Volume I
Summaries

PPG-909
UCLA-ENG-85-39

Technical Issues and Requirements of Experiments and Facilities for Fusion Nuclear Technology

FINESSE Phase I Report

December 1985

Fusion Engineering
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Volume I

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FINESSE Phase I Report

Technical Issues and Requirements of Experiments and Facilities for Fusion Nuclear Technology

Volume I

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CHAPTER 1

INTRODUCTION AND SUMMARY

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1. INTRODUCTION AND SUMMARY

1.1 Introduction

The magnetic fusion programs worldwide are all directed toward the eventual development of viable and attractive commercial fusion reactors. Fusion nuclear technology is critical to the accomplishment of this goal because it poses major engineering feasibility issues, and because it will very strongly impact fusion's ultimate economic, safety and environmental attractiveness. Enhanced research and development programs on fusion nuclear technology are necessary now because: 1) long lead times are required to perform the necessary experiments and obtain an adequate data base, and 2) early results are essential to defining major characteristics of viable and attractive fusion reactors, and hence providing timely feedback to plasma physics and confinement experiments.[1-10]

The development of a new technology, such as fusion energy, starts by a proposed application of a scientific principle and, if successful, ends with a commercial product. In between, there are many scientific and engineering activities whose characteristics depend on the specific technology being developed. However, three particularly important activity elements always take place, as illustrated in Fig. 1-1.

The first element involves conceptual design studies. In these studies, design options are examined and compared, based generally on a very limited data base. The product of design studies is an identification of those design concepts that appear to be promising, together with a preliminary description of such designs and their estimated performance. Information from design studies is necessary but not sufficient to implement a research and development (R&D) program. R&D implementation, which is the third activity in Fig. 1-1, refers to the step of constructing experimental facilities and performing experiments.

Experiment Planning, as illustrated in Fig. 1-1, is an important activity element in technology development and provides a crucial link between design studies and R&D implementation. The purpose of Experiment Planning is to develop an optimal R&D strategy based on detailed technical evaluations of key R&D issues, of experiments required to resolve the issues, and of capabilities and limitations of testing facilities.

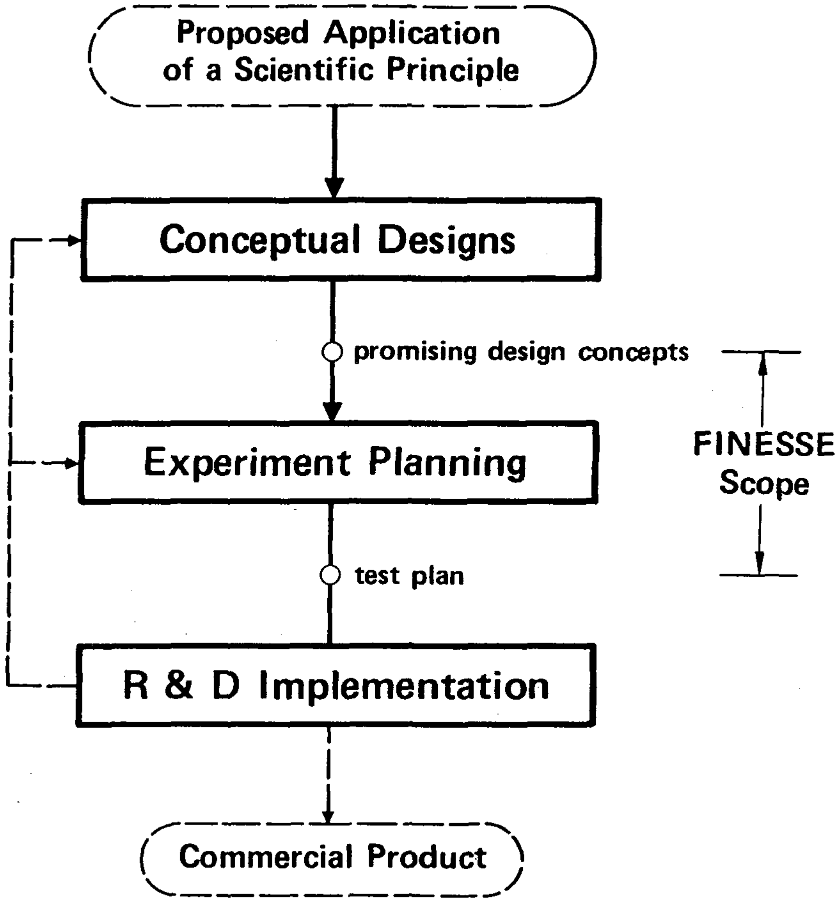


Figure 1-1. Role of Experiment Planning in technology development and illustration of the scope of FINESSE

FINESSE is concerned with evolving and performing an Experiment Planning process for fusion nuclear technology.(1) The primary fusion reactor components included in nuclear technology are those whose main functions are: 1) fuel production and processing, 2) energy extraction and use, and 3) radiation protection of personnel and components. These include blanket, plasma interactive components (such as first wall, limiter and divertor), radiation shield and tritium system. Non-nuclear components that are significantly affected by the nuclear environment include instrumentation and control, magnets, remote maintenance, and heat transport systems.

FINESSE Approach

An approach for Experiment Planning has been developed in FINESSE. The main elements of this approach, which is referred to as the FINESSE process, are shown in Fig. 1-2.

The primary input to the process is a set of promising design options for a particular technology component. The major output from the process is a technical test plan that identifies and quantifies the role, timing and characteristics of major experiments and facilities. The FINESSE process consists of four primary steps indicated in Fig. 1-2, namely: 1) characterization of issues, 2) quantification of experimental needs, 3) evaluation of facilities, and 4) development of a test plan. Experience from other technologies is an important input to the process, particularly in quantifying experimental needs and developing engineering scaling options. Programmatic considerations are important primarily for the last step concerned with the development of a test plan. The four steps in Fig. 1-2 are generally carried out sequentially, but considerable feedback and iterations among the steps have proved necessary. A brief summary of the focus of the technical effort in each step is given below.

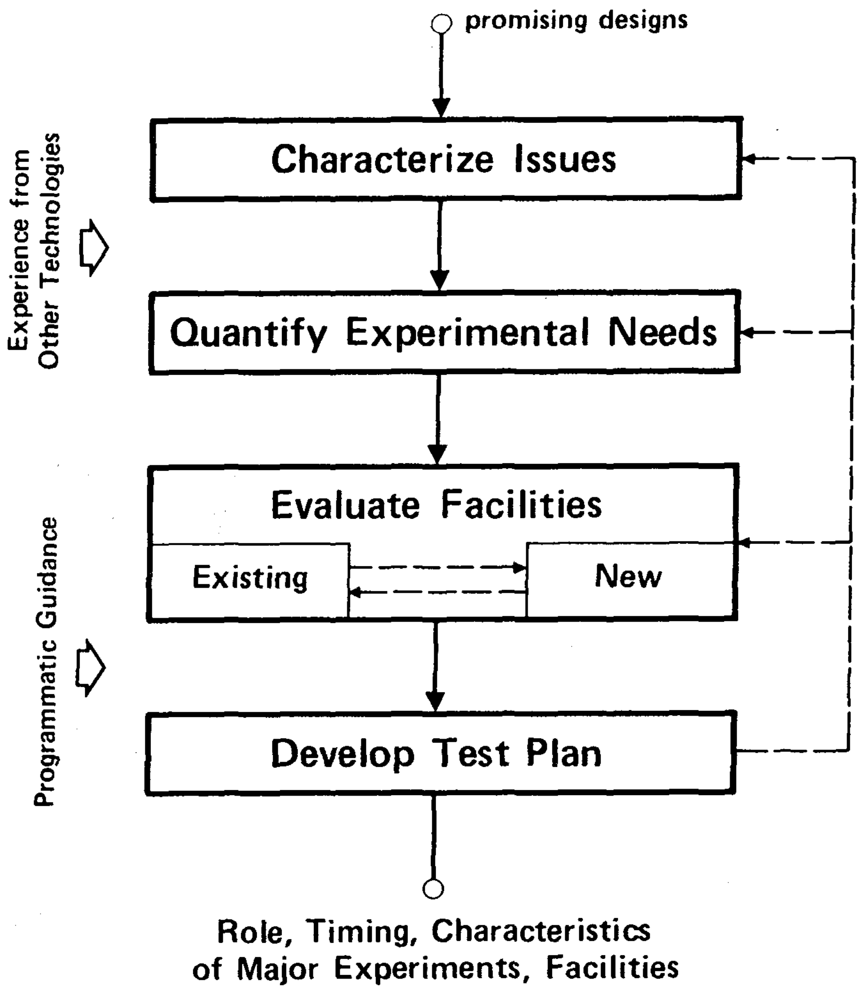


Figure 1-2. FINESSE process for Experiment Planning

The first step, which is concerned with the characterization of issues, involves the following technical investigations:

- assessment of accuracy and completeness of existing data and models;
- analysis of scientific/engineering phenomena to determine (anticipate) behavior, interactions and governing parameters in the fusion reactor environment;
- evaluation of the effect of uncertainties on design performance; and
- comparison of tolerable and estimated uncertainties.

This process element provides quantified understanding of the issues and their relative priorities.

The second step, which focuses on the quantification of experimental needs, involves:

- survey of needed experiments;
- exploration of engineering scaling options (engineering scaling is a process to develop meaningful tests at experimental conditions and parameters less than those in a reactor);
- evaluation of effects of scaling on usefulness of experiments in resolving issues;
- development of technical test criteria for preserving design-relevant behavior; and
- identification of desired experiments and key experimental conditions.

In the third step, the effort for evaluating facilities is directed initially at existing facilities and consists of: a) survey of the available facilities; b) evaluation of their capabilities and limitations; c) definition of meaningful experiments to be performed in such facilities; and d) estimation of costs for such experiments. Issues that cannot be resolved in existing facilities require the construction of new facilities. In evaluating the need for and in identifying new facilities, the effort is focused on: 1) exploring innovative testing ideas; 2) assessing the feasibility of obtaining the desired information, e.g., examining instrumentation limitations; 3) developing preliminary conceptual designs of facilities and estimating their costs; and 4) performing tradeoffs among experiments and facilities using parameters such as technical usefulness, time and cost.

In addition to information from the first three steps, the final step of developing a test plan requires input on programmatic considerations such as assumptions on budget and time constraints. The approach in this step depends on the complexity of the issues and the level of detail required in the test plan. In general, the approach involves developing a number of test program scenarios and comparing them in terms of risk, usefulness and cost.

Goals, Objectives and Assumptions

A principal goal of FINESSE is to provide recommendations, based on technical evaluations, for the types, sequences, and characteristics of major experiments and facilities that maximize technical benefits and minimize cost in a logically consistent path for fusion nuclear technology development. The ultimate goal of fusion R&D is the development of commercial fusion reactors. The FINESSE work completed to date and presented in this report has focused on R&D for the next fifteen years. The objective set for about the year 2000 is to provide adequate data base and prediction capability to permit: 1) a quantitative assessment of fusion energy economic, safety and environmental impact potential, and 2) the design and construction of experimental modules for testing in a fusion facility.[3,6]

The study has attempted to limit to a minimum the number of restrictive assumptions on the major characteristics of a commercial fusion reactor in order to assure the applicability of recommendations to a broad-based fusion technology development program. Nevertheless, a number of assumptions were made in the investigations presented in this report to keep the effort manageable. The key assumptions are:

- Electricity production is the primary purpose of the reactor. The impact of non-electric applications, e.g., hybrids, on nuclear technology R&D has not yet been investigated.

- Tokamaks and tandem mirrors are considered as the primary confinement concepts. Changes in R&D for reversed-field pinches[10] (RFP's) need further evaluation.

- Representative range for key parameters of commercial fusion reactors[7-9] considered in this work is shown in Table 1-1. The impact of variations in these parameters is discussed in the appropriate technical areas of the report.

- In developing a specific time schedule for the next 15-year test plan, it was assumed that no fusion device would be available for nuclear technology testing long before the year 2000. This assumption impacts primarily the pace rather than the type of near-term R&D activities.

Table 1-1. Representative Goal Ranges Considered in this Work for Commerical Reactor Parameters

Parameter	Range
Neutron Wall Load, MW/m^2	4-6
Surface Heat Flux at First Wall, MW/m^2	0.2-1
Average Heat Flux in High Heat Flux Components (e.g., limiter/divertor), MW/m^2	5-10
Plasma Burn Time	very long/continuous
Magnetic Field Strength in Blanket Region, T	5-7
Reactor Availability, %	80
First Wall/Blanket Lifetime Fluence, $MW{\cdot}y/m^2$	15-20

Types of Experiments and Facilities

The types of experiments required for a fusion nuclear component, e.g., blanket, can be classified into: 1) basic, 2) separate effect, 3) multiple interaction, 4) partially integrated, and 5) integrated tests. Figure 1-3 illustrates the role of these types of experiments and the strong interrelation between experiment and analytic modelling.

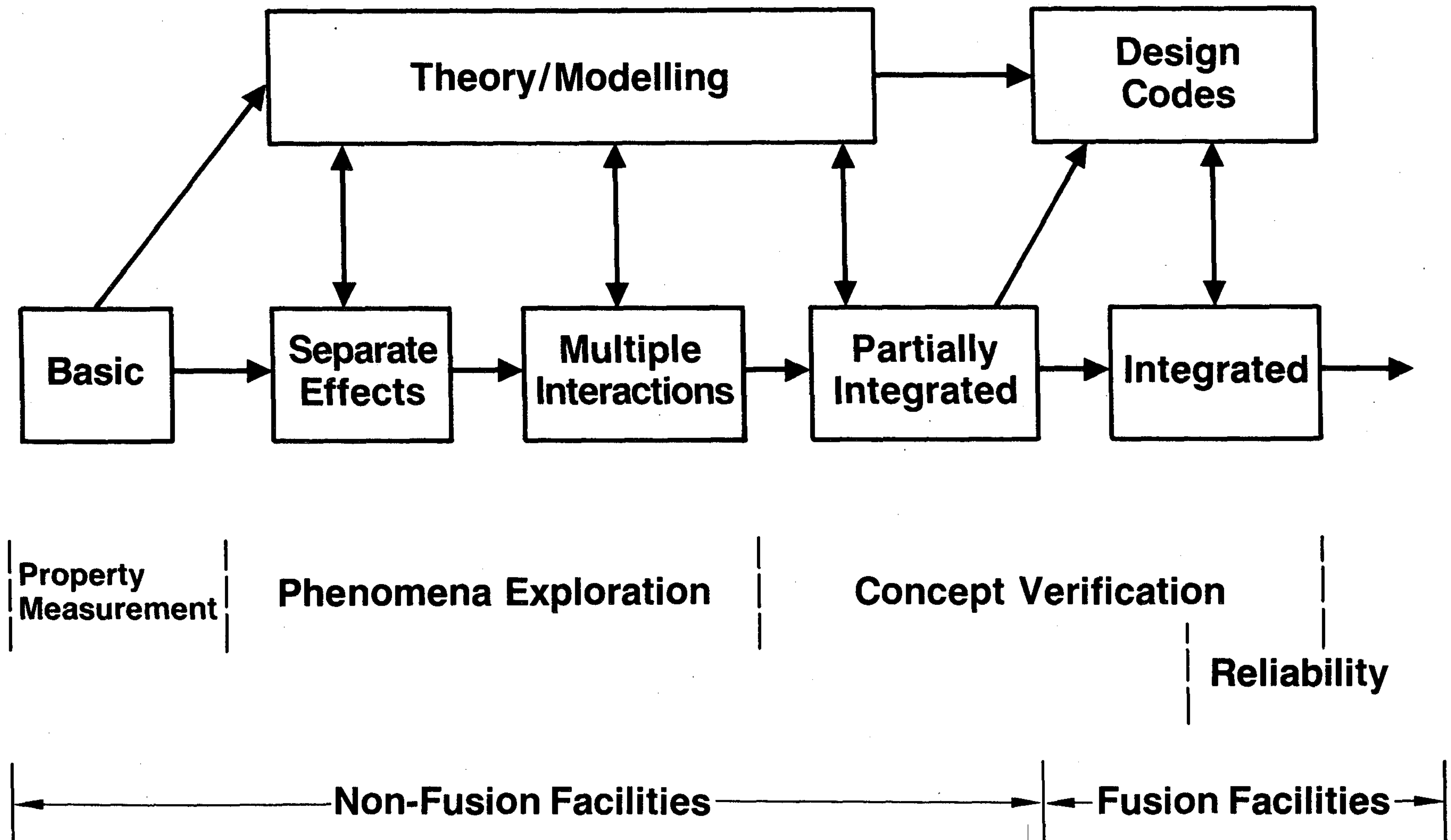


Figure 1-3. Types and role of experiments and facilities for Fusion Nuclear Technology

Basic tests measure basic property data. Single effect tests are experiments with a single environmental condition aimed at developing an understanding and models of a single phenomenon. Multiple interaction tests involve both interactions among the effects of multiple environmental conditions as well as direct interactions among different physical elements of the component. Partially integrated tests attempt to obtain integrated test information but without some key environmental condition. In integrated tests, all environmental conditions and physical elements are present.

The above classification is based on the degree to which environmental conditions (e.g., magnetic field, bulk heating, neutrons) and the physical elements (e.g., breeder, structure, coolant) of the component are simulated (or present) in the experiment. It should be noted that the level of integration in actual experiments spans a continuum and each of the above classifications represents a range of conditions.

As the level of integration in the experiment increases, more synergetic effects are observed, and the emphasis shifts from understanding and theoretical modelling to obtaining engineering data and empirical correlations.

The level of integration necessary for a design concept to be verified depends on the complexity of the component. For fusion nuclear components such as the blanket, it has been concluded that concept verification is unlikely prior to performing fully integrated tests.(1)

Basic, separate effect and multiple interaction experiments can be performed in non-fusion facilities. Completely integrated tests are possible only in fusion facilities. Beyond concept verification, the primary purpose of testing in a fusion device is to obtain data on component reliability.

Non-fusion facilities can be classified into:

- non-neutron test stands
- neutron producing facilities
  - fission reactors
  - accelerator-based neutron sources

For the purpose of the work reported here, a fusion facility can be any fusion device that is useful for nuclear technology testing.

International Cooperation

International cooperation has long been recognized as an important mechanism for maximizing progress in fusion.(2) There are particularly strong incentives for pursuing international cooperation on fusion nuclear technology (FNT). Among these reasons are:(2)

a) There are many areas of key R&D needs for FNT that are of common interest to all countries. These areas of common interest constitute opportunities for international cooperation derived from strong technical needs.

b) Substantial resources in terms of manpower and facilities are required to resolve the key FNT issues. International cooperation is thus desirable as a cost-effective, and in some cases necessary, means to conduct the R&D required in the many areas of FNT.

c) International cooperation can be an excellent mechanism to accelerate progress and enhance the prospects for success in development of credible and attractive fusion nuclear components. Effective coordination of intellectual and hardware resources in the world programs will permit more complete and faster exploration of promising options, identification of critical problems and development of attractive solutions.

A strong awareness of the importance of international cooperation has existed among FINESSE participants from the early phases of the study.(1) Therefore, the study has emphasized communication with scientists and engineers outside the U.S. The study has also attempted to maximize the usefulness of its results to the international community by emphasizing technical issues, design concepts and facilities that appear to be of global interest, and by avoiding overly restrictive development strategies or budget scenarios.

Organization

The study is led by UCLA and involves the following major organizations from the U.S.: Argonne National Laboratory (ANL); Hanford Engineering Development Laboratory (HEDL); TRW, Inc.; EG&G Idaho; Grumman Aerospace Corporation; McDonnell Douglas Astronautics Company, Sandia National Laboratory and Los

Alamos National Laboratory. Major support has also been provided to the study by Lawrence Livermore, Princeton Plasma Physics and Oak Ridge National Laboratories. An advisory committee consisting of senior members of the fusion community has provided an excellent mechanism for community-wide input to FINESSE.

FINESSE has benefited considerably from the productive participation of a number of scientists and engineers from the Canadian Fusion Fuels Technology Project, Japan Atomic Energy Research Institute, University of Tokyo, University of Kyoto, and Karlsruhe Nuclear Research Center. The presence of experts from outside the U.S. has helped FINESSE identify and address many technical areas of common interest to the international fusion community.

Scope of This Report

FINESSE was initiated approximately two years ago. The results of the first year effort were reported in an interim report in October 1984.(1) The interim report focused on: 1) detailed characterization and prioritization of technical issues; 2) investigation of general testing needs and quantification of key testing requirements; 3) evaluation of experience from other (fission and aerospace) technologies; and 4) evaluation of the capabilities and limitations of existing facilities.

This report presents results from the second year of the study. The focus of this report is on defining the role, characteristics, timing and costs of major experiments and facilities required over the next fifteen years for fusion nuclear technology development. This report has been written as a stand-alone document. Important information from the interim report has been briefly summarized wherever necessary. However, no attempt was made to reproduce the detailed technical analyses given in the interim report.

The report is divided into two volumes. Volume I has two chapters. Chapter 1 is a technical summary of all technical areas. Blanket R&D involves many complex problems and has received a significant part of the study effort. Therefore, Chapter 2 has been devoted specifically to an overview of the blanket test plan.

Volume II consists of Chapters 3 through 9, which contain detailed results in various technical areas. Chapters 3 and 4 focus on liquid metal and solid breeder blankets, respectively. Chapters 5, 6 and 7 consider, in respective order, the tritium processing system, plasma interactive components, and radiation shield. Chapter 8 addresses specific problems related to non-fusion irradiation facilities, namely fission reactors and point neutron sources.

Since the focus of the second year effort in FINESSE has been on nuclear technology R&D for the next fifteen years, most of the investigation has been concerned with non-fusion facilities. However, some effort was devoted to comparing various options for fusion test facilities. This comparative study is presented in Chapter 9.

1.2 Blanket

1.2.1 Introduction

The first wall/blanket is a particularly important fusion nuclear component that has a number of critical feasibility and attractiveness concerns. Blanket concepts can be divided into liquid breeders and solid breeders. Although the functional requirements (e.g., tritium breeding) and reactor operating conditions (e.g., neutron wall load) are similar for both classes of blankets, the critical issues are generally not. Consequently, the issues and associated experiments are discussed separately.

Within the uncertainties, it is not possible to determine whether solid or liquid breeder blankets are more attractive. Consequently, it appears prudent for the fusion program to retain both options, although a selection could be made at some point in the future when more information is available. In the test plans considered here, this selection is not explicitly made. Rather, separate test plans are presented that could develop solid and liquid breeder blanket concepts to the point of integrated fusion testing.

1.2.2 Solid Breeder Blankets

1.2.2.1 Issues and Testing Needs

The general classes of issues for solid breeder blankets are given in Table 1-2. These are based on the characteristics of solid breeder concepts (Table 1-3) from recent studies such as the Blanket Comparison and Selection Study (BCSS)[4,5]. Some of the design uncertainties resulting from these issues are large enough to make the blankets potentially impractical. The most important uncertainties are related to tritium breeding, tritium recovery, and breeder thermomechanical behavior. These are particularly large for solid breeder blankets because: 1) there is limited understanding of gas transport in irradiated solids, 2) complex designs are used to keep the low thermal conductivity solids within their temperature limits under substantial nuclear heating and neutron damage rates, and 3) the resulting designs have a significant amount of non-breeding structure, coolant, and other material.

Table 1-2. Generic Classes of Solid Breeder Blanket Issues

Tritium self-sufficiency
Breeder/multiplier tritium inventory and recovery
Breeder/multiplier thermomechanical behavior
Corrosion and mass transfer
Structural response and failure modes in fusion environment
Tritium permeation and processing from blanket

Table 1-3. Solid Breeder Blanket Characteristics Considered

Fusion-electric applications[a]
~ 5 MW/m^2 neutron wall load
~ 15 $MW\text{-}yr/m^2$ blanket fluence lifetime
Helium or water coolant
With or without beryllium neutron multiplier[b]
Austenitic (PCA) and ferritic (HT-9) structure
Separate purge and coolant streams
Breeder in plate, BIT or BOT geometry[c]

[a]Most of discussion also applicable to solid breeders for fusion hybrid blankets which operate in a similar temperature and neutron wall load regime.

[b]Only Li_2O is considered without multiplier.

[c]BIT = breeder-inside-tube; BOT = breeder-outside-tube.

For solid breeder blankets, the primary safety uncertainties are related to the behavior of the blanket under off-normal or transient conditions, and the control of tritium under normal operation. These issues would be addressed as part of the experimental and model development program within each of the technical issue areas defined in Table 1-2.

Tritium Self-Sufficiency: The tritium breeding ability of solid breeder blankets is reduced by the presence of the non-breeder materials. All solid breeder blankets are predicted to require ^6Li enrichment and a neutron multiplier, with the possible exception of Li_2O. Even so, within present uncertainties in data, modeling methods and design definition, it is not clear that any solid breeder blanket will be self-sufficient in tritium. Table 1-4 indicates the calculated 3-D tritium breeding ratio (TBR) for several BCSS blankets, and the estimated uncertainty in this TBR based on sensitivity studies. None of the blankets achieve a required TBR of 1.07 within the uncertainties.(11) The need for a neutron multiplier is a key issue for Li_2O.

Table 1-4. Achievable Tritium Breeding Ratios and Associated Uncertainties for BCSS Tokamak Blankets(11)

Concept	Achievable TBR	Uncertainty in Achievable TBR
$LiAlO_2$/salt/HT9-Be	1.24	0.22
LiPb/LiPb/V	1.30[a]	0.24
Li/Li/V	1.28	0.24
Li_2O/He/HT9	1.11	0.21
$LiAlO_2$/He/HT9/Be	1.04	0.19
Li/He/HT9	1.16	0.22
Flibe/He/HT9/Be	1.17	0.22
$LiAlO_2$/H_2O/HT9/Be	1.16	0.21

[a]Estimated for 90% ^6Li enrichment.

Tritium Recovery: The prediction of tritium behavior in solid breeder blankets requires understanding tritium transport, retention and chemical form in the breeder and multiplier material under the influence of the fusion environment. The importance and uncertainty of the various phenomena to the blanket tritium inventory is indicated in Table 1-5. The major contributors (by inventory and uncertainty) are the diffusivity, solubility and surface adsorption processes.

Tritium diffusion is anticipated to be a rate-controlling step in $LiAlO_2$ and other ternary ceramics. The uncertainty in the diffusivity can be much more than an order-of-magnitude, particularly at higher temperatures and burnups.

The TRIO experimental results(12) imply a surface adsorption in $LiAlO_2$ of between 0 and 5 wppm. The addition of sufficient protium to the purge stream can reduce the tritium surface inventory, but will affect the breeder chemical environment. This environment, and particularly the oxygen activity (effectively, the O_2 partial pressure), has a strong effect on the absorbed and adsorbed tritium.(5) However, the O_2 activity can vary over many orders-of-magnitude depending on the controlling thermodynamic system. For example, the ideal solution oxygen activity in 0.1 MPa helium at 1000 K is much less than 10^{-35} in a $LiAlO_2$/Be controlled system due to the formation of BeO; it is around 10^{-25} for equilibrium between iron and iron oxide (e.g., at the cladding surface) at 1000 K, and is 10^{-5} for a purge-controlled system with 10 ppm O_2 added. All these factors, plus others, will be present to some degree in solid breeder blankets, and the resultant local oxygen activity is not known.

Sufficient tritium is produced in the beryllium multiplier to also cause concern.(5) There is an inventory concern if the tritium simply accumulates in the beryllium, a coolant contamination problem if the tritium permeates directly into the coolant, and a breeder/multiplier chemical interaction concern if the beryllium is included in the breeder purge gas system. The same tritium transport phenomena apply as with solid breeder materials, but there is insufficient data to address their relative magnitude.

Thermal, Mechanical and Corrosion Behavior: The major issues associated with the mechanical interactions between the solid breeder, multiplier and structure are restructuring of the solid breeder, deformation and/or rupture

Table 1-5. Contributions to Blanket Tritium Inventory for BCSS[(5)] Tokamak Blankets[a]

Contributor	Blanket Inventory (g)				Uncertainty[b]
	Li_2O/He/HT9	$LiAlO_2$/He/HT9/Be	$LiAlO_2$/H_2O/HT9/Be	$LiAlO_2$/salt/HT9/Be	
Diffusivity	0.04	38	2300	2000	very large
Grain boundaries	~ 0	~ 0	~ 0	~ 0	very large
Solubility	134	~ 0	~ 0	~ 0	moderate
Surface adsorption:					large
H_2 added	~ 0	~ 0	~ 0	~ 0	
No H_2 added	1200	1200	1200	1200	
Pores/purge	0.04	0.04	~ 0	~ 0	large
Beryllium[c]	0	< 15000	< 15000	< 15000	upper bound
Coolant	0.00003	0.00001	53	500	large
First wall	19	19	19	19	very large
Blanket structure	1.1	1.2	9.3	42	very large

[a]Concept denoted by breeder/coolant/structure/multiplier.
[b]Rough estimate, varies between materials.
Moderate (≤25%), large (≤100%), very large (factor of 10)
[c]Upper limit refers to complete retention of tritium and end-of-life conditions.

of the structure, and changes in the heat transfer across the breeder/cladding interface. The primary driving forces are swelling (particularly for Li_2O and Be) and differential thermal expansion. The material can respond by deformation, creep, or fracture, but the extent of each is not known. Even though beryllium has been used in fission reactors, the available irradiated mechanical property data is generally an order of magnitude below the anticipated end-of-life blanket conditions, and the available high fluence swelling data is based on a few post-irradiation annealed specimens. Also, there are no completed experiments that indicate the extent and consequences of mechanical interactions or temperature gradients within the breeder.

The thermal behavior of the breeder is constrained by the relatively low thermal conductivity (~ 1-3 W/m-K) and upper temperature limits (~ 800-1000 °C) assumed for present solid breeder materials.(5) These are important to the blanket design, but present estimated values may be conservative. Table 1-6 illustrates the predicted large reduction in thermal conductivity for reasonable breeder conditions, and the limitations of present data. The upper temperature limits are not as easily defined, but are dependent on many processes such as sintering, creep, phase change, vapor phase transport or corrosion. Present limits are based on avoiding these processes, but it is not clear that they need have a net detrimental effect on the overall breeder performance. Particular material interaction concerns include the vapor phase transport and corrosion in Li_2O, and the kinetics of reactions between Be and the solid breeder.

Structural Response: The mechanical behavior of structural elements of the blanket determine its lifetime. Uncertainties in the loading (e.g., magnitude of magnetic field-induced forces) and response (e.g., radiation-induced creep stress relaxation, crack growth) must be accounted for by conservative designs. The mechanisms for component failure must be identified in order to determine and improve blanket reliability and safety.

Tritium Permeation and Processing: The permeation of tritium outside of the breeder zone is important for defining the coolant detritiation requirements, but the nature and effects of the chemical environment and surface conditions are uncertain. Also, uncertainties in the recovery of tritium from the primary breeder extraction stream include the incoming tritium form, the efficiency of the recovery process, and the tritium inventory in the recovery system.

Table 1-6. Variation of Thermal Conductivity with Breeder Conditions[13]

Breeder Condition[a]					Thermal Conductivity[b]	
ρ (%TD)	T (°C)	Form	Fluence (n/m^2)	p_{He} (MPa)	Li_2O (W/m-K)	γ-$LiAlO_2$ (W/m-K)
100	700	Sintered	0	--	4.8	2.6
85	700	Sintered	0	--	<u>3.8</u>	<u>2.5</u>[c]
85	400	Sintered	0	--	<u>5.2</u>	<u>2.5</u>
85	700	Sintered	10^{26}	--	2.6	1.6
87	700	Sphere-pac	0	0.1	1.4	1.2
87	700	Sphere-pac	0	0.6	2.2	1.7
87	700	Sphere-pac	10^{26}	0.1	1.2	1.1

[a]Density is % theoretical density;
Sphere-pac form is 100% TD spheres, 30-, 300- and 1200-μm diameters.
Fluence is with respect to fast neutrons.

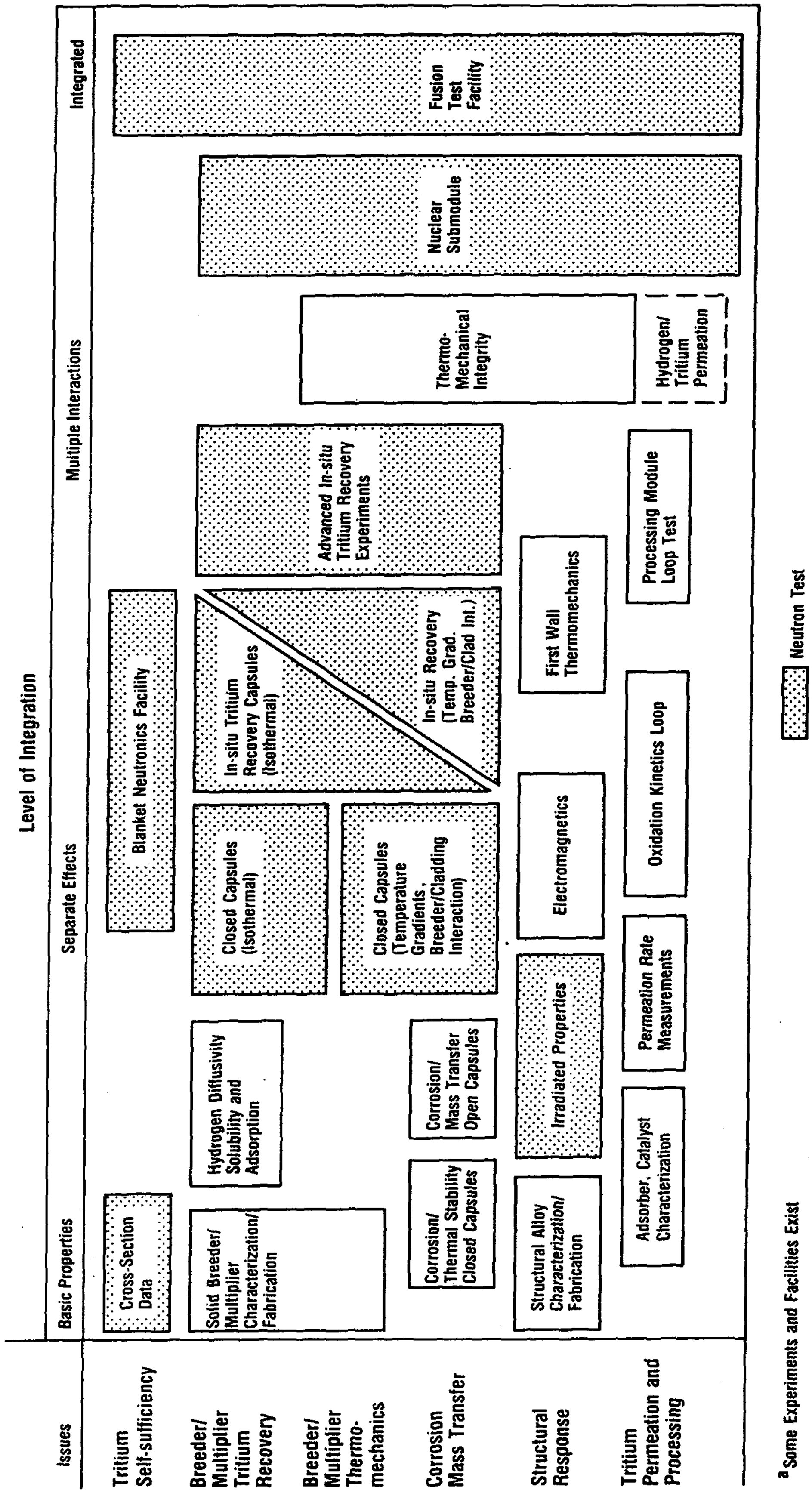
[b]Underlined values are measured, the rest are extrapolated.

[c]From recent data;[14] previously estimated value was 1.9 W/m-K.

1.2.2.2 Existing and Required Experiments and Facilities

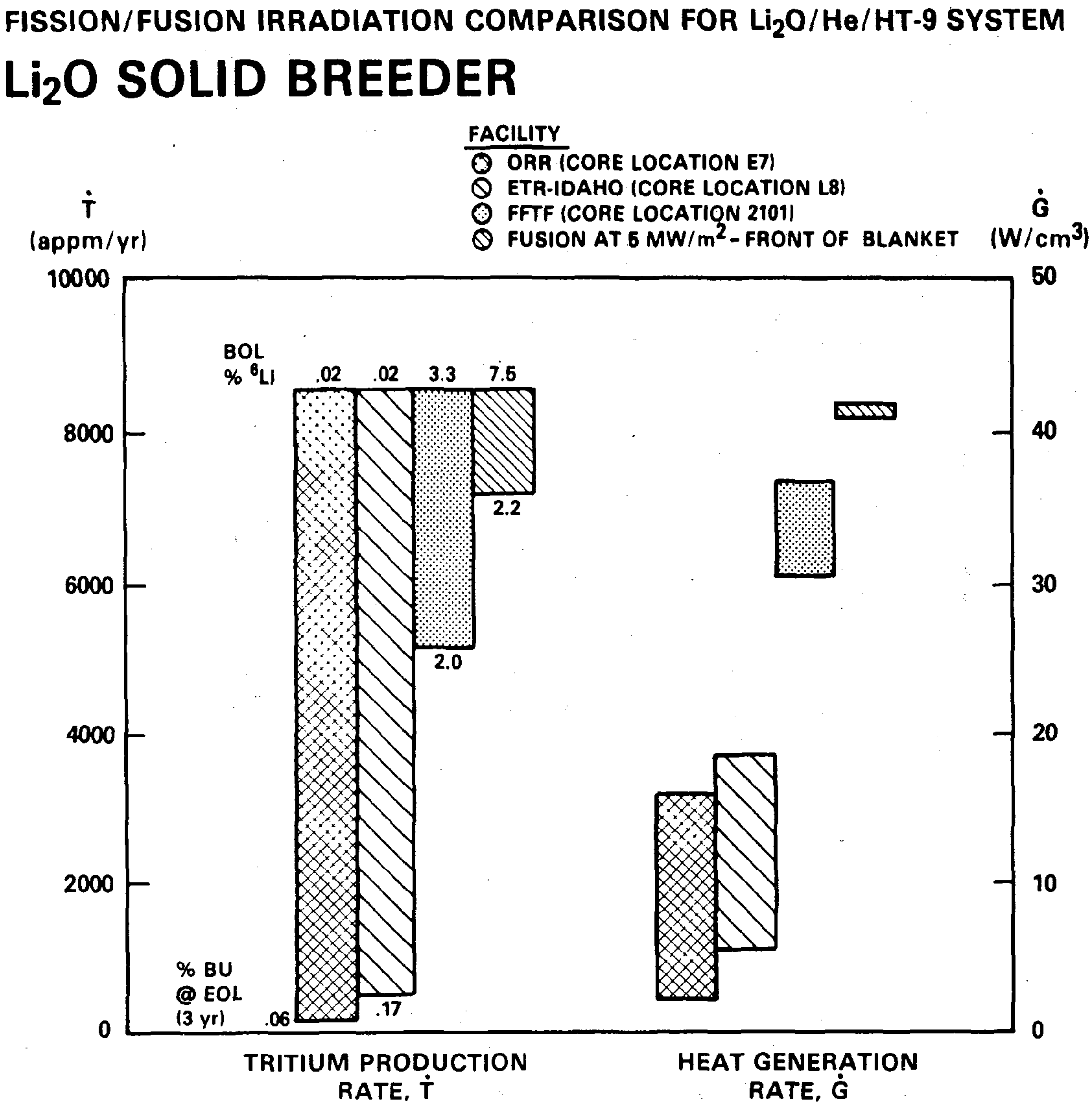
The issues can be addressed by a range of possible experiments as summarized in Fig. 1-4 and discussed below. The actual experiments will depend on particular test program assumptions and funding constraints. These tests are organized according to their level of integration, from basic properties, to phenomena exploration in separate and multiple effect tests, to concept verification in integrated fusion tests. In general, more than one experiment is needed to fully address each issue.

Since there is no general theoretical basis for scaling solid breeder behavior, the significant phenomena must be quantified by conducting tests at reactor-relevant conditions. Among the most important parameters are the tritium generation and heating rates. The ability of the ORR and ETR thermal reactors and of the FFTF fast reactor to match fusion conditions is shown in Fig. 1-5. By appropriate matching of ^6Li enrichment in the breeder material and the reactor, it is possible to simulate fusion tritium generation and heating rates within a factor of two. Furthermore, reactors with vented test



[a] Some Experiments and Facilities Exist

Figure 1-4. Types of experiments and facilities for solid breeder blankets[a]

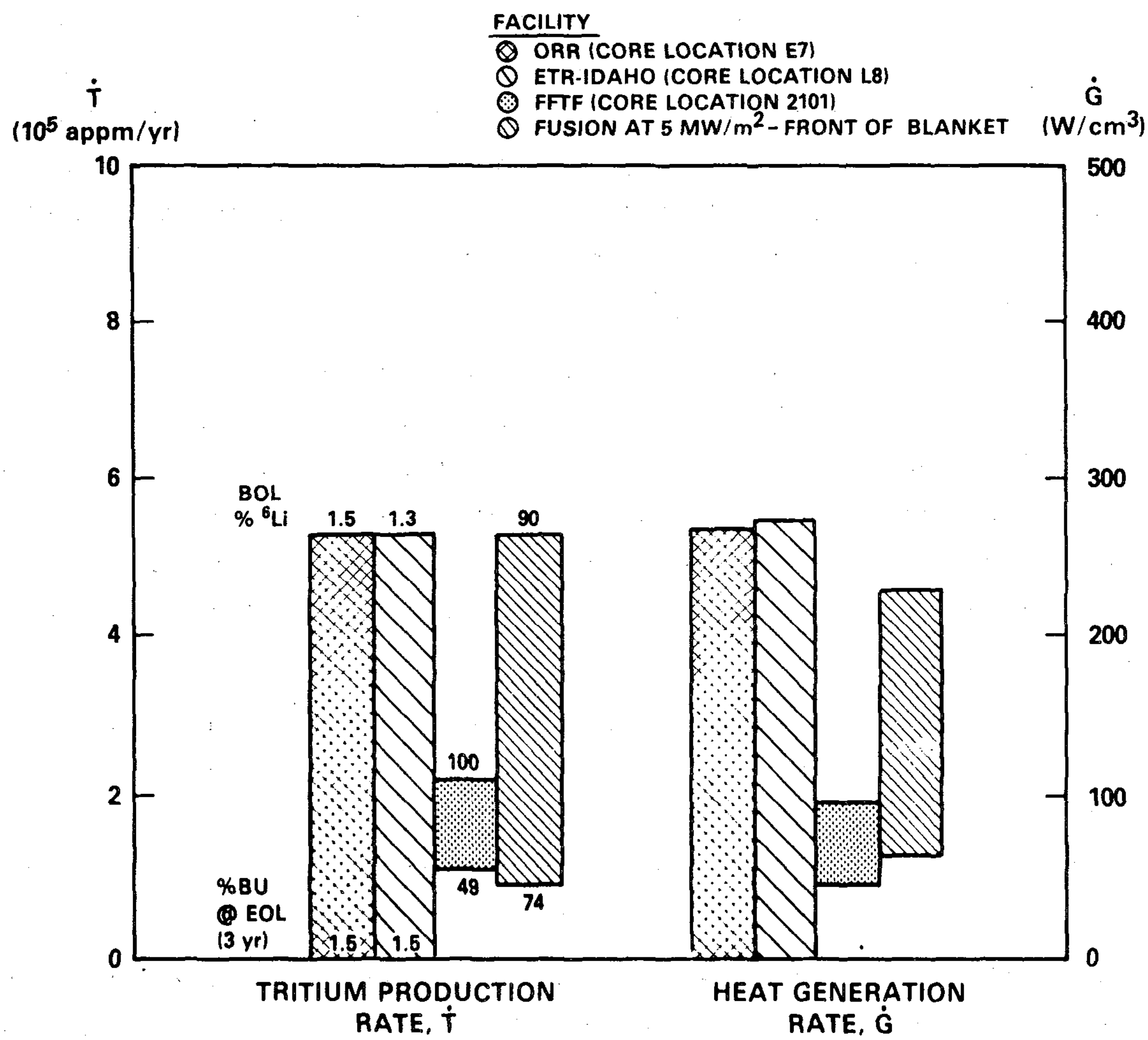


(a)

Figure 1-5(a). Comparison of tritium generation and heating rates for solid breeder irradiation in thermal, fast and fusion reactors, for Li_2O.

FISSION/FUSION IRRADIATION COMPARISON FOR $LiAlO_2/H_2O$/HT-9/Be SYSTEM

$LiAlO_2$ SOLID BREEDER



(b)

Figure 1-5(b). Comparison of tritium generation and heating rates for solid breeder irradiation in thermal, fast and fusion reactors, for $LiAlO_2$.

capabilities can also provide direct simulation of the purge environment. Therefore, nuclear testing in existing fission reactors is an important resource for solid breeder blankets.

Material Development and Characterization

The development of an attractive blanket depends strongly on the development of attractive blanket materials, particularly the solid breeder material itself. Material development refers to the process of identifying possible classes of materials, understanding the effects of material parameters on the properties, fabricating materials with the desired material parameters, and characterizing the material through measurement of its properties. This process continues throughout the overall experimental program, but the identification of desirable materials and material parameters is most effective when available early in the program.

Material parameters include the type of compound (e.g., Li_2O, Li_2ZrO_3, Li_8ZrO_6), crystal form (e.g., γ- or α-$LiAlO_2$), grain size, pore size, form, impurity and additive content, phase purity, and fabrication process. Lithium-bearing materials under active consideration include lithium oxides, aluminates, silicates, zirconates, and beryllates. Various completed and active irradiation experiments to characterize and understand these material parameters are summarized in Table 1-7.

The most important needs at present are for basic properties of all compounds (particularly measurements of tritium diffusion, tritium absorption, thermal conductivity, swelling, and thermal stability), the fabrication of sphere-pac forms, and an understanding of the importance of the various material parameters to the solid breeder properties.

All solid breeders require a neutron multiplier, with the possible exception of Li_2O. It is prudent to assume that some multiplier will be needed. While beryllium is the preferred material, the form of incorporating it into the blanket is uncertain (e.g., separate or mixed with breeder). Questions related to mechanical behavior, tritium retention and compatibility with the breeder need to be resolved, including the effects of material parameters. There is limited fusion-relevant data, and very few active experiments.

Table 1-7. Completed and Active Solid Breeder Material Irradiation Experiments[2,5,15]

Experiment	Ceramic	Grain size (μm)	Density (%TD)	Temperature (°C)	Li burnup (Max at.%)	Time Frame
Closed Capsule Experiments						
ORR (US)	Li_2O	< 47	70	750,850,1000	0.05	-
TULIP (US)	Li_2O	50	87	600	3	84
FUBR-1A (US)	Li_2O	6	85	500,700,900	1.5	84/85
	$LiAlO_2$	< 1	85,95	500,700,900	3	84/85
	Li_4SiO_4	2	85	500,700,900	2	84/85
	Li_2ZrO_3	2	85	500,700,900	2	84/85
FUBR-1B (US)	Li_2O	< 5	60,80	500,700,900	5	85/89
	Li_2O	< 5	80	500-700/1000		
	$LiAlO_2$	< 5-10	80	500,700,900	9	85/89
	(sphere-pac)		80	500-700/1000		
	Li_4SiO_4	< 5	80	400-500	9	85/89
	Li_8ZrO_6	< 5	80	600-700	7	85/89
	Li_2ZrO_3	< 5	85	520-620	7	85/89
ALICE (France)	$LiAlO_2$	0.35-13	71-84	400,600	-	85/86
DELICE (Germany)	Li_2SiO_3 (Li_4SiO_4)	-	65,85,95	400,600,700	< 0.02	85/86
EXOTIC (Neth./UK/ Belgium)	Li_2SiO_3	-	80	400,600	-	85/86
	Li_2O	-	-		-	85/86
	$LiAlO_2$	-	80		-	85/86
	Li_2ZrO_3	-	-		-	85/86
CREATE (Canada)	$LiAlO_2$	< 1	80,90	100	-	85/86
In-situ Tritium Recovery						
TRIO (US)	$LiAlO_2$	0.2 (50 μm particles, 0.9 cm thick annular pellet)	65	400,...,700	0.2	84/85
VOM-15H (Japan)	Li_2O	< 10	86	480,...,760	0.24	84
VOM 22/23 (Japan)	Li_2O	- (1.1 cm pebbles)	-	400-900	0.04	-
	$LiAlO_2$	- (1.1 cm pebbles)	-	400-900	0.1	-
LILA (France)	$LiAlO_2$	1-30 (1 cm diameter pellet)	78	375-600	< 0.02	86
LISA (Germany)	Li_2SiO_3	- (1 cm diameter pellet)	-		-	86
EXOTIC (Neth./UK/ Belgium)	$LiAlO_2$	- (1.4 cm diameter pellet)	80,95	400,600	< 0.4	86
	Li_2SiO_3	- (1.4 cm diameter pellet)	50	400,600	< 0.4	86
CRITIC (Canada)	Li_2O	- (1 cm thick annular pellet)	80	400-900	-	86

The development of high-strength irradiation resistant alloys for fission breeder reactors has led to particular alloys which are currently being evaluated and modified for fusion operation under the U.S. Fusion Materials Alloy Development and Irradiation Program (ADIP) and the Damage and Fundamental Studies Program (DAFS). The present structural material options most suitable for solid breeder blankets are an austenitic (PCA) and a ferritic (HT-9) steel. Low-activation versions are being considered. The use of high-temperature refractory materials depends on the development of suitable radiation-resistant alloys that are compatible with water, reactor-grade helium, and solid breeders under the projected operating conditions.[8] The development and characterization of structural alloys is a common need for all fusion nuclear components and is not discussed further here. Examples of recent experiments in the material irradiation program may be found in Ref. (16).

Tritium Recovery Experiments

The most important tests involve irradiation to provide internal tritium generation, heating and fluence effects. These can be either closed or open capsule tests using either isothermal specimens, pellets large enough to support reactor-relevant temperature gradients (or to achieve high center temperatures), and/or pellets with significant mechanical interaction with the container walls. The importance of an actively-controlled flowing gas environment has been demonstrated in recent experiments such as TRIO.[12] However, closed capsule experiments are cheaper and have proved useful for providing scoping data and irradiated specimens for subsequent property measurement.

A number of open capsule irradiations (Table 1-7) are also underway or have been completed. These tests are exploring a range of temperatures, temperature gradients, materials (primarily Li_2O, $LiAlO_2$, and Li_2SiO_3), material characteristics, container materials, burnups and sweep gas compositions and flow rates. As a result of these tests, a fairly wide-ranging data base will be available around 1990.

However, the planned tests will not address the combination of moderate-to-high burnup with a flowing purge gas under temperature gradients and breeder/clad interactions. Although these effects will be considered separately to some degree, synergistic effects and modeling inadequacies will make extrapolation to reactor-relevant combinations uncertain. Consequently, the next major class of tests should address these interactions. Such Advanced In-situ Tritium Recovery experiments could still be performed with relatively small capsules (~ 1-5 cm diameter), allowing multiple specimens at a given site or a distributed set of tests at different irradiation facilities. The importance of achieving significant burnup while limiting self-shielding in a fission reactor neutron spectrum leads to relatively long irradiation times and a preference for fast reactors. The test facilities must also be high flux and have enough test volume to be able to dedicate the space for the duration of these tests.

Breeder Thermomechanics Experiments

Although unirradiated tests of mechanical properties can be performed relatively easily with standard equipment, the important breeder/cladding interactions and breeder thermomechanical behavior are affected by radiation (swelling, creep) and larger geometrical/operating effects (settling, cyclic cracking). The radiation effects can be determined in the same tests as those described above for monitoring tritium recovery. Some scoping tests with temperature gradients and breeder/clad interactions are underway (e.g., FUBR-1B). However, several closed capsule tests dedicated to thermomechanical effects should be performed in order to allow complete instrumentation (e.g., thermocouples distributed inside the solid breeder) and to provide data to plan more complex in-situ recovery tests. A representative design is illustrated in Fig. 1-6, indicating that several breeder/cladding interactions can be considered in each capsule.

Corrosion and Mass Transfer Experiments

Although there are no major chemical reactivity concerns, temperature limits will exist based on material interactions leading to changes in composition, mechanical integrity or mass transfer. Experiments to determine these limits involve long-term tests of relevant materials and impurities at temperatures which will be achieved in many of the tritium recovery experiments (including those presently underway). However, for new and/or more reactive materials, separate unirradiated testing at relevant temperatures for long-

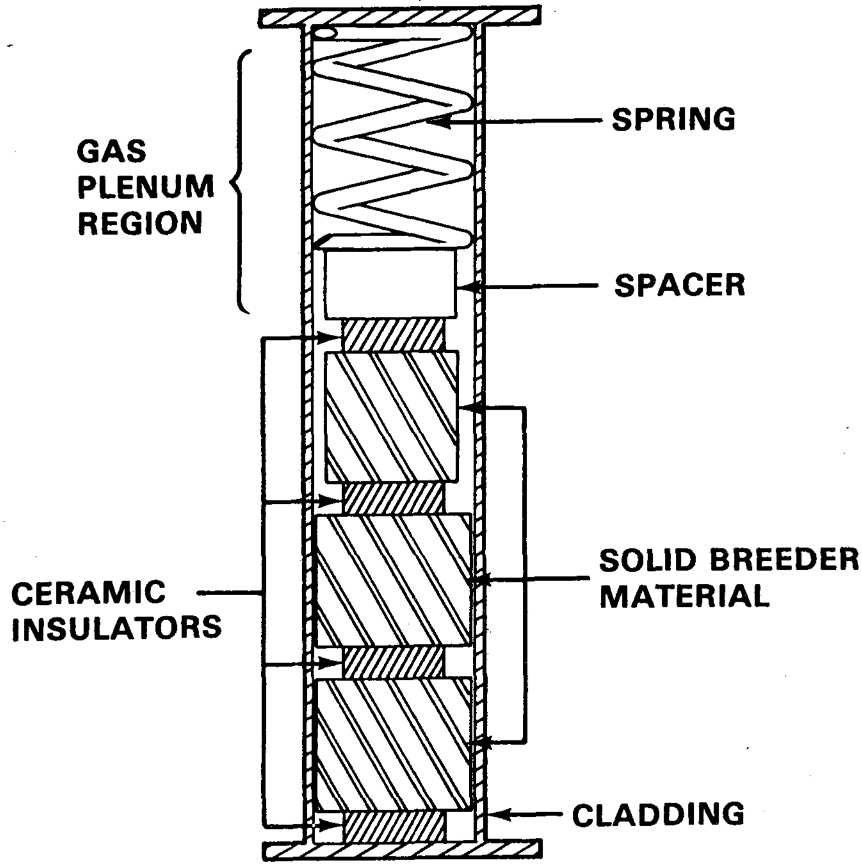


Figure 1-6. Schematic of capsule design to investigate the mechanical interaction between the solid breeder and the structural cladding.

time periods may provide cost-effective data to judge the feasibility of the material or to provide well-defined test conditions for model development. Particularly useful tests include mass transfer within and from Li_2O in a purge stream with hydrogen, the thermal stability and clad compatibility of lithium beryllates, and the interaction kinetics of beryllium with solid breeders and clad.

Multiplier Behavior Experiments

For beryllium or other solid neutron multipliers, the mechanical behavior and tritium retention under reactor conditions are significant uncertainties. Present U.S. effort on multipliers (other than for neutron cross-sections) is limited to planned beryllium creep tests. Experiments needed to address the many uncertainties include unirradiated property measurements, and irradiated closed and open capsules as with the solid breeder material. Fission reactors such as FFTF can provide reactor-relevant simulation of the helium and tritium production in beryllium. Although beryllium powder is chemically toxic, there is sufficient experience available to safely guide fabrication and experiments.

Structural Response Experiments

Many of the issues associated with structural behavior can be addressed by determining the irradiated properties of the materials through specimen tests in suitable irradiation facilities. The modeling basis for structural behavior is reasonably well-established from fission programs, but further model development is needed to provide simpler design tools, to describe particular phenomena and to establish appropriate design criteria for fusion conditions.

Separate unirradiated experiments could usefully address electromagnetic effects (such as steady-state forces on ferritic structures or transient forces on any structure) and the behavior of the first wall under high heat flux and cycling conditions. In the long-term, structural integrity and failure modes with full geometrical effects need to be determined by operation of submodules and/or full modules under reactor-relevant temperatures, pressures and irradiation effects. These more integrated tests are discussed later.

Tritium Breeding Experiments

The tritium breeding uncertainties range from cross-section uncertainties (particularly ^{7}Li at higher energies), to the achievable tritium breeding ratio and heating profile in blankets. The more important questions at present require the measurement of neutron spectra and reaction rates (tritium, heating, transmutations) under progressively more relevant blanket geometries to provide for verification of basic nuclear data, data libraries and neutronics analysis techniques. A well-calibrated 14-MeV neutron source is important, although high fluence is not. Existing facilities such as the

Fusion Neutron Source in Japan are able to address the major issues, and the present US/JAERI cooperative agreement should allow addressing these issues to the extent possible in a non-fusion test facility.

Tritium Permeation and Processing Experiments

Uncertainties associated with controlling tritium permeation and efficiently recovering the tritium from the purge stream are important because they relate to the quantity of tritium released during normal operation. Many of the issues associated with inventory, permeation rate and oxidation kinetics can be addressed in separate glove-box-scale experiments. The use of tritium provides finer accuracy, which may be particularly important for addressing issues at the low tritium partial pressures relevant to some applications. Processing system loop tests (including molecular sieves, oxidizers, getters, etc.) can be performed with reactor-relevant modules to explore tritum holdup, efficiency, lifetime, and general operations. These are reasonably small-sized experiments because of the modularity and size of the full-scale components.

Partially Integrated Experiments

Tests with a higher degree of integration but with a notable lack of one important condition can be considered for providing concept verification information. Non-neutron test stands, fission reactors and fusion devices can serve different roles. However, only a fusion device can provide fully integrated testing.

The non-neutron thermomechanical tests involve heat sources such as microwaves(18) and resistive wires to simulate bulk heating, and particle beams or radiant arcs for surface heating. The tests can range in size from single unit cells to full blanket modules. Although there are clearly limitations on the ability to simulate reactor heating profiles and irradiation effects, these tests are expected to be relatively inexpensive and can provide an opportunity to explore complex thermomechanical behaviors (e.g., gap conductance, flow distribution, thermal cycling), to benchmark design codes, and to study severe transients. The ability to perform such tests in irradiation facilities is limited by available test volume, by the costs of irradiation tests, and by reactor safety constraints. The value of non-neutron large-geometry tests is dependent on the degree to which geometrical details have been defined, on the importance of the related issues, and on the extent of the planned nuclear experiments.

Nuclear test assemblies designed for fission reactors can also provide the maximum concept verification possible in non-fusion devices. These include the important nuclear effects but would be limited in several respects, primarily test volume. A full-blanket module test would need about 1 m^3 of test volume, require extensive modifications to any operating fission reactor core, and still only achieve the equivalent of (at most) a 1 MW/m^2 heating rate in any existing reactor.(1) In-core assemblies could be placed in existing fission reactors like FFTF at reactor-relevant heating rates (2-5 MW/m^2), but would be limited to about a 10-cm diameter. These test assemblies would provide fairly realistic simulation of fusion conditions, with complete coolant and purge flow systems and instrumentation. These tests would provide a necessary amount of concept verification, but the degree of confidence achievable prior to fusion testing remains unclear.

1.2.2.3 Test Plan

The solid breeder blanket issues and the corresponding testing needs have some unique characteristics, especially with respect to liquid breeder blankets. First, there are a large number of potential breeder materials and material variables (e.g., grain size) that can be altered to produce unique properties. Secondly, the influence of geometry on the primary uncertainties is not large. The most significant uncertainties are related to basic properties or to local behavior (e.g., within a pellet). Thirdly, the influence of radiation on the behavior in general, and the uncertainties in particular, is large. Radiation damage and transmutation can substantially alter the original material. Finally, much of the important functional behavior of the solid breeder is not described by classical equations, but rather the controlling phenomena must be quantified by experiments. Therefore, the whole test program is more empirical, and it is more difficult to confidently scale from test conditions to reactor conditions.

Based on the key issues and testing needs, a number of broad tasks have been identified as key elements in the test program for solid breeder blankets. Each task consists of a number of experiments and related activities aimed at resolving one or more of the critical issues. The costs associated with each task have also been assessed. These tasks and estimated costs are summaried in Table 1-8 for a reasonably complete solid breeder blanket experimental program. Capital costs include design effort, materials, fabrication and construction of the facility and experimental apparatus. Operating costs include use of materials and energy, operating staff, and data acquisition. Both these costs include laboratory overhead. The cost of model development, data analysis and comparison with theory and blanket design studies are listed separately. The task duration includes design, fabrication, test and post-test examination. The total cost is the sum of the capital costs plus the operating cost over the testing phase (not the design and fabrication phase). For solid breeder blankets, existing fission reactors and point neutron sources are sufficient, and the costs do not include new nuclear facilities or neutron changes in existing facilities.

The objective of the solid breeder characterization and development task is to fabricate, characterize, and improve the properties of candidate breeder materials, including possibly closed or open capsule irradiation of material specimens. Present activity relevant to this task is about 6-9 M$/yr world wide. A similar level of effort should continue with an additional near-term effort on developing sphere-pac materials and consideration of novel materials such as lithium beryllates.

The objective and tests for the multiplier characterization and development task are similar to those for solid breeder materials. However, there is presently little experimental activity in this area. A reasonable program would be about 1-2 M$/yr, consistent with the pace and relative number of solid breeder materials being investigated.

The objective of the breeder thermal behavior task is to investigate thermomechanical behavior, heat transfer and corrosion/mass transfer. In the near term, a few unirradiated corrosion capsule tests and irradiated breeder/-clad interaction closed capsule tests are needed. In the longer term, a non-nuclear thermomechanical test facility could provide more complete testing of geometrical and transient related effects, although without irradiation. The need for such a facility and the complexity of the test will depend on the degree of design detail available, and on the extent and scope of the planned nuclear tests.

Table 1-8. Representative Costs of Major Solid Breeder Tasks Over the Next Fifteen Years[a]

Task	Capital cost (M$)	Operating cost (M$/yr)	Duration (years)	Total cost (M$)
Solid Breeder Characterization and Development (Fabrication, properties, closed and open capsule irradiations)	5-7	5-8	5 (initial)	30-50
Multiplier Characterization and Development (Fabrication, properties, closed capsule irradiations)	1-2	1-2	5	6-12
Blanket Thermal Behavior				
A. Breeder thermal behavior	0.8-1.5	0.8-1.5	3-5	3-8
B. Non-neutron (sub)module thermomechanics	3-8	0.8	4	5-10
Neutronics and Tritium Breeding				
A. Simple geometry	3-6	0.8-1.5	5	7-14
B. Engineering mockup	4-7	0.8-1.5	3	6-12
Advanced In-Situ Recovery (Two sequential subassemblies with multiple purged capsules)	3-5 each	0.8 each	6 each	12-16
Nuclear submodules (Two parallel submodules)	5-7 each	1-1.5 each	7 each	20-30
Analysis and Model Development	0	2-3	15	30-45

[a]1985 constant dollars; Neutron facility and neutron costs not included.

The objective of the neutronics and tritium breeding task is to verify and improve nuclear data, design methods and models by measuring tritium production rate and heating rate distributions in relevant blanket assemblies. Two phases of testing can be identified, with simple geometry and detailed geometry. Those tests require a 14 MeV neutron source. Relevant experiments are beginning in 1986 as part of the U.S./JAERI Fusion Breeder Neutronics Collaborative Program.

The objective of the advanced in-situ recovery tests is to study tritium recovery with local reactor-relevant conditions, specifically moderate-to-high burnup, temperature gradient, purge flow and breeder/cladding mechanical and chemical interactions. This task could be performed as one or more instrumented and purged subassemblies in fission reactors, depending on the available test volume and the size of the test matrix.

The objective of the nuclear submodule task is to verify overall behavior of a blanket submodule to the extent possible in a neutron environment. Fission reactor limitations constrain the size of the test piece to sections of a full blanket module.

These tasks are key elements of the test plan for the development of solid breeder blanket technology shown in Figure 1-7. Over the next 15 years, the plan emphasis gradually shifts from the understanding of material behavior and blanket phenomena to the development of predictive capabilities and, finally, to the verification of design concepts in a non-fusion environment. Accordingly, the development and characterization of solid breeder materials must continue since the resulting data support the selection of materials and impact the other tests. The assessment of tritium self-sufficiency should also continue with the next phase of the U.S./JAERI Fusion Breeder Neutronics Collaborative Program. In the near future (~ 1987), additional tasks must be started with respect to neutron multiplier material development (specifically beryllium) and to basic breeder thermal behavior in order to support material selections in about 5 years. The design of an advanced in-situ tritium recovery experiment would also begin, in order to quantify local design-related behavior under fusion-relevant conditions.

From this data base, a limited number of blanket concepts would be selected and verified to the extent possible in non-fusion facilities. The

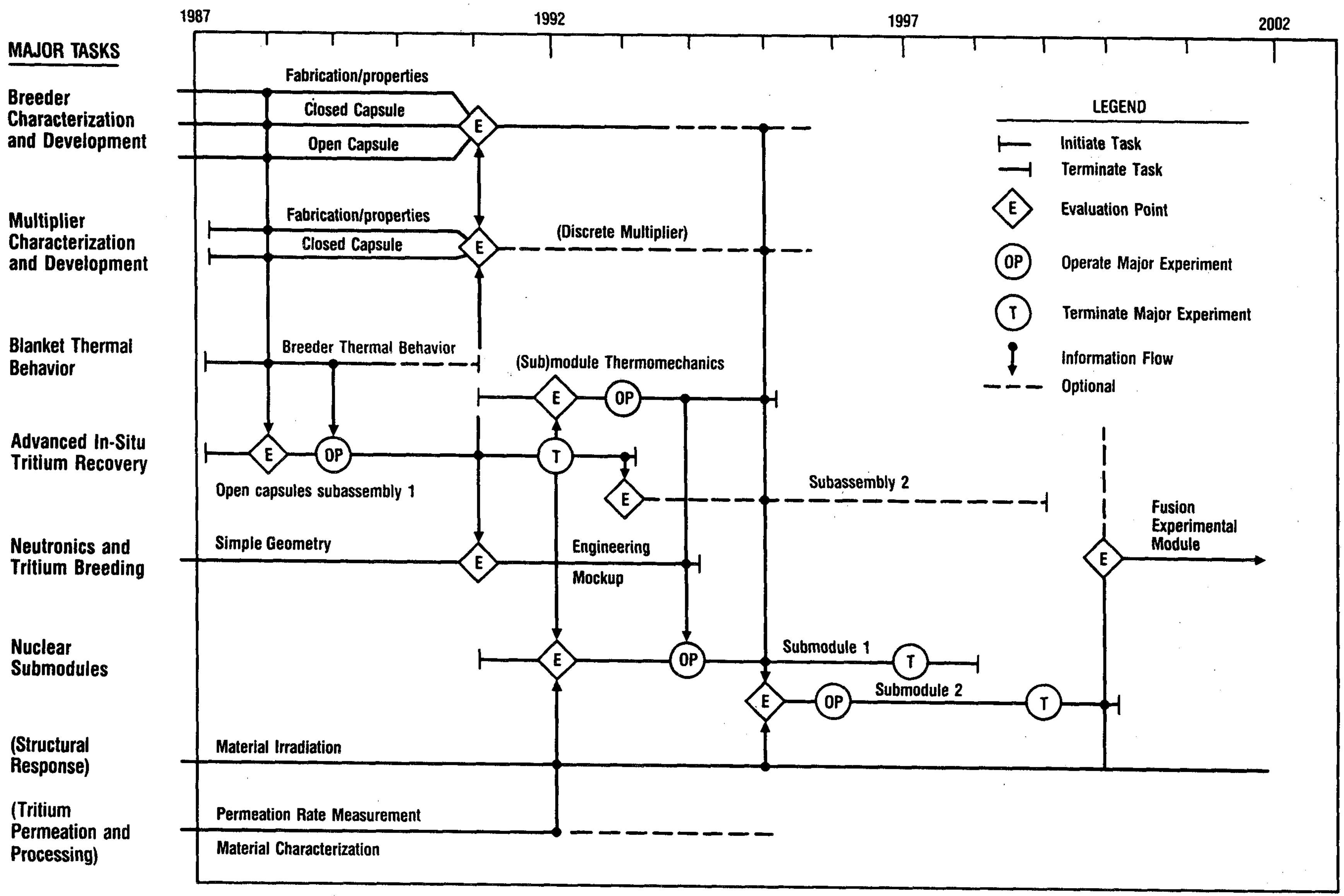


Figure 1-7. Test sequence for major solid breeder blanket tasks

tests include engineering mockup neutronics tests, non-neutron (sub)module thermomechanical integrity tests, and nuclear submodule tests in fission reactors.

This test plan should be sufficient to support an assessment of the feasibility and attractiveness of solid breeder blankets within the next 15 years at an estimated cost of 10-20 M$/yr. Assuming that solid breeder blankets are sufficiently attractive, the program would then be able to confidently proceed with the design of a blanket experimental module for a fusion test device.

1.2.3 Liquid Breeder Blankets

1.2.3.1 Issues and Testing Needs

Liquid breeder blankets encompass a variety of generic design variations, including self-cooled or separately-cooled, and insulated or uninsulated designs (see Table 1-9). The existence and seriousness of the major issues are stongly dependent on the particular blanket concept, and also depend on the operating conditions such as power density, magnetic field, surface heat flux, temperature and duct length. Generic issues have been defined to encompass the most promising blanket designs being considered today. These issues are listed in Table 1-10 and are discussed below. Issues relating to safety and/or transient effects are not listed separately, but rather they are considered an integral part of all the issues.

MHD Effects: Some of the largest uncertainties in self-cooled liquid metal blankets relate to magnetohydrodynamic (MHD) effects on velocity profiles, heat transfer, pressure drop, and mass transfer. The existing theory of the flow of conducting liquids in strong magnetic fields has established some general features of the flow, but large uncertainties remain in predicting key design parameters in complex geometries of fusion blankets. A particular concern is the large degree of uncertainty in characterizing the velocity profiles.

MHD effects are most strongly dependent on the geometry and on a small number of dimensionless parameters, the most important being the Hartmann number (M), the interaction parameter (N), and the wall conductance ratio (C). (The Hartmann number is proportional to the magnetic field and measures the dominance of the MHD force over viscous forces. Similarly, the interaction parameter measures the dominance of the MHD force over inertial forces.) Figure 1-8 indicates that the dimensionless region of representative existing experimental data is much lower than that found under actual reactor conditions. Most of the data has also been accumulated in very simple geometries. Data is needed both for higher values of M and N and also for geometries more representative of actual blanket configurations. In addition, due to the large potential impact that electrical insulators will have on the feasibility and design of liquid breeder blankets, early scoping tests should be performed to explore their potential problems and benefits.

Table 1-9. Liquid Breeder Blanket Design Options

Design Classes	Materials	Configuration
Self Cooled	Breeders	Radial Flow
With Insulated Wall	Lithium	Poloidal Flow
With Uninsulated Wall	17Li-83Pb	Toroidal/Poloidal
	Flibe	Helical Flow
Separately Cooled	Coolants	
	Self	
	Helium	
	Multiplier	
	None	
	Beryllium	

Table 1-10. Generic Liquid Breeder Blanket Issues

- Tritium Self-sufficiency
- Magnetohydrodynamic (MHD) Effects
  - Fluid Flow (including pressure drop)
  - Heat Transfer
- Material Interactions (e.g., Corrosion)
- Structural Response in the Fusion Environment
  - Irradiation Effects on Material Properties
  - Response to Complex Loading Conditions
  - Failure Modes
- Tritium Recovery and Control

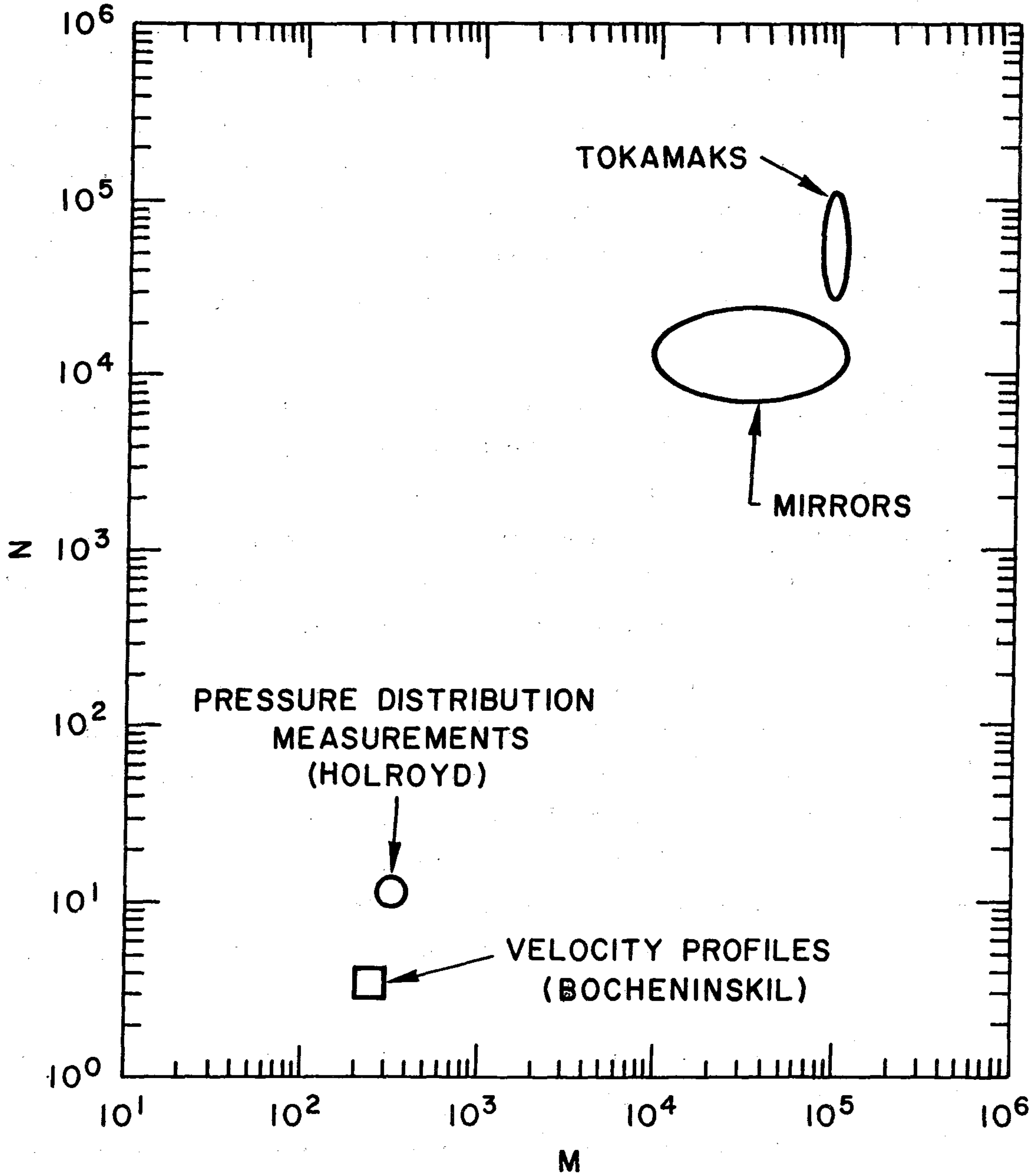


Figure 1-8. Hartmann number (M) and interaction parameter (N) ranges for existing data and reactor conditions.

Because of the impact of the magnetic field on the velocity profiles, the heat transfer characteristics are also strongly affected. MHD heat transfer can be predicted if the velocity profiles are sufficiently well known. However, the accuracy of velocity profile measurements and the ability to extrapolate such measurements to the more complex geometries of actual designs are serious concerns. Measurements of temperature profiles provide additional information that can be used directly to predict heat transfer and/or to provide a consistency check of velocity profile measurements. Thus, heat transfer experiments are considered an important supplement to fluid flow measurements. The engineering scaling requirements for testing include all of those for fluid flow testing, as well as the additional ones listed in Table 1-11.

Material Interactions: There are a large number of phenomena relating to material interactions, including both mass transfer and structural changes due to interactions among the coolant, breeder, and structural materials within the primary cooling loop. The importance of the issues depends strongly on the type of materials. Table 1-12 shows the most important material interaction issues for the materials shown in Table 1-9.

Compared to heat transfer and fluid flow, additional environmental conditions (such as materials, impurity levels, absolute temperature, temperature gradient, out-of-blanket geometry, and long-term exposure) can be critically important. Because of the complexity and material dependence, general models for predicting material interaction phenomena will likely be deficient. Thus, a number of experiments will be needed to develop empirical correlations for the behavior under relevant conditions, such as temperature and impurity levels.

A strategy for obtaining maximum relevant information on material interactions at relatively moderate cost is proposed. In the early stages of the test program, conventional material interaction loops (e.g., forced convection loops) will be operated with relevant materials and at relevant operating conditions, such as temperature and impurity level, but without a magnetic field. In parallel, experiments on MHD effects on fluid flow will provide information on the effects of magnetic field on the fluid operating parameters. The combined information from conventional loops and MHD fluid

Table 1-11. Engineering Scaling Requirements for Non-Neutron Heat Transfer Tests of a Liquid Metal First Wall/Blanket*

First Wall	Liquid Metal Blanket
• Negligible Axial Conduction $\frac{t}{\ell} \ll 1$	• Correct Velocity Distribution $M \simeq 10^4 \sim 10^5$; $N > 10^3$; $1 \gg C \gg M^{-1}$
• Negligible Bulk Heating $\frac{Qt}{2q} \ll 1$ or $\frac{k\,\Delta T_{fw}}{q\,t} \simeq 1$	• Suppression of Turbulence $Re < 60\ M$
	• Flow Entrance Length $\frac{\ell C^{1/2}}{a} \simeq 1$
	• Thermal Entrance Length $\frac{\ell \alpha}{va^2} \simeq 10^{-2}$
	• Suppression of Natural Convection $\frac{\rho g \beta \Delta T}{v \sigma B^2 C} \ll 1$
	• Negligible Axial Conduction $\frac{\alpha}{v\ell} \ll 1$

*Parameters used are:

a	channel radius or half-width	Re	Reynold's number
B	magnetic field strength	t	first wall thickness
C	wall conductance ratio	ΔT	temperature difference
g	acceleration of gravity	ΔT_{fw}	temperature difference across first wall
k	thermal conductivity	v	velocity
ℓ	axial length along channel	α	thermal diffusivity
M	Hartmann number	β	thermal expansion coefficient
N	interaction parameter	ρ	density
q	surface heat flux	σ	electrical conductivity
Q	volumetric heating rate		

of the issues relating to structural behavior are dominated by the material response under irradiation. These issues can be partially addressed in small, subscale test elements placed in fission reactors and other available neutron sources. The most desirable test facility for structural response issues is clearly a fusion reactor, in which the power density, fluence, spectrum, and key thermomechanical conditions can all be achieved simultaneously.

Tritium Recovery and Control: At present, tritium recovery is considered a critical issue for LiPb, but not for lithium. Acceptable extraction schemes have been proposed for lithium (for example, molten salt extraction[19]), with laboratory-scale experimental verification available. For 17Li-83Pb (LiPb) and LiF/BeF_2 salt (Flibe) the tritium solubility is so low that high partial pressures exist, which may result in unacceptable tritium permeation and release rates. An extremely high extraction efficiency is required, but not yet experimentally demonstrated. This is further complicated by a general lack of tritium-related data in LiPb and Flibe.

Tritium Breeding: Tritium breeding is not usually considered a feasibility issue for liquid breeder blankets. Self-cooled designs show a high breeding ratio, but in separately cooled designs, the breeding margin is smaller. Also, some reactor designs may have only partial blanket coverage (such as no breeding at the tokamak inboard side). The uncertainties in tritium breeding can be reduced through a program of basic nuclear data measurements, integral neutronics experiments, and improvement of calculational methods and codes.

1.2.3.2 Existing and Required Experiments and Facilities

Through examination of the issues and test requirements, the needed experiments and test facilities have been identified. Figure 1-9 shows a matrix of tests required to address the key issues for liquid breeder blankets, including some experiments which are already in progress. The required experiments and facilities are organized according to the classes of issues they resolve and their level of integration.

The test matrix represents a complete list of major types of experiments which are needed, but not all of them will necessarily be performed. Depending on funding constraints, choices of blanket materials and configurations,

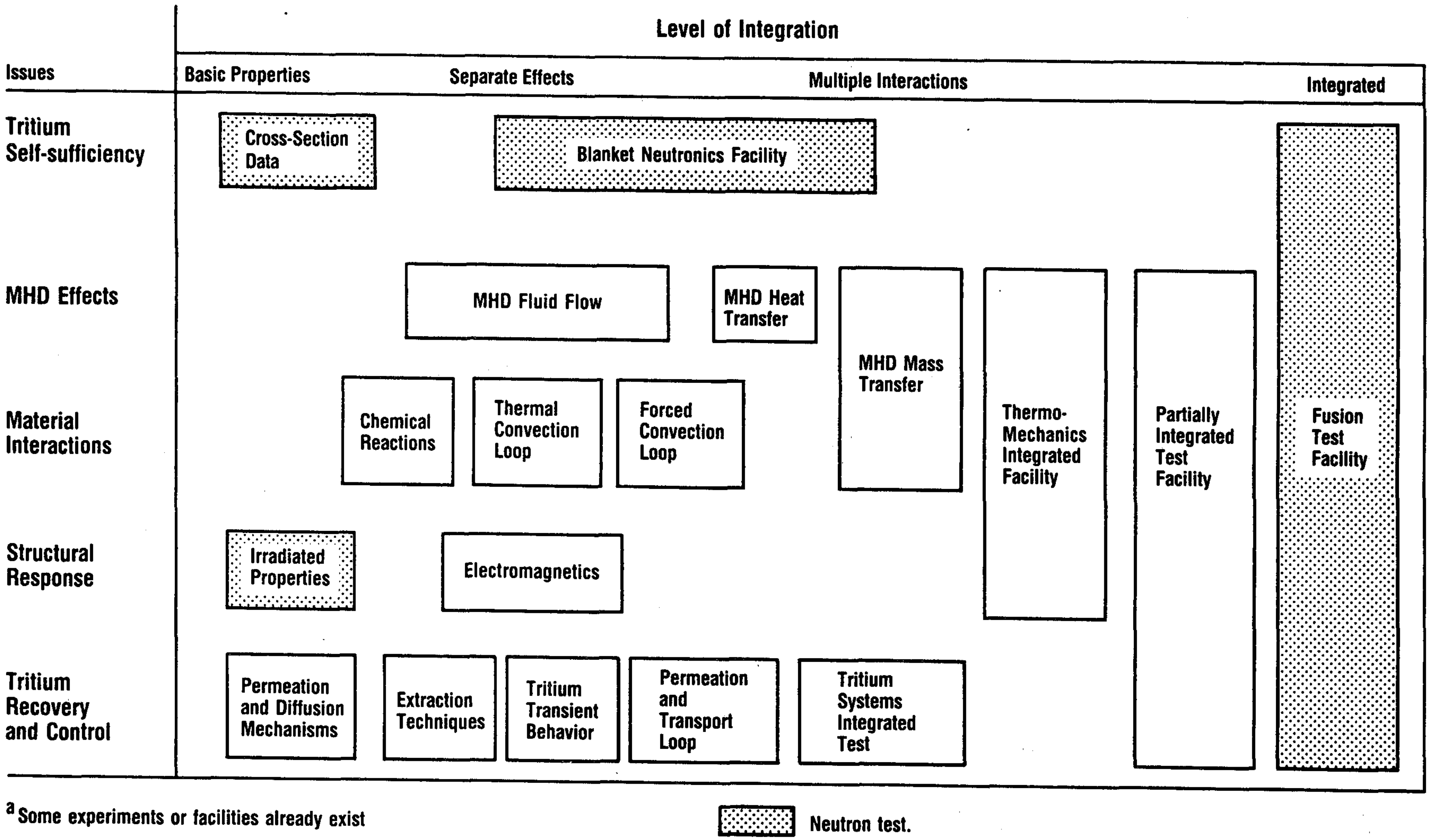


Figure 1-9. Types of experiments and facilities for liquid breeder blankets[a]

results of prior experiments, and time-dependent testing goals, only a subset of the proposed experiments may actually be performed. In addition, a complete testing program designed around these major experiments may include a number of smaller experiments not specifically listed in the figure. A complementary theory and model development program will also be required. The logic behind these choices is considered in Chapter 2.

Existing Experiments and Facilities

The experiments which have been performed or are underway in technical disciplines relevant to liquid breeder blanket issues are summarized in Table 1-13. The primary element in the current U.S. MHD program is the ALEX facility at Argonne National Laboratory (ANL). ALEX is capable of magnetic fields up to 2 Tesla in a field volume 1.8 m x 0.76 m x 0.15 m.[20] This will provide Hartmann numbers and interaction parameters much closer to reactor conditions than any previous experiment. Information expected to come from ALEX includes single channel pressure drops and velocity profiles in straight channels, bends, and magnetic field entrance regions. In addition to the measurements of pressure drop and velocity profiles, the MHD program at ANL will contribute to the development of velocity profile instrumentation. The ability to develop techniques to accurately measure velocity profiles at high magnetic field will have a large impact on the remainder of the MHD test program.

In the area of material compatibility, several corrosion loops are in operation. These loops provide valuable information for identifying compatible material combinations. However, large uncertainties remain in defining accurate temperature limits, the effects of impurities, and methods of controlling corrosion.

The remaining facilities listed in Table 1-13 include the FELIX electromagnetic test stand and FNS, which is a neutronics facility with a point neutron source. The experiments anticipated in these facilities should satisfy near-term requirements for data relevant to liquid breeder blanket issues.

Table 1-13. Summary of Existing U.S. Test Facilities for Liquid Breeder Blanket Research

LIQUID METAL MHD		
Location	Field Strength	Volume
ANL (ALEX)	2.0 T	1.83 m x 0.76 m x 0.15 m
LIQUID METAL CORROSION		
Location	Materials	Loop Type
ANL[a]	Li/304SS	Forced convection
ANL	17Li-83Pb/316SS, PCA, HT-9, 9Cr-1Mo	Forced convection
ORNL[b]	316SS, HT-9, Alloy 800	Thermal convection
ETEC[c] (BLIP)	Li/2-1/4Cr-1Mo	Forced convection
HEDL[d] (ELS)	Li/SS	Forced convection
UW[e]	Li/316SS	Forced convection
ELECTROMAGNETICS		
Location	Field Strength	Volume
ANL (FELIX)	1.0 T steady[f], 0.5 T pulsed	0.9 m diameter, 1.2 m long
BREEDER NEUTRONICS		
Location	Source Strength	
JAERI (FNS)[g]	2 x 10^{12} n/s	

[a] Argonne National Laboratory

[b] Oak Ridge National Laboratory

[c] Energy Technology Engineering Center

[d] Hanford Engineering Development Laboratory

[e] University of Wisconsin

[f] Capable of 4.0 T steady state and 1.0 T pulsed

[g] Cooperative U.S./Japan program

Required New Experiments and Facilities

While existing test facilities have begun to address critical liquid breeder blanket issues, there is need for a number of new facilities. A range of experiments have been explored to fulfill this need. Table 1-14 shows the relationship between these by specifying the principal features of the facilities and objectives of the experiments. In addition to ALEX, further experimentation on MHD in more complex geometries and under conditions closer to fusion reactor conditions will be necessary in order to develop an ability to predict fluid flow, heat transfer, and pressure drop behavior in some self-cooled blanket designs with complex flow paths. Two advanced liquid metal flow facilities, LMF1 and LMF2, have been examined. In LMF1, the emphasis is on developing a better understanding of the "microscopic" MHD behavior, especially the velocity profiles, in basic elements of relevant geometries. LMF2 is a facility which is directed at the measurement of more global parameters in representative blanket module designs, such as pressure drop and heat transfer coefficients. It would be especially crucial if microscopic measurements do not provide sufficient theoretical prediction capability.

It is not expected that results from the existing experiments in corrosion/mass transport will provide enough information for the development of fusion blankets. More corrosion loops will be required for thorough studies of fusion relevant materials, especially for refractory metals and bimetallic systems. The most critical information required includes dependence on temperature and impurities, loop effects, dependence on magnetic field, and methods of controlling corrosion/mass transport. Thermal convection loops (TCLs) can provide fundamental information on temperature and impurity dependencies at relatively low cost. Forced convection loops (FCLs) will be needed to obtain relevant velocities and also to simulate the effects of loop components such as pumps or heat exchangers. After studying the basic material interactions in TCLs and FCLs, experiments with strong magnetic fields will be needed to explore the effects of the magnetic fields on mass transport. A particular facility, called the MHD Mass Transfer Facility (MHDM), was defined with a large enough volume and field strength such that prototypical velocity features can be obtained.

Table 1-14. Features and Objectives of Major Liquid Breeder Experiments

	ALEX[a]	Magnetic Transport Phenomena Facilities: LMF[b]	Magnetic Transport Phenomena Facilities: MHDM[c]	TMIF[d]	PITF[e]
Features of Experiments	• Simple geometry of a channel	• Basic elements of relevant geometry	• Basic elements of relevant geometry • Relevant materials combination • Transport loop • Relevant T, ΔT, impurities, V • Long operating time per experiment	• Actual materials and geometry • Transport loop • Relevant environmental and operating conditions	• Prototypic blanket module • Transport loop • Prototypic environmental and operating conditions
	• Measure velocity profile, electric potential pressure drop (may be upgraded)	• Measure velocity and temperature profiles; pressure drop, temperature, electric potential	• Measure dissolution and deposition rates	• Measure integral quantities (ΔP, T, corrosion and deposition rates)	• Measure integral quantities
Objectives	• Develop and test velocity profile instrumentation in NaK environment • Validate MHD in simple geometry (basic heat transfer data may be possible in upgrade)	• Develop and test instrumentation • Validate MHD, MHD heat transfer • Design data (ΔP, T) for configuration screening and provide input information to design TMIF • Explore techniques to reduce ΔP and enhance heat transfer	• Develop and test instrumentation in relevant environment • Design data on MHD heat and mass transfer • Verify techniques to reduce corrosion and corrosion effects	• Design data for blanket test module • Confirm and refine configurations	• Engineering design data • Reliability data in non-fusion environment

[a]Exists (ANL)
[b]Liquid Metal Flow Facility
[c]MHD Mass Transfer Facility
[d]ThermoMechanical Integration Facility
[e]Partially Integrated Test Facility (may be an upgrade of TMIF)

Beyond the first 5-10 years of testing, experiments will become progressively more integrated as they treat a larger number of environmental conditions and components resembling actual reactor blankets. A class of experiments has been defined to provide information contributing to concept verification, rather than phenomena exploration alone (as with separate and multiple interaction experiments). Two types of tests with different missions have been considered for providing engineering data. Since their operation would occur after 5-10 years of more fundamental testing, it is difficult to anticipate the exact features of the facilities. However, the general features and objectives have been studied. A Thermomechanical Integration Facility (TMIF) is one particularly attractive concept. It combines thermal, hydraulic, materials, and structural issues in a system which includes the blanket, chemical control systems (inhibition and impurity control), primary cooling system components, and possibly even the tritium extraction systems. Table 1-15 shows preliminary design data for such a facility.

Another type of facility, called the Partially Integrated Test Facility (PITF), has also been defined. Since fully integrated testing in the fusion environment will be a very costly step in the development of fusion nuclear components, it is desirable to maximize the availability of the fusion device and the benefit of fusion testing. The PITF would be a full- or near-full-scale blanket with primary cooling system and tritium extraction system (with tritium or deuterium/hydrogen). For liquid breeder blankets, the omission of neutrons results in large cost savings, with many of the critical issues still addressed. For many important parameters, such as surface heat flux, velocity, and geometry, partially integrated experiments can provide a good simulation of the operating characteristics of a power reactor. These experiments should provide some useful information on failure modes and component reliability.

Table 1-15. Parameter Range for TMIF

Magnet Volume	3 m x 1 m x 0.5 m
Magnetic Field Strength	4-6 T
Average Coolant Velocity	0.1-0.2 m/s
Bulk Coolant ΔT	100-200 K
Volumetric Flow Rate	0.5-1.0 m^3/s
Total Heat Input (assuming no economizer)	10-40 MW

1.2.3.3 Test Plan

The test plan is a method to optimally resolve the issues and develop blankets whose feasibility and attractiveness can be predicted with adequate certainty. It also provides a framework for the selection and sequencing of experiments. Figure 1-10 shows a possible sequence of experiments for liquid breeder blankets. The major classes of facilities are listed as a function of time, indicating key evaluation points. The evaluation points generally include both selection of future experiments and narrowing and selection of materials and design choices. In the figure, four overlapping phases of experimentation are assumed. In the first phase (0-10 years), the primary goal is to identify a limited number of material choices through a program which broadly treats the most critical issues for the largest number of attractive blankets. The experiments include a variety of (simple and advanced) MHD fluid flow tests, materials compatibility loops, and tritium recovery and tritium breeding experiments in simple geometries.

In the second phase (5-15 years), the primary goal is to assess design configurations and design limits, and to select a small number of primary design candidates. Some of the single and multiple effects experiments continue into this phase, and more integrated facilities are initiated. One of these more integrated facilities is the TMIF, which explores thermal/hydraulics, materials compatibility, and some structural behavior under complex environmental conditions which include magnetic field and surface heating. Phase II also includes more advanced experiments on tritium recovery and control, such as a tritium transport loop and a blanket/tritium

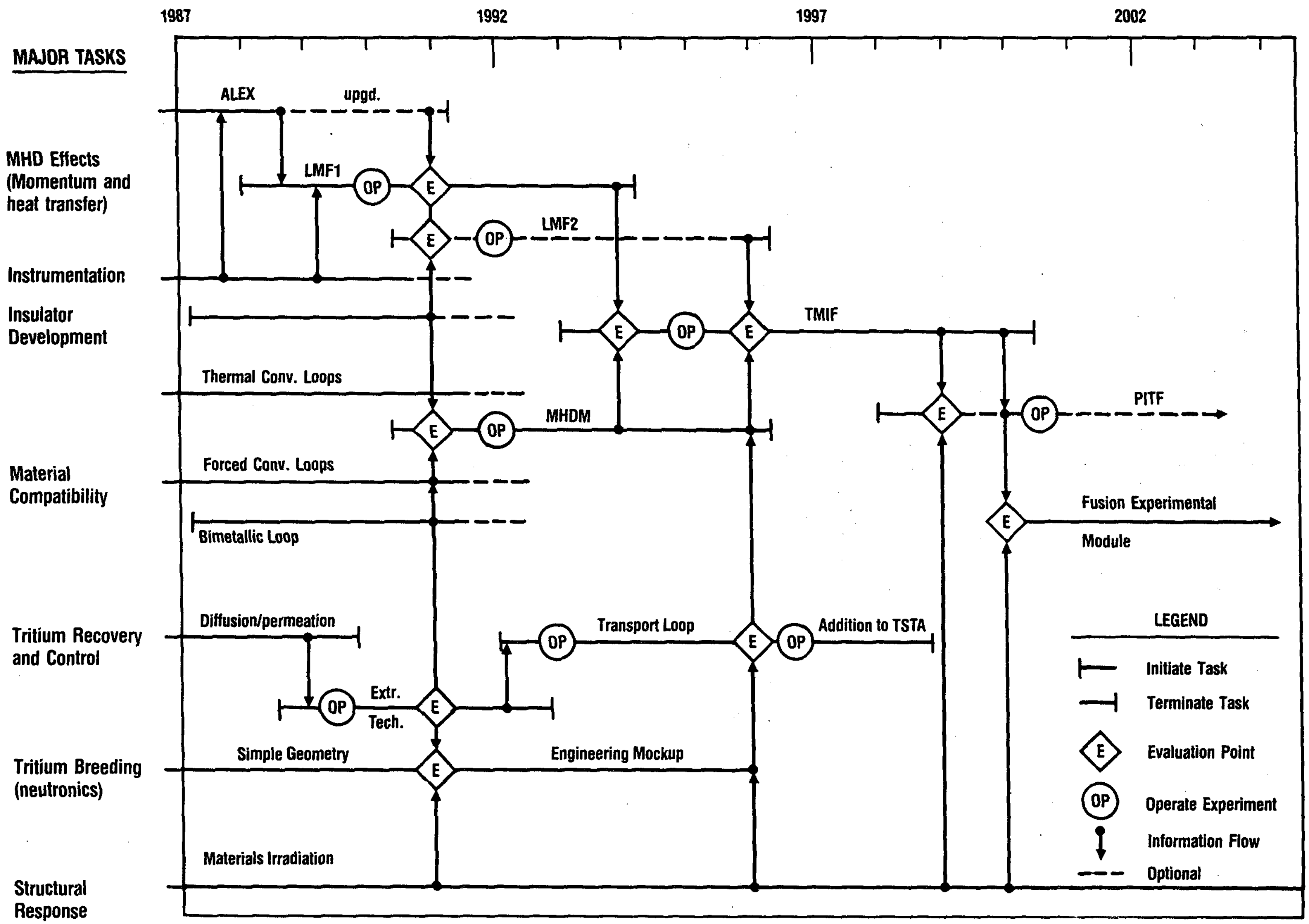


Figure 1-10. Test sequence for major liquid breeder blanket tasks

processing system interface experiment.

In the third phase (10-20 years), partially integrated testing will be carried out to verify prototypic designs under non-fusion conditions with the maximum number of environmental conditions possible. Finally, in the fourth phase, fusion testing will be used to operate prototypic blanket test modules under full fusion conditions.

The cost of the major facilities discussed above have been estimated in order to determine an approximate overall program cost to develop liquid breeder blankets. The numbers shown in Table 1-16 are intended as program costs. They represent all of the costs associated with the experimental program, including both capital and operating expenses. Capital costs include design effort, materials, fabrication, construction, and any expense directly related to the construction of the facility and the experimental apparatus. Annual operating costs include use of materials and energy, staff to operate the experiments, and data acquisition. The cost of modeling efforts, detailed comparison of experiments with theory, and blanket design studies have not been included as operating expenses. These are listed separately. A liquid breeder blanket program requires an average annual expenditure of about 10-20 million dollars.

Table 1-16. Representative Costs of Key Liquid Breeder Blanket Facilities

Item	Capital Cost[a] (M$)	Operating Cost[b] (M$/yr)	Duration (years)	Total Cost (M$)
Advanced liquid metal flow facility (LMF1)	7-10	0.5	4-6	10-15
Integral Parameter Experiment (LMF2)	7-10	0.5	4-6	10-15
MHD mass transfer facility (MHDM)	8-12	1.0	6-8	15-20
Thermal convection loops (~4)	2-4	0.8	4-6	5-9
Forced convection loops (~4)	4-6	0.8	4-6	7-11
Tritium extraction test (2)	2-3	0.4	3-4	3-5
Tritium transport loop test	6-8	0.6	5-7	9-12
Thermomechanical Integration Facility (TMIF)	20-25	2.0-3.0	8-10	35-60
Analysis and model development	--	2.0-4.0	15	30-60

[a]In 1985 constant dollars

[b]Does not include analysis of data

1.3 Tritium Processing and Vacuum Systems

The tritium processing and vacuum systems can be divided into four areas: (1) Fuel Processing, (2) Tritium Permeation, (3) Breeder Tritium Extraction, and (4) Vacuum Systems. The issues and testing needs for each area are discussed below.

1.3.1 Fuel Processing

The research and development path for tritium processing technology is rather different from the paths of other technologies. The reason for this is a unique set of circumstances in tritium technology that has resulted in both the need and the ability to build a "partially integrated test facility" for tritium processing relatively early in the fusion program schedule. This facility, the Tritium Systems Test Assembly (TSTA), is shown in block form in Fig. 1-11.

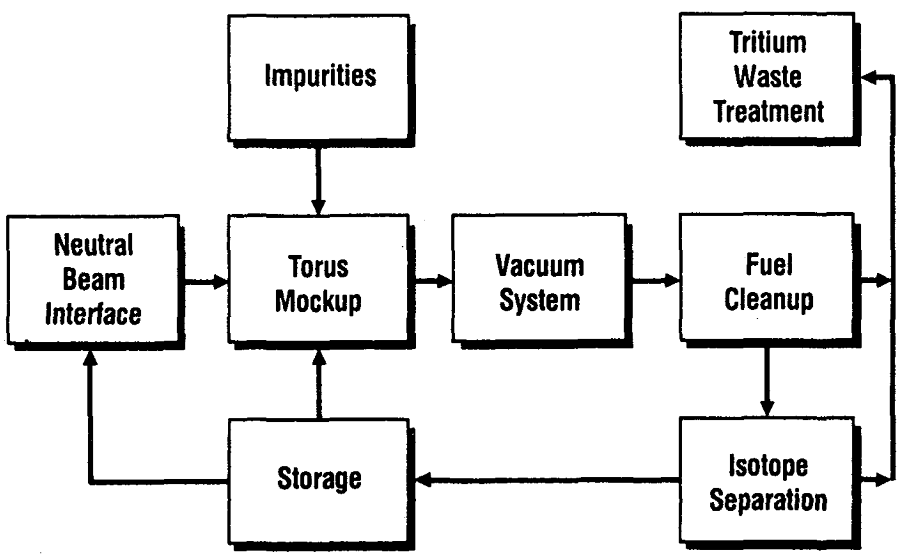


Figure 1-11. TSTA main process loop and auxiliary systems

With an existing facility, some effort necessarily shifts from the task of shaping a test facility to the task of shaping experiments, modifications, and interactions with other technologies to realize the maximum programmatic return from the capital investment in the facility. These factors have contributed to the directions and plans discussed in this report.

The critical technical issues in tritium processing tend to be issues dealing with integration of tritium systems and with the interfaces between tritium systems and other systems. These issues, briefly summarized, are:

A. Tritium monitoring and accountability: Two key aspects are the avoidance of neutron and gamma effects on monitors and the present uncertainty of regulatory requirements for accountability.

B. Impurity removal from fuels: Key aspects are defining impurity species and concentrations and defining tritium losses in processing.

C. Detritiation of room atmospheres and water coolant: Key aspects are defining the required cleanup time for room atmospheres and defining the input and required output concentrations for water detritiation systems.

D. Integrated system behavior: Key aspects are the reliability of complex and interrelated systems during the normal and off-normal operations.

1.3.2 Tritium Permeation

Two kinds of tritium permeation are important and must be understood - plasma driven and pressure driven. An understanding of permeation is important at many locations in a fusion reactor, across a variety of material interfaces, in a variety of materials, and under a range of environmental conditions. The conditions of experiments to gain the necessary understanding of tritium permeation issues are summarized in Table 1-17. The importance of neutron effects on tritium permeation needs to be evaluated.

Table 1-17. Key Parameters for Permeation Experiments

Key Parameters	Plasma Driven Permeation	Pressure Driven Permeation
Temperature (°C)	200 - 500[a]	200 - 500[a]
Temperature Gradient (°C/cm)	200 - 300	100 - 300
Pulse Lengths (s)	$10^2 - \infty$	$10^2 - \infty$
Neutron Fluence (dpa)	> 1	> 1
Tritium Wall Flux (cm^{-2} s^{-1})	10^{15}	--
Tritium Energy (eV)	≤ 1000	--
γ Radiation	Characteristic of metal under neutron irradiation	
Surface Effects	Characteristic of plasma edge	Characteristic of blanket purge and coolant system
Tritium Partial Pressure (Pa)	--	$10^{-7} - 10^{1}$[b]

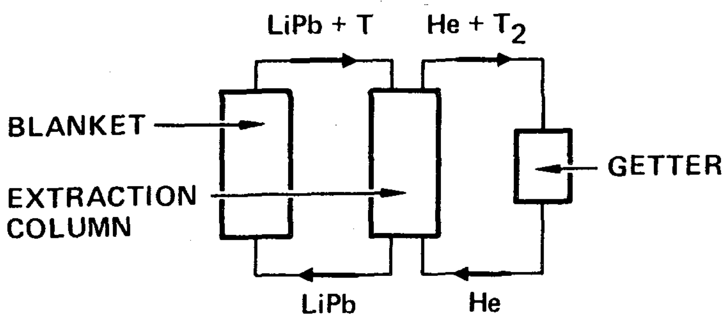
[a]Up to 750°C for vanadium; higher for some coatings.

[b]Dependent on blanket design.

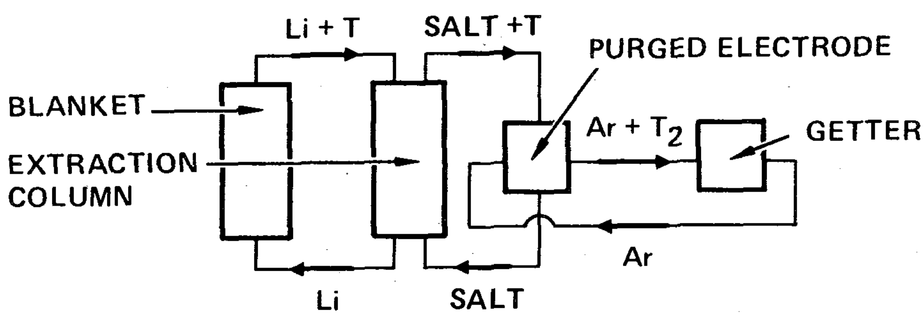
1.3.3 Breeder Tritium Extraction

Issues of breeder tritium extraction can be summarized according to the fluid used to transport tritium from the breeder. The potential carriers, in different breeder systems, are LiPb, Li, and He. Extraction of permeated tritium from water is also of interest.

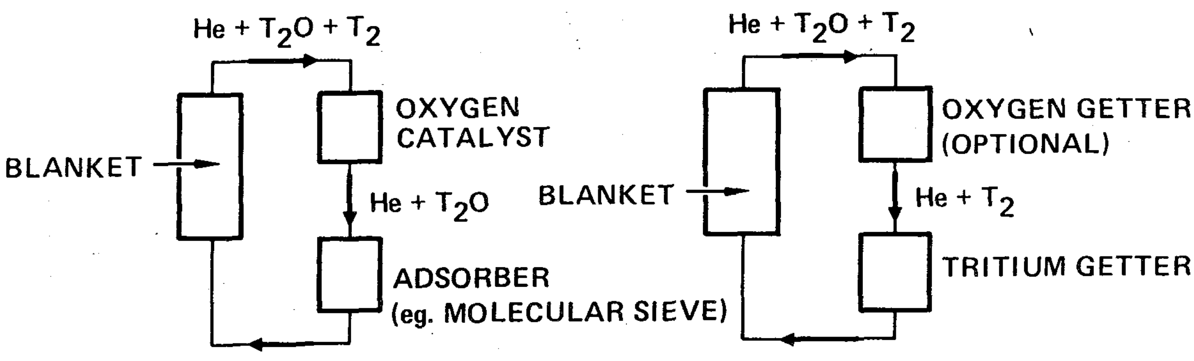
Possible process flow schematics and processing methods for each case are summarized in Fig. 1-12 and Table 1-18. The key experimental parameters for studying tritium extraction from each of the carrier fluids (i.e., basic breeder concepts) are summarized in Table 1-19. Experiments are needed to explore the feasibility of tritium recovery from the three potential carrier fluids under the sets of conditions listed, and to evaluate the operating



TRITIUM EXTRACTION FROM LiPb



TRITIUM EXTRACTION FROM Li



TRITIUM EXTRACTION FROM He

Figure 1-12. Schematic representation of tritium processing schemes

Table 1-18. Tritium Processing Methods for Different Tritium Carrier Fluids

T Carrier Fluid	Tritium Form T_2/HT	Tritium Form T_2O/HTO	Extraction Method[a]	Minimum T Concentration X_p (appm)[b]	Other Application of Method
LiPb	X		Vacuum degassing		
	X		<u>Extraction with counter-current He flow</u>	0.01-10[e]	None
	X		Permeation combined with catalytic oxidation		
Li	X		Absorption with solid getters		
	X		<u>Extraction with molten salt</u>	7	None
He	X	X[c]	<u>Absorption with solid getters</u>	10^{-5}	Fuel clean-up
	X[d]	X	<u>Adsorption with molecular sieves</u>	10^{-5}	Air detritiation
	X[d]	X	Freezing out in cold traps		
H_2O		X	Vapor phase catalytic exchange Liquid phase catalytic exchange Electrolysis		CANDU reactor coolant clean-up

[a]Preferred method underlined.

[b]Tritium concentration at processing system outlet from various design studies (LiPb) and experiments (Li,He).

[c]Additional process needed to decompose T_2O, HTO.

[d]Additional process needed to oxidize T_2, HT.

[e]Dependent on design and cost tradeoffs.

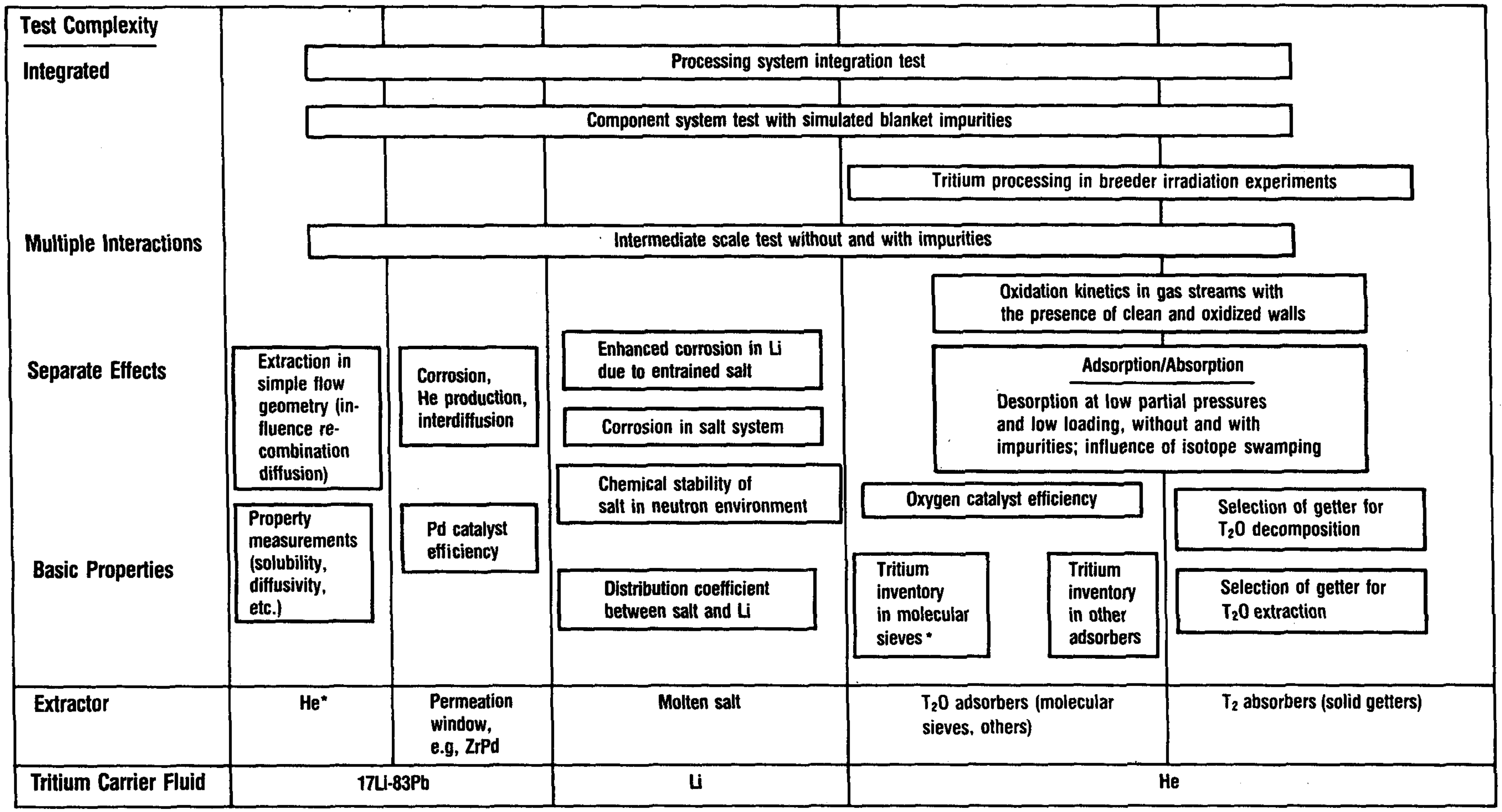
Table 1-19. Range of Inlet Parameters for Tritium Extraction Systems

	Tritium Carrier Fluid			
			He	
	LiPb	Li	BPS[a]	CPS[b]
T Composition:				
Fraction as HT, T_2 (%)	100	100	1-100	0-99
Fraction as HTO, T_2O (%)	0	0	0-99	1-100
Tritium Partial Pressure, P_{T2} (Pa)	10^{-4}-1	10^{-7}-10^{-5}	0.1-10	10^{-5}-10^{-2}
Hydrogen Partial Pressure, P_{H2} (Pa)	0-10	0	0-100	0-10
Oxygen Partial Pressure, P_{O2} (Pa)	0	0	0	0-10
Impurity Levels (appm)	< 10	> 10	< 10	< 10
Temperatures, T (°C)	400-600	450-600	300-500	275-510
System Pressure, P (MPa)	0.1-3	0.1-3	0.1	5

[a]Blanket Processing System
[b]Coolant Processing System

characteristics (including reliability and tritium inventory) of the applicable processing systems. These experiments, with few exceptions, do not require neutrons. The experiments are laid out in more detail in Fig. 1-13.

The experiments of less complexity (indicated in the table) can be done in gloveboxes, with relatively modest costs (≈ 10-100 K$), required to provide tritium handling capability. The more complex and integrated effect experiments require increasingly elaborate facilities. At some latter stage, interfacing must be done between the breeder extraction and the fuel reprocessing systems.



*Preferred solution; feasibility has to be proven

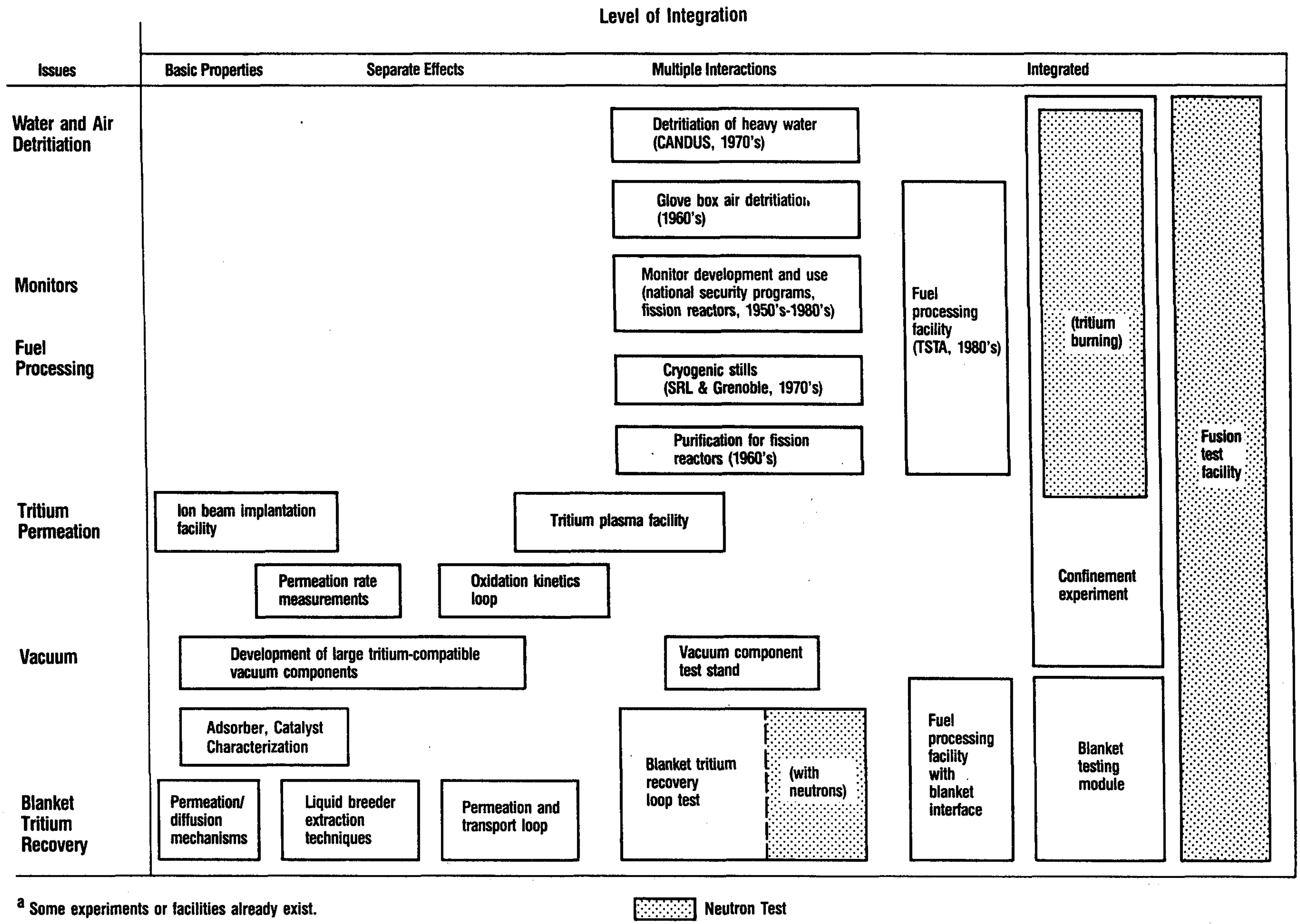
Figure 1-13. Breeder tritium extraction experiments

1.3.4 Vacuum Systems

Two issues related to vacuum systems have been identified. These are the optimum design and expected lifetime of cryopumps and the design and lifetime of large (1-m diameter) all-metal vacuum valves for the plasma chamber. Implicit in these is the issue of maintaining adequate chamber vacuum conditions.

1.3.5 Summary of Tritium/Vacuum Experiments and Facilities

Figure 1-14 displays a summary of experiments and facilities for the tritium processing and vacuum systems. The figure shows the types of experiments and facilities to address various issues for different levels of integration.



[a] Some experiments or facilities already exist.

Figure 1-14. Types of experiments and facilities for tritium processing and vacuum systems[a]

1.4 Plasma Interactive Components

Plasma-Interactive Components (PIC) include those components whose functional requirements and operating environment are strongly determined by the plasma. These components have important nuclear issues. The PIC elements of particular concern in nuclear technology are the impurity control and exhaust system and the in-vessel elements of the plasma heating and fueling systems (e.g., rf antenna).

1.4.1 Issues and Testing Needs

The PIC issues can be divided into the following categories: 1) particle exhaust, erosion and recycling; 2) high heat flux (HHF) removal and thermomechanical response; 3) disruptions; 4) tritium permeation and retention; and 5) irradiation effects.

The first category of issues is concerned with the coupled interaction of the plasma with the PIC surface, leading to uncertainties in edge conditions and surface lifetime. A particular concern is the sputtering erosion caused by the energetic plasma edge particles when they strike the surface of PICs. It is expected that the sputtered particles will enter the scrape-off region and possibly the main plasma, will be recycled and eventually will redeposit on other exposed surfaces. The redeposition process is important for reducing the impurity level in the plasma and for extending the erosion lifetime of PICs. The uncertainties also include surface conditioning methods in order to minimize plasma impurity influx due to outgassing and high voltage breakdown.

The uncertainties in the plasma edge physics conditions greatly affect the feasibility, performance and lifetime of PIC components. One such condition can be stated in terms of the plasma edge temperature. At low edge temperatures (< 50 eV), high-Z materials such as tantalum or tungsten are feasible because self-sputtering is a minor concern and they offer the potential for high performance, resistance to disruption and longer lifetime. At medium edge temperatures (~ 100-300 eV), self-sputtering is a major concern and the only suitable candidate materials are those with low Z; for example, beryllium or graphite. However, the erosion rate is very large and the lifetime critically depends on the rate and uniformity of redeposition.

The second category relates to the energy removal and recovery requirement and to the thermomechanical considerations associated with the high heat fluxes seen by PICs. Although PICs in mirror end plugs and in tokamaks are subject to average heat fluxes of about 0.2 to 0.5 kW/cm^2, peak heat fluxes of up to 2 kW/cm^2 are possible in some locations. The key issues in this category include coolant/surface heat transfer limits, coolant flow distribution and stability, channel erosion, thermal fatigue, bond integrity and heat source profile.

Water coolant has been used extensively in conceptual reactor design studies. Water appears to be the best possible candidate coolant for near-term fusion devices. However, exploring other coolants is desirable and may even be necessary in the long term. For example, if a liquid metal blanket is utilized, safety considerations preclude the use of water in components such as PICs which are near the blanket. The use of liquid metals in PICs involves most of the issues discussed for liquid metal blankets. Some of these issues, such as pressure drop, heat transfer and temperature level, appear more difficult for PICs because of the higher heat fluxes involved and of the necessity for thicker walls to withstand particle erosion.

Disruptions, which are observed in all current tokamaks, are characterized by a rapid reduction in the plasma current accompanied by the localized deposition of much of the plasma energy on an interior surface. There are uncertainties in the magnitude of the resulting induced forces (which determine to a large extent the required vacuum vessel and PIC structural support) and of the heat fluxes (which can cause surface vaporization and melt layer formation).

The extent of tritium permeation through and retention in PICs is a significant uncertainty in assessing the safety and tritium handling requirements of fusion reactors. Although a considerable data base has been generated for hydrogen isotope interaction with unirradiated structural first wall materials, little is known about the tritium interaction with low-Z materials such as graphite.

Finally, there are many uncertainties associated with radiation effects which are generic to all fusion systems. Of particular concern are the effects of radiation damage on the structural material mechanical properties (including swelling, ductility, strength, fatigue and crack growth behavior) and on the thermophysical properties such as thermal conductivity of plasma-side materials (coating or tiles), bonds and insulators.

It is important to distinguish between short and long-term radiation effects on PIC materials. There are some radiation effects that occur on a short time scale; for example, reduction in the thermal conductivity of graphite that occurs within days of operation of a reactor. Such short-term radiation effects are critical to establishing the feasibility of PIC design concepts. Other long-term radiation effects, e.g., swelling and embrittlement of some structural materials for the heat sink, relate to the component lifetime. The importance of such issues depends greatly on the resolution of erosion/redeposition issues which appear now to be more limiting than irradiation effects.

1.4.2 Parameter Ranges for Testing

The development of an experimental test plan to address the uncertainties associated with these PIC issues must accommodate both the near-term needs of plasma confinement experiments and the long-term needs of reactor systems. Thus, the parameter ranges for testing change with time and dictate the degree of attention required for a particular category of issues.

For instance, increasing pulse lengths coupled with larger amounts of injected power require that limiters or other PICs in direct contact with the plasma be actively cooled in the next generation of devices (e.g., JET, JT-60 and Tore Supra). Thus, problems of high heat flux removal and associated thermomechanical behavior require immediate attention, while erosion and redeposition, tritium permeation and retention, and effects of 14 MeV neutrons will remain secondary issues until large charged particle and neutron fluences can be realized. However, it is important not to ignore these latter issues because of the required lead time to understand their effects and develop solutions.

1.4.3 Existing and Required Experiments and Facilities

Figure 1-15 illustrates the testing requirement for each category of issues, organized according to the level of integration of the experiment. A summary of existing and required experiments and facilities is given below.

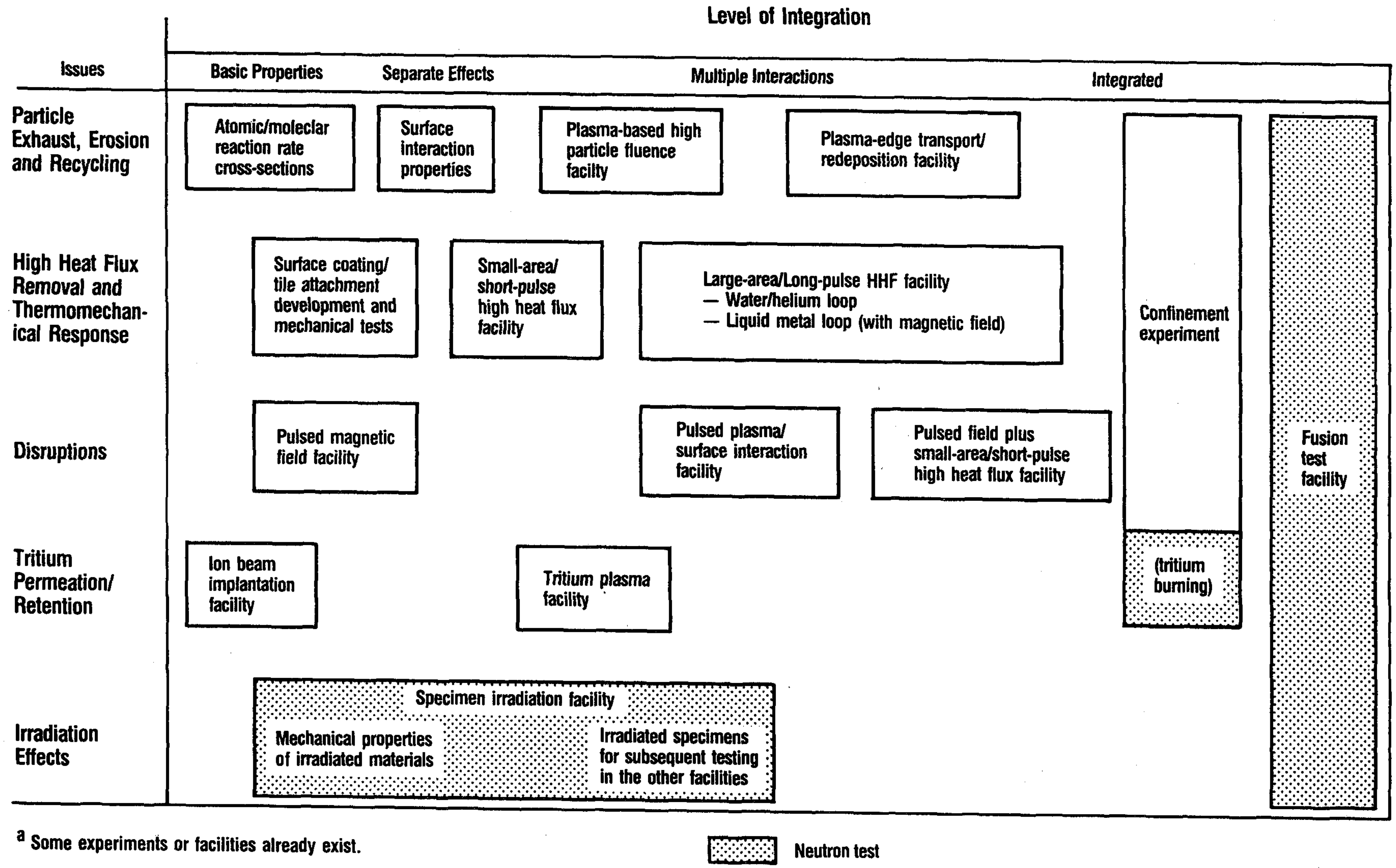


Figure 1-15. Types of experiments and facilities for plasma interactive components[a]

Survey of Existing Experiments and Facilities

The principal on-going work in the area of erosion/redeposition and conditioning is the post-plasma-exposure analysis of surfaces of PICs (removed from machines such as PLT, PDX, TFTR and TEXTOR). Analysis techniques, such as Rutherford backscatter and nuclear reaction analysis, give a detailed description of near-surface composition which allows for measurements of surface coating erosion rates, impurity deposition and hydrogen isotope implantation depths. These experiments, however, lack the large ion fluxes needed to evaluate the redeposition process and include no neutron effects. PISCES (presently operating at UCLA) is a high-particle-fluence experiment that can provide some information on erosion and redeposition.

Nuclear reaction analysis of post-plasma-exposure PIC surfaces also gives some information on the behavior of hydrogen isotopes in materials. The Tritium Permeation Experiment (TPX), located at SNLL, is studying the interaction of various materials with a tritium plasma and contributes greatly to the tritium permeation and retention data base but neutron effects are not included.

A considerable amount of high heat flux (HHF) testing is being conducted in response to the cooling problems associated with HHF components in existing and near-term confinement experiments. Operating test stands, which include a number of electron and ion beams, are used to measure material thermophysical properties, to evaluate bonding and brazing techniques, to study critical heat flux, to test heat removal capacity of coolant systems and to investigate channel erosion. HHF test stands have been used to study melt layer formation under pulsed high heat fluxes, while the induced currents and forces created by transient magnetic fields are investigated at the FELIX facility of ANL.

Existing facilities for neutron irradiation include fission reactors, such as EBR-II and FFTF, and some point neutron sources such as RTNS-II. Some material specimens are being irradiated, but much more useful testing can be done (subject to the limitations of the neutron energy spectrum and achievable fluence levels).

Required Experiments and Facilities

Uncertain basic material properties (such as sputtering yields and atomic and molecular reaction rate cross-sections) that affect the erosion and redeposition characteristics of candidate material, can be measured in well-calibrated and controlled test facilities. High particle fluence test facilities are needed to understand surface phenomena under high erosion rate conditions. Table 1-20 indicates that possible non-fusion facilities include a small facility to address local erosion and redeposition effects (e.g., PISCES), and a larger one (e.g., the proposed ICTF facility) with a more reactor-relevant edge conditions. However, there may be limits to the ability to usefully simulate reactor edge conditions outside of a reactor-relevant confinement device.

A basic program is needed to develop the special surface materials and bonding techniques assumed in many designs. The thermal and mechanical properties of the interface between the plasma-side and substrate materials under HHF conditions is of particular concern and will require testing. Small-area and short-pulse high surface heating facilities can provide data on the thermal resistance characteristics of candidate coating materials and bonds. However, larger area and long-pulse HHF facilities will be needed to understand and provide data on critical heat flux, heat transfer characteristics and stresses over useful sections of in-vessel components, and to understand the flow behavior of HHF coolants.

With respect to disruptions, measurements in confinement devices are needed to identify the characteristics of disruptions. Non-confinement experiments may also be included to understand and model the complex processes related to mechanical stresses, energy deposition and melt layer formation.

Tritium permeation and retention can be related to plasma-driven implantation processes or pressure-driven processes. The former is of more interest for PICs and the required tests include understanding the basic properties and behavior with ion beam implantation facilities as discussed earlier in the tritium processing system section. Beyond this, measurements in plasma-based facilities could be useful to explore effects under more realistic surface bombardment conditions. Advantages of using tritium instead of hydrogen or deuterium in these tests include the higher accuracy obtained with smaller amounts of tritium and the correct accounting of any isotopic effects.

Table 1-20. Parameters of the PISCES and ICTF High Particle Flux Facilities

Parameter	PISCES Maximum	PISCES Typical	ICTF Proposed
Gas	--	H,D,He,Ar	H,D,He,Ar
Operating Time (hr)	continuous	4-8	continuous
Ion Flux (H^+/m^2-s)	2×10^{23}	10^{22}-10^{23}	3×10^{22}[a]
Charge Exchange/Ion Flux	--	--	0.5
Energy Flux (MW/m^2)	10	--	2[a]
Density ($1/m^3$)	5×10^{19}	10^{18}	10^{19}
Plasma Area (cm^2)	100	50-80	300
Electron Temperature (eV)	25	6-20	50
Ion Energy/Temperature (eV)	1	0.5	50-200
Sample Bias (V)	1000	50-500	50-150
Magnetic Field (T)	0.2	0.025-0.08	5
Base Pressure (Torr)	--	5×10^{-8}	--
Ionization	disc cathode hollow cathode	10-50% 40-90%	-- --
Hydrogen Ionization Length (cm)	--	0.5-200	--
Iron Ionization Length (cm)	--	0.1-50	--

[a]At 15° angle of magnetic field with target; angles from 5-90° possible.

Irradiation effects are observed by the irradiation and associated testing of specimens in suitable neutron facilities. Fission reactors and some DT point sources are available and will be very useful but the relevance of the environment may limit the usefulness of these tests for some materials. Irradiated specimens can also be used in other facilities to measure synergistic effects such as the effects of irradiation on tritium trapping.

1.5 Radiation Shield

1.5.1 Introduction and Issues

The radiation shield must reduce the radiation damage and nuclear heating rates below the design criteria for the radiation sensitive components (such as superconducting magnets, some elements of plasma heating and exhaust systems, and diagnostic equipment). In addition, the biological dose should be less than the regulatory level. Although many shield designs exist, the design criteria are often not well established. Some of these design criteria are based on untested assumptions and incomplete models. The uncertainties will impact the construction and operating cost, availability, maintainability, and lifetime of the reactor.

Shielding uncertainties lead to design conservatism in order to provide a safety margin. A high degree of design conservatism could impose an unacceptably high cost on a test facility or reactor. However, reducing the prediction uncertainty in the shield performance imposes research costs.

The key issues relevant to the radiation shield have been defined by considering the sources of design uncertainties and are presented in Table 1-21. These are generic issues for the various blanket concepts and confinement systems. The requirements to resolve these issues are based on evaluation of the required accuracy; a review of existing experiments, data base, and design methods; and a definition of the type and characteristics of the experiments and facilities needed to resolve the issues. Since neutrons are critical in shielding experiments, the characteristics of point neutron, fission and fusion sources are discussed below. The requirements for the test module geometry were also investigated.

1.5.2 Status of Experiments, Data and Methods

An evaluation of the required and present level of accuracies is useful in planning the experiments. The present level of accuracy has been estimated based on the available experiments. LLNL has done spectrum measurements on spherical geometry which provide systematic data for many materials. These have been widely used to evaluate the adequacy of basic nuclear data and

Table 1-21. Radiation Shield Issues

1.	Design criteria of sensitive components in superconducting magnets, vacuum equipment, plasma heating systems and control system
2.	Effectiveness of bulk shield • composition, thickness of shield materials • deep penetration of high energy neutrons (14 MeV) including cross-section windows
3.	Penetrations and their shield effectiveness • streaming and partial shield • modeling procedure
4.	Occupational exposure • induced activity and dose distribution • radioactive corrosion materials • remote maintenance system
5.	Public exposure • sky shine • radioactive waste of shield materials
6.	Shield compatibility with blanket, heat transport system, and magnet, including assembly/disassembly and magnetic field penetration

processing methods. The bulk shield and streaming experiments performed at ORNL provided data needed to know the prediction accuracies of calculations. At JAERI, bulk shield experiments for SS316 with a thickness of 30-110 cm have been carried out. Neutron spectrum measurements have been performed at Osaka University for many shield materials with various thicknesses. No serious discrepancy has been observed except for certain induced activities. However, there is very little data to estimate the present accuracies for many nuclear responses.

Several reviews of the status of the data base and calculational methods are available. At present, the evaluated nuclear data files, ENDF/B-V, ENDL (U.S.), JENDL-2 and -3 (Japan), and EFF (EC), are extensive and widely used. International collaboration and data exchange are in progress. The processing methods seem to be satisfactory. Considerable efforts have been expended to produce activation data libraries and develop codes for computing activation levels. The reliability of numerical methods and the accuracy of the data should be evaluated by comparing predictions with experimental results.

1.5.3 Required Experiments and Facilities

The planning of issue-relevant experiments should be timely, systematic and cost effective. The types of shielding experiments are categorized as:

- measurements of differential nuclear data,
- neutron and gamma ray transport in bulk shield and penetrations, and response of shielding parameters,
- multiple or integral effects on components with complex geometry.

Examples of these experiments are presented in Table 1-22 for each important issue.

Since neutrons are critical in shielding experiments, the performance and specification of neutron source facilities are essential in planning the experiments. The basic experiments usually use small specimens, hence the required volume is small. Experiments on transport phenomena need relatively large volumes; for example, the area should be several mean free path lengths square and the thickness should be deep enough to achieve several orders-of-magnitude attenuation of shielding parameters. In the next 10-15 years, point or small volume sources will be used to resolve the issues. There are basically three options (cost estimates are shown in parenthesis):

- construction of a new point neutron source facility (10 $M)
- modification of conventional point source (2-5 $M)
- utilization of RTNS-II, FNS, and/or LOTUS.

The third option results in the lowest costs but requires changes in existing programs and also some small modification of the facilities.

In addition to point source testing, fission reactors seem to be attractive in some respects. There are some fission reactors built for shielding experiments which have test zones with large volumes and high fluences. Comparison calculations have been made to examine the possibility of using fission sources. The neutron spectra below a few MeV are quite similar for 14 MeV and fission sources through the whole shield region. It was found that most of the nuclear heating and dpa rates arise from the energy range below 2.5 MeV. Hence, fusion conditions can be simulated by fission sources. However, the simulation of gas production rates would be difficult due to their high threshold energy.

Table 1-22. Examples of Shielding Experiments

Issues	Basic	Separate Effect	Multiple and Partially Integrated Effect
Bulk Shield	Cross section of main nuclides [σ_t, $\sigma_e(E,\mu)$, resonance and window]	Attenuation in stainless steel, lead, tungsten, concrete, copper (10 ~ 100 cm)	Optimization of bulk shield
Penetration	$\sigma_e(E,\mu)$	Straight duct (L/D effect, source scanning Bent duct (shape, angle) Slit (step, width)	• Penetration shield • NBI port, RF port with structure • Divertor/limiter duct and exhaust • Coolant channels • Interaction of streaming holes
Induced Activity and Dose Rate	$\sigma_{n,x}$, decay data, gamma production, $P(E_n \rightarrow E_\gamma)$	Specimen irradiation Response function	• γ dose through bulk shield and penetration • Shut down dose distribution in D-T burning device • γ dose from corrosion product • Sky shine
Design Criteria	Damage rate	Specimen irradiation Response function	Radiation damage and heating parameters in various components
Measurement Technique Development			Several[a]

[a]See Chapter 7 for details.

1.5.4 Shielding Experiments in Fusion Facilities

Shielding experiments performed in a fusion test facility have many advantages with respect to the strength and volume of the source and neutron spectrum. The required operational mode of a fusion test facility and the test module geometry have been examined for shielding experiments. A tokamak type reactor has been considered as an example of a test facility with test locations on the outboard region but the results are generally applicable to other confinement systems.

Most of the neutronics measurements can be performed in a low fluence field (~ 1 MW·s/m^2 or less) but irradiation tests, such as induced activity measurements, need higher fluences to obtain data with a high accuracy. Foil activation measurements at deep locations in the shield need a fluence of about 100 MW· s/m^2. Both pulsed and quasi-steady operations are acceptable. Some consideration will be required on the activation levels of components and test modules, particularly for shutdown dose rate measurements. Low statistical errors and signal-to-noise (S/N) values are essential to obtain data with a high accuracy.

The geometrical requirement for a shield test module has been examined in order to minimize the size within a reasonable S/N value. The module is placed adjacent to the first wall. The calculations have been performed by 1-D and 2-D discrete-ordinates transport calculation codes. The dimensions obtained are 100 cm (thickness) x 140 cm (toroidal width) x 120 cm (poloidal height). This module can provide a test zone with a 40 x 40 cm surface area at the first wall and can simulate the radial profile of a full coverage case up to r = 80 cm within a deviation of 20% from the centerline values.

1.6 Non-fusion Irradiation Facilities

The best facility for the irradiation testing of materials and components is clearly a fusion device and, for large components, it may be the only option. However, it should be possible to utilize non-fusion irradiation facilities, fission reactors and accelerator-based sources, to resolve many of the issues. The capabilities and limitations of non-fusion irradiation facilities have been evaluated and are summarized below.

The irradiation environment of a fusion device consists of 14 MeV neutrons from the DT reaction plus a large fraction (approximately 80% at the first wall) of lower energy (collided) neutrons. The total flux is approximately 2×10^{15} n/cm^2-s at the first wall at a 4-MW/m^2 wall load.

The principal question regarding the use of fission facilities is the lack of high energy neutrons. When compared to the fusion environment, this results in fewer displacements per incident neutron, possible differences in the spatial configuration of defect production, and fewer transmutations per neutron. In particular, the ratio of transmutation rate to displacement rate is generally significantly lower. This rate is considered an important criterion in simulating fusion environment irradiation effects.

Modeling and actual experiments suggest that the effects of high energy displacement cascades are not qualitatively different from those of low energy cascades. The quantitative differences (per neutron) can be estimated from low fluence fission/fusion comparisons and modeling. Hence, fission reactors are believed to be useful for displacement damage studies of fusion materials.

Fission reactor studies of gaseous transmutants have concentrated on helium in metals and tritium in solid breeders. In nickel-bearing metals, helium is produced in a two-step reaction:

$$^{58}\mathrm{Ni} + n \rightarrow {}^{59}\mathrm{Ni} + \gamma \rightarrow {}^{56}\mathrm{Fe} + \mathrm{He}$$

It is possible to achieve near-prototypic helium generation rates in fission reactors for a wide range of nickel-bearing alloys by varying the nickel content and/or isotopic composition of the nickel.

Considerable materials testing in fission reactors is now underway or planned. Beyond material property questions, there are many issues related to the interaction of materials as they are combined in a system such as a solid breeder module. To demonstrate the utility of fission reactors in testing solid breeder systems (i.e., breeder and cladding), several cases were examined.

Two blanket designs were selected (Li_2O/He/HT-9 with natural enrichment ^6Li and $LiAlO_2$/H_2O/HT-9/Be with 90% ^6Li), along with three reactors (ORR, ETR, and FFTF). The objective of this study was to determine if prototypic dpa and helium generation rates could be achieved in the cladding material while simultaneously simulating tritium and heat generation rates in the solid breeder material. The results of that analysis for the Li_2O system are shown in Fig. 1-16. This figure compares the helium production rates and displacement rates for HT-9 and the tritium production and heat generation rates for the Li_2O solid breeder. The facilities evaluated were Fast Test Reactors (FTR), Light Water Reactors (LWR), and a fusion facility at 5 MW/m^2. The figure shows that a reasonable simulation can be achieved using currently available facilities and techniques for all but the helium generation in HT-9. To achieve the necessary helium generation in HT-9 will require using 75% enriched ^{59}Ni; the achievable rates with Ni doping are shown by the dotted lines. In the past, it has been prohibitively expensive to enrich nickel to these levels; however, an isotope separation process developed by TRW shows some promise and is currently being evaluated.

Although there are many fission test facilities available, they are limited in the size of a test module they can accept, roughly on the order of 10 cm (some considerably less). This limits solid breeder testing to small breeder modules or subsections.

Though it is possible to perform considerable irradiation testing of materials and components in a fission facility, it is important to note that this evaluation is based on known phenomena. It is possible, or even probable, that the higher energy neutrons will give rise to unanticipated phenomena. For this reason, fission reactor testing cannot be considered a complete substitute for high fluence testing in a fusion spectrum.

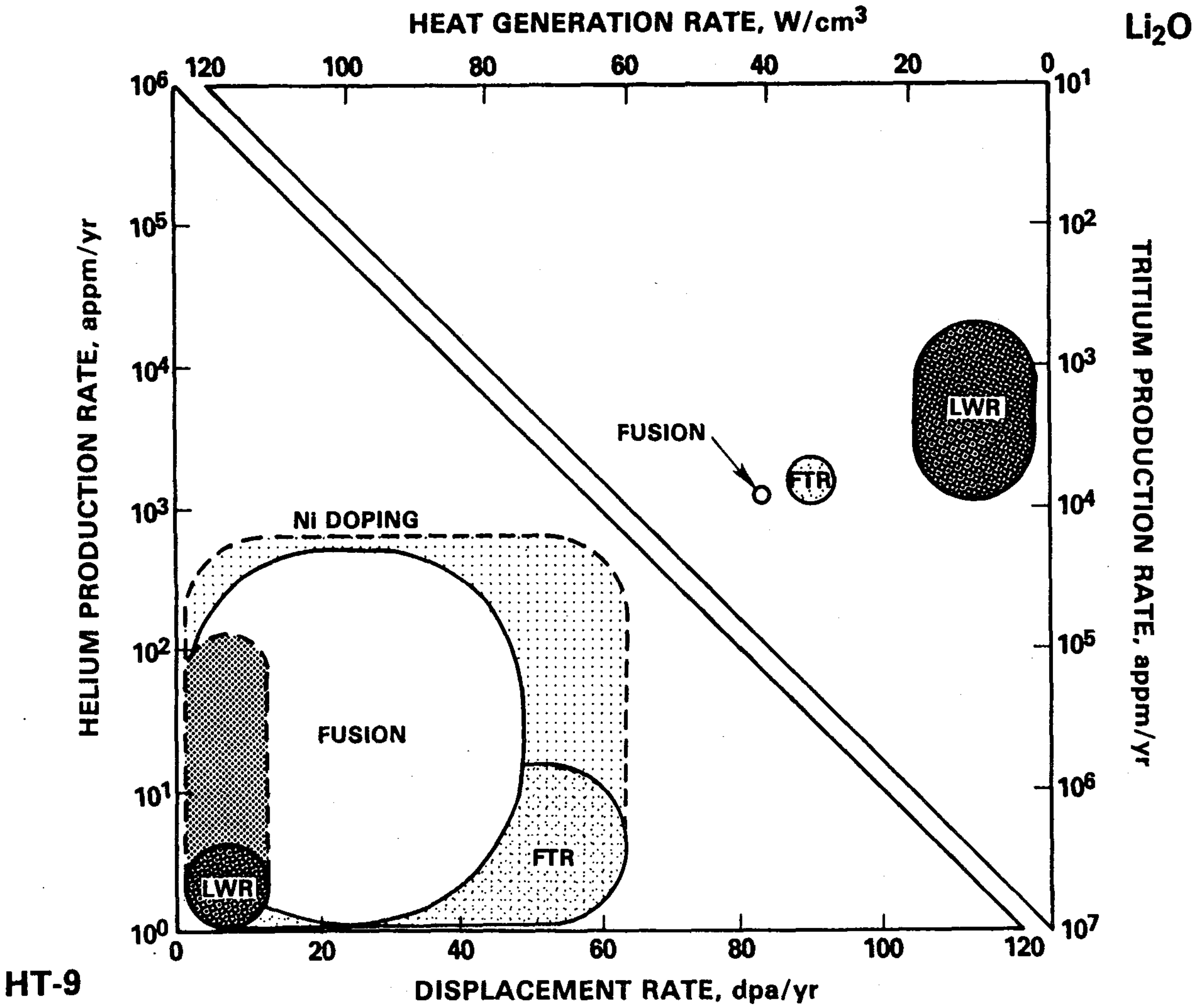


Figure 1-16. Simulation of Li_2O/He/HT-9 fusion blanket in fission reactors

The second type of non-fusion irradiation facilities, accelerator-based sources, are better able to match the fusion spectrum. These facilities produce neutrons through the interaction of accelerated charged particles and a target material. The ideal characteristics of such a source for fusion irradiation testing are:

1. Capability of producing high damage rates-on the order of 100 dpa/year. In a metal, this translates to a 14 MeV neutron flux of 10^{15} n/cm^2-s.

2. A spectrum which produces qualitatively the same damage as a fusion reactor. The rate and type of transmutations produced per dpa are the best indicator of this characteristic. The criteria most often used is a He/dpa ratio of 10 in steels.

3. Sufficient volume to test components as well as materials. Flux and spectrum gradients should be prototypic.

4. Availability over extended periods of time.

None of the currently available facilities meet all of these criteria; in fact, all are extremely deficient in one or more of the areas. The three facilities currently available within the U.S. are: 1) RTNS-II at LLNL, 2) UC-Davis Cyclotron facility, and 3) A-6 facility at the Los Alamos Meson Physics Facility (LAMPF).

The RTNS-II facility at LLNL generates neutrons by bombarding a tritiated target with deuterons. The resulting neutron spectrum yields damage similar to that predicted for a fusion device. The He/dpa ratio is 14 in iron and 12 in copper. The highest flux is estimated at 5 x 10^{12} n/cm^2-s at a distance of 0.35 cm from the target. The available volume with flux greater than 10^{12} n/cm^2-s is only 8 cm^3.

The UC-Davis Cyclotron facility produces neutrons by stopping 30 MeV deuterons in a thick beryllium target. The flux and useful volume is roughly the same as the RTNS-II facility; the spectrum peaks near 12 MeV in the forward direction. The facility availability is estimated at 2-3 days a week, which results in a large reduction in exposure compared to RTNS-II.

The LAMPF A-6 facility is the beam dump for the 800 MeV proton beam. This dump produces spallation neutrons with a spectrum similar to moderated fission spectrum plus a high energy tail that extends to the energy of the incident protons (800 MeV). Approximately 250 liters are available for neutron irradiations; the volume with flux greater than 10^{13} n/cm^2-s is much less but has not yet been determined.

The question of spectrum and damage production is important with respect to the LAMPF facility. The estimated He/dpa ratio in copper is 28, compared with 12 in RTNS-II. However, 95% of the helium is produced by neutrons over 20 MeV where the uncertainty in cross-sections is large. This illustrates the importance of the tail of high energy neutrons to the production of transmutations. It seems unlikely that the ratio of transmutations per dpa produced in the A-6 spectra would be similar for all important transmutants found in a fusion device. Certainly, there are some transmutation reactions that are energetically possible in a spallation spectrum which are below threshold in a fusion spectrum. The importance of such effects is not known.

To summarize, the RTNS-II and UC-Davis facilities provide a spectrum which is a reasonable approximation of a fusion spectrum; however, they are very limited in flux and available volume. LAMPF has considerably more volume and a somewhat higher flux, though still a factor of 10 lower than ideal. The spectrum, particularly the high energy tail, is such that more questions may be raised than answered. If an accelerator-based source is to be used for high fluence fusion technology development, a new facility is required.

The Fusion Materials Irradiation Test (FMIT) facility has been proposed to fill this need. While substantial development work has been completed on the accelerator and target, all work on completing the facility has been stopped. This facility would use a 35-MeV 100-mA steady-state beam of deuteron which impinges on a lithium target producing neutrons with a mean energy of 12 MeV and a broad peak at 14 MeV. The facility would produce a peak flux of 3×10^{15} n/cm^2-s. Calculations of He/dpa ratios are close to that expected from a fusion device first wall for all materials examined to date. The volume available at a flux greater than 10^{15} n/cm^2-s is 7.6 cm^3; for flux greater than 10^{14} n/cm^2-s, the volume is 480 cm^3. This is an adequate volume to complete a significant amount of materials testing but is too low for component testing.

It is possible to build a multiple beam accelerator which could increase the neutron production by a factor of 10. This would provide sufficient volume and flux to test some components as well as materials. It is also worth noting that some ion irradiation facilities are presently in use.

In conclusion, fission testing can provide a significant volume of high fluence data in materials, and there is some limited capability for testing of subcomponent size systems. The differences in spectrum between a fission and fusion environment preclude relying solely on fission testing, however, and it is still necessary to perform high fluence testing in a fusion environment. Accelerator-based sources can provide a fusion spectrum and will provide correlation between the fission and fusion environment. However, currently available accelerator sources do not have the capability to provide even moderate fluence levels.

1.7 Fusion Integrated Testing

1.7.1 Test Requirements

Some issues, such as failure modes and reliability, require an integrated test with full components in a fusion environment. In addition, most issues are affected in some way by the combination of all relevant environmental conditions. Without integrated testing of the nuclear components, there is substantial doubt that high availability could be achieved in a demonstration reactor or other fusion device that relied on these components.

The only suitable test facility for providing integrated testing is a fusion device. Other neutron sources have a number of significant differences such as limited test volume, neutron energy spectra differences, or absence of other environmental conditions. However, fusion test devices are expensive, particularly if reactor conditions are to be provided.

It is possible, in many cases for which the phenomena are sufficiently well-understood, to modify the design (e.g., coolant flow rate) of the test module in order to recover the important aspects of the testing issues, even though the test device parameters are not the same as those of a commercial reactor. However, a change of device parameters beyond certain limits results in the inability to maintain "act-alike" behavior. Through analyzing the behavior of components under altered device parameters and considering methods for scaling the observed behavior to that expected in a reactor, it is possible to identify a set of minimum requirements on the parameters of a fusion test facility in order for it to provide useful testing of nuclear technologies. Such analyses were performed for a range of blanket concepts. The resulting requirements are also expected to provide useful testing of the other nuclear components. These requirements are given in Table 1-23.

From a fusion technology development view-point, any fusion device that satisfies these requirements is acceptable.

1.7.2 Fusion Test Facilities

The need for such test facilities has been recognized and reflected in many design studies worldwide (e.g., FED, MFTF-α+T, TDF, INTOR, NET, FER). In some cases, the facility is viewed primarily as a reactor-relevant physics

Table 1-23. Requirements for Fusion Integrated Testing

Parameter	Reference Reactor	Test Facility Parameter Minimum	Test Facility Parameter Desirable
Neutron wall load (MW/m^2)	5	1	2-3
Surface heat load (MW/m^2)	1	0.2	0.2-0.5
Fluence ($MW/yr\text{-}m^2$)	15-20	1-2	3-6
Test port size (m^2 x m deep)	--	0.5 x 0.3	1 x 0.5
Total test surface area (m^2)	--	5	10-20
Plasma burn time (s)	Continuous	500	1000
Plasma dwell time (s)	None	< 100	< 50
Continuous operating time	Months	Days	Weeks
Availability (%)	70	20	30-50
Magnetic field strength (T)	7	1	3

device with additional engineering testing capabilities built-in or added as a later upgrade. In others, both technology development and physics experiments are comparable goals.

The primary purpose of the fusion devices considered here is to provide testing of the fusion nuclear technologies. This may change the facility characteristics and reduce costs from those usually anticipated for physics experiments. Physics information would of course be obtained, but the design is not constrained by the need to provide such data. For example, operating in a driven mode may be acceptable (particularly if it results in substantial reduction in the size and fusion power of the test device) for a technology facility, while ignition is a key goal of physics experiments. It is also possible for the technology test facility to be based on a different device concept than that of a reactor, although reactor relevance is still desirable. These technology-oriented devices are generically referred to here as Fusion Engineering Research Facilities (FERFs).

In this study, fusion test facilities were considered that could plausibly address the nuclear technology test requirements by or around the year

2000. In particular, tokamaks, mirrors and reverse field pinches (RFPs) have been considered as possible FERFs. However, within each of these device classes, there is a range of alternative approaches that depart from the conventional form and could improve the overall concept as a test facility. Organizations familiar with the individual concepts were provided minimum acceptable values of the device parameters and asked to generate devices that met or exceeded these requirements.

The representative engineering test facilities considered were:

(1) INTOR (1982 US FED/INTOR): A conventional reactor-relevant tokamak with ignited operation, inductively-driven current, RF heating and moderate-field superconducting magnets. INTOR can provide full integrated testing of plasma physics and technology, including electricity production.

(2) LITE FERF (TRW/MIT): A driven version of the LITE ignition experiments. The LITE tokamaks incorporate a high-field copper magnet and moderate beta within conventional tokamak physics assumptions. This device is able to operate in a normal mode with 1 MW/m^2 and 500 s pulse, with extended pulses (1000 s) and higher power (2 MW/m^2).

(3) "BEAN" FERF (PPPL): A tokamak with moderate-field copper coils, a bean-shaped plasma to access a stable high beta regime, and quasi-ohmic heating to ignition.

(4) IDT-DTFC (Energy Applications and Systems, Inc.): A toroidal plasma core configuration with joints on copper TF coils (and elsewhere) such that the entire fusion core can be replaced in a single operation. The example considered here is a small inductively-driven tokamak with ohmic heating and moderate beta.

(5) ST FERF (FEDC): A representative spherical torus configuration (i.e., a very low aspect ratio "tokamak") with a low fusion power, non-inductive current drive and a low magnetic field.

(6) TDF and MFTF-α+T (LLNL): Relatively recent tandem mirror designs with neutral-beam driven test cells within the central cell region. The end plug magnet and thermal barriers are similar to the TMX-U and MFTF-B plug designs. TDF can operate in a relatively high neutron wall load reference mode, plus a high plasma Q mode. MFTF-α+T is an upgraded version of MFTF-B

with the addition of a test cell, tritium burning capabilities, and (as assumed here) improved availability.

(7) RFP FERF (LANL/Phillips Petroleum): A representative reversed-field pinch configuration with copper coils and ohmic heating. Two RFP versions are considered, a 1 MW/m^2 neutron wall load reference version and a 5 MW/m^2 extended version.

The strengths and weaknesses of these concepts as fusion engineering research facilities were compared by characterizing each concept by a short list of distinct parameters that represent the overall attractiveness of each device. For technology testing, the major parameters are those that summarize performance (as a test facility) as a function of cost and risk. Since the designs were not necessarily consistent in assumptions or detail, some common assumptions were imposed with respect to availability, duty cycle, useful test area, lifetime, and capital and operating costs.

Table 1-24 gives the performance parameters of representative concepts, and Table 1-25 provides a summary of their overall performance, cost and risk. The primary performance parameters for technology testing are the irradiation capability, degree of required scaling, and burn length capability. The irradiation capability, or the ability to provide neutrons, includes the neutron wall load, device availability and test area (which is defined to include regions with adequate depth). Clearly, the larger these device parameters are, the quicker tests can be completed. These parameters can be traded amongst each other within certain bounds without affecting overall attractiveness. The ability to provide fusion reactor relevant conditions is expressed by the degree of required scaling. At 1 MW/m^2 neutron wall load, considerable but plausible extrapolation is required to predict operation at 5 MW/m^2 reactor levels so there is certainly an advantage in operating at higher neutron wall loads. In addition, many device parameters are at least indirectly related to neutron wall load, so larger neutron loads generally imply that all parameters are more reactor-relevant. Finally, present fusion concepts often operate in a pulsed mode (true steady-state operation has not been demonstrated for any fusion device). However, it is important that commercial fusion reactors operate in steady state or at least with long plasma burn lengths. Since pulsing introduces thermal and

Table 1-24. Performance Comparison of Fusion Engineering Research Facilities

	Tokamaks				Spherical Torus FERF	Tandem Mirrors		Reverse Field Pinch
	INTOR	LITE FERF	BEAN FERF	DTFC-IDT		TDF	MFTF-α+T	
Fusion power, MW	620	90	185	100	39	36	17	22-110
Electrical consumption, MWe	200	210-270	185	427	120	250	104	126-180
Neutron wall loading, MW/m^2	1.3	1.0-2.0	1.3	2.0	1.0	2.1	2.0	1.0-5.0
Surface heat flux, MW/m^2	0.1	0.1	0.2	0.9	0.1	0.3	0.1	3.5-4.4
First wall radius, m	1.2	0.8	0.75	0.59	0.59	0.3	0.25	0.3
First wall area, m^2	380	72	110	40	31	8	4	18
Accessible test area[a], m^2	38	7.2	11	4.0	3.1	4	2	3.5
Test port area/depth, m^2/m	2/1	1/1	1.5/0.8	1.2/1	1.6/0.8	1.6/0.8	0.8/0.8	1/0.3
Pulse length[b], s	200	500-1000	1000	520	SS	SS	SS	SS
Duty cycle (%)	80	90	90	90	100	100	100	100
Ultimate availability[a] %	35	45	45	45	45	45	45	45
Neutron fluence[c], $MW\text{-}yr/m^2$	3.3	4.0	4.7	7.3	4.0	8.5	8.1	4.0-20
External field on-axis, T	5.5	5.5	3-6	8	3	4.5	4.5	7-9

[a]Consistent estimate.

[b]Designs of tokamak devices, e.g., INTOR, with a plasma current drive for steady state (SS) operation were not explored here.

[c]Assuming total equal to 9 years at ultimate availability.

Table 1-25. Summary Characteristics of Fusion Engineering Research Facilities

	Tokamaks				Spherical Torus FERF	Tandem Mirrors		Reverse Field Pinch
	INTOR	LITE FERF	BEAN FERF	DTFC-IDT		TDF	MFTF-α+T	
Neutron wall load, MW/m^2	1.3	1.0-2.0	1.3	2.0	1.0	2.1	2.0	1.0-5.0
Fluence x Area/Year, MW-yr/yr	14	2.9	5.8	3.2	1.4	3.8	1.8	1.6-7.9
Pulse length, s	200	500-1000	1000	520	360,000	360,000	360,000	360,000
Physics risk[a]	2	1	7	3	8	2	2	10
Technology risk[a]	5	4	5	6	8	3	3	7
Total capital cost, M$	2800	900	1200	1200	700	1200	600	700-800
Annual operating cost, M$	251	112	155	169	74	123	56	68-117
Total cumulative cost[b], M$	5500	2000	2800	2900	1500	2500	1200	1400-2000
Total cost/useful neutron[c]	4	7	5	9	11	6	7	9-2
Useful neutrons/cost/"risk"[d]	4	3	2	1	1	3	3	1-2

[a]Larger values indicate higher risk; based on judgement of the required subsystem extrapolation.

[b]Assuming 3 years non-tritium/low-availability operation plus 9 years full-availability operation.

[c](Total cost)/(Annual fluence*area) rounded to nearest leading digit.

[d](Annual fluence*area)/(Total cost)(Physics+Technology Risk) rounded to nearest leading digit.

mechanical variations that can lead to fatigue or other effects, it is desirable to minimize these from the point of view of simulating reactor conditions. Thus, the third performance parameter is the burn length capability, or the pulse length here, since the present concepts all have high duty cycle and are assumed to be able to operate for 100 hrs continuously.

Summary risk parameters are desirable to represent "overall" physics and technology extrapolation from present data. A crude measure of "overall" risk is shown based on a cumulative assessment of the amount of extrapolation required for the major physics functions (e.g., plasma heating) and technology subsystems (e.g., magnets). The numerical values are based on zero "risk" points for a moderate extrapolation, one point for a large extrapolation (some additional testing required), and two points for a very large extrapolation (major experimental program needed).

The major cost parameters are the capital and annual operating cost. Although no detailed analysis was performed, some ground rules were adopted to provide consistency among the concepts. The direct capital cost was estimated by comparison with devices costed recently using FEDC/INTOR algorithms and based on the total power handled (electrical plus plasma) and on the fusion core size. Two possible cost-benefit figures-of-merit are also included: the cost per useful neutron (based on the total cost and the annual fluence/area product), and the useful neutrons per unit cost and "risk" (where risk is based on the sum of the physics and technology risk points). These cost-benefit parameters provide some normalization of the data but must be interpreted with due caution.

The results summarized in Tables 1-24 and 1-25 address two questions. First is the usefulness of the concept of devices for fusion technology testing. In this respect, it is clear that a wide variety of possible Fusion Engineering Reseach Facility concepts exist. All concepts considered provide reasonable performance for technology testing; the minimum requirements identified earlier are 1 MW/m^2 neutron wall load, 1 MW of irradiation capability (e.g., 1 MW/m^2 over 5 m^2 test area at 20% availability each year), with pulse lengths over 500 s. On the other hand, there is at present no facility design that can easily be built under present U.S. budget limitations without some international framework. A technology test facility may not be as costly as a combined physics/technology device, but is still an expensive proposition.

This is perhaps not surprising since costs are driven by the presence of neutrons and by the overall power level handled. With present concepts, ignited fusion devices (low electrical consumption) generally require high fusion power, while driven fusion devices (low fusion power) generally require high electrical power.

The second question is whether a particularly attractive technology test facility concept can be identified. If the facility must be built in the near term, then low risk is important, and the options are probably limited to either a moderate-beta, moderate-field tokamak or a tandem mirror with a simple test cell and end plugs. Tokamaks have a much more extensive data base, but tandem mirrors offer potentially lower device cost because they can access the lower limits of useful testing performance. Tokamaks have the advantage in unit cost-effectiveness. The cost per neutron figure-of-merit indicates the economy of scale; INTOR is the largest device and provides considerably more potential test area (although there are some questions as to its practical utility) without a corresponding large increase in cost. The spherical torus and reverse field pinch offer relatively low total power, but were also sufficiently small that the irradiation capability was limited. A high performance RFP could provide an interesting alternative if the high physics and technology risks are acceptable or can be reduced by other experiments.

Finally, several areas for improvement in fusion test facility designs are suggested by this comparison. The importance of reducing the total device power (fusion plus electrical) and maintaining a reasonable amount of test area is emphasized. Better assessments of the useful test volume and of the device costs are also needed to support a useful comparison. With respect to experiments, common high-risk technologies are the magnets and plasma interactive components. Development of these specific technologies could reduce these risk contributors and allow improved performance.

1.8 Concluding Remarks

FINESSE has developed and applied a process for the technical planning of experiments and facilities for fusion nuclear technology. The process involves: 1) characterization of issues, 2) quantification of testing requirements, 3) evaluation of facilities, and d) development of a test plan to identify the role, timing, characteristics and costs of major experiments and facilities.

The nuclear subsystems addressed are: a) blanket, including first wall; b) radiation shield; c) tritium processing system; and d) plasma interactive components. The technical issues and the development problems of the blanket are complex. The greater part of the FINESSE effort has been devoted to the blanket. The issues, experiments and facilities have been evaluated and the major features of a test plan have been developed for the blanket. Further evolution and additional details in the blanket test plan are expected. The radiation shield and tritium processing subsystems are much simpler and the required R&D resources are far less than the blanket. Accordingly, only general test plan considerations are developed. A major complication in plasma interactive components (PIC) is the strong interrelation to plasma physics and confinement experiments. This leads to many complex questions in developing a logically consistent and effective test plan for PIC. The features of the experiments presented here for PIC should be viewed as preliminary.

Significant advances have been made in understanding and characterizing the issues and required experiments and facilities for fusion nuclear technology (FNT). A general R&D framework for FNT has been developed. A major feature of this framework is the utilization of non-fusion facilities over the next fifteen years, followed by testing in fusion devices beyond about the year 2000, as illustrated in Fig. 1-17.

Basic, separate effect and multiple interaction experiments in non-fusion facilities will provide property data, explore and understand phenomena, and provide input to theory and analytic modelling development. The data base from non-fusion testing should be sufficient to: 1) quantitatively assess the economic, safety and environmental potential of fusion; and to 2) design and construct experiments for testing in a fusion device.

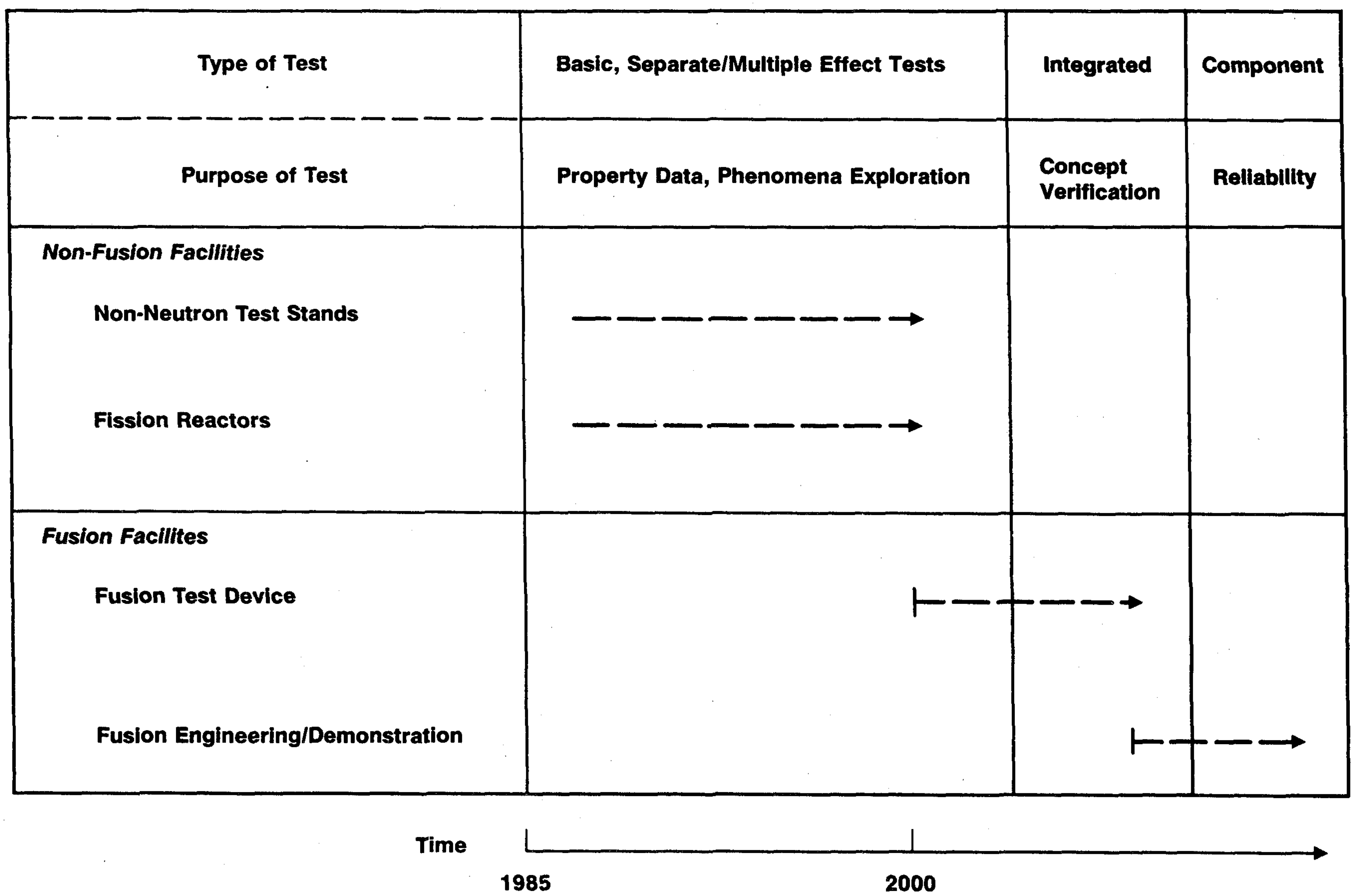


Figure 1-17. General framework for nuclear technology development

Experiments in fusion facilities can proceed in two phases. The first phase will focus on integrated testing of experimental modules to provide concept verification. Some of these modules can be partial simulation of the component while others provide an integrated simulation of all physical elements and environmental conditions within the component. Effective FNT integrated testing imposes certain requirements on some of the fusion device parameters (e.g., wall load, plasma burn time); these requirements have been quantified. Any fusion device that meets these requirements will satisfy the needs of nuclear technology testing. The second phase of testing in fusion facilities will focus on obtaining data on component reliability. System integration, in which interactions among components are present, is necessary for this advanced stage of component testing.

Blanket concepts can be divided into two generic classes: 1) solid breeder blankets, and 2) liquid breeder blankets. Within each class there are a number of design concepts that involve a variety of material and configuration choices. Analysis of issues suggests that both classes have significant engineering feasibility uncertainties, and so both liquid and solid breeders should be pursued. Further experimental and analytical effort is required to select viable concepts with the highest economic, safety and environmental attractiveness potential. The test plan developed in FINESSE emphasizes providing opportunities for innovation as well as obtaining information that can lead to early selection of material combinations and blanket configurations.

A major difference between liquid and solid breeder blankets is in the type of non-fusion facilities required. Fission reactors are the primary facilities for solid breeder blanket R&D, as they are the only means at present to provide the neutrons necessary for producing bulk heating, tritium, and radiation effects in experiments with significant volume. Liquid metal blanket issues are dominated by problems related to momentum, heat and mass transfer which can be addressed in non-neutron test facilities.

The blanket test plan defines the scope, technical characteristics, time sequence and costs of experiments, facilities and analysis. The required R&D effort defined in the test plan for the next 15 years has been summarized in terms of a number of major tasks. Each task consists of a number of facilities, experiments, and related activities aimed at resolving one or more of the critical issues.

To address the critical issues, a blanket R&D program requires an average expenditure of about 20 to 40 million dollars per year. The level of confidence in the details of the test plan and associated cost estimates are higher for the nearer term tasks. As with any test plan for a complex R&D program, the technical requirements and cost estimates for experiments and facilities beyond the next few years will need to be revised based on technical results and in response to changes in programmatic emphasis.

The R&D approach and pace for the tritium processing technology are quite different from those for other nuclear components. A unique set of circumstances have permitted advanced experimental investigation of the tritium processing issues early in the program. The Tritium Systems Test Assembly (TSTA) now in operation in the U.S., and other facilities being completed in Europe and Japan can be classified as "partially integrated" test facilities. Present plans for these facilities call for addressing the key issues of the tritium fuel processing. Two important tritium issues are not being addressed presently by TSTA-type facilities. These are 1) external blanket tritium extraction, i.e., extraction of bred tritium from the fluid used to transport it outside the blanket; and 2) tritium permeation in a number of reactor components. The tritium processing methods and associated issues are strongly dependent on the particular tritium carrier fluid. Small-scale experiments have been identified to resolve the issues of tritium extraction from helium, lithium and lithium-lead. A number of experiments have also been suggested to understand plasma-driven and pressure-driven tritium permeation issues.

The main issues for the radiation shield relate to: 1) the accuracy of neutronics prediction capabilities, and 2) the uncertainties in design criteria due to lack of data on radiation effects on some reactor components. Neither appears to be a fundamental feasibility issue at present. However, progress on these issues will help reduce design conservatism and lower the costs of fusion test facilities and reactors. The accuracy of neutronics predictions can be addressed by a) a modest program to improve basic nuclear data and calculational methods; b) integral experiments with a point neutron source; and c) maximum utilization of any fusion device that becomes available for design verification (any tritium-burning device can provide substantial information). The issue of design criteria can be addressed in existing facilities as part of the materials irradiation program for elements of radiation-sensitive components such as superconducting magnets and cryopumps.

Finally, it appears that there are special features of the R&D for fusion nuclear technology that are likely to facilitate international cooperation. As evident from present activities(2) in the world fusion programs and from the work reported here, many generically different options with a large number of distinct issues need to be addressed for fusion nuclear technology. The diversity of options and issues requires the utilization of different types of facilities. Many suitable and unique facilities exist in various countries. The immediate need for new facilities involves a number of modest cost facilities. These particular features of fusion nuclear technology R&D should facilitate developing international agreements that provide for equitable distribution of benefits and costs among the parties involved. For non-fusion testing over the next fifteen years, at least three options can be considered:

- Several nations could jointly sponsor the same, shared facilities and experiments.
- Individual nations could construct and operate separate but complementary facilities.
- Several nations could jointly sponsor a number of the larger facilities and experiments where strong common interest exists, while maintaining their own smaller or special interest experiments and facilities.

These options provide a flexible framework for planning international cooperation.

Fusion nuclear technology is an essential ingredient to bringing the attractive potential of fusion into realization. Effective international cooperation on nuclear technology will play a major role in advancing fusion.

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CHAPTER 2

BLANKET TEST PLAN

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2.1 Introduction

There are currently a large number of blanket options and a large number of issues associated with those options. Figure 2-1 indicates the most promising blanket options as they are viewed today. The test plan provides a method to optimally resolve the issues and develop blankets whose feasibility and attractiveness can be predicted with adequate certainty. It specifies the means by which decisions can be made regarding which blanket options should continue to be pursued and which experiments should be performed within the constraints imposed by time and budget. There are many complex aspects of the test plan involved in satisfying the objectives and constraints. These are considered in Appendix A, together with an analytic decision-making framework for cost/benefit/risk analysis.

In addition to providing a method to focus the research effort, the test plan defines a framework which spans the entire time period from the present to the time when a decision can be made on the ultimate attractiveness of fusion. This framework is crucial for the purpose of planning. Given our current understanding of the issues for the most promising blankets, it is possible to specify in detail the experiments and facilities required for the near term. However, the characteristics of experiments performed beyond the next 5-10 years will depend on the results of near term testing and also on future developments in blanket design - both the elimination of undesirable concepts and the addition of new, innovative concepts.

In order to perform experiments in a timely manner, it is desirable to anticipate at least some of the characteristics of planned experiments beyond the next 5-10 years, especially the cost. The test plan provides an indication of the expected directions of the long term testing program. This helps not only in the planning of future experiments, but it also provides impetus and direction for the near term testing efforts. For example, one of the primary objectives of near term experiments is to generate data to allow for reliable operation and meaningful testing in more integrated experiments, and to help select blanket designs which will be tested in later stages.

The suggested test plan for blanket research and development consists of four separate but overlapping phases, as shown in Table 2-1. Given the

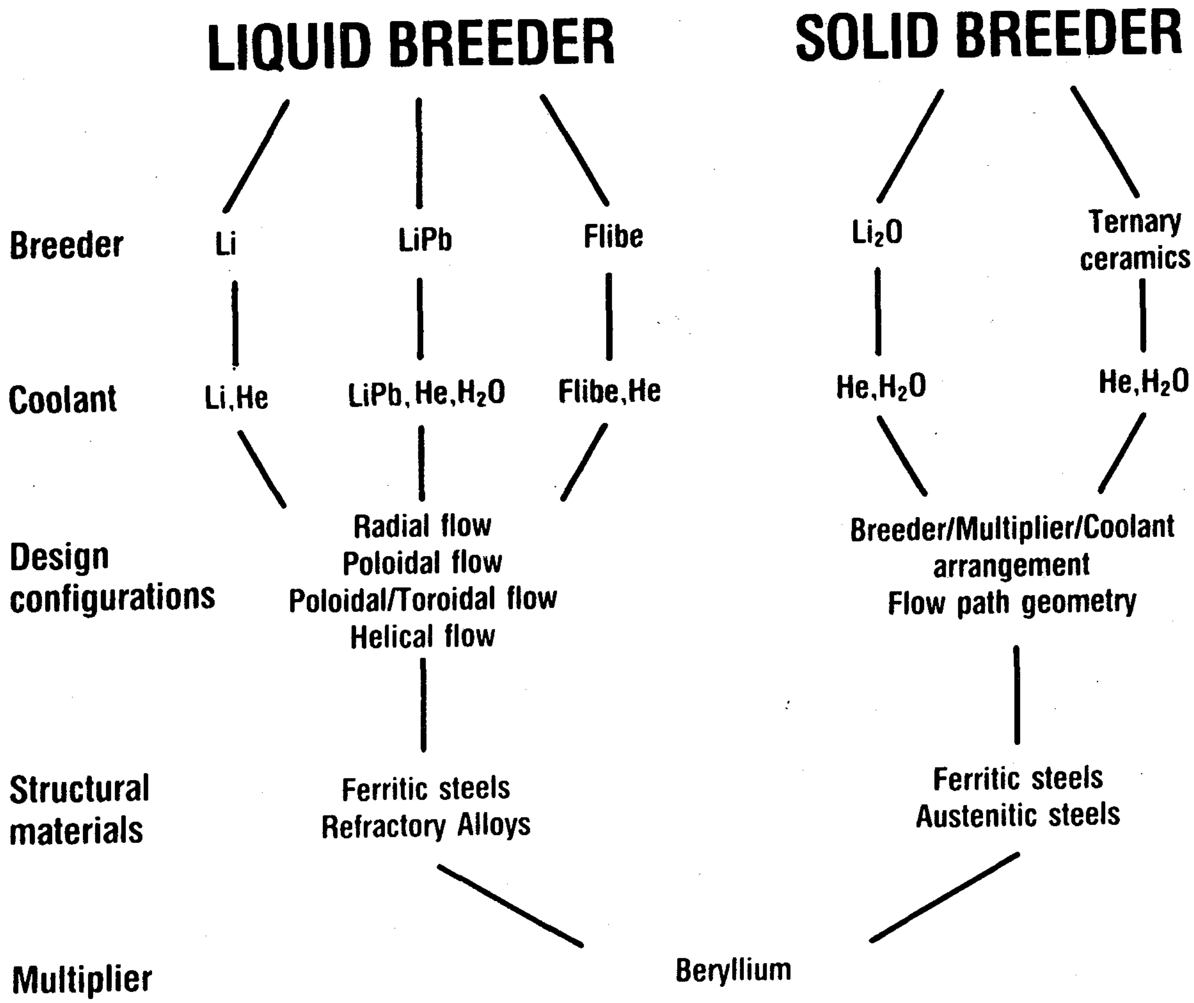


Figure 2-1. Primary blanket options (note: not all combinations of breeder/coolant/structure/multiplier provide attractive options)

Table 2-1. Phases of Blanket Testing

	Approximate Time Frame	Level of Integration	Primary Objectives	Milestones
Phase I	0-10 yrs	properties, separate effects	develop understanding of material behavior and blanket phenomena	select material combinations[a]
Phase II	5-15 yrs	multiple effects	understand phenomena, develop predictive capabilities for complex configurations	select blanket configurations
Phase III	10-20 yrs	partially integrated (non-fusion)[b]	design concept verification in non-fusion environment	select primary blanket design options for fusion testing
Phase IV	15-25 yrs	integrated	design concept verification in fusion facility	successfully operate test modules

[a]To the extent possible with limited high fluence irradiation data

[b]Non-fusion facilities for liquid breeders includes only non-neutron facilities; for solid breeders it includes both fission reactors and non-neutron facilities

currently wide range of options in materials and designs and the relatively immature state of understanding of blanket phenomena, the first phase is dedicated to the generation of scoping data for the widest possible range of options. Phase I emphasizes basic properties and separate effect tests. The ultimate goal of Phase I is to identify a limited number of prime candidate material combinations for further testing. Since the cost of test facilities becomes greater in later phases, it is important to reduce the number of candidate materials as much as possible, consistent with the desire to maintain some degree of breadth in the program. Because of the limited time available to reduce the number of candidate material combinations in Phase I, these decisions will probably be made without a large base of high fluence data on irradiated material properties. Selection and refinement of materials, incorporating irradiation data, will continue beyond Phase I.

The primary purpose of Phase II is to quantify local, design-related behavior under fusion-relevant conditions and to develop engineering data to enhance predictive capabilities. This provides a basis for assessing design configurations and limits, as well as further narrowing of material combination choices. Phase II emphasizes multiple effect/multiple interaction experiments.

Phase III provides concept verification to the extent possible in non-fusion facilities. This phase should support an assessment of the feasibility and attractiveness of blanket concepts for the purpose of selecting a limited number of candidates for testing in fusion facilities. The type of data obtained should be sufficient to adequately design blanket test modules for fusion testing.

Phase IV consists of the testing of blanket modules in a fusion device to provide design verification and obtain information on failure modes and reliability. The results from Phase IV should be the primary data base for a fusion engineering assessment and for the design and construction of demonstration reactors. Figure 2-2 illustrates the phases and overall objectives of the blanket test plan.

In the following two sections, more details of test plans are provided for both solid and liquid breeder blankets. These test plans are derived by considering the most critical issues for current blanket design concepts. In addition to the blanket designs, the issues depend on the assumed parameter ranges for the reactor environment in which blankets are expected to operate. Table 2-2 gives the parameter ranges which appear now to represent the most likely reactor conditions and which were used as the basis for defining the blanket issues and test plans.

Table 2-2. Representative Goal Reactor Parameter Ranges

Parameter	Range
Neutron wall load	4-6 MW/m^2
Surface heat load	0.2-1 MW/m^2
Fluence	15-20 $MW/yr\text{-}m^2$
Plasma burn time	Continuous
Continuous operating time	Months
Reactor Availability[a]	70-80%
Magnetic field strength	5-7 T

[a]Implies a required availability for the entire blanket of >95%; the required availability per blanket module is much higher.

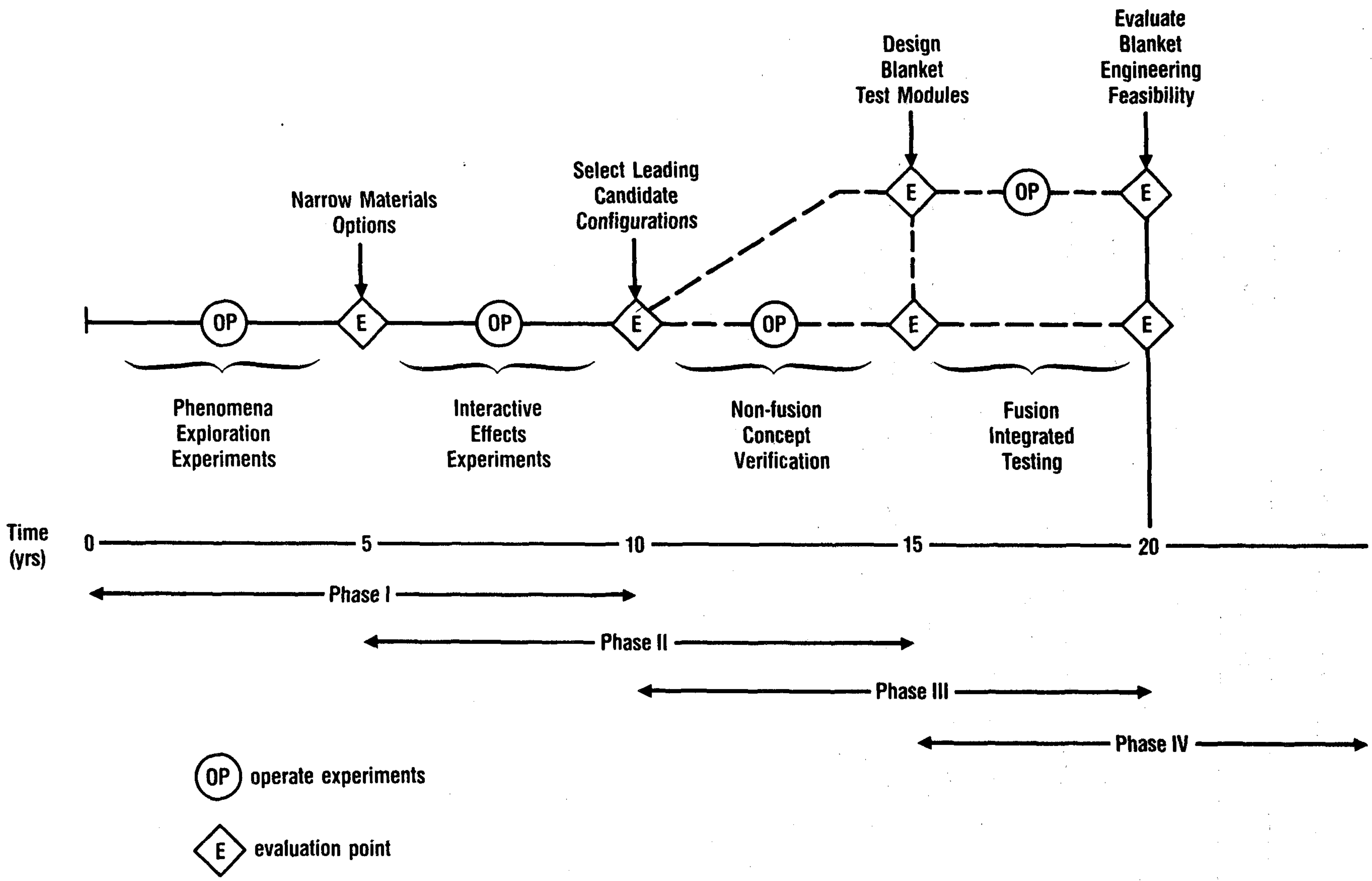


Figure 2-2. Overall objectives and phases of blanket test plan

2.2 Solid Breeder Blanket Test Plan

2.2.1 Introduction

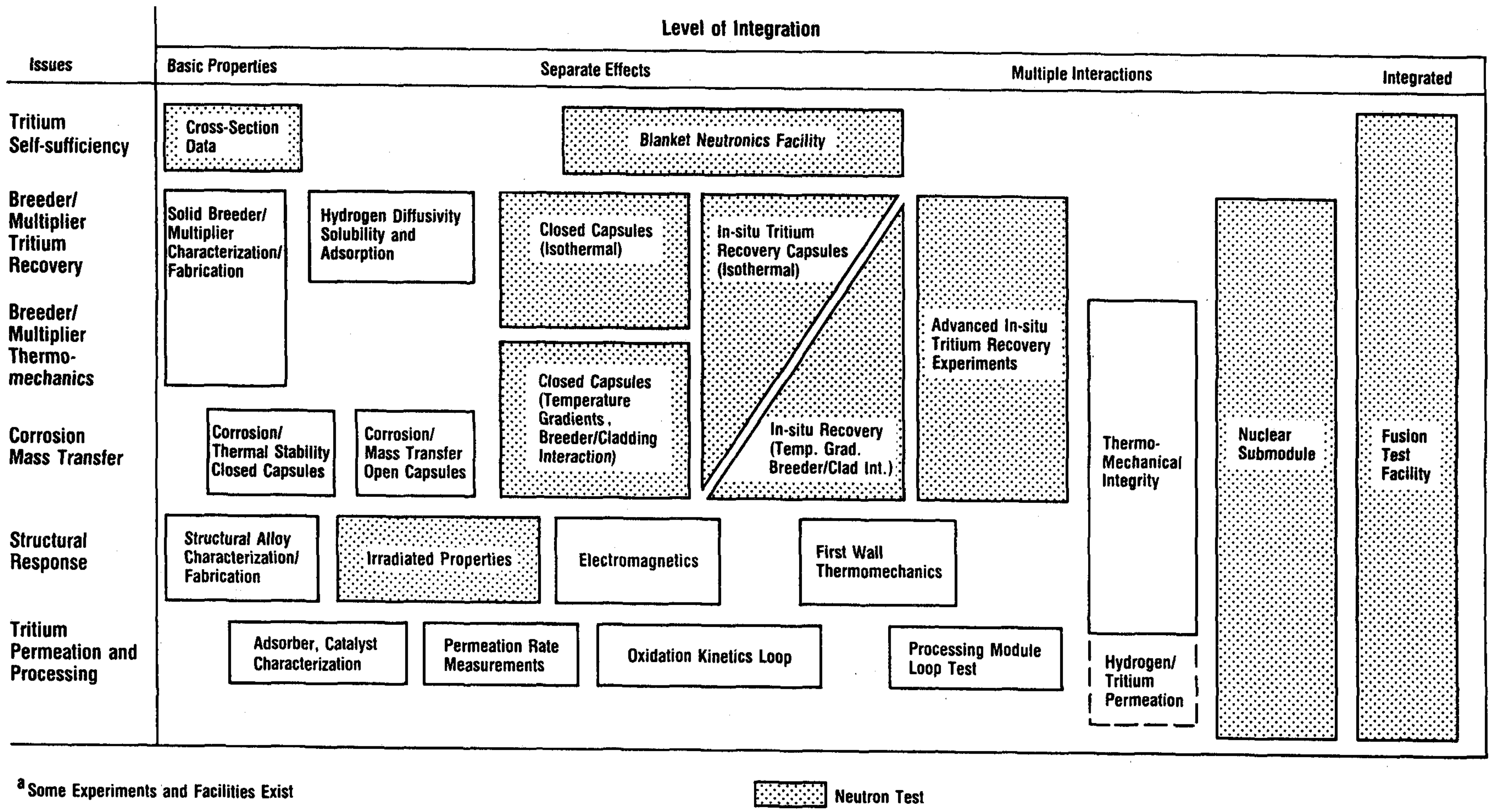
The most important concerns for solid breeder blankets are related to tritium breeding, tritium recovery and breeder thermomechanical behavior. These are important because: 1) there is limited understanding of tritium transport in irradiated solids; 2) fairly complex designs are used to keep these solids within their temperature limits under substantial heating and neutron damage rates; and 3) the resulting designs have a significant amount of non-breeding structure, coolant and other material. The major classes of issues related to these concerns are summarized in Table 2-3.

These issues can be addressed by a range of possible experiments as illustrated in Figure 2-3. The experiments are organized according to level of integration, from basic properties, to phenomena exploration in single and multiple effect tests, to concept verification in integrated fusion tests. In general, more than one experiment is needed to fully address each issue.

It may be observed that the solid breeder blanket issues and the corresponding testing needs have some unique characteristics, especially with respect to liquid breeder blankets. First, there are a large number of potential breeder materials and material variables (e.g., grain size) that can be altered to produce unique properties. Secondly, the influence of geometry on the primary uncertainties is not large. The most significant uncertainties are related to basic properties or to local behavior (e.g., within a pellet). Thirdly, the influence of radiation on the behavior in general, and

Table 2-3. Generic Classes of Solid Breeder Blanket Issues

Tritium self-sufficiency
Breeder/multiplier tritium inventory and recovery
Breeder/multiplier thermomechanical behavior
Corrosion and mass transfer
Structural response and failure modes in fusion environment
Tritium permeation and processing from blanket



[a] Some Experiments and Facilities Exist

Figure 2-3. Types of experiments and facilities for solid breeder blankets[a] the uncertainties in particular, is large. Radiation damage and transmutation can substantially alter the original material. Finally, much of the important functional behavior of the solid breeder is not described by classical equations, but rather the controlling phenomena must be quantified by experiments. Therefore, the test program is very empirical and it will be difficult to confidently scale from test conditions that differ from reactor conditions.

A test program for the development of solid breeder blankets can proceed with the four overlapping phases outlined earlier in Table 2-1. Given the wide range of possible materials, the initial emphasis is to explore phenomena and develop an understanding of how the material behavior is affected by the material variables. It is important to identify the prime candidate materials early since the results of subsequent testing will be material-specific and may not be easily applied to other materials. The purpose of the next phase is to quantify local design-related behavior under fusion-relevant conditions. This provides a basis for assessing configurations and limits. The third phase provides concept verification to the extent possible in non-fusion facilities. This should be sufficient to support an assessment of the feasibility and attractiveness of solid breeder blankets. The design codes that are to be developed and calibrated by this point in time would be used to define commercial reactor blankets. The fourth phase would emphasize the testing of components in a fusion device to provide design verification and obtain information on failure modes and reliability. These phases overlap in practice because the distinction between the experiments is not abrupt, and because test results can lead to suggestions for further earlier phase tests (e.g., additional measurements of material properties).

General tasks can be identified based on the issues. However, there is no one-to-one correspondence because the more integrated tests address multiple issues. The major tasks are identified in Table 2-4. Some tasks have several smaller associated subtasks or experiments, while other tasks involve a few large experiments. These tasks are defined in subsequent sections based on the issues, experiments and estimated costs discussed in more detail in Chapter 4.

Table 2-4. Major Solid Breeder Blanket Tasks for the Next 15 Years

Solid breeder material development and characterization

- Measurement of tritium retention and release, including effects of burnup, material parameters, and purge flow;
- Thermophysical and thermomechanical properties, including effects of irradiation and material variables;
- Development of sphere-pac material;
- Assessment of novel materials, such as lithium beryllates;
- Development of fabrication and recycling techniques.

Multiplier material development and characterization

- Measurement of swelling in beryllium irradiated at temperature, including effects of form and porosity,
- Measurement of tritium retention and release, particularly the effects of form and irradiation;
- Measurements of irradiation creep and mechanical properties;
- Development of low-loss-rate fabrication and recycling techniques.

Blanket thermal behavior

- Measurements of corrosion, mass transfer and chemical interaction kinetics, particularly for Li_2O and beryllium-containing materials;
- Measurements of breeder/multiplier temperature profile and thermomechanical effects of breeder/cladding interaction;
- Non-neutron blanket (sub)module thermomechanical integrity, including cycling, corrosion, normal transients, and severe transients.

Neutronics and tritium breeding

- Simple geometry mockups for important blanket material combinations;
- Engineering mockups of blanket designs and adjacent reactor sector.

Advanced in-situ tritium recovery

- Two or more instrumented and purged assemblies with multiple capsules.

Nuclear submodule experiments

- Two or more nuclear submodule assemblies

2.2.2 Solid Breeder Material Development and Characterization

The basic material in solid breeder blankets can be tailored to some degree to provide specific properties. The objective of this task is to fabricate, characterize, and improve the properties of candidate breeder materials. In addition to standard property measurements, this task includes closed or open capsule irradiation of material specimens. The immediate goal is to provide basic data for candidate breeder materials to support blanket designs and provide a basis for the selection of materials (e.g., Li_2O or $LiAlO_2$) and material parameters (e.g., grain size, sintered versus sphere-pac). In the long-term, this task will seek to optimize the properties of selected materials, and to develop fabrication techniques that can be extrapolated to commercial operation.

A sufficient data base on all candidate materials is needed to support an assessment of their feasibility (i.e., at least thermal stability, thermal conductivity and tritium diffusivity). Also, some understanding of the many material-related variables is necessary to identify directions for improving the properties. Particularly important are temperature, grain size, porosity and pore size distribution, impurities or additives, fabrication process, material form, burnup, container material, and purge gas flow rate and composition. The sphere-pac form offers attractive features, and material specimens in this form are needed for testing. The test program should also include novel materials such as lithium beryllates into the test matrix.

The completed and active experiments are summarized in Tables 2-5 and 2-6. A major evaluation and selection of materials will occur over the next 3-5 years as the results of present and new experiments become available (e.g., the FUBR-1B irradiation tests will not be completed until 1989). This selection of the primary candidate materials and material form will be needed for the Phase II and III experiments.

The experiments can be conducted in many laboratories (material development and property measurement) and fission reactors (irradiated tests), using standard equipment (except for the development of novel fabrication processes and material forms). However, there are a limited number of facilities that can perform precise measurements of certain properties such as thermodynamic data. The present level of activity relevant to this task is about 6-9 M$/yr worldwide, and a similar total level of effort should continue.

Table 2-5. Completed and Active Solid Breeder Material Characterization Irradiations in Closed Capsules

Experiment	Material Characteristics				Irradiation Environment				Time Frame
	Ceramic	Grain size (μm)	Density (%TD)	Li-6 (%)	Temperature (°C)	Li burnup (Max at.%)	Container	Reactor	
ORR	Li_2O	< 47	70	0.05	750,850,1000	0.05	-	ORR	-
TULIP (US)	Li_2O	50	87	93	600	3	35% Ni, PCA	EBR-II	84
FUBR-1A (US)	Li_2O	6	85	45	500,700,900	1.5	Ni/ Ce getter	EBR-II	84/85
	$LiAlO_2$	< 1	85,95	95	500,700,900	3	Ni/Ce	EBR-II	84/85
	Li_4SiO_4	2	85	70	500,700,900	2	Ni/Ce	EBR-II	84/85
	Li_2ZrO_3	2	85	70	500,700,900	2	Ni/Ce	EBR-II	84/85
FUBR-1B (US)	Li_2O	< 5 < 5	60,80 80	56 56	500,700,900 500-700/1000	5	Ni/ Ce getter	EBR-II	85/89
	$LiAlO_2$ (sphere-pac)	< 5-10	80 80	73,95 95	500,700,900 500-700/1000	9	Ni/Ce	EBR-II	85/89
	Li_4SiO_4	< 5	80	94	400-500	9	Ni/Ce	EBR-II	85/89
	Li_8ZrO_6	< 5	80	73	600-700	7	Ni/Ce	EBR-II	85/89
	Li_2ZrO_3	< 5	85	70	520-620	7	Ni/Ce	EBR-II	85/89
ALICE (France)	$LiAlO_2$	0.35-13	71-84	-	400,600	-	-	OSIRIS	85/86
DELICE (Germany)	Li_2SiO_3 (Li_4SiO_4)	-	65,85,95	-	400,600,700	< 0.02	SS	OSIRIS	85/86
EXOTIC (Neth./UK/ Belgium)	Li_2SiO_3	-	80	0.05-7.5	400,600	-	-	HFR	85/86
	Li_2O	-	-	-		-	-	HFR	85/86
	$LiAlO_2$	-	80	0.06-7.5		-	-	HFR	85/86
	Li_2ZrO_3	-	-	-		-	-	HFR	85/86
CREATE (Canada)	$LiAlO_2$	< 1	80,90	7.5	100	-	Quartz, SS, Ni	NRU	85/86

Table 2-6. Completed and Active In-situ Tritium Recovery Experiments

Experiment	Material Characteristics				Irradiation Environment						Time Frame
	Ceramic	Grain size (μm)	Density (%TD)	Li-6 (%)	Temperature (°C)	Li burnup (Max at.%)	Sweep Gas	Flow rate ($m^3/s\text{-}g$)	Container	Reactor	
TRIO (US)	$LiAlO_2$	0.2 (50 μm particles, 0.9 cm thick annular pellet)	65	0.55	400,...,700	0.2	He He + H_2 He + O_2	0.01-0.1	SS	ORR	84/85
VOM-15H (Japan)	Li_2O	< 10	86	7.4	480,...,760	0.24	He He + H_2	0.05	-	JRR-2	84
VOM 22/23 (Japan)	Li_2O	- (1.1 cm pebbles)	-	7.6	400-900	0.04	He He + D_2	-	SS	JRR-2	-
	$LiAlO_2$	- (1.1 cm pebbles)	-	25	400-900	0.1	He He + D_2	-	SS	JRR-2	-
LILA (France)	$LiAlO_2$	1-30 (1 cm diameter pellet)	78	7.5	375-600	< 0.02	He	-	Quartz,SS	SILOE	86
LISA (Germany)	Li_2SiO_3	- (1 cm diameter pellet)	-	-	-	-	-	-	-	SILOE	86
EXOTIC (Neth./UK/ Belgium)	$LiAlO_2$	- (1.4 cm diameter pellet)	80,95	0.6,7.5	400,600	< 0.4	He + H_2	-	SS,Ni	HFR	86
	Li_2SiO_3	- (1.4 cm diameter pellet)	50	0.6,7.5	400,600	< 0.4	He + H_2	-	SS,Ni	HFR	86
CRITIC (Canada)	Li_2O	- (1 cm thick annular pellet)	80	0.3	400-900	-	He He + H_2 He + H_2O He + O_2	-	Inconel-600	NRU	86

2.2.3 Multiplier Material Development and Characterization

Solid breeder blankets require a neutron multiplier, with the possible but uncertain exception of Li_2O. In addition, multipliers are useful for increasing energy production and may be required for other blanket concepts (liquid breeders) and applications (fissile fuel production). In many cases, the multiplier would be a separate, unmixed material. Therefore it is prudent to develop multiplier material options. The objective of this task is to fabricate, characterize, and improve the properties of candidate multiplier materials, including possibly closed or open capsule irradiation tests.

The primary candidate multiplier material is beryllium. The near-term subtasks are to measure the effects of irradiation (swelling, creep, and ductility), and tritium retention and release at reactor-relevant temperatures and fluences, including the effect of material form and porosity. The magnitude and direction of beryllium properties are dependent on the material form. Long-term tasks are to optimize the properties for the particular applications (e.g., beryllium pebbles, self-supporting metal rods), and to develop practical and economic (i.e., low loss rate) fabrication and recycling techniques. The latter are particularly important because of the limited beryllium resources.

Fabrication and property measurements can be performed with standard equipment, although the chemical toxicity of beryllium must be considered. The mechanical behavior under irradiation is dominated by helium production from the (n,2n) reaction (~1.7 MeV threshold). Tritium production in Be occurs by a high neutron energy reaction (which dominates in a fusion reactor), and a lower energy reaction (which dominates in a fission reactor). Calculations indicate that a fission reactor like FFTF can provide reactor-relevant helium and tritium production.

There is presently very little experimental activity with respect to multiplier development. A reasonable program would require about 1-2 M$/yr, consistent with the pace and relative number of materials being investigated in the solid breeder material development task. Because of the importance of multipliers and their properties in blanket design, these experiments should begin shortly in order to provide irradiation-related data in a reasonable time frame (5 years).

2.2.4 Neutronics and Tritium Breeding

The objective of this task is to measure the tritium production rate and heating rate distributions in order to verify and improve nuclear data, design methods and models. Two stages of testing can be identified: simple geometry mockups and engineering mockups.

Simple Geometry Mockups

Simple mockups would be conducted with geometrically simplified blanket modules that incorporate the primary breeder and blanket materials. A 14 MeV neutron source with sufficient strength (about 10^{12} n/cm^2-s) is the primary requirement. The most suitable existing 14 MeV neutron source with sufficient test volume, strength, availability, and source characterization is the Fusion Neutron Source (FNS) at JAERI. Other facilities exist in Japan (OCTAVIAN), Europe (LOTUS), and the U.S. (RTNS-II).

The blanket mockups can be expensive because of the material costs, particularly Li_2O, Be and ^6Li (although enriched lithium may not be necessary for these tests since the relevant ^6Li cross-sections are reasonably well-known). For two to three distinct breeder materials plus instrumentation, the capital cost is about 3-6 M$. Each experiment series would cost about 0.5 M$ for setup, operation, and post-test analysis, and up to two experiments a year could be performed with sufficient time for analysis. Such experiments are beginning in 1986 as part of the US/JAERI Fusion Breeder Neutronics Collaborative Program, initially with Li_2O. About 5 years would be needed to explore the major materials and combinations of present international interest.

Engineering Mockup

Some of the important uncertainties in the assessment of the tritium breeding and other neutronics parameters are associated with the effects of the geometrical details of the blanket and the surrounding reactor. Engineering mockup experiments are necessary to address these uncertainties.

These tests include partial coverage of the neutron source with a mockup of the reactor sector, plus a detailed blanket module design for measurement of the tritium and heat production profiles. As with the simpler blanket material mockup experiments, a 14 MeV neutron source is needed with comparable source characterization and intensity. Again, the Fusion Neutronics Source at JAERI is the most suitable neutron source, but the other facilities in Europe, Japan, and the U.S. can perform supporting subtasks.

The blanket mockups would be more expensive because of the required design detail in addition to the material cost; so the mockup of the blanket and surrounding sector brings the capital cost up to 4-7 M$. Each experimental series would be about 0.5-1 M$ for setup, operation and post-test analysis, with one or two series per year. These experiments would not begin until module and reactor concepts are reasonably well defined.

2.2.5 Advanced In-situ Tritium Recovery

The objective of these tests is to study in-situ tritium recovery with local reactor-relevant conditions, specifically moderate-to-high burnup, temperature gradient, purge flow, and breeder/cladding mechanical and chemical interactions. Relevant parameter ranges are given in Table 2-7 and an example multiple-capsule test assembly for FFTF is illustrated in Fig. 2-4.

Table 2-7. Parameters for Major Integrated Non-fusion Irradiation Experiments

	Advanced In-situ Tritium Recovery	Nuclear Submodule
Test geometry	Subassembly with multiple capsules	Blanket breeder section or unit cell
Material	Multiple	One per submodule
Temperature, °C	350-1200°C	Reactor blanket profile
Temperature gradients, °C/cm	100-1000	100-1000
Breeder thickness, cm	0.5-5	0.5-5
Purge gas	Helium, plus O_2, H_2 and/or H_2O	Helium, plus O_2, H_2 and/or H_2O
Purge flow rate, $m^3/s\text{-}g^a$	0.01-0.1	0.01-0.1
Burnup, at.% Li	3-10	3-10
Heat generation, MW/m^3	30-100	30-100
Irradiation time, yrs	1-3	1-3
Tritium production, T/Li-yr	0.01-0.5	0.01-0.5

[a]Normalized per gram of solid breeder material

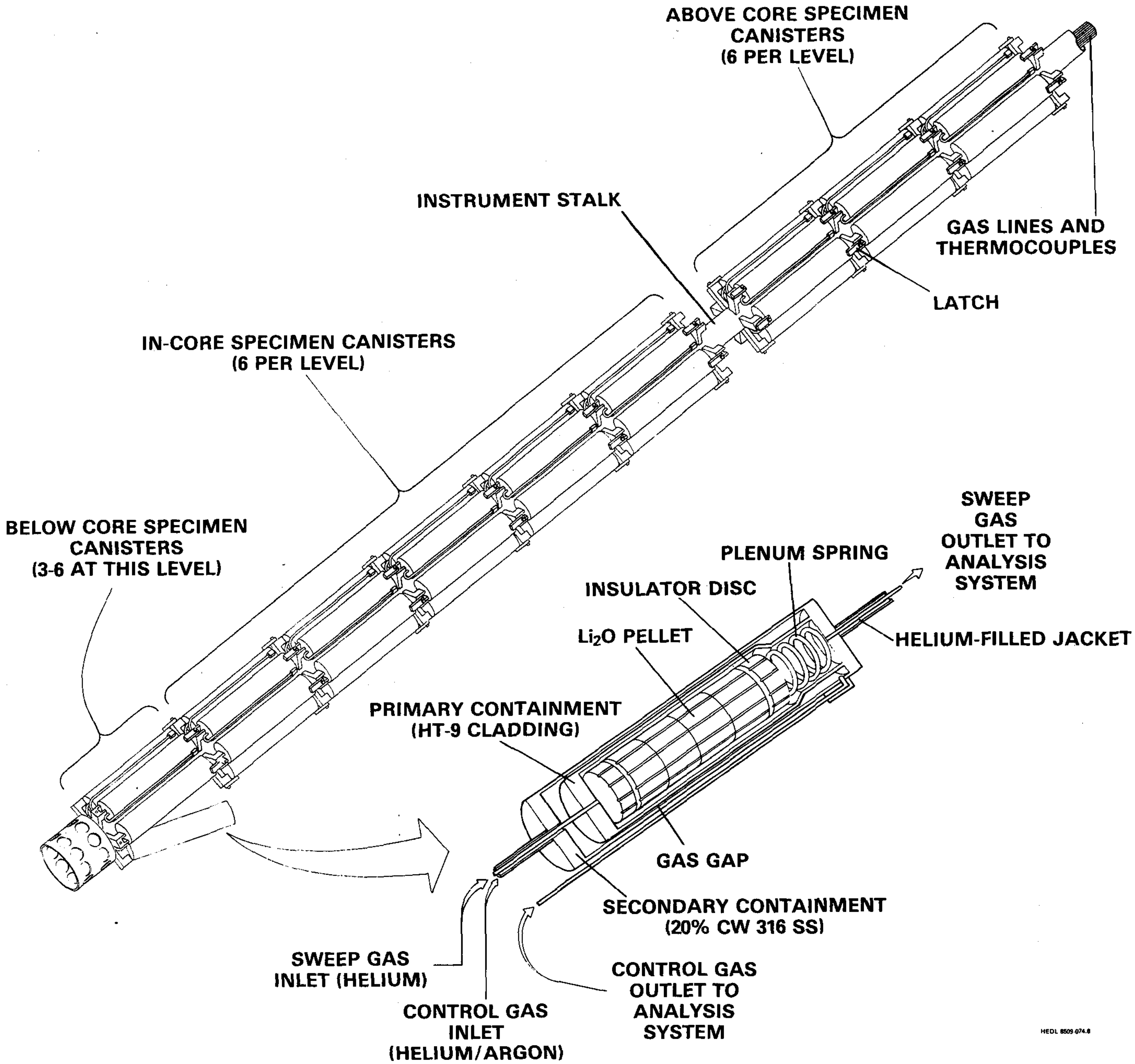


Figure 2-4. Representative advanced in-situ tritium recovery test assembly showing expanded view of a purged capsule

An important part of this task would be to investigate the design limits (e.g., upper temperature limits) and transient behavior.

This task could be performed as one or more instrumented and purged assemblies in fission reactors, depending on the available test volume and the number of materials and conditions to be tested. The facilities needed are reasonably high-flux and large-test-volume fission reactors with the ability to handle purged and instrumented assemblies. Fast neutron spectra are preferred in order to allow high ^{6}Li content without self-shielding, which is necessary for achieving reactor-relevant tritium production and heating rates, and for achieving 5-10% burnups within reasonable time periods. The most useful reactors appear to be FFTF and Phoenix, although the availability of Phoenix for such experiments is uncertain.

A subassembly containing several capsules (as in an FFTF-sized assembly) has a total cost of about 7 M$, with 3-5 M$ for design and fabrication, 0.8 M$ operating costs for a 3 year irradiation (not including neutron charges) and 1 M$ for post-test examination. The duration of the experiment is about 5-6 years, with 1-2 years design/fabrication, 3 years irradiation (e.g., 3-9% lithium atom burnup), and 1 year post-test examination.

2.2.6 Blanket Thermal Behavior

The objective of this task is to investigate thermomechanical behavior, heat transfer and material interactions. Two stages in testing can be identified: local breeder thermal behavior and blanket submodule or module thermomechanical integrity.

Breeder Thermal Behavior

This task would focus on local behavior (e.g., within the breeder pellet). Several experiments would be performed to measure breeder internal temperature profiles, gap conductance, breeder/cladding mechanical interaction, corrosion and mass transfer. Some tests could be performed with non-neutron experiments to provide scoping data and support model development, particularly for corrosion and mass transfer. However, irradiation effects are important and most experiments would be performed (eventually) in irradiated closed capsules.

The irradiated tests would be different from the tritium recovery capsule tests. For example, there is a larger emphasis on internal temperature gradients, on breeder/cladding mechanical interactions, and on monitoring the breeder internal conditions. The latter could require internal thermocouples or isotopically modified materials that might affect tritium recovery.

Many laboratories and fission reactors could provide suitable testing. The total cost is estimated as about 3-8 M$ over the next 3-5 years for several experiments, including design, fabrication, testing and data analysis, but not model development.

Non-neutron Thermomechanical Integrity

In the longer term, a non-neutron thermomechanical test facility could be built to provide more complete testing of geometrical and transient-related effects (possible in a full blanket module), although without irradiation effects. The need for this facility and the complexity of the tests will depend on the degree of design detail available, the extent of the planned nuclear tests (for example, whether they include transient effects), and the importance of the issues that would be addressed.

Such a facility could use RF heating, resistive wires, or a hot purge to simulate bulk heating (depending on the desired accuracy and complexity), and particle beams or radiant arcs for surface heating. This facility would be built later in the test program when more detailed designs would be available, and in support of the nuclear submodule experiments which are more limited in size and transient testing abilities. Some non-neutron testing of prototype nuclear submodule assemblies may even be necessary for final design and approval of the latter reactor tests.

A facility large enough to provide full power heating to a partial or complete blanket module would cost about 3-7 M$ to design and build. The test modules themselves would cost up to 1 M$ for a full blanket ($\sim 1\ m^3$).

2.2.7 Nuclear Submodule Experiments

The objective of this task is to provide non-fusion neutron testing of a blanket module, including more complex geometry effects. This is the maximum level of testing that can be achieved in a non-fusion test facility.

Fission reactor limitations constrain the size of the test piece to sections of a full blanket module, and make other environmental conditions such as magnetic fields and surface heating difficult to include. As with the Advanced In-situ Tritium Recovery tests, the number of useful test facilities is limited. Parameters for this test are also given in Table 2-7, and are similar to those of the Advanced In-situ tests since it is important to provide reactor-relevant conditions for both series of tests.

The cost for design and fabrication of each assembly is about 5-7 M$, plus an annual operating cost of 1-2 M$ for a 3-year irradiation test and 1 M$ for post-test examination. The duration of the experiment would be about 7 years, including design, fabrication, testing and post-test examination.

2.2.8 Other Solid Breeder Related Tasks

Structural Response

The most important element of determining the structure mechanical behavior in the fusion environment is the development and characterization of the structural alloys under irradiation. This is an important and active materials task that is not specific to solid breeder blankets. Obtaining high fluence data on candidate alloys is a long term task due to the lack of a high-flux, large-volume fusion neutron source.

Information on electromagnetic effects on the structure and on first wall thermomechanical behavior will be available from the parallel efforts to develop plasma interactive components for existing and planned fusion devices. Additional information on submodule and module mechanical behavior will be provided by the non-neutron thermomechanical integrity and nuclear submodule experiments. No major additional solid breeder specific tests are anticipated prior to fusion integrated testing.

Tritium Permeation and Processing

The control of tritium beyond the solid breeder and multiplier raises issues regarding tritium permeation and processing (from the blanket purge stream and coolant). Many of the uncertainties are not specific to solid breeder blankets and will be addressed as part of the development of tritium system technology. For example, permeation is important in most blankets, and extraction of tritium from a helium stream is also applicable to the fuel processing system and some stage of most liquid breeder blanket tritium extraction systems. However, some tests with specific solid breeder blanket conditions will be required. Some permeation data will be available from the various nuclear experiments (and possibly the non-nuclear thermomechanical integrity test).

2.2.9 Modeling Needs

The experiments should be supported by a strong program of model development in order to understand the test results and to improve predictions of the blanket behavior, which will in turn reduce the number of required future experiments. Since the major features (and design uncertainties) in solid breeder blankets are not expressed in terms of classical equations, modeling of solid breeder blankets generally emphasizes a mechanistic or semi-empirical approach. As indicated in Table 2-8, models for all the important phenomena must be developed or improved (by the inclusion of additional important variables, for example). These models must then be combined to form design-oriented codes.

2.2.10 Test Sequence and Logic

The experiments associated with these tasks can be formed into a test plan that indicates both a logical sequencing or relationship between the experiments, and an estimated program cost. Although both the test sequence and costs are more meaningful in the near-term, it is important to include an estimate of the long-term strategy and costs since program resources are constrained by time and funding.

Figure 2-5 illustrates a solid breeder test sequence that structures the experiments according to the test program phases described in Table 2-1, with initial emphasis on understanding material behavior and blanket phenomena, and a 15-year objective of concept verification in a non-fusion environment.

In this test plan, the development and characterization of solid breeder materials must continue since this data allows material selection which is most cost-effective at this point in the program. The assessment of tritium self-sufficiency should also continue through the next phase of the U.S./JAERI Fusion Breeder Neutronics Collaborative Program. Both of these tasks involve series of experiments in existing facilities.

Table 2-8. Solid Breeder Blanket Modeling Needs for the Major Tasks

Material development and characterization
Thermal conductivity as a function of microstructure
Swelling and creep of breeder and multiplier
Tritium diffusion with impurities and radiation damage
Breeder thermal behavior
Creep and swelling mechanical interaction between breeder and clad and effects on gap conductance and breeder/clad deformation
Breeder internal and external mass transfer and effects on microstructure
Advanced in-situ tritium recovery
Local oxygen activity and effects on tritium recovery
Time-dependent blanket tritium inventory and recovery with temperature profile, purge chemistry and irradiation effects
Nuclear submodule experiments
Inelastic fracture mechanics, plastic crack growth
Simple models for high fluence/high temperature failure-related phenomena (e.g., creep buckling, creep/swelling)
Tritium permeation into coolant
Neutronics and tritium breeding
Improvements in code capabilities for complex geometries (e.g., faster 3-D algorithms or accurate homogenization methods)

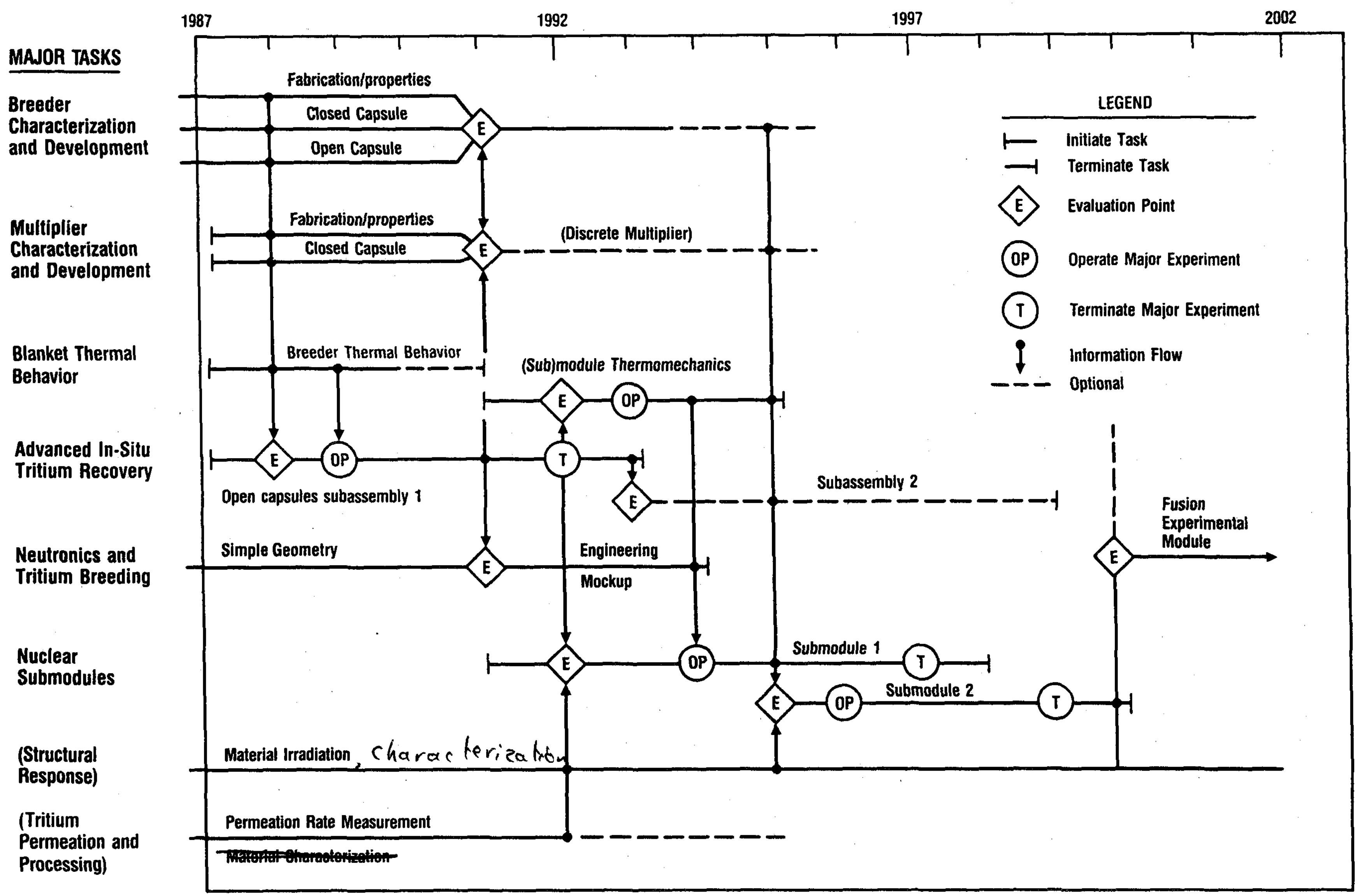


Figure 2-5. Test sequence for major solid breeder blanket tasks

In the near future (~ 1987), additional tasks must be started with respect to multiplier development and basic breeder thermal behavior in order to support material selections in about 5 years. The design of an advanced in-situ tritium recovery experiment would also begin, with initial focus on identifying the test facility and irradiation vehicle. The selection of the particular materials and test matrix could be made somewhat later, incorporating the latest results from the other tasks. This experiment could be placed in-reactor in 1989, with interim discharges (allowing examination and possibly replacement by alternate materials) at one-year intervals. Around 5 years into the program, a major evaluation of the materials should be made in order to select the most promising materials and assess the need for further development and characterization. Development of commercial-relevant fabrication and reprocessing methods might then begin. If the multiplier is to be incorporated into the breeder, then subsequent experiments should test the combined material.

Towards the end of this period (5-7 years into the program), more detailed solid breeder blanket concepts will be available and it will be appropriate to address design considerations for the next generation of design-relevant tests. These include neutronics tests of detailed engineering mockups with simulated partial coverage of the neutron source by a reactor sector, nuclear submodule tests, and non-neutron thermomechanical integrity tests of reasonably large sections (such as the nuclear submodule or larger). The latter would include transients, and would also serve to help design and license the nuclear submodule tests. A second or third nuclear submodule test would be useful to allow testing of alternate ideas and to provide backup against possible failure of the first nuclear submodule test.

At this point, information from the alloy development program on preferred structural alloys and their behavior, from the plasma interactive component program on electromagnetic effects and high heat flux thermomechanics, and from the tritium system program on tritium permeation and processing, would also be available. At any point beyond the first testing phase that assesses material behavior, a selection between liquid and solid breeder blankets could be made with some confidence. However, assuming a reasonably successful test program, an accurate assessment of the attractiveness of solid breeder blankets in general would not be available until the completion of the nuclear submodule experiments.

Table 2-9 summarizes the expected costs and duration of the major tasks. The costs are broken down into two categories: capital and operating costs. Capital costs include design effort, materials, fabrication, construction, and any expense directly related to the construction of the facility and/or the experimental apparatus. Operating costs are average annual costs for materials and energy, staff to operate the experiments, and data acquisition. The costs of modeling efforts and blanket design studies are listed as a separate task. The duration of each task includes design, fabrication, test operation and post-test examination. Some of these tasks incorporate several phased experiments, and the duration reflects the net length of the test series. The total cost is the sum of the capital costs plus the operating costs over the testing phase of the experiment.

Overall, this test plan should lead to a reasonable solid breeder blanket data base for a variety of materials and allow timely opportunity for innovation and design changes. It includes testing under all reasonable non-fusion conditions with substantial reactor-relevant conditions in many tests. The estimated cost of this solid breeder blanket test plan is about 10-20 M$/yr. These tests would be expected to support model development and allow calibration of design codes. The only remaining surprises, if any, from testing in a fusion device would be those effects that could not be reasonably anticipated from non-fusion tests. Thus it would be possible to assess the feasibility and attractiveness of solid breeder blankets around the year 2000.

Table 2-9. Representative Costs of Major Solid Breeder Tasks Over the Next Fifteen Years[a]

Task	Capital cost (M$)	Operating cost (M$/yr)	Duration (years)	Total cost (M$)
Solid Breeder Characterization and Development	5–7	5–8	5 (initial)	30–50
(Fabrication, properties, closed and open capsule irradiations)				
Multiplier Characterization and Development	1–2	1–2	5	6–12
(Fabrication, properties, closed capsule irradiations)				
Blanket Thermal Behavior				
A. Breeder thermal behavior	0.8–1.5	0.8–1.5	3–5	3–8
B. Non-neutron (sub)module thermomechanics	3–8	0.8	4	5–10
Neutronics and Tritium Breeding				
A. Simple geometry	3–6	0.8–1.5	5	7–14
B. Engineering mockup	4–7	0.8–1.5	3	6–12
Advanced In-Situ Recovery	3–5 each	0.8 each	6 each	12–16
(Two sequential subassemblies with multiple purged capsules)				
Nuclear submodules	5–7 each	1–1.5 each	7 each	20–30
(Two parallel submodules)				
Analysis and Model Development	0	2–3	15	30–45

[a] 1985 constant dollars; Neutron facility and neutron costs not included.

2.3 Liquid Breeder Blanket Test Plan

2.3.1 Introduction

The major classes of issues for liquid breeder blankets have been extensively studied in FINESSE and are listed in Table 2-10. A test plan has been developed to resolve these issues and develop predictive capabilities which address the technical uncertainties related to the issues. Figure 2-6 shows the types of major facilities and experiments in each issue category as a function of the degree of integration of the test. The elements in Fig. 2-6 can be viewed as "building blocks" for the test plan.

The four major phases of testing were identified in Table 2-1. The nature of the information sought in each phase gradually shifts from fundamental, scientific data to empirical, design-related data which will ultimately be required to support testing in a fusion environment. This structure is utilized because of the current absence of standard material and design choices. Design-independent data and material screening will be useful to identify the most attractive options, after which a more aggressive "product-oriented" program can be initiated. This phasing is important due to the general lack of data to demonstrate a clearly superior blanket design, and because some of the required types of experiments vary widely between different blanket options.

The candidate liquid breeder blanket material options are shown in Table 2-11. To meet the Phase I primary objective of material combination screening, the testing program should provide sufficient breadth to cover the

Table 2-10. Generic Liquid Breeder Blanket Issues

- Tritium Self-sufficiency
- Magnetohydrodynamic (MHD) Effects
  - Fluid Flow (including pressure drop)
  - Heat Transfer
- Material Interactions (e.g., Corrosion)
- Structural Response in the Fusion Environment
  - Irradiation Effects on Material Properties
  - Response to Complex Loading Conditions
  - Failure Modes
- Tritium Recovery and Control

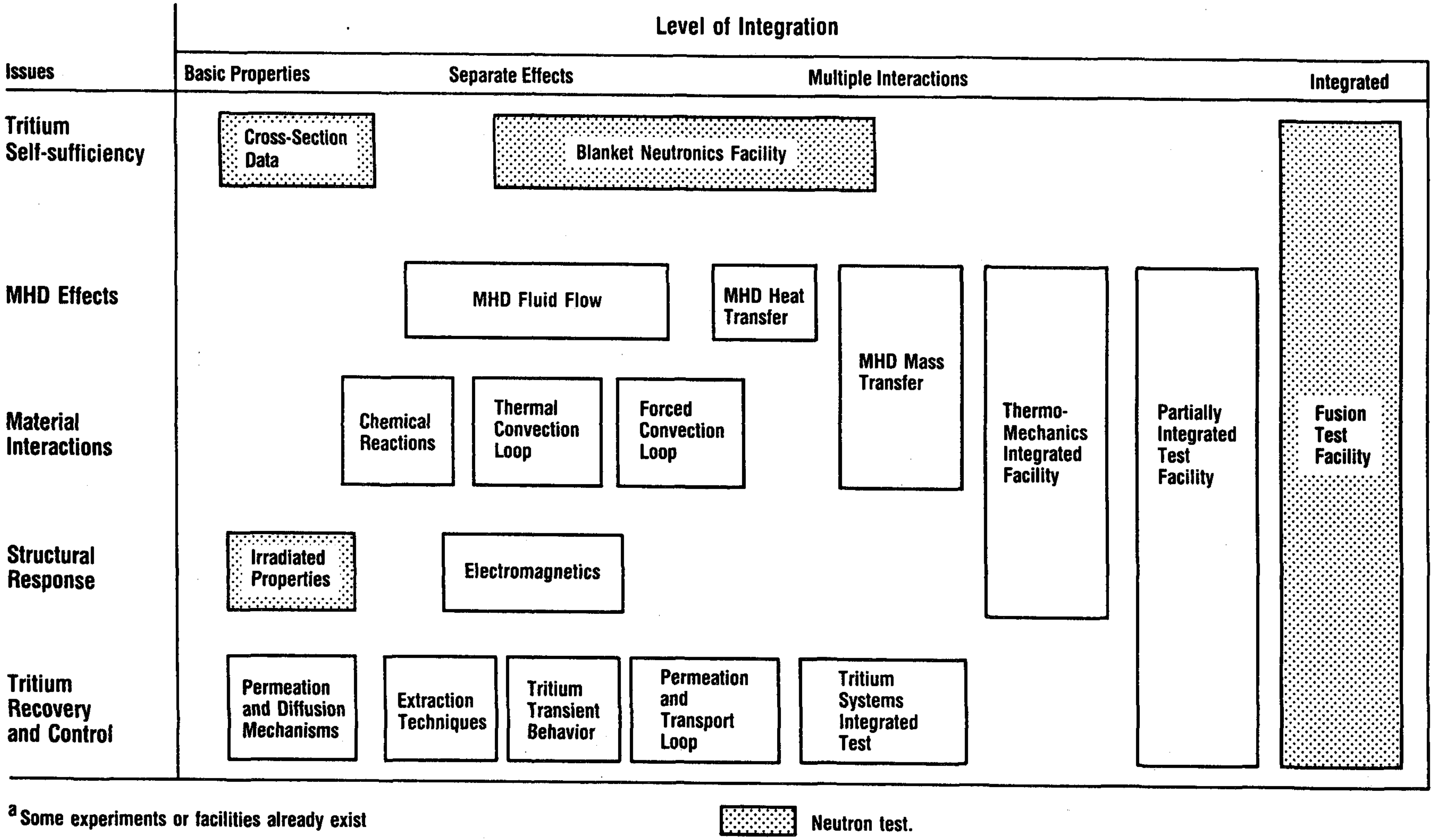


Figure 2-6. Types of experiments and facilities for liquid breeder blankets[a]

Table 2-11. Potential Material Combinations for Liquid Breeder Blankets

Breeders	Structures	Neutron Multipliers	Coolants
$\boxed{\text{Li}}$	SS	$\boxed{\text{Be}}$	$\boxed{\text{self}}$
$\boxed{\text{17Li-83Pb}}$	$\boxed{\text{HT-9}}$	$\boxed{\text{none}}$	$\boxed{\text{He}}$
$\boxed{\text{Flibe}}$	$\boxed{\text{V-alloy}}$		organic
			water
			Flibe

$\boxed{}$ most promising ones

critical feasibility issues for each major class of design. To maximize the breadth of the program in Phase I while limiting the expense, only the largest uncertainties for each blanket concept should be addressed. Table 2-12 shows the effect of coolant, breeder, and structural material choices on the dominant near-term issues for liquid breeder blankets. The primary objective of Phase I experiments is a narrowing of material combination choices and ranking of blanket concepts which will be used to allocate resources for further, more detailed investigations.

Although the emphasis in Phase I is the development of understanding and predictive capabilities, some concept screening tests may also be used to meet the milestone of material selection. Screening test data is characterized by attention to general behavior under relevant conditions rather than mechanisms under idealized conditions. In general, screening tests involve more interactive effects and are less well understood. Nevertheless, empirical data can have several benefits, including early detection of unanticipated problems and determination of global blanket performance parameters. An example of this kind of test would be a MHD mass transfer scoping test which compares mass transfer with and without a magnetic field in a simple geometry. An MHD pressure drop scoping test is another example.

The general features and requirements for experiments in the second and third phases of testing have been defined. However, the specific details have

Table 2-12. Effect of Coolant, Breeder and Structural Material Choices on Dominant Issues for Liquid Breeder Blankets

Liquid Metal Cooling	
Li or 17Li-83Pb	MHD effects (including viability of insulators)
	corrosion (including viability of inhibitors)
Coolant or Breeder	
Lithium	chemical reactivity
17Li-83Pb	tritium containment
Flibe	tritium containment
Helium	tritium containment
Structural Material	
Vanadium alloys	bimetallic mass transfer
	DBTT[a] (due to impurities, radiation, H, He)
Ferritic alloys	DBTT

[1]Ductile-to-Brittle Transition Temperature

not been evaluated as thoroughly as those of Phase I. Such details will be affected considerably by the results of Phase I experiments. The primary goal of Phase II is selection of blanket configuration; the goal of Phase III is initial design concept verification in non-fusion facilities. The primary facilities for Phases II and III are the Thermomechanical Integrated Facility (TMIF) and Partially Integrated Test Facility (PITF).

The generic issues in Table 2-10 have been used to define several tasks in the program. These are indicated in Table 2-13. Each of these tasks is described below, including the test logic and facilities.

Table 2-13. Major Tasks for Liquid Breeder Blankets

• MHD Effects Momentum and Heat Transfer Facilities Instrumentation Development Insulator Development
• Material Compatibility Thermal Convection Loops Forced Convection Loops MHD Mass Transfer Facility (MHDM)
• Tritium Extraction and Control Tritium Extraction Tests Tritium Transport Loop
• Tritium Breeding
• Structural Response and Failure Modes
• Thermomechanics Integration Facility
• Analysis and Model Development

2.3.2 MHD Effects

Self-cooled liquid metal blankets represent a large and potentially attractive class of blanket designs. Through the effects on fluid flow, many aspects of blanket behavior are impacted by magnetohydrodynamics (MHD), including pressure drop, heat transfer, mass transfer, and structural behavior.

The current state of knowledge of MHD effects is limited to very simple geometries and very low Hartmann and interaction numbers (a factor of 100-1000 away from the actual fusion environment). Therefore, the program to resolve MHD issues and develop a greater understanding should continue to progress through levels of complexity from simple duct flow in uniform magnetic fields to flow in structure and magnetic field geometries which are representative of actual blanket designs. As the program proceeds, additional interactions should be explored such as the influence of MHD velocity profiles on heat transfer and the combined influence of velocity and temperature profiles on mass transfer.

A cornerstone of the predictability of MHD effects is an understanding of the velocity profiles. If the electric current distributions and velocity profiles can be predicted theoretically, then many of the other important "macroscopic" parameters can be derived, such as pressure drop and heat transfer coefficients. Therefore, great importance should be attached to the development of velocity profile instrumentation, and measurement of velocity profiles in a variety of relevant geometrical configurations.

A prudent test plan, however, must account for the possibility that it may eventually prove impractical to directly measure velocity profiles with the required accuracy in complex configurations. More global information can be sought on integral quantities, such as pressure drop and heat transfer coefficients. These measurements would serve as a check on the validity of the velocity profile models and measurements, and also provide a back-up source of data if the velocity measurements turn out to be inadequate. This might happen, for example, if reliable velocity measurements cannot be extended to high fields, complex geometries, high temperatures, or lithium operation.

Models can also be developed in either a purely theoretical or a more empirical, or correlational, form. While empirical data tend to be very geometry dependent, a number of measurements on several design variations and a program of empirically based design improvement may be the most practical approach to resolving MHD issues.

Considering these factors, the MHD base program has been defined to include the following activities:

- exploration of local MHD behavior
- scoping tests for global behavior
- instrumentation development
- insulator development
  fabrication and mechanical integrity
  irradiation measurements
  compatibility scoping (for coatings and surface layers)

In addition, a program of theory development and design studies is recommended and is described later in this section.

Advanced Liquid Metal Flow Facility (LMF1)

The primary mission of the advanced MHD Liquid Metal Facility, LMF1, is to develop a better fundamental understanding of fluid flow behavior in blanket components. Although velocity profiles are the primary source of data, surface heat flux is provided to allow for the complementary measurement of temperature profiles and heat transfer coefficients. Temperature and pressure measurements will provide an integral check on the validity of the velocity profile measurements and also serve as a back-up source of data if the velocity profile measurements are unclear. The experiments will involve elements of complex geometries, e.g., expanding or contracting ducts, orifices, or bends with different alignments relative to the magnetic field.

Beyond simple velocity measurements for validation of MHD theory, the facility has a secondary mission to measure temperature profiles, explore heat transfer characteristics, and develop methods to improve heat transfer and fluid flow, such as geometric modifications, use of insulators, and flow tailoring. This secondary mission will likely follow a period of 2-4 years of basic velocity profile measurements.

The advanced MHD facility is designed for flexibility and to provide high capabilities for magnetic field strength and field volume. This allows the experiments to treat a wide variety of geometric configurations and reactor relevant conditions. Analysis of fluid flow behavior indicates that certain ranges of the Hartmann (Ha) and Reynolds (Re) numbers must be provided in order to maintain similar fluid flow behavior. As an example, Figure 2-7 indicates the acceptable operating region for the Hartmann and Reynolds numbers which provide fluid flow behavior similar to that expected to occur in the poloidal manifolds of the reference BCSS design. The criteria imposed include (1) suppression of turbulence, (2) size of inertial and shear layers less than 1/10 of the channel half-width, and (3) dominance of inertial forces over viscous forces. Additional criteria may be required, for example on the wall conductivity ratio, C. Other blanket designs may require different parameters for similarity.

From the figure, it can be concluded that the Hartmann number should be maintained higher than 10^3 or 10^4. This can be achieved in small channels (1-5 cm) if the magnetic field strength is above 1-2 Tesla. Other concerns also suggest a high magnetic field strength for relevant MHD behavior. For example, if ferritic materials are utilized, they must be fully saturated. In addition, experimental results indicate that it might be possible to generate

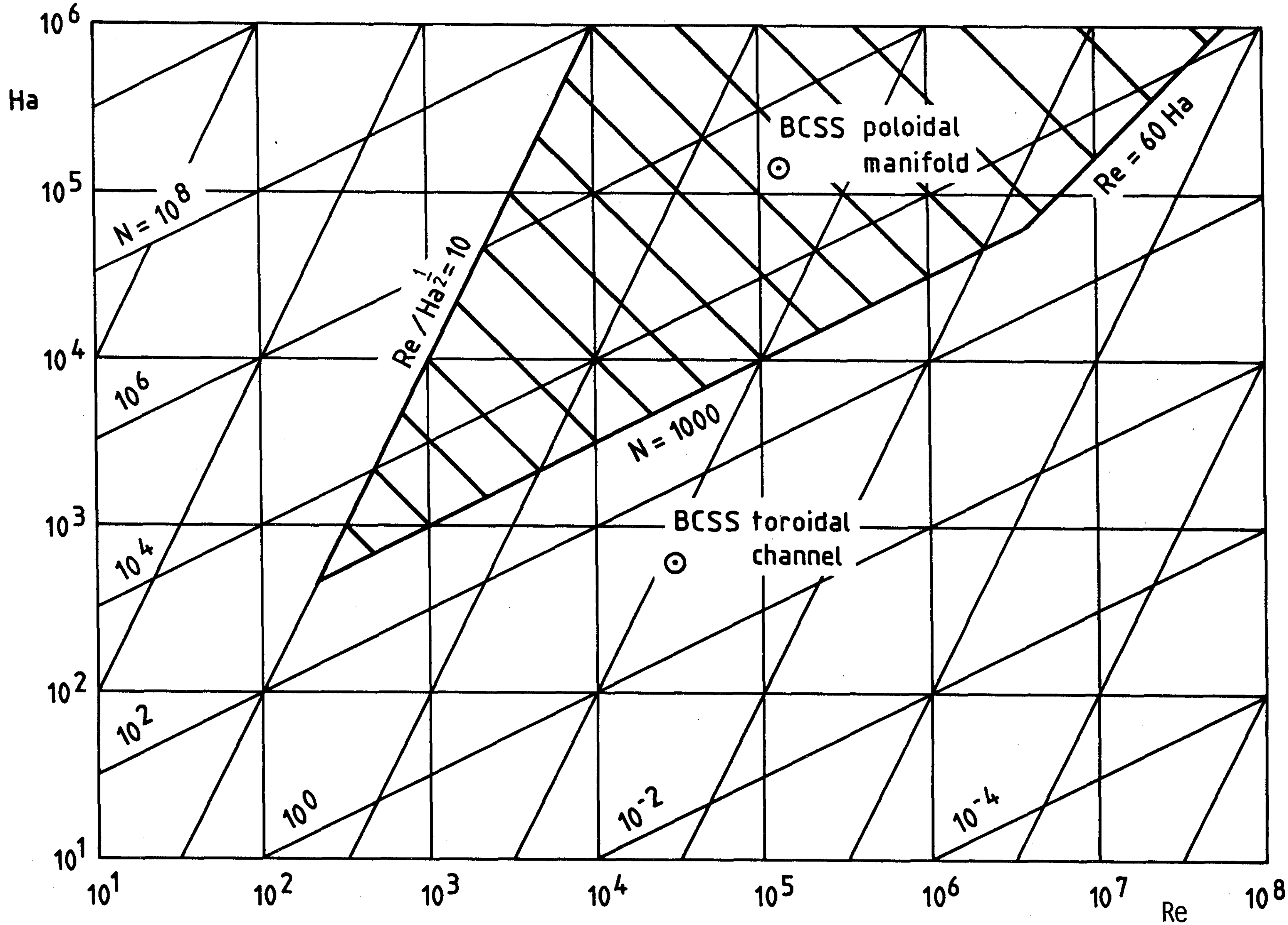


Figure 2-7. Required parameter ranges of the Hartmann number (Ha), Reynolds number (Re), and interaction parameter (N) for MHD fluid flow experiments.

and sustain some flow fluctuations at very high magnetic field. The possible impact on heat transfer may be critical and needs to be verified.

Another desired feature of the advanced MHD facility is expandability. Beyond Phase I, and assuming that self-cooled blankets continue to be strong candidates, MHD heat and mass transfer experiments will be required. By providing the capability for large volume, high temperature operation in the initial facility, the same facility can be used for the follow-up experiments with significant savings on the integrated cost.

The proposed primary facility parameter ranges are shown in Table 2-14. The facility would provide a large volume of moderately high magnetic field, power supplies for the magnets and for surface heating, heat rejection systems, instrumentation, and work space. It is envisioned that several different flow loops will be inserted into the test volume, making this a "user facility".

Integral Parameter Experiment (LMF2)

The above facility (LMF1) will focus on developing predictive capability by focusing on measurement and understanding of "microscopic" parameters such as velocity profiles. It is suggested that a series of experiments would also be performed with a greater focus on "macroscopic" parameters such as the pressure drop. Although in principle these experiments can be conducted in LMF1, practical considerations suggest that another facility, called LMF2, would be devoted to this purpose.

The "macroscopic" experiments will be of the scoping type to assess the seriousness of the MHD problem early in the program. Basic geometric elements representing prototypical blanket modules could be constructed and gross measurements made of pressure drop and temperature response. The extent to which our understanding would be furthered by this kind of test and the amount of data to support model development are less than the MHD "microscopic" parameter experiments in LMF1; however, integral benchmark data could be used to indicate the most serious problems and to provide in an empirical data base for design improvement.

The cost and time to perform such testing might be small enough to make it very attractive. Although the device parameters should be very similar to those of LMF1, much of the LMF1 instrumentation (which will be required to develop "microscopic" data such as flow profiles) can be avoided and the device need not provide as much flexibility as the LMF1. Additionally, the operating costs are expected to be reduced because of the smaller amount of associated analysis.

Table 2-14. Preliminary Parameter Ranges for the Advanced MHD Facilities (LMF1 and LMF2)

magnetic field strength	4-6 T
field volume	3 m x 1 m x 0.5 m
coolant velocity (in the test section)	0.05-0.1 m/s
pipe diameter in loop	10 cm
coolant temperature rise	100-200°C
total heat input	10 MW
approximate total cost	10-15 M$

The operation of two facilities, LMF1 and LMF2, early in the program provides considerable benefits in terms of obtaining information on a timely basis and the ability to carry out the many required experiments. However, the cost of two facilities may be too high for the fusion program in any one country. This is a good example of an area where international cooperation can be very effective. Other alternatives to dividing the mission between the two facilities might also be considered.

Instrumentation Development

The measurement of MHD velocity profiles has been performed in a number of experiments in NaK, mercury, and sodium. However, velocity profile measurements are expected to become more difficult at higher magnetic field, higher temperatures, and in corrosive liquids such as lithium. At high magnetic field, the flow becomes so strongly laminarized that the heat transfer becomes nearly independent of velocity. This makes standard instruments such as hot film probes ineffective. At high temperature and in corrosive liquids, fouling and desensitization of probes may become a serious problem. In addition to these environmental effects, the important characteristics of the velocity profiles themselves may be unmeasurable at very high field. It is anticipated that as the field increases, the thickness of boundary layers becomes smaller and a large part of the flow may be contained in extremely narrow layers. The spatial resolution of any available technique may not be adequate to discern the important characteristics of the flow.

The degree of success in instrumentation development will determine the type and accuracy of information obtainable from liquid metal experiments. Hence, early instrumentation experiments will have a profound impact on the type of facilities and experiments to be performed later in the program. Some work is already ongoing to improve existing measurement techniques and develop new ones. Because of the importance of measuring velocity profiles, a continued and stronger program of instrumentation development should be implemented. Facility requirements for instrumentation are smaller than for MHD measurements. Small, bench-top loops or existing MHD facilities could be used.

Insulator Development

MHD pressure drop can be significantly reduced through the use of electrically insulating coatings or laminates. Because of the high potential payoff, efforts should determine whether or not insulated structures can meet the requirements on compatibility and structural integrity under irradiation. Three kinds of scoping tests are recommended: (1) fabrication of the various proposed insulated structures (coatings and laminates) and simple mechanical testing, (2) mechanical testing after high fluence irradiation, and (3) compatibility tests in lithium and LiPb.

2.3.3 Material Compatibility

Material compatibility is a dominant concern for all self-cooled liquid metal blanket designs as well as helium-cooled designs with refractory alloy structure. Experiments are ongoing, as indicated in Table 2-15, but the current level of understanding of corrosion and mass transport is still very limited. Temperature limits and the impact of impurity levels and system-related effects (i.e., loop effects) are highly uncertain. Methods to control corrosion and mass transport have been proposed (for example, inhibitors or impurity control systems), but not sufficiently studied to indicate the likelihood of success or limits of applicability.

The material compatibility test plan is designed to provide data on temperature and impurity limits, to develop a better understanding of the important mechanisms, and to explore approaches to reduce mass transfer rates and structural property degradation.

Table 2-15. Summary of Existing U.S. Activities in Liquid Breeder Blanket Research

LIQUID METAL MHD		
Location	Field Strength	Volume
ANL (ALEX)	2.0 T	1.83 m x 0.76 m x 0.15 m
LIQUID METAL CORROSION		
Location	Materials	Loop Type
ANL[a]	Li/304SS	Forced convection
ANL	17Li-83Pb/316SS, PCA, HT-9, 9Cr-1Mo	Forced convection
ORNL[b]	316SS, HT-9, Alloy 800	Thermal convection
ETEC[c] (BLIP)	Li/2-1/4Cr-1Mo	Forced convection
HEDL[d] (ELS)	Li/SS	Forced convection
UW[e]	Li/316SS	Forced convection
ELECTROMAGNETICS		
Location	Field Strength	Volume
ANL (FELIX)	1.0 T steady[f], 0.5 T pulsed	0.9 m diameter, 1.2 m long
BREEDER NEUTRONICS		
Location	Source Strength	
JAERI (FNS)[g]	2×10^{12} n/s	

[a]Argonne National Laboratory

[b]Oak Ridge National Laboratory

[c]Energy Technology Engineering Center

[d]Hanford Engineering Development Laboratory

[e]University of Wisconsin

[f]Capable of 4.0 T steady state and 1.0 T pulsed

[g]Cooperative U.S./Japan program

Early experiments should focus on the largest uncertainties for the greatest possible number of material combinations, consistent with the objectives of Phase I. Table 2-16 shows the relevant environmental conditions for material compatibility as compared with heat and momentum transport. Table 2-17 compares the importance of these environmental conditions for different aspects of material compatibility, including local attack, dissolution and deposition. Temperature, temperature difference, material constituents, and impurity levels have the largest effects and should be emphasized in near term testing. When the basic material interactions are better understood and the velocity and temperature profiles have been determined from MHD testing, magnetic field effects on corrosion can be tested. This will be important only if mass transfer rates are determined to be dominated by liquid phase diffusion rather than solid phase diffusion and chemical reactions.

Following a series of inexpensive static capsule compatibility tests, material compatibility experiments should include at least one dynamic loop for each proposed primary coolant/structural alloy combination. These include lithium and 17Li-83Pb, ferritic steel, vanadium based refractory alloy, and possibly austenitic steel (although not currently a favored class of alloys). The tests with a refractory structure should be performed in a bimetallic loop, since reactor primary cooling systems will almost certainly not be fabricated out of vanadium alloy due to economic considerations. Ideally, each material combination would be tested for long periods of time at different temperature and impurity levels. For an accurate assessment, the experiments must start with a clean loop. In addition, every potential structural material actually represents a class of alloys (for example, ferritic steels include HT-9, 2-1/4Cr-1Mo, etc.). Since different alloys in the same class can exhibit very different material compatibility characteristics, it is desirable to test more than one specific alloy in an alloy class. Clearly, a very large number of loops is desirable; the actual number will depend on practical limits of funding and balance with other tasks in the program.

Other compatibility issues may be important depending on the design of the blanket and tritium extraction systems. If beryllium is contained in the blanket, then mass transfer and formation of intermetallic compounds may be important issues. If molten solt extraction is used, then the effects of associated impurities in the primary cooling system should be explored.

Table 2-16. Reactor Relevant Conditions Required for Testing Momentum, Heat, and Mass Transport Issues

	Momentum Transfer	Heat Transfer	Mass Transfer
Magnetic Field	X	X	X
Velocity	X	X	X
Geometry Inside the Magnetic Field	X	X	X
Temperature Gradient		X	X
Temperature			X
Impurity Level			X
Material			X
Long Time Exposure			X
Geometry Outside the Magnetic Field			X

Table 2-17. Relative Importance of the Different Environmental Conditions Required for Testing Corrosion and Mass Transport Issues

	Local Attack	Dissolution	Deposition
Magnetic Field	--	X	XX
Velocity	--	X	X
Geometry Inside the Magnetic Field	--	X	X
Axial Temperature Gradient	X	X	--
Temperature	XX	XX	XX
Impurity Level	XX	XX	XX
Material	XX	XX	XX
Long Time Exposure	X	X	X
Geometry Outside the Magnetic Field	--	--	XX

-- not very important (20-50% effect)
X important (factor of 2 or more)
XX very important (exponential)

The type of loop and the size are important considerations. Thermal convection loops have the advantage of being simple to construct. However, they require relatively large heat inputs to circulate the coolant, and reactor relevant velocities are not achievable. Forced convection loops include pumps of some sort which can generate a high fluid velocity. They also re quire large heat inputs to maintain a loop temperature difference, but economizers can be utilized to recover much of the energy. Another consideration in choosing the type and size of the loops is the nature of the information desired. Even though Phase I emphasizes basic physical understanding, the sheer complexity of corrosion phenomena indicates that a complete physical description of the phenomena may never be found. Therefore, it is more efficient to operate immediately in a more empirical mode, even in Phase I. The base program should consist of forced convection loops, with a smaller set of thermal convection loops to assist in the interpretation of the data and development of models.

The size of the loops is also a critical parameter, because the cost generally scales with size. Since a very large number of loops is desired, they should be as small as possible. Table 2-18 gives example values of the major facility parameters for a small forced convection loop. The size of the loop is also an important factor which determines impurity generation and transport. In order to provide information to benchmark effects near reactor scale, a small number of large loops should operate, with parameter ranges such as given in Table 2-19.

In order to provide data for material selection and more integrated tests within 5 years, the various loops must be operated in parallel beginning immediately.

Because of the influence of the magnetic field on velocity profiles, it is quite likely that material interactions between the coolant and structure will also be affected. Earlier studies have shown that the effect could be as large as a factor of 5-10, especially in localized regions. A mass transfer facility is proposed to explore the influence of MHD velocity effects on mass transfer, called the MHDM facility.

2.3.4 Tritium Recovery and Control

Tritium control issues include two major categories: permeation rates

Table 2-18. Parameter Ranges for Smaller Forced Convection Loops

velocity	10-50 cm/s
channel radius	1-2 cm
volumetric flow rate	30-500 cm^3/s
temperature difference	150-250°C
gross thermal power[a]	10-300 kW

[a]In the form of surface heating, not including economizer

Table 2-19. Parameter Ranges for Larger Forced Convection Loops

velocity	10-50 cm/s
channel radius	4-6 cm
volumetric flow rate	500-5000 cm^3/s
temperature difference	150-250°C
gross thermal power[a]	15-3000 kW

[a]In the form of surface heating, without economizer

and extraction techniques. These two sub-issues are related, since the type of extraction system will be matched to the requirements on limiting tritium release rates. For lithium, molten salt extraction has been proposed and seems to be feasible. Before developing extraction systems for LiPb and Flibe, it is necessary to obtain better measurements of fundamental properties (solubility and diffusivity), and to characterize the permeation behavior in order to set guidelines for the extraction system.

Permeation and Properties Measurement

The classical permeation rate is dependent on the square root of the pressure. However, as the partial pressure is reduced, permeation will change from diffusion-limited to surface-reaction-limited, and consequently, the pressure dependence will change from square root dependence to linear dependence. The partial pressure at which this change will occur depends on the temperature, surface condition, and hydrogen isotopic composition. Consequently, the use of a classical permeation relationship to define the acceptable tritium partial pressure may be too conservative and may result in an oversized tritium recovery system. Therefore, the permeation rate and pressure-dependence relationship must be established. This will provide the design goal for the tritium recovery system.

The key material properties required are tritium solubility, diffusivity, and surface recombination constants. The size and cost of the tritium recovery system can be defined using these properties.

Extraction Techniques

Tritium recovery issues are completely different not only for different breeding materials, but also for different recovery schemes. Therefore, small-scale extraction experiments are required for scientific verification of different techniques for different breeding materials. The three primary candidates for liquid breeders are lithium, LiPb, and Flibe. For lithium, the solubility is relatively high and the partial pressure of tritium is fairly low. Therefore, tritium permeation is not a serious problem and tritium can be feasibly extracted by processing only part of the coolant stream. The largest issue is to maintain the total tritium inventory in the coolant within reasonable limits. For LiPb and Flibe, the solubility of tritium is low and partial pressures extremely high. Therefore, in order to prevent large quantities of tritium from escaping the system, a very efficient extraction system must be developed. The entire coolant stream may have to be processed on each pass through the blanket. For Flibe, it may be possible to use additives (e.g., lithium droplets) to increase the solubility and reduce tritium losses.

Permeation and Transport Loop

Once the most feasible tritium recovery techniques have been identified, a large tritium permeation and transport loop can be constructed. The purpose of this experiment is to demonstrate tritium recovery and transport on a continuous basis under fusion-relevant loop conditions. Since the loop is attached to the tritium recovery system, all the material problems caused by the impurities introduced from the tritium recovery system will be tested here. The tritium permeation rate depends strongly on the oxide layer condition, which in turn depends strongly on the steam-side conditions. Therefore, the steam-side conditions will also need to be reactor relevant.

The tritium permeation and transport loop could eventually be connected to a fuel processing facility (such as TSTA) to provide complete testing of the entire tritium process.

2.3.5 Tritium Breeding

Blanket neutronics experiments are required in order to establish tritium self-sufficiency for liquid breeder blankets. However, the uncertainty in tritium self-sufficiency for most liquid breeder blankets is much less than for solid breeder blankets. Since the issue of tritium breeding is less critical, it will probably not be a major discriminating factor for near-term blanket selection.

The general types of neutronics experiments for liquid breeder blankets are similar to those for solid breeder blankets. Tritium breeding experiments are already ongoing, and the test plan is described in detail in Section 2.2.

2.3.6 Structural Response and Failure Modes

The major uncertainties in structural response for liquid breeder blankets include material behavior under irradiation, mechanical response under complex loading conditions, and failure modes. Irradiation effects on basic properties of individual materials is an important topic, but is not considered here. Structural response under complex loading conditions can be addressed, to a large extent, by the MHD effects experiments, since most of the uncertainties in loading conditions involve effects of the magnetic field. The nature of the response is intimately tied to irradiation effects, material choices, configurations, and correct loading conditions. The first fully adequate test of component structural responses will, therefore, require a fusion test facility.

2.3.7 Testing in Later Phases

Although experiments beyond the first 5-10 years of the test program cannot be defined in detail, there are certain key facilities which can be anticipated. After the first phase of thermal hydraulic and material interactions experiments, it is anticipated that material choices will be ranked and candidate geometric configurations will be developed for the most promising ones. A Thermomechanical Integration Facility (TMIF) should be constructed to test the combined influence of heat, mass, and momentum transport, as well as some of the structural issues under non-neutron conditions. The purpose of TMIF is to aid in the selection of a small number of leading configurations and to begin to develop empirical relations describing the global behavior of the blanket. TMIF will be a larger facility than the early MHD experiments, with more prototypical blanket geometries present. In addition, a large number of attached subsystems will be present to more accurately represent the actual thermal and material environment of the blanket. These include, for example, representative pumps and heat exchangers, impurity control systems, and possibly a model of a tritium extraction system (without tritium).

After the operation of TMIF, one or two leading blanket configurations should be selected and defined in great detail. A non-neutron Partially Integrated Test Facility (PITF) could be operated to simulate all environmental conditions except neutrons at full or near-full scale. This facility has characteristics similar to the TMIF, and may be built as an upgrade of that facility. Partially integrated testing will ensure that when fusion integrated testing of blanket modules is performed, failure modes due to non-neutron effects can be anticipated and eliminated. Since fusion testing is expensive, it is important that initial test modules have a reasonable degree of reliability.

To clarify the role of the various MHD facilities which have been described, Tables 2-20 and 2-21 show the characteristics, objectives, and main features of the major liquid metal facilities.

2.3.8 Modeling Needs

The testing program should be accompanied by a strong complementary program of modeling and design work. One of the primary reasons for perform-

Table 2-20. Characteristics of Major Liquid Breeder Experiments

Characteristic	ALEX[a]	Magnetic Transport Phenomena Facilities: LMF[b]	Magnetic Transport Phenomena Facilities: MHDM[c]	TMIF[d]	PITF[e]
Fluid	NaK (100°C)	NaK	actual materials	actual materials	actual materials
Testing volume (m x m x m)	1.83 x 0.76 x 0.15 (0.21 m^3)	3 x 1 x .05 (1.5 m^3)		3 x 1 x .05	3 x 1 x .05
Magnetic Field	2 T	4-6 T		4-6 T	4-6 T
Configuration	simple geometry	elements of complex geometry		submodule	prototypic

[a]Exists (ANL)

[b]Liquid Metal Flow Facility

[c]MHD Mass Transfer Facility

[d]ThermoMechanical Integration Facility

[e]Partially Integrated Test Facility

ing experiments is to aid in the development of predictive capabilities. Therefore, the development of modeling capabilities is crucial to satisfying the objectives of the test plan.

Table 2-22 summarizes the principle model development needs for liquid breeder blankets. The two areas in which modeling development is most needed are MHD effects and material interactions. MHD phenomena are derivable, in principle, from the basic MHD equations, including Maxwell's equations and the Navier-Stokes equation. The primary difficulty lies in their simultaneous solution in complex, 3-dimensional geometries, such as a reactor blanket. Analytic models have been developed for some simple, specialized problems, but a general 3-dimensional model will require extensive development of numerical or combined analytic/numerical approaches.

Because of the large number of complex effects which are involved in material interactions, a single unified model will probably not be possible. However, some aspects of corrosion and mass transfer can be studied and

Table 2-21. Features and Objectives of Major Liquid Breeder Experiments

		Magnetic Transport Phenomena Facilities			
	ALEX	LMF	MHDM	TMIF	PITF
Features of Experiments	• Simple geometry of a channel	• Basic elements of relevant geometry	• Basic elements of relevant geometry • Relevant materials combination • Transport loop • Relevant T, ΔT, impurities, V • Long operating time per experiment	• Actual materials and geometry • Transport loop • Relevant environmental and operating conditions	• Prototypic blanket module • Transport loop • Prototypic environmental and operating conditions
	• Measure velocity profile, electric potential pressure drop (may be upgraded)	• Measure velocity and temperature profiles; pressure drop, temperature, electric potential	• Measure dissolution and deposition rates	• Measure integral quantities (ΔP, T, corrosion and deposition rates)	• Measure integral quantities
Objectives	• Develop and test velocity profile instrumentation in NaK environment • Validate MHD in simple geometry (basic heat transfer data may be possible in upgrade)	• Develop and test instrumentation • Validate MHD, MHD heat transfer • Design data (ΔP, T) for configuration screening and provide input information to design TMIF • Explore techniques to reduce ΔP and enhance heat transfer	• Develop and test instrumentation in relevant environment • Design data on MHD heat and mass transfer • Verify techniques to reduce corrosion and corrosion effects	• Design data for blanket test module • Confirm and refine configurations	• Engineering design data • Reliability data in non-fusion environment

modeled in order to increase our understanding and predictive capabilities. These include dissolution and transport at solid-liquid interfaces, kinetics of reactions and transport processes, and primary cooling system global empirical modeling.

Table 2-22. Summary of Model Development Needs

<u>MHD Effects</u>

- analytic modeling of velocity profiles in straight ducts and simple geometries
- 3-dimensional MHD computer codes for the solution of velocity profiles in complex geometries
- semi-empirical design codes for prediction of fluid flow behavior in complex 3-dimensional geometries

<u>Material Interactions</u>

- dissolution modeling with solid phase, liquid phase, and interface transport processes
- modeling of the kinetics of material interaction processes
- primary cooling system loop modeling

<u>Tritium Transport</u>

- simple loop codes to predict transport and permeation rates in an integrated primary cooling system with extraction system
- surface absorption/desorption modeling for tritium permeation through metal barriers

2.3.9 Test Sequence and Logic

A possible sequence of testing has been defined using the experiments and facilities described above, and is presented in Figure 2-8. Several tasks are already underway, such as the ALEX facility, corrosion/mass transport loops, neutronics experiments at FNS, and others. The test plan calls for continuation of these experiments and in some cases an increase in the level of activity (for example, more corrosion/mass transfer loops are required). Two tasks which should be initiated immediately are insulator development and bimetallic mass transfer tests.

After approximately five years, decisions will be necessary regarding both the blanket materials and designs as well as the new facilities which should be built. One of the important decisions will be to determine the

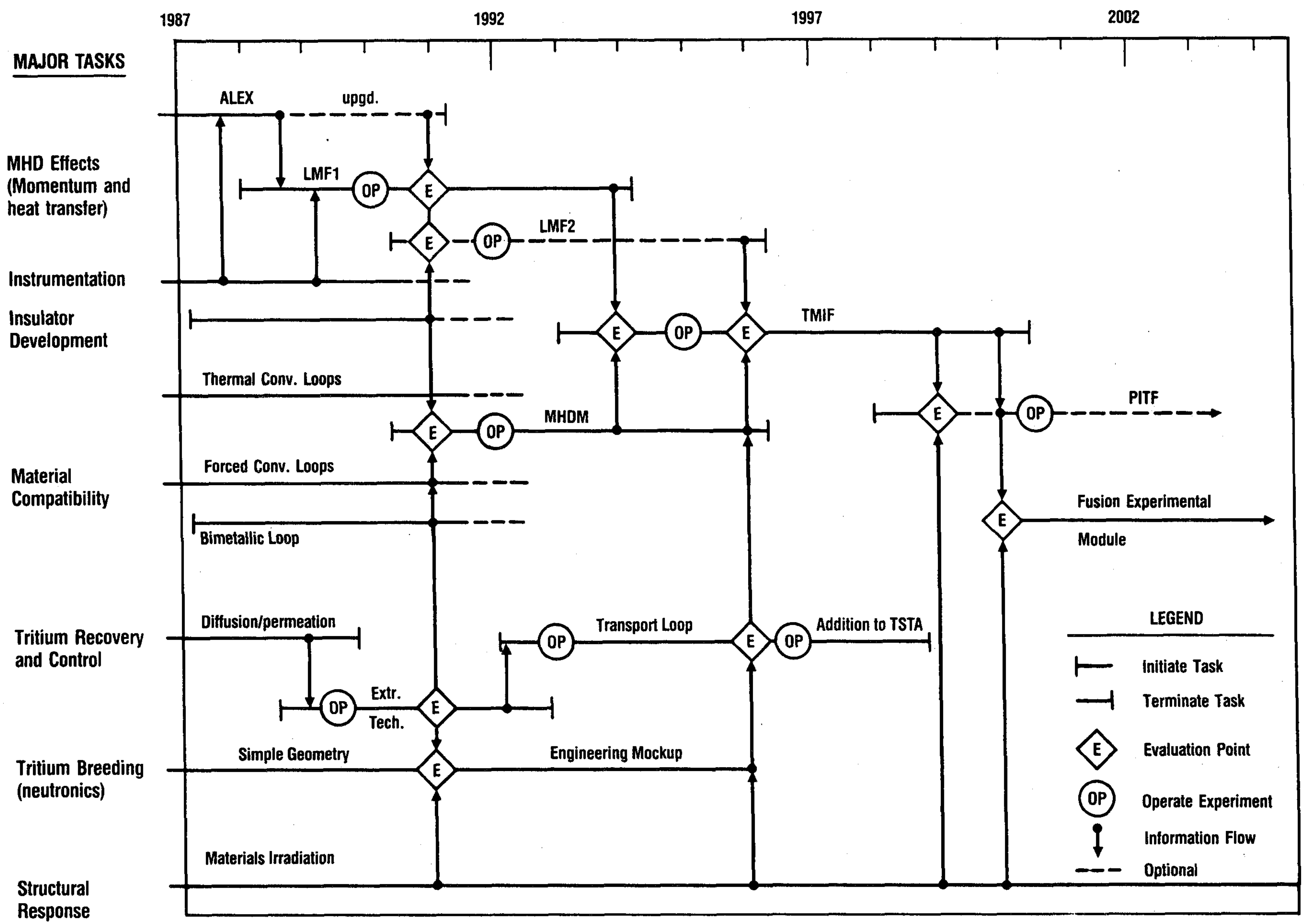


Figure 2-8. Test sequence for major liquid breeder blanket tasks

emphasis of the LMF facilities and the need for additional experiments and facilities, such as LMF2 and MHDM.

After approximately ten years, relatively detailed blanket designs should exist, and the TMIF can be designed in detail. By that time, the role that fusion testing will play should be more clearly defined. Together with the results of testing in TMIF, this information will lead to a decision on the role, timing, and need for PITF.

A rough estimate has been performed to determine the total cost of the test program and the results are given in Table 2-23. The costs are broken down into two categories: capital and operating costs. Capital costs include design effort, materials, fabrication, construction, and any expense directly related to the construction of the facility and the experimental apparatus. Annual operating costs include use of materials and energy, staff to operate the experiments, and data acquisition. The cost of analysis for the experiments, modeling efforts, blanket design studies, etc. have not been included as operating expenses. They are listed in Table 2-23 as a separate item. A liquid breeder blanket program requires an average annual expenditure of about 10-20 M$.

Table 2-23. Representative Costs of Key Liquid Breeder Blanket Facilities

Item	Capital Cost[a] (M$)	Operating Cost[b] (M$/yr)	Duration (years)	Total Cost (M$)
Advanced liquid metal flow facility (LMF1)	7-10	0.5	4-6	10-15
Integral Parameter Experiment (LMF2)	7-10	0.5	4-6	10-15
MHD mass transfer facility (MHDM)	8-12	1.0	6-8	15-20
Thermal convection loops (~4)	2-4	0.8	4-6	5-9
Forced convection loops (~4)	4-6	0.8	4-6	7-11
Tritium extraction test (2)	2-3	0.4	3-4	3-5
Tritium transport loop test	6-8	0.6	5-7	9-12
Thermomechanical Integrated Test Facility (TMIF)	20-25	2.0-3.0	8-10	35-60
Analysis and model development	--	2.0-4.0	15	30-60

[a]In 1985 constant dollars

[b]Does not include analysis of data