$^3$. The neutron spectrum is much different than for fusion with most neutrons having energies >1 MeV but a significant portion having energies reaching to $\sim 10^3$ MeV. The combination of flux, spectrum, and duty factor is sufficient to provide a displacement level of $\sim 1$ dpa/yr in copper. While of interest to materials science, the differences in flux, spectrum, and fluence are probably too large for multiple-effects testing or high exposure materials testing.

### *V.C.4. FMIT Facility at Hanford Engineering Development Laboratory*

The FMIT facility[16] 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 spectrum averaging around 14 MeV. Damage calculations indicated that the spectrum from 35-MeV deuterons on lithium was adequate for fusion materials evaluation. The FMIT would provide 10 cm$^2$ of test space with a neutron flux of $1 \times 10^{15}$ n/cm$^2$·s or greater. Approximately 500 cm$^3$ would be available at $1 \times 10^{14}$ 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. Note that FMIT construction has been postponed.

### *V.C.5. Large Volume Point Source Facility*

To explore the potential of point sources for multiple-effects testing, a device for producing a fusion-like neutron environment in a 23- × 23- × 23-cm$^3$ cube was considered. This was assumed to be the smallest size of interest to multiple-effects experiments. The approach taken was to scale up the FMIT accelerator to the maximum extent possible using relatively small extrapolations of existing technology. The result yielded a neutron flux of $\sim 5 \times 10^{13}$ 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 with considerably more flexibility than in other neutron-generating machines. In this "Super FMIT" source, 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 an rf quadrupole of the zero-mode type[17] 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-A beam can be obtained. The beam interacts with a large flowing lithium target that dissipates less energy per unit area than FMIT and therefore should be easier to engineer and operate.

In summary, the flux and testing volume capabilities of recent point neutron sources are given in Table XII. The primary point source for fusion materials testing in use in the United States is 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.

## V.D. Fission Reactors

One option for performing the engineering experiments needed for fusion development is to employ fission reactors. 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) have 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 XIII and discussed below.

Radiation damage simulation is a concern because of the difference between the fusion and fission reactor spectra. Fusion radiation damage in structure

TABLE XII

Point Neutron Sources for FMIT

| Facility | Status | Peak Flux ($n/cm^2 \cdot s$) | Testing Volume | Applicable to Multiple Effects |
|---|---|---|---|---|
| RTNS-II | In use for materials testing | $\sim 5 \times 10^{12}$ | $\sim 0.1\ cm^3$ | No |
| INS | Conceptual design completed; project terminated | $1 \times 10^{14}$ | $\sim 1\ cm^3$ | No |
| LAMPF A-6 | Operational | $1 \times 10^{13}$ | $\sim 0.02\ m^3$ | No |
| FMIT | Design completed; project deferred | $1 \times 10^{15}$ | $\sim 10\ cm^3$ | No |
| SUPER FMIT | Scoping study | $5 \times 10^{13}$ | $\sim 0.016\ m^3$ | Yes |

TABLE XIII

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. Nonnuclear conditions
   a. Magnetic field
   b. Surface heat
   c. Particle flux
   d. Mechanical forces
6. Reactivity considerations
7. Availability for testing
8. Cost

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 that 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 $^{58}Ni$, boron, or lithium; although this approach can improve the simulation of first-wall damage, it can introduce uncertainties in the material performance due to the effects of the additional elements. Overall, fission testing, therefore, is suited more for 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 (and tritium production) 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 XIV 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.

The total test volume that is available and potentially useful for fusion testing is significant in view of

TABLE XIV

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] ($n/cm^2 \cdot s$) |
|---|---|---|---|
| Slab[b] | $Li_2O/He/HT\text{-}9$ | --- | $1.4 \times 10^{15}$ |
| | | Cadmium | $1.5 \times 10^{15}$ |
| Slab | Li/Li/V | --- | $9.4 \times 10^{14}$ |
| | | Cadmium | $9.8 \times 10^{14}$ |
| Slab | $LiAlO_2/Be/H_2O/PCA$ | --- | $6.2 \times 10^{14}$ |
| | | Cadmium | $8.7 \times 10^{14}$ |
| Submodule[c] | $Li_2O/He/HT\text{-}9$ | --- | $5.1 \times 10^{14}$ |
| Submodule | $LiAlO_2/Be/H_2O/PCA$ | --- | $2.1 \times 10^{14}$ |

[a]Water-moderated, plate-fueled test reactor assumed.
[b]Core side.
[c]In core.

the number and sizes of tests that 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 United States and U.S. plus foreign reactors are shown in Tables XV and XVI. The numerical entries give the number of locations that 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 $10^{14}$ $n/cm^2 \cdot s$. 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 a 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 the Oak Ridge Research Reactor (ORR) by modifications that would increase the reactor power. Other reactors could perhaps provide slab test locations by conversion of thermal columns.

TABLE XV

Number of Existing Acceptable In-Core Test Locations in United States (U.S. and Foreign) Reactors

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

TABLE XVI

Numbers of Existing Acceptable Slab Test Locations in United States (U.S. and Foreign) Reactors

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

It is desirable for fission tests to include nonnuclear conditions such as mechanical forces, surface heating, magnetic field, or particle flux. 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 the 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 that 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 that were examined, the effect of the reactor core thickness on the net reactivity effect was found to be large (Fig. 19). As a point of reference, an average control rod is worth approximately +\$2.50. This implies that small reactor cores (<30 cm thick) will have great difficulty in accommodating such tests. Large cores (>50 cm thick) can certainly accommodate them, and medium cores (30 to 50 cm thick) may require some modifications.

Whether or not reactors will be available for testing when needed is also an issue that can affect program strategy. Most facilities contacted during the study will be in operation for the indefinite future; exceptions were the Experimental Breeder Reactor-II

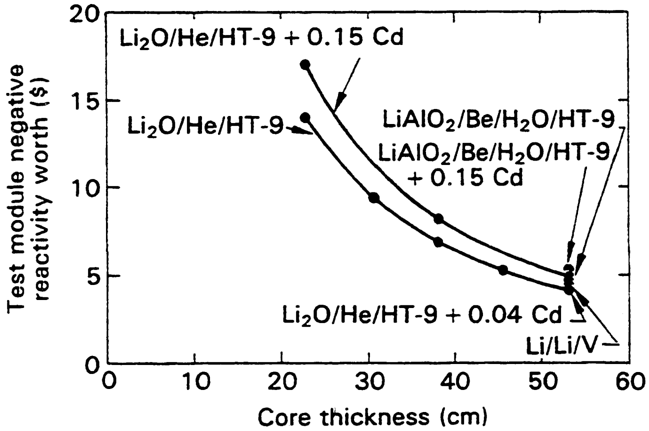



Fig. 19. Slab test module negative reactivity worth as a function of core thickness.

(EBR-II) (with a projected life of ten more years) and power burst facility (PBF) (with a projected life of one more year). All indicated that test spaces would be available, given proper programmatic priority. In view of the plans for EBR-II and PBF, as well as the recent mothballing of the engineering test reactor (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 nonnuclear conditions 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 part of the fusion R&D program. Although it cannot completely replace or eliminate the need for fusion testing (except in 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.

## VI. FUSION FACILITIES FOR NUCLEAR EXPERIMENTS

### VI.A. 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 are large, in the range of 10 to 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. Even if neutrons are not critical in certain tests to simulate 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 Sec. V.D, 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 VI.B is a summary of the potential of tandem mirrors as a Fusion Engineering Research Facility (FERF). Section VI.C 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 Sec. VI.D.

### VI.B. Tandem Mirror Test Facilities

#### *VI.B.1. Test Facility Concepts*

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 >10 yr ago that single-cell mirrors might be attractive as nuclear research test facilities. A device, called FERF, was designed[18] 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[19] 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. The 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 Tandem Mirror Experiment facility (TMX) at LLNL.

A more detailed TDF design was developed[20] by LLNL and other fusion organizations during 1982-1983. This design, shown in Fig. 20, would have a total capital cost in the range of \$1 to 1.5 billion. It would provide two blanket test module ports and a substantial area for neutron damage testing to fluences of 5 to 10 $MW \cdot yr/m^2$. To address the problem of trapped particle modes in TMRs, the TDF electrostatic plugs were moved outward 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 Mirror Fusion Test Facility-B (MFTF-B) configuration.

Two different options for achieving microstability in TDF were considered. With the first, stream stabilization,[21] 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,[22] based on stability by sloshing ions with the stream gas removed, has potential to provide performance consistent with the engineering parameters shown in Table XVII. In this "beam-fueled" mode, the electron temperature and lifetime double, 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-$\alpha$+T] upgrade of MFTF-B,[23] would combine plasma confinement objectives with nuclear experiment and test objectives. The approximately \$450 million, 11-MW (fusion) MFTF-$\alpha$+T upgrade would be a D-T-burning machine with significant $\alpha$-power deposition in the central cell. It would incorporate many recent ideas that are expected to result in TMR 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 the MFTF-$\alpha$+T operation would focus on technology development and low fluence integrated nuclear testing with one or more (at higher cost) beam-driven test cell inserts. During the nuclear test phase, the machine would be operated in a low confinement mode, but would provide a relatively high (~2 $MW/m^2$) neutron wall loading at an ~10% availability for test periods up to ~100 h. The \$450 million MFTF-$\alpha$+T facility would be fully shielded and remotely maintained. Unlike TDF where

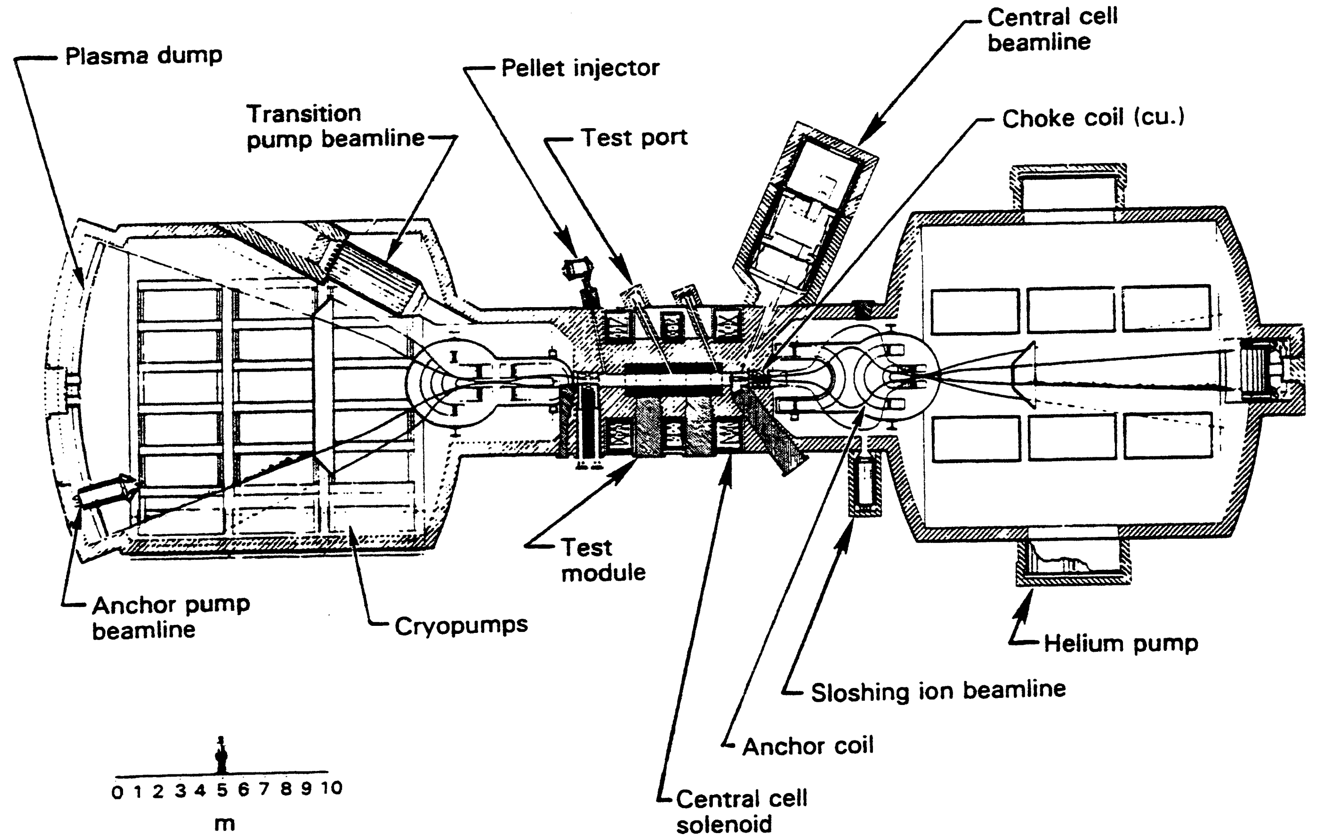



Fig. 20. Cross section of TDF machine. (Some components are shown rotated out of plane for clarity and only one of the eight center cell beamlines is shown.)

the insert would be the entire central cell, in MFTF-$\alpha+T$ the insert would be embedded in the central cell.

Another design study, TASKA-M, was recently completed by Kernforschungszentrum Karlsruhe, FRG (KfK), in conjunction with the University of Wisconsin.[24] 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 on near-term physics assumptions and mid-1980s 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 in the axial direction. Nevertheless, TASKA-M, with a projected direct cost of less than $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-$\alpha+T$, 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, octupole end plugs have been proposed for MHD stabilization in TMRs. 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 to 3 $MW/m^2$.

*VI.B.2. Test Facility Assessment*

A preliminary assessment of the ability of an MFTF-$\alpha+T$ or TDF class FERF to resolve the nuclear issues was performed. This assessment focused on the ability of tandem mirror FERF options, in conjunction with complementary fusion and nonfusion facilities, to play a role in tokamak development. Specifically, it is known that the tokamak and tandem mirror geometries, magnetic field profiles, and plasma side conditions (i.e., heat flux and erosion) are quite different. These issues have been addressed to some extent and it appears that many aspects of tokamak blanket performance can be achieved in scaled-down tandem mirror test modules. In most cases, a radiant heat flux

TABLE XVII

TDF Engineering Parameters

| Parameters | Quantity |
|---|---|
| Overall machine | |
| Full power run length (h) | ~1000 |
| Availability [% (life average)] | 30 |
| Design life (FPY) | 5.4 |
| Total capital cost [in millions of (1982) dollars] | ~1000 |
| Plasma | |
| Length (central cell) (m) | 8.0 |
| Radius (m) | 0.15 |
| Peak beta (%) | 40 |
| Test zone | |
| 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 |
| Tritium | |
| Consumption rate (g/h) | 0.23 |
| Inventory (kg) | ~0.3 |
| Vacuum | |
| Base pressure (Torr) | $5 \times 10^{-6}$ |
| Total pump speed (ℓ/s) | $1.3 \times 10^{8}$ |
| Magnets | |
| Superconducting | |
| Material | Nb-Ti |
| Peak conductor field (T) | 8 |
| Peak center field (T) | 4.5 |
| Resistive | |
| Material | Copper alloy |
| Peak center field (T) | 15 |
| Neutral beams | |
| Mode | Continuous |
| Energy (max) (keV) | 80 |
| Power | |
| Central cell (MW) | 51 |
| Pumping (MW) | 7.0 |
| Sloshing (MW) | 0.8 |
| rf sources | |
| Type | Electron cyclotron resonance heating |
| Frequency (GHz) | 35/60 |
| Power (MW) | 1.0/0.6 |

must be applied to the test module first wall to simulate the typically high surface heat flux in tokamaks. A preliminary design of a tungsten filament resistive heater indicates that such a capability is possible.

Differences between the capabilities of an MFTF-$\alpha$+T class facility and those of a TDF class facility primarily relate to overall expected availability. A TDF class FERF (ultimate fluence of 5 to 10 MW·yr/$m^2$) would provide much greater operating time than an MFTF-$\alpha$+T class FERF (ultimate fluence ~1 MW·yr/$m^2$) and, consequently, is more attractive for tests involving extended fluence effects (>4 MW·yr/$m^2$).

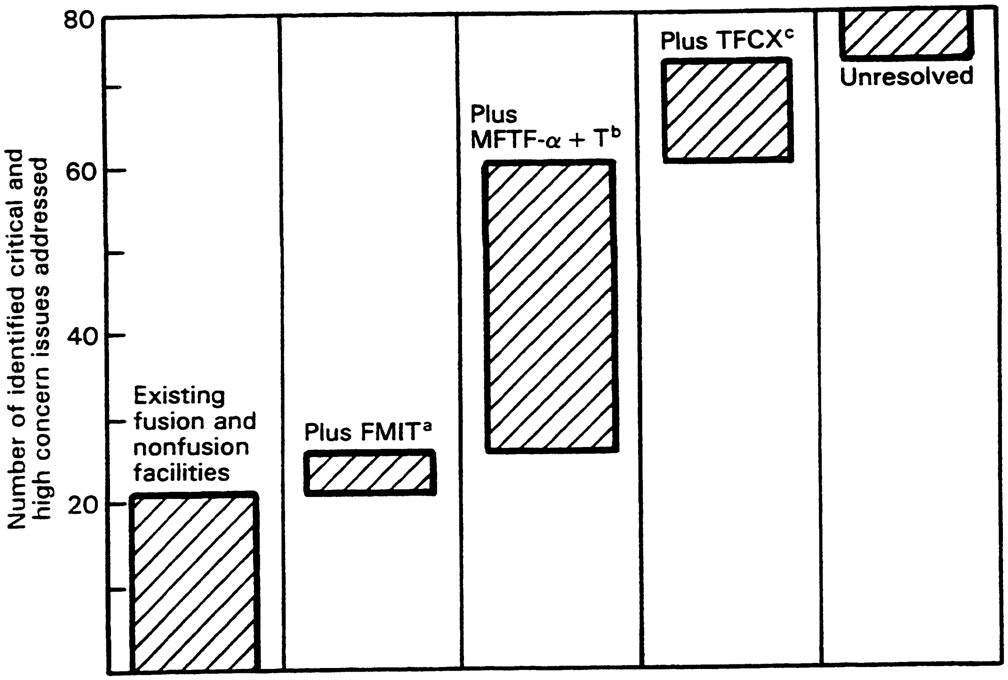
To perform this assessment, each of the 80 critical or high concern nuclear issues identified earlier (Sec. II) was considered with respect to the contributions that could be made by other experimental facilities (e.g., fission reactors), the adequacy of the tandem mirror test environment, and fluence requirements.

As indicated in Fig. 21, an MFTF-$\alpha$+T facility, with complementary facilities [e.g., existing fusion and nonfusion facilities, FMIT, Tokamak Fusion Core Experiment (TFCX)] might address over 90% of the FINESSE nuclear issues (i.e., including those associated with tokamaks, and not limited to blanket/first-wall issues). The remaining issues, which involve fluence effects on large-scale components, cannot be addressed in MFTF-$\alpha$+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-$\alpha$+T and TDF with those of recent tokamak test facility designs, the Fusion Engineering Device-R (FED-R) design[25] and the International Tokamak Reactor (INTOR) design.[26] Such a comparison is presented in Tables XVIII and XIX. Note that the assumptions required to construct this table are uniform and consistent, but may not reflect the published values. Nevertheless, the trends are expected to be relatively valid.

As shown, the tandem mirror facilities have the 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 and with a total fusion power that is reduced by at least an order of magnitude or more. The tandem mirror component test area can be increased, 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 several types of experiments, most notably, those that involve the dynamics of tritium recovery from solid breeder blankets.

The rough cost comparisons of Table XIX can be used to demonstrate that MFTF-$\alpha$+T, TDF, and the tokamak alternatives represent distinct cost categories. The MFTF-$\alpha$+T upgrade would have a low capital cost and an annual operating cost that is expected to be dominated by personnel costs. As a result, the cumulative (i.e., life cycle) cost is expected to be of 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




[a]Small size of FMIT box reflects limited breakout of irradiation issues.
[b]Low fluence fusion nuclear test facility.
[c]Assuming superconducting TF coils.

Fig. 21. FINESSE issue resolution in the context of a multifacility development plan including MFTF-α+T.

by the tokamaks, which cost 60 to 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 toroidal field (TF) coils would require a large electrical input. Thus, the cost differential between MFTF-α+T and a large tokamak ETR is five- to sixfold, and their respective capabilities should be viewed in this light.

### VI.C. Tokamak Test Facilities

#### *VI.C.1. Test Facility Concept*

To assess the advantages and disadvantages of a tokamak nuclear test facility, a study was performed to identify a 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 follow:

TABLE XVIII

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·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 XIX

Cost Comparison of Various Fusion Engineering Facility Candidates

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

[a]At 50 mil/kW(electric)h.
[b]Assumes INTOR TBR and blanket coverage of 50%.
[c]At \$20 000/g.
[d]Estimate.

1. nuclear performance
   a. test volume at least 0.5 m in depth from frontal area, with at least 10 $m^2$ exposed to the fusion neutron current
   b. neutron wall loading at least 1 $MW/m^2$ and preferably 3 $MW/m^2$
   c. lifetime fluence capability of 1 to 10 $MW \cdot yr/m^2$
   d. surface heat load over 80 $W/cm^2$
   e. ease of test module installation and replacement
2. duty factor
   a. burn time at least 500 s (preferably steady state)
   b. dwell time between pulses $<100$ s
   c. continuous operating period $>1$ week
3. cost constraints
   a. capital cost less than \$1000 million for the complete facility
   b. minimum operating cost, i.e., electrical power consumption <200 MW (electric) and tritium consumption $\leq 5$ kg/yr.

The last constraint implies a fusion power under 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.

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 (OH) regime; this assumption should be valid if the externally injected power does not significantly exceed the ohmic power.

The following selections were made to ensure 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 were chosen, principally because of the inability to shield superconducting coils in a compact device.
2. Ignited operation was selected 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. Radio-frequency heating (ion cyclotron waves) was selected 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 minimized 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 maximized the flux swing available for current startup and for driving 1000-s pulses.

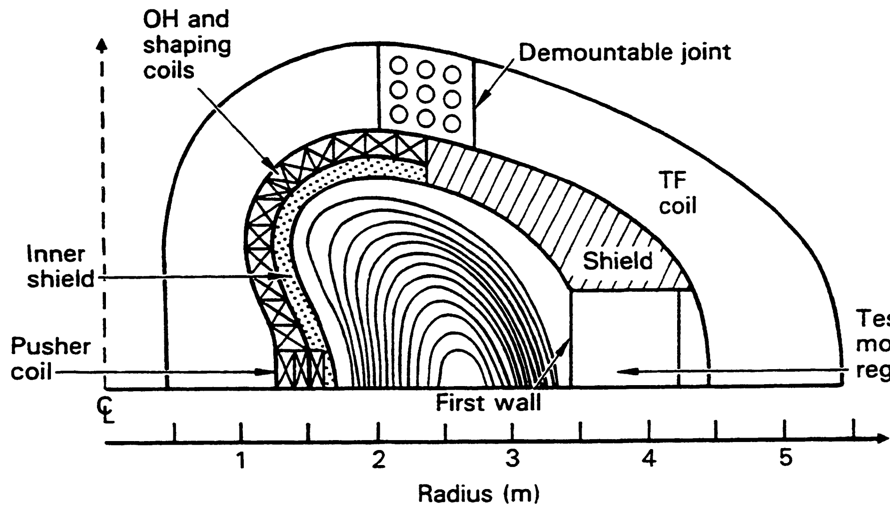



Fig. 22. Partial elevation view of tokamak nuclear test facility.

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

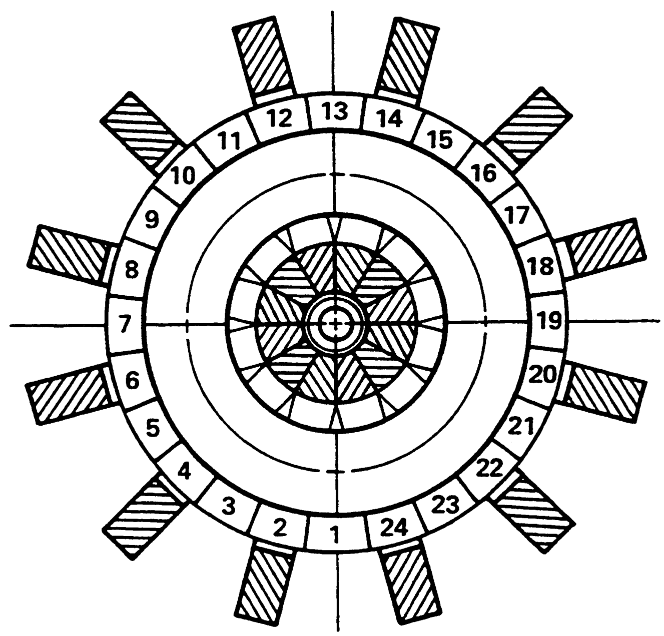
8. Pumped limiters were selected to avoid the additional size and complexity of magnetic divertors.

Table XX gives the principal geometric parameters and performance characteristics. Figure 22 shows an elevation view schematic, and Fig. 23 shows how the device sectors are utilized.

To achieve ignition mainly by OH, 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.

The "pusher coil" required for indenting the bean plasma is located in the inboard blanket/shield region and dissipates ~25 MW of power. The current-driving solenoid is also located inside the TF coils in order to 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 (PF) 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 mill/kWh) and $75 million for tritium (at $15 000/g). A significantly different (and still unidentified) design approach would be required to reduce




| 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 |

Fig. 23. Utilization of tokamak nuclear test facility sectors.

TABLE XX

Illustrative Tokamak Nuclear Test Facility

| Parameter | Startup Phase | Burn Phase |
|---|---|---|
| Geometry | | |
| 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 |
| Plasma | | |
| $B$ at plasma axis (T) | 5.6 | 2.8 |
| $\langle\beta\rangle$ | 0.04 | 0.23 |
| $\langle$Temperature$\rangle$ (keV) | 3.0 | 15 |
| $\langle$Density$\rangle$ ($10^{14}$/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 |
| Magnets | | |
| TF horizontal bore (m) | 3.5 | 3.5 |
| TF vertical bore (m) | 5.3 | 5.3 |
| TF coil material | Copper | Copper |
| Maximum $J$, TF coil (kA/cm$^2$) | 0.93 | 0.47 |
| TF coil loss (MW) | 490 | 122 |
| PF coil loss (MW) | 60 | 40 |
| Power production | | |
| Fusion power (MW) | 60 | 185 |
| First-wall area (m$^2$) | 129 | 129 |
| $\langle$Neutron wall load$\rangle$ (MW/m$^2$) | | 1.15 |
| At outboard (MW/m$^2$) | | 1.3 |
| Circulating power (MW) | 550 | 190 |

the fusion power and circulating power to the 100-MW level.

### *VI.C.2. Test Facility Assessment*

The major plasma physics and engineering advances required relative to the expected performances of Tokamak Fusion Test Reactor (TFTR), Joint European Torus (JET), and JT-60 are associated with achieving and sustaining ignited, high-beta operation over pulse lengths of the order of 1000 s, reactor level $n\tau_E$ in the near-OH regime, reactor level temperatures at smaller $n\tau_E$, and the effectiveness of ion cyclotron resonance heating auxiliary heating. The stability of bean-shaped plasma has been shown at relatively low-$\beta$ 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. feasibility of achieving and maintaining $\beta =$ 0.20 to 0.25
2. 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. effectiveness of ceramic insulation in magnet application
4. lifetime of the first wall under plasma erosion
5. impossible maintenance of the inboard OH and shaping coils, 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.

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.

## VI.D. Availability Considerations for Fusion Engineering Facilities

Many components to be included in the first fusion engineering R&D facilities will have little or no engineering precedence. This will be particularly true of nuclear components which, despite the best efforts in the design and prefusion 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 for iterative design/test/fix programs aimed at improving the nuclear component reliabilities. An apparent paradox results, however, because those nuclear components that would be targeted in a reliability improvement program depend on 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. Although the breeding modules would be designed for high reliability, they would be essentially unproven and necessarily complex. Any one of these could fail such that the overall facility could not operate. Consequently, they have the potential to negatively influence the blanket test module development program by reducing the overall device availability.

Two studies that address these concerns are 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.

### *VI.D.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. High confidence in component performance in entirely new applications, however, 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[26] 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 TF coils, are not expected to fail during the lifetime of the facility (a design assumption) and have required MTBF periods that 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 10-full-power-year (FPY) 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 $R = \exp(-\tau/\mathrm{MTBF})$, where $\tau$ is the operational period (10 yr in this case), it follows that for $R = 0.978$, the individual TF coil MTBF must be 450 operating years. For 80% confidence in this MTBF, a $3.5 \times 450 = 1575$ yr test might be required. Although this is reduced tenfold 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, since blanket removal is expected to be a relatively routine maintenance operation. In this case, a minimal EDF 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 1 month results in a required MTBF of ~10 yr. This implies a typical test period of 34 yr. If equal credit can be taken for 60 modules, tested in parallel, however, the required test period would be reduced to a manageable 0.5 FPY.

These results are illustrated in Fig. 24, 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 on 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.

### *VI.D.2. Reliability Development*

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), U.S. Department of Defense systems development documents[27,28] suggest the following general form of a parametric relationship between the component development/test time and the achieved component MTBF:

$$\mathrm{MTBF} = ct^m ,$$

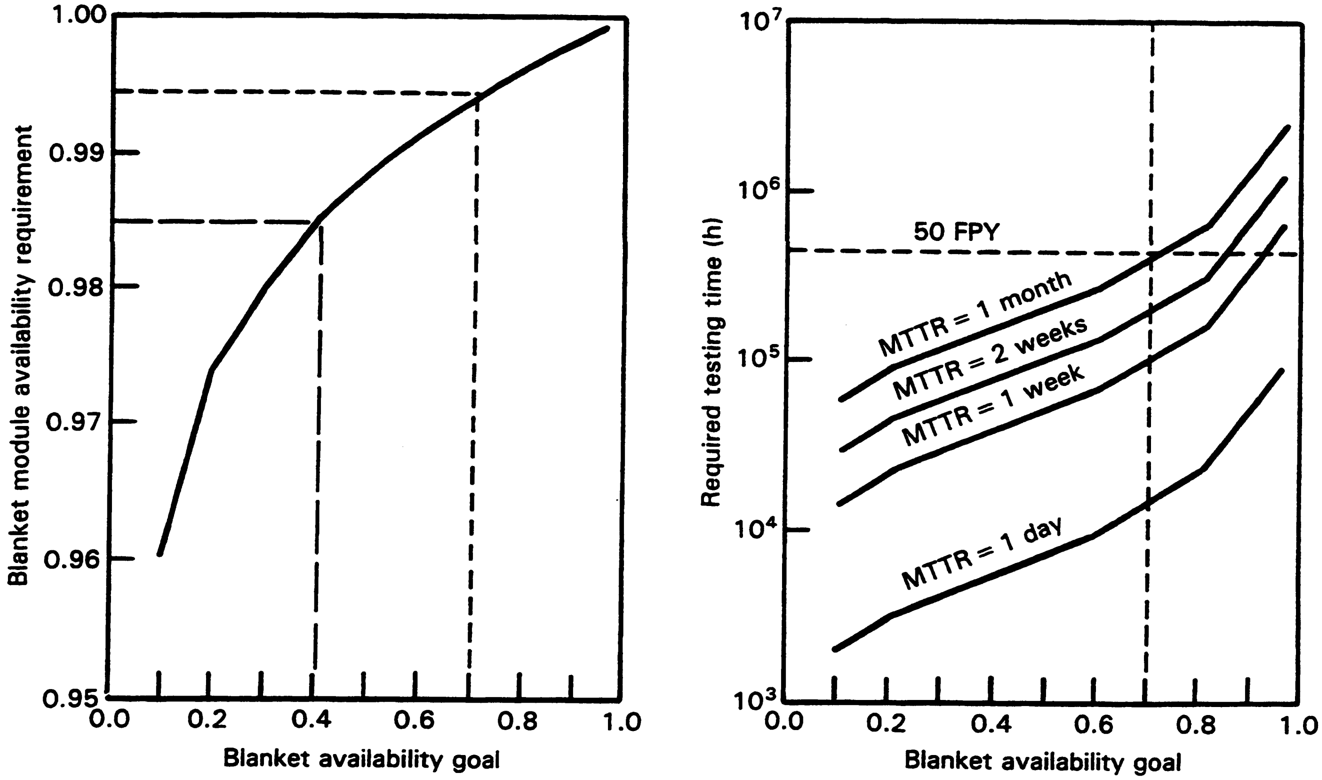



Fig. 24. Blanket module availability and typical testing requirements meeting the 80% confidence level (3.5 × MTBF).

where

$t$ = testing time

$m$ = testing improvement exponent (typically $0.1 \leq m \leq 0.6$)

$c$ = constant determined by the initial component MTBF and initial testing time.

The above equation presents an MTBF improvement model that achieves a goal MTBF through a more or less aggressive (depending on $m$) testing and development program but does not demonstrate the achieved MTBF in the statistical sense described in Sec. VI.D.1.

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 Sec. VII for discussion):

$$\text{TFTR/TFTR-U} \rightarrow \text{INTOR} \rightarrow \text{DEMO} \quad (1)$$

and

$$\text{TFTR} \rightarrow \left.\begin{matrix}\text{TFCX} \\ \text{FERF}\end{matrix}\right\} \rightarrow \text{ETF/EDF} \ . \quad (2)$$

The initial facilities in these pathways are expected to be oriented primarily toward confinement physics goals, while the intermediate engineering facilities (e.g., INTOR, TDF, and MFTF-$\alpha$+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 that 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 6 of 60 modules would be used for testing and development. In comparison, the FERF class would include only the latter six modules.

In the second pathway, the complications caused by relying on unproven tritium breeding modules are avoided by operating the engineering test facilities (ETFs) 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 caused by a failure of *in situ* tritium breeding modules. Consequently, it is expected that a FERF 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 FERF, respectively. The result would be the achievement of a goal MTBF sufficient to support an ETF/EDF or DEMO initial availability (after a 3-yr startup phase) of 20 to 30%, depending on the reliability of the other systems.

For each engineering facility, availability models were developed to describe the influence of blanket 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 calendar time required to achieve a relatively low but acceptable blanket module MTBF of 10 yr (87 600 h) in the two pathways was calculated, based on the parameters shown in Table XXI, and is plotted in Fig. 25.

Pathway 2 achieves the 10-yr blanket 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. Parametric studies[1] performed over the parameters shown in Table XXI indicate that the relative performance of the two pathways is not expected to change.

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

TABLE XXI

Key Assumptions in the Availability Analysis

| | Blanket Test Modules | Blanket Tritium Breeding Modules |
|---|---|---|
| Initial MTBF (yr) | 1 | 2.9 |
| Initial test experience (day) | 31 | 99 |
| MTTR (week) | 2 | 4 |
| Goal MTBF (yr) | 10 | 10 |
| Test improvement factor | 0.50 | 0.10 |
| Experience factor[a] | 0.50 | 0.50 |

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

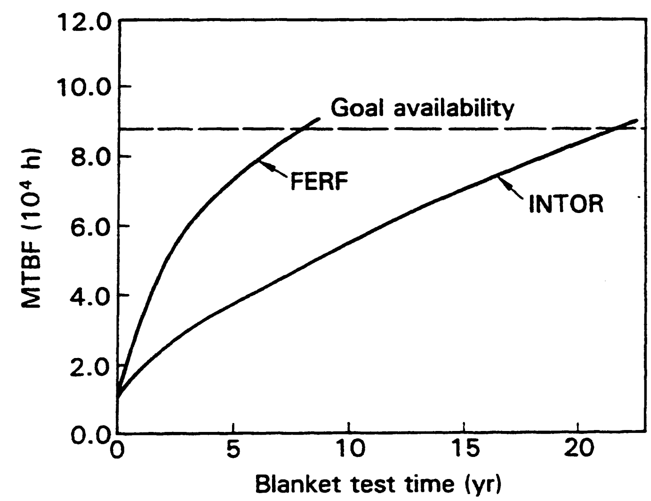



Fig. 25. Higher FERF availability leads to more test time and faster MTBF growth.

## VII. FUSION R&D SCENARIOS

### VII.A. Planning Considerations

In constructing an R&D scenario for fusion, 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:

1. Must every near-term, D-T-burning fusion facility operate in a physics mode that is presently perceived to be extrapolatable to a reactor-relevant "strategic goal" of the program? For example, will a nonignited, beam-driven physics mode be acceptable for an early technology facility if it results in a lower cost and/or risk?

2. Will early experimental fusion technology facilities be required to provide much or all of the tritium fuel required to sustain their own operation?

3. 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?

4. 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?

5. 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 R&D pathway scenarios will not require that early technology facilities operate in a reactor-relevant physics mode. For example, some pathways include FERFs, which do not necessarily feature reactor-relevant fusion plasma physics but operate in parallel with reactor-relevant plasma burning experiments (PBXs).

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,[3] with a 500-MW fusion power level and an ultimate capacity factor of 40% would burn ~12 kg of tritium per year. As shown in Fig. 26, a steady supply of 2.4 kg/yr tritium from the Canadian nuclear program,[29] starting in 1988, would sustain this level of operation only if an internal TBR of ~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 to $20 000/g, the annual operating cost for tritium alone would be in the range of $120 to $240 million/yr.

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

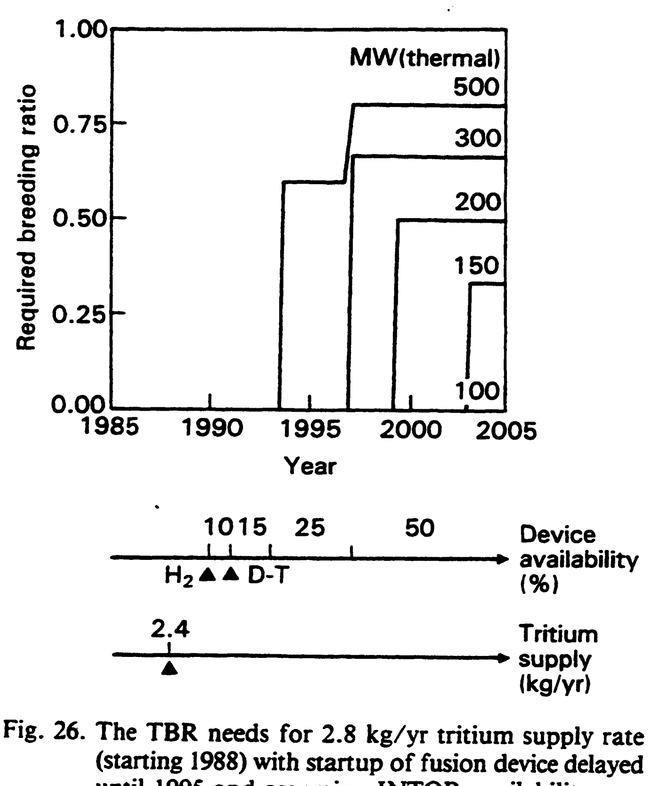



Fig. 26. The TBR needs for 2.8 kg/yr tritium supply rate (starting 1988) with startup of fusion device delayed until 1995 and assuming INTOR availability.

The number of combinations of blanket coolants, tritium breeding materials, and structural materials presently being considered by the fusion community is enormous[7] and the cost impact of continuing to carry several options will increase as the program progresses to performing blanket experiments on fusion technology facilities. 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. Some possible objectives are:

1. 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 the feasibility and attractiveness of candidate concepts can be evaluated and compared.

2. Identify design flaws and early failure modes through BOL screening tests.

3. Implement reliability improvement goals in addition to the identification of early design flaws and failure modes by conducting iterative design/test/fix/test sequences.

4. Generate sufficient data to support a high statistical confidence in the reliability of components by 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 on 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 that is 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:

1. $M = 1$ 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.
2. $M = 2$ or 3 should significantly reduce the above concerns.
3. $M > 3$ will provide good test statistics but 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 sources (e.g., FMIT class but possibly more modest) that 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 that 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 on 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 that will be used in the first generation of fusion facilities.

## VII.B. Fusion Experimental and Test Facilities

As a first step toward the evaluation of fusion technology development scenarios, a set of generic definitions for major fusion facilities was developed. These definitions, summarized in Table XXII, 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 that are included in the preliminary evaluation move directly from a PBX to an ETF followed by an EDF. Other pathways include an early FERF, but eliminate the first-generation ETF and replace it with a more advanced ETF that can be upgraded to a first-generation 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 XXII.

Other facilities, which reflect possible upgrades, are:

1. SFE/SFE-U: Scientific feasibility experiments (SFEs) such as TFTR and MFTF-B might be upgraded for scenarios that 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.
2. 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 that would be a key part of the FERF experimental program, but other components would be investigated, including PICs, I&C 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 ~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.

## VII.C. Fusion Development Pathways and Evaluations

A list of generic fusion development pathways is provided in Table XXIII. Typical examples for

TABLE XXII

Descriptions of Major Fusion Development Pathway Facilities

| | PBX | FERF | ETF | 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 |
| *Minimum* fluence/flux goals | Negligible | ~1 MW·yr/m$^2$ at ~1 MW/m$^2$ | ~4 MW·yr/m$^2$ at ~1.5 MW/m$^2$ | ~8 MW·yr/m$^2$ at ~3 MW/m$^2$ |
| *Minimum* 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 test (≥4 MW·yr/m$^2$) can be risky |
| Facility examples | TFCX (Ref. 2), LITE (Ref. 30), MFTF-α+T (Ref. 23) | MFTF-α+T (Ref. 23), TDF (Ref. 22), FED-R (Ref. 25) | INTOR (Ref. 3) and FPD (Ref. 31) | DEMO (Ref. 32) |

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 XXIII) 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 Sec. VI.B).

Three pathway milestones that provide a uniform set of development pathway goals have been defined:

1. *Engineering feasibility milestone:* achievement of a 2 MW·yr/m$^2$ neutron fluence over several tests and test articles in a FERF or ETF.

2. *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%.

3. *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.

"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 EOL conditions in the first commercial-scale fusion reactor. The years in which these milestones are achieved depend on 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.

The following tokamak R&D pathways were subjected to a *preliminary* evaluation: A1, B1, D2, E2, and F2. Pathways A1 and B1 are "conventional" pathways that 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

TABLE XXIII

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-$\alpha$ → FPD → EDF

C. SFE → PBX/FERF → ETF/EDF
  1. T: TFTR → TFCX/TFCX-U → ETF/EDF
  2. M: MFTF-B → MFTF-$\alpha$+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. $\left.\begin{array}{r}\text{SFE} \rightarrow \text{PBX}\\ \text{FERF}\end{array}\right\} \rightarrow \text{ETF} \rightarrow \text{EDF}$
  1. T: $\left.\begin{array}{r}\text{TFTR} \rightarrow \text{TFCX}\\ \text{MFTF-}\alpha\text{+T}\end{array}\right\} \rightarrow \text{ETF} \rightarrow \text{EDF}$
  2. T: $\left.\begin{array}{r}\text{TFTR} \rightarrow \text{TFCX}\\ \text{TDF}\end{array}\right\} \rightarrow \text{ETF} \rightarrow \text{EDF}$
  3. M: None

F. $\left.\begin{array}{r}\text{SFE} \rightarrow \text{PBX}\\ \text{FERF}\end{array}\right\} \rightarrow \text{ETF/EDF}$
  1. T: $\left.\begin{array}{r}\text{TFTR} \rightarrow \text{TFCX}\\ \text{MFTF-}\alpha\text{+T}\end{array}\right\} \rightarrow \text{ETF/EDF}$
  2. T: $\left.\begin{array}{r}\text{TFTR} \rightarrow \text{TFCX}\\ \text{TDF}\end{array}\right\} \rightarrow \text{ETF/EDF}$
  3. T: $\text{TFTR} \rightarrow \left.\begin{array}{r}\text{TFCX}\\ \text{FERF (Tokamak)}\end{array}\right\} \rightarrow \text{ETF/EDF}$
  4. M: $\text{MFTF-B} \rightarrow \left.\begin{array}{r}\text{MFTF-}\alpha\\ \text{TDF}\end{array}\right\} \rightarrow \text{ETF/EDF}$

G. $\left.\begin{array}{r}\text{SFE} \rightarrow \text{PBX}\\ \text{FERF}\end{array}\right\} \rightarrow \text{EDF}$
  1. T: $\text{TFTR} \rightarrow \left.\begin{array}{r}\text{TFCX}\\ \text{FERF (Tokamak)}\end{array}\right\} \rightarrow \text{EDF}$
  2. M: $\text{MFTF-B} \rightarrow \left.\begin{array}{r}\text{MFTF-}\alpha\\ \text{TDF}\end{array}\right\} \rightarrow \text{EDF}$

assumed in all pathways that the issue of irradiation damage calibration is adequately resolved prior to construction of the EDF.

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 XXIV, pathway A1, featuring a 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 ~5 yr, 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 that appear to be promising include those that feature a PBX and a low fluence FERF (e.g., MFTF-$\alpha$ + T), 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.

### VII.D. International Implications

Clearly, enhancing the fusion technology R&D effort implies a near-term funding requirement that is in excess of current fusion funding levels of any of the large national and multinational programs in the United States, 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 an attempt has not been made to define an international strategy, some reasonable observations regarding possible frameworks for such a venture follow.

First, many types of experiments and facilities could be involved in an international strategy. These might include (but would not be limited to) nonnuclear test stands, partially integrated test facilities, point accelerator neutronics facilities, fission test reactors, accelerator-based irradiation damage facilities, SFEs for advanced confinement concepts, PBXs for established confinement concepts, and FERFs.

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.

TABLE XXIV

Summary of Preliminary Evaluations

| | Pathway | | | | |
|---|---|---|---|---|---|
| | A1 | B1 | D2 | E2 | F2 |
| Fusion facilities | SFE-U<br>ETF<br>EDF | PBX<br>ETF<br>EDF | SFE-U<br>FERF<br>ETF<br>EDF | PBX<br>FERF<br>ETF<br>EDF | PBX<br>FERF<br>ETF/EDF |
| Overall operational risk | High | Moderate to high | Moderate to high | Low | Low |
| Nuclear testing/development risk | High | High | Low to moderate | Low to moderate | Low to moderate |
| Engineering feasibility milestone date | 2005 | 2008 | 2002 | 2002 | 2002 |
| Engineering demonstration milestone date | 2020 | 2023 | 2028 | 2028 | 2023 |
| Near-term funding required | Low to moderate[a] | Moderate | Moderate | High | High |
| Overall funding[b] required | Low | Moderate | Moderate to high | High | Moderate |
| 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.

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

1. Several nations could jointly sponsor the same, shared facilities and experimental programs.
2. Individual nations could construct and operate separate but complementary facilities.
3. 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.

## VIII. 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. Second, many fusion nuclear components perform multiple new and unique functions, e.g., simultaneous tritium production and extraction, 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
3. safety and environmental impact, a crucial acceptance criterion for the public.

During the first year of FINESSE, the principal nuclear technology issues were 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 Sec. II and are documented in detail in the FINESSE interim report.

New knowledge is required to understand and resolve these known and unknown fusion nuclear issues. New knowledge can be acquired only through new experiments accompanied by intensive theoretical and modeling 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 (a) basic, (b) single-effect, (c) multiple-effect/multiple interaction, (d) partially integrated, and (e) 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 bounded 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. The 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 D-T 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 nonnuclear 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 nonnuclear test stands can satisfy an important part of fusion nuclear technology R&D. It appears, however, 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 $1 \times 1 \times 0.5$ 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/s, 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 R&D 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 increase (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. Table XXV shows preliminary conclusions on requirements for the primary parameters of a fusion test device that appear at 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 XXV indicate the importance of high-power density (wall load ~2 to 3 $MW/m^2$), long plasma burn time (>500 s), and surface area available for testing (~10 to 15 $m^2$) in a fusion test device. High fluence (4 to 10 $MW \cdot yr/m^2$) is important for near-EOL prediction, but critical information about many interactive effects that are critical to feasibility issues can be learned at lower fluences (~1 to 2 $MW \cdot yr/m^2$).

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 overall fusion R&D. 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.

TABLE XXV

Preliminary Requirements on Key Parameters of a FERF

Wall load
- Minimum: >1 $MW/m^2$
- Substantial benefits: 2 to 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 $W/cm^2$
  TMR 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 to 2 $MW \cdot yr/m^2$
  Very important: 2 to 4 $MW \cdot yr/m^2$
  Important: 4 to 6 $MW \cdot yr/m^2$
  Desirable: 6 to 10 $MW \cdot yr/m^2$

Minimum size of test assembly
- Interactive tests:
  ~$0.2 \times 0.2 \times 0.1$ m
- Integrated tests:
  $1 \times 1 \times 0.5$ 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 to 20 $m^2$

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, Next European Torus (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 to 600 MW) to achieve ignition and/or reasonably high wall loads (>1 MW/m$^2$), but requires low fluence (<0.01 MW·yr/m$^2$). On the other hand, nuclear technology experiments require low fusion power (~20 to 50 MW) but high fluence (~2 to 10 MW·yr/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 (greater than $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 outside the scope of this work, 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. A preliminary scoping study of the potential of tokamaks and tandem 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 of fusion power, an ~1.15 MW/m$^2$ neutron wall load, an ~1000-s pulse length, a major radius of 2.55 m, and an aspect ratio of 3.4. The device requires an average beta of ~23% and circulating power of ~190 MW. While such a device appears considerably more attractive than previous tokamak test facility designs, its capital 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 to 50 MW of fusion power at ~1 to 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 tandem 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. 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 to deepen our understanding of the implications of various options for fusion R&D 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 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 a partially integrated test facility (PITF). This is a facility 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 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. 27. 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.

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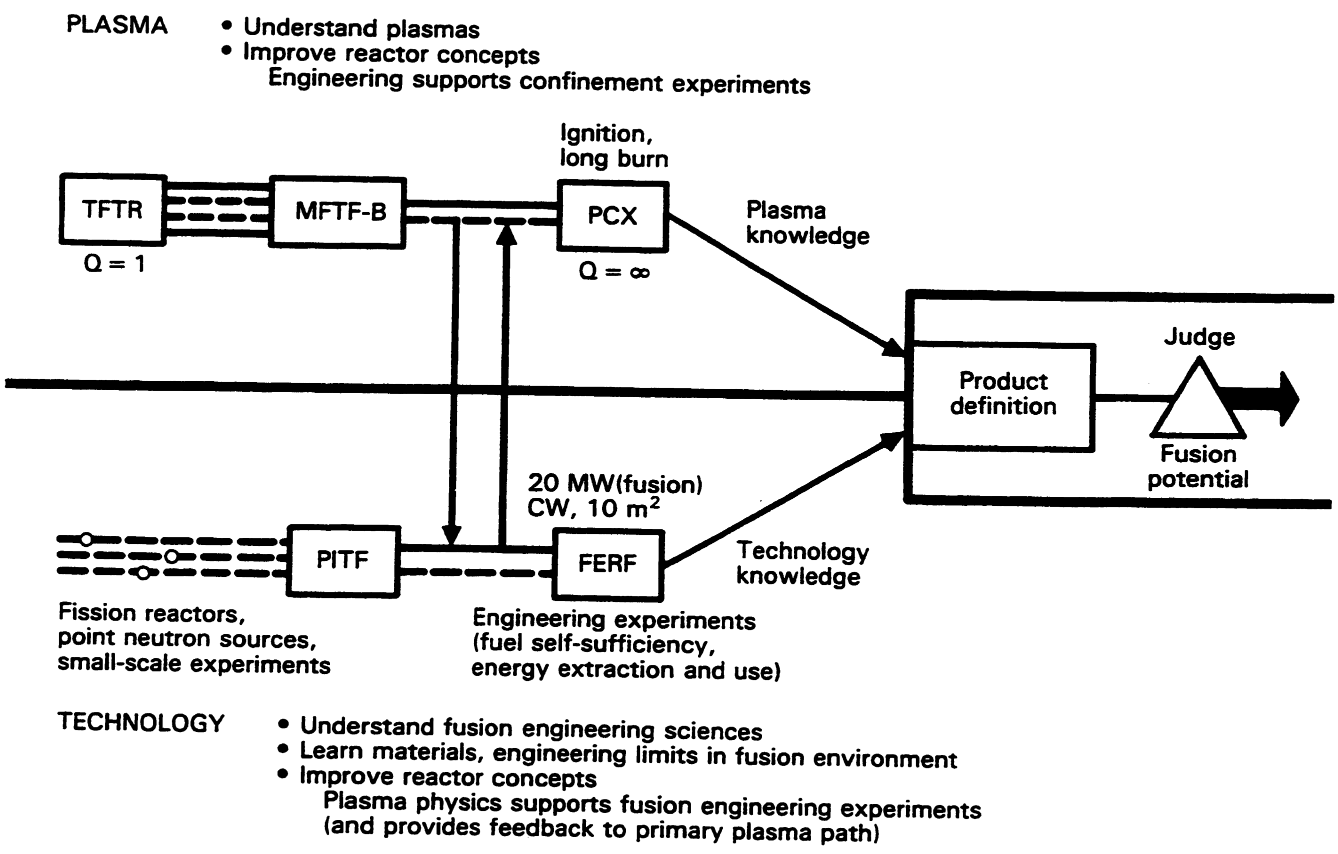


Fig. 27. Key elements of a plausible approach to the next step in fusion R&D.

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