SPECIAL TOPIC

TECHNICAL ISSUES AND REQUIREMENTS OF EXPERIMENTS AND FACILITIES FOR FUSION NUCLEAR TECHNOLOGY

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ABSTRACT. The technical issues, development problems and required experiments and facilities for fusion nuclear technology have been investigated. The results have been used to develop a technical framework for a test plan that identifies the role, timing, characteristics and costs of major experiments and facilities. A major feature of this framework is the utilization of non-fusion facilities over the next 15 years, followed by testing in fusion devices beyond about the year 2000. Basic, separate effect and multiple interaction experiments in non-fusion facilities will provide property data, explore phenomena and provide input to theory and analytic modelling. Experiments in fusion facilities can proceed in two phases: (1) concept verification and (2) component reliability growth. Integrated testing imposes certain requirements on fusion testing device parameters; these requirements have been quantified. The nuclear subsystems addressed in the study are: (a) blanket and first wall; (b) tritium processing system; (c) plasma interactive components; and (d) radiation shield. The two generic classes of liquid and solid breeder blankets have significant engineering feasibility issues, and new experimental data must be obtained before selection of an attractive design concept. Liquid metal blanket issues are dominated by problems related to momentum, heat and mass transfer, which can be addressed in non-neutron test facilities. Solid breeder blanket issues are, however, dominated by the effects of radiation, including heating, transmutation and damage, which can be reasonably addressed in fission reactors. The tritium processing uncertainties are primarily related to the control and recovery systems, and most can be addressed in existing and planned non-neutron facilities. A dominant feature of plasma interactive components is the strong interrelation to both plasma physics and nuclear technology. Required facilities include thermomechanical test stands and confinement devices with sufficiently long plasma burn. The radiation shield poses no feasibility issues, but improved accuracy of predictions will reduce design conservatism and lower costs.

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CONTENTS.

1. INTRODUCTION

Fusion nuclear technology is critical to the development of viable and attractive commercial fusion reactors because it poses major engineering feasibility issues, and because it will strongly impact fusion's ultimate economic, safety and environmental attractiveness. Enhanced research and development programmes 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 reactor characteristics, and hence providing timely feedback to plasma physics and confinement experiments.

The development of a new technology, such as fusion energy, starts with a proposed application of a scientific principle and, if successful, ends with a commercial product. In between, three important activities always take place, as illustrated in Fig. 1. In the first activity, design options are examined and compared, generally based on a very limited data base. The objective of design studies is to identify the most promising concepts, 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) programme. R & D implementation, the third activity in Fig. 1, refers to constructing experimental facilities and performing experiments.

Experiment Planning provides the crucial link between design studies and R & D implementation. The purpose of Experiment Planning is to develop and direct a cost effective research strategy, and is particularly important for complex, expensive and long term efforts such as controlled nuclear fusion. This paper summarizes the development and application of an Experiment Planning process for fusion nuclear technology. This work has been performed as part of the FINESSE study [1, 2].

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

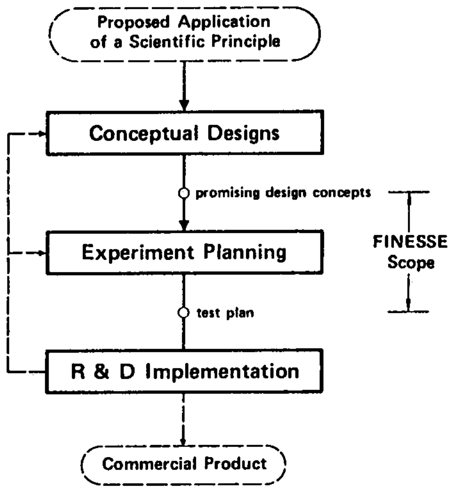


FIG. 1. Role of experiment planning in technology development.

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.

1.1. Approach

The main elements of the approach adopted for Experiment Planning are shown in Fig. 2. The primary input to the process is a set of promising design options for a particular component. The major output is a test plan that identifies and quantifies the role, timing and characteristics of major experiments and facilities. The process consists of four steps: (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 are generally carried out with considerable feedback and iteration.

The first step, characterization of issues, involves: (1) assessment of the accuracy and completeness of existing data and models; (2) analysis of phenomena to determine (or anticipate) behaviour, interactions and governing parameters in the fusion reactor environment; (3) evaluation of the effect of uncertainties on design performance; and (4) comparison of tolerable and estimated uncertainties. This process provides quantified understanding of the issues and their relative priorities.

The second step, quantification of experimental needs, involves: (1) survey of needed experiments; (2) exploration of engineering scaling options (engineering scaling is the process of developing useful tests at experimental conditions less than those in a reactor); (3) evaluation of effects of scaling on usefulness of experiments in resolving issues; and (4) identification of desired experiments and experimental conditions.

In the third step, existing facilities are evaluated with respect to: (1) their capabilities and limitations; (2) meaningful experiments that could be performed in such facilities; and (3) estimated costs for such experiments. Issues that cannot be resolved in existing facilities require the construction of new facilities. In evaluating new facilities, the effort is focused on: (1) exploring innovative testing ideas; (2) assessing the feasibility of obtaining the desired information, e.g. instrumentation limitations; (3) developing conceptual facility designs and cost estimates; and (4) comparing possible experiments and facilities based on technical usefulness, time and cost.

In addition to information from the first three steps, the final step towards a test plan requires input on programmatic considerations such as budget and time constraints. This step depends on the complexity of the issues and the level of detail required in the test

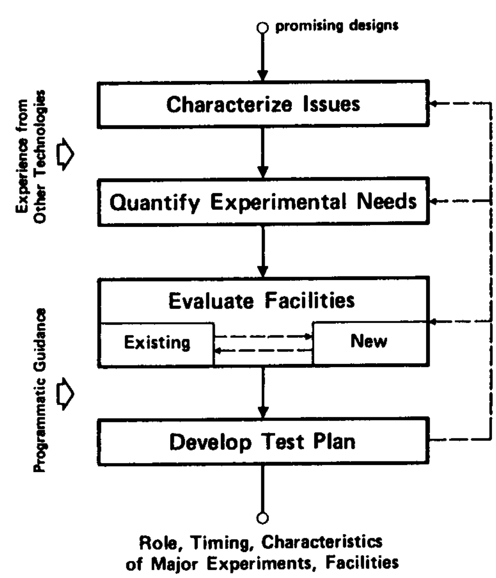


FIG. 2. Process for experiment planning.

TABLE I. REPRESENTATIVE GOAL RANGES CONSIDERED FOR COMMERCIAL REACTOR PARAMETERS

Parameter	Range
Neutron wall load ($MW \cdot m^{-2}$)	4–6
Surface heat flux at first wall ($MW \cdot m^{-2}$)	0.2–1
Average heat flux in high heat flux components (e.g. limiter/divertor) ($MW \cdot m^{-2}$)	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 ($MW \cdot a \cdot m^{-2}$)	15–20

plan. In general, several test plans could be developed and then compared in terms of risk, usefulness and cost.

1.2. Goals, objectives and assumptions

The principal goal of this study is to recommend 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. A previous study [1] has explored the issues and testing needs for fusion nuclear technology in greater detail. In the present study, the issues are briefly reviewed and the bulk of the effort concentrates on experiments, facilities, and the overall test plan.

The ultimate goal of fusion R & D is the development of commercial fusion reactors. The results presented here have 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. This objective is consistent with existing worldwide plans [3, 4].

The study minimized restrictive assumptions on the major characteristics of a commercial fusion reactor in order that the results apply to a broad-based fusion technology development program. Nevertheless, a number of assumptions were made to keep the effort manageable. These are:

(1) Electricity production is the primary purpose of the reactor. The impact of non-electric applications, e.g. hybrids, has not yet been investigated.
(2) Tokamaks and tandem mirrors are considered as the primary confinement concepts. Changes in R & D for reversed field pinches (RFPs) [5] need further evaluation.
(3) Representative ranges for key parameters of commercial fusion reactors [6–8] are shown in Table 1.

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

1.3. Types of experiments and facilities

The types of experiments can be classified into: (1) basic, (2) separate effect, (3) multiple interaction, (4) partially integrated, and (5) integrated tests. Figure 3 illustrates the role of these types of experiments and the strong interrelation between experiments and analytic modelling. This 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 present or simulated in the experiment.

Basic tests measure property data. Single effect tests are experiments with one primary controlled environmental condition, and are aimed at developing models and understanding of single phenomena.

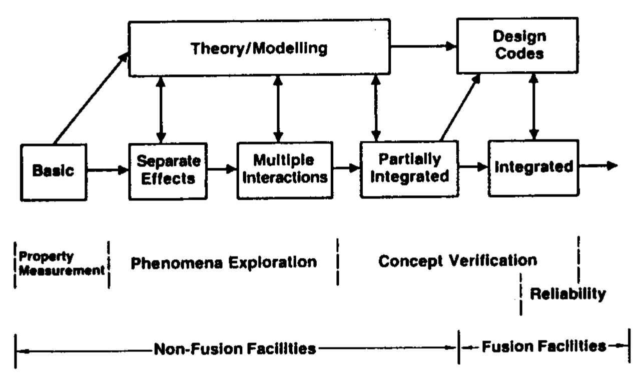


FIG. 3. Types and role of experiments and facilities for fusion nuclear technology.

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 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 synergistic 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, concept verification is unlikely before performing fully integrated tests [1]. While 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 and neutron producing facilities, which include fission reactors and 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.

1.4. International co-operation

International co-operation has long been recognized as an important mechanism for maximizing progress in fusion. There are particularly strong incentives for pursuing international co-operation on fusion nuclear technology, including [9]:

(a) There are many areas of key R & D needs for fusion nuclear technology that are of common interest to all countries.
(b) Substantial resources of both manpower and facilities are required to resolve the key issues. International co-operation is thus a desirable, and in some cases necessary, means for cost sharing.
(c) International co-operation can accelerate progress and enhance the prospects for successful development of attractive fusion nuclear components. Effective co-ordination of intellectual and hardware resources in the world programmes will permit faster and more complete exploration of promising options, identification of critical problems and development of attractive solutions.

This study attempts 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.

2. BLANKET

2.1. Introduction

The first wall/blanket is a particularly important fusion nuclear component. It will play a dominant role in determining the ultimate attractiveness of fusion energy sources in terms of economics, safety and environmental impact. All presently proposed blanket concepts have critical feasibility and attractiveness concerns. The issues, experiments and facilities are most complex for the blanket. Therefore, this component has received the greatest attention in this study.

Numerous blanket design studies have been conducted worldwide. Many of these studies have been carried out in the context of conceptual designs for power reactors (see Refs [8, 10–15] for examples) or for near term engineering testing facilities such as INTOR [16], NET [17], and FER [18]. Recently, the Blanket Comparison and Selection Study (BCSS) [19] in the USA and another study [20, 21] carried out by a number of organizations in Europe have focused explicitly on comparisons of blanket concepts. It is clear from these studies that there is no universally accepted blanket concept. The preferences for various blanket concepts vary among researchers, organizations and countries. Blankets are in the early stages of development and it is likely that new design concepts will emerge as a result of future innovation and better understanding of the technical problems based on new experimental data and analysis.

For the purpose of planning experiments and facilities, we have attempted in this study to select representative classes of designs which appear to pose generic issues and whose experimental and facility needs can cover the fundamental data base required for blanket R & D. The identified issues and testing plans will need to be revised in the future as new results become available.

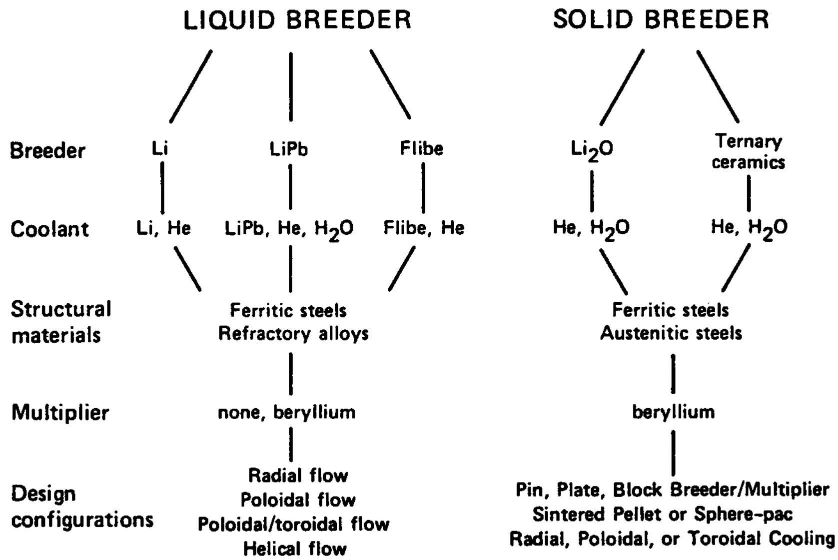


FIG. 4. Primary blanket options.

Blanket concepts can be divided into liquid and solid breeders. Within each class there are a number of distinct material and configuration options. Figure 4 shows the primary options of most interest to many of the world fusion programmes.

Solic breeders include Li_2O and ternary ceramics such as $LiAlO_2$, Li_2SiO_3 and Li_2ZrO_3. Helium and water cooled solid breeder blankets appear to be of interest in the EC, Japan, Canada, USA, and USSR. Martensitic and austenitic steels have been considered as structural materials. The use of vanadium with helium and water has been considered in BCSS [19], but this option was given a lower priority because of compatibility concerns. All solid breeders, with the possible exception of Li_2O, require a neutron multiplier for adequate tritium breeding [22]. The use of Zr_5Pb_3 was considered in STARFIRE [8], but subsequent analysis in the early stages of BCSS [23] concluded that the neutron multiplication in Zr_5Pb_3 in a practical engineering design does not provide adequate margin to compensate for the uncertainties in tritium breeding [22]. Lead has been considered for INTOR and appears to be suitable for fusion devices with a low neutron wall load of $\sim 1\ MW \cdot m^{-2}$. However, for fusion power reactors with much higher wall loading, the combination of high power density, low melting point and relatively poor thermal conductivity make lead unsuitable as a neutron multiplier. Beryllium appears to be the best neutron multiplier. It is worth noting that the USSR designs which are focused primarily on fusion–fission hybrids also utilize beryllium as a neutron multiplier.

There are solid breeders and coolants other than those discussed above that have been considered in the literature; for example, Li_7Pb_2 as a breeder and certain molten salts as coolants with solid breeders. These were explicitly assessed in the interim report of BCSS [23] and given much lower priority for various reasons. Furthermore, they do not presently appear to be of wide interest. Therefore, they were not included in this study.

Liquid breeders include liquid metals and molten salts. Lithium has been considered most extensively in the USA, while lithium-lead (17Li–83Pb) is of interest in Europe and the USA. Liquid metal blankets can be classified into (a) self-cooled designs in which the liquid metal serves as the breeder and the coolant, and (b) separately cooled designs in which the liquid metal serves only as the breeder and a separate coolant such as helium and water are used. Self-cooled liquid metals are presently receiving considerable attention in the USA. There appears to be more emphasis on separately cooled designs in Europe.

Many issues are similar for self-cooled and separately cooled designs. For example, MHD effects on fluid flow are of critical concern in self-cooled liquid metal blankets. In separately cooled designs, the breeder must still be circulated for external tritium extraction. Although the required speed is low for lithium, it approaches that of the coolant for Li–Pb because of its low tritium solubility. Therefore, many of the fundamental aspects of MHD effects on fluid flow need also to be understood for separately cooled design. designs.

Austenitic steels are generally considered inviable structural materials for liquid metal blankets. Martensitic steels and advanced alloys are generally preferred.

In the USA, vanadium alloys are normally regarded as representative of the advanced alloys.

Issues related to structural materials, including radiation effects, are of considerable importance. These issues are treated here to the extent necessary to address the overall blanket performance. However, specific problems related to testing the end of life radiation effects on the structural material alone are not treated in detail.

There are a variety of possible configurations for blankets. Many issues are affected by the configuration selected. In carrying out this study, specific blanket designs had to be used as examples in order to provide a more in-depth quantitative analysis of the technical issues. Some of the best concepts proposed in BCSS [19] were selected for this purpose. However, in characterizing and ranking the issues, and in defining the experiment and facility needs, attempts were made to include the generic features of as many of the primary candidate blanket concepts as possible.

The critical issues and the experiment and facility needs differ greatly between liquid breeder and solid breeder blankets. Considerable savings in the R & D programme can be realized if one of these two classes is selected now. However, such selection at present will entail very high risks because both classes have critical feasibility and attractiveness issues. Consequently, it appears prudent for the fusion programme to retain both options, although a selection can be made 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.

2.2. Solid breeder blankets

2.2.1. Issues and testing needs

The general classes of issues for solid breeder blankets are given in Table II. These are based on the characteristics of solid breeder concepts from recent studies (e.g. Refs [1, 2, 21–24]). 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 behaviour. 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 materials.

TABLE II. SOLID BREEDER BLANKET ISSUES

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

2.2.1.1. Tritium self-sufficiency

Most solid breeder blankets require ^{6}Li enrichment and a neutron multiplier for adequate tritium breeding, with the possible exception of Li_2O. Even so, it is not clear that present solid breeder blanket concepts provide reactor self-sufficiency in tritium. Table III indicates the calculated 3-D tritium breeding ratio (TBR) for several blankets, and the estimated uncertainty in this TBR based on sensitivity studies. None of the blankets achieve a TBR with a margin sufficient to compensate for all estimated uncertainties [22].

The need for a neutron multiplier is a key issue for Li_2O. In all multiplied solid breeders, however, the tritium breeding is affected by the form in which the multiplier is incorporated – which also affects the

TABLE III. ACHIEVABLE TRITIUM BREEDING RATIOS AND ASSOCIATED UNCERTAINTIES FOR SEVERAL TOKAMAK BLANKETS [22]

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

TABLE IV. CONTRIBUTIONS TO BLANKET TRITIUM INVENTORY FOR SEVERAL [23] 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
He + H_2/He only	~0/1200	~0/1200	~0/1200	~0/1200	large
Pores/purge	0.04	0.04	~0	~0	large
Beryllium[c]	0	<4000	<4000	<4000	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 ten).
[c] Upper limit refers to complete retention of tritium, end of life conditions, and 1% tritium breeding occurring in Be.

TABLE V. VARIATION OF THERMAL CONDUCTIVITY WITH BREEDER CONDITIONS [29]

Breeder condition					Thermal conductivity[a]	
Density (%TD)	Temp. (°C)	Form	Fluence ($n \cdot m^{-2}$)	Helium pressure (MPa)	Li_2O ($W \cdot m^{-1} \cdot K^{-1}$)	γ-$LiAlO_2$ ($W \cdot m^{-1} \cdot K^{-1}$)
100	700	Sintered	0	–	4.8	2.6
85	700	Sintered	0	–	*3.8*	*2.5*[b]
85	400	Sintered	0	–	*5.2*	*2.5*
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] Values in italics are measured, the rest are extrapolated.
[b] From recent data; previously estimated value was 1.9 $W \cdot m^{-1} \cdot K^{-1}$.

tritium and thermal behaviour. An accurate assessment of the tritium breeding margin would thus indicate whether blankets without distinct multipliers were possible and, if not, what level of physical separation was acceptable.

2.2.1.2. Tritium recovery

The prediction of tritium behaviour 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 are indicated in Table IV. The major contributors are the diffusivity, solubility and surface adsorption processes.

The uncertainty in the diffusivity can be much more than an order of magnitude, particularly at higher temperatures and burnups [25]. The soluble tritium inventory is believed to be large only for Li_2O, where it is reasonably well measured [26]. The surface inventory could be large for all breeder materials and is sensitive to surface conditions and the breeder chemical environment, particularly the oxygen activity (effectively, the O_2 partial pressure) [27]. The O_2 activity can, however, vary over many orders of magnitude, depending on the controlling thermodynamic system and the reaction kinetics.

Sufficient tritium is produced in the beryllium multiplier to also be of concern (about 2 g per day in a 5000 MW(th) reactor). The same tritium transport phenomena apply as with solid breeder materials, but there are scarcely any data to assess their magnitude. Even if the tritium is released from the multiplier, the tritium must be removed by the coolant or purge streams. These may lead to coolant contamination or breeder/multiplier chemical interaction concerns.

2.2.1.3. Thermal, mechanical and corrosion behaviour

The major issues associated with the mechanical interactions between the solid breeder, multiplier and structure are: cracking and restructuring of the solid breeder; changes in heat transfer across the breeder/cladding interface; and deformation and/or rupture of the structure (e.g. Ref. [28]). 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. There are no completed experiments that indicate the extent and consequences of mechanical interactions or temperature gradients within the breeder. Beryllium has been used in fission reactors, but the available irradiated mechanical property data are generally at low temperature ($\sim 100°C$) and low fluence compared to the anticipated end of life blanket conditions.

The thermal behaviour of the breeder is constrained by the relatively low thermal conductivity (~ 1-$3\ W\cdot m^{-1}\cdot K^{-1}$) and upper temperature limits (~ 800–$1000°C$) assumed for present solid breeder materials. These are important to the blanket design, but present values are only roughly estimated. For example, Table V illustrates the predicted large reduction in thermal conductivity for reasonable breeder conditions, and the limitations of present data. Other characteristics such as the breeder upper temperature limits are not even well defined, but depend on many processes such as sintering, creep, phase change, vapour phase transport and corrosion.

2.2.1.4. Structure response

The mechanical behaviour 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.

2.2.1.5. Tritium permeation and processing

The permeation of tritium outside of the breeder and into the coolant is an important safety concern, but the nature and effects of the chemical environment and surface conditions are uncertain. Also, uncertainties in the recovery of tritium from the breeder purge stream include the tritium form (gas or oxide, amount of added H_2 or O_2), the efficiency of the recovery process, and the tritium inventory in the recovery system.

2.2.2. Existing and required experiments and facilities

The issues can be addressed by a range of possible experiments, as summarized in Fig. 5 and discussed below. The actual experiments will depend on particular test programme assumptions and funding constraints. These tests are organized according to their level of integration, from basic properties, to phenomena in separate and multiple effect tests, to concept verification in integrated fusion tests.

Since there is no general theoretical basis for scaling solid breeder behaviour, the significant phenomena must be quantified by conducting tests at fusion reactor relevant conditions. Among the most important parameters are the nuclear effects, particularly tritium generation and heating. The ability of fission reactors to match fusion conditions is shown in Fig. 6. By appropriate ^6Li content in the breeder material, and choice of fission reactor, it is possible to simulate fusion tritium generation, heating and lithium atom burnup over reasonable irradiation periods. Reactors with vented test capabilities can also provide direct simulation of the purge environment. Overall, nuclear testing in existing fission reactors is an important resource for solid breeder blankets [30].

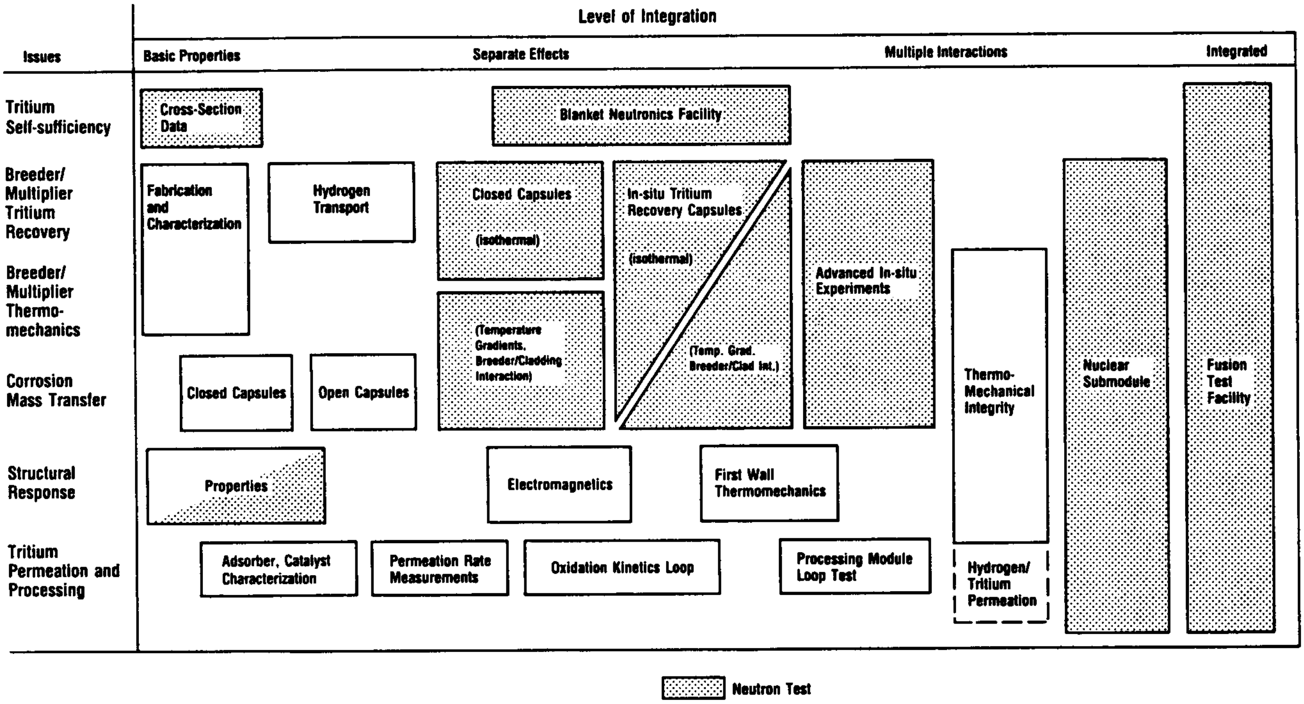


FIG. 5. Types of experiments and facilities for solid breeder blankets (some experiments and/or facilities already exist).

2.2.2.1. 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 these experiments is to fabricate, characterize, and improve the properties of candidate breeder materials. The measurement of tritium recovery and thermal behaviour in closed and open capsule irradiation of material specimens is an integral part of this task. Various completed and active irradiation experiments to characterize and understand material parameters are summarized in Table VI. 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. These variables include 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 programme should also include mixtures of multiplier and solid breeder.

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. The present structural material options under evaluation in the USA for solid breeder blankets are an austenitic steel (Primary Candidate Alloy, or PCA) and a martensitic steel (HT-9). The use of high-temperature refractory materials in solid breeder blankets 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 [31]. The development and characterization of structural alloys is a common need for all fusion nuclear components and is not discussed here.

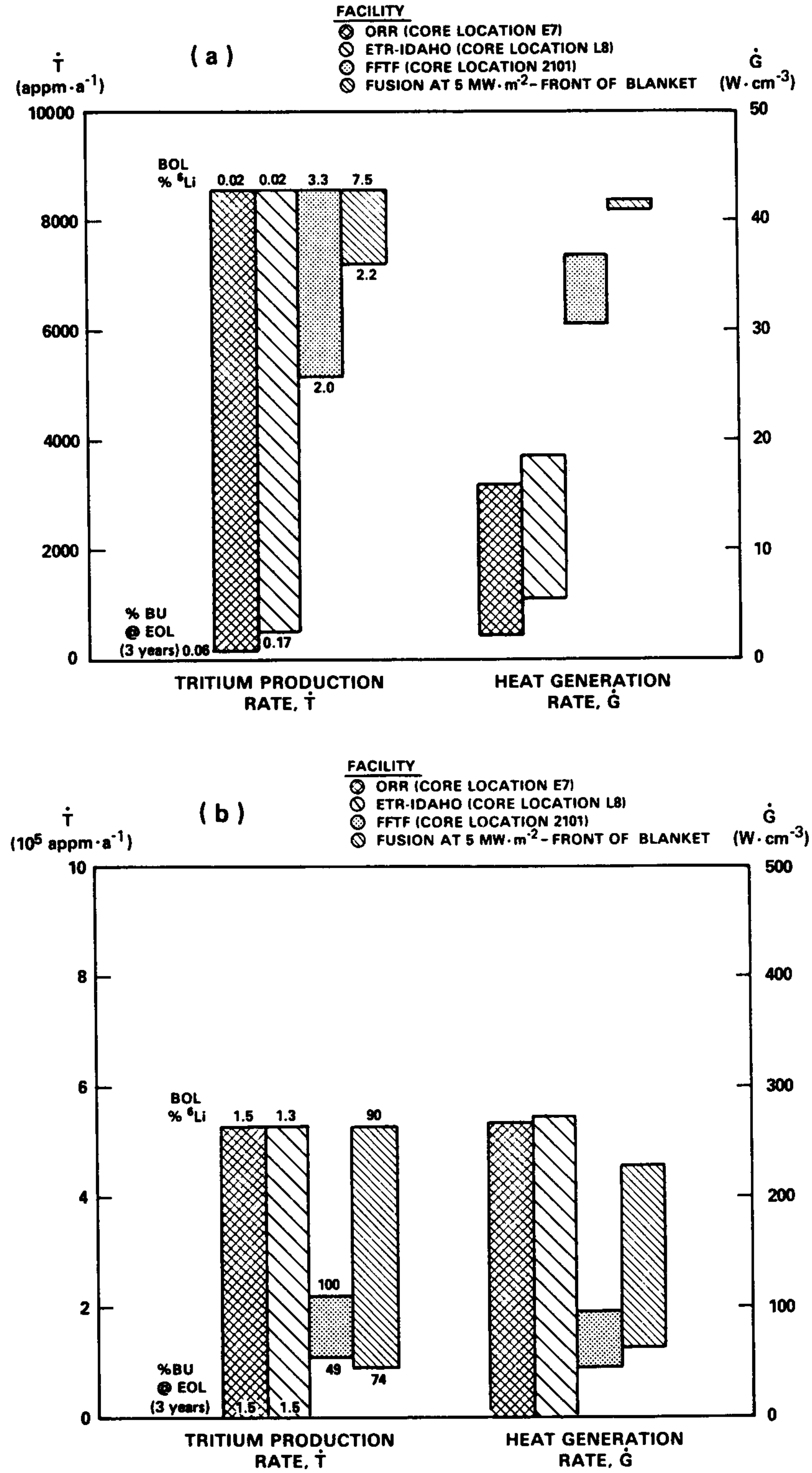


FIG. 6. Comparison of tritium generation ($\dot{T}$) and heating rates ($\dot{G}$) for (a) Li_2O and (b) $LiAlO_2$ irradiation in thermal (ORR and ETR), fast (FFTF), and fusion reactors at beginning of life (BOL) and end of life (EOL) three years later. The ^6Li content is initially set to produce the same BOL tritium generation rate as in a fusion reactor. The total per cent lithium burnup at EOL is also shown.

TABLE VI. COMPLETED AND ACTIVE SOLID BREEDER MATERIAL IRRADIATION EXPERIMENTS

Experiment	Ceramic	Grain size (μm)	Density (%TD)	Temperature (°C)	Li burnup (Max at.%)	Time Frame
Closed Capsule						
ORR (US)	Li_2O	< 47	70	750,850,1000	0.05	84
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	1,10	80,95	400,600	-	85/86
	Li_2O	5-10	80,90	400,600	-	85/86
	$LiAlO_2$	1,8	80	400,600	-	85/86
CREATE (Canada)	$LiAlO_2$	-	60-90	100	<0.05	85/86
	Li_2O	-				
---	$LiAlO_2$	-	-	300,...,800	-	85/86
(USSR)	Li_7Pb_2	-	-	-	-	
In-situ Tritium Recovery						
TRIO (US)	$LiAlO_2$	0.2	65	400,...,700	0.2	84/85
		(50 μm particles in pellets)				
VOM-15H (Japan)	Li_2O	< 10	86	480,...,760	0.24	84
VOM 22/23 (Japan)	Li_2O	-	-	400-900	0.04	86
		(4 mm dia. pellets)				
	$LiAlO_2$	0.5	77	400-900	0.1	86
		(4 mm dia. pellets)				
LILA (France)	$LiAlO_2$	0.4-13	78	375-600	< 0.02	86
		(1 cm dia. pellet)				
LISA (Germany)	Li_2SiO_3	30-80	86,93	550,600	< 0.02	86
		(1 cm dia. pellet)				
EXOTIC (Neth./UK/ Belgium)	$LiAlO_2$	1,8	80	400,600	< 0.4	86
		(1 cm thick annular pellet)				
	Li_2SiO_3	1,10	80,95	400,600	< 0.4	86
		(1 cm thick annular pellet)				
CRITIC (Canada)	Li_2O	20	80	400-950	0.15	86
		(1 cm thick annular pellet)				
	$LiAlO_2$	1	85	400-1000	0.15	87
		(sphere-pac)				

2.2.2.2. Multiplier material development and characterization

The objective of multiplier development and characterizations 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, but lead (alone or as solid Li_7Pb_2) and Zr_5Pb_3 have been considered. 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. 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 behaviour under irradiation is dominated by helium production from the (n, 2n) reaction (~1.7 MeV threshold). Tritium production in Be occurs because of a high neutron energy reaction (which dominates in a fusion reactor), and a lower energy reaction (which dominates in a fission reactor). The latter reaction allows a fission reactor like FFTF to provide reactor relevant helium and tritium production.

2.2.2.3. Tritium recovery experiments

The most important tests involve irradiation to provide internal tritium generation, heating and fluence effects. These can be 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 [32]. 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 (see Table VI) are also underway or have been completed. These tests are exploring a range of temperatures, temperature gradients, breeder materials, 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, 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 in-core capsules.

Relevant parameter ranges are given in Table VII. (Also shown are parameter ranges for a partially integrated nuclear submodule test, described below). An important part of these experiments would be to investigate design limits (e.g. upper temperature limits) and transient behaviour.

TABLE VII. 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 ($°C \cdot cm^{-1}$)	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 ($m^3 \cdot s^{-1} \cdot g^{-1}$)[a]	0.01–0.1	0.01–0.1
Burnup (at.% Li)	3–10	3–10
Heat generation ($MW \cdot m^{-3}$)	30–100	30–100
Irradiation time (a)	1–3	1–3
Tritium production (T/Li·a)	0.01–0.5	0.01–0.5

[a] Normalized per gram of solid breeder material.

These experiments 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, large test volume fission reactors with the ability to handle purged and instrumented assemblies. There is some preference for fast neutron spectra 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.

2.2.2.4. 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 behaviour are affected by radiation (swelling, creep) and 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).

2.2.2.5. Corrosion and mass transfer experiments

Experiments to determine temperature limits based on material interactions involve long term tests of relevant materials and impurities at temperatures which will be achieved in many of the tritium recovery experiments. However, for new and/or more reactive materials, separate unirradiated testing at relevant temperatures for long time periods can provide cost effective data to judge the feasibility of the material or to provide well defined test conditions for model development. Useful tests include mass transfer within and from Li_2O in a purge stream with hydrogen, and the interaction kinetics of beryllium with solid breeder and clad.

2.2.2.6. Structural response experiments

The most important element of determining the structure mechanical behaviour 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.

The modelling basis for structural behaviour is reasonably well established from fission programmes, 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 ferromagnetic steel structures or transient forces on any structure) and the behaviour of the first wall under high heat flux and cyclic 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. The latter, more integrated tests are discussed later.

2.2.2.7. Neutronics and tritium breeding

The objective of neutronics experiments is to measure the tritium production rate and heating rate distributions in order to verify and improve nuclear data, design methods and models [33, 34]. Two stages of testing can be identified: simple geometry mockups and engineering mockups.

Simple mockups would be conducted with geometrically simplified blanket modules that incorporate the primary breeder and blanket materials. A well calibrated 14 MeV neutron source with sufficient strength (about 10^{12} n$\cdot$cm$^{-2}\cdot$s^{-1}) 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 the Japan Atomic Energy Research Institute (JAERI). Other facilities exist in Japan (OCTAVIAN), Europe (LOTUS), and the USA (RTNS-II). Such experiments are beginning in 1986 as part of the US/JAERI Fusion Breeder Neutronics Collaborative Program, initially with Li_2O [35]. About five years will be needed to explore the major materials and combinations of present international interest.

Some of the important uncertainties in the assessment of 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

TABLE VIII. SOLID BREEDER BLANKET MODELLING NEEDS FOR THE MAJOR TASKS

Material development and characterization
Tritium diffusion with impurities and radiation damage
Thermal conductivity as a function of microstructure
Swelling and creep of breeder and multiplier
Breeder thermal behaviour
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)

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. These experiments would not begin until module and reactor concepts are reasonably well defined.

2.2.2.8. Tritium permeation and processing experiments

Many of the issues associated with tritium inventory, permeation rate and oxidation kinetics can be addressed in separate glovebox scale experiments. The use of tritium directly accounts for any isotope effects and provides finer accuracy, which may be necessary at the low tritium partial pressures relevant to some applications. Concerns over irradiation effects on trapping and transport will require ion beam and/or fission reactor irradiation facilities.

Processing system loop tests (including molecular sieves, oxidizers, getters, etc.) can be performed with reactor relevant modules to explore tritium holdup, efficiency, lifetime, and general operations. These are reasonably small sized experiments because of the modularity and size of the full scale components.

2.2.2.9. Partially integrated experiments

More complex tests with more relevant geometry, size, and environmental conditions can provide some concept verification information. Non-neutron test stands, fission reactors, and fusion devices can serve different roles. However, only a fusion reactor can provide fully integrated testing.

Non-neutron thermomechanical tests with non-nuclear heat sources such as microwaves [36] or resistive wires can be used to test up to full blanket modules. Although there are clearly limitations on the simulation of reactor heating profiles and irradiation effects, these tests provide an opportunity to explore complex thermomechanical behaviours (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 contraints. 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.

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. The 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.

Nuclear test assemblies for fission reactors can provide the maximum concept verification possible in non-fusion devices [37]. 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 $MW \cdot m^{-2}$ heating rate in any existing reactor. In-core assemblies could be placed in existing fission reactors like FFTF at reactor relevant heating rates (2–5 $MW \cdot m^{-2}$), but would be limited to about 10 cm diameter. Such test assemblies could provide fairly realistic simulation of many fusion conditions, with complete coolant and purge flow systems and instrumentation.

2.2.3. Modelling needs

The experiments should be supported by a strong programme of model development in order to understand the test results and to improve predictions of the blanket behaviour, 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, modelling of solid breeder blankets generally emphasizes a mechanistic or semi-empirical approach. As indicated in Table VIII, 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.

2.2.4. 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 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. The test plan is represented here by a set of time-dependent objectives and a possible sequence of experiments.

In addition to providing a method of focusing 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 five to ten 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 five to ten years, especially the cost. The test plan provides an indication of the expected directions of the long term testing programme. This helps not only in the planning of future experiments, but also provides impetus and direction for the near term testing efforts.

TABLE IX. OBJECTIVES AND MILESTONES OF THE FOUR PHASES OF BLANKET TESTING

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

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

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.

2.2.4.1. Experiment objectives

A test programme for the development of solid breeder (and liquid breeder) blankets can proceed with the four overlapping phases outlined in Table IX. 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 phasing is important due to our present inability to demonstrate a clearly superior blanket design, and because some of the required types of experiments vary widely between different blanket options.

In the first phase (zero to ten years), the primary goal is to explore phenomena and develop an understanding of material behaviour. 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 second phase (five to fifteen years) is to quantify local design related behaviour under fusion relevant conditions. This will allow an assessment of design configurations and design limits, and selection of 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.

The third phase (ten to twenty years) provides concept verification to the maximum extent possible in non-fusion facilities. This should be sufficient to support an assessment of the feasibility and attractiveness of 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 'early phase' tests (e.g. additional measurements of material properties).

2.2.4.2. Sequence of experiments

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. Second, the influence of geometry on the primary uncertainties is not strong. The most significant uncertainties are related to basic properties or to local behaviour (e.g. within a pellet). Third, the influence of radiation on the behaviour, in general, and the uncertainties, in

TABLE X. 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, particularly those containing both breeder and multiplier;
- 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 behaviour

- 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

particular, is strong. Radiation damage and transmutation can substantially alter the original material. Finally, much of the important functional behaviour of the solid breeder is not described by classical equations, but rather the controlling phenomena must be quantified by experiments. Therefore, the whole test programme 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 programme for solid breeder blankets. Each task consists of a number of experiments and related activities as described earlier and summarized in Table X.

Figure 7 illustrates a solid breeder test sequence that structures the experiments according to the test programme phases described in Table IX, with initial emphasis on understanding material behaviour and blanket phenomena, and a 15-year objective of concept verification in a non-fusion environment.

The development and characterization of solid breeder materials must continue since these data allow material selection which is most cost effective at this point in the programme. The completed and active irradiation experiments were summarized in Table VI. A major evaluation and selection of materials will occur over the next three to five 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 subsequent experiments.

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.

The assessment of tritium self-sufficiency should also continue through the next phase of the US/JAERI Fusion Breeder Neutronics Collaborative Program. Both this task and the development of solid breeder materials involve series of experiments in existing facilities.

In the near future (~1987), additional tasks must be started with respect to multiplier development and basic breeder thermal behaviour in order to support material selections in about five years. There is presently very little experimental activity with respect to multiplier development. 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

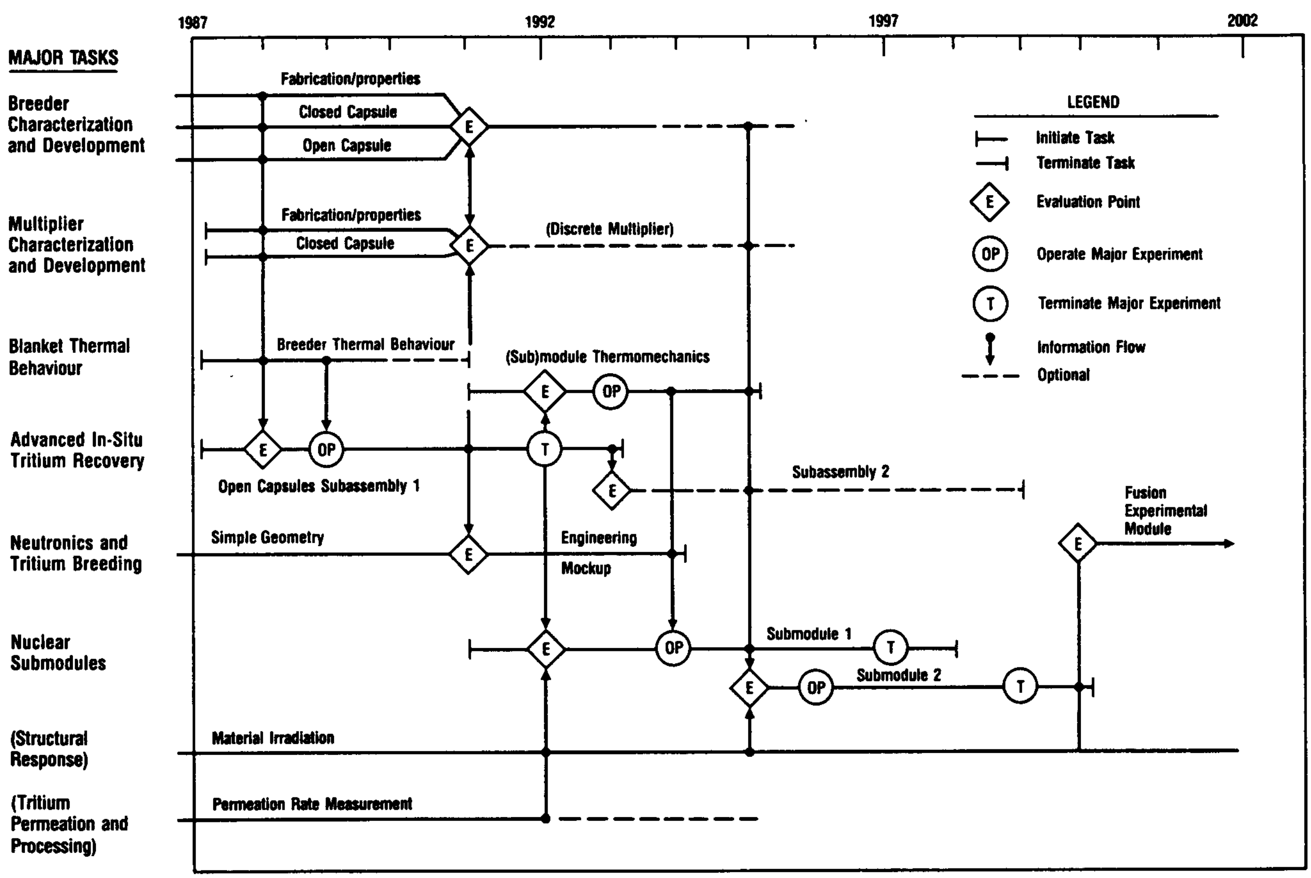


FIG. 7. Test sequence for major solid breeder blanket tasks.

TABLE XI. REPRESENTATIVE COSTS OF MAJOR SOLID BREEDER TASKS[a]

Task	Capital cost (M$)	Operating cost (M$$\cdot a^{-1}$)	Duration (years)	Total cost (M$)
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 behaviour				
A. Breeder thermal behaviour	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.

frame (five 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 five years into the programme, 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 (five to seven years into the programme), 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 programme on preferred structural alloys and

TABLE XII. 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

TABLE XIII. EFFECT OF COOLANT, BREEDER, AND STRUCTURAL MATERIAL CHOICES ON THE 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
Water	tritium containment and safety
Structural material	
Vanadium alloys	bimetallic mass transfer
	DBTT[a]
Ferritic/martensitic steels	DBTT[a]

[a] Ductile-to-Brittle Transition Temperature (changes due to impurities, radiation, H, and He).

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

Overall, this test plan should lead to a reasonable solid breeder blanket data base for a variety of materials and allow opportunity for timely innovation and design changes. It includes testing under all reasonable non-fusion conditions with substantial reactor relevant conditions in many tests. The estimated total cost of this solid breeder blanket test plan is about 10–20 million US $ per year. Representative costs for the tasks are given in Table XI. These tests would be expected to support model development and allow calibration of design codes. Assuming that solid breeder blankets are sufficiently attractive, the programme would then be able to confidently proceed with the design of a blanket experimental module for a fusion test device.

2.3. Liquid breeder blankets

2.3.1. Issues and testing needs

A number of large uncertainties exist in the behaviour of liquid breeder blankets, leading to uncertainties in their feasibility and attractiveness. Generic issues have been defined to encompass the most promising blanket designs. These issues are listed in Table XII and are discussed below. Issues relating to safety and/or transient effects are not listed separately, as they are considered an integral part of each issue.

Liquid breeder blankets encompass a variety of generic design variations and material combinations. Current designs include a number of self-cooled and separately cooled options with widely varying geometries (see Fig. 4). Table XIII indicates the effect of material choices on the dominant near term issues. The existence and seriousness of the major issues depend not only on the particular blanket concept, but also on the operating conditions such as power density, magnetic field, surface heat flux, temperature and duct length.

2.3.1.1. MHD effects

Some of the largest uncertainties in self-cooled liquid metal blankets relate to magnetohydrodynamic (MHD) effects. Many aspects of blanket behaviour, including pressure drop, heat transfer, mass transfer, and structural behaviour are impacted through the effects of the magnetic field on fluid flow. The existing theory of the flow of conducting liquids in strong magnetic fields has established some general features of the flow [38–40], 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. In separately cooled blanket designs, the MHD pressure drop can be greatly reduced. However, the liquid breeder must still be circulated for tritium removal, in some cases (e.g. LiPb) at speeds approaching that of self-cooled blankets. Prediction of MHD velocity profiles and heat transfer characteristics will be needed in virtually all liquid metal designs.

Because of the influence of the magnetic field on the velocity profiles within the blanket, 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 an important supplement to fluid flow tests. The engineering scaling requirements for heat transfer testing include all of those for fluid flow testing, as well as several additional ones [2], in particular (for most blanket designs):

(1) correct velocity distributions;
(2) suppression of turbulence;
(3) suppression of natural convection;
(4) similar flow entrance length;
(5) similar thermal entrance length;
(6) negligible axial conduction;
(7) negligible bulk heating in the first wall.

Some of these are automatically satisfied if the fluid flow requirements are met.

Electrically insulated coating and laminates have been suggested as a possible means of reducing the MHD pressure drop and thereby alleviating the problems with blanket cooling and pressure stresses. The effectiveness and survivability of MHD insulators must be demonstrated. The most serious lifetime concerns relate to compatibility with the coolant and effects of irradiation. If insulators are shown to be feasible, then MHD experiments must be appropriately modified to account for the new designs which emerge.

2.3.1.2. Material interactions

Material compatibility is a dominant concern for nearly every liquid breeder blanket design; however, the nature and importance of the issues depend strongly on the materials. There are a large number of phenomena relating to material interactions, including both mass transfer and structural degradation [41]. Table XIV shows the most important material interaction issues.

Compared to heat transfer and fluid flow, additional environmental conditions can be critically important, such as materials, impurity levels, absolute temperature, temperature gradient, out-of-blanket geometry, and long term exposure. Because of the complexity and material dependence, general models for predicting material interaction phenomena will likely be deficient.

TABLE XIV. MATERIAL INTERACTION ISSUES

A. Mass transport

1. Wall erosion from dissolution and spalling
2. Erosion/redeposition phenomena, including tube plugging and activation product transport

B. Structural property degradation

1. Liquid metal embrittlement
2. Removal of alloying elements
3. Embrittlement and surface chemistry changes due to interstitial element transport
4. Stress-corrosion interactions
5. Radiation-corrosion interactions

Thus, experiments are needed to develop empirical correlations for the behaviour under fusion relevant conditions.

To confidently design blankets, temperature and impurity limits must be adequately established. Impurity control techniques should also be explored. While a number of methods to control corrosion and mass transport have been proposed (inhibitors, coatings, getters, etc.), further studies are required to indicate the likelihood of success and the limits of applicability.

2.3.1.3. Tritium recovery and control

Tritium issues include two major categories: permeation rates and extraction techniques. These two subissues are related since the type of extraction system will be matched to the limits on the tritium release rate.

The determination of permeation rates and pressure dependence will provide the design goal for the tritium recovery system. 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 to linear dependence. The partial pressure at which this change occurs 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.

Tritium extraction issues vary widely for different breeding materials and also for different recovery schemes. The three candidate liquid breeding materials are lithium, 17Li-83Pb (LiPb), and LiF/BeF_2

salt (Flibe). For lithium, the tritium solubility is high and the partial pressure relatively low. Partial processing of the coolant stream is feasible while still maintaining acceptable tritium permeation rates. An acceptable extraction scheme has been proposed for lithium (molten salt extraction [42]), with laboratory scale experimental verification available. Other potentially viable techniques, such as permeation windows and gettering, have been proposed and are under study [43, 44]. The largest remaining issue is to maintain the total tritium inventory in the coolant within reasonable limits.

For LiPb and Flibe the tritium solubility is low and the partial pressures are extremely high. To prevent large quantities of tritium from escaping from 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. This is further complicated by a general lack of tritium related data on LiPb and Flibe. Before developing extraction systems for LiPb and Flibe, it will be necessary to obtain better measurements of fundamental properties (solubility and diffusivity), and to characterize the permeation behaviour.

2.3.1.4. Structural response

Structural issues involve uncertainties in both the loading conditions and the response to those loads. The principal loading conditions include stresses caused by pressure, thermal gradients, steady state or transient electromagnetic forces, and neutron induced swelling. Many of the major issues relating to structural loading conditions are dominated by MHD effects, which should be addressed in MHD experiments.

The structural response to the loads is dominated by material behaviour under irradiation. Some of these issues can be partially addressed in small, subscale test elements placed in fission reactors and other available neutron sources. However, 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.

2.3.1.5. Tritium breeding

Tritium breeding is not usually considered a feasibility issue for self-cooled liquid breeder blankets. However, in separately cooled designs, the breeding margin is much smaller and tritium breeding can be a major issue. Also, some reactor designs may have only partial blanket coverage (such as not breeding at the tokamak inboard side).

The uncertainties in tritium breeding can be reduced through a programme of basic nuclear data measurements, integral neutronics experiments, and improvement of calculational methods and codes. These experiments are substantially the same as those described in Section 2.2 for solid breeder blankets.

2.3.2. Existing and required experiments and facilities

Through examination of the key issues and test requirements, a complete matrix of needed experiments and test facilities has been identified. Figure 8 shows this matrix of tests for liquid breeder blankets, including some experiments which are already in progress. (Existing experiments in technical disciplines relevant to liquid breeder blanket issues are summarized in Table XV.) 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, 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 programme designed around these major experiments may include a number of smaller experiments not specifically listed in the figure, such as instrumentation development. A complementary theory and model development programme will also be required. More details on the test plan are provided in Section 2.3.4.

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 9 shows the parameter ranges covered by existing data: those accessible in operating experiments, and those expected under actual reactor conditions [45–48]. Only very recently have experiments begun to approach values of M and N close to those found under actual reactor conditions. Information expected to come from ALEX in the USA and MALICE in Belgium includes single channel

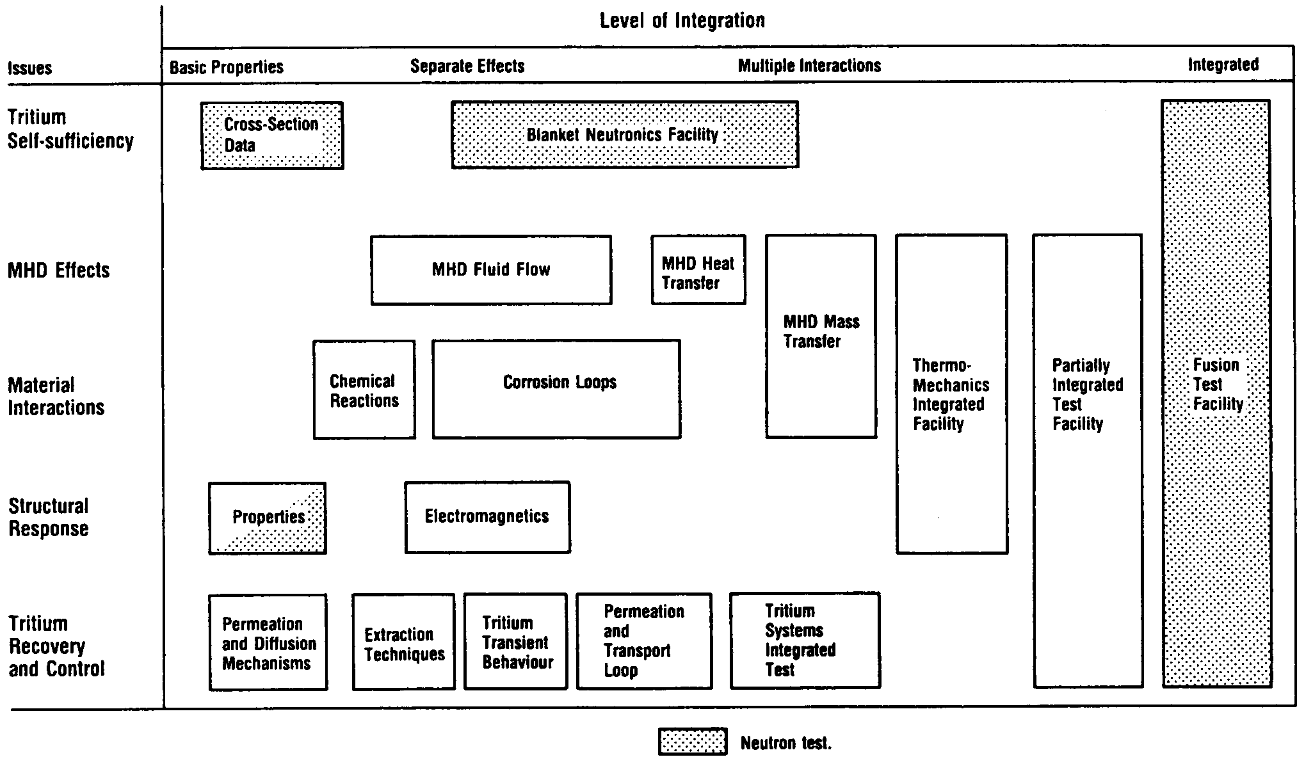


FIG. 8. Types of experiments and facilities for liquid breeder blankets (some experiments and/or facilities already exist).

TABLE XV. SUMMARY OF MAJOR EXISTING TEST FACILITIES FOR LIQUID BREEDER BLANKET RESEARCH

LIQUID METAL MHD		
Location	*Field strength (T)*	*Volume (m × m × m)*
ANL, USA (ALEX)	2	1.83 × 0.76 × 0.15
Mol, Belgium (MALICE)	2	1.0 × 0.1 × 0.1
Inst. Physics (Riga, USSR)	2	0.25 × 0.25 diameter
LIQUID METAL CORROSION		
Location	*Materials*	*Loop type*
ANL (USA)	Li/304SS, LiPb/304SS, LiPb/carbon steel	Forced convection
ORNL (USA)	Li, LiPb/various steels, LiPb/316SS	Thermal convection
Mol (Belgium)	Li/316SS, Li/Fe-9Cr	Thermal and forced convection
CEA (France)	LiPb/316SS	Thermal convection
Kurchatov, Efremov (USSR)	Li/316SS, Li/20Cr-45Ni, Li/Fe-13Cr	Forced convection

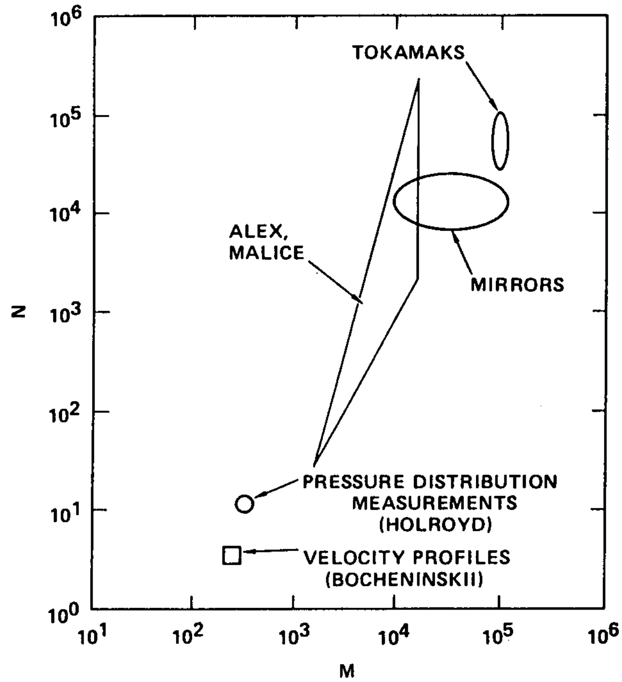


FIG. 9. Hartmann number (M) and interaction parameter (N) ranges for existing data and reactor conditions.

pressure drops and velocity profiles in straight channels, bends, and magnetic field entrance regions. Velocity profile instrumentation is also being developed. The ability to accurately measure velocity profiles at high magnetic field will have a large impact on the remainder of the MHD test program. The data provided by ALEX and similar facilities will be essential in order to assess the feasibility of self-cooled liquid breeder blankets and also provide a basis for the design of advanced concepts. Depending on developments in reactor and blanket designs, more experimental data may be needed at higher values of M and N, as well as for geometries more representative of actual blanket configurations.

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 major facilities are also listed in Table XV [2, 49].

Existing tritium recovery experiments are fewer in number. While several potential extraction schemes have been proposed, experimental verification is generally lacking. In some cases (primarily LiPb), uncertainties in basic data even restrict our ability to perform design and scoping studies.

While existing test facilities have begun to address critical liquid breeder blanket issues, there is need for a number of new facilities. These are classified according to the principal issues addressed. A separate class is considered for partially integrated tests, which provide simultaneous testing of several different issues.

2.3.2.1. MHD experiments

A range of experiments have been explored to fulfil the need for further testing of MHD related effects. The base programme proposed for MHD fluid flow and heat transfer consists of two flow loops plus complementary tests to develop instrumentation and MHD insulators. Table XVI shows the relationship between the major facilities by specifying the principal features and objectives of the experiments. Also included in the table are an MHD mass transfer facility and two partially integrated test facilities (TMIF and PITF), which are considered later in this section. Table XVII indicates approximate ranges for the major experimental parameters.

Beyond the ALEX class of facility, experimentation on MHD effects should progress to more complex geometries and conditions closer to the fusion reactor environment. This is particularly important in order to develop the ability to predict fluid flow, heat transfer, and pressure drop behaviour in 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 behaviour, especially the velocity profiles, in basic elements of relevant geometries. 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. 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.

In addition to 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.

TABLE XVI. 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 • NaK	• 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)

TABLE XVII. CHARACTERISTICS OF MAJOR LIQUID BREEDER EXPERIMENTS

Characteristic	ALEX[a]	Magnetic transport phenomena facilities		TMIF[d]	PITF[e]
		LMF[b]	MHDM[c]		
Fluid	NaK (100°C)	NaK	actual materials	actual materials	actual materials
Testing volume (m × m × m)	1.83 × 0.76 × 0.15 (0.21 m^3)	3 × 1 × 0.5 (1.5 m^3)		3 × 1 × 0.5	3 × 1 × 0.5
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.

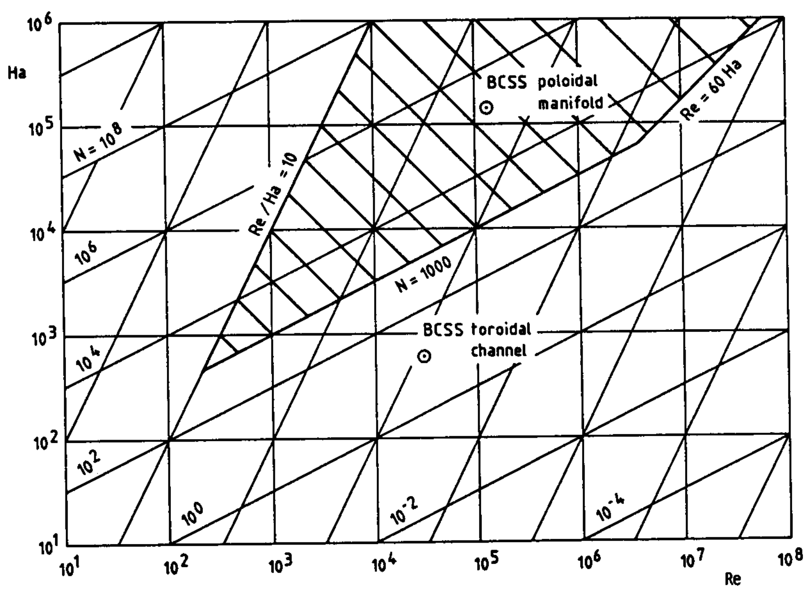


FIG. 10. Required parameter ranges of the Hartmann number (Ha), Reynolds number (Re), and interaction parameter (N) for MHD fluid flow experiments.

This secondary mission will likely follow a period of two to four years of basic velocity profile measurements.

LMF1 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 behaviour indicates that certain ranges of the Hartmann (Ha) and Reynolds (Re) numbers must be provided in order to maintain similar fluid flow behaviour. As an example, Figure 10 indicates the minimum acceptable operating region for the Hartmann and Reynolds numbers which provide fluid flow behaviour similar to that expected to occur in the poloidal manifolds of the reference BCSS design [19]. 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. Also, other blanket designs may require different parameters for similarity.

From the figure, it can be concluded that the Hartmann number must be maintained higher than 10^3 or 10^4. This can be achieved in small channels

TABLE XVIII. PRELIMINARY PARAMETER RANGES FOR THE ADVANCED LIQUID METAL FLOW FACILITIES (LMF1 AND LMF2)

Magnetic field strength (T)	4–6
Field volume	3 m × 1 m × 0.5 m
Coolant velocity ($m\cdot s^{-1}$) (in the test section)	0.05–0.1
Pipe diameter in loop (cm)	10
Coolant temperature rise (°C)	100–200
Total heat input (MW)	10
Approximate total cost (million US $)	10–15

(1 to 5 cm) if the magnetic field strength is above 1 to 2 T. Ideally, the Hartmann number and interaction parameter should match reactor values in order to provide correct non-dimensional scaling. Other concerns also suggest a high magnetic field strength for relevant MHD behaviour. For example, if ferromagnetic materials are utilized, they must be fully saturated. In addition, experimental results indicate that it might be possible to generate 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 LMF1 facility is expandability. After simple fluid flow experiments, 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 XVIII. The experiments require 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 envisaged that several different flow loops will be inserted into the test volume, making this a 'user facility'.

While LMF1 will focus on developing predictive capability by concentrating on measurement and understanding of 'microscopic' parameters such as velocity profiles, it is suggested that a series of experiments should also be performed with a greater emphasis on 'macroscopic' parameters such as the pressure drop and heat transfer coefficients. Although, in principle, these experiments can be conducted in LMF1, practical considerations suggest that another facility, called LMF2, would be devoted to this purpose. Macroscopic measurements would serve as a check on the validity of the velocity profile models and measurements, and also provide a backup 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.

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 an empirical database 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.

The operation of *two* facilities, LMF1 and LMF2, early in the programme 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 programme in any one country. This is a good example of an area where international co-operation can be very effective. Other alternatives to dividing the mission between the two facilities might also be considered.

The measurement of MHD velocity profiles is a key issue which will determine to a large extent the type of information, accuracy, and benefits of testing for MHD fluid flow phenomena. While a number of experiments have been performed at low field in NaK, mercury, and sodium, velocity profile measurements are expected to become more difficult at higher magnetic field, higher temperatures, and in corrosive liquids such as lithium and LiPb. 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 [50]. 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.

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 programme 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.

MHD pressure drop can be significantly reduced through the use of electrically insulating coatings or laminates. Owing to the large potential impact that electrical insulators will have on the feasibility and design of liquid metal blankets, early scoping tests should be performed to explore their potential problems and benefits. Initial efforts should be placed on determining whether or not insulated structures can meet requirements on compatibility and structural integrity under irradiation. Three kinds of scoping tests are recommended: 1) fabrication of the 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.

2.3.2.2. Material interaction experiments

Results from existing experiments on corrosion/mass transport will not provide enough information to adequately define operating limits and aid in the selection of materials. More corrosion loops will be required, especially for refractory metals and bimetallic

TABLE XIX. 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 XX. 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).

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. After studying the basic material interactions in convection loops, 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 (see Tables XVI and XVII).

Table XIX shows the relevant environmental conditions for material compatibility as compared with heat and momentum transport, and Table XX 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 it is determined that mass transfer rates are dominated by liquid phase diffusion rather than solid phase diffusion and chemical reactions.

Material compatibility experiments would ideally include at least one loop for each proposed primary coolant/structural alloy combination. The material systems currently being considered include lithium, 17Li-83Pb, and Flibe breeder/coolants, helium coolant, water coolant, and structural materials of ferritic/martensitic steel, vanadium based refractory alloy, and possible austenitic steel (although not a favoured class of alloys).

The tests with a refractory structure should be performed in a bimetallic loop since reactor primary cooling system will almost certainly not be fabricated out of vanadium alloy because of economic considerations. It is more likely that the out-of-blanket piping, pumps, heat exchangers, tritium recovery components, chemical control systems, etc. will be fabricated from some type of steel. In this case, a dominant issue will be impurity transport between materials. Establishing design limits on temperature and impurity levels will be more difficult in such a multi-component system, and more extensive impurity control systems will probably be needed as compared to simple steel systems.

Every potential structural material actually represents a class of alloys (for example, ferritic/martensitic 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 programme.

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

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 five to ten, especially in localized regions [51]. The MHD facility is proposed to explore the influence of MHD velocity effects on mass transfer.

2.3.2.3. Tritium recovery and control

Needed experiments related to tritium recovery and control cover varying levels of integration, including:

(1) basic properties and mechanisms
(2) tritium extraction techniques
(3) permeation and transport in the coolant loop
(4) integration of extraction and tritium processing systems.

For lithium, the fundamental tritium properties are fairly well known and plausible extraction systems have been proposed. However, before developing extraction systems for LiPb and Flibe, it is necessary to obtain better solubility and diffusivity data. For all breeders, it is important to improve our knowledge of the permeation mechanisms and behaviour at low partial pressure in order to set guidelines for the extraction system.

Small scale extraction systems must then be operated in order to verify their performance. Since the tritium recovery issues are completely different for different breeding materials and recovery schemes, separate verification tests are required.

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.

2.3.2.4. Tritium breeding

Blanket neutronics experiments are required 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. 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 Section 2.2.

2.3.2.5. Structural response and failure modes

The major uncertainties in structural response for liquid breeder blankets include material behaviour 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 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.

2.3.2.6. Partially integrated experiments

Beyond the first five to ten 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 five to ten years of more fundamental testing, it is difficult to anticipate the exact features of the facilities. However, certain key features and objectives have been studied.

A Thermomechanical Integration Facility (TMIF) is one particularly attractive concept. It tests the combined influence of heat, mass, and momentum transport issues, as well as some structural issues, in a non-neutron environment. The purpose of TMIF is to aid in the selection of a small number of leading configurations and to begin the development of empirical relations describing the global behaviour of the blanket. TMIF will be a larger facility than the early MHD experiments, and will use more prototypical blanket geometries. Because of the presence of a number of attached subsystems, the thermal and material environment of the blanket will be more accurately represented. These subsystems include, for example, primary cooling system elements (such as pumps and heat exchangers), chemical control systems (inhibition and impurity control), and possibly a tritium extraction system operating with injected protium.

Another type of facility, designated the Partially Integrated Test Facility (PITF), has also been defined. The PITF would be a full- or near-full-scale test which simulates all environmental conditions except neutrons. In addition to the blanket, the primary cooling system, chemical control systems and tritium extraction system (with tritium or deuterium/protium) would be present. For liquid breeder blankets, the omission of neutrons results in large cost savings, with many of the critical issues still addressed. Partially integrated experiments can provide a good simulation of the operating characteristics of a power reactor for many important parameters, such as surface heat flux, velocity, and geometry. These experiments should provide some useful information on failure modes and component reliability. 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 benefits of fusion testing.

The PITF facility has characteristics similar to the TMIF, and may be built as an upgrade of that facility.

TABLE XXI. SUMMARY OF LIQUID BREEDER BLANKET MODEL DEVELOPMENT NEEDS

MHD effects	
	Analytic modelling of velocity profiles in straight ducts and simple geometries
	Three-dimensional MHD computer codes for the solution of velocity profiles in complex geometries
	Semi-empirical design codes for prediction of fluid flow behaviour in complex three-dimensional geometries
Material interactions	
	Dissolution modelling with solid phase, liquid phase, and interface transport processes
	Modelling of the kinetics of material interaction processes
	Primary cooling system loop modelling
Tritium transport	
	Surface absorption/desorption modelling for tritium permeation through metal barriers
	Loop code to predict transport and permeation rates in an integrated primary cooling system with extraction system

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. Tables XVI and XVII show the characteristics, objectives, and main features of the partially integrated experiments as well as the other major liquid metal facilities.

2.3.3. *Modelling needs*

The development of modelling capabilities is crucial to satisfying the objectives of the test plan. One of the primary reasons for performing experiments is to aid in the development of predictive capabilities. Therefore, the testing programme should be accompanied by a strong complementary programme of model development, analysis and design work.

Table XXI summarizes the principal model development needs for liquid breeder blankets. The two areas in which modelling 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, three-dimensional geometries, such as a reactor blanket. Analytic models have been developed for some simple, specialized problems, but a general three-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 modelled 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 modelling.

2.3.4. *Test plan*

The objectives of the liquid breeder blanket test plan are similar to those of solid breeder blankets as described in Section 2.2.4. However, the experiments are quite different in several ways. For liquid breeder blankets, most of the key issues can be resolved in non–neutron test facilities. These include, for example, fluid flow, heat transfer, structural response, and material interaction test stands. Another important difference is the greater uncertainty in material behaviour for solid breeders as opposed to liquid breeder blankets.

The same phasing is used for both solid and liquid breeder blankets. The experiments in the first phase (zero to ten years) include a variety of MHD fluid flow tests, materials compatibility loops, tritium recovery and tritium breeding experiments in simple geometries. Some of the single and multiple effects experiments continue into the second phase (five to fifteen years), and more integrated facilities are initiated. One of these more integrated facilities is the TMIF (Thermomechanical Integration Facility), which explores thermal hydraulics, materials compatibility, and some structural behaviour 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 processing system interface experiment. In the third phase (ten to twenty 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.

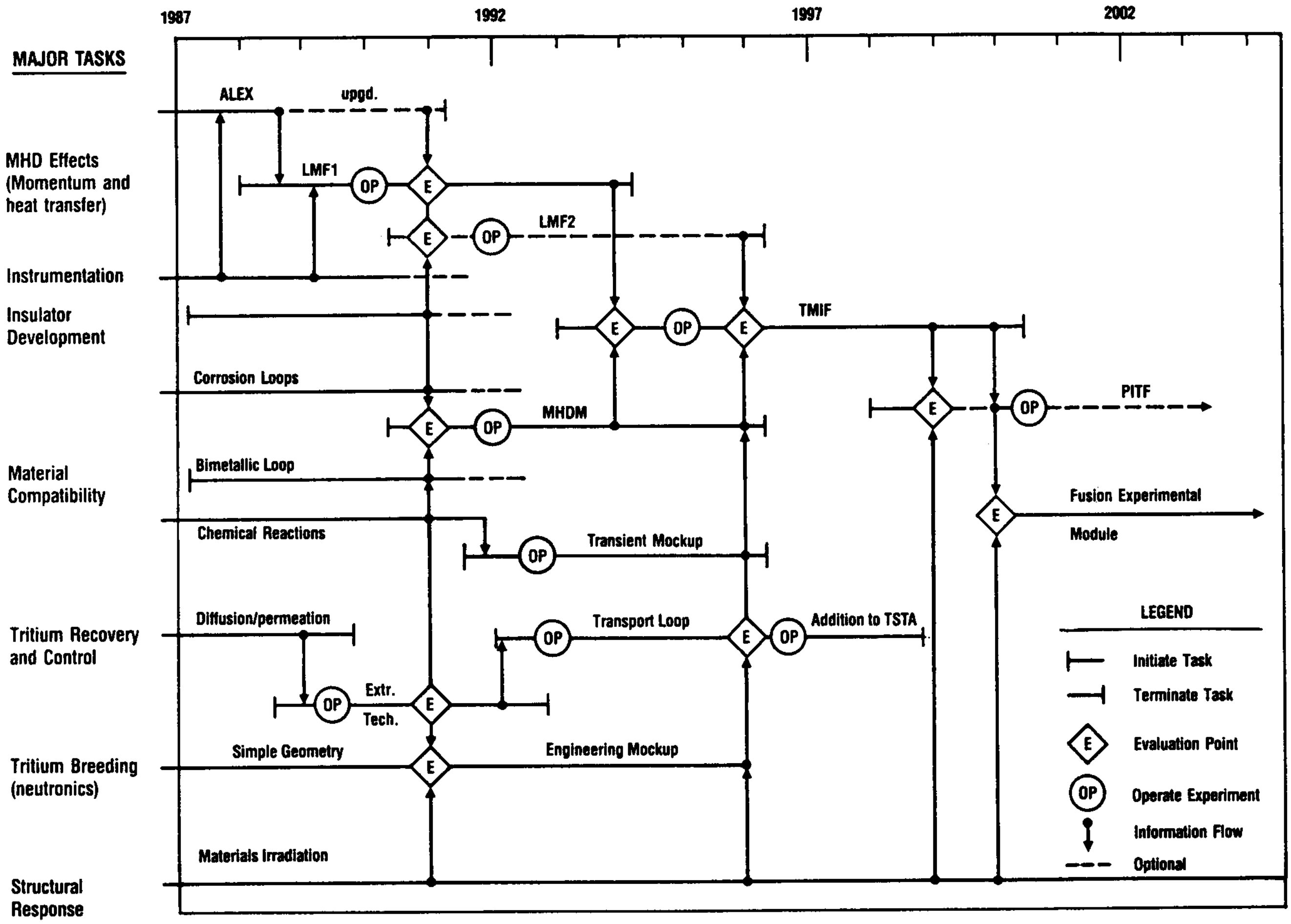


FIG. 11. Test sequence for major liquid breeder blanket tasks.

TABLE XXII. REPRESENTATIVE COSTS OF KEY LIQUID BREEDER BLANKET FACILITIES

Item	Capital cost[a] (million US \$)	Operating cost[b] (million US \$ · a^{-1})	Duration (a)	Total cost (million US \$)
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
Corrosion loops (~8)	2–4	0.8	10–15	10–16
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 US dollars.
[b] Does not include analysis of data.

2.3.4.1. Sequence of experiments

The sequence of experiments for liquid breeder blankets is constructed from the list of major facilities and experiments presented in Fig. 8. The elements in Fig. 8 can be viewed as 'building blocks' for the test plan. In Fig. 11, the major classes of facilities are shown 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.

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. Two of the important decisions will be to determine the 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.

2.3.4.2. Costs

The cost of the major facilities discussed above have been estimated to determine an approximate overall programme cost to develop liquid breeder blankets. The results are given in Table XXII. These numbers represent all of the costs associated with the experimental programme, including construction, experimental hardware, staff, overhead, etc. The costs in the table are broken down into two categories: 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 analysis, modelling efforts, blanket design studies, etc. have not been included as operating expenses. These are listed in Table XXII as a separate item. A liquid breeder blanket programme requires an average annual expenditure of about ten to twenty million US dollars.

3. TRITIUM PROCESSING AND VACUUM SYSTEMS

The tritium processing and vacuum system issues are divided into six categories: (1) fuel processing, (2) blanket tritium recovery, (3) tritium permeation, (4) room atmosphere detritiation, (5) tritium monitoring and accountability, and (6) reactor relevant system development. The specific issues and testing needs for each category are discussed below.

3.1. Fuel processing

A unique set of circumstances in tritium technology has resulted in both the need for and the ability to build a reactor scale tritium fuel processing facility ($\sim$1 kg per day tritium flow rate) relatively early in the US fusion programme schedule. Such a facility is needed owing, in part, to the anticipated D–T operating mode of TFTR [52] scheduled around 1989, and other near term machines, such as CIT [53], which will be operated with D–T fuelling. The ability to operate such a facility in this technology area is primarily due to the fact that the fuel processing system can operate independently of the neutron environment and machine parameters. There is also a large database inherited from the national security programme which made the construction of a test facility for fuel processing feasible. This facility is the Tritium System Test Assembly (TSTA) [54], shown schematically in Fig. 12.

The TSTA is designed to test a simulated D–T fusion reactor fuel processing system. The size of the facility is of the same order as is required for a commercial fusion reactor. TSTA is currently demonstrating a range of tritium monitoring, handling, and control actions that will be essential for the licensing and safe operation of experimental and commercial fusion reactors. The batch handling tritium technology needed for near term experimental devices is also being developed and demonstrated concurrently at the facility.

The elements of TSTA are shown in Fig. 12. It consists of two primary parts. The first part includes the neutral beam interface, impurity injection and torus mockup. This represents the output from the plasma exhaust and can only be simulated at this stage. The real parameters of this stage will be obtained only after the operation of a D–T fusion reactor. The second

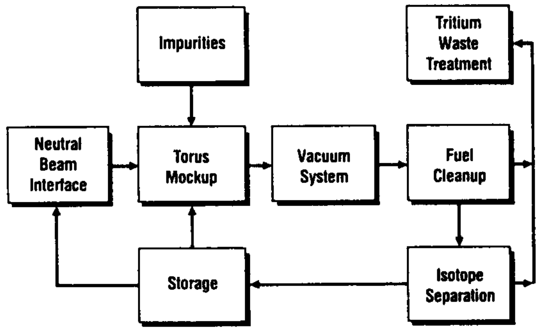


FIG. 12. TSTA main process loop and auxiliary systems.

part includes the vacuum system, fuel cleanup, isotope separation storage and tritium waste treatment. Adequate information exists to indicate how each individual step can function. The critical technical issues in tritium processing tend to be issues dealing either with integration of tritium systems or with the interfaces between tritium and other systems. Another goal for TSTA is to test both components and systems for their reliability and availability under semi-relevant conditions. Therefore, the TSTA is a 'partially integrated test facility' for the fuel process system.

3.2. Blanket tritium recovery

The blanket tritium recovery system affects both the blanket system design and the tritium process system. The tritium inventory, tritium partial pressure and tritium recovery loop requirements have a major impact on blanket neutronics, tritium containment and blanket mechanical design. On the other hand, the condition of the tritium carrier stream, such as flow rate, impurity level, pressure, etc. has a major effect on the tritium processing system. Issues of the breeder tritium extraction system can be categorized 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 tritium from water may also be important if it is present.

It appears that no feasibility issue exists for blanket tritium recovery systems. The key problem is the trade-off among blanket tritium inventory ($\leqslant$ order of kg), tritium leakage ($\leqslant$ order of 10 $\mathrm{Ci \cdot d^{-1}}$), and tritium recovery system cost ($\leqslant$ order of 10 million US \$). Another potential issue is the possible impact on blanket design, such as complexity caused by the requirement of purge gas for solid breeders; or the impact on Li blankets caused by the salt carried back from the molten salt extractor. Even for the same breeding material, the critical issue can be completely different for different recovery methods.

Possible process flow schematics and processing methods for each case are summarized in Fig. 13 and Table XXIII [43, 55, 56]. The key experimental parameters for studying tritium extraction from each of the carrier fluids are summarized in Table XXIV [23]. Experiments are needed to explore the feasibility of tritium recovery from the three primary carrier fluids under these conditions, and to evaluate the operating characteristics (including reliability and tritium inventory) of the processing systems. These experiments, with few exceptions, do not require neutrons. The experiments are laid out in greater detail in Fig. 14.

The less complex experiments indicated in Fig. 14 can be done in glove boxes, with relatively modest costs required to provide tritium handling capability (10 000–100 000 US \$ additional). The more complex and integrated experiments require increasingly elaborate facilities. At some later stage, interfacing must be done between the breeder extraction and the fuel reprocessing systems.

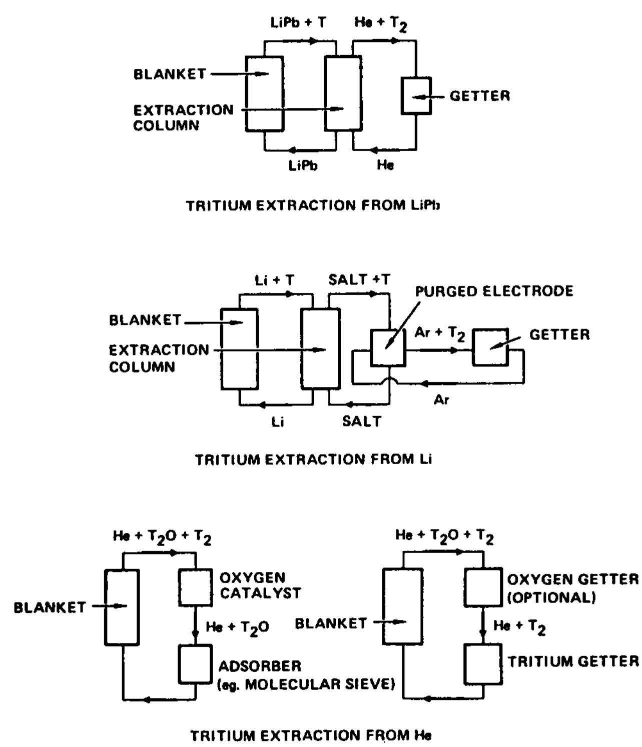


FIG. 13. Schematic representation of tritium processing schemes.

TABLE XXIII. TRITIUM PROCESSING METHODS FOR DIFFERENT TRITIUM CARRIER FLUIDS

T carrier fluid	Tritium form		Extraction method[a]	Outlet T concentration[b] X_p (appm)	Other application of method
	T_2/HT	T_2O/HTO			
Li	X		*Extraction with molten salts*	7	None
	X		Absorption with solid getters	–	
LiPb	X X		*Extraction with countercurrent He flow*	0.01–1	None
			Vacuum degassing	–	
	X		Permeation combined with catalytic oxidation	–	
He	X	X[c]	*Absorption with solid getters*	10^{-5}	Fuel cleanup
	X[d]	X	*Adsorption with molecular sieves*	10^{-5}	Air detritiation
	X[d]	X	Freezing out in cold traps	–	
H_2O		X	Vapour phase catalytic exchange	0.6	CANDU reactor coolant cleanup
		X	Liquid phase catalytic exchange	–	
		X	Electrolysis	–	

[a] Preferred method italicized.

[b] Tritium concentration at processing system outlet. This value is very dependent on design and cost trade-offs. Values given are from various design studies (LiPb), experiments (Li,He), and the CANDU Darlington TRF design (H_2O).

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

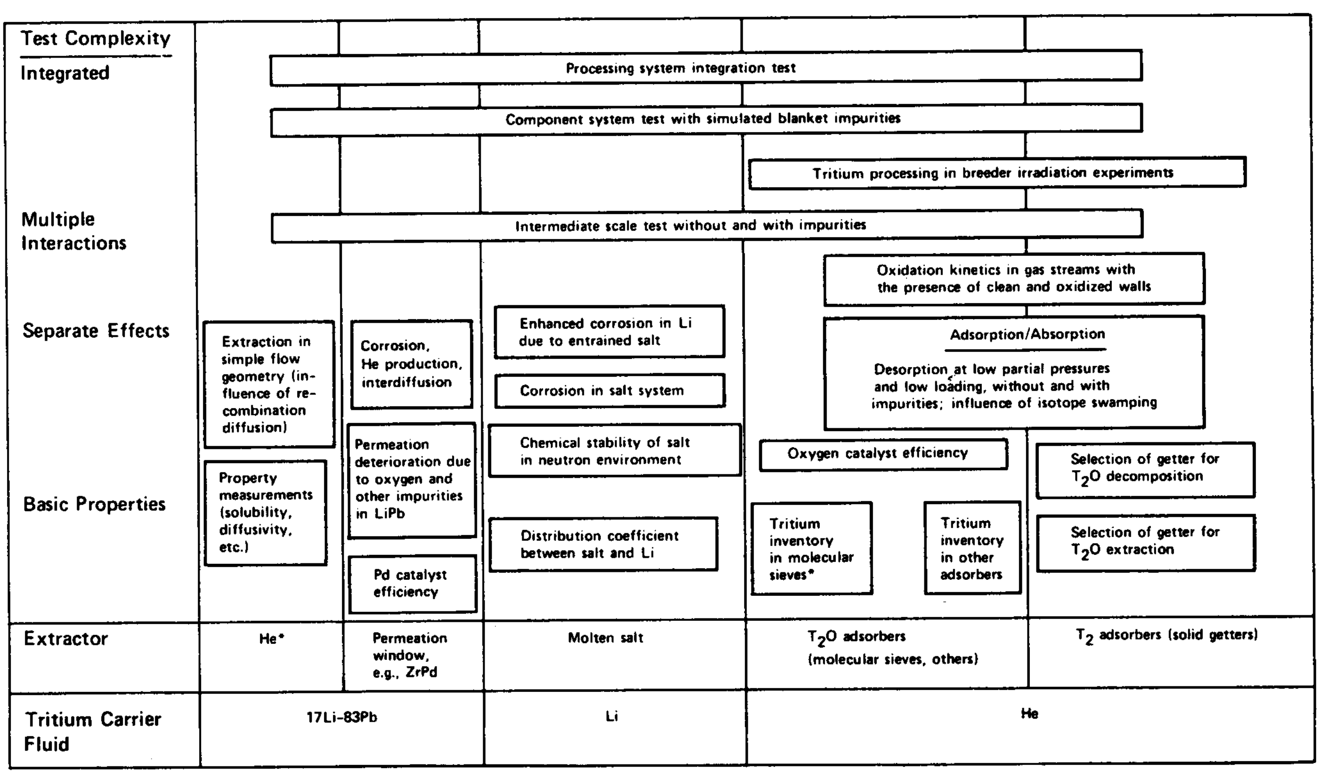
[d] Additional process needed to oxidize T_2, HT.

TABLE XXIV. 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)	<1	0	0	0–10
Impurity level (appm)	<10	>10	<10	<10
Temperature, 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.



*Preferred solution; feasibility has to be proven

FIG. 14. Breeder tritium extraction experiments.

TABLE XXV. 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^{-1})	200–300	100–300
Pulse length (s)	10^2–∞	10^2–∞
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.

3.3. Tritium permeation

There are two types of permeation which may cause different problems, i.e. plasma driven permeation [57] and pressure driven permeation [58]. The driving force for plasma driven permeation is the energetic particles striking in-vessel components. Regions in which plasma driven permeation is important include the first wall and impurity control systems. The driving force for pressure driven permeation is the tritiated gas pressure. The key region in which pressure driven permeation is important is the primary heat exchanger. The conditions under which tritium permeation must be understood are summarized in Table XXV [58].

There are two issues associated with plasma driven permeation: tritium leakage and tritium inventory. Tritium leakage is usually not a serious problem if the coolant is also the breeding material (or if the coolant is He), because the amount of tritium from the plasma is usually small. However, if water is used as a coolant, such as in impurity control systems or first walls for high wall loading machines, tritium recovery from the water may prove to be costly [59]. Tritium inventory may be a serious problem if the structure is at relatively low temperature (typically, about half the melting temperature). Radiation damage in the structural material may create trapping sites which may lead to a large tritium inventory [60].

Pressure driven permeation will cause problems with tritium containment. The key region in which this will be a problem is the main heat exchanger, where the requirement of good heat transfer also results in high mass transfer [61].

The key problem in the resolution of permeation issues is to identify the 'critical' pressure of the primary coolant, which is a function of the heat exchanger design. The blanket tritium recovery system has to be designed to achieve this critical pressure, as well as other design criteria, such as acceptable cost. The major uncertainty of defining the critical pressure is the permeation pressure relationship at low tritium partial pressure. It is an accepted theory that permeation changes from pressure square root dependence to pressure linear dependence as the tritium partial pressure is reduced [62]. It is unknown, however, at what tritium partial pressure this changeover occurs. Uncertainties also exist in the effect of surface conditions on the permeation pressure relationship. A recent measurement reported that pressure square root dependence was still obeyed at tritium partial pressures as low as 10^{-7} Pa [63] – far lower than predicted. The surface condition and the isotopic concentration [61, 64] will all affect this relationship. The effect of surface condition is also important to plasma driven permeation, since it changes the recombination constant [59].

Another important issue to be resolved is the relationship of the chemical form of tritium to the permeation. It has generally been assumed that tritium in the oxide form (T_2O) will not permeate. This is an important assumption for helium cooled designs [65]. Some recent experimental results show that tritium permeates equally whether it is in the elemental form or oxide form [66], according to the reaction:

$$T_2O + M \rightarrow MO + 2T$$

in which M is the metallic component of the structure.

The permeation/diffusion mechanism is an important issue to be resolved in the early stage of fusion development. The understanding of the permeation rate dependence on partial pressure, isotopic ratio, tritium chemical form and wall surface conditions will determine the feasibility of the tritium recovery process.

3.4. Room atmosphere detritiation system (RADS)

A room atmosphere detritiation system is required to reduce tritium leakage from the nuclear system during normal operation as well as during an accident. The detritiation capacity required depends on reactor design, building design, reactor maintenance procedure, room entry requirements, tritium oxidation kinetics, and mass transfer rate between the room air and the building wall [8]. Related facilities exist for the CANDU reactors. The facilities contain air drying beds, but no catalytic oxidation step. Catalytic oxidation and air drying are the two steps needed in room air detritiation for fusion. Nevertheless, useful information can be extracted from the CANDU operating [67] experience.

3.5. Tritium monitoring and accountability

Both environmental and regulatory requirements dictate the need for tritium monitoring and accounting. One method for accomplishing global monitoring is through the use of an overall mass balance method, but this may not be sufficient. Monitoring in individual components will be very difficult due to the high mobility of tritium, especially in the high temperature environment, and due to the inaccessibility of a large number of components, such as the first wall.

Substantial progress has already been made toward resolving this issue [68]. However, most of the progress has been made in the measurement of tritium concentration in gases or liquids for environmental and safety purposes. No method has been developed for on-line measuring of tritium concentration within the solid structure for in-reactor components. This on-line (or 'near real time') accounting is much more difficult than that for fission fuel processing [69] due to the much wider dispersal of fusion fuel (tritium) over all the reactor components. The uncertain regulatory aspect of this problem makes its resolution more difficult. Depending on eventual regulatory requirements, the problem could be of medium or of critical concern.

TSTA is an excellent facility to develop on-line accounting methods. It is unlikely, however, that comprehensive on-line accounting can ever be adequate for a fusion facility. The most promising method is to combine global monitoring, together with computer modelling (a tritium system code), to estimate the mass balance over individual components [22].

3.6. Reactor relevant system development

Reactor system development includes both large component development as well as tritium system integration. At this stage, all the relevant components have been tested separately. The purpose of this task is engineering testing.

Large component development is needed to assess operating lifetime for fuel processing components such as plasma chamber vacuum valves and plasma exhaust cryopumps. Vacuum valves for this service must be large and tritium compatible. Such valves that are capable of maintaining their performance over several years with repeated cycling do not now exist. The possibility of particulate formation due to sputtered material in the plasma exhaust further complicates valve design. Questions regarding the compound cryopumps under development for exhausting the plasma chamber include the effect of tritium and the number and type of regeneration cycles on the adsorbents (molecular sieve or charcoal) and adsorbent binders used in these pumps. The particulates in the exhaust may also have important impacts on pump lifetime.

Before commissioning a fusion device, the entire tritium processing system will have to be combined. The components included are the fuel processing unit, blanket processing unit, water processing unit, as well as the room atmosphere detritiation system. System integration is the last stage of testing before the fusion device to ensure reliable operation of its tritium systems.

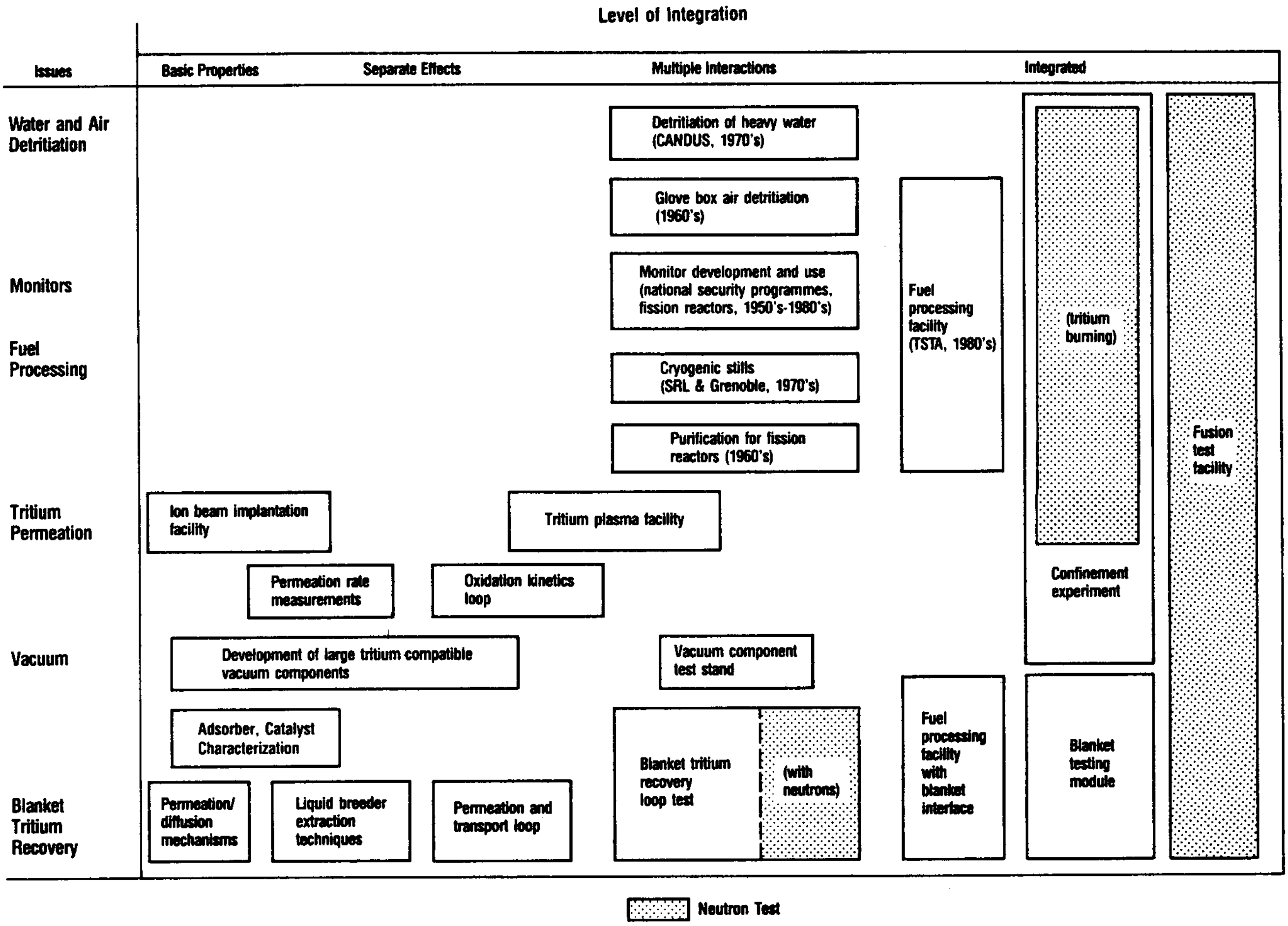


FIG. 15. Types of experiments and facilities for tritium processing and vacuum systems (some experiments and/or facilities already exist).

3.7. Experiments and facilities in tritium and vacuum

Figure 15 summarizes the types of experiments and facilities needed to resolve the technical issues outlined in the tritium and vacuum area. (Note that the axes are reversed as compared with Fig. 14). Some experiments or facilities already exist, these being primarily in the processing and permeation areas. In processing, monitors and detritiation, existing work is indicated by dates. The vacuum work has not been started for the most part. The work in blanket tritium recovery is shown in greater detail in Fig. 14.

Three substantial new (non-existing) facilities (other than fusion devices) are indicated in Fig. 15. All of these are relatively low cost, each one being less than 5 million US $ to construct.

The three facilities are a vacuum test stand, a blanket tritium recovery loop, and a fuel processing facility with blanket interface. The vacuum stand should include tritium handling capability and plasma particulate simulation. The recovery loop also requires tritium handling capability. The last – the blanket/processing interface – could be an add-on at TSTA approximately in the mid-1990s.

4. PLASMA INTERACTIVE COMPONENTS

4.1. Introduction and issues

The term plasma interactive component (PIC) is generally used to refer to any component inside of the vacuum vessel that is in direct contact with the plasma. Examples of such components include pumped limiters, divertor plates, RF antennas and their associated shields, wall armour for neutral beam dumps, and the entire inner surface of the vacuum vessel. Each of these components is installed on a confinement device to accomplish a specific purpose, such as plasma heating or particle exhaust. To perform these tasks on near term devices, PICs must operate under very high heat flux loads. In the future, a reactor environment will also include large charged particle and energetic neutron fluences. The presence of these components in the machine influences the character of the boundary layer and through such processes as neutral recycle in the edge, hydrogen permeation and release, and impurity generation and transport, ultimately influences the central plasma particle and energy confinement times.

The objectives of the PIC technology development are: (1) to support the needs of confinement testing in the near term, and (2) to develop components for eventual use in fusion reactor systems. At the same time, there are two scientific objectives that must be pursued to achieve these technology goals. These are: (1) to develop a database of fundamental information on the materials that may be used and the processes that take place in the vicinity of plasma interactive components, and (2) to investigate the complex coupling that occurs between these fundamental processes and its effects on PIC operation. Considerable technology development will be required to ensure that PICs are capable of meeting the demands imposed by the experimental programme in confinement physics and eventually by the harsh edge plasma environment of a fusion reactor. The specific issues requiring attention in the PIC area are: (1) active cooling under high heat flux operation, (2) erosion/redeposition processes, (3) tritium permeation and retention, (4) disruption survivability, (5) helium trapping and exhaust, and (6) irradiation effects. Individually, each of these issues presents serious technological difficulties for PICs, but the complexity of the problems is greatly compounded by the synergism that exists between these processes. Each of these issues is discussed below and the test programmes in each area reviewed. While some testing in each of these areas on a single effects basis is ongoing, multiple effects tests are expensive, and no activity has been initiated at this level of complexity. Upon completion of this review a generic test plan is presented which outlines the testing needs required to ensure the availability of reliable PICs.

4.2. Parameter ranges for PIC testing

Development of an experimental test plan for PICs is complicated by the requirement to 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 application. The testing environment is most easily defined by reviewing the machine parameters for confinement experiments now operating and for those planned for the future. Machine parameters for large tokamaks operational at present and for those planned for the future are summarized in Table XXVI [70–79]. These machine parameters define the operating regimes anticipated for PICs, as shown in Table XXVII, which is subdivided into parameters for: (1) present machines, (2) near term machines, (3) those scheduled for operation in the early 1990s through 2000, and (4) those planned for 2000 and beyond. This evolution of confinement parameters places specific requirements on PICs. For example, problems in high heat flux obviously require immediate attention. 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, which includes JET, JT-60 ASDEX-U, and Tore Supra. While heat fluxes in near term confinement experiments will be quite high (2–4 $kW \cdot cm^{-2}$), detailed investigation of synergistic effects will not be possible. Thus, heat removal techniques and materials devised for use in these machines can only be considered as possible solutions for long pulse, burning core experiments or reactor applications. Further reference will be made to the parameter ranges listed in Table XXVII in connection with the discussion of specific issues for PICs presented below.

4.3. Candidate materials for PICs

To date, graphite, beryllium, and titanium carbide coated graphite have been primarily used in tokamak PIC applications. Each has problems which may prohibit its use in reactor systems. One notable problem for graphite is irradiation swelling and subsequent tritium retention in the porous material. But in the near term, advanced graphite structures, such as pyrolitic graphite and multidirectional woven graphite matrices, are very attractive for high heat flux or high strength applications [80–82]. For beryllium, microcracking due to thermal cycling, surface melting and neutron induced helium production are serious problems, but beneficial effects of Be gettering have been observed on ISX [83] and Be limiters are being fabricated for use in JET [71]. TiC coatings have performed satisfactorily on experiments such as ALT-I [80] but impurity generation through sputtering may limit their usefulness. Alternative plasma side materials include B, BC, SiC, BeO and some higher Z options such as W and Ta. The development of plasma spray technology

TABLE XXVI. LARGE TOKAMAK MACHINE PARAMETERS [70–79]

Device	TFTR	JET	JT-60	T-15	Doublet III-D	ATF	Tore Supra	ASDEX-U	CIT	NET	FER
Country	USA	EC	Japan	USSR	USA	USA	France	FRG	USA	EC	Japan
First plasma	Dec. 82	Jun. 83	Apr. 85	Late 85	Jan. 86	87	Mid 87	Late 87	92	92	95
Major radius (m)	2.55	2.96	3.0	2.4	1.67	2.1	2.42	1.65	1.22	6.4	5.5
Minor radius (m)	0.85	1.25	0.95	0.70	0.67	0.30	0.70	0.50	0.45	1.64	1.1
Elongation	1.0	1.0–1.6	1.0	1.0	1.0–2.0	–	1.0	1.6	1.8		1.5
Toroidal field (T)	5.2	3.5	4.5	3.5–4.5	2.2	2.0	4.5	4.0	10.4	5.7	5.7
TF coil conductor	Cu	Cu	Cu	Nb_3Sn	Cu	Cu	NbTi	Cu	Cu		Nb_3Sn/NbTi
Working gas	H/D–T(?)	H/D–T	H	H	H	H/D	H/D	H/D	D–T	D–T	D–T
Plasma current (MA)	2.5–3.0	4.8	2.7	1.4–2.0	3.5–5.0	–	1.7	2.0	9.0–10.0	10.0	5.3
Pulse length (s)	1.0–3.0	10–20	5–10	5.0	2.0–5.0	5–CW	30	10–120	3.1	200–1000	100

Injected power (WM)	38	46	30–40	5.5–8	20	4	21	12	10–20	–	30
Neutral beam	27	16	20–30	5.5–8	12–14		7				
ICRH	8	30	3		6		6–9	12	10–20		
LHRH	3		8				5–8				
ECRH				4–6	2						
Fuelling	gas puff. pellet inj.	gas puff.	gas puff.	gas puff. pellet inj.	gas puff. pellet inj.	gas puff. pellet inj.	gas puff. pellet inj.	gas puff. pellet inj.	gas puff. pellet inj.	gas puff. pellet inj.	gas puff. pellet inj.
Impurity control	gettering	wall cond.	mag. lim. wall cond.	wall cond.	pump lim. stubby div.	pump lim. nat. div.	wall cond. pump lim. ergodic div.	divertor limiter	divertor		pol. div.
Fusion power (MW)	–	–	–	–	–	–	–	–	300	1170	440
Peak wall load ($kW\cdot cm^{-2}$)	1.0–2.0	2.0	2.0	1.0–2.0	2.0	1.0–2.0	2.0–4.0	2.0–3.0	1.0	1.5	1.0

Abbreviations used for fuelling and impurity control:
Gas puffing; pellet injection; wall conditioning; magnetic limiter; pump limiter; stubby divertor; natural divertor; ergodic divertor; poloidal divertor.

TABLE XXVII. OPERATING PARAMETERS FOR PLASMA INTERACTIVE COMPONENTS

Time-scale	Present machines (0–3 years)	Near term (3–8 years)	Early 1990s	Long term (beyond 2000)
Field strength (T)	3–6	3–6	5–10	5–10
Plasma current (MA)	0.5–3.0	2.0–4.0	2.0–7.0	5.0–11.0
Injected power (MW)	5–15	15–50	15–50	15–50
Peak heat flux ($kW \cdot cm^{-2}$) (normal operation)	0.4–5.0	0.5–5.0	1.0–5.0	1.0–5.0
Pulse length (s)	1–5	5–30	30–300	CW
Number of cycles	10^5	10^5	10^6	–
Neutron wall loading ($MW \cdot m^{-2}$)	–	–	1.0–5.0	1.0–5.0
Neutron fluence (dpa)	~0	1	25	60
Fusion power (MW)	–	–	300–500	500–1200
Surface materials	C,TiC-coated C,Be	Be,BeO, SiC,C	Be,W,Ta	Be,W,Ta,V
Structural materials	Stainless steels, Ni alloys	Stainless steels, Ni alloys, Cu alloys	Cu alloys, Refractory metals	Cu alloys, Refractory alloys
Bonding/attachments	Mechanical	Mechanical, bonded, plasma spray	Bonded, plasma spray	Bonded, plasma spray
Coolant types	None	H_2O	H_2O	H_2O, liquid metals, others?
Disruption characteristics				
Current decay time (ms)	0.1–1.0	1.0–10	5.0–200	5.0–200
Thermal quench time (ms)	0.02–0.3	0.3–3.0	3.0–100	3.0–100
Peak heat flux ($kW \cdot cm^{-2}$)	80–500	500	500	500

may allow for the use of very thick (> few centimetres) surface coatings [84, 85]. Ceramic compounds that have a stable molten phase, such as BeO, BC, MgO, AlO, TiB, TiC and VC are suitable for plasma spray. Composite coatings of ceramics and metals, cermets, such as SiC/Al and SiC/Ni, have been plasma sprayed and look very promising for high heat flux applications. The physical and mechanical properties of these coatings depend strongly on the fabrication method and on the product form. Therefore, extensive testing of physical properties for these coatings will have to be performed. It should also be noted that for non-elemental and composite materials the physical properties may be largely degraded, because of surface reconfiguration during the erosion/redeposition process.

4.4. High heat flux testing

As is pointed out in Table XXVII, existing and near term confinement experiments are facing difficult problems in active cooling of high heat flux components. Of immediate interest are pump limiter front face and leading edge surfaces, divertor neutralizer plates, RF antenna shields and neutral beam dump armours. To ensure reliable operation of these components a considerable amount of high heat flux (HHF) testing is being conducted. Operating test stands include a number of electron and ion beams and the RF plasma chamber at Oak Ridge National Laboratory [84–88]. Some have been constructed specifically for HHF testing of fusion materials and components. Others have been constructed as part of the neutral beam development programme, where heat fluxes as high as $5.0\ \mathrm{kW \cdot cm^{-2}}$ have been successfully removed from water cooled, swirl tube enhanced copper tube beam dump designs for 30 s pulses [86]. These facilities could be converted to accommodate PIC testing, although major modifications would be required to incorporate adequate diagnostics.

Most notable among the facilities specifically developed for PIC testing is the Plasma Materials Test Facility (PMTF) at Sandia National Laboratories [87,88]. There, the electron beam can deliver up to 30 kW on targets with heat fluxes ranging from $0.3–30\ \mathrm{kW \cdot cm^{-2}}$. Pulse lengths range from 50 ms, for the investigation of thermal shocks, up to continuous operation. Sample areas as large as $100\ \mathrm{cm^2}$ may be irradiated, together with high velocity, high pressure water cooling. The ion beam test chamber at PMTF has recently become operational and is capable of delivering up to 800 kW on large (up to several square metres) actively cooled targets, with heated areas as large as $800\ \mathrm{cm^2}$ receiving as much as $2.0\ \mathrm{kW \cdot cm^{-2}}$.

Presently, these facilities are being used to: (1) measure the thermophysical properties of materials for which the database is poor, such as Be, and of advanced graphites and novel coatings produced by plasma spray; (2) evaluate brazing and bonding techniques; (3) study critical heat flux onset and augment heat transfer techniques; (4) study melt layer formation and stability in disruption simulations; (5) test the heat removal capacity of coolant systems designed for use in PIC applications; and (6) study coolant channel erosion caused by flow cavitation and coolant impurities. While these tests are necessary and provide valuable information, they all fall into the category of single effects testing. An electron beam facility under construction at HEDL [84] will be capable of some multiple effects testing. The system couples HHF testing ($0.3–60\ \mathrm{kW \cdot cm^{-2}}$) with neutron irradiation of the samples. Fluence levels are low, and more multiple effects testing of this type is required.

In the next generation of machines, tokamaks, stellarators, RFPs, or mirrors, energy removal is required. In the long term, however, energy recovery is an additional goal. Owing to its excellent heat transfer capabilities and database, water is the preferred near term coolant. Long term options include other coolants such as helium, liquid metals, organics or molten salts. These alternative coolants are considered for reasons of safety, efficiency, tritium breeding, and design simplicity, and the choice of PIC coolant is closely related to the primary blanket coolant choice.

The primary uncertainties in heat transfer limits are critical heat flux (water, organics, molten salts), pressure drop (liquid metals), and flow stability (all) [89]. If heat transfer augmentation is used, it will also be necessary to determine the corresponding heat transfer and pressure drop correlations. For water coolant at heat fluxes up to about $2\ \mathrm{kW \cdot cm^{-2}}$, there is reasonable confidence for design predictions, although additional one-sided CHF measurements at long L/D ratios are desirable [90]. Liquid metals, on the other hand, will require substantial development efforts to demonstrate reliable operation at these heat flux levels because of uncertainties in MHD effects, especially in complex geometries.

At the higher heat fluxes and associated higher coolant velocities, flow distribution, flow stability and channel erosion are principal problems. The velocities at which the onset of flow and channel erosion problems occur is not well defined. Practical experience with water suggests $20\ \mathrm{m \cdot s^{-1}}$ as an upper limit for straight channels. For comparable inertial force, this suggests limits of 300 $m \cdot s^{-1}$ for He (5 MPa), 30 $m \cdot s^{-1}$ for Li, and 20 $m \cdot s^{-1}$ for organic coolants. Higher flow velocities also require that particular attention be paid to the design of headers and structural supports. Practical helium flow limits of about one-third sonic speed also imply about 300 $m \cdot s^{-1}$ at 5 MPa, though higher pressures may be possible. For organic and molten salt coolants, further uncertainties include the control of fouling under high temperature or irradiation induced decomposition (e.g. by use of smooth geometries and coolant chemistry control).

For these energy removal and recovery issues, steady state operation of large heated surfaces at high heat flux loads must be demonstrated. Small area tests may miss channel length effects (e.g. short lengths may be in the entry length regime), multiple channel effects (e.g. flow instabilities, including stagnant and reversed flow), header effects on pressure losses and flow distribution, and mechanical rigidity (which affects flow induced vibrations) of the component. Optimization of heat flux removal capability is desirable for PICs because it allows for more compact hardware, with associated advantages in ease of removal, in minimizing plasma contamination, and in minimizing parasitic neutron absorption, which aids in tritium breeding.

4.5. Tritium permeation and retention

In present day devices, the repeated exchange of hydrogen isotopes between the plasma and surrounding walls is a dominant factor in plasma refuelling, and it will control the critically important issues of tritium inventory and permeation in D-T reactors. Tritium inventory is a near term safety and environmental problem that must be addressed for the tritium operation of TFTR and JET, and for the design of the next generation of experiments. In advanced long pulse/high duty cycle reactor designs, tritium permeation through plasma interactive components into the coolant can create a severe economic penalty as well.

A considerable database has been generated for hydrogen isotope interaction with structural first wall materials such as austenitic stainless steels and Inconel alloys. However, little is known about the interaction of low Z limiter materials, such as graphite and beryllium, with tritium. TFTR and JET have used graphite exclusively for plasma-interactive components [91]. The current CIT design lists graphite as the baseline material [76]. Yet, graphite is a porous material with an affinity for hydrogen. Much controversy exists in the literature over fundamental tritium properties, such as the surface and transgranular diffusivity, bulk solubility, trap concentrations and binding energies. A detailed, reliable model of tritium uptake and retention for graphite in a fusion reactor environment is currently not available. The database and understanding of tritium interaction with other candidate low atomic number materials (e.g. beryllium, TiC, SiC, etc.) are as bad as or worse than for graphite.

For near term devices (TFTR through CIT), the critical requirements are experiments and theory for tritium interaction with low atomic number materials and coatings. Measurements should be carried out with hydrogen/deuterium accelerators and plasma discharge devices, as well as direct experiments in tritium facilities. Tritium experiments involve the complementary techniques of gaseous (T_2), atomic (T°), and ion (T^+) charging of materials. Tests using tritium offer many advantages over protium/deuterium in detectability, as well as being a direct measurement rather than a simulation using isotopic substitution. Existing facilities include H/D accelerator laboratories [92–94], H/D plasma devices [95, 96], and tritium laboratories such as the Tritium Plasma Experiment at Sandia [97]. Extensive theory and computer modelling of tritium retention and release from these low atomic number materials must accompany the experimental programme. The results of modelling must also be benchmarked against measurement of hydrogen isotope retention (including tritium from D–D reactions) in limiter materials exposed to the high power, reactor grade plasmas of TFTR, JET, etc.

In commercial reactor devices, tritium permeation through the surface of first wall and impurity control components into the coolant may lead to contamination of the coolant to unacceptable levels. Tritium inventory in these advanced devices is a less serious issue because much larger inventories will exist in other components such as breeding blankets. Safety issues associated with large on-site tritium inventories and provisions for handling tritium contaminated components must be addressed regardless of the inventory in first wall and impurity control components. The three material options to reduce tritium permeation in a power reactor are: (1) select lower permeability materials, (2) coat the plasma surface with a high recycling material, (3) coat the coolant side with a low permeability material.

Serious deficiencies in the materials database make precise calculations of tritium permeation rates impossible at this time. Scoping studies have indicated that the range of uncertainty in the materials database leads to coolant conditions that range from the case where no tritium processing is required up to a potential for large economic expenditures for tritium processing equipment.

To adequately define the magnitude of the tritium permeation problem and to develop solutions, significant progress must also be made in characterizing and defining: (1) production of trapping sites for tritium by neutron irradiation of the material; (2) changes in surface permeability and plasma side recycling due to erosion/redeposition processes; (3) release of tritium through interfaces (e.g. from the material surface to the coolant); and (4) designs that act as tritium permeation barriers. Until this R&D programme is completed, meaningful calculations of tritium permeation cannot be carried out.

4.6. Helium issues

Helium exhaust is an essential requirement for operating fusion reactors. Overall particle exhaust experiments and He injection experiments examining differential H/He pumping have been investigated using divertor [75] and pumped limiter [98] configurations. For commercial reactor applications the problem of helium exhaust from the system is complicated by helium implantation in limiter or divertor surfaces. In fact, it has been proposed that the trapping of helium under layers of redeposited material may be an effective mechanism for ash removal [99]. Helium will also be produced in first walls and in PICs through energetic (n,α) reactions, and through the decay of tritium resident in the material. Helium migration and void formation degrade the structural integrity and physical properties, such as thermal conductivity, of most materials. Consequently, all aspects of helium implantation, production and migration must be understood for materials to be used in PICs. Ongoing technology development in these areas is minimal, though some investigations of He retention in redeposited surfaces are underway [96].

4.7. Erosion/redeposition

Erosion/redeposition is an issue generic to all confinement systems. Energetic plasma edge particles will strike the surface of plasma interactive components and cause sputtering erosion. It has been observed that the sputtered particles enter the scrape-off region in the main plasma, then recycle, and eventually redeposit on other exposed surfaces [100, 101]. Calculations show that the redeposition process is important for extending the lifetimes of plasma interactive components [102, 103], even though redeposited materials may exhibit properties which are inferior to those of the as-fabricated material. It is difficult to estimate the changes in physical properties to be expected because redeposited surfaces of sufficient thickness have not yet been produced. At present, there is little erosion/redeposition in plasma devices because of the short pulses and low availability of present experimental devices. In addition, the plasma scrape-off conditions are somewhat different from those expected in an ignition device or a reactor where particle and energy fluxes will be much greater.

The principal ongoing work in this area is the post-plasma-exposure analysis of limiter surfaces and other PICs. Analysis techniques used include Rutherford backscatter, nuclear reaction analysis, and proton induced X-ray analysis [92–94]. These techniques combine to give a detailed description of near surface composition before and after plasma exposure. This allows for measurements of surface coating erosion rates, impurity deposition and hydrogen isotope implantation depths. Such measurements have been carried out on PICs removed from PLT, PDX, TMX-U, TFTR, TEXTOR, ISX and other machines [92]. The information obtained provides valuable insight into the erosion and impurity transport processes as well as information on hydrogen isotope retention in the materials. These measurements greatly contribute to the fundamental database, but the experiments lack the large ion fluxes needed to evaluate the redeposition process and include no neutrons.

Experiments have been proposed which attempt to reproduce the complex environment of the plasma edge region outside of a confinement device. Most notable among US experiments is PISCES [96], which is presently operating at UCLA, and the ICTF proposed by researchers at ANL [104]. Each facility has the capability to investigate problems in fusion technology and problems pertaining to plasma science. The technology issues addressed include: (1) testing of erosion/redeposition properties of candidate surfaces materials; (2) determination of the operating limits for plasma side application of high Z materials such as W or Ta (for instance, at what plasma temperature runaway self-sputtering cascades will be initiated); (3) evaluation of the sputtering and redeposition behaviour of alloys and compounds; and (4) testing of self-pumped helium removal concepts using V, Mo, Ni or other metals to trap helium beneath a continuously deposited surface layer. The physics issues addressed in experiments of this type include: (1) investigations of high recycling regimes; (2) optimization of limiter and divertor geometries; (3) evaluation and analysis of conditioning techniques in a controlled environment;

and (4) provision of a controlled testing environment for benchmarking computational models. The ICTF experiment produces conditions much more representative of the plasma edge environment but is also a much more expensive facility (estimated cost, 2–5 million US \$). Such a facility provides very valuable data, but it should be noted that irradiation effects are not included.

4.8. Disruptions and runaway electrons

A disruption can generally be described as a rapid reduction in the plasma current accompanied by the localized deposition of much of the plasma energy on an interior surface. Disruptions are observed in all tokamaks and the phenomenon is poorly understood. Consequently, elimination of disruptions for future machines cannot be assured and plasma interactive components must be designed to withstand the induced forces and high heat fluxes which accompany these events.

From the point of view of damage to in-vessel components, disruptions can be divided into two phases: (1) the thermal quench phase, which produces localized high heat fluxes, and (2) current decay phase, which induces large forces on the vacuum vessel and on in-vessel components. In considering the thermal quench in component design for near term applications, it is generally assumed that one-half of the stored plasma energy is deposited upon a single component, such as a single limiter module. The time-scale estimated for this energy deposition in operating tokamaks such as TFTR is on the order of 0.3 ms [85]. Further, the energy deposition is observed to be highly localized, and heat fluxes well in excess of 10 $kW \cdot cm^{-2}$ should be expected in future machines.

Heat fluxes of this magnitude can cause surface vaporization and melt layer formation. The interaction between the vaporized surface material and the incident plasma is not fully understood although some testing and modelling in this area has recently been completed [105, 106]. Considerable additional effort should be expended in this area since the outgoing vapour cloud has the potential to shield the surface and prevent more extensive damage. The difficult problem of stability analysis of the surface melt layer is compounded by the effects of the induced eddy current forces generated during the current decay phase. This, too, is an area requiring further analysis and testing, although the relative time dependence of the two phases of the disruption is not known.

The forces produced during the current decay phase determine, to a large extent, the structural support required for the vacuum vessel and for plasma interactive components. The major forces generated during this phase result from coupling of the induced eddy currents in the vessel or other plasma interactive components with the externally applied magnetic fields. The magnitude of these forces is primarily determined by the current decay time, τ_I. For operating tokamaks, τ_I ranges from 1 to 10 ms and it has been observed that τ_I increases with plasma current. Consequently, for advanced tokamak design studies, it is usually assumed that the current decay time will be on the order of 500 ms [81]. Direct measurement of eddy current strengths and patterns has been made on FELIX at Argonne National Laboratory [107, 108]. This facility includes a 1.0 T solenoidal field with an accompanying 0.5 T dipole field which ramps in 10 ms. The test stand includes a wide range of diagnostics and data acquisition systems.

The best hope for eliminating disruptions as a serious problem in tokamaks is the detection of disruption precursors, followed by the activation of soft landing feedback systems [71]. Obviously, research efforts aimed at the design of durable plasma interactive components must continue. If plasma interactive components capable of surviving multiple disruptions cannot be designed, future tokamaks may require frequent component replacement. This will seriously affect the availability of these devices.

Like disruptions, runaway electrons present serious survivability or lifetime problems for PICs. Runaways are a threat during low density operation and can also be generated as part of a disruptive discharge. There are numerous examples of severe damage to tokamak components which are attributed to runaway electrons. Runaways appear to be produced in small beamlets with energies of 10–50 MeV. Damage analysis from JET [109] indicated that incident energy densities were as high as 80 $MJ \cdot m^{-2}$, with deposition times of 4 to 20 ms. Runaways appear to be confined to very narrow regions of the scrape-off layer (0 to 5 mm), but upon impacting a PIC, large angle scattering during the slowing down process makes coolant lines within several centimetres of the plasma edge susceptible to melting. Runaways are, in fact, one of the most serious concerns in the design of actively cooled limiters for use on Tore Supra [81]. As with disruptions, avoidance of runaway electron formation is the most attractive option for dealing with this problem on future machines. However, while their presence greatly complicates heat removal, runaways do not

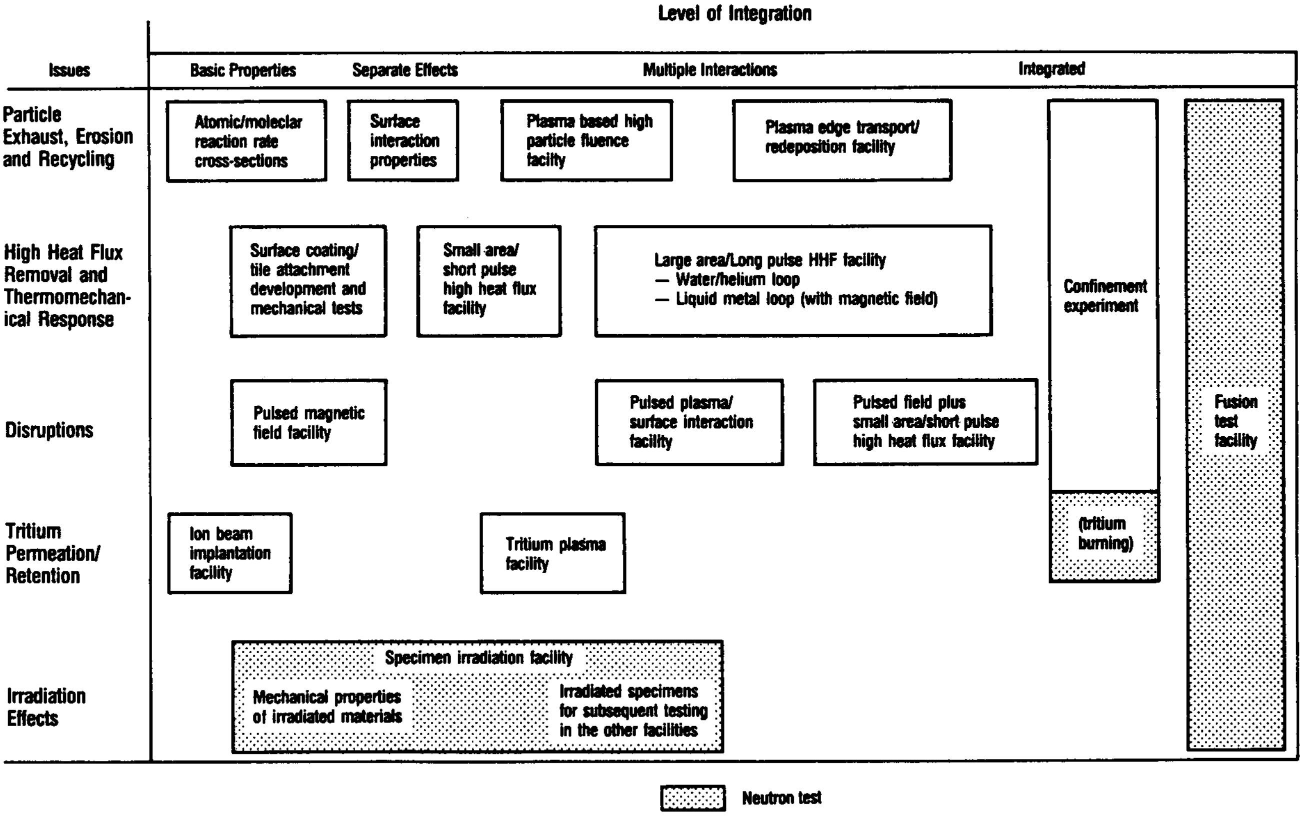


FIG. 16. Types of experiments and facilities for plasma interactive components (some experiments and/or facilities already exist).

appear to represent an insurmountable problem for PICs.

4.9. Radiation effects

Radiation damage is unavoidable in D–T fusion reactors and will impact reactor operation by determining the lifetime and operating limits of PICs and other system components. Of primary concern to PIC testing is the energy spectrum and fluence levels that can be achieved in the tests. Radiation damage due to 14 MeV neutrons will affect most of the thermal and mechanical properties of materials. In the case of structural alloys, the primary concerns are the effects on the mechanical properties, including ductility (impact, tensile, and creep), strength, fatigue and crack growth behaviour, and dimensional instability resulting from void or gas bubble swelling and radiation induced creep. These effects are also concerns for the plasma side materials, but the degree of concern is not as high since the plasma side materials do not carry primary loads. The integrity of bonds is expected to be degraded by radiation damage. In the case of non-metals and insulators, the thermophysical properties, such as conductivity and electrical resistance, will also be degraded. Existing facilities for neutron irradiation include EBR-II [110], FFTF [111] and other fission reactors equipped for sample irradiation.

Beyond these fundamental concerns, materials considered for use in reactor applications must be evaluated in terms of the synergistic effects that neutron irradiation might have on surface conditioning, the erosion/redeposition process, hydrogen isotope permeation and retention in the material, helium transport, and void formation. Again, this requires testing at the multiple effects level. Such testing can be accomplished by rotating samples between various single effects test stands or by material testing in a long pulse, burning core fusion experiment.

4.10. Evaluation of experimental needs

The unresolved issues for plasma interactive components and the ongoing testing efforts in this area have been briefly reviewed. Testing programmes in high heat flux, in basic tritium absorption and release, in erosion/redeposition, and in disruption simulations and modelling are providing some fundamental information, although much more could be done in these areas.

While some single effects testing is ongoing, the fundamental database in each of the general problem areas identified for PICs is deficient. The database on synergistic effects is nonexistent. The testing required to remedy this situation is outlined in Fig. 16. Emphasized here is the need to extend the level of activity in basic properties and single effects testing. Detailed understanding of these fundamental processes is essential in order to deal successfully with the more complex synergisms that occur in the edge plasma region of a fusion reactor. Valuable information would be obtained from the types of multiple effects tests listed in Fig. 16. These tests incorporate many, but not all, of the variables found in the edge plasma environment. Fully integrated testing, including all the complexities of the fusion reactor environment, may only be possible in a long pulse, burning plasma experiment with ample time dedicated to testing of fusion nuclear technology issues.

5. RADIATION SHIELDING

5.1. Introduction and issues

The role of radiation shielding is to protect the reactor components, the reactor operators and the public from intolerable levels of radiation exposure. The most sensitive components are superconducting magnets, some elements of plasma heating and exhaust systems, and instrumentation and control. Shielding must reduce the radiation damage and the nuclear heating of these components as well as the biological dose below the design criteria or regulatory level. Though many conceptual shielding designs exist, their design criteria differ by factors of up to an order of magnitude. Some of these design criteria are based on untested assumptions and incomplete models, and the uncertainties are not well evaluated. These uncertainties will affect construction and operation costs, availability, maintainability, and the life of the reactor. For example, the radiation level should be reduced by a factor of 10^5-10^6 in the regions from the first wall to the back of the shield. Since a reduction by a factor of $\sim 10^2$ can be expected in the blanket region, shielding should add a further reduction factor of at least 10^3-10^4, and even larger reduction is required for protection of personnel.

The key issues of radiation shielding [2] are categorized in Table XXVIII. These are generic for the various blanket concepts and confinement systems. Although additional issues may exist for a specific reactor design, such as high power density systems, these issues have a lower priority in test planning. The issues are briefly discussed below.

TABLE XXVIII. 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
    - modelling 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

(1) The design criteria for the sensitive components define the allowable radiation level at each component. These criteria should be determined from irradiation tests.

(2) The bulk shield plays the principal role in the protection of the magnet system and against biological exposure [112]. Most of the effort in fusion reactor shielding research has focused on design and analysis of the bulk shield. The main consideration in designing the shield involves the determination of optimized compositions and thickness of the shielding. Database uncertainties will strongly affect the evaluation of the effectiveness of the bulk shield [113]. These uncertainties arise from deep penetration of high energy neutrons (14 MeV peak) and γ-rays, and/or from the cross-section minima in particular energy ranges.

(3) The most difficult problems encountered in fusion reactor system shielding are those related to radiation streaming through penetration holes and slits. Open penetrations will directly cause serious damage to other reactor components resulting from leakage of neutrons and γ-rays through these penetrations. Streaming can also occur because of unexpected malfunction of a particular reactor component. This is one of the dominant problems in radiation shielding of fission reactors. The modelling procedure of penetration and partial shields is not simple and should be tested through experiments.

(4) Occupational dose issues affect reactor maintainability and availability. Radiation exposure due to induced activity after shutdown involves the contribution of γ-rays emitted from the reactor system and building (excluding exposure due to tritium leakage). At present, large discrepancies can be observed in the induced activation cross-section libraries.

Another possible source of radioactivity is radioactive corroded materials carried by coolant from highly irradiated components (e.g. first wall) to outside the shield. Large uncertainty exists regarding the amount of corrosion products, particularly in the case of liquid metal cooled blankets. Coolant pipes and heat exchangers with high radiation levels may require additional lead shielding. The development of remote operation systems and robotic techniques, with a resistance to high radiation, is needed to reduce radiation doses.

(5) The exposure level of the public from fusion power plants is a high safety concern. The selection of shield material is important to decrease the long lived radioactive waste. Radiation through sky shine will cause direct exposure to the public. This issue may not be serious for a next generation fusion facility, but will be important in power reactors.

(6) Shields are set in the inboard and outboard zones between the blanket and toroidal field coil of a tokamak. They must be fitted in these locations and the slit width between modules should be small enough to keep streaming low. Mechanical interactions can lead to mechanical failures [2].

(7) Larger uncertainties due to the modelling procedure, transport and response function calculation methods and database need a higher safety margin and conservatism. Most of the single and multiple effect testing will provide the data to be used for improvement of this software. By considering these issues in some detail, test requirements to resolve them could be addressed.

This work has evaluated the existing experiments, database, methods and facilities with respect to radiation shielding and attempted to anticipate the type and characteristics of experiments needed to resolve the issues. For shielding experiments, providing an adequate neutron source is an essential part of planning. The performance of the neutron source facility will constrain the experimental programme and the quality and quantity of data obtained. A point neutron source, a fission source and a fusion source have been compared based on their respective usefulness. Numerical investigation of the geometrical requirements has been performed for the shielding test matrix in a fusion test facility.

5.2. Required accuracy and present status

Uncertainties in predicting the shield performance 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 a reactor. However, the prediction uncertainty in the shield performance requires further research and development costs. Therefore, the desired optimum prediction accuracy can be determined from a cost–benefit analysis [112, 113].

The present status can be assessed by reviewing previous experiments, present data and calculational methods, and the results of sensitivity studies. A complete assessment is given in Ref. [2]. There are several examples of recent shielding experiments: (1) spectrum measurements, performed at LLNL, on spherical geometry for many materials [114–116], which have been used [117] to evaluate the adequacy of cross-sections and processing codes; (2) bulk shield and streaming

TABLE XXIX. REQUIRED ACCURACIES AND PRESENT STATUS IN NUCLEAR DESIGN OF FUSION REACTORS

Location/response	Required accuracy	Present status
First wall/divertor		
Nuclear heating	total 2%, local 10%	50%
Atomic displacement	10%	
Gas production	10%	
Transmutation	20%	
Induced activity	30%	50% to factor of 3
Blanket		
Tritium production rate	gross 3–5%, local 10%	gross 10%, local 20%
Nuclear heating	20%	
DPA	20%	
Gas production	20%	
Induced activity	50%	factor 2–5
Bulk shield		
Nuclear heating	20%	factor 2–5
DPA	30%	
Induced activity	factor 2	factor 5–10
Superconducting magnet		
Nuclear heating	gross 30%, local 50%	factor 5–10
DPA	gross 30%, local 50%	
Gas production	gross 50%, factor 2	
Dose	gross 30%, local 50%	
Induced activity	factor 2	
Penetration functional equipment (e.g. vacuum, pump, RF, and NBI)		
Nuclear heating	gross 30%, local 50%	gross factor 2, local factor 10
DPA and gas production	50%	
Induced activity	factor 2	
Reactor room (outside the shield and inside the reactor bldg.)		
Biological dose during operation	factor 3	
Biological dose after shutdown	factor 2	
External biological dose (outside plant site)	factor 3	

experiments [118–120], performed at ORNL, to obtain estimates of prediction accuracies; (3) bulk shield experiments [121] at JAERI for stainless steel with a thickness of 30 to 110 cm, as well as a number of streaming experiments [122]; and (4) neutron spectrum measurements [123], performed at Osaka University, for many shield materials with various thicknesses. At present, there are a number of evaluated nuclear data files in several countries. Examples include ENDF/B [124] in the USA, JENDL [125] in Japan, and EFF in Europe. Discrete ordinates computer codes are generally used for one- and two-dimensional transport problems, while Monte Carlo codes are normally used for three-dimensional transport problems. There are several computer programs for calculating nuclear responses, such as nuclear heating and induced activation. The current status of sensitivity and error analysis is given in Ref. [126].

An investigation was made for the required accuracies in predicting the important nuclear parameters in the blanket, shield and other reactor components where radiation is of concern, and the results are shown in Table XXIX. The radiation source originates in the plasma region; hence, the prediction accuracy of source characteristics is very important. Uncertainty in the source prediction propagates to all other nuclear responses. The required accuracy should be a few per cent. For the first wall and blanket region, good accuracies are required to predict severe radiation damage and the tritium breeding ratio. The achievable prediction accuracy generally decreases with increasing distance from the first wall, and is reflected in the required accuracies in Table XXIX.

Nuclear responses of the superconducting magnet in the toroidal field coils need relatively high prediction accuracies because of high radiation sensitivity. In a tokamak, the magnet is protected by the inboard bulk shield with a thickness of ~60–80 cm. Since the nuclear data for neutron and gamma ray transport and response functions have at least ~10% uncertainties, it is very difficult to reduce overall uncertainties below 20 to 30% after such a thick shield.

A factor of about two to three is shown for the required accuracies for the biological dose. Better accuracies are, of course, desirable but they are not practically achievable because of the large radiation attenuation required and the many possible streaming paths involved. Furthermore, present prediction accuracies are estimated based on comparison with experimental results. Some results are shown in the right column of Table XXIX. No serious discrepancy has been observed except for certain induced activities. However, there are very few data to estimate the present accuracies for many nuclear responses.

To achieve the accuracies cited in Table XXIX, the issues described in the preceding section should be resolved and verified through experiments.

5.3. Required experiments and facilities

5.3.1. Experiments

The types of shielding experiments needed include: (1) basic experiments for measurements of differential nuclear data such as cross-sections and secondary neutron energy distribution, (2) single effect simple geometry experiments for neutron and gamma ray transport and nuclear responses such as heating in the bulk shield and penetrations, and (3) multiple effects or partially integrated experiments with complex geometries to provide more detailed simulation of shielding components. Examples of these experiments are shown in Table XXX. Although the experiments are categorized according to the shield issues, the interrelations among the issues and experiments should be observed. Key points of the experiments listed in Table XXX are discussed below.

Basic experiments are common to all issues. They usually use small specimens and can be performed in the conventional accelerator or point neutron source facilities. The number of test articles is large, hence prioritization is necessary. The table shows the kind of nuclear data which mainly affect the uncertainty of the issue concerned. The required accuracies of the data, of course, depend on isotopes, type of reactions, energy range and required accuracies of integral quantities [113]. The uncertainty in predicting the activation dose is mainly caused from that in the response function at present. Details of the nuclear data needs are discussed in Ref. [127].

5.3.1.1. Bulk shield

As indicated in Table XXX, single and multiple effect experiments are needed for various shielding materials and configurations for the bulk shield. The main constituents of the bulk shield should be examined on an attenuation profile in the range of ~3–7 orders of magnitude for both neutron and gamma rays because prediction errors propagate and exponentially increase with distance from the front surface. The measured parameters include energy spectrum, threshold reaction rate, dose rate and heating rate. The optimum configurations of materials would be selected based on sensitivity analysis and should be verified experimentally. The bulk shield measurements provide good benchmark problems for transport calculation codes and nuclear data if they can be performed in simple geometries and with well identified source conditions.

5.3.1.2. Penetrations

Design of penetration and associated shielding is essential and most difficult. No data are available for the verification of design accuracies. Experiments have to start from fundamentals and then proceed systematically to understand the phenomena and to evaluate the design method. They are useful to find and examine semi-empirical approximations. Table XXX shows the

TABLE XXX. 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

typical geometries required in penetration experiments. Streaming experiments through circular straight ducts will show the fundamental aspect of penetration. The dependence on the ratio L/D (length to diameter) is a main design concern. If a point neutron source is used, the volume effect should be examined. The bending duct and slit are the basic areas considered to reduce streaming. Effectiveness experiments would be carried out by varying the shape, the number of bends, the angle of bend and the distance between bend points.

Since the prediction accuracy for streaming will not be good (there is no data to evaluate the accuracy), even if the three-dimensional Monte Carlo method is applied, partial mock-up experiments prior to detailed design are needed. The local dose rate distribution is important as well as the gross exposure rate in order to protect sensitive components.

The penetration shield should attenuate not only the direct radiation component from the burning plasma region during the operational phase, but also the gamma dose from the highly activated penetration components during the shutdown period. The coolant channels may cause streaming if helium gas is used or the coolant removed after shutdown. The multiple effects of these streaming holes and slits in different components could cause unexpected streaming paths, therefore, the predictable geometrical configurations for such a case should be tested before final design.

5.3.1.3. Induced activity

Exposure dose to workers is caused by the induced activity of reactor components, building and coolant. Accurate estimation of radioactivity spatial distribution, strength, and time dependence is fairly difficult, and experimental verification is not easy either. The D–T burning device can provide data for shutdown dose distribution. The radioactive corrosion product dose rate could be estimated from corrosion transport experiments and irradiation tests of first wall and plasma interactive components materials. The distribution of deposited materials and trapping efficiency in the cleanup system are not known, especially for liquid metal coolants [2]. The important part of the public exposure dose rate may come from sky shine in fusion power reactors. Some neutron source facilities have the potential to verify this effect.

5.3.1.4. Measurement technique and diagnostic development

Experimental uncertainties should be small enough to verify the prediction accuracy of the radiation field and the response. Present reliability and accuracy of the detecting system are not satisfactory for this purpose. Specifically, measurement techniques of the nuclear heating rate, damage rate and multifoil activation (MFA) need further development. The effect from the magnetic field should be tested for counter type detectors.

Since the experimental values must be compared with predicted results obtained by calculations, all the experiments should be well identified for source characteristics, boundary conditions, and geometrical and isotopic configurations. Results from the basic experiments are compiled to produce the data base required for neutronics calculations; the single effect and part of the multiple effect experiments offer the differential and integral data to examine the data base, the processing method, and transport and response calculation methods. Experiments in complex geometries will help in verifying design accuracies and in providing experimental justification of the shield design. Since the present work focuses on planning the experimental needs, and some needs for improvement were briefly mentioned before, the task required in the development of data and methods is not discussed here. Improvements in modelling, methods, and optimization are, however, the key issues to achieve the required accuracies within a reasonable design cost.

5.3.2. Facilities

Since neutrons are critical in all shielding experiments, the performance and specifications of the neutron sources are important in planning experiments. Possible neutron sources are fission reactors, accelerator based point neutron sources and fusion sources. These are characterized on the basis of neutron spectrum, flux and fluence, available volume and geometry, and operating cost. Different experiment stages will need different conditions. Since basic experiments usually need small specimens, the volume required is also small; multiple and partially integrated experiments need larger volumes. Experiments based on transport phenomena need relatively large volumes; for example, the area is several mean free path lengths squared and the thickness should be large enough to achieve several orders of magnitude of attenuation.

5.3.2.1. Point neutron source facility

There are many point neutron sources around the world which are usable for shielding experiments, although their usability is limited by their source strength or flux and fluence of 14 MeV neutrons.

Present D–T sources, such as RTNS–II (LLNL) [129], OKTAVIAN (Osaka University) [123], and LOTUS (Switzerland) [130] generate neutrons with a strength $\sim 10^{12}$–10^{13} n·s^{-1}. All have been used for neutronics and basic data measurements except RTNS–II.

Although the fluence of point neutron sources is lower than that of a fusion test facility, some benefits are expected. These benefits include well identified source characteristics, ease of access and a large available volume. Basic and single effect experiments are suitable for point sources since requirements on the flux and fluence are not high in shielding experiments. Even bulk shield experiments could be performed with a thickness of more than 100 cm of stainless steel [119]. In the next ten to fifteen years, point or small volume neutron source facilities will be mainly used to resolve these issues before the construction of a fusion facility.

There are basically three options for point neutron sources (cost estimates are shown in parentheses):

(1) construction of new facility (10 million US dollars)
(2) modification of conventional point source (about 2 to 5 million US dollars)
(3) utilization of RTNS–I, RTNS–II, FNS or LOTUS.

The third option has the lowest cost, but requires changes in existing programmes and also some small modification of the facilities.

5.3.2.2. Fission neutron source

In addition to point neutron source testing, fission reactors seem to be attractive in some respects. In the past, many fission reactors have been built and used for shielding experiments. Of course, the most serious problem in using fission sources is that they do not have 14 MeV peak spectra, but have large soft components below 1 MeV. The MeV range neutron flux can be increased with the use of convertors such as enriched uranium. These have test zones with large volumes and high fluence. Since no systematic study has examined the possibility of using fission neutron sources, we have compared some characteristics of fission and fusion sources.

Comparison calculations were made for a stainless steel shield using both a 14 MeV source and a fission source normalized to the same number of neutrons. It was found that the attenuation profiles for some nuclear responses are quite similar for the two sources, although the absolute values are different. The neutron spectra below a few MeV are quite similar for the 14 MeV and fission sources through the whole shield region. Most of the nuclear heating and dpa rates in the deeper shield regions arise from the energy range below 2.5 MeV. Hence, fusion conditions can be simulated, to a reasonable extent, by fission sources. However, the simulation of gas production rates would be difficult, because of their higher threshold energy.

5.4. Shielding experiments in fusion facilities

Shielding experiments performed in a fusion facility have many advantages with respect to strength and volume of the source and neutron spectrum. Tritium burning experiments, such as in TFTR, JET, or in a future ignition device could provide integral shielding data, such as activation level and dose rate, but would be limited in measured parameters.

A fusion engineering facility such as NET, FER, or FERF will provide much more capabilities for performing extensive shielding experiments. The requirements for shielding experiments on the operation mode and device parameters of a fusion test facility have been evaluated. 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, too.

Most of the neutronics measurements can be performed in a low fluence field ($\sim$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 of 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) × 140 cm (toroidal width) × 120 cm (poloidal height). This module can provide a test zone with a 40 × 40 cm surface area at the first wall and can simulate the radial profile of a full coverage case up to a depth of 80 cm into the shield within a deviation of 20% from the centreline values.

6. NON-FUSION IRRADIATION FACILITIES

The development of fusion technology will require engineering testing in a neutron environment. The best facility for this irradiation testing would be a fusion device. However, for technology development non-fusion irradiation facilities such as fission reactors and accelerator-based sources could be effectively utilized. The usefulness of these non-fusion irradiation facilities will depend to some extent on the research and development path chosen (and vice versa). Therefore, it is important to examine the technical and programmatic constraints on non-fusion irradiation testing in order to examine their advantages and disadvantages.

When exploring non-fusion irradiation experiments for fusion technology development, careful consideration must be given to the experiment's capability to match the anticipated environment for the fusion device. Five characteristics associated with the non-fusion irradiation facility should be considered: radiation damage, power density, lithium burnup rate, test volume, and non-nuclear induced environment.

Both the neutron energy spectrum and flux impact radiation damage to fusion reactor components. This damage arises primarily from atomic displacements and helium production. For a 'typical' fusion device, the neutron spectrum consists of 14 MeV neutrons from the DT reaction plus a large fraction of lower energy neutrons. Also, total neutron fluxes of $2 \times 10^{15} n \cdot cm^{-2} \cdot s^{-1}$ can be anticipated at the first wall for certain fusion device concepts (neutron wall loadings of $\sim$4 $MW \cdot m^{-2}$). The neutron energy spectrum in fission reactors is much softer than that anticipated in a fusion device. In fission reactors, the neutron fluxes with energies greater than a few MeV are negligible. When compared to the fusion environment the fission reactor neutron spectrum produces fewer displacements per incident neutron, probably results in differences in the spatial configuration of defect production, and produces different transmutations. Nevertheless, modelling and actual experiments suggest that the effects of high energy displacement cascades are not qualitatively different from those of low energy cascades produced in fission reactors. Therefore, atomistic damage may be correlated through the use of 'dpa' as the damage unit in the structural material.

The other source of radiation damage is gaseous transmutants (particularly helium in structural materials and helium/tritium in solid breeder materials). In fission reactors it is possible to approach simultaneously prototypic damage and helium production rates in nickel bearing structural materials through the use of isotopic tailoring and/or spectral tailoring. In nickel bearing materials, helium is produced by the following two-step reaction in fission reactors:

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

Increasing the initial ^{59}Ni concentration increases the helium production rate. Also, decreasing the neutron energy increases the helium production rate for a given neutron flux.

The power distribution (i.e. heat generation rates) and lithium burnup rates are strongly coupled in both a fusion device and non-fusion neutron irradiation testing. Only neutron and/or gamma fluxes can provide bulk heating to virtually all the materials simultaneously. And only the neutron flux can provide simultaneous in-situ tritium production and bulk heating, which are vital to solid breeder blanket technology experiments. Again, spectral tailoring can be employed to obtain fusion relevant power density and tritium production profiles. However, it is difficult to match the overall anticipated magnitude of the power density and lithium burnup rate. Figure 17 summarizes these observations for the damage in the structural material and the power density/lithium burnup rate in the solid breeder material for a reference Li_2O/He/HT9 fusion blanket concept using natural lithium. The facilities evaluated were Fast Test Reactors (FTR), Light Water Reactors (LWR), and a fusion facility at 5 $MW \cdot m^{-2}$. The figure shows that a reasonable simulation can be achieved by using currently available facilities and techniques for all but the helium generation in HT9. To achieve the necessary helium generation in HT9 will require using 75% enriched ^{59}Ni; the achievable rates with nickel doping are shown by the dotted lines. In the past, it has been prohibitively expensive to enrich nickel to these levels; however, newer processes of isotope separation are currently being evaluated.

The available test volume for fusion technology testing is limited in non-fusion test facilities. In fission reactors, the test modules are limited in size to a cylindrical envelope of $\sim$10 cm diameter by 1 m in length for acceptable neutron fluxes. Therefore, the in-core testing is limited to scaled breeder modules. Testing outside the core is feasible for larger sized experiments; however, such testing would require significant modification to existing reactors, and the neutron flux would be significantly lower than desired. The effect of the fusion blanket test module, which would be a strong neutron absorber, on the reactivity balance in the fission test reactor is also a concern

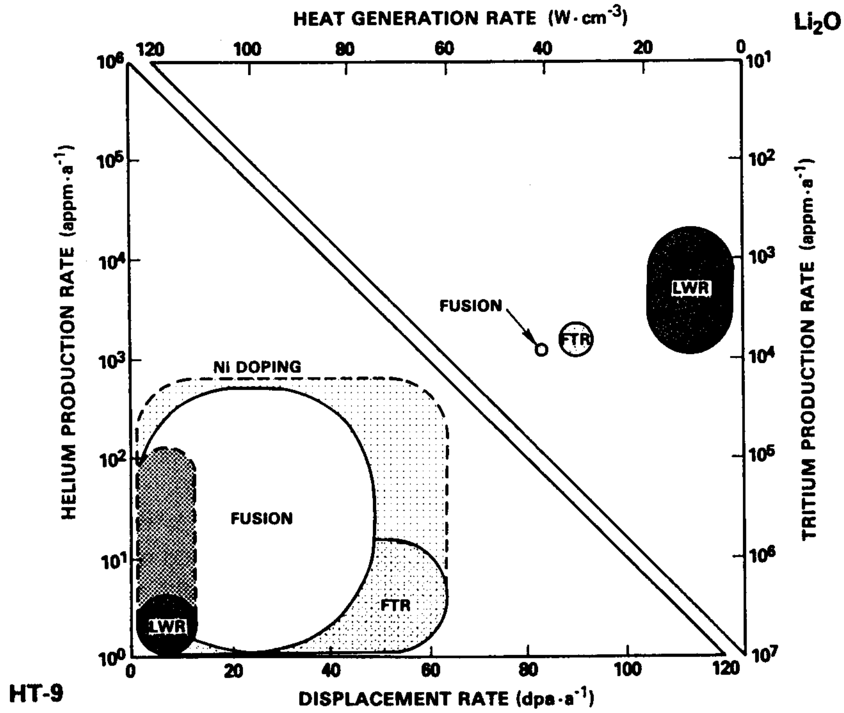


FIG. 17. Simulation of $Li_2O/He/HT9$ fusion blanket in fission reactors.

which may limit the number of available test reactors to those having large cores (>50 cm diameter).

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

Thus far, our discussions have been limited to the suitability of fission reactor testing. Accelerator based neutron sources can also play a role in fusion technology development. These facilities produce neutrons through the interaction of accelerated charged particles with a target material. The three facilities currently available within the USA are: (1) RTNS–II at Lawrence Livermore National Laboratory (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 those predicted for a fusion device. The He/dpa ratio is 14 in iron and 12 in copper which is comparable to that anticipated in the fusion environment. The highest flux is estimated at 5×10^{12} $n \cdot cm^{-2} \cdot s^{-1}$ at a distance of 0.35 cm from the target. The available volume with flux greater than 10^{12} $n \cdot cm^{-2} \cdot s^{-1}$ 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 are 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 litres are available for neutron irradiations; the volume with flux greater than 10^{13} $n \cdot cm^{-2} \cdot s^{-1}$ 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.

The Fusion Materials Irradiation Test (FMIT) facility has been proposed to fill the need for a high fluence accelerator based source for fusion technology development. 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 deuteron beam impinging on a lithium target to produce a peak neutron flux of 3×10^{15} $n \cdot cm^{-2} \cdot s^{-1}$ with a mean energy of 12 MeV and a broad peak at 14 MeV. 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} $n \cdot cm^{-2} \cdot s^{-1}$ is 7.6 cm^3; for flux greater than 10^{14} $n \cdot cm^{-2} \cdot s^{-1}$, 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 ten. 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 in use at present.

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

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

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, nor do they provide adequate test volume for engineering testing.

7. FUSION INTEGRATED TESTING

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.

If the fusion test device conditions are different from commercial reactor conditions, it may still be

TABLE XXXI. REQUIREMENTS FOR FUSION INTEGRATED TESTING [2, 131]

Parameter	Reference reactor	Test facility parameter	
		Minimum	Desirable
Neutron wall load ($MW \cdot m^{-2}$)	5	1	2–3
Surface heat load ($MW \cdot m^{-2}$)	1	0.2	0.2–0.5
Fluence ($MW \cdot a^{-1} \cdot m^{-2}$)	15–20	1–2	3–6
Test port size ($m^2 \times$ m deep)	–	0.5 × 0.3	1.0 × 0.5
Total test surface area (m^2)	–	5	10–20
Plasma burn time (s)	continuous	500	>1000[a]
Plasma dwell time (s)	none	<100	<50
Continuous operating time	months	days	weeks
Availability (%)	70	20	30–50
Magnetic field strength (T)	7	3	5

[a] Steady state preferred.

possible to modify the design of the test module (e.g. coolant flow rate) in order to retain the important aspects of the testing issues. However, too large a change of device conditions results in the loss of 'act-alike' behaviour. Through analysing the behaviour of components under altered device conditions and considering methods for scaling the observed behaviour to that expected in a reactor, it is possible to identify a set of minimum fusion test facility requirements in order to provide useful testing of nuclear technologies. Such analyses were performed for a range of blanket concepts [2, 131]. The resulting requirements are also expected to provide useful testing of the other nuclear components. These requirements are summarized in Table XXXI. From a fusion technology development viewpoint, any fusion device that satisfies these requirements is acceptable.

7.2. Reliability considerations

Many components in the first fusion test facilities will have little or no engineering precedence. This will be particularly true of nuclear components which, despite the best design effort and prefusion testing, will not yet have produced a high degree of confidence in their estimated reliabilities. Most likely, early fusion engineering facilities would be used for interative design/test/fix programmes aimed at improving component reliabilities. An apparent paradox results, however, because those nuclear components that would be targeted in a reliability improvement programme depend on the reliable performance of other nuclear components in the system.

One example of this would be the development of blanket modules in a high fusion power facility (e.g. INTOR, NET), which must also breed its own tritium in many tritium breeding modules. Although the breeding modules would be designed for high reliability, they would be essentially unproven and probably complex. Any one of these could fail such that the overall facility could not operate (e.g. loss of vacuum boundary, loss of tritium supply).

The implementation of a test programme to develop high statistical confidence in a reliability data base prior to engineering demonstration is clearly a desirable goal, but can be very difficult in practice because of the requirement for an extended test period. The INTOR critical issues study [132] concluded that the achievement of an 80% statistical confidence level in a given component mean time between failures (MTBF) in the constant failure rate regime of operation (i.e. random failure probability) would typically require a cumulative test period of 3.5 times the MTBF.

For blanket modules, the reliability requirements would be somewhat relaxed, since blanket removal is expected to be a relatively routine maintenance operation. In this case, a minimal fusion test facility blanket system availability goal might be ~60%. Since a facility such as INTOR might have six blanket modules per TF coil sector (60 total), the required

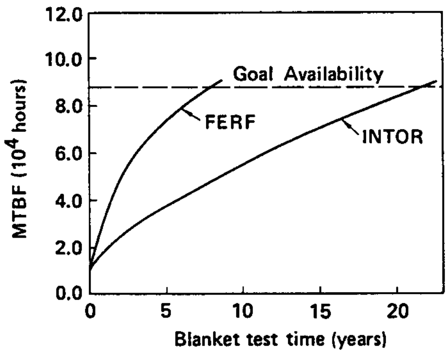


FIG. 18. Higher FERF availability leads to more test time and faster MTBF growth.

availability for individual components might be $(0.6)^{1/60} = 0.9915$ or 99.15%. For a typical repair/replace time of one month, this implies a required MTBF of ~10 a, and a test period of 34 a. If equal credit can be taken for 60 modules tested in parallel, the required test period would be reduced to a manageable 0.5 FPY.

In addition to establishing reliability, tests which result in component reliability improvement are also of interest. Although an accurate prediction cannot be made, some systems have been observed to follow a power law relation between the component testing time and the achieved MTBF [2].

Two development pathways have been studied. The first pathway is based on a high fusion power facility, such as an ignited conventional tokamak like INTOR (~600 MW) to achieve engineering testing. This facility would develop and test reactor blankets in 10% of the blanket area, while the remainder would be simpler tritium breeding modules to supply the device's tritium requirements.

In the second pathway, engineering testing is conducted in a low fusion power engineering research facility, referred to here as a FERF. This facility would be able to use external tritium supplies, but would test the same number of reactor blankets. This avoids relying on unproven tritium breeding modules. In contrast, an INTOR class facility would have reduced availability, because of the increased likelihood of failure of the in-situ tritium breeding modules.

Figure 18 shows the calendar time required to achieve a blanket MTBF of 10 a (87 600 h) for the two pathways, based on representative parameters given in Ref. [2]. A FERF class facility, with higher blanket availability, is able to reach the goal availability in much less time than the INTOR class facility. Parametric studies performed indicate that the relative results are not expected to change for plausible initial assumptions.

7.3. Fusion test facilities

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 be obtained, of course, 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.

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, including unconventional forms that might improve the concept as a fusion test facility. Organizations familiar with the individual concepts were provided minimum acceptable values of the device parameters (Table XXXI) and asked to generate devices that met or exceeded these requirements.

The representative engineering test facilities are summarized below. More details on the devices considered are described in Ref. [2], or in the references listed.

(1) *INTOR:* A conventional reactor relevant tokamak with ignited operation, inductively driven current, RF heating and moderate field superconducting magnets. It can provide full integrated testing of plasma physics and technology. Devices in this class include INTOR [16], NET [17], and FER [18]. The 1982 INTOR design is used here to represent this general class of facilities.

(2) *LITE FERF [133]:* A driven version of the LITE ignition device design. 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 $MW\cdot m^{-2}$ and 500 s pulse, with extended pulse (100 s) and high power (2 $MW\cdot m^{-2}$) modes.

TABLE XXXII. PERFORMANCE COMPARISON OF FUSION ENGINEERING RESEARCH FACILITIES

	Tokamaks					Tandem mirrors		
	INTOR	LITE FERF	BEAN FERF	DTFC-IDT	Spherical torus FERF	TDF	MFTF-α+T	Reverse field pinch
Fusion power (MW)	620	90	185	100	39	36	17	22–110
Electrical consumption (MW(e))	200	210–270	185	427	120	250	104	126–180
Neutron wall loading ($MW \cdot m^{-2}$)	1.3	1.0–2.0	1.3	2.0	1.0	2.1	2.0	1.0–5.0
Surface heat flux ($MW \cdot m^{-2}$)	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] ($MW \cdot a \cdot m^{-2}$)	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 nine years at ultimate availability.

(3) *"BEAN" FERF [2]:* 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 [134]:* 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 [135]:* 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 [136]:* 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:* A representative reversed field pinch configuration with copper coils and Ohmic heating. Two RFP versions are considered, a 1 $MW \cdot m^{-2}$ neutron wall load reference version and a 5 $MW \cdot m^{-2}$ extended version.

Table XXXII summarizes major design parameters of the fusion facilities considered. Since the designs were prepared independently and are not necessarily consistent, common assumptions were imposed here with respect to availability, duty cycle, useful test area, lifetime, and capital and operating costs. Some assumptions were to impose uniformity on parameters that cannot be easily quantified. For example, the ultimate availability was assumed to be 45% for all devices but INTOR, which was reduced to 35% because of its higher complexity with tritium breeding driver blankets. Other assumptions imposed a uniform equation on parameters, such as the first wall test area, and capital and operating costs. Details are given in Ref. [2].

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 address performance (as a test facility) as a function of cost and risk.

Table XXXIII summarizes the overall performance, cost and risk of these concepts. The three 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, is defined as the product of neutron wall load, device availability and test area (which only includes regions with adequate depth). These parameters can be traded against each other within certain bounds without affecting the overall irradiation capability. Secondly, the ability to provide fusion reactor relevant conditions is expressed by the degree of required scaling. From results obtained at 1 $MW \cdot m^{-2}$ neutron wall load, considerable but plausible extrapolation is required to predict behaviour at 5 $MW \cdot m^{-2}$ reactor levels. As the neutron wall load decreases, the amount of extrapolation increases and the value of the data decreases. Since many device parameters are at least indirectly related to neutron wall load, higher neutron loading generally implies that all parameters are more reactor relevant. Consequently, the neutron wall load is an approximate global measure of the amount of scaling required. 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 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 h continuously.

Summary risk parameters are desirable to represent 'overall' physics and technology extrapolation from present data. A crude measure of 'overall' risk is shown in Table XXXIII based on a consistent and cumulative assessment of the amount of extrapolation required for the major physics functions (e.g., plasma heating) and technology subsystems (e.g. magnets) [2]. The numerical values are based on zero 'risk' points for moderate extrapolation, one point for a large extrapolation (some additional testing required), and two points for a very large extrapolation (major experimental programme needed).

The major cost parameters are the capital and annual operating costs. Although no detailed analysis was performed, ground rules were adopted to provide consistency among the concepts. The direct capital

TABLE XXXIII. SUMMARY CHARACTERISTICS OF FUSION ENGINEERING RESEARCH FACILITIES

	Tokamaks					Tandem mirrors		
	INTOR	LITE FERF	BEAN FERF	DTFC-IDT	Spherical Torus FERF	TDF	MFTF-α+T	Reverse field pinch
Irradiation capability ($MW \cdot a \cdot a^{-1}$)	14	2.9	5.8	3.2	1.4	3.8	1.8	1.6–7.9
Neutron wall load scaling ($MW \cdot m^{-2}$)	1.3	1.0–2.0	1.3	2.0	1.0	2.1	2.0	1.0–5.0
Burn length capability (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
Tritium consumption ($kg \cdot a^{-1}$)	5.8[e]	2.0	4.1	2.2	0.97	0.90	0.42	0.55–2.8
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 three years non-tritium/low availability operation plus nine years full availability operation.
[c] (Total cost)/(annual fluence X area) rounded to nearest leading digit.
[d] (Annual fluence X area)/(total cost) (physics+technology risk) rounded to nearest leading digit.
[e] Assuming TBR = 0.6.

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 [2]. 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 XXXII and XXXIII 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 Research Facility concepts exist. All concepts considered provide reasonable performance for technology testing; the minimum requirements identified earlier are 1 $MW \cdot m^{-2}$ neutron wall load, 1 MW of irradiation capability (e.g. 1 $MW \cdot m^{-2}$ over 5 m^2 test area at 20% availability each year), with pulse lengths over 500 s. It is also clear that a technology test facility may not be as costly as a combined physics/technology device, but is still expensive. This is perhaps not surprising since costs are driven by the presence of neutrons and by the total power level. 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. The cost per neutron figure of merit indicates the economy of scale; conventional tokamaks (e.g. INTOR) are large and provide considerably more potential test area without a correspondingly large increase in cost, although there may be limited practical utility of test areas over $\sim$20 m^2. The spherical torus and reverse field pinch offer relatively low total power and therefore cost, 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 would reduce the risk and improve performance for all concepts.

8. CONCLUDING REMARKS

The timely development of fusion nuclear technology is critical to the development of fusion energy. A process has been developed and applied 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 4) development of a test plan to identify the role, timing, characteristics and costs of major experiments and facilities.

The nuclear subsystems addressed are: 1) blanket, including first wall; 2) tritium processing system; 3) plasma interactive components (PICs); and 4) radiation shield. The technical issues and the development problems of the blanket are complex, hence the greater part of this study 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. The radiation shield is much simpler, and the tritium processing subsystems have a larger existing technology base, so their 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 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.

A major feature of the R&D framework for fusion nuclear technology is the utilization of non-fusion facilities over the next fifteen years, followed by testing in fusion devices beyond about the year 2000. Basic, separate effect and multiple interaction experiments in non-fusion facilities will provide property data, explore and explain phenomena, and provide input to theory and analytic modelling development. The database from non-fusion testing should be sufficient to: 1) quantitatively assess the economic, safety and environmental potential of fusion; and 2) design and construct experiments for testing in a fusion device.

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 fusion nuclear technology integrated testing imposes certain requirements on some of the fusion device parameters (e.g. neutron 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 solid breeder and liquid breeder blankets. Within each class there are a number of design concepts that involve a variety of material and configuration choices. 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.

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 programme requires an average expenditure of about 20 to 40 million US dollars per year. The level of confidence in the details of the test plan and associated cost estimates is higher for the near term tasks. As with any test plan for a complex R&D programme, the technical requirements and cost estimates for experiments and facilities beyond the next few years will need to be revised on the basis of 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 (including a large technology base, small system size, and independence from neutrons) have permitted advanced experimental investigation of the tritium processing issues early in the programme. The Tritium Systems Test Assembly (TSTA), now in operation in the USA, can be classified as a 'partially integrated' test facility. Present plans for this and related international 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. Experiments are also needed to understand plasma driven and pressure driven tritium permeation issues.

The main issues for the radiation shield relate to the accuracy of neutronics prediction capabilities and 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: 1) a modest programme to improve basic nuclear data and calculational methods; 2) integral experiments with a point neutron source; and 3) 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 programme 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 co-operation. As evident from present activities [9] in the world fusion programme 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 co-operation.

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

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