FUSION NUCLEAR TECHNOLOGY TESTING REQUIREMENTS*

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ABSTRACT

The technical issues and requirements of experiments and facilities for fusion nuclear technology (FNT) have been investigated in the FINESSE study. The nuclear subsystems addressed are: a) blanket, b) radiation shield, c) tritium processing system, and d) plasma interactive components. Emphasis has been placed on the important and complex development problems of the blanket. The study developed and applied a process for technical planning of FNT that 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 general R&D framework developed for FNT consists of three stages. The first involves obtaining property data and exploring phenomena in non-fusion facilities such as fission reactors and non-neutron test stands. The second stage concerns concept verification and will focus on integrated testing of experimental modules in fusion facilities. Some of these modules can provide partial simulation of the component while others provide an integrated simulation of all physical elements and environmental conditions within the component. Effective FNT integrated testing imposes certain requirements on some of the fusion device parameters such as the wall load and plasma burn time. These requirements have been quantified. The third stage 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.

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1. INTRODUCTION

Fusion nuclear technology is a critical element in the development of fusion. Many of fusion's remaining unresolved issues are posed by nuclear technology. These issues relate to a) feasibility, a primary acceptance criterion for the science and technology communities; b) economics, a primary acceptance criterion for the utility industry; and c) safety and environmental impact, a crucial acceptance criterion for the public.

The technical and programmatic issues in the research and development (R&D) of fusion nuclear technology (FNT) are presently being investigated in the FINESSE study. The study has developed and applied a technical process for defining the major characteristics of major experiments and facilities for FNT. 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 iterations.

The study is led by the University of California, Los Angeles, with major participation of key U.S. organizations. The study is also benefiting from active participation of scientists and engineers from Canada, Europe and Japan. FNT offers unique opportunities for international cooperation. Therefore, the FINESSE study has attempted to maximize the usefulness of its results to the international community by emphasizing technical issues, design concepts and facilities that appear to be of global interest.

Detailed results from FINESSE have been published in Ref. [1]. In these reports, the technical issues, experiments, facilities and test plans are addressed for four nuclear subsystems: a) blanket/first wall, b) radiation shield, c) tritium processing system, and d) nuclear elements of plasma interactive components. In addition to non-fusion facilities, the study has investigated the requirements and options for fusion testing devices.

This paper provides an overview of selected results from FINESSE. Section 2 summarizes the general framework for FNT R&D. Section 3 presents the technical issues and major characteristics of the required non-fusion facilities including non-neutron test stands, fission reactors, and point neutron sources. The greater part of Section 3 is devoted to the complex problems and requirements of the blanket. The radiation shield, tritium processing and plasma interactive components are briefly addressed. Section 4 defines the requirements of the major parameters and technical features of fusion devices from the nuclear technology testing standpoint. Fusion facilities represent a particularly important topic because of many complex aspects concerning the cost, benefit and risk tradeoffs for such devices. A number of options for fusion test facilities are identified and compared. Considerable details on all topics presented here are available in the FINESSE reports [1].

2. FRAMEWORK FOR FNT R&D

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 FNT development. The ultimate goal of fusion R&D is the development of commercial fusion reactors. Representative goal ranges considered for commercial reactors are given in Table I.

A major feature of the R&D framework developed for FNT is the utilization of non-fusion facilities over the next 15 years, followed by testing in fusion devices beyond about the year 2000. No fusion device with significant nuclear testing capability is assumed to be available prior to the year 2000. This is consistent with the presently proposed schedule for major devices such as NET in Europe, FER in Japan, and an internationally constructed engineering test reactor.

The types of experiments required for FNT can be classified into: 1) basic, 2) separate effect, 3) multiple interaction, 4) partially integrated, and 5) integrated tests. 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. Figure 1 shows the role of these types of experiments. The strong interrelation between experiments and modeling is also illustrated in the figure.

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 an experiment increases, more synergistic effects are observed, and the emphasis shifts from understanding and theoretical modeling to obtaining engineering data and empirical correlations. The level of integration necessary for a design concept to be verified depends on the complexity of the component. For fusion nuclear components such as the blanket, it has been concluded that concept verification is unlikely prior to performing fully integrated tests [1]. While basic, separate effect and multiple interaction experiments can be performed in non-fusion facilities, completely integrated tests are possible only in fusion facilities.

Basic, separate effect and multiple interaction experiments in non-fusion facilities over the next 15 years will provide property data, explore and understand phenomena, and provide input to theory and analytic modeling. The data base from non-fusion test facilities should be sufficient to: 1) quantitatively assess the economic, safety and environmental potential of fusion; and 2) select, design and construct experiments for testing in a fusion device.

Non-fusion facilities can be classified into: a) non-neutron test stands, b) fission reactors, and c) accelerator-based neutron sources. The role of each type of facility has been evaluated in FINESSE [1]. Significant differences are found. For example, fission reactors are the primary facilities for solid breeder blankets, while non-neutron test stands are the most important facilities for liquid metal blanket R&D.

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 provide partial simulation of the component, while others provide an integrated simulation of all physical elements and environmental conditions within the component. Effective FNT integrated testing imposes certain requirements on some of the fusion device parameters (e.g., neutron wall load, plasma burn time); these requirements have been quantified and are presented in Section 4. Any fusion device that meets these requirements will satisfy the needs of FNT 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. Figure 2 illustrates the role of facilities and sequence of R&D in this framework.

3. TESTING IN NON-FUSION FACILITIES

3.1 Blanket Technology

The blanket is a particularly important fusion nuclear component, which simultaneously provides the functions of energy conversion and recovery, fuel breeding, and partial shielding. At the same time, its close proximity to the burning plasma leads to high heat flux, severe radiation loads, and high magnetic fields. This harsh and complex environment, together with the multiple functions the blanket must perform, leads to a number of critical feasibility and attractiveness concerns.

Blanket concepts can be divided into liquid breeders and solid breeders. Within each class, there are a number of distinct material and design options, as shown in Figure 3. Although the functional requirements and reactor operating conditions are similar for both classes of blankets, the critical issues are generally not. Consequently, the issues and associated experiments are discussed separately. Within the uncertainties, it is not possible to determine whether solid or liquid breeder blankets are more attractive. Consequently, it appears prudent for the fusion program to retain both options, although a selection could be made at some point in the future when more information is available.

3.1.1 Solid Breeder Blankets

3.1.1.1 Issues and Testing Needs. The most important uncertainties for solid breeder blankets are related to tritium breeding, tritium recovery, and breeder thermomechanical behavior (see Table II). These uncertainties are large for solid breeder blankets because: 1) there is limited understanding of tritium transport mechanisms 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. Safety uncertainties involve the behavior of the blanket (and blanket materials) under off-normal or transient conditions, and the control of tritium under normal operation.

For adequate tritium breeding, most solid breeder blankets require ^{6}Li enrichment and a neutron multiplier, with the possible exception of Li_2O. Even so, within present uncertainties in the data, modeling methods and design definition, it is not clear that present solid breeder blanket concepts can provide reactor self-sufficiency in tritium. The tritium breeding ratio (TBR) for several blankets has been calculated using a 3-D model [2], and the uncertainty estimated from sensitivity studies. None of the blankets achieves a high enough breeding ratio to assure a required TBR of 1.07 within the uncertainties.

The need for a neutron multiplier to enhance breeding 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 tritium and thermal behavior. An accurate assessment of the tritium breeding margin would thus indicate whether blankets without distinct multipliers are possible and, if not, what level of physical separation is acceptable.

The prediction of tritium behavior in solid breeder blankets requires understanding tritium transport, retention, and chemical form in the breeder and multiplier material under the influence of the fusion environment. One important parameter is the total blanket tritium inventory. The major contributors to the inventory are related to diffusion, solubility, and surface adsorption. The uncertainty in the diffusivity can be many orders-of-magnitude, particularly at higher temperatures and burnups. The soluble tritium inventory is believed to be large only for Li_2O, where it is reasonably well measured. In contrast, 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) [3]. However, the O_2 activity can 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/day in a 5000 MW_{th} reactor) [3]. The same tritium transport phenomena apply as with solid breeder materials, but data to assess their magnitude are not available. Even if the tritium is released from the multiplier, then the tritium must be removed by the coolant or purge streams. These two options could lead to either coolant contamination or breeder/multiplier chemical interaction concerns.

The major issues associated with the mechanical interactions among the solid breeder, multiplier and structure include restructuring of the solid breeder, deformation and/or rupture of the structure, and changes in the heat transfer across the breeder/cladding interface. The primary driving forces are swelling (particularly for Li_2O and Be) and differential thermal expansion. The blanket materials can respond by deformation, creep, or fracture, but the extent of each is not known. Further experiments are needed to 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 is generally at low temperature (~ 100°C) and low fluence compared to the anticipated end-of-life blanket conditions.

The thermal behavior of the breeder is constrained by the relatively low thermal conductivity (~ 1-3 W/m-K) and upper temperature limits (~ 800-1000 °C) presently assumed for solid breeder materials. These parameters are poorly defined due to uncertainties in processes such as sintering, creep, phase change, vapor phase transport or corrosion.

Important but less critical issues for solid breeder blankets are mechanical behavior of the structural elements and tritium permeation. The mechanical behavior of structural elements of the blanket determines 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.

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 primary breeder extraction stream (purge stream) include the tritium form, the efficiency of the recovery process, and the tritium inventory in the recovery system.

3.1.1.2 Required Experiments and Facilities. The issues can be addressed by a range of possible experiments as summarized in Figure 4 and discussed below. A number of experiments are already completed or are ongoing, including both closed and open capsule irradiation experiments such as TULIP, FUBR, ORR, TRIO, and a number of similar experiments outside the U.S. Additional tests are proposed to bridge the gap between existing data and the data base needed to assess the overall feasibility of solid breeder blankets. The tests span all levels of integration, from basic properties, to phenomena exploration in separate and multiple effect tests, to concept verification in integrated fusion tests. Much more detailed analysis of the required experiments can be found in Ref. [1].

Since there is no general theoretical basis for scaling solid breeder behavior, the significant phenomena must be quantified by conducting tests at fusion reactor relevant conditions. Among the most important parameters are the tritium generation and heating rates for the solid breeder materials, and helium generation and displacement rates for structural materials.

Figure 5 compares the helium production and displacement rates for HT-9 and the microscopic tritium production and heat generation rates for the Li_2O solid breeder in a Li_2O/He/HT-9 blanket using natural enrichment lithium. The facilities evaluated were Fast Test Reactors (FTR), Light Water Reactors (LWR), and a fusion facility at 5 MW/m^2. The figure shows that a reasonable simulation can be achieved using currently available facilities and techniques for all parameters except the helium generation. It is possible to achieve near-prototypic helium generation rates in fission reactors for a wide range of nickel-bearing alloys by varying the nickel content and/or isotopic composition of the nickel. To achieve the necessary helium generation in HT-9 will require using 75% enriched ^{59}Ni; the achievable rates with Ni doping are shown by the dotted lines. In the past, it has been prohibitively expensive to enrich nickel to these levels; however, newer processes of isotope separation are currently being evaluated.

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

One of the fundamentally important tasks for solid breeder blankets is material development and characterization for both the solid breeder material and the multiplier. The basic materials can be tailored to some degree to provide specific properties. Therefore, material improvement is an important part of this task. A sufficient data base is needed for their thermal behavior (thermal stability, thermal conductivity), tritium behavior (tritium diffusivity and retention), and mechanical properties (swelling, creep, and ductility). Some understanding of the many material-related variables is also necessary to identify directions for improving the properties. Particularly important are temperature, grain size, porosity and pore size distribution, impurities or additives, fabrication process, material form, burnup, container material, and purge gas flow rate and composition.

A number of tritium recovery experiments are underway and will provide a fairly wide-ranging data base around the year 1990. However, the planned tests will not address certain synergistic effects, such as the combination of moderate-to-high burnup with a flowing purge gas, temperature gradients, and breeder/clad interactions. Consequently, advanced in-situ tritium recovery tests should be planned to investigate synergistic effects, design limits, and transient behavior. These experiments could be performed as one or more instrumented and purged assemblies in fission reactors.

Tritium breeding tests are a special class of experiments which can be performed utilizing point neutron sources. These tests are needed to verify and improve nuclear data, design methods, and models. Simple mockups using a point neutron source with Li_2O have already been initiated under the U.S./JAERI Fusion Breeder Neutronics Collaborative Program [4]. While such experiments should continue, more complex engineering mockups will also be needed to address uncertainties associated with the geometric details of the blanket and surrounding reactor. These tests include partial coverage of the neutron source with a mockup of the reactor sector, plus a detailed blanket module design for measurement of the tritium and heat production profiles.

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 serve different roles in that regard. Non-neutron thermomechanical tests with heat sources such as microwaves or resistive wires have been explored to test up to full blanket modules. While there are clearly limitations on the simulation of reactor heating profiles and irradiation effects, these tests provide an opportunity to explore complex thermomechanical behaviors (such as gap conductance, flow distribution, and 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, cost, and reactor safety concerns.

Nuclear test assemblies for fission reactors can provide the maximum concept verification possible in non-fusion devices. These include the important nuclear effects, but would be limited in several respects, primarily test volume. In-core assemblies could be placed in existing fission reactors like FFTF at reactor-relevant heating rates (2-5 MW/m^2), but would be limited to about 10-cm diameter. Ultimately, testing in a fusion device will provide complete concept verification information.

3.1.2 Liquid Breeder Blankets

3.1.2.1 Issues and Testing Needs. A number of large uncertainties also exist in the behavior of liquid breeder blankets. Generic issues which encompass the most promising blanket designs are listed in Table III and discussed below.

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

MHD effects are most strongly dependent on the geometry and on a small number of dimensionless parameters, the most important being the Hartmann number (M), the interaction parameter (N), and the wall conductance ratio (C). (The Hartmann number is proportional to the magnetic field and measures the dominance of the MHD force over viscous forces. Similarly, the interaction parameter measures the dominance of the MHD force over inertial forces.) Figure 6 indicates that most existing experimental data are at values of M and N much lower than that found under actual reactor conditions. In addition, most of the data has also been accumulated in very simple geometries. More data is needed for higher values of M and N, and also for geometries more representative of actual blanket configurations.

Material compatibility is a serious 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. 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. Thus, experiments are needed to develop empirical correlations for temperature and impurity limits. 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 study is required to indicate the likelihood of success and the limits of applicability.

Tritium control issues include two major categories: permeation rates and extraction techniques. These two sub-issues are related, since the type of extraction system will be matched to the limits on the tritium release rate. Tritium extraction issues vary widely for different breeding materials and also for different recovery schemes. For lithium, the tritium solubility is high and the partial pressure relatively low. Only partial processing of the coolant stream is required while maintaining acceptable tritium permeation rates. The largest remaining issue is to maintain the total tritium inventory in the coolant within reasonable limits. For LiPb (17Li-83Pb) and Flibe (LiF/BeF_2 salt), the tritium solubility is low and the partial pressures extremely high. In order to prevent large quantities of tritium from escaping the system, a very efficient extraction system must be developed. The entire coolant stream may have to be processed on each pass through the blanket. This is further complicated by a general lack of tritium-related data in 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 behavior.

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 largest 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 behavior 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.

Tritium breeding is not usually considered a feasibility issue for self-cooled liquid breeder blankets. However, if a separate coolant is required or if a design utilizes only a partial coverage breeding blanket (such as no breeding at the inboard side of a tokamak), then fuel self-sufficiency cannot be guaranteed.

3.1.2.2 Existing and Required Experiments/Facilities. Through examination of the key issues and test requirements for liquid breeder blankets, a complete matrix of needed experiments and test facilities has been identified. Figure 7 shows this matrix of tests, including some experiments which are already in progress. The required experiments and facilities are organized according to the classes of issues they resolve and their level of integration. More details on the test plan are provided in Ref. [1].

A range of new experiments have been explored to fulfill the need for further testing of MHD related effects. Figure 8 shows the relationship between the major facilities. Included in this table are two fluid flow/heat transfer facilities (LMF), an MHD mass transfer facility, and two partially integrated test facilities (TMIF and PITF).

Beyond ALEX, 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 behavior 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 fundamental understanding of "microscopic" MHD behavior, especially the velocity profiles, in basic elements of complex geometries. If the electric current distributions and velocity profiles can be predicted, then most of the other "macroscopic" parameters can be derived. LMF1 has a secondary mission to measure temperature profiles, explore heat transfer characteristics, and develop techniques for improved heat transfer and fluid flow. The device parameters of both facilities (i.e., field strength and volume) are designed to be high enough to allow experiments to treat a wide variety of geometric configurations under reactor relevant conditions.

LMF2 has been designed to emphasize "macroscopic" parameters, such as the pressure drop and local heat transfer coefficients. Although in principle these experiments can be conducted in LMF1, practical considerations suggest that a second facility should be devoted to this purpose. Macroscopic measurements serve as a check on the validity of velocity profile models and measurements, and also provide a backup source of data if the velocity profile 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.

While several small corrosion/mass transport loops are now in operation, results from existing experiments will not provide enough information for the development of fusion blankets. More loops will be required for thorough studies of fusion relevant materials. The most critical information required includes dependence on temperature, impurities and magnetic field; loop effects; 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 magnetic fields on mass transport. A particular facility, called the MHD Mass Transfer Facility (MHDM), was defined for this purpose.

Needed experiments related to tritium recovery and control include: 1) basic properties and mechanisms, 2) tritium extraction techniques, 3) permeation and transport loops, and 4) integrated tests of extraction and tritium processing systems. Because tritium behavior is very material-specific, separate experiments will be needed for each potential breeder, including Li, LiPb, and Flibe.

Beyond the first 5-10 years of blanket testing, experiments will become progressively more integrated as they treat a larger number of environmental conditions and components resembling actual reactor blankets. Two types of tests with different missions have been considered for providing engineering data - TMIF and PITF. The exact features of these facilities are difficult to anticipate, since they depend on future experimental results and designs. However, certain key features and objectives have been studied.

TMIF tests the combined influence of heat, mass, and momentum transport issues as well as some structural issues in a non-neutron environment. It will be a larger facility than the early MHD experiments, with 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, chemical control systems, and possibly a model of the tritium extraction system (without tritium).

PITF would be a full- or near-full-scale test which simulates <u>all</u> environmental conditions except neutrons. For liquid breeder blankets, the omission of neutrons results in large cost savings, while many critical issues can still be addressed. These experiments should provide some useful information on failure modes and component reliability. PITF has characteristics similar to TMIF, and may be built as an upgrade of that facility. Partially integrated testing will ensure that when fusion integrated testing of blanket modules is performed, failure modes due to non-neutron effects can be anticipated and eliminated.

3.2 Radiation Shield

The key issues relevant to the radiation shield have been defined by considering the sources of design uncertainties and are presented in Table IV. These are generic issues for the various reactor concepts. Recommendations on the characteristics of experiments and facilities required to resolve these issues are based on an evaluation of the required accuracy of predictions and the status of existing data and design methods. Since neutrons are critical in shielding experiments, the characteristics of point neutron, fission and fusion sources are investigated in Ref. [1].

The types of shielding experiments required include: 1) measurements of differential nuclear data; 2) neutron and gamma ray transport in bulk shield and penetrations, and nuclear responses; and 3) multiple or integral effects on components with complex geometry. Examples of these experiments are given in Ref. [1] for each important issue. Experiments on basic data usually use small specimens; hence, the required volume is small. Experiments on transport phenomena need relatively large volumes; for example, the area should be several mean free path lengths square and the thickness should be deep enough to achieve several orders-of-magnitude attenuation of shielding parameters.

In the next 10-15 years, point or small volume sources will be used to address the issues. There are basically three options (cost estimates are shown in parenthesis):

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

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

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

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

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

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

3.3 Tritium Processing and Extraction Systems

3.3.1 Tritium Processing and Vacuum

Most of the critical technical issues in tritium processing deal with integration of tritium systems or with the interfaces between tritium systems and other systems. These issues include:

1) Tritium Monitoring and Accountability. Two key aspects are the avoidance of neutron and gamma effects on monitors and the present uncertainty of regulatory requirements for accountability.

2) Impurity Removal from Fuels. Key aspects are defining impurity species and concentrations and defining tritium losses in processing.

3) Detritiation of Room Atmospheres and Water Coolant. Key aspects are defining the required cleanup time for room atmospheres and defining the input and required output concentrations for water detritiation systems.

4) Integrated System Behavior. Key aspects are the reliability of complex and interrelated systems during the normal and off-normal operations.

Figure 9 summarizes the experiments and facilities required for the tritium processing and vacuum systems.

3.3.2 Breeder Tritium Extraction

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

Possible process flow schematics and processing methods for each case are summarized in Figure 10 and Table V. The key experimental parameters for studying tritium extraction from each of the carrier fluids (i.e., basic breeder concepts) are given in Ref. [1]. Experiments are needed to explore the feasibility of tritium recovery from the three potential carrier fluids under the sets of conditions listed, and to evaluate the operating characteristics (including reliability and tritium inventory) of the applicable processing systems. These experiments, with few exceptions, do not require neutrons. The experiments are laid out in more detail in Ref. [1].

3.4 Plasma Interactive Components

The plasma interactive components (PIC) of particular concern in nuclear technology are the impurity control and exhaust system and the in-vessel elements of the plasma heating and fueling system (e.g., rf antenna). A major complication in the PIC is the strong interrelation to plasma physics and confinement experiments. This leads to many complex questions in developing a logically consistent and effective test plan for PIC. Limitations of space preclude treatment of this important subject in this paper. Ref. [1] presents results of investigation of the key technical issues and required experiments and facilities.

4. TESTING IN FUSION FACILITIES

4.1 Test Requirements

Some issues, such as failure modes and reliability, require an integrated test with complete components in a fusion environment. In addition, most issues are affected in some way by the combination of all relevant environmental conditions. The only suitable test facility for providing integrated testing is a fusion device. However, fusion test devices are expensive, particularly if reactor conditions are to be provided.

It is possible, in many cases for which the phenomena are sufficiently well-understood, to modify the design of the test module (e.g., coolant flow rate) in order to recover the important aspects of the testing issues, even though the test device parameters are not the same as those of a commercial reactor. However, a change of device parameters beyond certain limits results in the inability to maintain "act-alike" behavior. By analyzing the behavior of components under altered device parameters and by considering methods for scaling the observed behavior to that expected in a reactor, it is possible to identify a set of minimum requirements on the parameters of a fusion test facility in order for it to provide useful testing of nuclear technologies. Such analyses were performed for a range of blanket concepts [1,5]. The resulting requirements are also expected to provide useful testing of the other nuclear components. These requirements are given in Table VI.

From a fusion technology development viewpoint, any fusion device which satisfies these requirements is acceptable.

4.2 Reliability Considerations

Many components in the first fusion engineering facilities will have little or no engineering precedence, particularly nuclear components. Most likely, early fusion engineering facilities would be aimed at improving the nuclear component reliabilities. An apparent paradox results, however, because those nuclear components that would be targeted in a reliability improvement program depend on the reliable performance of other nuclear components in the system.

The implementation of a test program 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 due to the requirement for an extended test period. The INTOR critical issues study [6] concluded that for a given component the achievement of an 80% statistical confidence level in the 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 MTFB.

For blanket modules, since a fusion test facility might have six blanket modules per TF coil sector (60 total), the required availability for individual components might be $(0.6)^{1/60} = 0.9915$ or 99.15%. Since the component availability is given by the MTBF/(MTBF + MTTR), where the MTTR is the mean time to repair or replace, a typical MTTR of 1 month results in a required MTBF of $\sim$ 10 yr. This implies a typical test period of 34 yr. If equal credit can be taken for 60 modules, tested in parallel, however, the required test period would be reduced to a manageable 0.5 FPY.

In addition to testing for confidence in an estimated level of 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 [1].

Two development pathways can be considered. The first pathway would be 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 simple 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 due to the increased likelihood of failure of the required in-situ tritium breeding modules.

Figure 11 shows the calendar time required to achieve a blanket MTBF of 10 yr. (87600 h) for the two pathways, based on the parameters listed in Table VII. 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 over the parameters shown in Table VII indicate that the relative results are not expected to change.

4.3 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, of course, be obtained, but the design is not constrained by the need to provide such data. For example, operating in a driven mode may be acceptable for a technology facility, while ignition is a key goal of physics experiments. It is also possible for the technology test facility to be based on a different device concept than that of a reactor, although reactor relevance is still desirable. These technology-oriented devices are generically referred to here as Fusion Engineering Research Facilities (FERFs).

In this study, fusion test facilities were considered that could plausibly address the nuclear technology test requirements by or around the year 2000. In particular, tokamaks, mirrors and reverse field pinches (RFPs) have been considered as possible FERFs.

The representative engineering test facilities considered were:

1) <u>INTOR (1982 US FED/INTOR)</u>: A conventional reactor-relevant tokamak with ignited operation, inductively-driven current, RF heating and moderate-field superconducting magnets.

2) LITE FERF (TRW/MIT): A driven version of the LITE ignition experiments. The LITE tokamaks incorporate a high-field copper magnet and moderate beta within conventional tokamak physics assumptions.

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

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

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

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

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

The strengths and weaknesses of these concepts as fusion engineering research facilities were compared by characterizing each concept with a short list of distinct parameters which represent the overall attractiveness of each performance (as a test facility) as a function of cost and risk. Since the designs were not necessarily consistent in assumptions or detail, some common assumptions were imposed with respect to availability, duty cycle, useful test area, lifetime, and capital and operating costs. Table VIII gives the performance parameters of representative concepts, and Table IX provides a summary of their overall performance, cost and risk.

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

The major cost parameters are the capital and annual operating costs. The direct capital cost was estimated by comparison with devices costed recently using FEDC/INTOR algorithms and based on the total power handled (electrical plus plasma), and on the fusion core size. Two possible cost-benefit figures-of-merit are also included: the cost per useful neutron (based on the total cost and the annual fluence/area product), and the useful neutrons per unit cost and "risk" (where risk is based on the sum of the physics and technology risk points). These cost-benefit parameters provide some normalization of the data but must be interpreted with due caution.

It is clear from Tables VIII and IX that a variety of possible Fusion Engineering Research Facility concepts exist. All concepts considered provide reasonable performance for technology testing (compared with Table VI). On the other hand, these technology test facilities may not be as costly as a combined physics/technology device, but are still expensive. This is perhaps not surprising since costs are driven by the presence of neutrons and by the overall power level handled. With present concepts, ignited fusion devices (low electrical consumption) generally require high fusion power, while driven fusion devices (low fusion power) generally require high electrical power.

If a technology test 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; INTOR is the largest device and provides considerably more potential test area without a correspondingly large increase in cost, although there may be limited practical utility of test areas over ~ 20 m^2. The spherical torus and reverse field pinch offer relatively low total power, but were also sufficiently small so 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.

In summary, the attractiveness of a particular FERF concept depends strongly on the ability to minimize its total device power (fusion plus electrical), while maintaining a reasonable test area and neutron wall loading.

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

LIST OF FIGURES

FIGURE

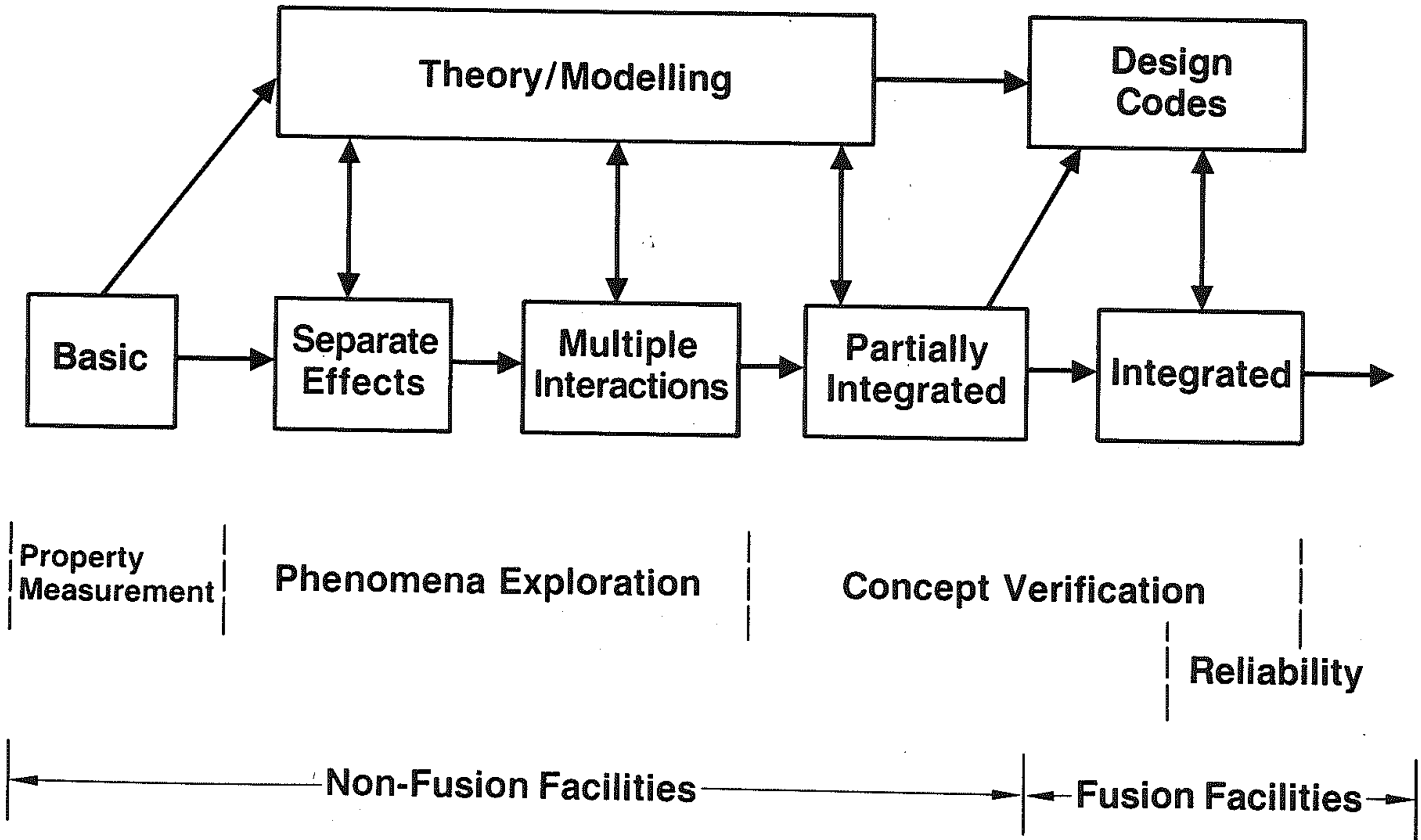


FIGURE 1. Types and role of experiments and facilities for fusion nuclear technology

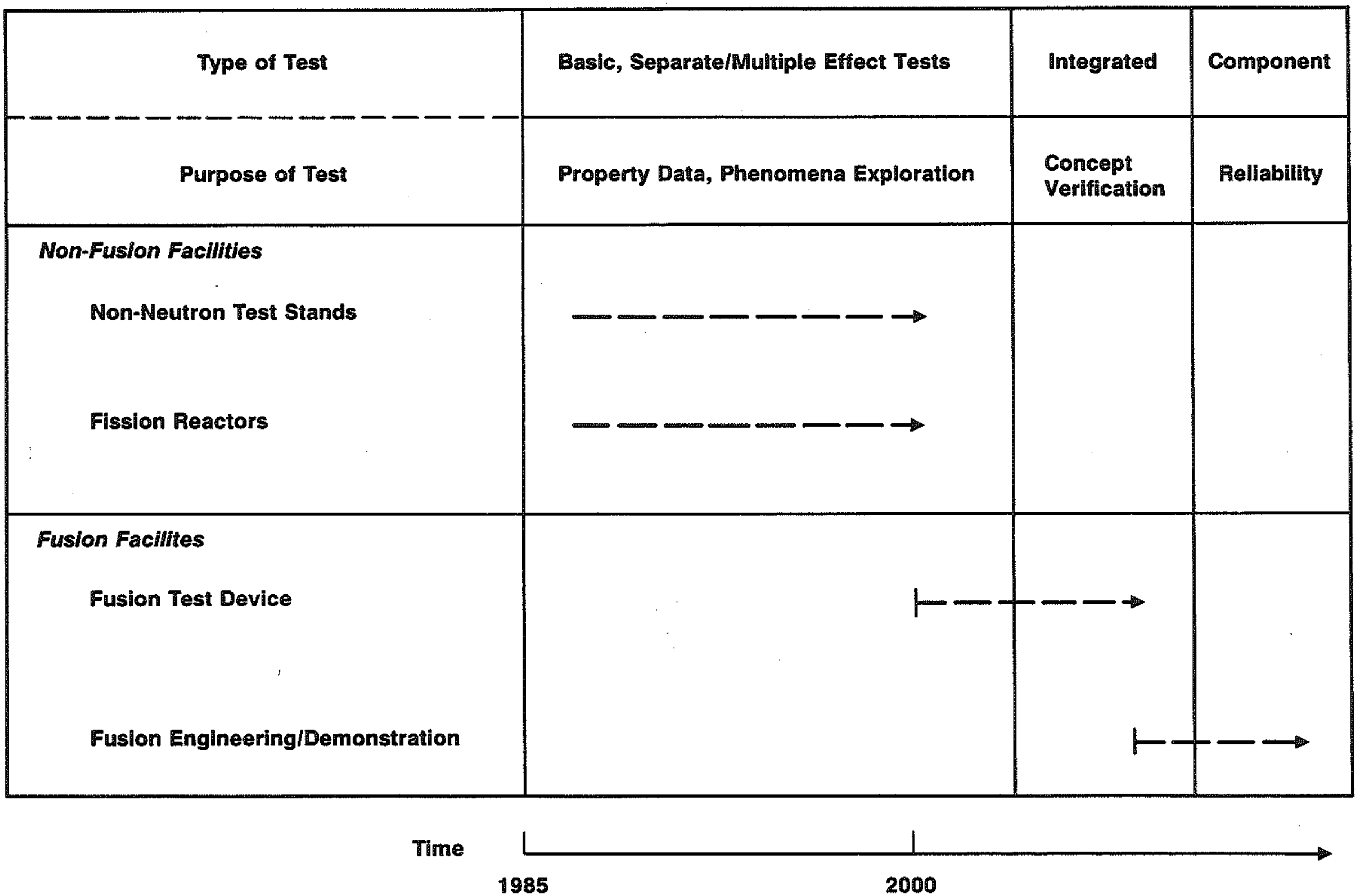


FIGURE 2. Role and timing of non-fusion and fusion facilities for fusion nuclear technology R&D

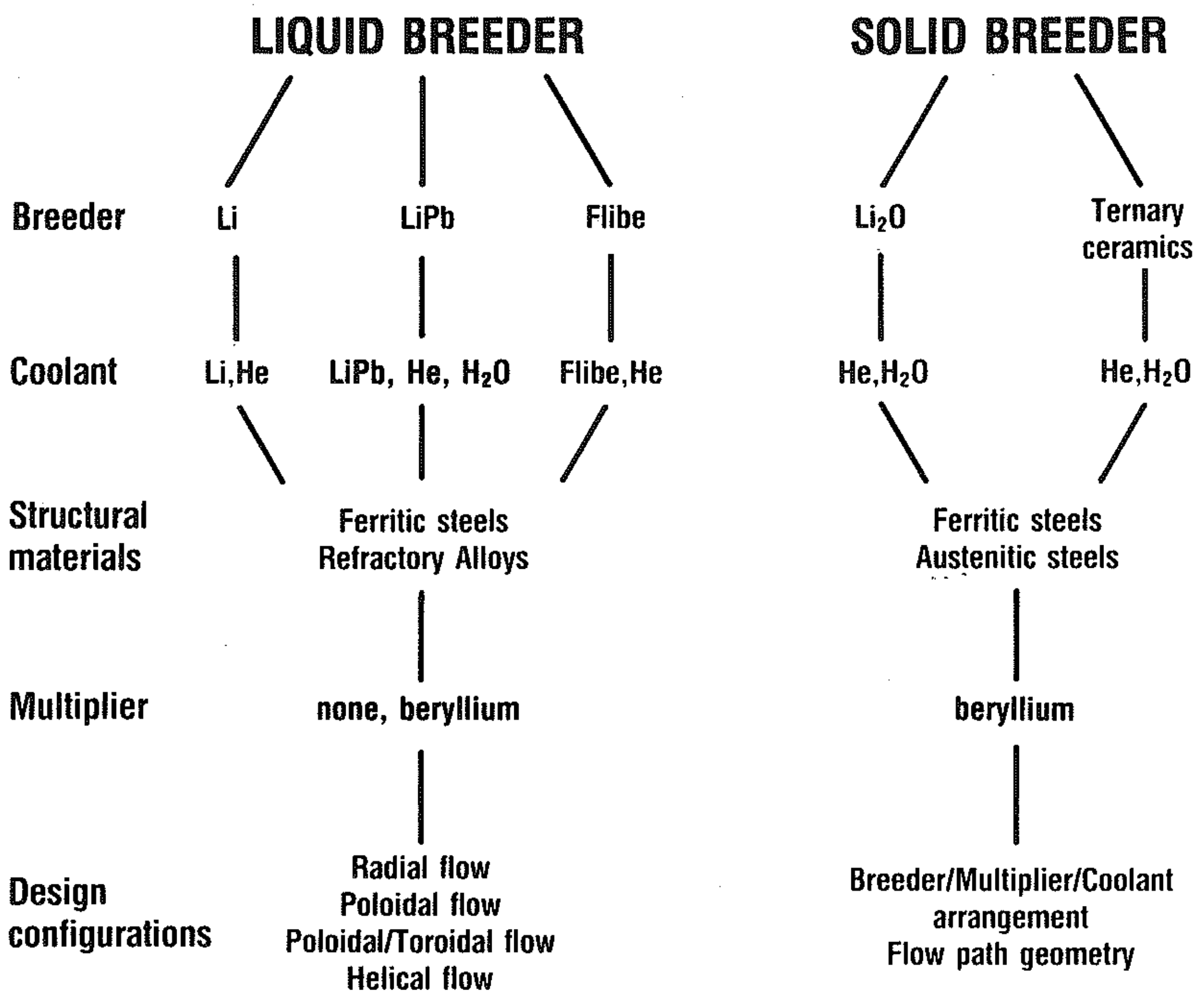


FIGURE 3. Primary blanket options

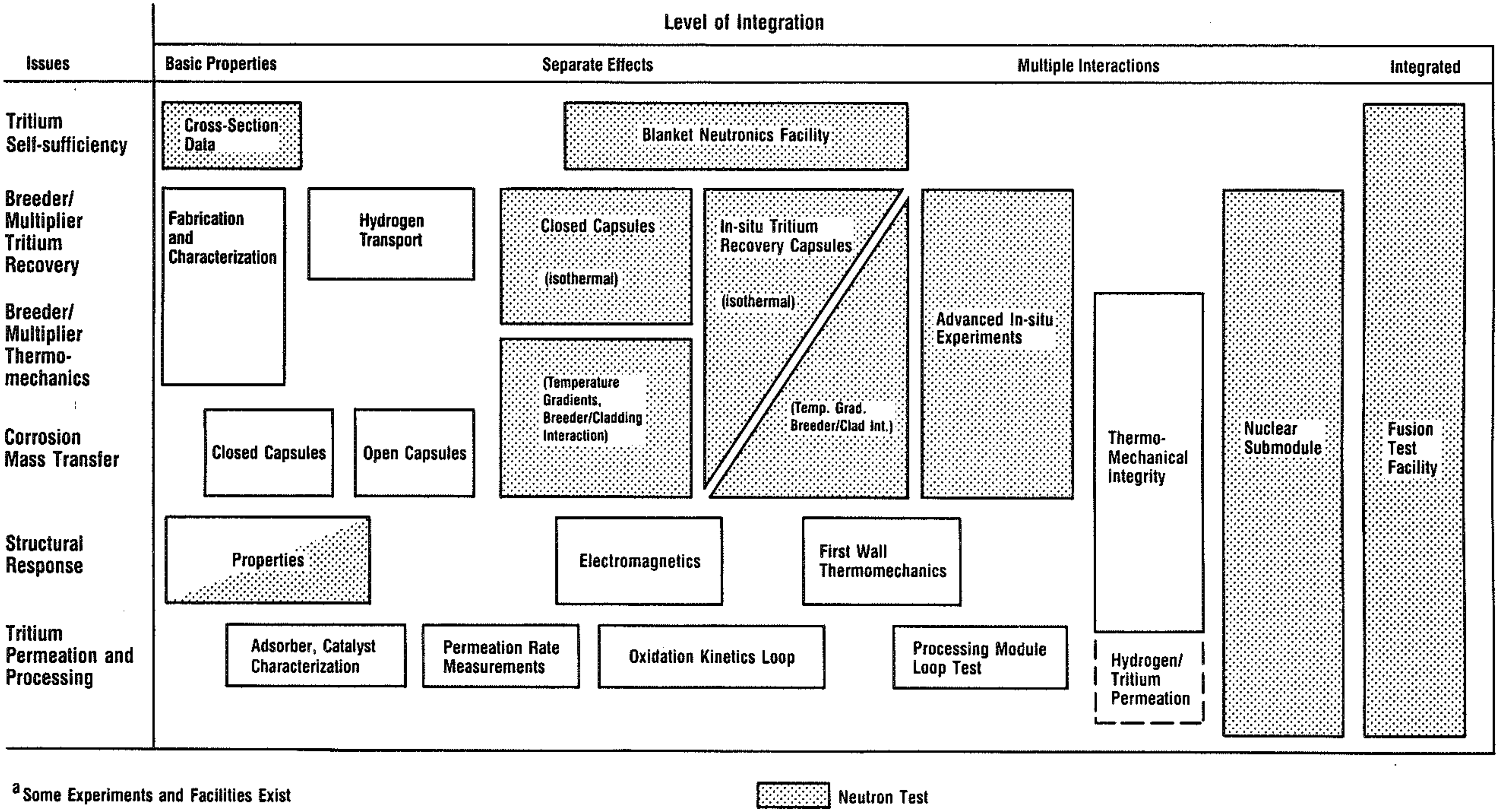


FIGURE 4. Types of experiments and facilities for solid breeder blankets[a]

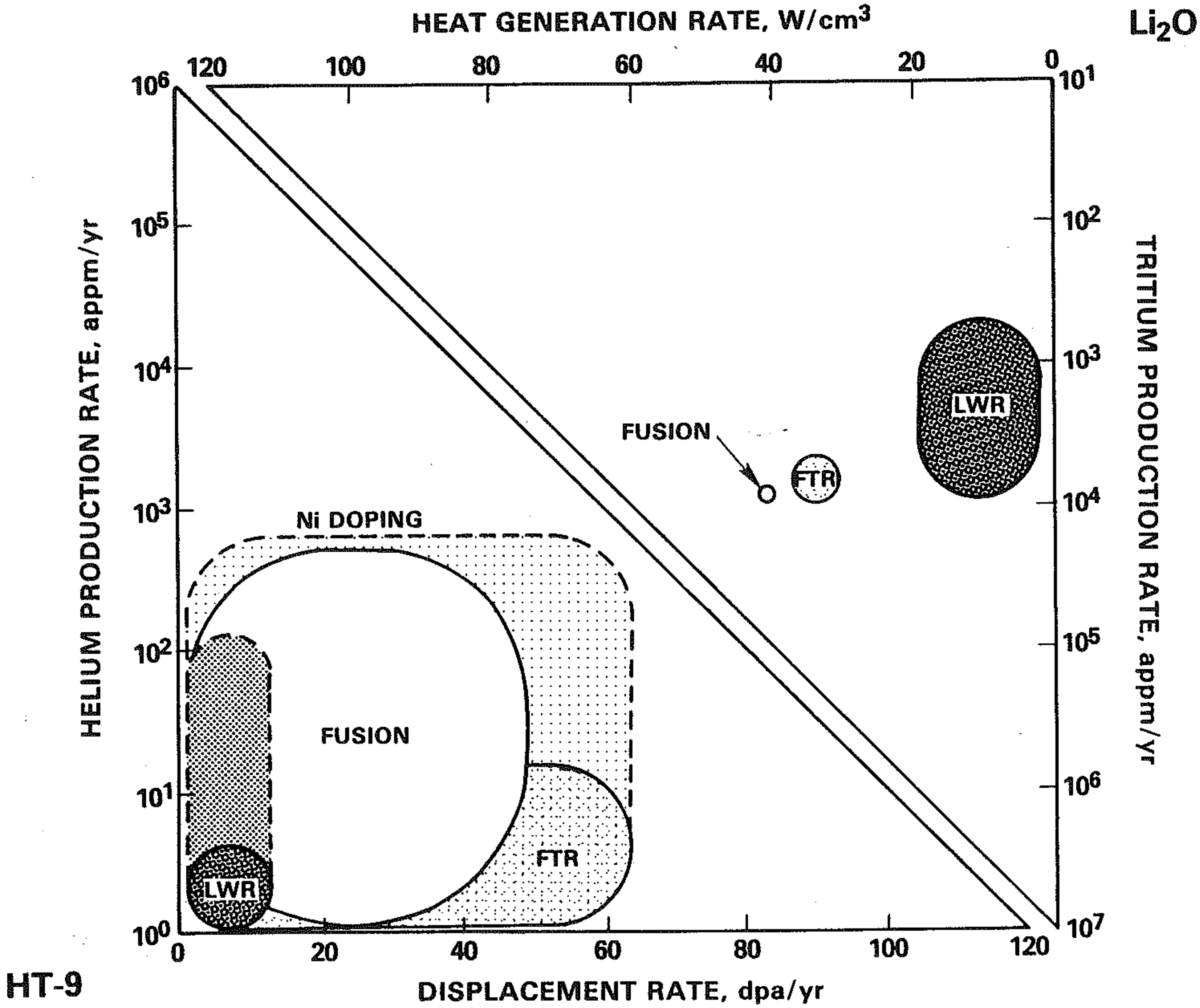


FIGURE 5. Simulation of Li_2O/He/HT-9 fusion blanket in fission reactors

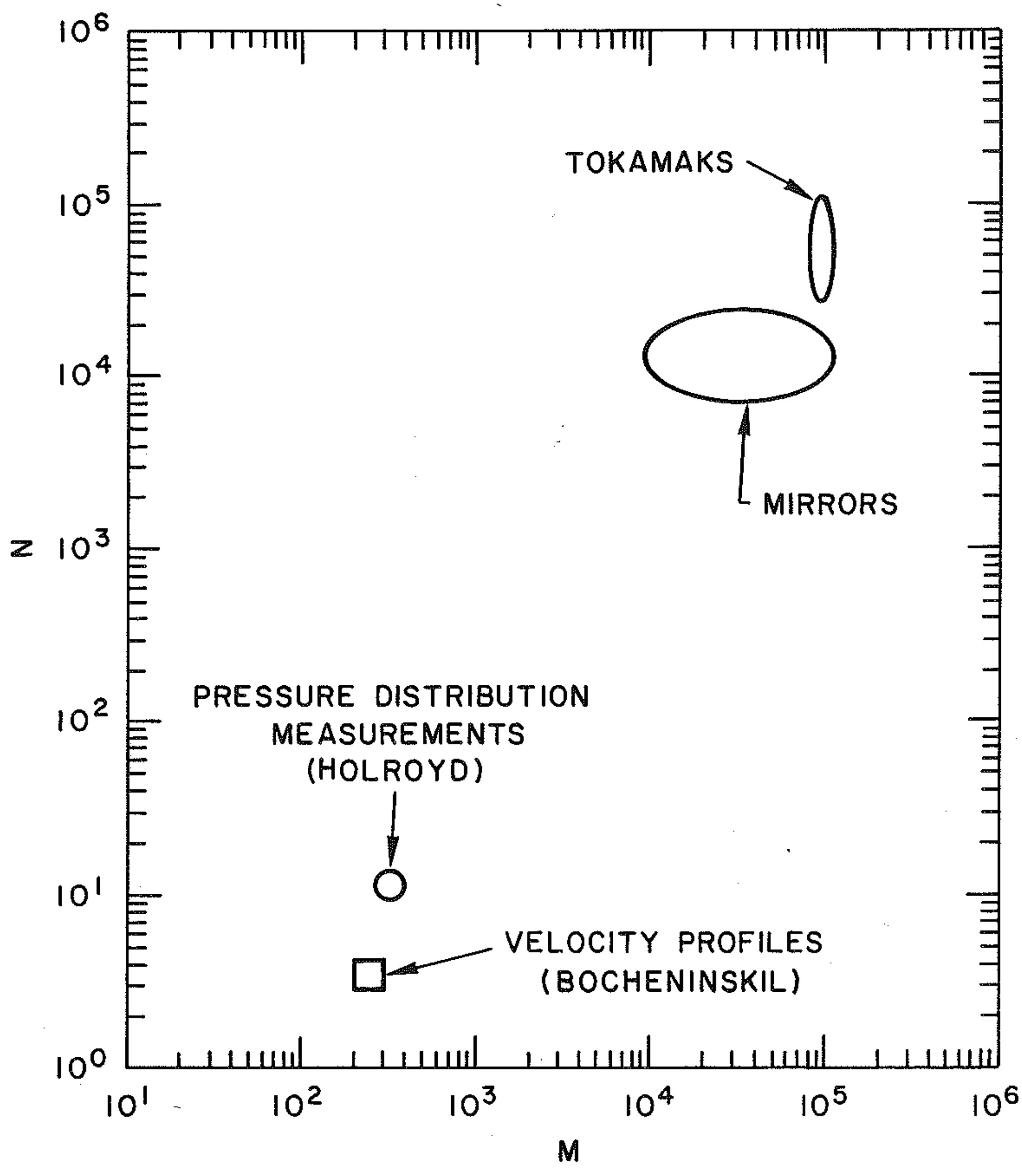


FIGURE 6. Hartmann number (M) and interaction parameter (N) ranges for existing data and reactor conditions

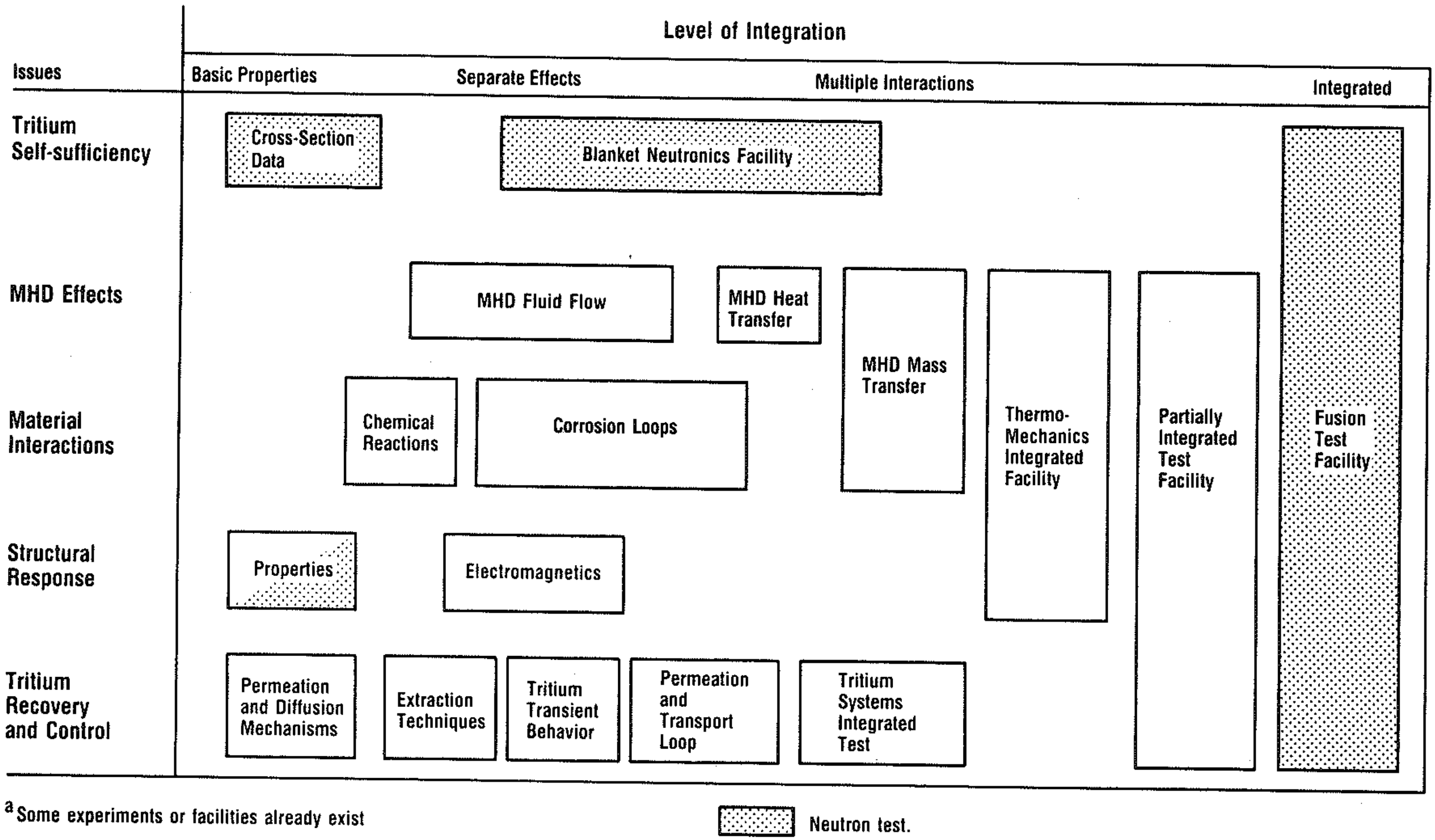
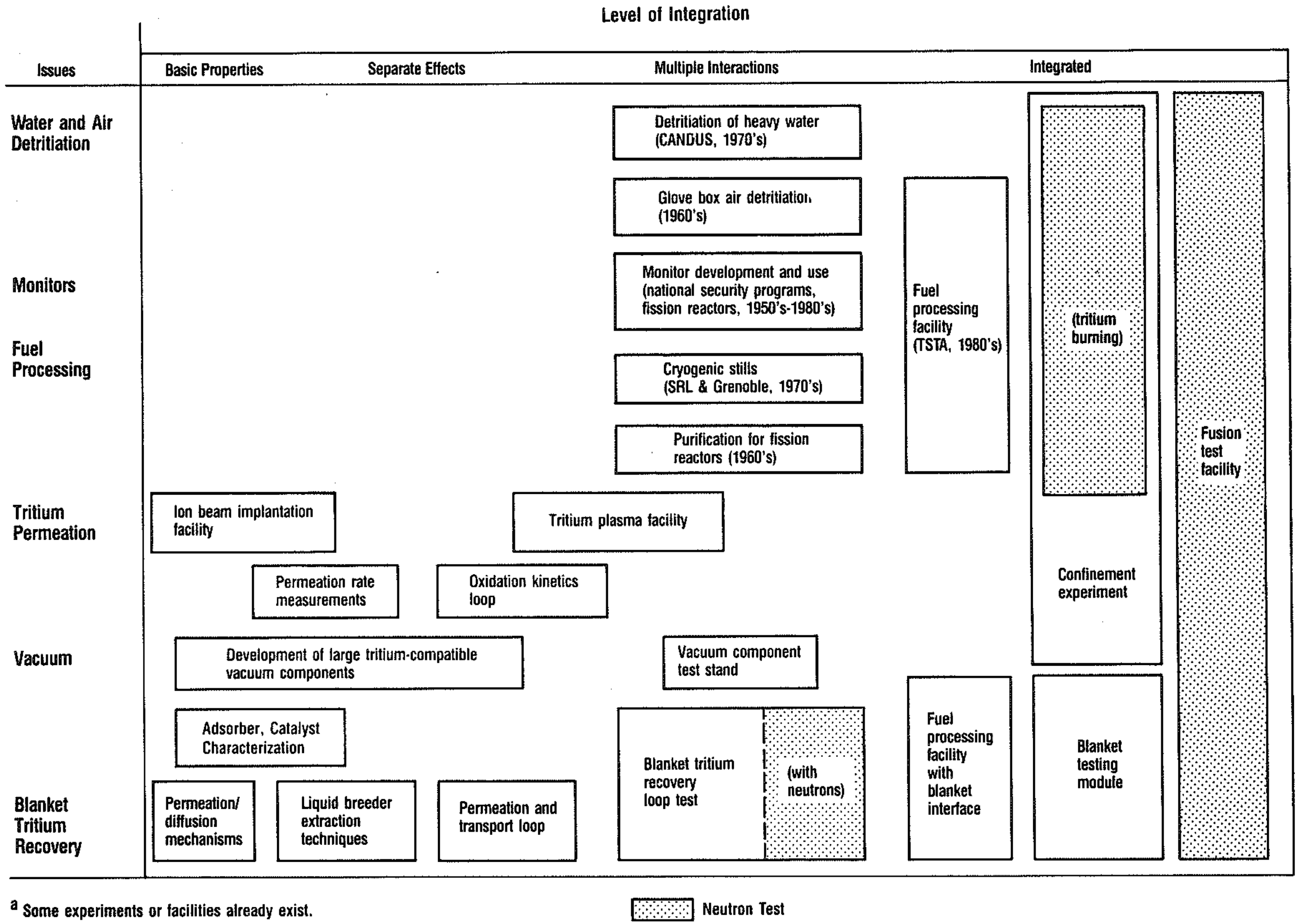


FIGURE 7. Types of experiments and facilities for liquid breeder blankets[a]

	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)

FIGURE 8. Features and objectives of major liquid breeder experiments



[a] Some experiments or facilities already exist.

FIGURE 9. Types of experiments and facilities for tritium processing and vacuum systems[a]

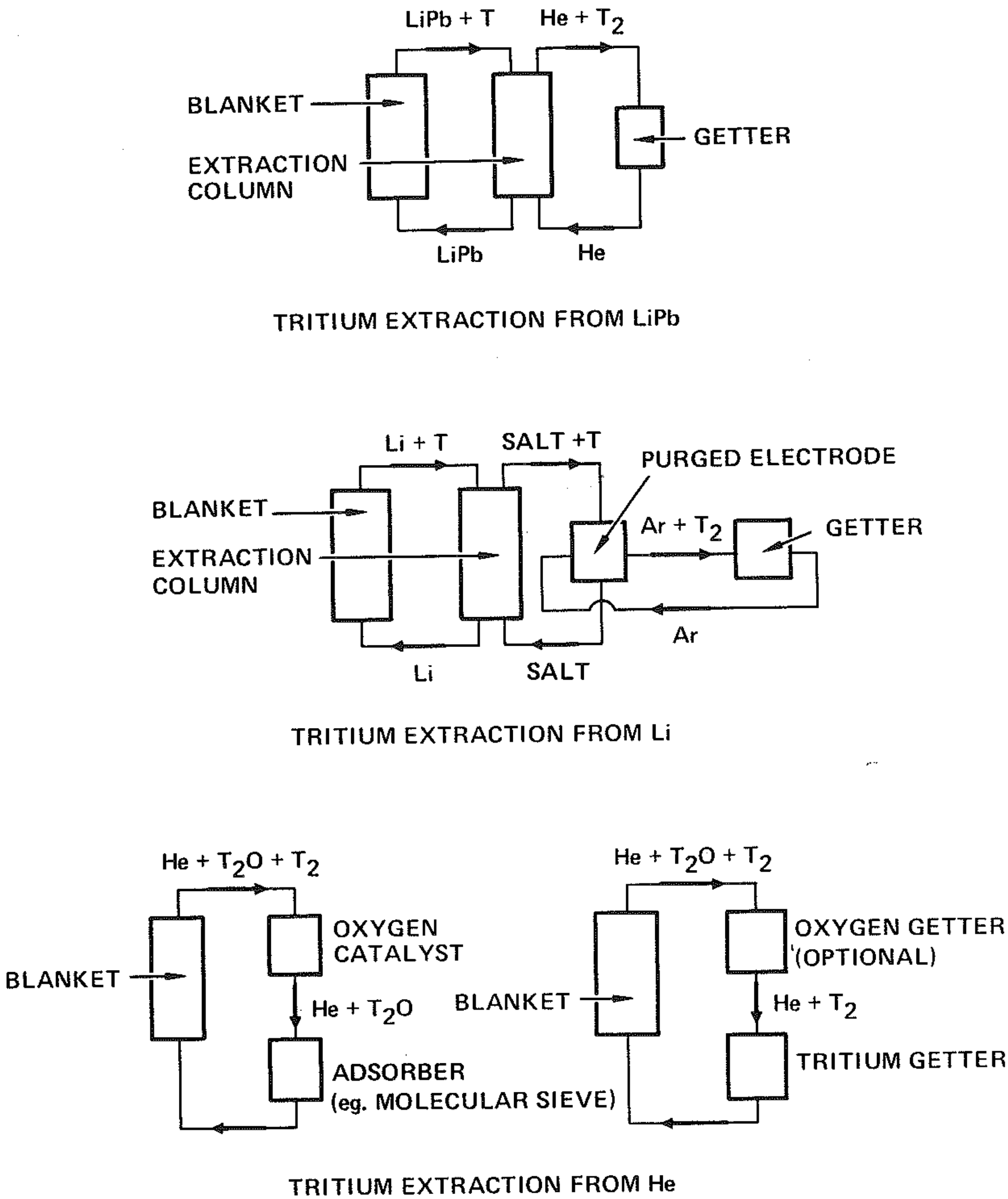


FIGURE 10. Schematic representation of tritium processing schemes

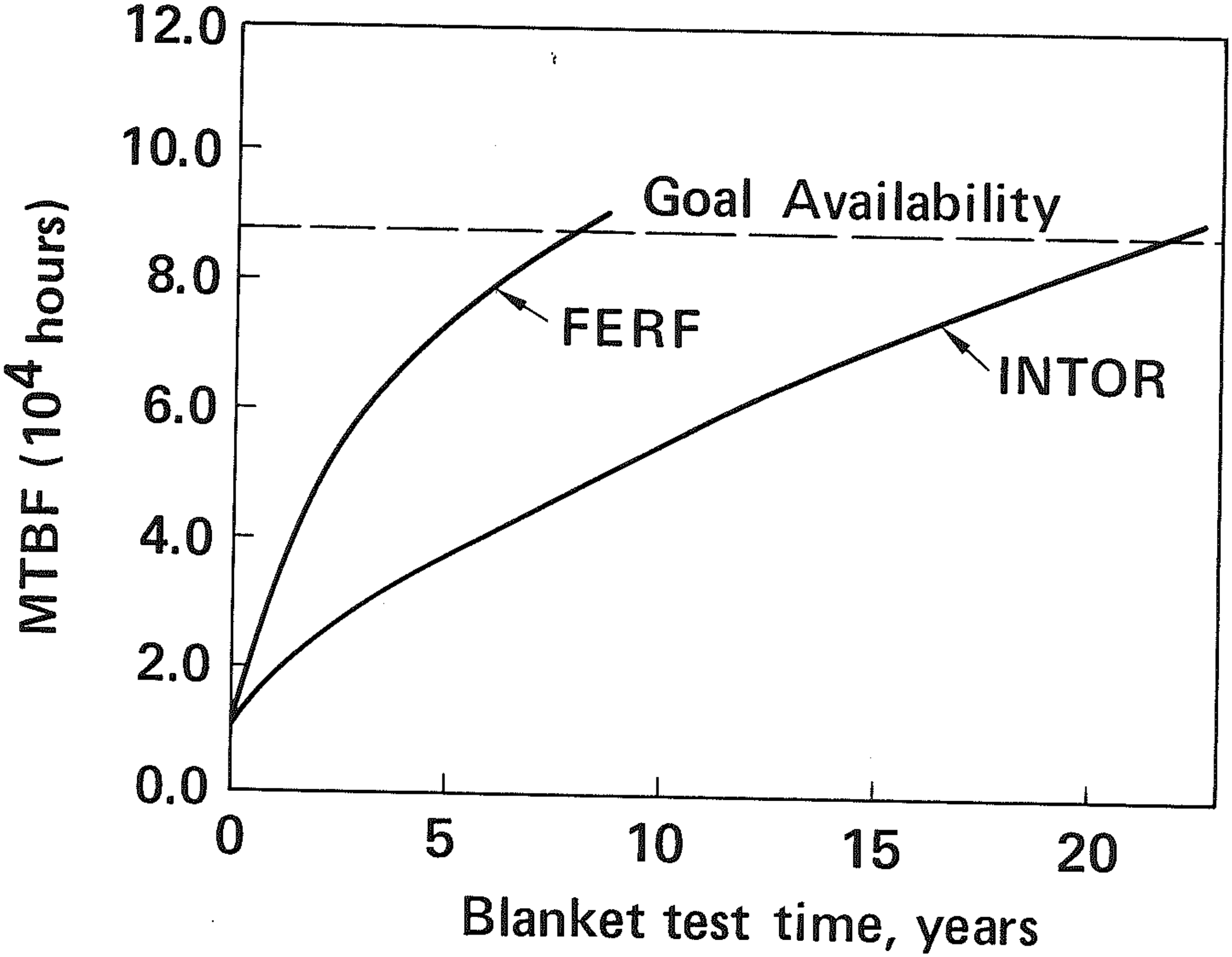


FIGURE 11. Nuclear component reliability growth in high availability (FERF) versus low availability (INTOR) test device

LIST OF TABLES

TABLE I. COMMERCIAL REACTOR PARAMETERS CONSIDERED IN FINESSE AS REPRESENTATIVE GOAL RANGES

Parameter	Representative Range
Neutron wall load, MW/m^2	4-6
Surface heat flux at first wall, MW/m^2	0.2-1
Average heat flux in high heat flux components (e.g., limiter/divertor), MW/m^2	5-10
Plasma burn time	very long/ continuous
Magnetic field strength in blanket region, T	5-7
Reactor availability, %	80
First wall/blanket lifetime fluence, $MW{\cdot}y/m^2$	15-20

TABLE II. GENERIC SOLID BREEDER BLANKET ISSUES

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

TABLE III. GENERIC LIQUID BREEDER BLANKET ISSUES

- Magnetohydrodynamic (MHD) effects
  - Fluid flow (including pressure drop)
  - Heat transfer
- Material interactions (e.g., corrosion)
- Tritium recovery and control
- Structural response in the fusion environment
  - Irradiation effects on material properties
  - Response to complex loading conditions
  - Failure modes
- Tritium self-sufficiency

TABLE IV. RADIATION SHIELD ISSUES

Radiation protection criteria of sensitive components (superconducting magnets, vacuum equipment, plasma heating systems and control system)
Effectiveness of bulk shield
- composition, thickness of shield materials
- deep penetration of high energy neutrons (14 MeV) including cross-section windows
Effectiveness of penetration shielding
- streaming and partial shield
- modeling procedure
Occupational exposure
- induced activity and dose distribution
- radioactive corrosion materials
- remote maintenance system
Public exposure and waste management
- sky shine
- radioactive waste of shield materials
Shield compatibility with blanket heat transport system and magnet, including assembly/disassembly and magnetic field penetration

TABLE V. 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		<u>Extraction with molten salts</u>	7	None
	X		Absorption with solid getters	—	
LiPb	X X		<u>Extraction with counter-current He flow</u>	0.01-1	None
			Vacuum degassing	—	
	X		Permeation combined with catalytic oxidation	—	
He	X	X[c]	<u>Absorption with solid getters</u>	10^{-5}	Fuel clean-up
	X[d]	X	<u>Adsorption with molecular sieves</u>	10^{-5}	Air detritiation
	X[d]	X	Freezing out in cold traps	—	
H_2O		X	Vapor phase catalytic exchange	0.6	CANDU reactor coolant clean-up
		X	Liquid phase catalytic exchange	—	
		X	Electrolysis	—	

[a]Preferred method underlined.

[b]Tritium concentration at processing system outlet. This value is very dependent on design and cost tradeoffs. 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 VI. REQUIREMENTS FOR FUSION INTEGRATED TESTING

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

TABLE VII. KEY ASSUMPTIONS IN THE AVAILABILITY ANALYSIS

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

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

TABLE VIII. PERFORMANCE COMPARISON OF FUSION ENGINEERING RESEARCH FACILITIES

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

[a]Consistent estimate.

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

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

TABLE IX. SUMMARY CHARACTERISTICS OF FUSION ENGINEERING RESEARCH FACILITIES

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

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

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

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

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