TECHNICAL REQUIREMENTS OF EXPERIMENTS AND FACILITIES FOR FUSION NUCLEAR TECHNOLOGY[a]

M. ABDOU (UCLA), P. GIERSZEWSKI (CFFTP), M. TILLACK (UCLA),
J. GROVER (HEDL), R. PUIGH (HEDL), D.K. SZE (ANL), D. BERWALD (TRW, Inc.)
University of California, Los Angeles
(213) 206-0501

ABSTRACT

The technical issues and requirements of experiments and facilities for fusion nuclear technology (FNT) have been investigated. 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. A technical planning process for FNT has been developed and applied, including four major elements: 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.

I. INTRODUCTION

A process has been developed and applied to 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: a) blanket, including the first wall; b) radiation shield; c) tritium processing system; and d) plasma interactive components. Particular emphasis has been placed on the complex technical issues and development problems of the blanket.

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

[a]Work supported by U.S. Department of Energy under Grant No. DE-FG03-86ER52123.

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

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., wall load, plasma burn time); these requirements have been quantified. The second phase of testing in fusion facilities

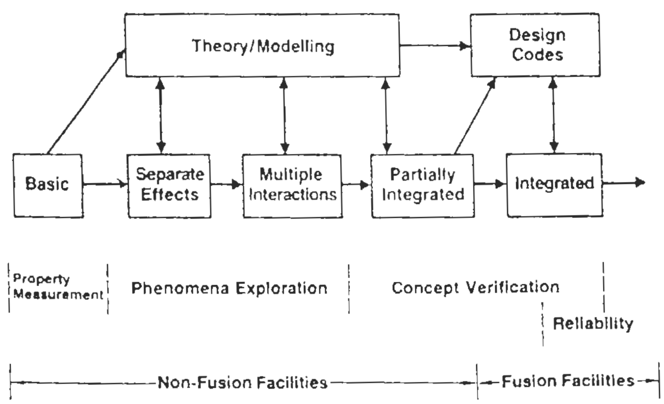


Fig. 1. Types and role of experiments and facilities for fusion nuclear technology

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.

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 two generic classes: 1) solid breeder blankets, and 2) liquid breeder blankets. Within each class there are a number of design concepts that involve a variety of material and configuration choices (see Fig. 2). Both classes have significant engineering feasibility uncertainties, and so both liquid and solid breeders should be pursued. 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

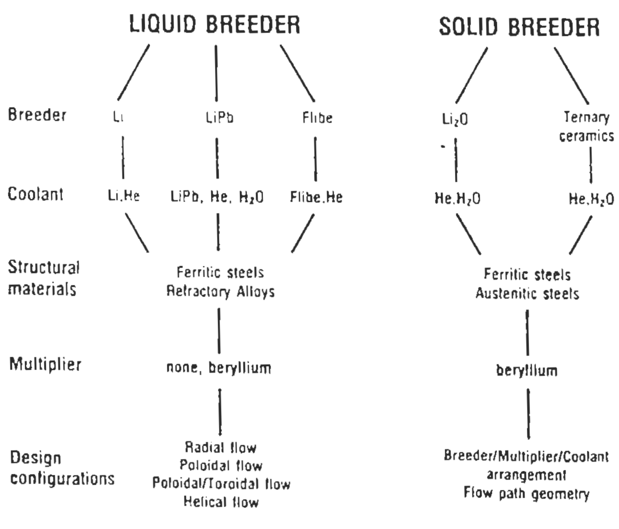


Fig. 2. Primary blanket options

for the next 15 years has been summarized in terms of a number of major tasks. Each task consists of a number of facilities, experiments, and related activities aimed at resolving one or more of the critical issues.

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

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

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

A major complication in plasma-interactive components is the strong interrelation to both plasma physics and nuclear technology. The suggested experiments span several technical disciplines such as surface physics, plasma confinement, thermal hydraulics, material irradiation and fabrication.

The key features of the test plan developed for fusion nuclear components are based on technical analysis of issues and experimental requirements, and evaluation of the capabilities and limitations of facilities. Estimates of cost and time required to resolve the key issues suggest a strong need for considerable enhancement of fusion nuclear technology programs.

II. BLANKET/FIRST WALL

II.A Solid Breeder Blankets

The most important uncertainties for solid breeder blankets are related to tritium breeding, tritium recovery, and breeder thermomechanical behavior (see Table 1). 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. The characteristics of these issues have been examined in detail in Ref. [1].

Table 1. Generic Solid Breeder Blanket Issues

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

The issues can be addressed by a range of possible experiments as summarized in Fig. 3 and discussed below. The figure presents the required experiments and facilities in a logical sequence which accounts for several factors important to test planning, such as the time to acquire and utilize data, the need for informa-

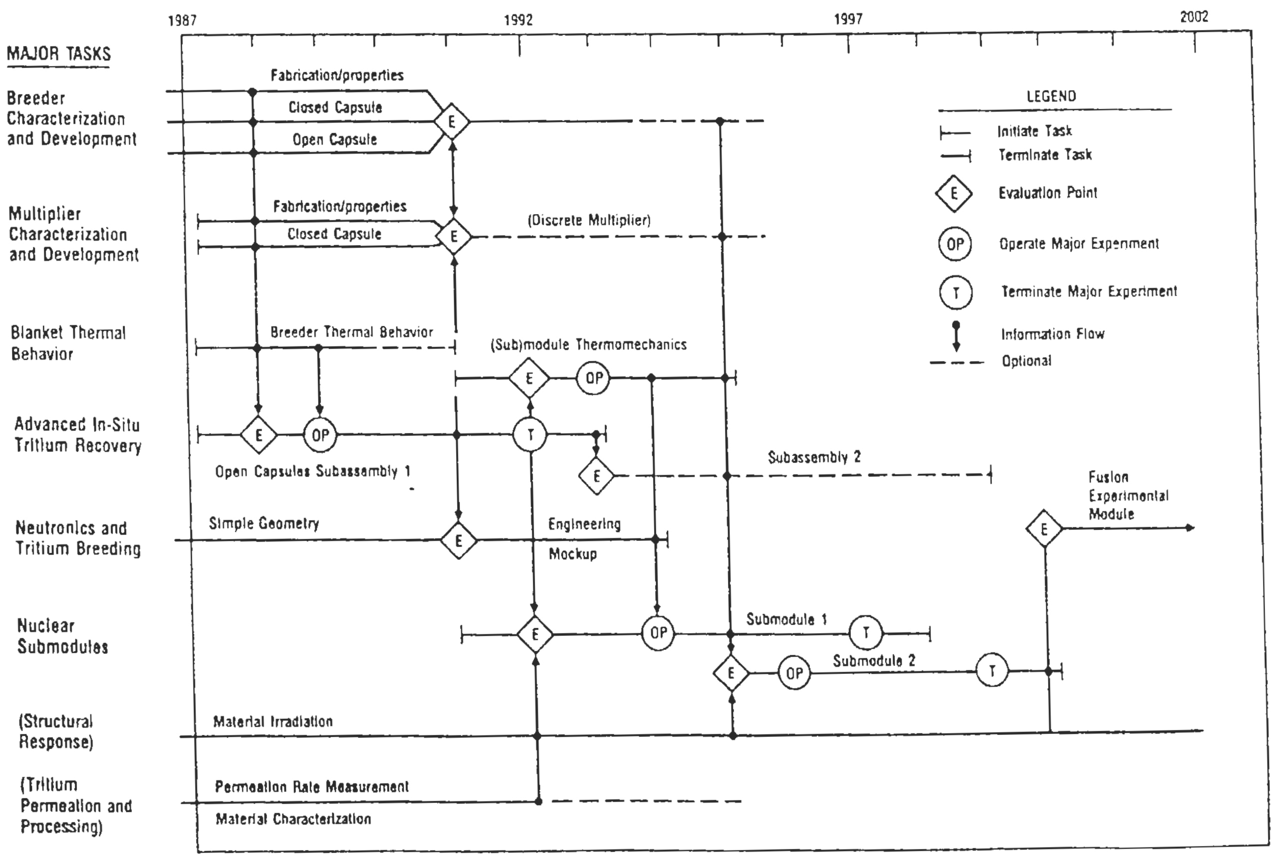


Fig. 3. Test sequence for major solid breeder blanket tasks

tion from prior experiments, and balanced yearly expenditures. 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 4 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. Figure 4 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

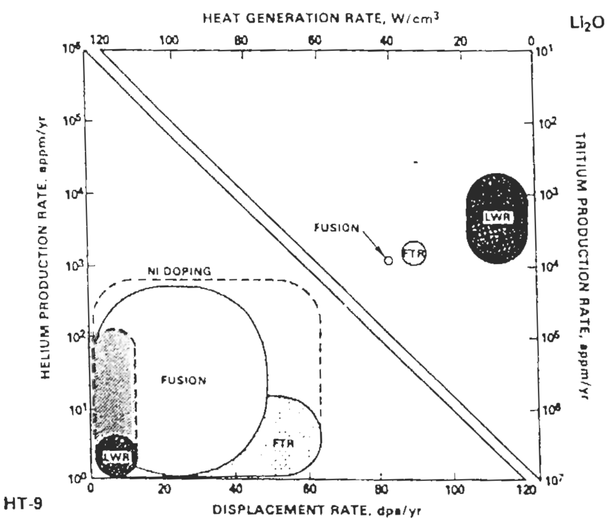


Fig. 4. Simulation of Li_2O/He/HT-9 fusion blanket in fission reactors

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 sub-sections. 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 (thermal 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.[2] Experiments are also being performed or planned in other countries. 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

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

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

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.

The tasks described above are summarized in Table 2 and total costs are estimated. These numbers are intended as total program costs--they represent all of the costs associated with the experimental program, including construction, experimental hardware, staff, overhead, etc. The costs in the table are broken down into two categories: capital and operating costs. Capital costs include design effort, materials, fabrication, construction and any expense directly related to the construction of the facility and the experamental apparatus. Annual operating costs include use of materials and energy, staff to operate the experiments, and data acquisition. The cost of analysis, modeling efforts, blanket design studies, etc. have not been included as operating expenses. These are listed in Table 2 as separate items. The solid breeder blanket program requires an average annual expenditure of about 10-20 million dollars.

II.B Liquid Breeder Blankets

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 3 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. Material compatibility is a serious concern for nearly every liquid breeder blanket design; the nature and importance of the issues depend strongly on the materials. A variety of phenomena relating to both mass transfer and structural degradation are involved.

Table 4. Representative Costs of Key Liquid Breeder Blanket Facilities

Item	Capital Cost[a] (M$)	Operating Cost (M$/yr)	Duration (years)	Total Cost (M$)
Advanced liquid metal flow facility (LMF1)	7-10	0.5	4-6	10-15
Integral Parameter Experiment (LMF2)	7-10	0.5	4-6	10-15
MHD mass transfer facility (MHDM)	8-12	1.0	6-8	15-20
Corrosion loops (~ 8-10)	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 dollars

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.

Table 5. 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 and disassembly and magnetic field penetration

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):

- modification 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 can be performed in a low fluence (~ 1 MW·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.

IV. TRITIUM PROCESSING AND EXTRACTION SYSTEMS

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.

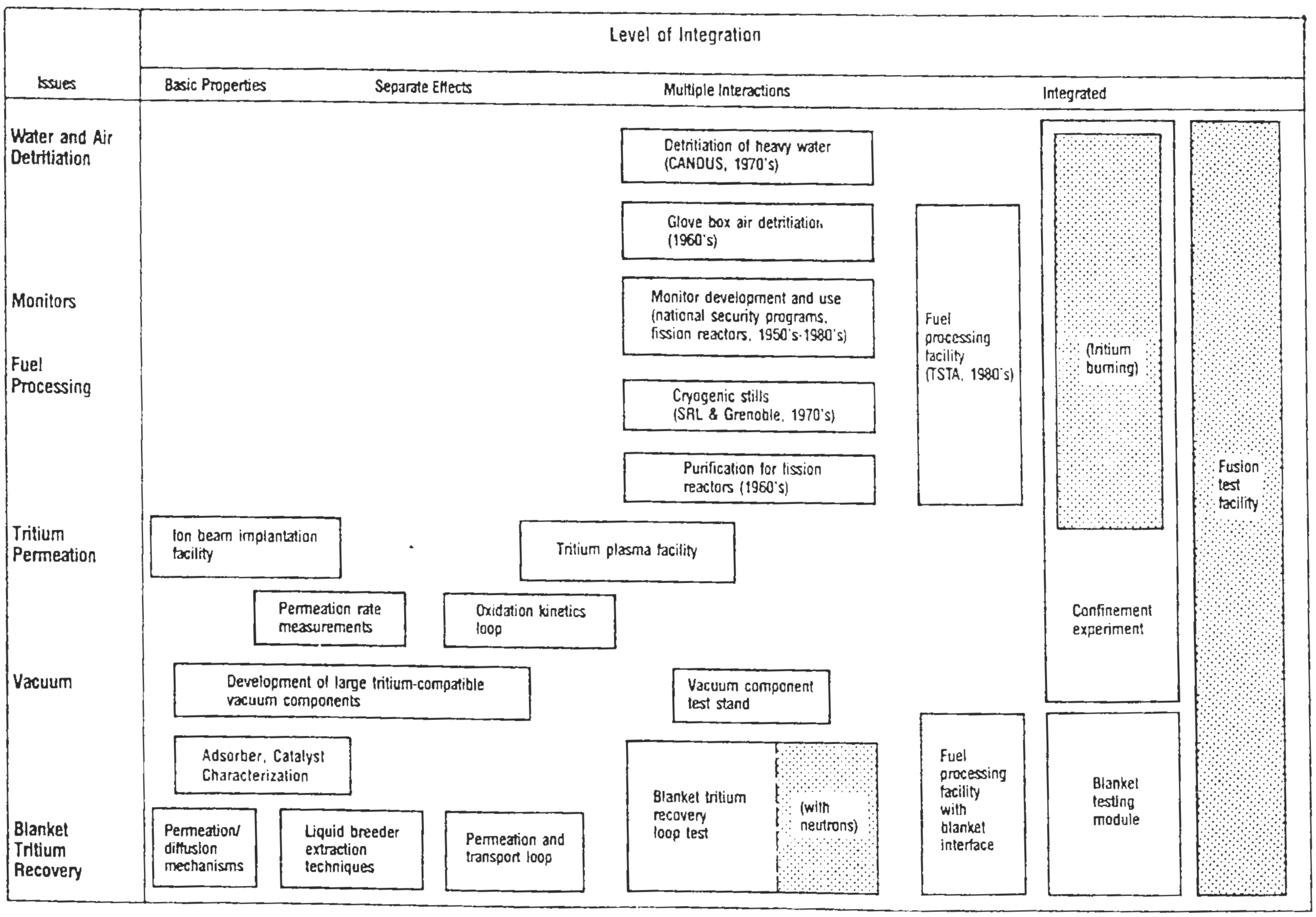
3) Detritiation of Room Atmospheres and Water Coolant. Key aspects are defining the required cleanup time for room atmospheres and defining the input and required output concentrations for water detritiation systems.

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

Figure 6 summarizes the experiments and facilities required for the tritium processing and vacuum systems.

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

Possible process flow schematics and processing methods for each case are summarized in Fig. 7 and Table 6. The key experimental parameters for studying tritium extraction from each of the carrier fluids (i.e., basic breeder



[a] Some experiments or facilities already exist.

Fig. 6. Types of experiments and facilities for tritium processing and vacuum systems[a]

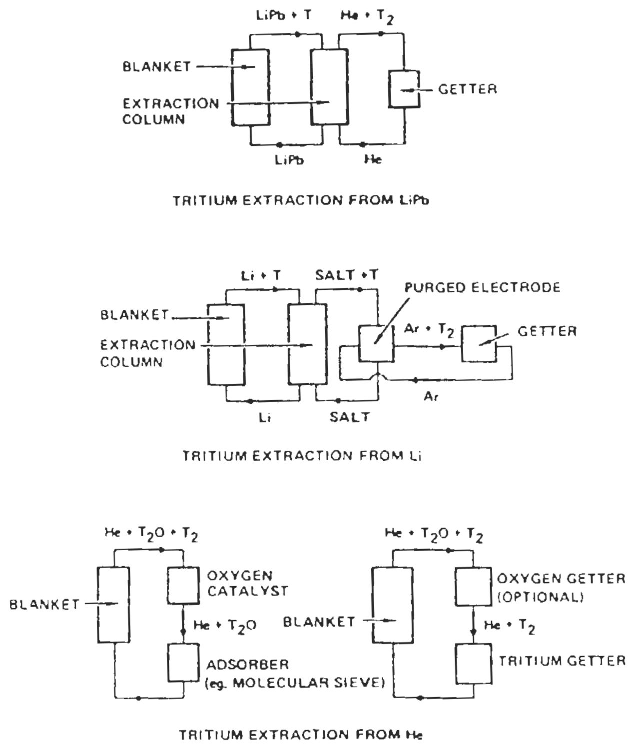


Fig. 7. Schematic representation of of tritium processing schemes

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].

V. 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. Reference [1] presents results of investigation of the key technical issues and required experiments and facilities.

Table 6. Tritium Processing Methods for Different Tritium Carrier Fluids

T Carrier Fluid	Tritium Form T_2/HT	Tritium Form T_2O/HTO	Extraction Method[a]	Outlet T Concentration[b] X_p (appm)	Other Application of Method
Li	X		Extraction with molten salts	7	None
	X		Absorption with solid getters	--	
LiPb	X X		Extraction with counter-current 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 clean-up
	X[d]	X	Adsorption with molecular sieves	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.

VI. TESTING IN FUSION FACILITIES

VI.A. 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.[1,3] The resulting requirements are also expected to provide useful testing of the other nuclear components. These requirements are given in Table 7.

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

VI.B. 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[4] 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 ratio MTBF/(MTBF+MTTR), where the MTTR is the mean time to repair or replace, a typical MTTR of 1 month results in a required MTBF of ~ 10 yr. This implies a typical test period of 34 yr. If equal credit can be taken for 60 modules, tested in parallel, however, the required test period would be reduced to a manageable 0.5 FPY.

Table 7. Requirements for Fusion Integrated Testing

Parameter	Reference Reactor	Test Facility Parameter	
		Minimum	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

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 8 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 8. 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 8 indicate that the relative results are not expected to change.

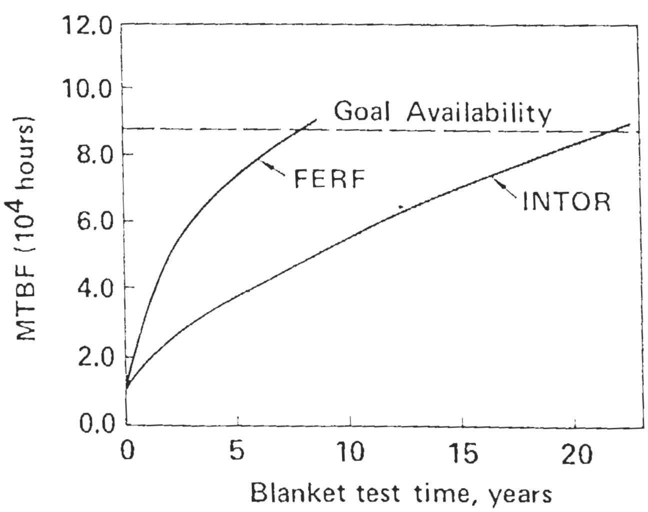


Fig. 8. Nuclear component reliability growth in high availability (FERF) versus low low availability (INTOR) test device

Table 8. 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, $0 \leq n \leq 1$.

VI.C. 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) INTOR (1982 U.S. FED/INTOR)
2) LITE FERF (TRW/MIT)
3) "BEAN" FERF (PPPL)
4) IDT-DTFC (Energy Applications and Systems, Inc.)
5) ST FERF (FEDC)
6) TDF and MFTF-α+T (LLNL)
7) RFP FERF (LANL/Phillips Petroleum)

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 9 gives the performance parameters of representative concepts, Table 10 provides a summary of their overall performance, cost and risk.

Summary risk parameters are desirable to represent "over-all" 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 recently costed 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 9 and 10 that a variety of possible Fusion Engineering Research Facility concepts exist. All concepts considered provide reasonable performance for technology testing (compared with Table 7). 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

Table 9. Performance Comparison of Fusion Engineering Research Facilities

	Tokamaks				Spherical Torus	Tandem Mirrors		Reverse Field Pinch
	INTOR	LITE FERF	BEAN FERF	DTFC-IDT	FERF	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-yr/m^2	3.3	4.0	4.7	7.3	4.0	8.5	8.1	4.0-20
External field on-axis, T	5.5	5.5	3-6	8	3	4.5	4.5	7-9

[a]Consistent estimate.

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

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

Table 10. Summary Characteristics of Fusion Engineering Research Facilities

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

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

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

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

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

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.

REFERENCES

1. M.A. ABDOU, et al., "Technical Issues and Requirements of Experiments and Facilities for Fusion Nuclear Technology (FINESSE Phase I Report)" University of California, Los Angeles, PPG-909, also UCLA-ENG-85-39, December 1985. See also M.A. Abdou, et al., "Technical Issues and Requirements of Experiments and Facilities for Fusion Nuclear Technology," to be published in Nuclear Fusion.

2. T. NAKAMURA and M. ABDOU, "Summary of Recent Results from the JAERI/U.S. Fusion Neutronics Phase I Experiments," to be presented at the 7th Topical Meeting on the Technology of Fusion Energy, Reno, NV, June 1986; also to be published in Fusion Technology.

3. M. ABDOU, P. GIERSZEWSKI, G.D. MORGAN, Eds., "Report on the International Workshop on Fusion Nuclear Technology Testing and Facilities (UCLA, March 10-13, 1985)," University of California, Los Angeles, PPG-872, also UCLA-ENG-85-17 and UCLA-FNT-5, March 1985.

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