ANS Reprint

BLANKET ENGINEERING

KEYWORDS: *volumetric neutron source, blanket, nuclear testing*

RESULTS OF AN INTERNATIONAL STUDY ON A HIGH-VOLUME PLASMA-BASED NEUTRON SOURCE FOR FUSION BLANKET DEVELOPMENT

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Received February 2, 1995
Accepted for Publication April 20, 1995

An international study conducted by technical experts from Europe, Japan, Russia, and the United States has evaluated the technical issues and the required testing facilities for the development of fusion blanket/first-wall systems and has found that some of the key requirements for the engineering feasibility of blanket concepts cannot be established prior to extensive testing in the fusion environment. However, because of availability and low cost, testing in nonfusion facilities (e.g., fission reactors and laboratory experiments) serves a critical role in blanket research and development (R&D) and reduces the risks and costs of the more complex and expensive fusion experiments. A comprehensive analysis shows that the fusion testing requirements for meeting the goal of demonstrating a blanket system availability in DEMO > 50% are as follows: a 1 to 2 MW/m^2 neutron wall load, a steady-state plasma operation, a >10-m^2 test area, and a fluence of >6 MW·yr/m^2. This testing fluence includes 1 to 3 MW·yr/m^2 for concept performance verification and >4 to 6 MW·yr/m^2 for component engineering development and reliability growth/demonstration. Reliability and availability analyses reveal critical concerns that need to be addressed in fusion power development. For a DEMO reactor availability goal of 50%, the blanket availability needs to be ~80%. For a mean time to recover from a failure of ~3 months, the mean time between failure (MTBF) for the entire blanket must be >1 yr. For a blanket that has 80 modules, the corresponding MTBF per module is 80 yr. These are very ambitious goals that require an aggressive fusion technology development program. A number of scenarios for fusion facilities were evaluated using a cost/benefit/risk analysis approach. Blanket tests in the International Thermonuclear Experimental Reactor (ITER) alone with a fluence of 1 MW·yr/m^2 can address most of the needs for concept verification, but it cannot adequately address the blanket component reliability growth/demonstration testing requirements. An effective path to fusion DEMO is suggested. It involves two parallel facilities: (a) ITER to provide data on plasma performance, plasma support technology, and system integration and (b) a high-volume plasma-based neutron source (HVPNS) dedicated to testing, developing, and qualifying fusion nuclear components and material combinations for DEMO. For HVPNS to be attractive and cost effective, its capital cost must be significantly lower than ITER, and it should have low fusion power (~150 MW). Exploratory studies indicate the presence of a design window with a highly driven plasma. A testing and development strategy that includes HVPNS would decisively reduce the high risk of initial DEMO operation with a poor blanket system availability and would make it possible—if operated parallel to the ITER basic performance phase—to meet the goal of DEMO operation by the year 2025. Such a scenario with HVPNS parallel to ITER provides substantial savings in the overall R&D cost toward DEMO compared with an ITER-alone strategy. The near-term cost burden is negligible in the context of an international fusion program with HVPNS and ITER sited in two different countries.

I. INTRODUCTION

In early 1994, the International Energy Agency (IEA) decided to initiate an international study on a high-volume plasma-based neutron source (HVPNS) for fusion blanket development. An international group was assembled with representatives from the European Union, Japan, the Russian Federation, and the United States to conduct the study according to two phases: phase 1, to address the need for HVPNS from the viewpoint of fusion nuclear technology (FNT) development, to reach a consensus on the mission for HVPNS, and to determine the general testing capabilities, design features, and operating parameters required for a HVPNS to accomplish its mission, and phase 2, to identify candidate plasma-based device concepts that could potentially meet the HVPNS requirements and assess the technical and economic feasibility of the leading concepts. At the invitation of IEA, Professor Mohamed Abdou from the University of California, Los Angeles, led the phase 1 effort. This paper summarizes the technical results of the HVPNS phase 1 activities.

Phase 1 was conducted as a user community assessment by specialists in FNT development with participants from the European Union, Japan, the Russian Federation, and the United States. In assessing the mission and requirements of an HVPNS, Phase 1 efforts identified and addressed the following tasks:

1. a definition of the demonstration reactor (DEMO) and long-range fusion development strategies leading to DEMO
2. the FNT database needs for DEMO
3. the testing capabilities of the International Thermonuclear Experimental Reactor (ITER) and nonfusion facilities for meeting the FNT database needs of DEMO
4. the potential for arriving at a DEMO with too high of a technological risk (e.g., too low of a probability of achieving target availability levels) based on the nuclear technology databases from testing in only ITER and nonfusion facilities
5. the DEMO technological risk reduction benefits derived from HVPNS nuclear component and material combination testing that complements nuclear testing in ITER and nonfusion facilities
6. optimal strategies for the phasing of ITER, HVPNS, and nonfusion facility operation and testing programs.

This paper is organized as follows. Section II identifies and compares the DEMO goals in the European Union, Japan, the United States, and the Russian Federation. Section III briefly reviews the technical issues for FNT with the major focus being the blanket (B)/first-wall (FW) system, which determines the critical path in FNT development. Section IV evaluates the role and limitations of nonfusion facilities. Extensive testing in fusion facilities is found to be necessary; Sec. V quantifies the FNT requirements for such fusion testing. Section VI discusses the possible mission, objectives, and design guidelines for HVPNS. Notice that HVPNS is often abbreviated as VNS. A number of scenarios for fusion facilities prior to DEMO with ITER and various possibilities for VNS are quantitatively compared using a cost/benefit/risk analysis approach in Sec. VII. Section VIII presents the conclusions of the study.

II. DEMO GOALS

II.A. Introduction

The term DEMO refers to a thermonuclear fusion demonstration power plant based on the magnetic confinement of plasmas with a deuterium-tritium (D-T) fuel cycle. The results of DEMO and DEMO-relevant planning and study activities by the fusion programs of the European Union, Japan, the Russian Federation, and the United States have been used to develop guidelines for the definitions of DEMO with regard to its schedule, its mission and objectives, and its major parameters and features. These guidelines for defining DEMO were developed for the purposes of this paper and do not represent official positions on a DEMO definition by any of the fusion programs.

II.B. Fusion Program Perspectives

II.B.1. European Union Perspective

The most recently published document expressing general European Union views on DEMO are contained in a report for the Commission of the European Communities by the Fusion Programme Evaluation Board, which was chaired by Professor Colombo.[1]

The European Union foresees a stepwise strategy toward a prototype commercial power plant involving, after Joint European Torus (JET), a next-step experimental device (e.g., ITER) and then a DEMO. With regard to the timescale for the start of DEMO operation, Ref. 1 reports the date 2025 for DEMO startup. However, this date will be re-examined by the European Union to fit the likely timetable for ITER construction and operation.

The DEMO should be capable of producing significant amounts of electricity while taking due account of environmental constraints. While DEMO would include all the key technical elements of a power-generating reactor, Ref. 1 indicates that some technological tasks would still have to be performed in DEMO. This strategy would likely impact the overall availability of the device in the initial phase of operation. For example, the neutron fluence goal for the first DEMO blanket might be $\sim 5\ \mathrm{MW \cdot yr/m^2}$ while the long-term goal would be $>10\ \mathrm{MW \cdot yr/m^2}$.

Currently, there is no planned European Union effort to further define the mission and objectives or the major parameters and features of DEMO. Table I summarizes historical positions regarding the major parameters and features for DEMO.

II.B.2. Japan Perspective

The "Third Stage Fusion R&D Plan"[2] indicates that fusion will be developed with the aim of contributing to the energy supply in the latter half of the next

TABLE I

Fusion Program Guidelines for DEMO Major Parameters and Features

Parameter of Feature	European Union	Japan	Russian Federation	United States
Plasma mode of operation	Aim for steady state; determine whether long-burn pulsed operation can be tolerated	Steady state	Both pulsed and steady state are being considered	Steady state
Tritium fuel cycle: global	Self-sufficient	Self-sufficient	Self-sufficient	Self-sufficient
Tritium breeding ratio (TBR)	TBR > 1.0	TBR > 1.0	TBR: 1.05 to 1.1	TBR > 1.0+ addition for doubling time
Power output	Significant amounts of electricity	3 GW fusion power	<1.5 GW(electric)	Hundreds of megawatts(electric)
Neutron wall loading (MW/m^2)	2 to 3	Up to 5.0	2 to 3	2 to 3 average 3 to 4 peak
Availability	Depends on DEMO mission; could be >50% for reactor island	70%	>60%	50% net plant goal[a]
Thermal efficiency	Unspecified	30 to 40% net	>40%	>30% net
Blanket lifetime goal ($MW \cdot yr/m^2$)	Depends on specific DEMO goals; could be 5 for first blanket and >10 long term	Up to 7	15 to 20	10 to 20
Environmental consideration	Due account of environmental constraints	Low-activation materials	Low-activation materials; recycling and refabrication of DEMO materials	Low-activation materials; recycling and refabrication of DEMO materials

[a]An initial stage of lower availability is acceptable provided the goal availability is reached and sustained for several years.

century. According to this plan, if the research and development (R&D) in the experimental reactor stage (e.g., ITER) progresses well, the operation of the DEMO can be expected in the 2020 to 2030 period, and commercialization of fusion power can be expected by the middle of the next century. A recent draft pamphlet[3] showing the annual progress of fusion research at the Japan Atomic Energy Research Institute (JAERI) reports the operation of DEMO to be around 2030.

Reference 2 indicates that the mission of the DEMO phase of fusion R&D is to demonstrate in a plant scale the technological feasibility of realizing a high-energy multiplication steady-state plasma, of extracting energy generated from the plasma, and of converting the energy into electricity. A DEMO would demonstrate all the technologies necessary for a commercial fusion power reactor but would not necessarily be economically competitive. The DEMO technologies would be sufficient to achieve tritium breeding and power generation, reliable operation and maintainability, and benign environmental and safety aspects.

In the PROTO reactor phase of fusion R&D (i.e., a prototype commercial fusion power plant), the reactor load and utilization factors would be enhanced, and by the efficient utilization of the reactor power, the overall plant energy efficiency should be improved with the aim of demonstrating that a fusion reactor has a sufficient economic capability as an energy generation plant.

Table I provides examples of DEMO major parameters and features based on the foregoing discussion.

II.B.3. Russian Federation Perspective

The Russian Federation strategy for developing fusion on the path toward the practical use of fusion energy is based on three sequential basic steps: an experimental reactor (e.g., ITER), DEMO, and a commercial power reactor. The engineering foundation of DEMO must be based on ITER and its testing program. The proposed start of DEMO operation is 2025.

A conceptual study of a DEMO was begun in 1992 with the goals of choosing key parameters for DEMO based on the database from ITER, of performing a conceptual design of main DEMO systems, of specifying requirements for the ITER testing program, and of providing the technical basis to design blanket test modules for ITER. Both pulsed and steady-state modes of operation are being considered for DEMO.

DEMO is envisioned as an electricity-producing fusion reactor that demonstrates reliable and safe operation of all systems, provides a basis for estimating the economics of a commercial reactor, confirms the plasma physics basis of a commercial reactor, and demonstrates the ecological advantages of fusion. Table I summarizes basic technical requirements for DEMO.

II.B.4. United States Perspective

The U.S. Department of Energy Fusion Policy Advisory Committee recommended that the U.S. fusion program become energy oriented, with the goals of an operating DEMO by 2025 and an operating commercial power plant by 2040 (Ref. 4). It was acknowledged that achieving the 2025 goal for DEMO operation would require substantial increases in U.S. fusion program budgets from 1990 levels. Since budgets have not increased to the required levels, DEMO operation will likely be delayed beyond 2025. The DEMO schedule is being evaluated to bring it in line with anticipated future budget levels.

In the current U.S. strategy for magnetic fusion energy development, DEMO follows immediately after ITER and is based primarily on the physics and technology databases derived from the operation of and testing programs for ITER, a material test facility, and the Tokamak Physics Experiment (TPX). The role of DEMO is envisioned as one of producing net electricity and of providing the technical basis to proceed with a prototype commercial power plant.

A study of DEMO was initiated in 1992 under the STARLITE program, which is scheduled to produce a preconceptual design for a DEMO by the end of 1995. The current phase of STARLITE is a concept formulation activity that is addressing DEMO mission, goals, requirements, and features based largely on the viewpoints of U.S. utilities, industry, and regulatory agencies. The first workshop on the subject of DEMO reached the following conclusions about DEMO characteristics: DEMO shows for the first time all systems working as a full-scale integrated unit, it addresses issues of dependability and reliability and is large enough that the step to a prototype commercial plant leaves no open questions about scalability, it establishes the licensing procedures and rules for a fusion power plant, it demonstrates public acceptability and cost viability, it demonstrates feasibility and acceptable costs for decontamination and decommissioning, it demonstrates that the industrial infrastructure exists to serve the needs of the end-users, and it uses the same technology as is planned for the first commercial power plant.

A consensus on the mission, goals, milestones, and general features of DEMO will be developed by early 1995. By the end of 1995, a more detailed set of DEMO features and subsystems will be developed, and an initial assessment will be made of DEMO availability based on reliability analysis of DEMO subsystems.

At the present time, it is speculative to judge the outcome of this DEMO study activity. However, based on the results of previous studies of DEMO, Table I provides judgments on guidelines until STARLITE has completed its work on these matters.

III. TECHNICAL ISSUES AND TYPES OF TESTING

Fusion nuclear technology is the technology necessary to simultaneously

1. convert the fusion energy into heat and to efficiently extract this heat and convert it to a useful product
2. produce, extract, and recycle tritium to close the fuel cycle
3. provide the vacuum boundary for the plasma-containing chamber
4. provide radiation protection to components, personnel, and public.

III.A. Technical (Testing) Issues

Table II lists the FNT components as well as other components affected by the nuclear environment in fusion systems. Among FNT components, blankets determine the critical path to DEMO. The primary blanket options presently being considered worldwide as candidates for DEMO are summarized in Table III. These can be classified into (a) solid breeders, (b) self-cooled liquid-metal breeders, and (c) separately cooled liquid-metal breeders. Both helium and pressurized water are considered as coolants for solid breeders. Two types of liquid metals are being considered: lithium and lithium-lead. In self-cooled concepts, the same liquid metal serves as the breeder and coolant. For separately

TABLE II

FNT Components and Other Components Affected by the Nuclear Environment

1. B/FW[a] components
2. Plasma interactive and high heat flux components
a. Divertor, limiter
b. rf antennas, launchers, and wave guides
3. Shield components
4. Tritium processing systems
5. Instrumentation and control systems
6. Remote maintenance components
7. Heat transport and power conversion systems

[a]The blanket determines the critical path to FNT development.

TABLE III

Worldwide Blanket Options for DEMO*

Breeder	Coolant	Structural Material
Solid breeders Li_2O, Li_4SiO_4, Li_2ZrO_3, Li_2TiO_3	Helium or H_2O	Ferritic steel, vanadium alloy, SiC composites
Self-cooled liquid-metal breeders Lithium, LiPb	Lithium, LiPb	Ferritic steel, vanadium alloy with electric insulator, SiC composites with LiPb only
Separately cooled liquid-metal breeders		
Lithium	Helium	Ferritic steel, vanadium alloy
LiPb	Helium or H_2O	Ferritic steel, vanadium alloy, SiC composites

*Almost all concepts use beryllium as the neutron multiplier.

cooled concepts, helium is considered as a coolant for both lithium and LiPb while pressurized water is considered as a coolant only with LiPb. Only three classes of structural materials are presently considered as candidates for DEMO and commercial reactors: martensitic steels, vanadium alloys, and SiC composites.

Fusion nuclear technology testing issues have been identified and characterized in previous studies (e.g., Refs. 5 through 13). These issues include feasibility issues and attractiveness issues. Feasibility issues are those whose negative resolution will have the following impact:

1. may close the design window
2. may result in unacceptable safety risk
3. may result in unacceptable reliability, availability, or lifetime.

Attractiveness issues are those whose negative resolution will have the following impact:

1. reduced system performance
2. reduced component lifetime
3. increased system cost
4. less desirable safety or environmental implications.

A summary of the testing issues for the B/FW system is shown in Table IV. Many issues are common to all types of blankets. Examples are tritium self-sufficiency, allowable operating temperatures, reliability and failure modes, effects, and rates. However, the specific details of all the issues are different. A very brief summary of key issues for different types of blankets is given as follows.

For solid breeder blankets, the major classes of issues include

1. tritium self-sufficiency
2. breeder/multiplier/structure interactive effects under nuclear heating and irradiation
3. tritium inventory, recovery, and control; development of tritium permeation barriers
4. thermal control
5. allowable operating temperature window for breeder
6. failure modes, effects, and rates
7. mass transfer
8. temperature limits for structural materials and coolants
9. mechanical loads caused by major plasma disruption
10. Response to off-normal conditions.

For self-cooled liquid metal blankets, including concepts with a separate first-wall coolant, the main feasibility issue is the electrical insulation between the flowing liquid metal and the load-carrying duct walls. The most attractive solution is insulating coatings on the duct surface. The coating of the structural materials is also required as a tritium permeation barrier in separately (particularly water-) cooled concepts. For both electrical insulation and tritium permeation barriers, the common issues for coatings are fabrication technology, stability, and long-term performance under irradiation in the presence of temperature and stress gradients.

The tritium control issue is different for lithium and Pb-17Li. Tritium extraction is a key issue for lithium while tritium permeation is a primary issue for Pb-17Li. Activation of Pb-17Li under neutron irradiation is a

TABLE IV

List of B/FW Testing Issues

- Structure
  - Changes in properties and behavior of materials
  - Deformation and/or breach of components
    - Effect of first-wall heat flux and cycling on fatigue or crack growth–related failure
    - Magnetic forces within the structure (including disruptions)
    - Premature failure at welds and discontinuities
    - Failures due to hot spots
    - Interaction of primary and secondary stresses and deformation
    - Effect of swelling, creep, and thermal gradients on stress concentrations (e.g., in grooved surfaces)
    - Failure due to shutdown residual stress
    - Interaction between surface effects and first-wall failures
    - Self-welding of similar and dissimilar metals
  - Tritium permeation through the structure
    - Effectiveness of tritium permeation barriers
    - Effect of radiation on tritium permeation
  - Structural activation product inventory and volatility
  - Hermiticity of SiC
- Coolant
  - MHD pressure drop and pressure stresses
  - MHD and geometric effects on flow distribution
  - MHD insulating coating fabrication, integrity, and in situ self-healing
  - Stability/kinetics of tritium oxidation in the coolant
  - Helium bubble formation leading to hot spots
  - Coolant/purge stream containment and leakage
  - Activation products in Pb-Li
  - Liquid-metal purification
- Breeder and purge
  - Tritium recovery and inventory in solid breeder materials
  - Liquid breeder tritium extraction
  - Temperature limits and variability in solid breeder materials
    - Temperature limits
    - Thermal conductivity changes under irradiation
    - Effect of cracking
    - Effect of LiOT mass transfer
  - Breeder behavior at high burnup/high dpa
- Coolant/structure interactions
  - Mechanical and materials interactions
    - Corrosion
    - Mechanical wear and fatigue from flow-induced vibrations
    - Failure of coolant wall due to stress corrosion cracking
    - Failure of coolant wall due to liquid-metal embrittlement
  - Thermal interactions
    - MHD effects on first-wall cooling and hot spots
    - Response to cooling system transients
    - Flow sensitivity to dimensional changes
  - Coolant/coatings/structure interactions
- Solid breeder/multiplier/structure interactions
  - Solid breeder mechanical and materials interactions
    - Clad corrosion from breeder burnup products
    - Strain accommodation by creep and plastic flow
    - Swelling driving force
    - Stress concentrations at cracks and discontinuities
    - Thermal expansion driving force
  - Neutron multiplier mechanical interactions
    - Beryllium swelling (swelling driving force in beryllium)
    - Strain accommodation by creep in beryllium
    - Mechanical integrity of unclad beryllium
  - Thermal interactions
    - Breeder-structure and multiplier-structure interface heat transfer (gap conductance)
- General blanket
  - D-T fuel self-sufficiency
    - Uncertainties in achievable breeding ratio
    - Uncertainties in required breeding ratio
  - Tritium permeation
    - Permeation from breeder to blanket coolant
    - Permeation from beryllium to coolant
    - Permeation characteristics at low pressure
  - Chemical reactions
  - Tritium inventory
  - Failure modes and frequencies
  - Nuclear heating rate predictions
  - Time constant for magnetic field penetration for plasma control
  - Blanket response to near blanket failures
  - Assembly and fabrication of blankets
  - Recycling of irradiated lithium and beryllium
  - Prediction and control of normal effluents associated with fluid radioactivity
  - Liquid-metal blanket insulator fabrication, effectiveness, and lifetime
  - Tritium trapping in beryllium

concern, especially in the case of a liquid-metal spill, because of the production of the alpha-emitter ^{210}Po. However, recent investigations[8] have shown that the ^{210}Po problem might have been previously overestimated. This may require an on-line bismuth-removal technique. Corrosion and mass transfer are issues for both Li and Pb-17Li. Temperature limits for the structural material and coolant are key issues. For the lithium/vanadium, the heat transport system outside the blanket must be constructed of a different structural material because vanadium is not economical to use outside the blanket. Interstitial impurity transfer in such bimetallic loops is a key concern. Large stored chemical reactivity of lithium is a serious issue if water cannot be excluded from the system.

Water/liquid-metal interaction is an issue for water-cooled Pb-17Li blankets. Transient electromagnetics is an issue for liquid-metal blankets particularly in the case of plasma disruption. The large electrical currents, which can be induced in a liquid metal, combined with magnetic field can lead to large forces and stresses in the blanket.

A summary of the critical issues of FNT, which stresses the key functional aspects of the fusion reactor that must be resolved through testing, is given in Table V.

III.B. General Testing Requirements

Fusion nuclear technology development up to the DEMO requires testing to resolve the many known issues as well as presently unknown ones. The term "test" is used here in a generic sense to mean a process of obtaining information through physical experiment and measurement, i.e., not through design analysis or computer simulation. The testing needs for FNT were also addressed in previous studies (e.g., Refs. 5 through 12). However, these studies focused more on testing in nonfusion facilities while here we are more concerned with testing in fusion facilities. Definitive testing for decisive resolution of the issues requires that all loading conditions of the fusion environment and interactions among all physical elements of the components be adequately simulated. The key fusion environmental conditions are indicated in Table VI. However, in a realistic R&D program, particularly for fusion where no appropriate facilities now exist, tests proceed from simple measurements to more complex prototypes to reduce cost.

The testing types are distinguished by the relevant components and by the level of integration of the test. For each component, there is a set of tests ranging from property measurements to component verification. The test categories adopted here are basic, single-effect, multiple-effect/multiple-interaction, partially integrated, integrated, and component tests. Table VII summarizes the description of these categories. Note that the level of integration provides a rough measure of test complexity and an approximate indication of the chronological order.

TABLE V

Summary of Critical R&D Issues for FNT

D-T fuel cycle self-sufficiency
Thermomechanical loadings and response of blanket components under normal and off-normal operation
Material compatibility
Identification and characterization of failure modes, effects, and rates
Effect of imperfections in electric (MHD) insulators in self-cooled liquid-metal blanket under thermal/mechanical/electrical/nuclear loading
Tritium inventory and recovery in the solid breeder under actual operating conditions
Tritium permeation and inventory in the structure
Radiation shielding: accuracy of prediction and quantification of radiation production requirements
Plasma-facing component thermomechanical response and lifetime
Lifetime of first-wall and blanket components
Remote maintenance with acceptable machine shutdown time

TABLE VI

Key Fusion Environmental Conditions for Testing Fusion Nuclear Components

Neutrons (fluence, spectrum, spatial and temporal gradient)
Radiation effects (at relevant temperatures, stresses, loading conditions)
Bulk heating
Tritium production
Activation
Heat sources (magnitude, gradient)
Bulk (from neutrons)
Surface
Particle flux
Magnetic field
Steady field
Time-varying field
Mechanical forces
Normal
Off-normal
Thermal/chemical/mechanical/electrical/magnetic interactions
Synergistic effects
Combined environmental loading conditions
Interactions among physical elements of components

TABLE VII

Test Categories for Blanket R&D

Basic test
Basic or intrinsic property data
Single material specimen
Examples: thermal conductivity and neutron absorption cross section
Single-effect test
Explore a single effect, a single phenomenon, or the interaction of a limited number of phenomena to develop understanding and models
Generally a single environmental condition and a clean geometry
Examples: (a) pellet-in-can test of the thermal stress/creep interaction between solid breeder and clad, (b) electromagnetic response of bonded materials to a transient magnetic field, and (c) TPR in a slab of heterogeneous materials exposed to a point neutron source
Multiple-effect/multiple-interaction test
Explores multiple environmental conditions and multiple interactions among physical elements to develop understanding and prediction capabilities
Includes identifying unknown interactions and directly measuring specific global parameters that cannot be calculated
Two or more environmental conditions and more realistic geometry
Example: testing of an internally cooled first-wall section under a steady surface heat load and a time-dependent magnetic field
Partially integrated test
Partial integration test information but without some important environmental conditions to permit large cost savings
All key physical elements of the component and not necessarily full scale
Example: liquid-metal blanket test facility without neutrons if insulators are not required. (For concepts requiring insulators, tests without neutrons are limited to multiple effect.)
Integrated test
Concept verification and identification of unknowns
All key environmental conditions and physical elements, although often not full scale
Example: blanket module test in a fusion test device
Component test
Design verification and reliability data
Full-size component under prototypical operating conditions
Examples: (a) an isolated blanket module with its own cooling system in a fusion test reactor and (b) a complete integrated blanket in an experimental power reactor

Figure 1 illustrates a loose chronological order of tests for a major nuclear component such as the blanket, although some overlap will occur. For example, some multiple-effect tests can continue parallel to integrated tests. A very important conclusion from the results given later in this paper that must be stressed here is that integrated and component tests can be performed only in fusion devices. However, tests in the fusion environment do not have to be out of the fully integrated type. For example, a test article simulating a portion of the blanket to examine a particular group of multiple effects can be designed for testing in the fusion environment.

IV. ROLE AND LIMITATIONS OF NONFUSION FACILITIES

Nonfusion facilities can and should play a role in FNT R&D because of availability and low cost. Information from testing in nonfusion facilities can help reduce the risks and costs of the more complex, integrated tests in the fusion environment. However, a major point to be stressed here is that tests in nonfusion facilities have very serious limitations. Blanket concepts cannot be verified in nonfusion facilities, not to mention component engineering development and reliability growth. Nonfusion facility tests cannot replace the need for a comprehensive testing program in fusion facilities. Nonfusion facilities can be classified into (a) nonneutron test stands, (b) fission reactors, and (c) point neutron sources. Each of these is discussed briefly in Secs. IV.A, IV.B, and IV.C.

IV.A. Nonneutron Test Stands

The role of nonneutron test stands is in the area of basic property data, single-effect experiments, and

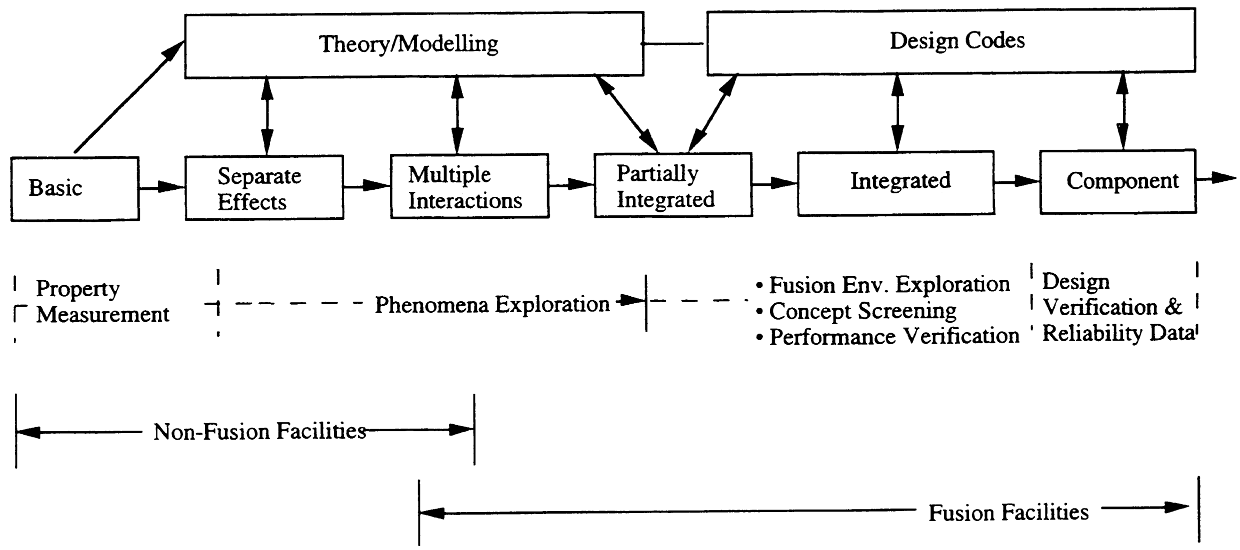

Fig. 1. Types and roles of experiments and facilities for FNT.

some of the multiple-effect/multiple-interaction tests for which the neutron field is not important. Since neutrons are the only practical source of nuclear heating in a large volume and they are also necessary to simulate radiation effects, the value of nonneutron test stands is limited. Studies in the early 1980s assumed that magnetohydrodynamic (MHD) tests without neutrons for liquid-metal concepts are possibly able to perform concept verification tests. Such an assumption is no longer valid. It is clear now that the toroidal magnetic field in tokamaks will most likely be high (>12 T at the coils). Therefore, electrical insulators must be used inside the blanket to reduce the MHD drop to an acceptable level. Concepts for self-healing coatings (e.g., aluminum oxide with LiPb) have been proposed. Fundamental feasibility issues that relate to the imperfections in such coatings are (a) the speed at which they occur, (b) the speed at which they heal, and (c) their effect on MHD pressure drop. These problems are strongly dependent on nuclear heating effects (e.g., temperature and stress magnitude and gradient) as well as radiation damage effects. Furthermore, experience from other technologies indicates that coatings constitute some of the most challenging problems. One of the reasons is the mismatch of the thermal expansion coefficients between the substrate and the coating. This problem can become even more pronounced when different types of materials are chosen, such as a ceramic coating on a metallic substrate, because of the difference of response of the material type to loads: Metals are mostly ductile while ceramics are brittle. Consequently, when a ceramic coating and a substrate are put under the same load, the substrate might deform plastically while the coating may crack in a brittle matter. This scenario most commonly results in the initiation of cracks on the ductile substrate. Given that the electric insulating coating is a ceramic-type material, the difference of response of the substrate (metal) and in the coating in a neutron environment further complicates the issue. Furthermore, complications based on geometric features such as bends, corners, and joints will introduce additional variations in stress distribution and stress concentrations. Therefore, the R&D on insulators should proceed in two steps. First, screening of candidate coatings is performed in laboratory experiments, and possibly some small-scale experiments in fission reactors, to identify promising concepts that perform well in a single-effect environment. Second, experiments with a prototypical test section in an environment that combines neutrons and a magnetic field are necessary to finally confirm the feasibility of self-cooled liquid-metal concepts. Such a combination with the large test volume required is practically available only in a fusion test facility, as will be clear from Secs. V and V.B.

The foregoing examples do not argue against tests in a nonneutron environment. They only emphasize the fact that feasibility of blanket concepts cannot be established prior to testing in the fusion environment. Experiments in nonneutron test stands are relatively low in cost, and they are important and useful in reducing the large costs and risks associated with future tests in the fusion environment.

IV.B. Fission Reactors

Fission reactors provide neutrons in a limited volume and are thus suited to some FNT experiments. Table VIII summarizes the capabilities of fission reactors available in the United States, Canada, Russia, and Europe for blanket tests. Testing in fission reactors suffers from serious limitations, which are listed in Table IX.

TABLE VIII

Capabilities of Available Fission Reactors for Blanket Tests

Reactor	Location	Reactor Power (MW)	Fast Flux ($n/cm^2 \cdot s$)	Thermal Flux ($n/cm^2 \cdot s$)	Dimension of Irradiation Channel (cm)	Effective Core Height (cm)
EBR-II	United States	62	2.0×10^{15}		7.4 (circular)	36
HFIR	United States	100	1.5×10^{15}	2.3×10^{15}	3.7 (circular)	51
ATR	United States	250	1.9×10^{14}	8.8×10^{14}	6.05 (seven flux traps)	122
RBT-10	Russia	10	4.4×10^{13}	2.3×10^{13}	15.8 × 23.7	35
SM-3	Russia	100	2.2×10^{14}	8.8×10^{13}	6 and 16 (circular)	35
IVV-2M	Russia	20	9.3×10^{13}	5.5×10^{13}	14.7 × 25.5	50
Phénix	France	250	1.3×10^{15}		12.67 (hexagonal)	85
OSIRIS	France	70	5.0×10^{14}	1×10^{14}	8.4 (circular)	60
SILOE	France	35	5.0×10^{14}	4.0×10^{14}	8.0 (circular)	60
BR-2	Belgium	60	6.0×10^{14}	1.0×10^{15}	20 (circular)	96
HFR	The Netherlands	20	5.0×10^{14}		14.5 (circular)	60
JRR-2	Japan	10	1.0×10^{14}	1.0×10^{14}		
NRU	Canada	125	4×10^{13}	2.4×10^{14}	10 (circular)	300

Most serious is the small test volume. For example, there is no fission reactor now operating anywhere in the world that can provide a test location with a ≥15-cm equivalent circular diameter at a fast neutron flux equivalent to 1 MW/m^2 wall loading ($\geq 1 \times 10^{15}$ $n/cm^2 \cdot s$). This limitation, together with some safety aspects of fission reactors, also makes the simulation of nonnuclear effects such as magnetic field and mechanical forces very difficult or impossible. Another set of problems arises from the difference between the fission and fusion reactor neutron and secondary gamma-ray spectra. These differences lead to difficulties in simulating the magnitude, profile, and time-dependent behavior of reaction rates such as helium and tritium production, as well as power density and atomic displacements.

TABLE IX

Key Limitations of Fission Reactors

Small test volume
Small size per location
Small number of existing locations
Lack of fusion-related (nonneutron) conditions
Magnetic field
Surface heat
Particle flux
Mechanical forces
Accessibility
Lack of fusion-related radiation damage parameters
Neutron spectra
Helium-to-dpa ratio
Types and rates
Lack of fusion-related power density
Magnitude
Spatial profile
Lack of fusion-related lithium burnup rate
Magnitude
Spatial profile
Reactivity considerations limits on size and type of experiments
Availability of fission test reactors for testing (rapid downward trend)

Despite these limitations, fission reactor testing is extremely useful for near-term FNT experiments. It is suited for some multiple-effect tests that depend on nuclear effects and are less sensitive to nonnuclear effects. Examples are tests of a unit cell of a solid breeder blanket to investigate tritium release behavior and some aspects of breeder/structure interactions.

IV.C. Accelerator-Based Neutron Sources

Accelerator-based neutron sources produce neutrons in such a small volume that they are normally called point neutron sources. Deuterium-tritium point sources produce 14-MeV neutrons, hence the correct fusion spectra, but their yield in existing facilities is limited technologically to $\sim 10^{13}$ n/s. Such a yield results in a very low neutron flux. Even at a small distance as close as 5 cm to the target, the neutron flux is more than five orders of magnitude lower than that in a fusion reactor with a 1 MW/m^2 wall load. Furthermore, the life of the target is limited to a <100-h irradiation. Therefore, the usefulness of D-T point neutron sources is limited to neutronics experiments, e.g., measurements of tritium production rates (TPRs).

The flux is too low to produce nuclear heating or reactions at a rate that would permit other engineering experiments, e.g., thermomechanics testing, or measurements of significant radiation effects. An example of a state-of-the-art D-T point neutron source is the Fusion Neutronics Source (FNS) facility in Japan.[14] The capabilities of FNS are compared in Table X with those from recent D-T shots in the Tokamak Fusion Test Reactor[15] (TFTR). The Joint European Torus[16] (JET) provides performance comparable to TFTR. It is interesting to note that even present plasma physics devices could provide several orders of magnitude higher neutron flux than D-T point neutron sources. The key problem with the present tokamaks is obviously the plasma pulse length as well as the number of plasma cycles per day.

Other proposals for accelerator-based neutron sources have been made. The most prominent is a proposal for a deuterium-lithium (D-Li) source in which neutrons are produced by bombarding a flowing lithium target with energetic (~30- to 40-MeV) deuterons. The deuterons interact with the lithium jet atoms either losing part of their energy through Coulomb interactions or producing nuclear reactions some of which produce neutrons, $^{7}\mathrm{Li}(d,np)^{7*}\mathrm{Li} \rightarrow \mathrm{T} + \alpha$, $^{7}\mathrm{Li}(d,2n)^{7*}\mathrm{Be} \rightarrow {}^{3}\mathrm{He} + \alpha$, $^{7}\mathrm{Li}(d,n)^{8}\mathrm{Be}$, $^{7}\mathrm{Li}(d,3n)^{6}\mathrm{Be}$, and other reactions.

The design of a D-Li source Fusion Materials Irradiation Test was started[17,18] in the late 1970s in the United States and was later terminated during construction because of a combination of funding problems and technological issues. Recently, an international activity under the auspices of IEA was started[19–21] to examine the need and issues for a D-Li source called the International Fusion Materials Irradiation Facility (IFMIF). Examples of analyses of neutronics characteristics of IFMIF-type facilities are given in Refs. 21, 22, and 23.

One advantage of such a source is the existing experience with accelerators. Another potential advantage is the possibility of performing accelerated testing of radiation damage effects in material specimens if a high neutron flux can be produced at a reasonable cost. However, there are a number of technical issues that affect the usefulness of a D-Li source for FNT and material development. These include (a) the neutron spectrum, (b) the steep flux gradient, and (c) the surface area and volume available for testing.

The D-Li neutron source produces neutrons with energies from electron volts up to ~50 MeV. This is compared with the fusion D-T reaction where neutrons are produced within a narrow energy range at ~14 MeV. The neutron spectrum from the D-Li reaction varies with the incident deuteron energy. As shown in Table XI (see Ref. 22), the fraction of the neutrons above 15 MeV increases from 8 to 15.7% when the incident deuteron energy is increased from 30 to 40 MeV. The average neutron energy is ~6 MeV for a 35-MeV deuterium beam. The low-energy component of the D-Li source may be able to simulate qualitatively the neutron spectrum created by backscattering into a fusion reactor first wall. However, the high-energy component (>15 MeV) in the D-Li neutron spectrum is of concern. There, high-energy neutrons can induce reactions with high-energy thresholds that are not accessible to the lower energy neutrons of the D-T fusion reactor spectra. Furthermore, the accuracy of nuclear data above 14 MeV is generally poor. So, the concern here is whether radiation effects observed with D-Li neutron spectra can be accurately correlated to those in a fusion reactor.

An accelerator-based neutron source produces a neutron yield that is highly anisotropic. Furthermore, the neutron spectra are dependent on the angle (relative to the beam direction). This leads to gradients in the neutron flux in all directions at the test sample. Of particular concern are the directional gradients in the plane perpendicular to the direction of the deuteron beam. Figure 2 from Gomes research[23] shows the displacements-per-atom (dpa) rate in a direction perpendicular to the beam. Gradients in the direction along the beam are much steeper. At the first wall of the tokamak, the gradients in the toroidal direction are very small, and in the poloidal direction, they are typically <0.1%/cm. The flux gradient at the test samples with the D-Li source can be reduced for a given test area by increasing the beam focus area. However, this reduces the magnitude of local neutron flux.

The most serious issue that severely limits the usefulness of a D-Li source is the available space for testing and the type of tests that can be performed. This

TABLE X

Comparison of Present D-T Point Neutron Source (FNS) to Present Plasma-Based Device (TFTR)

	TFTR	FNS
Neutron yield	2×10^{18} n/shot	5×10^{12} n/s
Pulse length	~1 s	Variable
Irradiation frequency	~10 cycle/day	~10 h/day
Neutron flux (n/cm^2·s)	At the first wall: 2×10^{12}	At 5 cm from target: 6.4×10^{9} At 1 m from target: 1.6×10^{7}

TABLE XI

Neutron Generation Rate and Average Neutron Energy from D-Li Source

	Incident Deuteron Energy		
	30 MeV	35 MeV	40 MeV
Total neutron generation rate for a 250-mA deuteron beam (n/s)	6.46×10^{16}	8.36×10^{16}	1.035×10^{17}
Average neutron energy (MeV)	5.36	6.06	6.71
Percentage of neutrons born in each energy range			
0 to 15 MeV	91.9	88.1	84.3
15 to 50 MeV	8.1	11.9	15.7

problem has not received in the literature the comprehensive analysis required to judge the merits of a D-Li source. Key points related to this testing space issue are briefly treated below.

Optimization studies for the D-Li source suggest a 250-mA beam with 35-MeV deuterons. Figure 2 (from Ref. 23) shows the dpa rate per full-power year (FPY) in the direction perpendicular to the beam. Table XII shows the test area, and Table XIII shows the test volume obtainable with a 35-MeV, 250-mA D-Li source. Table XII shows the maximum surface area available for testing with rates of radiation damage indicators, e.g., dpa equivalent to that attainable with a given neutron wall load at the first wall of a tokamak reactor. The results show that the maximum surface area available for testing is 200 cm^2 (obtainable with a beam spot area 20×20 cm^2) at an equivalent neutron wall load of 1 MW/m^2. The maximum test area with an equivalent neutron wall load of 3 MW/m^2 is only 50 cm^2 (obtainable with a beam spot area of 10×10 cm^2).

Clearly, such a test area is not suitable for module or even submodule testing. Therefore, D-Li sources

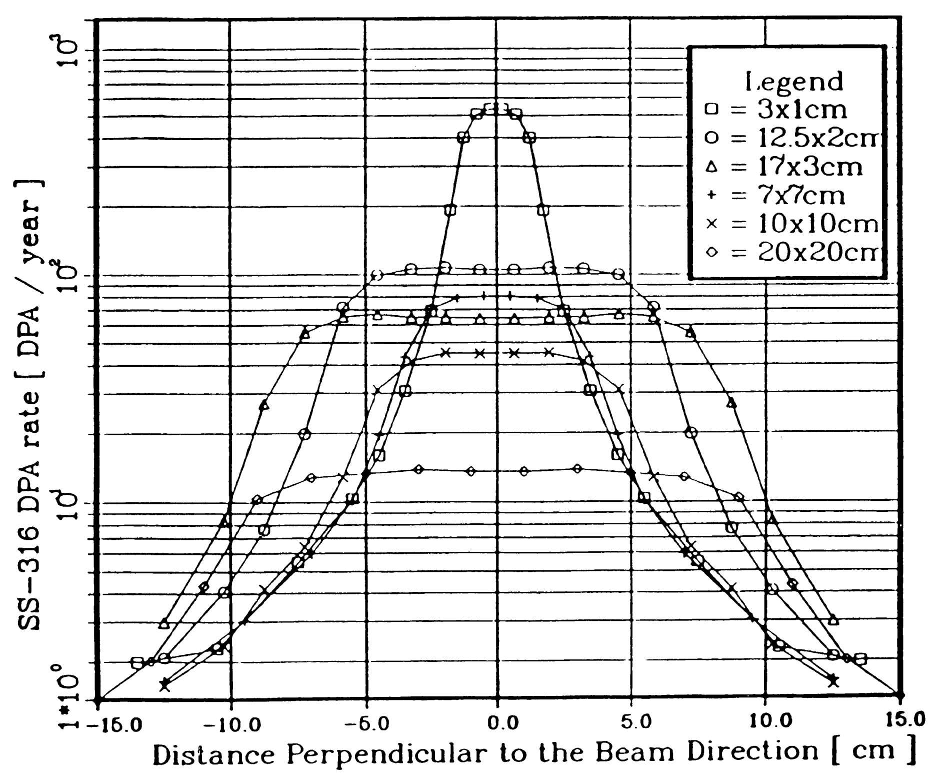

Fig. 2. Gradient of the Type 316 stainless steel dpa rate perpendicular to the beam. Beam current = 250 mA, and deuteron energy = 35 MeV.

TABLE XII

Surface Area Available for Testing with D-Li Neutron Source with 35-MeV, 250-mA Deuteron Beam to Simulate First-Wall Conditions of a Fusion Reactor

(Equivalent) Neutron Wall Load (MW/m^2)	Maximum Surface Area Available for Testing[a] (cm^2)	Comments
1	200	Possible with beam spot area 20×20 cm^2
3	50	Possible with beam spot area 10×10 cm^2

[a]Area perpendicular to beam direction.

cannot play a major role in the engineering development of FNT components and compatible material combinations. The question to be addressed here is whether such a source can alone fulfill all the material irradiation science needs. The relevance of this question to this study is to assess whether plasma-based fusion test facilities need to also address part of material specimen tests in addition to component tests.

Irradiation testing of material specimens is a useful tool to supplement component tests. However, irradiation of small specimens alone without parallel component tests is not meaningful for component development because specimens will not simulate the critical environmental conditions such as material interfaces (e.g., coolant-breeder-structure), temperature and stress gradients, and joints. If component tests are carried out, then parallel tests on specimens are useful when a large number of specimens are irradiated to investigate the response of a number of candidate materials under a variety of conditions. Table XIV summarizes the space requirements for material specimen tests. Information in Table XIV was first developed in INTOR (Ref. 24) and subsequently improved in FINESSE (Refs. 5 and 6) and ITER-conceptual design activity studies.[25,26] Table XIV is limited to structural materials and assumes that four candidate metallic alloys are to be investigated. Some observations are in order here. First, nonstructural materials such as breeder and multiplier materials are not suited for specimen tests because their issues (e.g., tritium transportation in solid breeders) require large volumes. Second, silicon carbide (SiC) composites represent a leading candidate for structural materials in DEMO. It is only one of two materials (the other being vanadium alloy) that can meet the low-activation and low-decay-heat requirements for attractive safety and environmental impact. The test volume required for ceramic matrix composites is much larger than that for metallic alloys because the fiber matrix behavior is not uniform; e.g., it is much different at a bend section from that in a straight section. Therefore, requirements for testing SiC composites are excluded from Table XIV. Third, for specimen tests to be useful, they have to be irradiated in a controlled environment, e.g., well-defined temperature. Controlling the temperature of the specimen requires cooling. Therefore, the irradiation volume required for the test matrix is much larger than that obtained by summing up only the volume of the specimens. Practical requirements of cooling, support, and instrumentation will considerably increase the test volume requirements.

TABLE XIII

Test Volume Available with dpa Rate per Year Greater Than a Specified Threshold for D-Li Neutron Source with 35-MeV, 250-mA Deuteron Beam

	Beam Cross-Sectional Area ($cm \times cm$)	
dpa/yr[a]	10×10 (cm^3)	20×20 (cm^3)
30	10	0
20	100	0
10	300	7

[a]Assuming a plant factor of 70% and stainless steel as typical material.

Based on the foregoing points, one concludes that the test volume defined in Table XIV is a minimum for the material science irradiation specimen matrix. Table XIV shows that more than 30 000 specimens are needed with a volume >2000 cm^3. This volume does not include the additional space needed for cooling, support, instrumentation, and other functions.

Table XIII shows the test volume available with dpa rate greater than a specified threshold for a D-Li source. With a 20- × 20-cm^2 beam focus, only 7 cm^3 is available with a dpa rate of 10/yr. With a 10- × 10-cm^2 beam focus area, higher dpa rates are possible but still at a very small test volume. The maximum test volumes at 30, 20, and 10 dpa/yr (with a 70% plant factor) are 10, 100, and 300 cm^3, respectively. These volumes are to be compared with the requirements of >2000 cm^3 in Table XIV for four candidate structural material science specimen irradiations. The dpa rate for typical candidate structural materials in a tokamak first wall

TABLE XIV

Space Requirements for Material Property Specimen Tests for Material Science Information on Radiation Effects on Four Candidate Structural Materials

Test Description	Specimen Configuration	Size (mm)	Material Variables	Irradiation Environment				Multiplicity	Total Specimens	Volume/Specimen (cm^3)	Total Volume (cm^3)
				Temperature	Fluence	Flux	Stress				
Charpy-V	$\frac{1}{3}$ Charpy V-notch	(3.3 × 3.3 × 23.6)	4	7	4	1	0	8	896	0.26	235
Tensile	Flat	(0.76 × 25.4 × 5.0)	4	7	4	1	0	15	1 680	0.10	162
Creep	Tube	(4.57 diameter × 23.0)	4	7	1	1	6	1	168	0.38	63
Swelling	Disk	(3.18 diameter × 0.25)	144	7	4	1	0	6	24 192	0.002	48
Fracture toughness	Compact tension	(16 diameter × 2.5)	4	7	4	1	0	16	1 792	0.5	901
Stress corrosion cracking	CERT SS-3	(0.76 × 25.4 × 5.0)	4	7	4	1	0	24	2 688	0.10	259
Fatigue	Constant amplitude/high cycle	(6.35 diameter × 38)	4	7	4	1	0	3	336	1.2	404
Total for all specimens (does not include volume for coolant and support)								---	31 752	---	2072

is 12 per MW·yr/m^2. The lifetime goals for the first wall are ~10 and 20 MW·yr/m^2, i.e., ~120 and 240 dpa, respectively, for the DEMO and commercial reactors. Therefore, the D-Li source with 10 dpa/yr (assuming a 70% plant availability factor) with a 300- to 400-cm^3 test volume will require ~12 and 24 yr to achieve end-of-life (EOL) irradiation for DEMO and commercial reactors, respectively.

Several important conclusions can be reached regarding the usefulness and limitations of a D-Li neutron source:

1. Present concepts for the source are clearly limited in both neutron flux/power density and test area/volume; representative maximum test area/volume are 200 cm^2/300 cm^3 at an equivalent neutron wall load of 1 MW/m^2. Such a wall load is comparable only to ITER and is a factor of ~3 lower than that for DEMO/power reactors.

2. It is clearly not suitable for testing submodules of components.

3. It is not suitable for testing important nonstructural materials, such as breeder and multipliers, as the key issues for such materials require testing in a volume (e.g., tritium release and transport in solid breeders).

4. It can be used for some structural material irradiation specimen testing; the major advantage relative to ITER is expected to be higher availability (~70% compared with <10% in ITER); however, the test volume is not sufficient to do all the required material science specimen irradiation tests. Since the flux in a D-Li source test region is not high, considerations of a test space–test time matrix need to be carefully analyzed. Some of the material science specimen tests will need to be performed in fusion testing facilities.

5. Results from specimen irradiation tests are generally meaningful only if performed parallel to component tests; therefore, an IFMIF-type facility will be useful only if submodule tests and module tests are carried out in parallel in fusion facilities.

IV.D. Summary of Role and Limitations of Nonfusion Facilities

Assessment of the overall contribution of nonfusion facilities to the development of FNT is important. Table XV summarizes the capabilities of nonfusion facilities for simulation of key conditions for fusion nuclear component experiments. The most important conditions are (a) neutron effects (radiation damage, and tritium and helium production), (b) bulk heating (nuclear heating in a significant volume), (c) nonnuclear conditions (e.g., magnetic field, surface heat flux, particle flux, and mechanical forces), (d) conditions for simulating thermal/mechanical/chemical/electrical interactions, and (e) conditions for integrated tests and synergistic effects. A very important conclusion is that

TABLE XV

Capabilities of Nonfusion Facilities for Simulation of Key Conditions for Fusion Nuclear Component Experiments

	Neutron Effects[a]	Bulk Heating[b]	Nonnuclear[c]	Thermal/Mechanical/ Chemical/Electrical[d]	Integrated Synergistic
Nonneutron test stands	No	No	Partial	No	No
Fission reactor	Partial	Partial	No	No	No
Accelerator-based neutron source	Partial	No	No	No	No

[a]Radiation damage, and tritium and helium production.
[b]Nuclear heating in a significant volume.
[c]Magnetic field, surface heat flux, particle flux, and mechanical forces.
[d]Thermal/mechanical/chemical/electrical interactions (normal and off-normal).

nonfusion facilities are not able to simulate partially integrated or integrated conditions. Their capabilities are limited mostly to single environmental conditions and some multiple-effect/multiple-interaction experiments.

From the FNT development viewpoint, the most important question is the contribution of facilities to resolving the critical issues, which were presented earlier in Table V. Table XVI shows the contribution of nonfusion facilities to resolving the FNT critical issues. The most striking result is that there is no critical issue that can be fully resolved by testing in nonfusion facilities alone. The second most striking conclusion is that there are critical issues for which no significant information can be obtained from testing in nonfusion facilities. An example is identification and characterization of failure modes, effects, and rates. Therefore, the feasibility of blanket concepts cannot be established prior to testing in fusion facilities. The word "partial" in Table XVI designates a contribution that is substantial when supplemented by fusion tests; otherwise, in the absence of fusion tests, no judgment can be rendered on the resolution of the critical issue.

TABLE XVI

Contribution of Nonfusion Facilities to Resolving Critical Issues for FNT Component Performance Demonstration*

Critical Issue	Nonneutron Test Stands	Fission Reactors	Accelerator-Based Neutron Sources	
			D-T	D-Li
D-T fuel cycle self-sufficiency	None	Small	Partial[a]	None
Thermomechanical loadings and response of blanket components under normal and off-normal operation	Small	Small	None	None
Materials compatibility	Some	Some	None	Small
Identification and characterizations of failure modes, effects, and rates	None	None	None	None
Effect of imperfections in electric (MHD) insulators in self-cooled liquid-metal blanket under thermal/mechanical/electrical/nuclear loading	Small	Small	None	Small
Tritium inventory and recovery in the solid breeder under actual operating conditions	None	Partial	None	None
Tritium permeation and inventory in the structure	Some	Partial	None	Small
Radiation shielding: accuracy of prediction and quantification of radiation protection requirements	None	Small	Partial	Small
Plasma-facing component thermomechanical response and lifetime	Some	Some	None	Some
Lifetime of first-wall and blanket components	None	Partial	None	Partial[a]
Remote maintenance with acceptable shutdown time	None	None	None	None

*Note that all these facilities make important contributions to important issues that require only single-effect or limited multiple-effect tests. This table focuses only on critical issues for component performance demonstration, which generally require extreme multiple-effect and integrated tests.
[a]Partial: substantial contribution when supplemented by fusion test; not sufficient in the absence of fusion tests.

One should emphasize again that the foregoing conclusions do not suggest that nonfusion facilities should not be used. They only suggest that their usefulness in resolving the critical issues is severely limited. Nonfusion facilities can and should be used to narrow material and design concept options and to reduce the costs and risks of the more costly and complex tests in the fusion environment. The cost of tests in nonfusion facilities tends to be much smaller than that expected in the fusion environment, with the only possible exception being tests in a D-Li source since none exists at present and both the capital and operating costs are substantial.

The key conclusion here is that FNT development does require fusion testing facilities.

V. FNT REQUIREMENTS FOR TESTING IN FUSION FACILITIES

Section IV.D shows that nonfusion facilities, albeit useful, are severely limited in simulating the key conditions for fusion nuclear component experiments and development (see Table XV). Based on results from Table XVI, one sees that clearly nonfusion facilities are unable to fully resolve any of the critical FNT issues. It is therefore very clear that testing in fusion facilities is an absolute necessity to develop fusion nuclear components. The key questions are (a) how should the tests in the fusion environment be structured to effectively develop and qualify FNT components for DEMO? and (b) what are the requirements of FNT tests on the major parameters and characteristics of suitable fusion test facilities?

V.A. Testing Stages and Framework

Figure 1, shown earlier, illustrates a loose chronological order of tests for major nuclear components such as the blanket. Tests in nonfusion facilities are limited to single-effect and some multiple-interaction tests. Fusion tests need to cover several multiple-interaction tests, integrated tests, and component tests.

In partial analogy to experience from technology development in other fields, we propose that testing and development of FNT (primarily the blanket) in fusion facilities proceed in three stages: (a) initial fusion "break-in" in the fusion environment, (b) concept performance verification, and (c) component engineering development and reliability growth as illustrated in Fig. 3. Notice that FNT components such as the blanket have never been tested before on any fusion facility. Therefore, the first stage should be focused on calibration and exploration of the fusion environment as well as testing and development of experimental techniques and diagnostic tools (for example, the questions of how to measure and collect data and interpret and extrapolate results and the effects of the fusion environment on instrumentation tools). Submodules, rather than modules, should be used to save cost in this stage. Part of the fusion environment exploration is screening a number of candidate design concepts. Only a limited number of concepts are tested in the second stage, which aims at performance verification. Modules should be used in this stage to ensure that all the key aspects of subsystem interactions are tested. Results of tests in stage II should permit selection of a very small number of concepts. This number should be 2 or 3. It is risky to select one concept before performing reliability growth tests in stage III. In the meantime, since stage III tests are complex, costly, and time consuming, the number of concepts should not exceed three. Stage III tests focus on true engineering development where actual prototypical components are tested to verify the final component design and to obtain data on reliability. As shown in Appendix A, tests, particularly reliability tests, may show excessive failures and/or unacceptable performance. Therefore, an aggressive design/test/fix iterative program is needed. More details on failure rates and reliability growth testing will be given in Appendix A. The extensive reliability testing required to achieve blanket availability goals is one of the primary reasons why blanket testing determines the critical path for FNT development.

V.B. Testing Requirements for Major Parameters of Fusion Facilities

Satisfactory testing of the blanket in the fusion environment imposes important requirements for the design of the fusion testing facility in at least two areas: (a) major parameters and (b) engineering design. The major parameters of concern are those that have a major impact on both the usefulness of the tests and the cost of the device. The requirements of the engineering design include providing capabilities for fast insertion and removal of test modules; access to the many coolant, tritium-processing, and instrumentation lines; and suitably located space and facilities for ancillary equipment to support the test program (e.g., heat rejection system, tritium processing facility, and purification and chemical control systems and instrumentation systems).

The FNT testing requirements for the major parameters for fusion facilities have been analyzed in several major studies.[6,7,25-32] International workshops have also helped to develop consensus on many of these requirements. However, recent interest in scenarios for fusion development facilities and the evolution of the ITER design during the engineering design activity (EDA) have made it necessary to review in more detail the FNT testing requirements. A summary of the results for the FNT requirements on major parameters for testing in fusion facilities is given in Table XVII. The requirements given in Table XVII are driven by the goal of providing the database necessary to construct the blanket for DEMO.

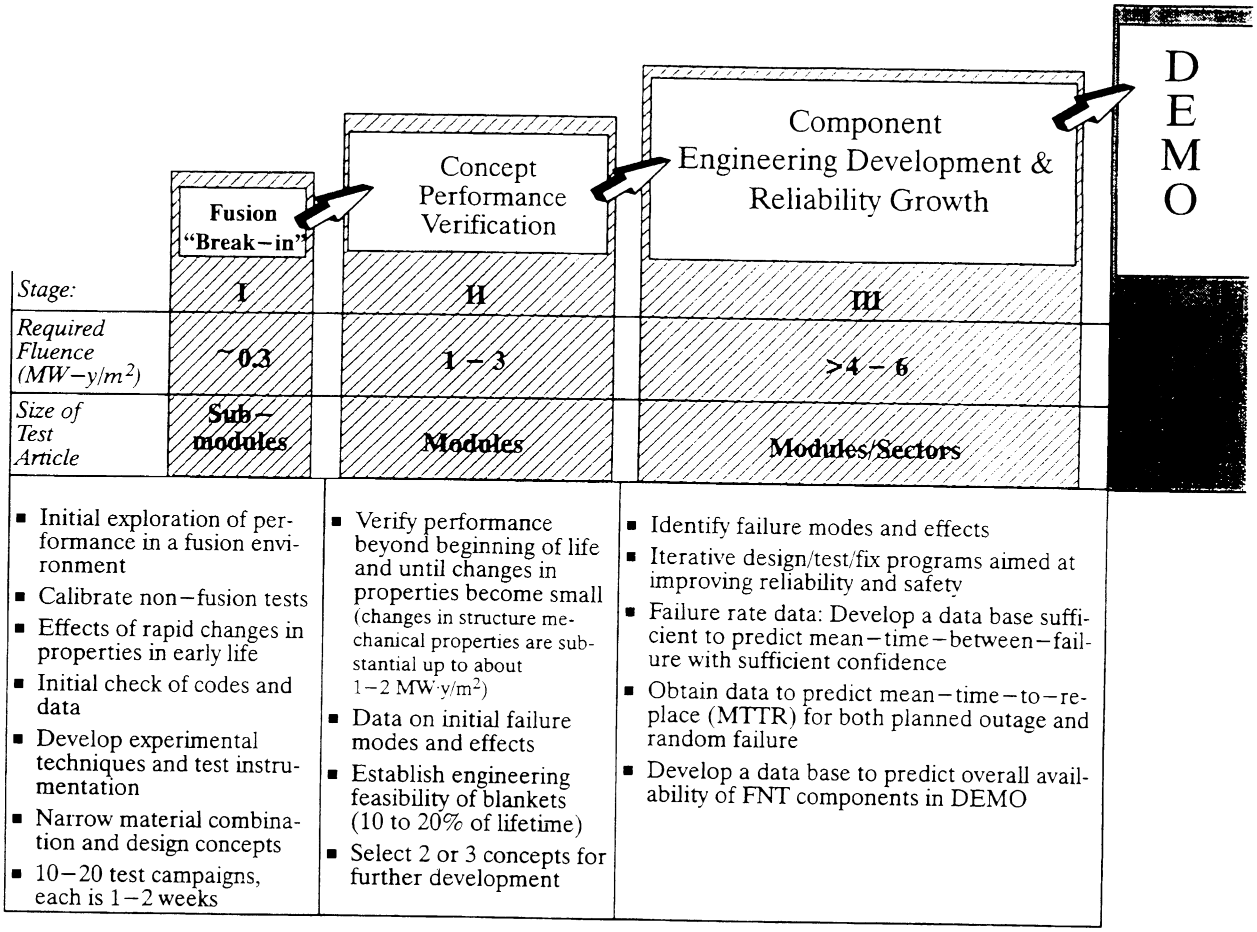

Fig. 3. Stages of FNT testing in fusion facilities.

There are other important requirements that are not given in Table XVII, such as the value of the magnetic field in the blanket test region (e.g., to test liquid-metal blankets or the effects of ferritic steel on magnetic performance), surface heat flux, and minimum test area per module. We limited Table XVII to those requirements that appear to be major discriminating factors in the selection among options for fusion testing facilities. Other parameters not given in Table XVII are either implied or can be deduced from those already given or do not appear to be crucial discriminating factors in the selection among options for fusion testing facilities. The technical basis for the values given in Table XVII are briefly summarized in Secs. V.B.1, V.B.2, and V.B.3.

V.B.1. Neutron Wall Load

The minimum acceptable neutron wall load is derived from two factors: (a) engineering scaling considerations and (b) trade-offs between device availability and wall load for a given testing fluence and testing time.

Volumetric heating in the blanket is directly proportional to the wall load. Most thermomechanical and tritium-related phenomena in the blanket strongly depend on the temperature and stress profiles, which in turn are directly dependent on the heating rates. Since the wall load in a fusion test facility is likely to be much lower than that in DEMO (~3 MW/m^2) and commercial plants (~4 to 5 MW/m^2), engineering scaling considerations[6,7,31] are crucial. Useful testing at a reduced wall load, relative to DEMO and reactor conditions, is possible by altering the design and operating parameters of the test modules. Test modules must "act like" rather than "look like" a DEMO module. Generally, the coolant bulk average temperatures are easy to maintain by varying the coolant speed and flow rate. Temperature distributions within components are much more difficult to maintain. Some control over temperature distributions can be obtained by changing the thickness of blanket elements within the blanket as well as the overall dimensions of the test module. However, very large changes in sizes lead to new effects and an

TABLE XVII

FNT Requirements for Major Parameters for Testing in Fusion Facilities with Emphasis on Testing Needs to Construct DEMO Blanket

Parameter	Value
Neutron wall load (MW/m^2)	1 to 2
Plasma mode of operation	Steady state[a]
Minimum COT (weeks)	1 to 2
Neutron fluence at test module ($MW \cdot yr/m^2$)	
Stage I: initial fusion break-in	0.3
Stage II: concept performance verification	1 to 3
Stage III: component engineering development and reliability growth	4 to 6
Total neutron fluence for test device ($MW \cdot yr/m^2$)	>6
Total test area (m^2)	>10
Magnetic field strength (T)	>2

[a]If steady state is unattainable, the alternative is long plasma burn with plasma duty cycle >80%.

overall geometry that is much less representative than a real DEMO/commercial blanket.

Engineering scaling techniques are found to be useful, particularly in simulating individual effects. However, two important conclusions are reached. First, engineering scaling techniques require that for any one given blanket design, several test modules must be designed, each focusing on a different group of phenomena, effects, and technical issues. Second, confidence in the extrapolation of test results to the DEMO and commercial reactors drops sharply when the wall load in the test facility is reduced by a factor of >2 to 3 relative to that in DEMO/commercial power plants. This is because it becomes very difficult to design "act-like" test modules using engineering scaling rules to maintain important performance parameters such as stresses and temperatures; hence, many phenomena cannot be preserved (see Ref. 5 for more details). Therefore, a neutron wall load of 1 to 2 MW/m^2 is necessary in the fusion test facility. A higher wall load in the test facility will increase the confidence level in extrapolating the test results to DEMO. The surface heat flux has a major influence on blanket thermomechanics, particularly for the first wall. Thus, prototypical ratios of the surface to bulk heating should be preserved.

Another requirement for the wall load is the need to achieve a reasonable fluence in a given calendar time. The integrated neutron wall load I is given by $I = P_{nw}At$ where P_{nw} is the neutron wall load, A is the device availability, and t is the operating period. As discussed later, the goal of FNT testing should be to reach ~6 $MW \cdot yr/m^2$ in 12 calendar years. Table XVIII shows the relationship between wall load and availability. The device availabilities required at wall loads of 1 and 2 MW/m^2 are 50 and 25%, respectively. The present ITER-EDA design[33] plans on achieving a <10% availability. Consequently, a higher wall load is needed. However, since the fusion device size and cost increase with wall load, improvements in achievable device availability are also necessary.

V.B.2. Fluence and Test Area

Fluence is one of the most critical parameters of primary interest to testing. The magnitude of this fluence will have a substantial impact on the selection and design of fusion testing facilities.

Fluence requirements for FNT were developed by considering the following factors:

1. time required to perform basic and multiple-effect experiments to observe groups of phenomena and to resolve technical issues associated with particular aspects of the blanket design, e.g., tritium release in solid breeders and thermomechanical interactions

TABLE XVIII

Neutron Wall Load and Availability Required to Reach 6 $MW \cdot yr/m^2$ Goal Fluence in 12 Calendar Years

Neutron Wall Load (MW/m^2)	Availability[a] (%)
1	50
1.5	33
2	25
2.5	20

[a]For pulsed plasma operation, this becomes the product of availability and plasma duty cycle. Therefore, at any given wall load, higher availability would be required.

2. time required to observe integrated behavior past the beginning of life (BOL) and during periods of significant radiation-induced changes in material properties and component behavior
3. time required to obtain data on key issues related to long-term component and system behavior such as corrosion and mass transfer, chemical interactions, stress relaxation, breeder burnup and tritium buildup, and containment
4. time required to obtain data on failure modes, effects, and rates
5. time required to perform the three stages of initial fusion break-in, concept verification, and component engineering development and reliability growth tests. The reliability growth testing phase is the most demanding on fluence requirements.

Before we proceed further, some definitions are necessary to ensure clarity. *Machine lifetime fluence* I_d refers to the time-integrated neutron wall load at the first wall during the machine lifetime:

$$I_d = P_{nw} \cdot A_d \cdot t_d \ ,$$

where

P_{nw} = average neutron wall load at the first wall of the fusion testing facility (MW/m^2)

A_d = machine availability averaged over t_d

t_d = machine lifetime (yr).

Test module fluence I_m is the time-integrated wall load as received at the front (first) wall of the test module:

$$I_m = P_{nw} \cdot A_m \cdot t_m \cdot T \ ,$$

where

P_{nw} = average neutron wall load at the first wall of the fusion testing facility (MW/m^2)

A_m = machine availability integrated over t_m

t_m = time during which a test module is placed in the machine

T = transmission factor (equivalent fraction of neutron wall load that reaches the test module).

In the literature, the integral wall load is quite often referred to as fluence. Despite the obvious misnomer here (fluence can be obtained from the foregoing expressions for I_d and I_m by replacing P_{nw} with the total neutron flux), we will occasionally follow the literature, relying on the units to make the distinction clear (megawatts per year per square metre for the time-integrated wall load I and neutrons per square metre for the true fluence).

The testing facility lifetime fluence should be much greater than the test module fluence because normally no test module is inserted for the entire lifetime of the machine t_d and because the transmission factor T is always less than unity. In the test program as currently envisaged, there are three stages of nuclear testing: initial fusion break-in, concept performance verification, and reliability growth. Different test articles may be used in each stage. During testing, some test articles are likely to fail or require replacement, also limiting the time any single test article can be irradiated.

Tests may be specified with isolation from the plasma for reasons of safety, reliability, and ease of maintenance. The existence of plasma-facing components, first-wall and multiple-containment structures for some tests reduces the neutron flux and energy spectrum at the test module. Reductions in neutron effects may be as much as a factor of 2 at the location of the tests due to a typical 1- to 2-cm steel and water enclosure.

The initial fusion break-in phase cumulative fluence at the test articles has been derived by considering several aspects. One of these is the testing time required for individual and multiple-effect tests at BOL. Examples include thermomechanical and tritium release tests. Rapid changes occur at BOL under irradiation in the range of 0 to 0.3 $MW \cdot yr/m^2$ (beyond this fluence, important changes still occur but at a slower rate). This is one reason for selecting 0.3 $MW \cdot yr/m^2$ as the fluence goal for the initial fusion break-in. Another reason is derived from the time to reach equilibrium for certain phenomena. Many phenomena such as tritium release and tritium permeation to the coolant, which will be discussed later, reach equilibrium in ~1 to 2 weeks. Therefore, each test campaign must be performed with continuous machine operation (100% availability) for ~1 to 2 weeks. About ten test campaigns are needed to perform tests under different conditions (temperature, flow rates, chemistry, etc.) to fully explore relevant phenomena and submodule behavior. If one assumes P_{nw} in the fusion testing facility to be ~1 to 2 MW/m^2, the initial fusion break-in phase requires a fluence in the range of 0.2 to 0.7 $MW \cdot yr/m^2$. Consequently, the 0.3 $MW \cdot yr/m^2$ specified for the initial fusion break-in tests is at the lower end of what is needed.

Concept performance verification is aimed at verifying performance beyond BOL and in the regime where changes in properties nearly saturate. Since concept verification testing results will be used to sharply reduce the number of specific blanket design concepts to only 2 or 3, it is necessary that testing in this stage be long enough to observe behavior under near-steady-state conditions. It is essential that the system behavior be observed when long equilibrium-time phenomena, such as corrosion and mass transfer, tritium permeation and containment, stress relaxation, and a variety of radiation effects, have reached some type of equilibrium. Table XIX presents a summary of expected radiation-induced effects in blankets in the 0 to 3 $MW \cdot yr/m^2$ fluence. Changes in mechanical properties of structural materials start to saturate at ~2 $MW \cdot yr/m^2$. During the concept verification stage, it is neither necessary nor

TABLE XIX

Summary of Expected Radiation-Induced Effects on Blankets Under Normal Operating Conditions in the 0 to 3 MW·yr/m^2 Fluence Range*

0 to 0.1 MW·yr/m^2 (at test module)
Some changes in thermophysical properties of nonmetals occur below 0.1 MW·yr/m^2 (e.g., thermal conductivity).
0.1 to 1 MW·yr/m^2 (at test module)
Several important effects become activated in the range of 0.1 to 1 MW·yr/m^2.
Radiation creep relaxation
Solid breeder sintering and cracking
Possible onset of breeder/multiplier swelling
Helium embrittlement
Changes in ductile-brittle transition temperature (DBTT)
1 to 3 MW·yr/m^2 (at test module)
Numerous individual effects and component (element) interactions occur here, particularly for metals, e.g.,
Changes in DBTT
Changes in fracture toughness
Helium embrittlement
Breeder and burnup effects
Breeder and multiplier swelling
Breeder/clad interactions

*Long-term radiation effects are not included.

practical to test components to their design EOL. However, it is desirable to test for a sufficiently long time, e.g., one-third to one-half of the projected life to provide confidence in concept selection. Therefore, a fluence of 1 to 3 MW·yr/m^2 is suggested for the concept verification phase.

The third stage of testing, namely, component engineering development and reliability growth (CEDAR) tests, is concerned with integrated behavior and endurance tests. The focus here is primarily on failure modes, effects, and rates. Because these tests are very demanding and require integrated component tests, the number of concepts to be tested should be limited. However, selecting one design concept at the end of concept verification, i.e., the beginning of the reliability testing stage, involves unacceptable risks because attaining the desired reliability goals may not be possible for a given concept regardless of how much testing and modifications in the design are made. Therefore, the number of blanket concepts at the beginning of the third stage should be two or three.

The required fluence during the CEDAR stage can be derived in several ways as follows.

1. *Experience from fission technology*. In the development of fast breeder reactors in Germany, irradiation of 2% of the total number of fuel pins up to roughly 40% of the goal lifetime fluence was specified as a prerequisite for the decision to start construction of DEMO. This corresponds in fusion to a 24-m^2 blanket test area for ~4 MW·yr/m^2 (assuming a 1200-m^2 first-wall area and a 10 MW·yr/m^2 lifetime in DEMO.

However, the reliability of the fusion blanket system must be higher than that of a fission reactor core because (a) the fission reactor can tolerate a rather large number of defective fuel pins (e.g., water purification systems in pressurized water reactors are designed to cope with 1% defective fuel rods) and (b) in contrast, the fusion reactor must be shut down immediately if one of the blanket modules leaks or if there is a local malfunction in the cooling system (leaks from blankets affect the vacuum environment necessary for plasma operation). Consequently, a malfunction in a blanket module requires a blanket exchange since an in situ repair is generally not possible. The time for this exchange [mean time to repair or replace (MTTR)] has been estimated to be at least 1 month for a machine designed for a fast blanket exchange. Therefore, the required mean time between failures (MTBF) of blanket modules must be exceptionally long. Indeed, MTBF required for blanket modules is substantially much longer than the lifetime as limited by neutron fluence. For example, for DEMO availability of 50%, the blanket system availability needs to be ~80%. For MTTR = 1 month, the MTBF for a blanket module is 26 yr for a blanket system with 80 modules. The demonstration of a higher reliability requires a larger number of test articles and/or a longer testing time.

2. *Reliability growth and demonstration methodology*. Reliability analysis and statistical methods have been used with great success to determine reliability testing requirements in aerospace, defense, and other industries.[34,35] We have attempted in this study to derive quantitative guidelines for testing requirements, including fluence, by applying available reliability analysis methods to the fusion blanket reliability testing problem.

Since the subjects of failures, availability analysis, and reliability growth and demonstration testing are neither widely studied nor commonly practiced in fusion, we have devoted Appendix A to this topic. Failures and reliability are among the most serious concerns in the engineering development of a component. The analysis shows that they will be even more so for FNT because (a) the mean time to recover from a failure is relatively long, (b) the surface area of the first wall is relatively large, and (c) the vacuum environment will not tolerate operation with leaks from blanket modules. All of these factors require that the failure rate be very low, or alternately, that MTBF be very long. Therefore, reliability growth and demonstration testing is extremely important for blanket development.

The results of the fluence required for reliability testing are given in Appendix A and can be briefly summarized for our purposes here. The results show that demonstrating a DEMO reactor availability of 60%, which implies a blanket system availability of >90%, requires a >20 MW·yr/m^2 testing fluence. Such a high testing fluence is practically unattainable because it greatly exceeds the estimated lifetime expected for any blanket to be developed in the time frame of interest and it cannot be achieved in a reasonable time with a fusion testing device that has a 1 to 2 MW·yr/m^2 wall load and a 30% availability. The results also show that the benefits increase with the neutron fluence at a relatively high rate up to a testing fluence of ~5 MW·yr/m^2. Beyond this fluence, the rate of increase in benefits becomes much slower. Therefore, we have selected ~4 to 6 MW·yr/m^2 as a target for fluence testing, which makes it possible to demonstrate a DEMO reactor availability of 50% with the optimistic assumptions of MTTR = 1 week and the simultaneous testing of 12 modules for a given blanket concept. If a larger sample size (i.e., more test modules) is used, there will be a saving in the test time required. However, the test time saving per additional sample decreases as the sample size increases (see Appendix A). With 4 to 6 MW·yr/m^2, the achievable availability is lower (>30%) with the more realistic assumption of MTTR = 1 month. The subject is examined in more detail in Appendix A.

We note that the number of test modules that should be tested simultaneously can possibly be reduced by testing in the so-called enhanced regime. In this testing regime, the test module is intentionally designed to increase, to some extent, destructive factors (e.g., stress or temperature) in the places of most probable failure. Those places are usually determined by accumulated energy estimation or stress analysis. Comparison of normal and enhanced tests in nonfusion environments could provide an estimate of the relative probability of failure. Thus, a smaller number of test modules would be needed in the fusion environment for a specified error in estimating the failure probability. One potential problem here is the lack of adequate nonfusion facilities. However, the concept of enhanced testing needs to be addressed in future studies.

If one considers the fluence requirements of the three stages of testing, i.e., initial fusion break-in (0.3 MW·yr/m^2), concept performance verification (1 to 3 MW·yr/m^2), and reliability growth (4 to 6 MW·yr/m^2), the total fluence required for FNT testing is >6 MW·yr/m^2.

The minimum surface area at the first wall for a test module is ~0.36 m^2 (60 × 60 cm) based on engineering scaling considerations. For example, to reproduce DEMO first-wall thermomechanical behavior under a testing neutron wall load of 1 MW/m^2, one must increase the lobe radius of a breeder-in-tube blanket concept from 0.35 to 0.5 m (Ref. 5). On the other hand, the test article for a partially integrated submodule test of any blanket concept should be large enough to address wall-end region effects on the performance issues. For a blanket module first-wall area of 1 × 1 m, this implies that a minimum surface area, including the spaces for preserving neutronics boundary conditions, of 0.6 × 0.6 m^2 is required. Some blanket concepts, e.g., those with self-cooled liquid-metal breeders or a ceramic matrix composite structure, might require a larger test module area. Assuming two to three blanket concepts to be tested in parallel during the reliability testing stage and 12 test modules per concept, one finds the total testing area required at the first wall to be >10 m^2. This area is also sufficient for the initial fusion break-in stage and concept verification stages. The initial fusion break-in stage will have a larger number of concepts, but the size of the test submodules can be smaller. During concept verification, four to six concepts may be tested, but the number of modules per concept can be only four to five.

V.B.3. Plasma Cycle Parameters and COT

Two areas of time-related parameters have a major impact on testing. The first is the plasma mode of operation, specifically the plasma burn and dwell times. The second is the minimum continuous operating time (COT), i.e., the minimum time required for continuous operation of the device with 100% availability.

At present, the designs for DEMO and commercial reactors are based on steady-state plasma operation because pulsing increases the capital cost[12,36,27] and has a large negative impact on reactor component reliability and failure rate. Therefore, steady-state plasma operation is desirable for FNT testing to simulate well the DEMO reactor environment. However, devices such as ITER are based on a pulsed plasma mode of operation. We examined the effects of plasma pulsing on blanket testing, and we attempted to derive requirements on the plasma burn time t_b, dwell time (t_d), and plasma duty cycle $t_b/(t_b + t_d)$.

Pulsing results in time-dependent changes in the environmental conditions for blanket testing, such as volumetric nuclear heating, surface heating, poloidal magnetic field, and the production of tritium and other neutron-induced reactions, and leads to several negative effects on testing, including

1. difficulty obtaining and sustaining equilibrium conditions for processes with long time constants
2. difficulties in maintaining equilibrium conditions during the dwell time because of the very short time constant for thermophysical parameters (e.g., temperature and temperature gradients)
3. undesirable changes in behavior that are not representative of equilibrium conditions
4. difficulty interpreting and extrapolating data.

Key blanket test issues to be affected by time-dependent environmental changes include thermal and fluid processes, structural response, and tritium release and inventory. The characteristic time constants calculated for these processes are shown for typical solid breeder and liquid-metal blankets in Tables XX and XXI. The characteristic time constant provides an indication of how fast a response will rise during the plasma startup and burn and how quickly it will decay during plasma shutdown and dwell time. For a given response F, the time-dependent maximum response after reaching quasi-equilibrium in a multiple number of back-to-back cycle operations (the number of cycles required to reach quasi-equilibrium is ~1/duty cycle) is calculated as

$$F_{max} = \frac{1 - e^{-t_b/\tau_c}}{1 - e^{-(t_b/\tau_c + t_d/\tau_c)}} \ ,$$

and the minimum response is written as

$$F_{min} = \frac{1 - e^{-t_b/\tau_c}}{1 - e^{-(t_b/\tau_c + t_d/\tau_c)}} e^{-t_d/\tau_c} \ ,$$

where F is a nondimensional response normalized to the equilibrium value and τ_c is the characteristic time constant. The allowable variation in a response during a specific test should not be any greater than 5% because

TABLE XX

Characteristic Time Constants in Solid Breeder Blankets

Process	Time Constant
Flow	
Solid breeder purge residence time	6 s
Coolant residence time	1 to 5 s
Thermal	
Structure conduction (5-mm metallic alloys)	1 to 2 s
Structure bulk temperature rise	
5-mm austenitic steel/water coolant	~1 s
5-mm ferritic steel/helium coolant	5 to 10 s
Solid breeder conduction	
Li_2O (400 to 800°C)	
10 MW/m^3	30 to 100 s
1 MW/m^3	300 to 900 s
$LiAlO_2$ (300 to 1000°C)	
10 MW/m^3	20 to 100 s
1 MW/m^3	180 to 700 s
Solid breeder bulk temperature rise	
Li_2O (400 to 800°C)	
10 MW/m^3	30 to 70 s
1 MW/m^3	80 to 220 s
$LiAlO_2$ (300 to 1000°C)	
10 MW/m^3	10 to 30 s
1 MW/m^3	40 to 100 s
Tritium	
Diffusion through steel	
300°C	150 days
500°C	10 days
Release in the breeder	
Li_2O 400 to 800°C	1 to 2 h
$LiAlO_2$ 300 to 1000°C	20 to 30 h

TABLE XXI

Characteristic Time Constants in Liquid-Metal Breeder Blankets

Process	Time Constant
Flow	
Coolant residence time	
First wall (V = 1 m/s)	~30 s
Back of blanket (V = 1 cm/s)	~100 s
Thermal	
Structure conduction (metallic alloys, 5 mm)	1 to 2 s
Structure bulk temperature rise	~4 s
Liquid breeder conduction	
Lithium	
Blanket front	1 s
Blanket back	20 s
LiPb	
Blanket front	4 s
Blanket back	300 s
Corrosion	
Dissolution of iron in lithium	40 days
Tritium	
Release in the breeder	
Lithium	30 days
LiPb	30 min
Diffusion through:	
Ferritic steel	
300°C	2230 days
500°C	62 days
Vanadium	
500°C	47 min
700°C	41 min

small changes in some fundamental quantities result in large changes in important phenomena; i.e., a 5% change in the solid breeder temperature results in a factor of 5 change in the tritium diffusion time constant. Therefore, if we are to preserve a response within 95% of equilibrium value [i.e., $F_{min} = 92.5\% < F(t) < F_{max} = 97.5\%$], the calculation based on the aforementioned equations suggests that

$$t_b > 1.1\tau_c$$

and

$$t_d < 0.05\tau_c \ .$$

This guideline makes the requirements on dwell time particularly difficult. For example, it requires keeping the dwell time to no more than 15 s for a front zone of solid breeder blanket designs to maintain the temperature variation to within 5% of equilibrium value. Moreover, the goal of a test is not just to reach equilibrium but to stay at equilibrium long enough to observe behavior. This has led to a consideration of burn time requirements approaching $3\tau_c$.

Desirable values for the burn and dwell times can be derived from the time constant approximations. A point to note is that testing in the fusion facilities involves interrelated phenomena with widely varying time constants. Thus, the burn time must be longer than $3\tau_c$ for important processes with the longest time constants. The dwell time should be shorter than $0.05\tau_c$ for the processes with the shortest time constants. From calculations in Tables XX and XXI, the burn time needs to be several days, and the dwell time should not exceed a few seconds. Clearly, steady-state operation is essential.

In a tokamak designed strictly for pulsed operation, the dwell time is determined by many considerations including the time to evacuate the plasma chamber and, more importantly, the time to cool down and reset the poloidal coils. Obtaining a short dwell time in a machine with pure inductive current drive is not possible. For example, ITER-EDA has a 1200-s dwell time.

Figure 4 shows the maximum and minimum temperature response of Li_2O in a position inside a breeder blanket test module under the ITER pulsed conditions of $t_b = 1000$ s and $t_d = 1200$ s with plasma startup and shutdown times of 50 and 100 s, respectively.[38] Figure 4 shows that the breeder temperature barely reaches steady state during the burn and drops to the inlet coolant temperature during the dwell time (the coolant inlet temperature was kept constant during the dwell by external means). Figure 5 shows the effect of the dwell time on the tritium release and inventory in Li_2O. Long dwell times will make the interpretation of tritium release very difficult and could lead to the occurrence of phenomena not otherwise accessible in steady-state operation.

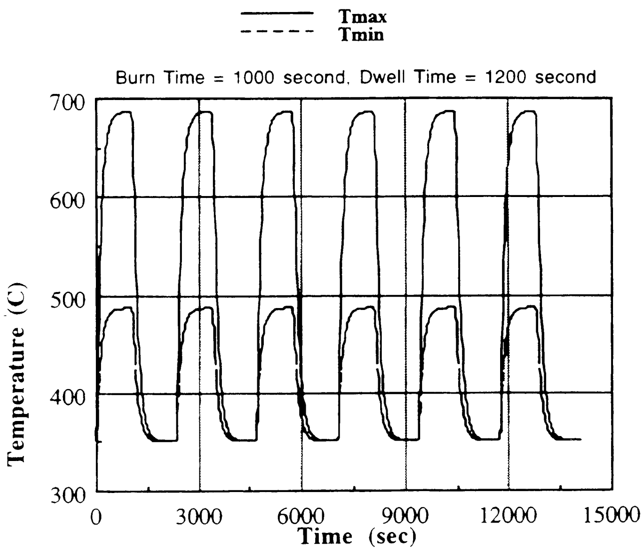

Fig. 4. Scaled-up Li_2O breeder temperature response to 1 MW/m^2 pulsed wall load (blanket front position $q''' = 9.4$ MW/m^3).

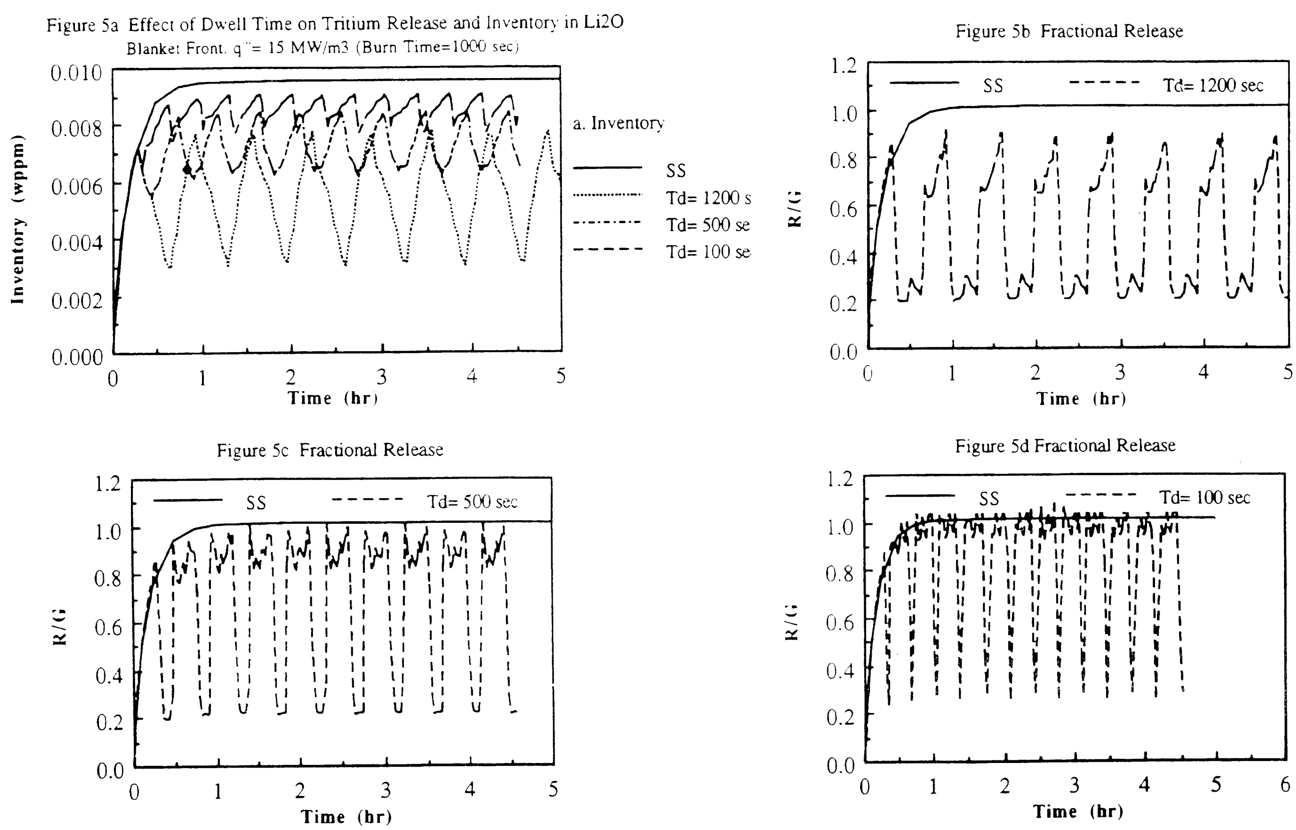

Fig. 5. Effect of dwell time on tritium release and inventory in Li_2O.

There is a need to provide many periods for test campaigns. During each period, the device must operate continuously (i.e., at 100% availability or load factor). This COT is for steady-state plasma operation or back-to-back plasma cycles in a pulsed system. The COT allows continuous operation of test modules to reach equilibrium and to observe cumulative effects, e.g., some radiation-induced changes, failures, and other nuclear phenomena. This COT is calculated to be ~1 to 2 weeks. Based on the time constants shown earlier, we find that shorter periods will result in a loss of substantial test information.

We conclude that steady-state plasma operation is very highly desirable for FNT testing. If pulsing is unavoidable, then the plasma duty cycle should be >80% with long plasma burn to achieve equilibrium for the most important processes.

VI. NEED FOR VNS AND DEFINITION OF OBJECTIVES AND DESIGN GUIDELINES

Section IV has clearly shown that testing in nonfusion facilities, albeit useful, cannot resolve the critical issues for FNT. Fusion facilities are required to test, develop, and qualify FNT components and to demonstrate short MTTR for DEMO. These testing requirements have also been quantified for the three stages of fusion testing: initial fusion break-in, concept verification, and component engineering development and reliability growth. Table XVII and Fig. 3 summarize the FNT primary requirements for the major parameters for testing in fusion facilities. The key requirements are a 1 to 2 MW/m^2 neutron wall load, steady-state plasma operation, many periods of continuous operation (100% availability) with each period 1 to 2 weeks, at least 6 $MW \cdot yr/m^2$ of neutron fluence, and >10 m^2 of test area at the first wall.

VI.A. Role of ITER

The key question now is how to satisfy these FNT requirements for fusion testing, specifically, what fusion facilities can best serve the FNT development needs. Since ITER is already in the EDA phase, it is prudent to examine first whether ITER can satisfy the FNT testing needs. Parameters of ITER (Ref. 33) are compared with those of the present devices of TFTR and DEMO in Table XXII.

Table XXIII summarizes the major R&D tasks to be accomplished prior to DEMO:

1. plasma performance
2. system integration
3. plasma support systems
4. material and FNT component performance and reliability and changeout cycle.

As designed in EDA, ITER (Ref. 33) will accomplish tasks 1, 2, and 3 with the possible exception of noninductive current drive and steady-state plasma operation. Task 4 will not be addressed adequately in ITER. This should be clear from comparing the FNT requirements in Table XVII to the ITER parameters listed in Table XXII. The primary reasons ITER cannot satisfy the FNT fusion testing and development requirements are

1. pulsed operation with low-duty cycle
2. low device availability
3. low fluence
4. short continuous operating time
5. small number of blanket test ports.

As shown in Sec. V, FNT testing requires steady-state plasma operation, and if this cannot be realized, the plasma duty cycle must be >80%. From Table XXII, ITER has a burn length of 1000 s, a dwell time of 1200 s, and a plasma duty cycle of ~45%. Therefore, based on the analysis in Sec. V, we find that ITER plasma mode of operation does not meet the FNT testing requirements.

The neutron fluence at the first wall of ITER is 0.1 $MW \cdot yr/m^2$ during 12 yr of a basic performance phase

TABLE XXII

Comparison of Parameters for Present Plasma Devices (TFTR/JET), ITER, and DEMO

	TFTR/JET	ITER	DEMO
Neutron wall load (MW/m^2)	<0.2	1	2 to 3
Plasma burn length (s)	1	1000	Steady state (or hours)
Plasma dwell time (s)	Very long	1200	0 (or <100 s)
Fuel cycle	Limited	Partial (fuel consumer)	Complete, self-sufficient
Thermal conversion efficiency (%)	0	0	>30
Net plant availability (%)	<1	1 to 10	>50
Fluence ($MW \cdot yr/m^2$)	$\sim 10^{-4}$	0.1 BPP 1.0 EPP	10 to 20

TABLE XXIII

Major R&D Tasks to be Accomplished Prior to DEMO

1. Plasma
   - Confinement
   - Impurity control and exhaust (divertor)
   - Disruption control
   - Current drive
2. System integration
3. Plasma support systems
   - Magnets
   - Heating
4. FNT components and materials (blanket, first wall, and high-performance divertors)
   - Material combination selection
   - Performance verification and concept validation
   - Show that the fuel cycle can be closed
   - Failure modes and effects
   - Remote maintenance demonstration
   - Reliability growth
   - Component lifetime
   - Mean time to recover from failure

(BPP) and 1 $MW \cdot yr/m^2$ during an additional 12-yr extended performance phase (EPP). Therefore, ITER fluence is 1.1 $MW \cdot yr/m^2$ compared with the ~6 $MW \cdot yr/m^2$ required for FNT testing (see Table XVII). Consequently, ITER alone cannot provide a database sufficient enough for the construction of FNT components in DEMO. The risk to the DEMO of relying on only ITER's low fluence is unacceptably large and will be quantified in Sec. VII.C.

Many (~100) periods of COT are required for FNT; i.e., at 100% availability, each period is 1 to 2 weeks. In ITER, the 0.1 $MW \cdot yr/m^2$ during the 12 yr of BPP means that the total operating time is <5 weeks, i.e., only ~3 full-power day/yr.

Section VII.C quantifies the technological risks to DEMO from relying only on the database from ITER. We also quantify its impact on DEMO of long time delays in schedule. The main point here is that ITER alone cannot provide a database sufficient to construct FNT components for DEMO.

VI.B. HVPNS Mission, Objectives, and Design Guidelines

The results in Secs. IV and VI.A strongly indicate that there is a definite need for HVPNS, which is a fusion facility to test, develop, and qualify FNT components and material combinations for DEMO. Such a facility must be a fusion facility because a prototypical environment must be provided and because plasma-based neutron sources are the only ones capable of providing neutrons in an appropriate test volume, as discussed in Sec. IV. We will occasionally abbreviate HVPNS as VNS.

The HVPNS mission is to complement ITER as a dedicated fusion facility to test, develop, and qualify FNT components and material combinations for DEMO. The blanket determines the critical path for FNT development and is a major focus for FNT testing in VNS. The design and material combination options to be tested are those that have a high potential for meeting the DEMO goals in safety, environmental impact, economics, reliability, and dependability. More detailed objectives and a testing strategy for VNS can be defined as follows:

1. stage I: initial fusion break-in
   a. initial exploration of performance in the fusion environment
   b. calibrate nonfusion tests against performance in the fusion environment
   c. observe effects of rapid changes in properties in early life
   d. initial check on codes and data
   e. test and develop experimental techniques and instrumentation
   f. narrow material combinations and design concepts in the fusion environment
2. stage II: concept performance verification
   a. verify performance beyond BOL and until changes in properties become small (changes in structure mechanical properties are substantial to ~2 $MW \cdot yr/m^2$)
   b. data on performance under normal operating conditions (temperature, stress, pressure drop, etc.) and under off-normal conditions (e.g., plasma disruption)
   c. data on initial failure modes and effects
   d. establish engineering feasibility of blankets (up to ~10 to 20% of lifetime)
   e. select two or three concepts for further development
3. stage III: CEDAR
   a. identify failure modes and effects
   b. iterative design/test/fix programs aimed at improving reliability and safety
   c. failure rate data: obtain a database sufficient to predict MTBF with sufficient confidence
   d. obtain data to predict mean time to replace (MTTR) for both planned outage and random failure
   e. obtain a database to predict overall availability of FNT components in DEMO.

The next question is what type of fusion facility VNS should be and what are the major parameters of VNS. Table XVII indicates that VNS clearly must have the following parameters to meet FNT development requirements:

1. neutron wall load of 1 to 2 MW/m^2
2. steady-state plasma operation
3. COT of 1 to 2 weeks
4. total neutron fluence of $\geq$6 $MW \cdot yr/m^2$ (0.3, 1 to 3, and 4 to 6 $MW \cdot yr/m^2$ for stages I, II, and III, respectively)
5. total test area at the first wall of $>$10 m^2.

One observation that can be made here is that FNT testing requires ~10 m^2 of test area at a 1 to 2 MW/m^2 neutron wall load, i.e., total fusion power of only ~20 MW. In contrast, plasma ignition in tokamaks requires >1500 MW of fusion power. Table XXIV compares the plasma ignition physics in tokamaks and FNT testing requirements. Plasma ignition physics requires ~1500 MW fusion power with a total integrated burn time of ~15 days. The tritium consumption, and hence the tritium supply requirement, for ignition physics is only ~3.5 kg. In contrast, FNT testing requires only ~20 MW of fusion power but a long test time of ~5 FPY. Because of the low fusion power, the tritium supply required for 5 FPY of FNT testing remains modest, ~5.6 kg. Combining the missions of plasma ignition testing (A) and FNT testing (B) into one facility leads to combining the large power requirements of A with the long test time of B; therefore, the tritium supply requirement becomes very large, ~420 kg. To put the magnitude of this tritium supply in perspective, consider the cost. At today's price of $20 million/kg, the cost of tritium for the combined (A + B) scenario is $8.4 billion, which is clearly unaffordable (and not justifiable). A more serious issue is the availability of the tritium supply. Since tritium production facilities for weapons have been shut down in the United States and Russia and since the half-life for tritium radioactive decay is only 12.3 yr, it is reasonable to deduce that no supply will be available from such a source in the time frame of 2006 to 2020. The only known supply is from the operation of heavy water-moderated Canada deuterium uranium (CANDU) reactors in Canada. This supply is estimated[39] at 2.5 kg/yr, which is clearly not sufficient for the combined (A + B) scenario but is more than adequate for the two separate facilities of A and B.

If a combined (A + B) facility were to be built, a tritium-producing blanket must first be constructed to internally produce tritium in such a facility. The problem here is that such a scenario assumes that a breeding blanket can be designed, constructed, and operated reliably and safely before obtaining the required database. The technical logic in such a scenario is flawed.

The foregoing discussion leads to the following points:

1. Although we derived the need for VNS from a detailed examination of FNT technical issues and an evaluation of facility capabilities, there is another way to arrive at the need for VNS. This is based on a comparative evaluation of a scenario of two separate facilities, one for FNT testing and the other for plasma ignition testing, to another scenario that combines ignition and FNT testing. It is worth noting that such a comparative evaluation was performed in earlier research[5,6,40] and led to a conclusion in favor of the two-separate-facilities approach.

2. A key requirement that should be imposed on VNS is that the fusion power should be kept small to minimize the tritium supply requirements. This suggests that the fusion power of VNS should be <150 MW to keep the annual tritium consumption at $\leq$2 kg/yr, if one assumes that the VNS overall availability is 30% and that ~20% of the wall area will be used by blanket test modules. Implicit in this guideline is that a base breeding blanket whose sole function is to produce tritium should not be used in VNS. Use of unproven technologies in VNS should be avoided to the maximum possible extent.

3. Limiting the fusion power in VNS to 150 MW or less requires that the plasma in VNS be in a driven

TABLE XXIV

Comparison of Physics and Nuclear Technology Requirements for Testing and Impact on Required Tritium Supply

Scenario	Fusion Power[a] (MW)	Integrated Burn Time[a]	Tritium Consumption (kg)
A. Separate facility for plasma ignition	1500	15 days	3.5
B. Separate facility for FNT	20	5 yr	5.6
(A + B) combined into one facility[b]	1500	5 yr	420

[a]Physics and FNT requirements are very dissimilar.
[b]Combining large power and high fluence leads to large tritium consumption requirements.

mode with $Q \sim 1$ to 3 (Q = fusion power output/drive power in).

Designing for maintainability and high availability is both an objective and a requirement for VNS. The required testing fluence of ~6 MW · yr/m² in 12 yr with wall loads in the range of 1.5 to 2 MW/m² is achieved by having the device availability in the range of 25 to 30% (see Table XVIII). As discussed earlier, achieving such a range of availability is by itself an important objective as a step toward DEMO. Involved in such a task is developing the failure recovery and remote maintenance techniques and safety procedures to reduce the device downtime. Table XXV summarizes the ground rules suggested for evolving VNS design concepts.

VI.C. Types of Confinement Concepts for VNS

There are two types of magnetic confinement concepts that can be considered for plasma-based VNS, namely, mirrors and tokamaks. One option, proposed by Kruglyakov et al., for a mirror-type facility, is called the gas dynamics trap (GDT). This concept has the advantage of reasonable confidence in its technical feasibility. Unfortunately, the maximum testing area available with GDT is ~0.5 to 0.75 m². Thus, a single GDT cannot provide the surface area required for FNT testing (>10 m²; see Table XVII). Hence, it is suitable for VNS if the cost is low enough that construction of several devices is cheaper than a single tokamak HVPNS.

Tokamaks appear to offer the most attractive approach to VNS at present. A driven plasma is acceptable for VNS since FNT testing requires only that neutrons be produced steadily over a large area, regardless of whether neutrons are produced by ignited or driven plasmas. This fact is a key reason why an attractive design envelope can be identified for VNS. At Q (ratio of fusion power to plasma input power) of ~1 to 3, one can show that a tokamak with TFTR/JET types of devices supplemented by noninductive current drive and a divertor can satisfy FNT requirements and provide VNS at a relatively low cost.

A number of design options for tokamak HVPNS are outlined in Appendix B for both standard as well as very low aspect ratios. Tokamak designs with normal-conducting toroidal field (TF) coils (TFCs) result in the smallest size and the desired low fusion power.

VII. COST/BENEFIT/RISK ANALYSIS OF SCENARIOS TO DEMO WITH AND WITHOUT VNS

The purpose of this section is to quantitatively compare various scenarios for fusion facilities from now to DEMO with and without VNS. Our particular focus is

TABLE XXV

Guidelines for Evolving HVPNS Design

HVPNS mission	
To serve as a test facility for FNT and to provide a database sufficient to construct FNT components for DEMO.	
Testing requirements	
HVPNS must satisfy the following FNT testing requirements:	
Wall load	1 to 2 MW/m²
Neutron fluence	≥6 MW · yr/m²
Plasma mode of operation	Steady state or long plasma burn with duty cycle >80%
Minimum test area per test article	0.36 m²
Total test area[a]	>10 m² (up to ~20 m²)
Device availability	>25%
Minimum COT[b]	1 to 2 weeks
Magnetic field at the test region	>2 T
Design features/constraints	
HVPNS design should be consistent with the following features/constraints:	
Configuration, remote maintenance, and other design features must emphasize the reliability of basic device components and rapid replacement of device components and test articles.	
Device must be able to test all candidate blanket concepts for DEMO including liquid metal and beryllium.	
The fusion power must be low enough that the tritium consumption does not exceed that available from external sources (e.g., the fusion power should be <150 MW with 30% of the first wall occupied by test modules).	
The capital cost of HVPNS should be kept as low as possible (e.g., <25% of that for ITER).	
The power consumption of the HVPNS site (e.g., from normal copper coils, current drive, etc.) should be kept reasonably low (e.g., <700 MW).	

[a]However, test devices that can satisfy part of the total testing area requirements should be considered in a cost/benefit/risk analysis.
[b]Periods with 100% availability.

on the fusion nuclear testing necessary to construct the DEMO nuclear components. A cost/benefit/risk analysis is conducted as the basis for comparing the various scenarios.

The scenarios considered here are shown in Fig. 6. Obviously, other variations of these scenarios are possible. However, we used experience and knowledge to limit the scenarios to the ones essential for understanding the impact of adding or eliminating VNS. The scenarios are as follows.

VII.A. Fusion Facility Scenarios

Scenario I: ITER alone. In this scenario, ITER as presently envisaged is considered to be the only facility available for nuclear testing prior to DEMO; ITER will have two phases: BPP and EPP. We considered each to run for 12 yr. We did not explicitly account for the ITER downtime between BPP and EPP. If this period is 2 yr, for example, the EPP duration can be shortened to 10 yr at the same total fluence with little impact on the comparative result derived here.

Scenario II: ITER (BPP only) + VNS. In this scenario, VNS operates parallel to the BPP phase of ITER. The EPP phase is eliminated.

Scenario III: ITER (BPP + EPP) + VNS. This scenario is the same as scenario II except ITER will also operate the second phase (EPP). The VNS operation can also extend beyond the initial 12 yr.

Scenario IV: ITER + VNS delayed. In this scenario, ITER operates for the two phases BPP and EPP as in scenarios I and III. However, the VNS start of operation is delayed to coincide with the beginning of ITER EPP.

Scenario Ib: ITER alone but at high fluence. This is the same as scenario I except the fluence accumulated in ITER is much higher here: 0.3 $MW \cdot yr/m^2$ during BPP and 3 $MW \cdot yr/m^2$ during EPP.

In addition to the foregoing scenarios, another interesting and promising scenario will be discussed later. In this scenario, VNS is operated parallel to ITER BPP for 12 yr, beyond which data from VNS are used to construct a hot DEMO-type blanket on ITER, which makes ITER operation during the second phase a "pre-DEMO" type. Time and resources have not permitted full evaluation of this interesting ITER pre-DEMO scenario. The quantitative comparison for scenarios I through IV plus Ib will be addressed here first followed by some limited results for the ITER pre-DEMO scenario.

We address below the areas of cost/benefit/risk analysis that are among the most important factors to decision makers, namely,

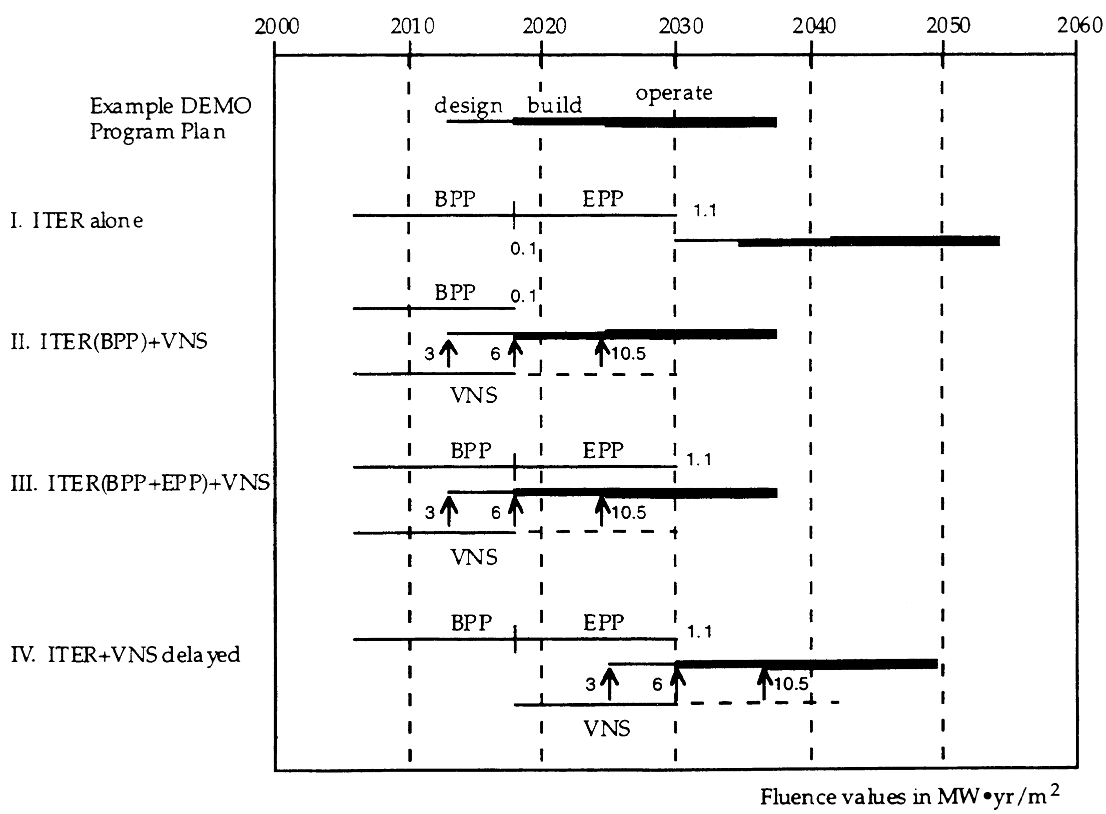

Fig. 6. Scenarios for major fusion devices leading to a DEMO.

1. *Time schedule to DEMO*. When can DEMO begin operation under the different scenarios?
2. *Technical risk*. What are the technical risks to DEMO associated with the various scenarios?
3. *Costs*. What is the total cost of the R&D program to DEMO with the various scenarios? What is the impact on the near-term financial requirements?

VII.B. Time Schedule

The date for the beginning of fusion DEMO operation is important from technical and programmatic viewpoints. In "roll backward" technical planning, the date of the DEMO is a key factor in determining the pace of R&D to DEMO. In "roll forward" technical planning, the date of the DEMO must be consistent with the technical results and schedule achievable with the planned R&D program.

From a programmatic viewpoint, the date for the DEMO signals when fusion can reach its goal. Since fusion R&D is funded to produce a practical energy source, the date for the DEMO is critical because

1. It shows when the public can expect the new energy source to play a role in a power-dependent economy. Nearer term options generally receive higher priority in public funding.

2. It provides an indication of the total cost of R&D. With worldwide spending of approximately \$1.2 billion/yr on fusion R&D, tens of years of delay in DEMO operation can substantially increase the cost of developing fusion.

Most world programs state the year 2025 as a target date for the beginning of DEMO operation. We used this as an example case in Fig. 6. We also assume 7 yr of construction and 3 yr of final engineering design. For the example case, the final design begins in the year 2015, and construction begins in the year 2018. From results in Sec. VI.B, ~3 and 6 $MW \cdot yr/m^2$ of testing for DEMO nuclear components must be available by the beginning of the final design and at the beginning of construction, respectively.

The scenarios (I through IV and Ib) considered here, illustrated in Fig. 6, vary considerably in the achievable time schedule to DEMO. Scenario I, ITER alone, achieves only 1 $MW \cdot yr/m^2$ by the end of EPP, i.e., by the year 2030. The clear conclusion here is that ITER alone cannot provide a sufficient database to construct with reasonable confidence the DEMO nuclear components. The risk of constructing the DEMO with such low levels of FNT testing is unacceptably high, as will be quantified shortly.

Even if one were to accept such a very high risk, the DEMO operation with the ITER-alone scenario is the year 2042, as illustrated in Fig. 6. Therefore, the conclusion here is that the ITER-alone strategy results in an unacceptably high level of risk, and even with such a risk, the DEMO operation is delayed by 17 yr.

In contrast, the scenario with VNS parallel to ITER BPP, i.e., scenario II, meets the FNT testing requirements, provides high confidence (to be quantified shortly) in DEMO, and allows the DEMO operation to begin on schedule, i.e., by the year 2025.

Table XXVI provides for all the scenarios considered here, the DEMO start date, the test fluences achievable prior to the beginning of DEMO construction, and DEMO operation. A qualitative measure of confidence in DEMO is also indicated, which will be addressed quantitatively in Sec. VII.C.

Scenario III is the same as scenario II, with VNS operating parallel to ITER BPP, except that scenario III assumes that ITER will also continue operation into the EPP phase. Figure 6 and Table XXVI indicate that adding EPP to ITER clearly has very little effect on the DEMO start date or the confidence level in the FNT components in DEMO. This point, to be discussed in more detail later, seems to indicate that the second phase of ITER (EPP), as presently planned, costs very much (approximately \$500 million/yr) and achieves few benefits, compared with VNS.

Scenario IV is similar to scenario III except that the VNS start of operation is delayed to begin after the end of ITER BPP. This scenario achieves the same level of confidence as scenarios II and III with VNS, but it delays the start of DEMO operation to the year 2037. In addition, this delay of VNS precludes the use of VNS information to improve ITER EPP; e.g., it eliminates the possibility of converting the ITER EPP phase into pre-DEMO with full hot reactor relevant blankets.

VII.C. Technical Risk

The evaluation of technical risk is a crucial tool in decision making. Here, we evaluate risk by quantifying the probability of meeting the technical objectives of DEMO, with the focus on the nuclear components. However, another critical aspect of risk, which is not considered here, is the possible programmatic consequences of excessive premature failures of components. Among the potential burdens are the time and resources to fix the problems encountered, programmatic disenchantment, and the problem of erasing "bad data." The last item, "bad data," stems from experiences drawn from the fission industry. Current fission technology is replete with examples of poorly characterized data sets, often generated with nonprototypical tests to meet near-term needs. However, this has in many cases left a legacy of large scatter and uncertainty in data sets used by regulatory agencies to predict worst-case performance limits. In short, it is difficult to make the case to regulatory agencies for retroactively separating good data from bad, with the extremes controlling conservative predictions of behavior.[41]

As discussed previously, one of the important requirements set by industry and utility for DEMO is the

TABLE XXVI

DEMO Start Date and Testing Fluence Achieved for Various Scenarios

Scenario	DEMO Operation Start Date	Fluence at Start of DEMO Construction ($MW \cdot yr/m^2$)	Fluence at Start of DEMO Operation ($MW \cdot yr/m^2$)	Confidence Level in DEMO[a]
I: ITER only	2042	1	1	Very low
II: ITER/BPP + VNS	2025	6	10.5	High
III: ITER + VNS	2025	6	10.5	High
IV: ITER + delayed VNS	2037	6	10.5	High
Ib: ITER only, high fluence	2042	3	3	Low

[a]Confidence level is quantified under technical risk in Sec. VII.C.

demonstration of dependability and reliability. As discussed in Sec. II, a DEMO availability goal of 60% is typically used in worldwide fusion studies based on private sector requirements according to experiences from current conventional power plants. The DEMO reactor availability is given by (see Appendix A)

$$\text{DEMO } \textit{reactor availability} = A_R = \frac{1}{1 + \sum_i (\textit{outage risk})_i} ,$$

where i represents a reactor component and the outage risk is defined as

$$\textit{outage risk}_i = \text{MTTR}_i \times \textit{failure rate}_i = \text{MTTR}_i/\text{MTBF}_i ,$$

where MTTR_i is the mean downtime to recover from a failure in component i and MTBF_i is the MTBF for component i.

Achieving a low-outage-rate operation requires that high reliability of the component system and good accessibility for maintenance and repair (low failure rate and mean time to repair) be achieved. While the mean time to repair (MTTR) is determined by whether the reactor design configuration characteristics can be maintained in accordance with prescribed procedures and resources, a low component failure rate necessitates the need of a long MTBF. The parameters that directly affect the percentage of time that a system is available for use are MTBF and MTTR. Notice that a number of combinations may be possible to achieve the same desired level of system availability. A component can be designed and built to have high MTBF with respect to MTTR, or ease of maintenance can be designed into the system, which would result in short maintenance times. Achieving the desired MTTR is influenced by environment, cost, and other external constraints. The most practical way to achieve high availability is to supplement the design for reliability with a design for efficient and rapid repair and a high degree of maintainability. However, as shown in Appendix A, MTTR for tokamaks is predicted to be long, which necessitates that MTBF be long to achieve the desired availability.

We cannot be assured with a high degree of confidence that the reliability of the blanket concept selected will be adequate for DEMO. Indeed, clearly, at the outset, an extensive component development effort is required. The reliability level of components is established at the design phase, and subsequent testing and production will not raise the reliability without a basic design change/modification or improvement. The way to measure component reliability is to test completed products under conditions that simulate real life. Unproven component reliabilities can be estimated from the proven reliabilities of components of similar design and application, if such design and applications exist. However, high confidence in component performance in entirely new applications, such as fusion, can be obtained only from testing in relevant environments. One simply cannot assess reliability without data, and of course, the more data available, the more confidence one will have in the estimated reliability level.

For each of the scenarios defined earlier in this section, the risk to the DEMO can be quantified in at least one critical area: the DEMO blanket system availability (subsequently DEMO reactor availability). Typically, availability goals, such as those shown in Table A.I, would be used to establish blanket module availability requirements, and subsequent testing would be used to confirm the achievement of such requirements to a specified level of statistical confidence. Two approaches were adopted to quantify the comparative measures:

Approach I. Calculate the blanket system availability and the corresponding DEMO reactor availability achievable with 80% confidence.

Approach II. Calculate the confidence level in achieving the DEMO blanket system and reactor availability goals as given earlier, i.e., DEMO reactor availability of 60% and the alternative case of 30%.

Mathematically, we need to define a framework for capturing the notion of confidence. The most commonly used framework is the Poisson model, also known as the constant failure rate model. The Poisson distribution can be used to relate the number of testing failures, the confidence level, the testing time, and the estimated MTBF. The Poisson model asserts that the component fails at random points in time but with a constant long-term average occurrence rate. Although the Poisson model might not really describe the failure behavior of the FNT components (in particular, material properties change with time because of irradiation effects), it is widely used and often is quite adequate. In principle, the component failure rates are measurable, and if we in fact had vast amounts of testing data, we would have measured them closely enough. Figure 7 shows the upper statistical confidence level as a function of the test time in MTBF multiples and the number of failures that occurred during the test.[28] As shown, if an initial MTBF is assumed and a test is conducted for two times the MTBF with the result of two failures, then the confidence level (i.e., the probability that the actual MTBF is greater than or equal to the estimated value) for the assumed MTBF would be only ~30%. If larger numbers of failures occur, then very much longer test times are required to give high confidence that the actual MTBF is greater than or equal to the estimated MTBF. Conversely, if very few failures occur during the test period, high confidence levels can be provided with relatively short test periods. This implies that quick reliability confirmation might be obtained if the "as-demonstrated" MTBF of the component is higher than its required MTBF.

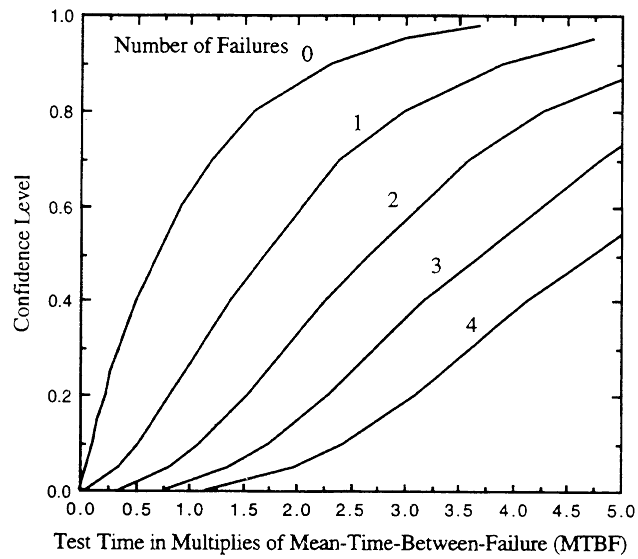

Fig. 7. Upper statistical confidence level as a function of test time in multiples of MTBF for time-terminated reliability tests (Poisson distribution). Results are given for different numbers of failures.

VII.C.1. Results of Approach I

The results for approach I are summarized in Table XXVII and Figs. 8 and 9 (with the computational method described in Sec. A.III). One failure was assumed during the entire period of the test. At such a stage of development, failures are expected to occur. Note, however, that if more than one failure is assumed, it will take a much longer test time, i.e., a higher fluence, to achieve the DEMO availability goal. Strictly speaking, the ITER-alone scenario provides a fluence that is barely sufficient for the FNT testing stages of initial fusion break-in and concept verification and therefore does not provide any real component reliability growth and demonstration testing. However, for each scenario, a fluence level of 0.3 MW·yr/m^2 is considered for initial fusion break-in testing while the remaining fluence is dedicated for the reliability growth/demonstration testing. To facilitate the comparison, consider an experience factor of 0.8 and 12 test modules in both VNS and ITER. The experience factor is meant to reduce the total test credit for parallel tests to account for the fact that similar failure causes may

TABLE XXVII

Summary of DEMO Reactor Availability (%) Obtainable with 80% Confidence Compared with Calendar Year in the Various Scenarios*

Scenario	MTTR = 1 week[a]			MTTR = 1 month		
	2013	2018	2025	2013	2018	2025
I: ITER alone	0	0	7.1	0	0	1.8
II. ITER (BPP) + VNS	42.3	47.4	53.8	15.2	23.1	31.0
III: ITER(BPP + EPP) + VNS	42.3	47.4	54.5	15.2	23.1	31.9
IV: ITER + delayed VNS	0	0	37.5	0	0	14.8
Ib: ITER alone (high fluence)	0	0	25.2	0	0	8

*There are 12 test modules, one failure during the test, and an experience factor = 0.8.
[a]MTTR = mean time to repair.

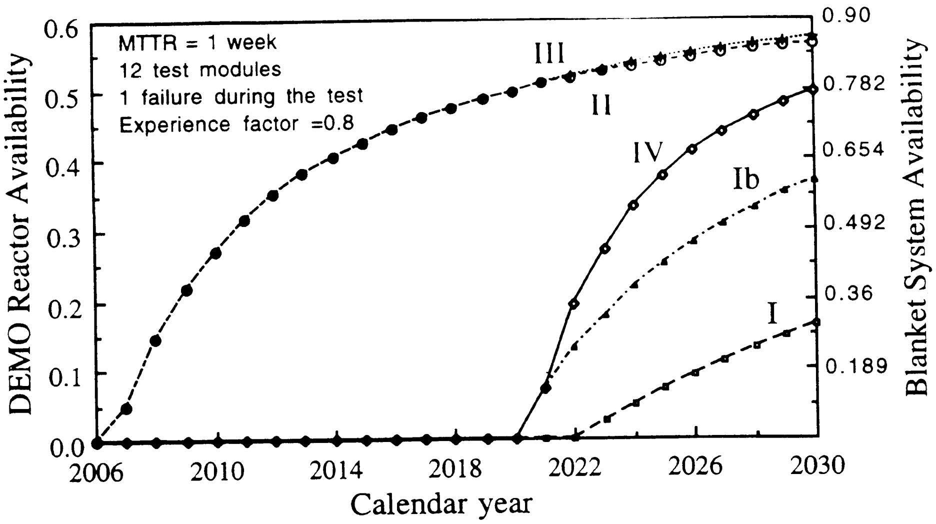

Fig. 8. The DEMO reactor availabilities obtainable with 80% confidence for different testing scenarios, MTTR = 1 week (scenario I is ITER only; scenario II is ITER BPP + VNS; scenario III is ITER + VNS; scenario IV is ITER + delayed VNS; and scenario Ib is ITER only, high fluence).

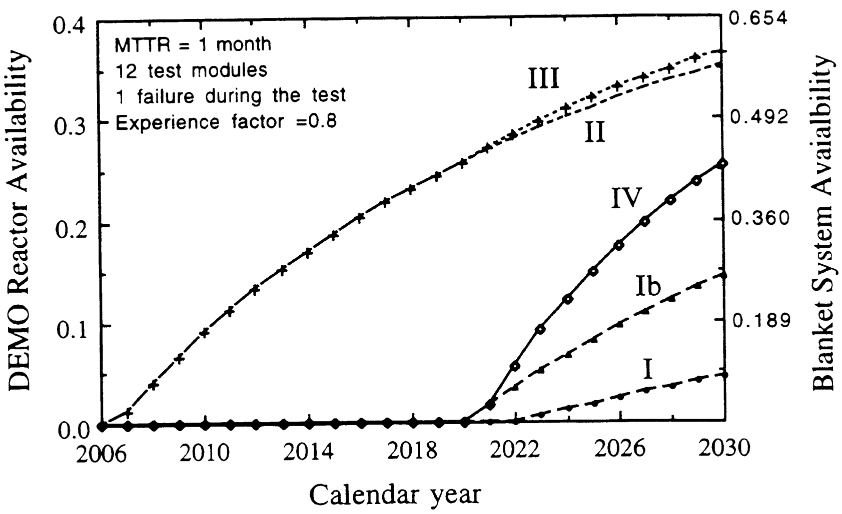

Fig. 9. The DEMO reactor availabilities obtainable with 80% confidence for different testing scenarios, MTTR = 1 month (scenario I is ITER only; scenario II is ITER BPP + VNS; scenario III is ITER + VNS; scenario IV is ITER + delayed VNS; and scenario Ib is ITER only, high fluence).

be observed in different blanket test modules. The maximum value for the experience factor is 1.0. A lower experience factor reduces the test benefits of parallel tests (see Appendix A). The results show that testing in the ITER-alone scenario could confirm with an 80% confidence level for the achievement of a DEMO reactor an availability of only ~7.1% for MTTR = 1 week by the year 2025. This reduces to ~1.8% if MTTR equals 1 month. In contrast, with scenario II (i.e., VNS operating parallel to ITER BPP), it is possible to confirm with an 80% confidence level the achievable DEMO reactor availability of ~54% if MTTR = 1 week and of ~31% if MTTR = 1 month. The results for scenario III suggest that testing in ITER EPP as presently planned does not provide any significant increase in DEMO reactor availability achievable with an 80% confidence level. Furthermore, the results for scenario IV indicate that delaying VNS would delay the start of DEMO operation at the same confidence level. Therefore, VNS makes it possible to come close to demonstrating the achievable DEMO goals without another machine between ITER and DEMO if MTTR = 1 week. If longer machine shutdown times are required such as MTTR = 1 month, a DEMO reactor availability of 31% can be certified with a VNS device. Without VNS, the confidence to proceed with DEMO is too low to be acceptable.

VII.C.2. Results of Approach II

Approach II of determining the risk in achieving the DEMO availability goals provides another useful perspective. The results show that with scenario II (ITER/VNS parallel strategy), there is substantial confidence (~63%) in achieving a DEMO reactor availability of 60% if MTTR = 1 week. In contrast, even with this optimistic assumption about MTTR, there is no appreciable level of confidence (<1%) that the DEMO will achieve this goal with scenario I of the ITER-alone strategy. However, notice that confirming an 80% confidence (or greater) in the achievable parameters is generally required for major and critical projects such as DEMO. To ascertain that the risk associated with the achievable reactor availability goal can be acceptable, we examined reactor availabilities of 50% in addition to 30%. Figure 10 shows the confidence level in achieving a DEMO reactor availability of 50% for different FNT testing scenarios, if one assumes MTTR = 1 week. The ITER-alone strategy (scenario I) provides nearly zero confidence by the year 2018 (end of BPP) and by the year 2025 for a DEMO availability of 50%. Conversely, scenario II with VNS achieves 72% confidence by the year 2018 and 91% confidence by the year 2025. The high fluence ITER-alone scenario Ib results in a confidence of only 14% for a DEMO availability of 50% at the year 2025. If the more realistic value of MTTR = 1 month is considered, the confidence level with the ITER-alone strategy becomes even lower. But, with VNS, the confidence level would be adequate for confirming the achievable reactor availability of 30% and would be insufficient for ensuring an availability of 50% achievable by the year 2025, as shown in Table XXVIII and Fig. 11. The increment in the confidence level by employing the ITER EPP nuclear testing is negligible.

VII.C.3. Pre-DEMO Scenario

The foregoing analyses suggest that the blanket (and other FNT component) tests in ITER alone cannot demonstrate an availability larger than a few percent (~4%) even if the quality of ITER testing (i.e., steady state instead of high rate of pulsing, etc.) were improved. Tests in VNS will demonstrate much higher confidence for much higher DEMO availability. However, high confidence in demonstrating the ultimate goal set by industry for DEMO, i.e., ≥60%, does not appear possible. This points to the need for a pre-DEMO.

A pre-DEMO device would be a device between ITER and DEMO. If such a device were of the same type as ITER (size, fusion power, etc.), it would not improve the situation for several reasons:

1. ITER would have exhausted the external tritium supply, and there would be no tritium left to operate the pre-DEMO device.

2. The testing information from ITER would not allow a full hot breeding blanket to be constructed with acceptable confidence in the pre-DEMO device.

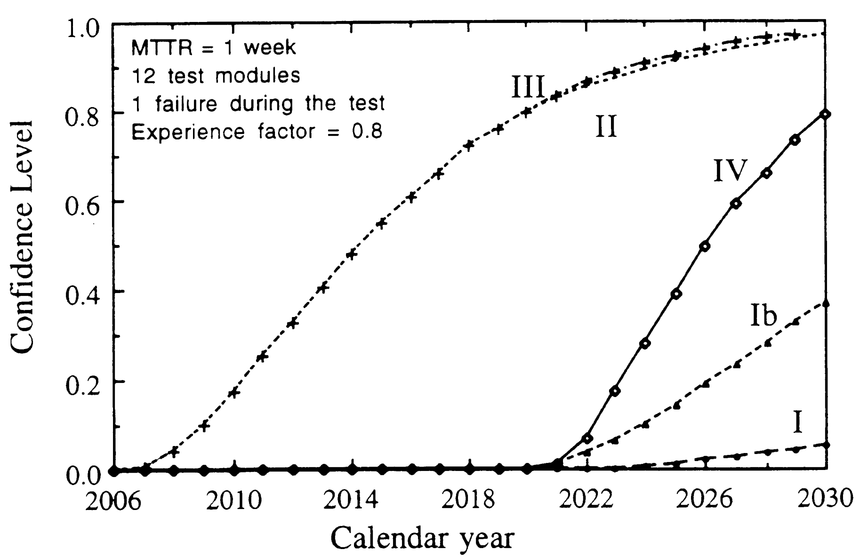

Fig. 10. Confidence levels of confirming DEMO reactor availability of 50% for different testing scenarios, MTTR = 1 week (scenario I is ITER only; scenario II is ITER BPP + VNS; scenario III is ITER + VNS; scenario IV is ITER + delayed VNS; and scenario Ib is ITER only, high fluence).

TABLE XXVIII

Summary of Confidence Level (%) Obtainable in DEMO Reactor Availability*

Scenario	I		II		III		IV		Ib	
DEMO availability	30%	50%	30%	50%	30%	50%	30%	50%	30%	50%
MTTR = 1 week[a]										
6 test modules										
2013	0	0	80	19	80	19	0	0	0	0
2018	0	0	~100	46.38	~100	46.38	0	0	0	0
2025	3.8	0	~100	67.0	~100	70.16	76.56	17.1	40.6	5.7
MTTR = 1 week										
12 test modules										
2013	0	0	~100	40.7	~100	40.7	0	0	0	0
2018	0	0	~100	72.0	~100	72.0	0	0	0	0
2025	8.8	0	~100	91.0	~100	92.0	>97	38.9	70.6	14
MTTR = 1 month										
12 test modules										
2013	0	0	40.2	4.56	40.2	4.56	0	0	0	0
2018	0	0	63	11.6	63	11.6	0	0	0	0
2025	1.0	0	86	23.89	87.5	25.69	32.17	4.4	11.1	1.7

Note: There are one failure during the test and an experience factor = 0.8.
[a]MTTR = mean time to repair.

3. A pre-DEMO device would add a very substantial burden to the total cost of fusion R&D and would further delay the DEMO operation to the point that fusion would not play a role in world energy production in the twenty-first century.

Alternately, the pre-DEMO device could be either an upgraded ITER machine or the DEMO machine itself, operated during the initial phase not as a DEMO reactor but as a blanket test facility. In the first approach, ITER would have to be modified by installing the selected hot breeding blankets including the required external systems for the extraction of heat and tritium.

This scenario is illustrated in Fig. 12. If VNS operates parallel to ITER BPP and the results from ITER

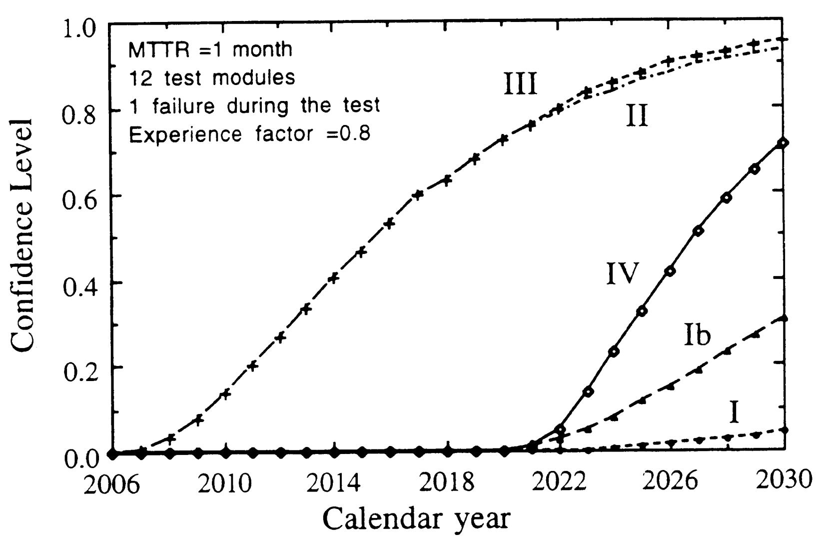

Fig. 11. Confidence levels of confirming DEMO reactor availability of 30% for different testing scenarios, MTTR = 1 month (scenario I is ITER only; scenario II is ITER BPP + VNS; scenario III is ITER + VNS; scenario IV is ITER + delayed VNS; and scenario Ib is ITER only, high fluence).

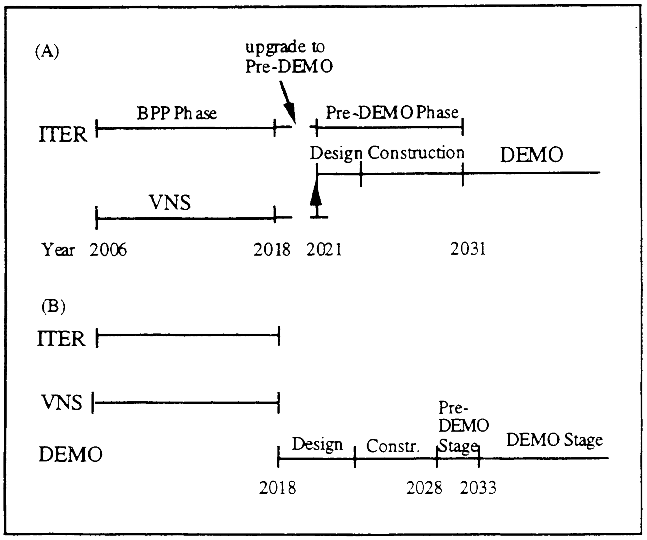

Fig. 12. Promising scenario of (A) VNS parallel to ITER BPP and ITER second phase becomes a pre-DEMO with full hot breeding blanket and (B) VNS parallel to ITER BPP and operating DEMO for 5 yr as pre-DEMO.

and VNS are good in all technical areas, then there will be sufficient data to construct a full hot breeding blanket on ITER, which allows the ITER second phase to become a pre-DEMO with full system integration. Obviously, this saves the cost of a completely new pre-DEMO device after ITER. It also shortens the time schedule. Of particular importance, this scenario will further increase the confidence in the DEMO achieving its goals.

As shown in Fig. 13, an achievable DEMO reactor availability of 41% can be confirmed with 80% confidence after 10 yr of ITER/pre-DEMO operation, assuming MTTR = 1 month. An additional increase in reactor availability to 56% can be recognized at the same confidence level if ITER/pre-DEMO can be operated at a DEMO neutron wall load of 3 MW/m^2.

VII.C.4. Summary of Technical Risk

Key conclusions from the technical risk assessment can be summarized as follows:

1. Blanket tests in ITER cannot demonstrate a blanket system availability larger than a few percent. The confidence to proceed with DEMO based on FNT testing in ITER alone is too low to be acceptable.

2. The VNS makes it possible to come close to demonstrating the achievable DEMO reactor availability goal of 50% with sufficient confidence without a machine between ITER and DEMO if MTTR = 1 week. If longer machine shutdown times are required such as MTTR = 1 month, a DEMO reactor availability of 31% can be certified with a VNS device as opposed to 1.8% with the ITER-alone strategy.

3. The contribution of blanket tests in the presently envisaged ITER EPP to the reliability testing is negligible.

4. In optimizing system design, maintainability requirements (the length of MTTR) and maintainability

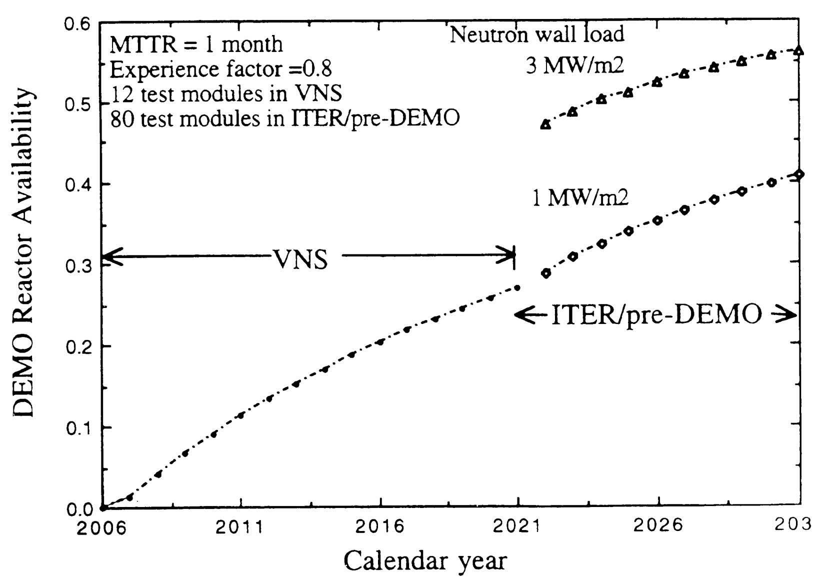

Fig. 13. The DEMO reactor availabilities achievable with 80% confidence with the scenario in which VNS operates parallel to ITER BPP, and the second phase of ITER is upgraded to pre-DEMO using VNS testing data (results are for MTTR = 1 month).

design criteria are critical. If MTTR is >1 week, the reliability improvement in blanket components (increase in MTBF) must be more substantial.

5. With the ITER-alone strategy, the problem of a high technical risk to DEMO cannot even be credibly resolved by assuming another pre-DEMO device between ITER and the DEMO.

6. Operating VNS parallel to ITER BPP makes it possible to envision a credible scenario in which the second phase of ITER is upgraded to a pre-DEMO operating mode in which a fully integrated system, including a hot breeding blanket, is tested. This scenario allows reasonable confidence in meeting the DEMO goals.

VII.D. Costs

All of the foregoing considerations clearly indicate that VNS is not only desirable but is a necessary element in the success of the world fusion R&D program toward DEMO. The question is whether it adds a substantial financial burden. Below, we address cost considerations that show that VNS is affordable and most likely will result in substantial savings in the overall cost of R&D toward DEMO.

Two aspects of financial considerations were addressed: (a) the total cost of fusion R&D from now until DEMO and (b) expenditure profile, i.e., the annual cost and whether it peaks to an unaffordable level in certain years. With regard to the expenditure estimate, Table XXIX provides the costing assumptions used in the calculation. They are for comparative and illustrative purposes and are not meant to be precise numbers. The capital cost for ITER is approximately \$8 billion in 1994 dollars. Relative to ITER, VNS has a smaller first-wall surface area by a factor of ~20 less than that of ITER, which leads to a significant reduction in the capital cost. To be specific, we note that the VNS design envelope with normal copper coils has an estimated capital cost in the range of 15 to 25% of that of ITER. We use here the upper value of 25%. The estimated operating cost of ITER is approximately \$400 million/yr. Relative to this, we estimate the VNS operating cost to be approximately \$200 million/yr including the power consumption cost. The tritium supply cost is calculated at \$20 million/kg.

TABLE XXIX

Costing Assumptions for Scenario Evaluations

Device	Capital Cost	Operating Cost	Tritium Supply Cost
ITER	\$8 billion	\$400 million/yr	BPP: \$15 million/yr EPP: \$150 million/yr
VNS	\$2 billion	\$200 million/yr	\$36 million/yr

VII.D.1. Total R&D Cost

The results in Table XXX show the total capital cost and the operating and tritium supply costs for the various scenarios. The lowest cost strategy for fusion R&D is with scenario II: VNS parallel to ITER BPP. The uncertainties in the cost estimate are not critical here. The key point is that VNS, besides being necessary from a technical standpoint, does not really add a cost burden; it actually provides cost savings. Another indication of the cost savings of operating VNS parallel to ITER is a minimum 17-yr reduction in the period from now to DEMO. At present, the world expenditure on fusion R&D is \$1.2 billion; this shortening of time to DEMO made possible by VNS provides additional savings of approximately \$20 billion. This cost savings becomes possible with VNS in addition to substantially reducing the high risk to the DEMO associated with the ITER-alone scenario.

It should be obvious that if the ITER-alone scenario is to be compared with the VNS/ITER parallel facilities scenario on the same risk level, one should consider another facility (pre-DEMO) between ITER and DEMO. This scenario results in very large additional capital and operating costs of DEMO, it delays DEMO operation to the year 2054, and it results in only the same confidence level as that achievable with VNS for a DEMO by the year 2025.

VII.D.2. Near-Term Cost

Another point on cost is whether constructing and operating VNS parallel to ITER will impose a substantial financial burden during the years of construction. Such a burden will be substantial if one country builds both ITER and VNS. However, in the context of an international fusion program, VNS will not impose a significant burden if two key points are realized: (a) ITER and VNS will be sited in two different countries instead of in the same country and (b) the host party for a facility will pay 50% or more of the capital cost for this facility, as presently being discussed for ITER.

Table XXXI summarizes the construction and annual operating costs for party X that hosts ITER and for party Y that hosts VNS. The ITER host party will pay \$4.96 billion of which only \$0.33 billion, i.e., <10%, is the additional burden due to VNS. The VNS host Y will pay a total cost of \$2.96 billion, which is substantially lower than that to be paid for hosting ITER. The benefits to both parties X and Y cannot be quantified at present, but they appear comparable. Since VNS will deal with the FNT component development and engineering issues that are most critical to DEMO, the experience gained from hosting VNS is tremendous. Finally, from a programmatic viewpoint, the scenario with parallel ITER and VNS should make it easier to agree on siting by providing more than one opportunity to the parties.

TABLE XXX

Total Cost to Start of DEMO Construction*

	I ITER Only	II ITER BPP + VNS	III ITER + VNS	IV ITER + Delayed VNS	Ib ITER Only, High Fluence
Capital cost					
ITER	8	8	8	8	8
VNS	---	2	2	2	---
Operating cost					
ITER BPP	4.8	4.8	4.8	4.8	4.8
ITER EPP	4.8		4.8	4.8	4.8
VNS		2.4	2.4	2.4	
^{3}T supply cost					
ITER BPP	0.18	0.18	0.18	0.18	0.18
ITER EPP	1.8		1.8	1.8	5.4
VNS		0.43	0.43	0.43	
Total cost	19.6	17.8	24.4	24.4	23.2

*In billions of dollars.

TABLE XXXI

Construction Costs by Party for ITER, VNS, and Other Facilities

	Total Cost	Cost to Party That Hosts ITER	Cost to Party That Hosts VNS	Cost to Party with No Site
Construction cost ($ billion)				
ITER	8	4	1.33	1.33
VNS	2	0.33	1	0.33
IFMIF	0.8	0.13	0.13	0.13
Other	2	0.5	0.5	0.5
Total construction cost ($ billion)	12.8	4.96	2.96	2.29

The final point to be remembered about cost is that if correcting and improving a design through development is considered expensive, correcting it by changing a production run as a result of field experience is even more expensive.

VIII. CONCLUSIONS

VIII.A. FNT Issues and Testing Needs

With regard to fusion issues and testing needs, the following can be stated.

1. Fusion nuclear technology development has important feasibility and attractiveness issues for realizing fusion power. A serious R&D program with a clear strategy and goals for FNT development must now be a high priority for the world's fusion energy development programs.

2. Physics, engineering, and economic constraints as well as industry and utility requirements for fusion demonstration power plants (DEMO) make it possible to define the major parameters and characteristics for a tokamak DEMO. Such a DEMO is now the stated goal of most of the world's fusion R&D program. The DEMO goals for fuel self-sufficiency, safety, environmental impact, and plant availability permit deriving quantitative goals for FNT R&D. The blanket system is found to determine the critical path to the development of FNT components for DEMO.

3. The goal of the blanket development is to simultaneously achieve tritium self-sufficiency, efficient energy conversion and heat extraction, acceptable failure rates, adequate radiation protection, and attractive safety and environmental features, under operation in the complex fusion environment.

4. Adequate performance verification and engineering development require prototypical test articles (e.g., materials, configurations, and size) and testing environment. Multiple interactive effects among the physical elements of the blanket (e.g., breeder/structure/coolant/multiplier/electric insulators/tritium barriers/tritium carrier fluid) and the elements of the fusion environment (e.g., neutrons, bulk heating, surface heating, tritium production, magnetic field, mechanical forces, and vacuum) represent the major testing issues.

VIII.B. Role of Nonfusion Facilities

Nonfusion facilities provide a cost-effective approach to performing single- and multiple-effect tests. Hence, they play an important role in providing basic data, screening of blanket concepts, and establishing the infeasibility of some blanket concepts, prior to performing the more complex and expensive fusion tests. However, the engineering feasibility of blanket components cannot be established prior to extensive testing in the fusion environment. None of the critical issues can be fully resolved by testing in nonfusion facilities alone. Nonneutron test stands, fission reactors, and accelerator-based neutron sources (including the D-Li source) are unable to simulate the multiple effects of the fusion environment, and they cannot provide adequate space to test articles with relevant material combinations, configurations, and dimensions.

VIII.C. Fusion Testing Requirements

The FNT testing in fusion facilities should proceed in three stages: (a) initial fusion break-in, (b) concept performance verification, and (c) component engineering development and reliability growth. Extensive analysis shows that the FNT fusion testing requirements are a 1 to 2 MW/m^2 neutron wall load, steady-state plasma operation, 1- to 2-week periods of continuous operation (i.e., 100% device availability), and >10 m^2 of test area. The testing fluence required is >6 $MW \cdot yr/m^2$ for enabling the demonstration of a blanket system availability in DEMO >50% (0.3, >1, and >4 to 6 $MW \cdot yr/m^2$ for stages 1, 2, and 3, respectively). The component engineering development/demonstration and reliability growth stage is the most demanding on FNT testing.

VIII.D. Blanket Failures and Demo Availability

With regard to blanket failures and DEMO availability, the following can be stated.

1. Availability analysis reveals critical concerns in fusion power development; some of these concerns can be addressed by changes in blanket and machine design, but most must be addressed by extensive testing to realize the DEMO availability goals and to address critical questions concerning the practicality and economics of tokamak power systems. For a DEMO reactor availability goal of 50%, the blanket availability must be ~80%. The mean time to replace (MTTR) or recover from a failure and MTBF are the parameters that directly affect availability. Shorter MTTR lowers the required MTBF to achieve a given availability goal. For MTTR = 3 months, the blanket MTBF must be >1.0 FPY; i.e., only one failure anywhere in the blanket is allowed for about every 1 yr of operation. For a blanket that has 80 modules, the corresponding MTBF per module is 80 FPY. These are very ambitious goals. Experience from nonfusion technologies shows that achieving such long MTBFs requires very extensive testing and development.

2. Some of the important conclusions regarding failure modes, failure rates, and reliability growth testing are

a. The capability of replacing the FW/B in as short a time as possible must be a design goal for fusion devices.

b. Design concept selection and improvement for FW/B must aim at improving reliability (e.g., minimize welds, brazes, joints, and total tube length).

c. A serious reliability/availability analysis must be an integral part of the design process.

d. Research and development programs must be based on quantitative goals for reliability (type and number of tests, test duration, and prototypicality).

e. Reliability growth/demonstration testing in fusion devices will be the most demanding, particularly on the number of tests and the time duration of tests (>10 m^2 and ~6 $MW \cdot yr/m^2$ for blankets).

f. Reliability testing should include identification of failure modes and effects, aggressive iterative design/test/analyze/fix programs aimed at improving reliability, and the obtainment of failure rate data sufficient to predict MTBF.

VIII.E. ITER-Alone Scenario

With regard to the ITER-alone scenario, the following can be stated.

1. As presently envisaged, ITER alone cannot satisfy the FNT fusion testing requirements listed earlier because of pulsed operation with a low duty cycle, low fluence, a short continuous operating time, low device availability, and a small number of blanket testing ports.

2. For the presently envisaged ITER strategy based on EPP with a fluence of 1 $MW \cdot yr/m^2$ and 10 m^2 (to be checked) of test area, blanket tests in ITER alone

enable DEMO blanket concept performance verification but cannot demonstrate a blanket system availability in DEMO higher than 4%.

3. In addition to the high risk to DEMO, an ITER-alone strategy will result in long delays in the commitment to DEMO construction. The development schedule to DEMO becomes problematic.

VIII.F. Scenarios with HVPNS

With regard to scenarios with HVPNS, the following can be stated.

1. A DEMO availability of >30% can be demonstrated by adding blanket tests in a HVPNS characterized by the following parameters: average neutron wall load of 1 to 2 MW/m^2, maximum neutron fluence ≥ 6 $MW \cdot yr/m^2$, testing space at the first wall ≥ 10 m^2, and device availability >25%.

2. Presentations made to the study participants during the phase 1 effort on candidate HVPNS concepts seem to show that an attractive design envelope for HVPNS exists. A small size ($R < 2$ m) tokamak with normal-conducting TFCs and a driven ($Q \sim 2$ to 3) steady-state plasma meets the FNT testing requirements with a capital cost expected to be <25% that of ITER. (The design of HVPNS was outside the scope of phase I. Presentations were made by volunteers from the United States, the European Union, and the Russian Federation. The study participants did not address the specifics of any design.)

3. An effective path to fusion DEMO involves two parallel fusion facilities: (a) ITER, to provide data on plasma performance, plasma support technology, and system integration, and (b) HVPNS, to test, develop, and qualify fusion nuclear components and material combinations and to demonstrate an acceptable MTTR for DEMO.

4. A testing strategy employing such an HVPNS would decisively reduce the high risk of initial DEMO operation with a poor blanket system availability and would make it possible, if operated parallel to ITER BPP, to meet the goal of DEMO operation by the year 2025.

5. With an ITER/HVPNS strategy, blanket tests in ITER BPP are still very important for fusion scoping tests requiring lower fluence, short-term performance tests, and testing large blanket modules up to the size of a segment at low fluence.

6. The contribution of blanket tests in the presently envisaged ITER EPP to the reliability testing is very small compared with that obtainable in HVPNS. If HVPNS is operated parallel to the ITER BPP, several scenarios for better utilization of the ITER EPP can be envisaged and should be studied further. An example is the use of HVPNS testing information to construct a hot DEMO-type breeding blanket on ITER after the end of BPP to operate the second phase (EPP) of ITER in a pre-DEMO mode.

7. The parallel path strategy with ITER at large fusion power, low fluence, and VNS at low fusion power and high fluence reduces the tritium consumption and external supply problem to an acceptable level.

8. A scenario with HVPNS parallel to ITER (BPP) provides cost savings in the overall R&D toward DEMO compared with an ITER-alone strategy. The near-term cost burden is small in the context of an international fusion program with HVPNS and ITER sited in two different countries.

VIII.G. Figures of Merit

In determining an attractive design envelope for VNS, cost/benefit/risk analysis and trade-off studies should be conducted. Suggested figures of merit include the following:

1. extent of meeting FNT requirements (wall load, fluence, test area, etc.)
2. total capital and operating costs
3. contribution to nuclear testing for DEMO components
4. additional contributions to satisfying DEMO database requirements other than testing
5. minimal R&D to construct HVPNS
6. confidence in achieving HVPNS goals
7. contributions to ITER (e.g., reduced technological burden and possible cost savings)
8. contributions to improvements in the development schedule to DEMO.

APPENDIX A

FAILURES AND RELIABILITY TESTING IN FUSION FACILITIES

One of the most serious concerns in the engineering development of a component, particularly for a new technology, is failure. Failure is defined here as the ending of the ability of a design element to meet or continue its function before its allotted lifetime is achieved, i.e., before reaching the operating time for which the element is designed.

Causes of failures include errors in design, manufacturing, assembly, and operation; lack of knowledge and experience; insufficient prior testing; and random occurrence despite available knowledge and experience.

Experience from other technologies shows[42,43] that the failure rate λ during the lifetime of a component for a fully developed technology generally looks like a "bathtub" curve, as shown schematically in Fig. A.1. High failure rates are experienced during early life, which decrease with time until a steady-state value λ_b at the "bottom of the bathtub" is reached. This steady-state value λ_b remains generally constant with time until near the end of the component life when the failure rate increases with time during the "wear-out" period. The value of λ_b may actually decrease or increase moderately during operation. A key question for FNT development is the value of λ_b for the blanket: What is the goal value for λ_b and how can it be achieved through testing? Experience shows that the value of λ_b for the new technology is high and decreases with testing during the R&D phase, as illustrated in Fig. A.1. Such a reduction in failure rate λ, or equivalently an increase in MTBF (MTBF = $1/\lambda$) is achieved through a reliability growth program that involves a test-analyze-fix strategy.

The term "reliability" here implies that a component satisfies a set of performance criteria while under specified conditions of use over a specified period of time. The objective of this section is to quantify the reliability goals for the DEMO blanket and to derive quantitative requirements of reliability growth/demonstration testing in fusion facilities prior to constructing the DEMO blanket. Such a testing program proceeds from measurements of unexpected performance, investigation of failure modes and consequences, and identification of the optimum product and ends with a demonstration of satisfactory performance.[35,44,45] While the component lifetime is mainly determined by the fluence limitation (i.e., damage level) that leads to performance degradation, MTBF represents an arithmetic average life of all units in a population. As we will shortly see, the MTBF requirements are much more demanding on the blanket test program than the design lifetime.

Our approach here to evaluating the requirements of the reliability growth/demonstration program for fusion blankets is as follows:

1. determine the DEMO reactor availability goal
2. determine a corresponding goal for the availability of the blanket system and for the blanket modules
3. determine a target MTBF for blanket modules
4. quantify both the test times and the number of test articles that would be required to confirm that the specified target MTBF is met.

A.I. GOAL MTBF (AND MTTR) FOR DEMO BLANKET

The availability allocation among components of a fusion reactor system for the achievement of target availability has been performed in INTOR (Ref. 46), STARFIRE (Ref. 37), and Next European Torus[47] (NET). These studies have shown that blanket system

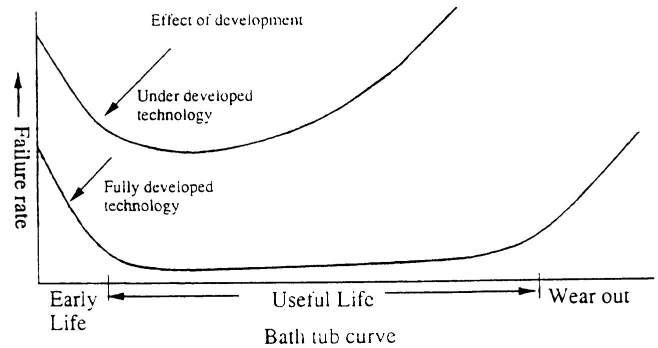

Fig. A.1. Failure rate characteristics for developed and underdeveloped technologies.

availabilities of ~97 and ~99% are required to meet reactor availability goals of 60% (Ref. 47) and 77% (Ref. 46), respectively. Notice that these values are estimated based largely on expert opinion and the data obtained from the experience of nonfusion technology. Although this study does not attempt to assess availability apportionment for achieving a goal availability, the required minimum availabilities of the blanket system and other FNT components are projected to help identify the testing needs.

An availability assessment requires a complete description of the plant, including possible failure modes and consequences. This description is then used to construct a model for the plant operation, which is in turn used to calculate the plant availability for given operating options and for given data on failure rates, repair times, and scheduled maintenance. While all these elements are yet less defined for DEMO, the reactor availability is approximated as

$$\text{DEMO } \textit{reactor availability} = A_R = \frac{1}{1 + \sum_i \textit{outage risk}_i} ,$$

where the outage risk is defined as the failure rate times the mean time to repair and i represents a component. To derive the blanket system and other FNT component availability requirements for achieving a target DEMO availability, we further assumed that the reactor availability goal is determined by the six major components (i.e., B/FW system, divertor/limiter system, heating and current drive system, magnets, vacuum pumping and vacuum vessel, and fueling and fuel cycle system), which at their fully developed stage for DEMO would carry the same amount of outage risk in the reactor unavailability. Such a component apportionment has led to a required FW/B system availability of 90% (or FW/B system outage risk of 0.111) for achieving a DEMO reactor availability of 60%. The impact of the lack of FNT component development is then evaluated by the higher outage risks and consequently by the lower reactor availabilities, as shown in Table A.I. In this case, a blanket system availability of 49% is projected to achieve a DEMO reactor availability of 30%.

The blanket system availability goals, such as those shown in Table A.I, can be used to establish blanket module availability requirements. A blanket system is viewed as a system consisting of a number of blanket modules in series for which the failure of any blanket module causes failure of the blanket system and the failure of any blanket module is entirely independent of the failure of any other blanket module.

The availability of the blanket system is defined as the probability that the blanket system at time t will be available. Thus, the blanket system availability at any time t can be written as

$$A_{BS} = \frac{uptime}{uptime + downtime} = \frac{\mathrm{MTBF}_{BS}}{\mathrm{MTBF}_{BS} + \mathrm{MTTR}_{BS}} = \frac{1}{1 + \lambda_{BS}\mathrm{MTTR}_{BS}} \enspace ,$$

where MTBF_{BS} is the MTBF for the blanket system, which is equal to the reciprocal of failure rate λ, and MTTR_{BS} is the mean time to replace, i.e., the downtime of the reactor to replace, or fix, the failed portion of the blanket system. To determine the blanket module availability A_n, we will assume that the MTTR_{BS} for the blanket system is equal to the MTTR_n for the blanket module. This means that blanket replacement operations for the blanket can be performed in parallel rather than in series in case of simultaneous failures in any number of modules. Thus, the relationship between A_{BS} and A_n can be written as

$$A_n = \frac{n}{(n-1) + \dfrac{1}{A_{BS}}} \enspace ,$$

TABLE A.I

Requirements on Blanket System Availability as a Function of Reactor Availability

DEMO Reactor Availability (%)	FW/B System Availability (%)
60	90
51.4	80
50	78
43.5	70
36	60
30	49
23	30
10	20
5	10

where A_n is the blanket module availability defined as

$$A_n = \frac{1}{1 + \lambda_n \mathrm{MTTR}_n} \enspace .$$

The target MTBF per module is shown in Table A.II for MTTR values of 1 and 2 weeks and 1 and 2 months. Note that current estimates of MTTR are 3 months or greater. Such a long MTTR leads to longer MTBF. We assume that future improvements in configuration and maintenance will lead to shorter MTTR. For the 30% DEMO reactor availability, the MTBF (module) varies from 1.5 to 12.65 FPY for MTTR = 1 week to MTTR = 2 months. For a 60% DEMO reactor availability, the MTBF (module) is ~14 FPY for MTTR = 1 week and becomes ~119 FPY for MTTR = 2 months.

The results in Table A.II have serious implications, particularly for the DEMO reactor availability of 60%, which is commonly assumed worldwide. For this A (reactor) = 60%, the required MTBF per blanket module is much longer than the design life of the blanket (10 to 20 MW·yr/m^2, which is ~3 to 7 FPY at $P_{nw} \sim 3$ M/m^2). For MTTR = 1 week and 80 modules, the goal MTBF for the blanket module needs to be ~14 FPY, i.e., more than two times longer than the design lifetime. For a more likely case in which MTTR equals 1 month, MTBF (module) is 60 FPY and MTBF (blanket system) ~ 0.8 FPY. This means only one failure in the entire blanket system is allowed per calendar year. This is an extremely ambitious goal compared with the state of the art discussed in Sec. A.II. As also shown later, the testing requirements for demonstrating such long MTBF appear to be extremely demanding. This is why we are considering here a different scenario for the DEMO, as discussed in Sec. II, which assumes the DEMO will have two stages. The first has an initial target availability of 30%, and it reaches 60% only in the second stage.

One additional observation on the results of Table A.II is that the mean time to replace a failed blanket module (MTTR) has tremendous influence on the target blanket MTBF for a given availability. We considered the range of MTTR = 1 week to 2 months.

TABLE A.II

Required DEMO Blanket Module MTBF_n as a Function of Mean Time to Repair (MTTR) for Two Values of DEMO Reactor Availability*

MTTR	MTBF_n (FPY)	
	A (reactor) = 60%	A (reactor) = 30%
1 week	13.9	1.5
2 weeks	27.4	2.9
1 month	60	6.5
2 months	119	12.65

*Number of modules n = 80.

Analysis shows that it is difficult to reduce the MTTR to 1 week. The operations required to replace a failed blanket module are many and complex (de-energizing the magnets, filling the vacuum vessel with inert gas, breaking seals in the vacuum vessel, disconnect, removal, insertion, reconnect, etc.). In addition, when a module fails, one needs to identify the failure consequences (e.g., the distortion of the module geometry) on the maintenance operation. There are also many safety-related precautions and operations. Therefore, 1 week appears a low value for MTTR. However, values of 1 to 2 months have a very serious impact on the required MTBF and achievable availability. The results here and in Sec. VII.C suggest that achieving short MTTR is crucial to the ultimate economic viability of the tokamak system. A key conclusion here is that all aspects related to MTTR must be addressed in machine design and in fusion testing. Data on achievable MTTR need to be obtained from fusion test facilities.

A.II. ESTIMATES OF FAILURE RATES

Given the target MTBF values for the blanket DEMO in Sec. A.I, we ask the following key question: What do we expect the failure rate to be based on current knowledge? Unfortunately, our current database from fusion systems is nonexistent since no blanket was ever tested or operated. An indication of expected failure rates can be obtained from using data in other technologies. Data from steam generators and fission reactors appear relevant and have recently been used by Bünde et al.[48] to assess failure rates in fusion systems. We considered in this study a range of blanket options for the DEMO, particularly those with high-pressure coolant. We assumed that the size of DEMO is similar to that of ITER-EDA (Ref. 33), with a first-wall surface area of $\sim 1200\ m^2$. We assumed 80 blanket modules. The number of modules affects only the failure rate per module but does not have a major influence on the total failure rate for the blanket system.

Table A.III shows the estimated failure rates using data compiled by Bünde et al.[48] from steam generators and fission reactors. Mean and high values for unit failure rate units (i.e., per unit length of weld or pipe) are given in Table A.III. The estimated length and number of elements per blanket module are also given in Table A.III. The overall failure rate per blanket module is estimated to be in the range of 7×10^{-6} to 1×10^{-4}/h. Thus, the MTBF (module) is in the range of 1 to 16 yr, and the MTBF for the overall blanket systems is 0.01 to 0.2 yr; i.e., there will be ~5 to 80 failures somewhere in the blanket per year.

It is instructive to compare MTBF estimates based on what has been achieved to date in mature nonfusion technologies to those that must be achieved in fusion DEMO. Table A.IV presents a comparison of what is expected versus what is required for the blanket MTBF. The MTBF values are shown for the blanket module and the blanket system, which consists of 80 modules. The expected MTBF is based on results in Table A.III, i.e., based on those failure modes and failure rates that we know from the mature technologies of steam generators and fission reactors are likely to exist in fusion DEMO blankets. The expected MTBF values in Table A.IV do not account for the additional failure modes for the fusion specific system, as will be discussed later. The required values of MTBF in Table A.IV are those that must be achieved to meet certain availability goals for the blanket. We show the required MTBF in Table A.IV for two cases of DEMO reactor availability: 30 and 60%. For each case, MTBF values are given for

TABLE A.III

Estimated Failure Rate for Typical Blanket Based on Data from Nonfusion Technologies*

Blanket Element	Number or Length of Elements per Blanket Module	Unit Failure Rate[a]		Failure Rate per Blanket Module (1/h)	
		Mean	High	Mean	High
Longitudinal welds	66 m	5.0E−8/h·m[b]	5.0E−7/h·m	3.3125E−6	3.3125E−5
Butt welds of pipe	462	5E−9/h·weld	1E−7/h·weld	2.31E−6	4.62E−5
Pipes (straight)	2.75 km	5E−10/h·m	1E−8/h·m	1.375E−6	2.75E−5
Pipe bend	28	1E−8/h·bend	3.5E−7/h·bend	2.8E−7	9.8E−6
Overall failure rate per module (1/h)		7×10^{-6} to 1×10^{-4}			
Calculated MTBF per module (yr)		1 to 16			
Calculated MTBF for blanket system (yr)		0.01 to 0.2			

*Failure rates given here do not include fusion-specific failure modes.
[a]Failure rates are based on experience from nonfusion technologies.[48]
[b]Read as 5×10^{-8}.

TABLE A.IV

Comparison of Expected Blanket MTBF to That Required in DEMO

	MTBF (yr)	
	Blanket Module	Blanket System
Expected		
Expected (for fully developed technology based on steam generator and fission reactor data)[a]	1 to 16	0.01 to 0.2
Required		
DEMO availability = 30%		
MTTR = 1 week	1.5	0.02
1 month	6.5	0.08
2 months	12.65	0.16
DEMO availability = 60%		
MTTR = 1 week	13.9	0.17
1 month	60	0.75
2 months	119	1.49

[a]Estimates here do not account for additional failure modes specific to the fusion environment.

different values of the MTTR, i.e., the downtime to recover from a blanket failure.

The results in Table A.IV are striking and have very serious consequences for many aspects of fusion R&D. Required MTBF values for the DEMO blanket module are in the range of 1.49 to 12.65 yr for MTTR in the range of 1 week to 2 months for the case of a DEMO reactor availability of 30%. These are within the range of expected values, which is 1 to 16 yr. For the DEMO reactor availability goal of 60%, the MTBF per blanket module with the shortest time estimated for an MTTR of 1 week falls in the range of expected values. However, the MTBF per blanket module increases to 60 and 119 yr at MTTR = 1 and 2 months, respectively. These values are much greater than the 1 to 16 yr range of expected values. In other words, assuming a realistic time for MTTR of 1 month, the MTBF value required to achieve a DEMO reactor availability of 60% is much longer than those expected to be achievable. This suggests that a blanket with a low enough failure rate to achieve a DEMO reactor availability goal of 60% appears to be an ambitious goal.

Note that the expected values derived here are based on data from steam generators and fission reactors. The primary failure rate in steam generators appears to come from failures in welds. Since steam generators represent mature technologies with tens of thousands of components in operation, the failure rate per unit length of weld in fusion systems cannot be expected to be any lower. Consequently, the only prudent method to reduce the failure rate in fusion blankets is to reduce the number and length of welds. This should be a key factor in the design of blankets and in the selection among blanket concepts.

Another serious concern is that the failure rates in Table A.III account for only a very limited number of known failure modes. Very little work has been done to date to identify failure modes in FW/B systems. Table A.V lists some of the possible failure modes that should be of concern. For example, in self-cooled liquid-metal blankets, cracks or other imperfections may prove to be a failure mode that occurs at high frequency, and the large flow channel area in the tokamak geometry will magnify the problem. On the other hand, self-healing insulator coatings may function perfectly with a very low failure rate. The problem is that we do not know. There has been little FNT R&D. Fusion testing can provide the answer to such critical questions.

It is reasonable to ask whether the failure rate in fusion blanket systems can be expected to be lower or higher than in steam generators and fission reactors. A quantitative answer is beyond the scope of this research but should be seriously addressed in the future, most importantly by generating a database from actual tests of blankets in the fusion environment. Our concern is that failure rates may be much higher in fusion blankets because they appear to be much more complex than in steam generators and the core of fission reactors because of the following points:

1. larger numbers of subcomponents and interactions (tubes, welds, breeder, multiplier, coolant, structure, insulators, tritium recovery, etc.)
2. more damaging, higher energy neutrons
3. other environmental conditions: magnetic field, vacuum, tritium, etc. (for example, a leak from the first wall or blanket module walls into the

TABLE A.V

Examples of Possible Failure Modes in B/FW for Solid and Liquid Breeder Blanket Concepts

Cracking around a discontinuity/weld
Crack on shutdown (with cooling)
Solid breeder loses functional capability due to extensive cracking
Cracks in electrical insulators (for liquid-metal blankets)
Cracks, thermal shock, vaporization, and melting during disruptions
First-wall/breeder structure swelling and creep leading to excessive deformation or first-wall/coolant tube failure
Environmentally assisted cracking
Excessive tritium permeation to worker or public areas
Cracks in electrical connections between modules

vacuum system results in failure, while in steam generators and fission reactors, continued operation with leaks is often possible)

4. Reactor components must penetrate each other; many penetrations must be provided through the blanket for plasma heating, fueling, exhaust, etc.

5. the ability to have redundancy inside the B/FW system is practically impossible.

Some important concluding remarks regarding this topic of failure modes, failure rates, and reliability growth testing are as follows:

1. The capability to replace first wall and blanket (individual modules as well as the entire FW/B system) in a reasonable time *must* be a design goal for fusion devices.

2. Design concepts for FW/B (and other components) must aim at improving reliability. One of the most effective directions is to minimize features that are known to have a high failure rate (e.g., minimize or eliminate welds, brazes, and tube length).

3. A serious reliability and availability analysis must be an integral part of the design process.

4. The R&D program must be based on quantitative goals for reliability (type of tests, prototypicality of test, number of tests, and test duration).

5. Reliability growth testing in fusion devices will be the most demanding (particularly on the number of tests and time duration of tests). Reliability testing should include identification of failure modes and effects, aggressive iterative design/test/fix programs aimed at improving reliability, and obtainment of failure rate data sufficient to predict MTBF

A.III. RELIABILITY TESTS AND CONFIDENCE LEVEL

The term "reliability" is defined as the ability of an item to perform for a stated period of time. The principal purpose of reliability tests is to determine whether the product meets a specific reliability criterion. Reliability tests can be either sequential or fixed length. With the sequential approach, test termination is generally after either the test has exhibited few enough failures at some point in time during the test for a pass decision or enough failures have occurred to make a fail decision. Based on a study of sequential test plans,[49] the INTOR critical issues study concluded that the achievement of 80% confidence in a given MTBF in the constant failure rate regime of operation would typically require a cumulative test period of 3.5 times the MTBF (Ref. 46). The other alternative is either fixed-length or fixed-failure tests. Because most test situations have schedule/time constraints, time-terminated tests are the preferred choice for fixed-length test strategies. A fixed-length test plan is particularly appropriate when the total test time must be known in advance. Such a test design assuming a constant failure rate can lead to the selection of the Poisson distribution for the test analyses. (The nonhomogeneous Poisson process with Weibull intensity can be used when the failure rate is considered to change as a function of system usage.[50]) As shown in Fig. 7 (see Sec. VII), the Poisson distribution is used to relate the upper statistical confidence level as a function of test time in multiples of MTBF and the number of failures experienced during the tests.[51] By utilizing this test plan, the primary objectives here are to determine the blanket test time and the test area in fusion facilities, which are required to meet certain goals for MTBF.

The total test time T in multiples of MTBF given in the horizontal axis is calculated as

$$T = \frac{Nt \times N^{1-\alpha}}{\Phi_{NW}} ,$$

where

N = number of test modules

t = test fluence per test module

α = experience factor

Φ_{NW} = DEMO neutron wall load (3 MW/m^2).

The experience factor is meant to reduce the total test time of Nt by a factor of $N^{1-\alpha}$ taking account of the fact that similar failure causes may be seen in different blanket modules.[53] Based on this test plan, we were able to calculate the confirmatory DEMO reactor availability at 80% confidence as a function of fluence on test module and number of modules tested. For all cases, we used that data in Tables A.I and A.II that correlate reactor availability, blanket availability, MTBF, and MTTR. In all cases, we assumed the number of blanket modules in DEMO to be 80.

Figure A.2 shows the DEMO reactor availability achievable with 80% confidence and, if one assumes one failure during the test, as a function of fluence on test modules. Results are shown for two cases of 6 and 12 test modules and for two cases of MTTR = 1 week and 1 month. Several important observations can be made from the results. The MTTR is again clearly a critical parameter. If MTTR = 1 month or longer, the DEMO reactor availability will be <40% even for a fluence of 10 $MW \cdot yr/m^2$. Increasing the number of modules provides an opportunity to possibly observe different failure modes and to improve statistics. However, the same failure may occur in more than one module. Therefore, the increase in experience from testing with the number of test modules is less than linear. For all calculations, an experience factor of 0.8 was assumed.

The fluence requirement on the test modules is critical. Figure A.2 indicates that clearly the achievable DEMO blanket availability, and hence the DEMO reactor availability, increases substantially with testing

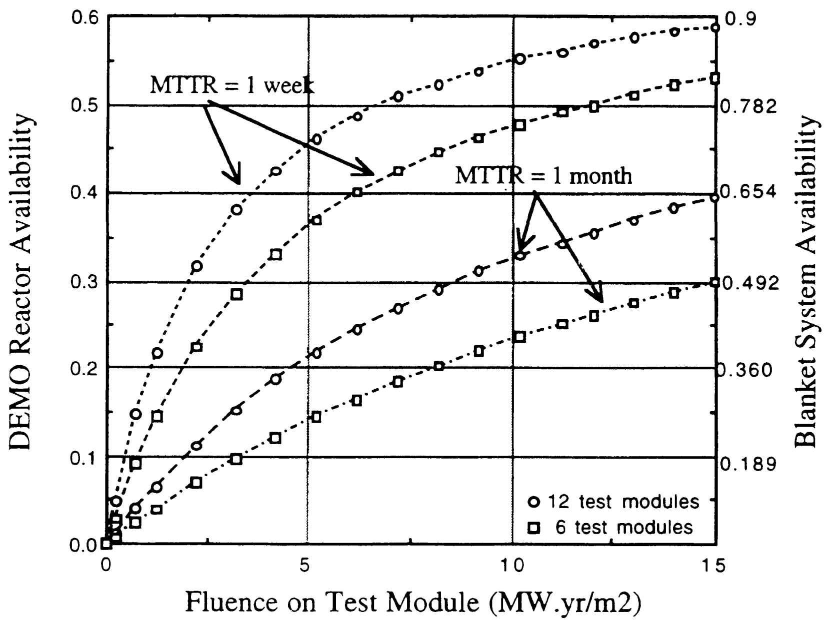

Fig. A.2. The DEMO reactor and blanket system availabilities obtainable at 80% confidence as a function of testing fluence on the blanket test module for MTTR = 1 week and 1 month.

fluence. For MTTR = 1 week, increasing the testing fluence from 1 to 6 $MW \cdot yr/m^2$ increases the DEMO availability from 19 to 48% with 12 test modules and from 12 to 39% for 6 test modules. For MTTR = 1 month, a testing fluence of 1 $MW \cdot yr/m^2$ leads to reactor availability of only 5.6% with 12 test modules, but increasing the testing fluence to 6 $MW \cdot yr/m^2$ increases the DEMO reactor availability to 24%.

Notice that as the test fluence increases beyond ~6 $MW \cdot yr/m^2$, the rate of increase in reactor availability per unit of additional testing fluence decreases. The rate of improvement in reactor availability becomes even smaller at higher fluences, >10 $MW \cdot yr/m^2$. Since the blanket design lifetime may be limited to ~10 $MW \cdot yr/m^2$, testing will become difficult at such high fluence.

A number of key conclusions are important from the results here. First, achieving a fluence of ~6 $MW \cdot yr/m^2$ at the test modules with ~6 to 12 test modules is crucial to achieving a DEMO reactor availability in the 40 to 50% range with 80% confidence. Second, achieving a DEMO reactor availability of 60% may not be possible with 80% confidence for any practical blanket test program. Third, the length of downtime to recover from random failures must be by itself one of the critical objectives for testing in fusion facilities.

APPENDIX B

TOKAMAK VNS DESIGN ENVELOPE

In this appendix, we summarize the performance guidelines established in Sec. VI, identify the physics and engineering assumptions for the VNS consistent with the present tokamak database, and determine the range of VNS device parameters. A number of design options are considered, based on the use of superconducting or normal-conducting TFCs. An updated version of the SuperCode[52] will be utilized for this purpose.

The basic variations of tokamak VNS designs include

1. superconducting TFCs and adequate inboard radiation shield to protect the superconducting magnets
2. multiturn normal-conducting TFCs and adequate inboard radiation shield to limit damage to TFC insulators and normal conductor requiring standard aspect ratios ($R_0/a \geq 2.5$)
3. single-turn normal-conducting TFCs and essentially no inboard nuclear shielding, permitting $R_0/a \leq 2$.

These design options were considered recently[53–55] for application in fusion development. The present study utilizes common assumptions to define the envelope for VNS and to produce information useful in comparing the merits of these options in future studies.

B.I. PERFORMANCE GUIDELINES

Performance guidelines that determine the parameters for a tokamak VNS are summarized as follows:

1. fusion power < 150 MW (in some cases, fusion power up to 400 MW was considered to assess the impact on design)
2. neutron wall load = 1.0 to 2.0 MW/m^2
3. device fusion neutron fluence at the first wall ≥ 6 MW·yr/m^2
4. maximum site power requirement ≤ 700 MW
5. steady state (or long plasma burn with duty cycle ≥ 80%
6. device load factor (duty cycle·availability) ≥ 25%
7. surface area (at the first wall) for test module ≥ 10 m^2
8. FW/B/shield thickness
   a. inboard distance from first wall to inboard leg of TFCs: Δ_I = 83, 44, or 4 cm for the three design options
   b. outboard thickness: $\Delta_0 \geq 1.0$ m
9. no breeding blanket except the test modules.

Among these guidelines, those that strongly affect the design parameters are the maximum fusion power (for option 1), the required neutron wall load (for all options), the maximum site power consumption (for normal-conducting options), the minimum surface area for test module (for option 3), and the inboard material thickness between the plasma and the TFC (all options). Plasma duty cycle, burn duration, and availability determine the rate of tritium consumption as well as the usefulness to technology testing. The outboard shield thickness affects the torus and the magnet sizes.

B.II. PHYSICS ASSUMPTIONS

The plasma physics assumptions and operating conditions for the tokamak VNS are summarized in Table B.I and compared with available information[56] on ITER. A lower bound of 0.6 m in the plasma minor radius is imposed to ensure that the plasma temperature in the core can exceed a minimum of ~10 keV, as obtained in DIII-D (Ref. 57). The subscript "95" refers to the surface containing 95% poloidal flux in the plasma. Somewhat higher elongations (κ_{95} = 2.1), safety factors ($q_{95} \geq 3.5$ to 4.5), and plasma Troyon beta factor ($g_T \leq 3.5$, with β defined relative to the average magnetic field in the plasma[58]) are assumed[59-65] for the VNS than those assumed for ITER. The confinement improvement factor H_f (relative to the ITER-89P scaling[61]) assumed for VNS is ≤2.5, as indicated in present experiments for H-mode plasmas.[62]

Steady-state noninductive current drive will be unavoidable for the tokamak VNS to achieve a plasma duration of 1000 s or more. Owing to the finite space for the solenoid in modest-size VNS tokamaks, the plasma duration maintainable by induction alone is limited. The magnitude of the noninductive current drive efficiency required for VNS is assumed to be similar to those already achieved to date using neutral beam and/or radio-frequency (rf) injection ($I_{CD}/P_{CD} \leq 0.3 \cdot 10^{20} \cdot$ A/W·m^2) (Refs. 63 and 64). Minimum-size tokamaks are obtained when the solenoid is eliminated. This will lead to the requirement for noninductive initiation and

TABLE B.I

Plasma Assumptions and Operating Conditions for Tokamak VNS

	Tokamak VNS	ITER[a]
Plasma minor radius, a (m)	≥0.6	2.8
Plasma elongation, κ_{95}	≤1.6 to 2.1	≤1.65
Plasma triangularity, δ_{95}	≤0.2 to 0.3	0.3
Plasma edge safety factor, q_{95}	≥3.5 to 4.5	2.9
Normalized plasma beta, g_T (%·m·T/MA)	≤3.5	2.0
Confinement H_f relative to ITER-89P scaling	≤2.5	2.5
Current drive coefficient (10^{20}·A/W·m^2)	≤0.3	N/A
TF ripple at outboard plasma edge (±%)	≤0.5	≤1.0
Core ash buildup factor, τ_a/τ_{burnup}	≤0.1	~0.2
Plasma effective charge, Z_{eff}	≤2.0	≤2.0
Average plasma surface heat exhaust flux (MW/m^2)	≤1.1	0.55
Maximum fraction of plasma radiation loss	≤0.5	≤0.3
Divertor heat flux factor, f_{div} (MW·T$^{0.5}$/m$^{1.5}$)	≤20	20
× point to divertor distance in elevation plane (m)	1.0	2.0

[a]For nominal operation of the BPP of ITER (Ref. 57).

rampup of the plasma current. There is a significant database for such operations,[65,66] which suggests a viable and potentially low-cost option for consideration in future VNS studies.

The TF ripple at the outboard plasma edge will affect the confinement of suprathermal ions in that region. Ripples more than ±0.5% have been estimated to lead to significant losses of these ions and possible damage to the first-wall components.[67] A large fraction of such ions are expected in VNS because of the steady-state drive powers. A somewhat more stringent limit than the ITER assumption is therefore required.

The ratio of the confinement time for fusion ash (thermalized fusion alpha particle) to the D-T fuel burnup time (ash production time), τ_a/τ_{burnup}, controls the ash concentration in the plasma core.[68] High ash concentration increases the size and cost of a fusion device to maintain constant performance. For VNS, τ_a, which is similar to the core particle confinement time, is expected to be smaller than that for ITER, leading to a lower ash concentration. An ash concentration of ~20% is presently estimated for ITER ($\tau_a \geq 15\tau_E \sim 50$ s, $\tau_{burnup} \sim 150$ s) for a helium recycling coefficient of 0.95 to 0.98 (Ref. 68), where τ_E is the plasma core energy confinement time. An ash concentration of 3 to 10% is therefore expected for VNS ($\tau_a \sim 15\tau_E \sim$ 6 to 20 s, $\tau_{burn} \sim 150$ s). A lower ash concentration will tolerate a higher impurity content and possibly a higher impurity radiation loss.

The plasma heat flux averaged over the plasma surface is tentatively assumed to be limited to within a factor of 2 of those anticipated for ITER. For plasmas with successful divertors, the loading on the first wall is expected to be primarily due to radiation (bremsstrahlung, impurity line, and synchrotron), which is assumed to be up to ~50% of the total plasma heating power for VNS. This defines the maximum average heat loading on the first wall. Under clean plasma conditions, however, this radiation loss can be limited to 10 to 15%, allowing up to 90% of the total plasma heating power to enter the divertors.

The heat flux loading in the divertor channel (and hence the chamber wall or the divertor plate, depending on the divertor concept) should not far exceed that anticipated for ITER. Given similar levels of plasma purity, temperature, and density at the plasma edge, and assuming similar divertor geometric configurations, the divertor heat flux F_{div} scales roughly as the divertor heat flux factor f_{div}:

$$F_{div} \propto f_{div} \equiv P_{heat} B^{\gamma}/R_0^{1.5} q_{95}^{0.5} \quad ,$$

where P_{heat} is the total kinetic power entering the divertor chamber and γ depends on the cross-field diffusion in the plasma scrape-off layer (SOL). For purpose of comparison, the total plasma heating power is assumed to enter the divertor(s). The magnitude of F_{div} can be limited if anomalous cross-field diffusion is assumed in the SOL (Ref. 69), making $\gamma = 0.5$. However, if the cross-field diffusion remains constant among all tokamaks (such as the 2 to 3 m^2/s assumed in ITER), making $\gamma = 0$, the divertor heat flux factor f_{div} would be higher for the smaller lower field tokamaks. The physical distance between the × point and the divertor plate is assumed to be 1 m, about half that available in the present ITER design.

Plasma power and particle handling, noninductive current drive, and dependence on plasma aspect ratio are areas of high leverage in determining the size and cost of VNS in the present physics database. The attractiveness of the tokamak VNS will depend on the outcome of the ongoing physics tests in these areas, as well as the design configuration and engineering features that can satisfy the performance guidelines while minimizing cost, engineering, and technology risks.

B.III. DESIGN CONFIGURATION AND ENGINEERING FEATURES

Tokamak VNS design configuration and engineering features are driven by the nuclear testing requirements discussed in Sec. VI. These features are summarized in Table B.II and depicted in Figs. B.1 and B.2 for the VNS options with normal-conducting TFCs. Ensuring the capability to achieve high fluence (≥6 MW·yr/m^2) and high load factor (≥25%) will require ready access for repair or replacement of the critical components in the VNS toroidal chamber. These components include divertor plates, first-wall protection tiles, and nuclear test modules.

Features common to all neutron-producing tokamaks include inboard shielding to protect magnets with electrical insulation, outboard shielding to minimize reactor hall activation and to ensure personnel safety and access, accessible and removable blanket test modules at the outboard midplane, and removable divertor cassettes between the TFCs (Ref. 71).

As shown in Figs. B.1 and B.2, jointed demountable TFCs are used to ensure the ability to disassemble and replace all key components of the VNS tokamak with normal-conducting TFCs (Refs. 55 and 72). This also permits the placement of superconducting poloidal field coils (PFCs) internal to the TFC enclosure. To maximize the wall area available for nuclear test modules in all options, one must place the outboard PFCs at the maximum possible distance from the midplane permitted by the removable divertor modules. The average current densities for the inner and outer legs of the TFCs are limited to 3.0 kA/cm^2 and 1.0 kA/cm^2, respectively, subject to temperature rises up to 150°C in separate pressurized coolant channels.

To ensure adequate rigidity of the multiturn joints in the TFCs (Fig. B.1), we configured the inboard shielding to carry a fraction of the out-of-plane loads, a feature similar to that used in the Small Fusion Development Plan concept.[55] In the case of a single-turn TF magnet

TABLE B.II

Key Design Configuration and Engineering Features for Tokamak VNS

Configuration/Features	Superconducting	Multiterm Normal Conducting	Single-Term Normal Conducting
Total inboard shield material thickness (cm)	72	23	3
Total outboard shield/blanket thickness (cm)	100	100	100
Number of outboard TFC legs	12	8	8
Number of removable divertor modules	12	8	8
Elevation of outboard PFCs	× point	× point	× point
Jointed demountable TFCs	No	Yes	Yes
Average TF inner winding current density (kA/cm^2)	3.7	3.0	1.9 to 2.1[a]
Average TF outer winding current density (kA/cm^2)	3.7	1.0	1.0
TFC load path through radiation shield	No	Yes	Yes
PFC location compared with TFC bore	External	Internal	Internal

[a]Averaged over the entire center leg, which is hourglass shaped at the midsection (Fig. B.2).

(Fig. B.2), the outboard shield is configured to carry essential loads.

The configuration for VNS using superconducting TFCs is expected to be roughly similar to that of ITER, although much smaller in size and plasma current. Key differences in contrast with the normal-conducting cases include a significantly thicker inboard shield (72 cm thick plus spacing), 12 TFCs, separate load

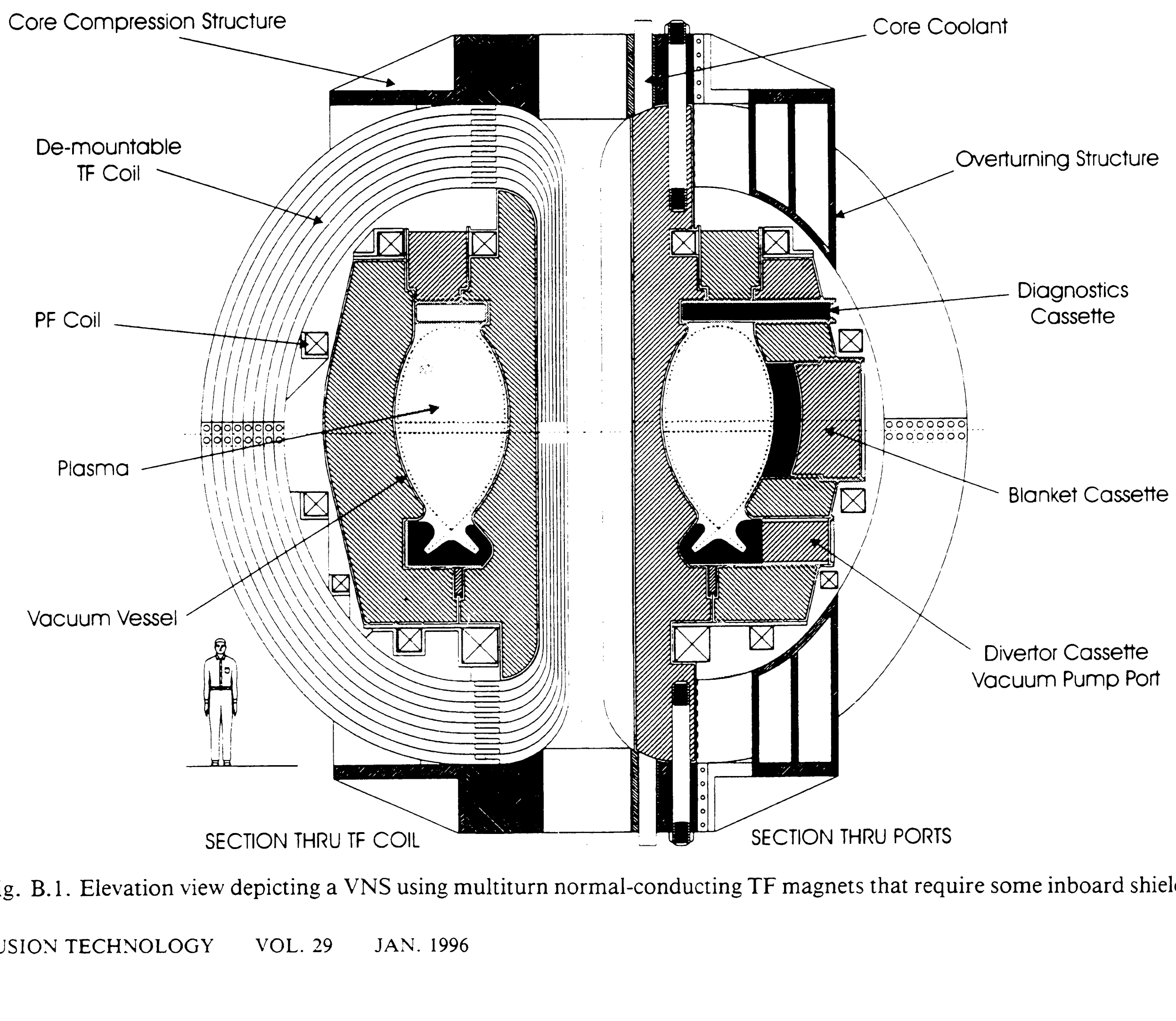

Fig. B.1. Elevation view depicting a VNS using multiturn normal-conducting TF magnets that require some inboard shielding.

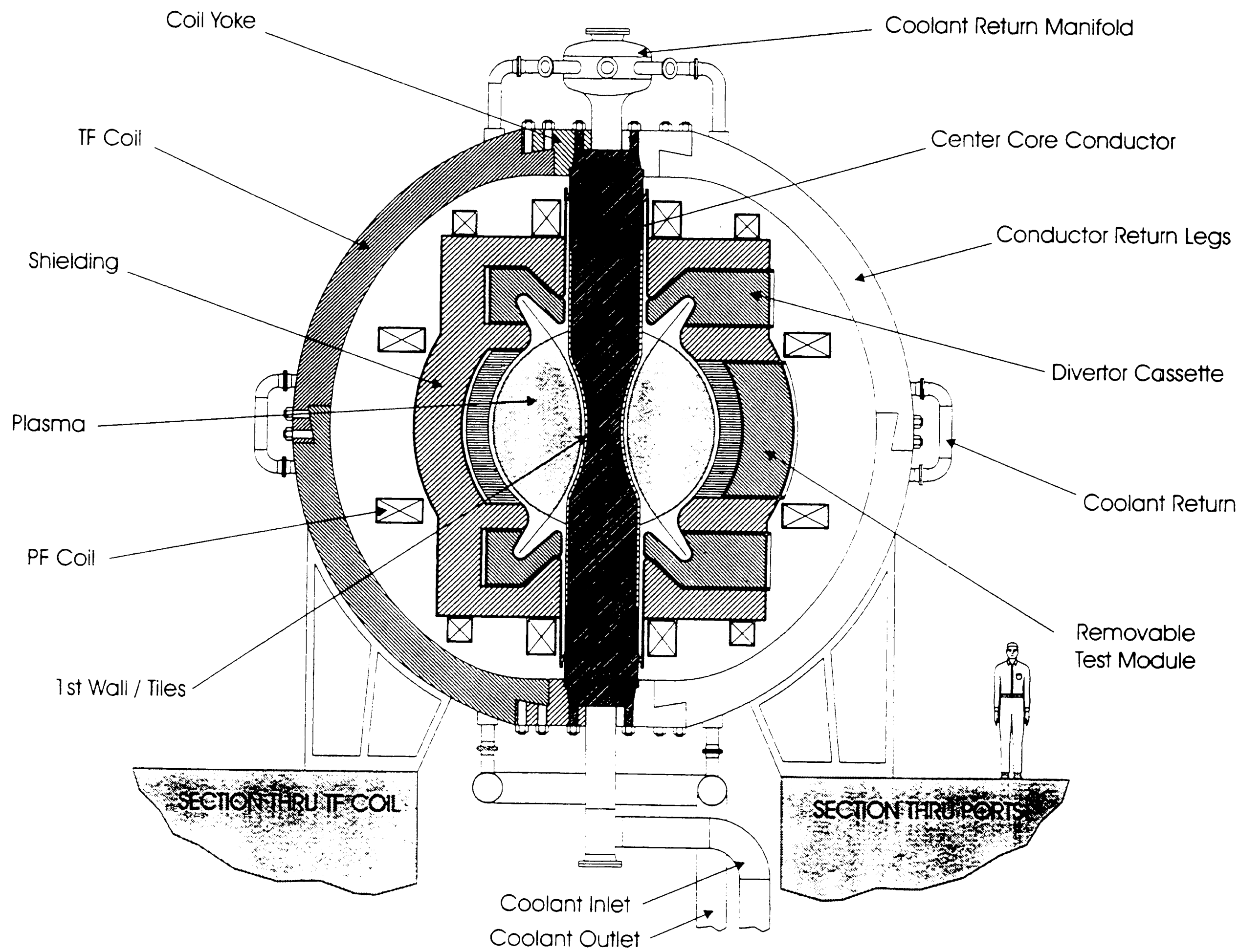

Fig. B.2. Elevation view depicting a VNS using single-turn normal-conducting TF magnets that require no inboard shielding.

paths for the TFCs and the shield structure, and PFCs external to the TFCs. The current densities in the superconducting windings are assumed to be close to those assumed in the ITER design concept.[57]

B.IV. VNS DESIGN ENVELOPE

The preceding assumptions of the VNS plasma and device concepts define the constraints within which desirable design parameters can be estimated. We use the SuperCode,[53] modified to account also for the regime of low plasma aspect ratios ($R_0/a \leq 2$) (Ref. 56) and the use of normal-conducting TFCs. The physics and engineering models in the code are up to date and consistent with the ITER design assumptions.[57] The code permits the determination of design parameters that produce the optimum value for a figure of merit, such as device size or scaled cost, subject to the constraints discussed earlier.

The key results for typical VNS designs providing a neutron wall loading of 1 to 2 MW/m^2 are summarized in Table B.III together with the ITER parameters for comparison. The radial build calculated for these designs for the lower wall loading are given in Fig. B.3, consistent with Figs. B.1 and B.2 in the gaps and thicknesses of the indicated elements. The superconducting VNS option mimics the buildup configuration concept of ITER.

Relative to ITER, the VNS with superconducting TFCs has typically about one-half the device linear size (17 m overall) and one-quarter the plasma current (6.4 MA) and fusion power (370 MW). It is comparable to ITER in toroidal field (7.7 T), average density (1.5×10^{20} m^{-3}), average temperature (9.5 keV), and steady-state power consumption (370 MW). Here, the steady-state consumption includes power to maintain the plasma drive input (140 MW, at an efficiency of 50%), magnet cryogenic systems, and operation of the VNS device and test facility (~15% of total consumption). The plasma fusion amplification required for the VNS is modest ($Q \approx 2.6$) and corresponds to an ignition parameter of $\langle T \rangle_n \langle n_e \rangle \tau_E \approx 6.6 \times 10^{20}$ keV/m$^3 \cdot$s, which is a factor of ~3 below that required for ignition.

TABLE B.III

Key Parameters for VNS with Superconducting, Multiturn Normal Conducting, and Single-Turn Normal Conducting TF Magnets, and ITER

	ITER[a]	Superconducting	Multiturn Normal Conducting	Single-Turn Normal Conducting
Average neutron wall load (MW/m^2)	~1.0	1.0	1.0 to 2.0	1.0 to 2.0
Major radius, R_0 (m)	7.75	4.64	1.88 to 2.0	0.79 to 0.81
Minor radius, a (m)	2.8	1.05	0.6	0.6
Plasma current, I_p (MA)	24	6.4	5.3 to 6.4	9.4 to 10.4
Externally applied toroidal field, B_{t0} (T)	6.0	7.7	4.6 to 6.0	2.0 to 2.4
Volume average density, $\langle n_e \rangle$ (10^{20} m^{-3})	1.1	1.5	1.9 to 2.2	0.95 to 1.3
Density-average temperature, $\langle T \rangle_n$ (keV)	11	9.5	9.4 to 12.6	16
Divertor heat flux factor, f_{div} ($MW \cdot T^{0.5}/m^{1.5}$)	20	17[b]	17 to 24[b]	12 to 21[b]
Drive power, P_{drive} (MW)	0	140	51 to 60	19 to 29
Fusion power, P_{fusion} (MW)	1530	360	109 to 231	32 to 65
Electric power consumption, peak/steady state (MW)	800/400	370	700	130 to 180
Outboard accessible wall area (m^2)	TBD	56	35 to 36	20
Number of ports for plasma drive	N/A[c]	3	2	2
Number of ports for nuclear test modules	N/A	9	6	6
Test module cross section $w \cdot h$ ($m \cdot m$)	N/A	$3.1 \cdot 1.5$	$2.5 \cdot 1.8$	$2.1 \cdot 1.2$
Plasma volume (m^3)	~2000	150	28 to 30	10 to 11
Plasma surface area (m^2)	~1150	250	72 to 77	27 to 28
First-wall area, including inboard (m^2)	~1300	290	79 to 85	30 to 31

[a]Parameters chosen for the BPP of the ITER outline design.[57]
[b]Double-null poloidal divertors assumed.
[c]N/A designates not applicable.

Relative to ITER, this VNS is a factor of ~13 lower (150 m^3) in plasma volume and a factor of 4 lower (250 m^2) in plasma surface area. The divertor heat flux factor f_{div}, assuming anomalous diffusion in the SOL ($\gamma = 0.5$) and double-null divertors, is estimated to be ~17 $MW \cdot T^{0.5}/m^{1.5}$, comparable to the ITER value. The total wall area accessible from outboard between the outer TFC legs and the outboard PFCs is estimated to be ~56 m^2.

A significant reduction in device linear size (to ~10 m overall) from the superconducting option is obtained by using multiturn normal-conducting TFCs, in spite of the doubled wall loading. The latter permit a reduction in the inboard radiation shield (from 83 to 44 cm). The values for the plasma current and density remain similar to those for the superconducting option. Reductions in plasma drive power (to 51 to 60 MW) and fusion power (to 109 to 231 MW) are significant without leading to a significant change in the ignition parameter $\langle T \rangle_n \langle n_e \rangle \tau_E$ (≈ 5.1 to 8.4×10^{20} $keV/m^3 \cdot s$). The (double-null) divertor heat flux factor f_{div}, assuming anomalous diffusion in the SOL ($\gamma = 0.5$), remains similar to the superconducting case and ITER. A major drawback for this option, however, is the large increase in power consumption (700 MW), which is dominated by the normal-conducting TFCs that produce 5.3 to 6.4 T at the major radius of 1.9 to 2.0 m. A wall area of ~35 m^2 is accessible from the outboard side in this device.

The use of a single-turn normal-conducting inner leg for the TFCs permits the elimination of the inboard radiation shield, which leads to a further reduction in device size for constant neutron wall loading. The device is reduced to ~7 m overall in linear size, the major radius being ~0.8 m. The values for the plasma current, temperature, and density remain similar to the preceding case, but a large reduction in the toroidal field (to 2.0 to 2.4 T) is seen and is a result of the low aspect ratio ($R_0/a \sim 1.3$) (Ref. 61). Further reductions in plasma drive power (to 19 to 29 MW) and fusion power (to 32 to 65 MW) are obtained, now with a reduced ignition parameter ($\langle T \rangle_n \langle n_e \rangle \tau_E \approx 4.4$ to 5.5×10^{20} $keV/m^3 \cdot s$). The relatively small change in the fusion amplification Q (to 1.7 to 2.2) results from the reduced plasma volume (to 11 m^3) and a large contribution (~30%) in fusion power from a strong suprathermal ion component, which accounts for ~40% of the plasma pressure. Neutral beam injection heating

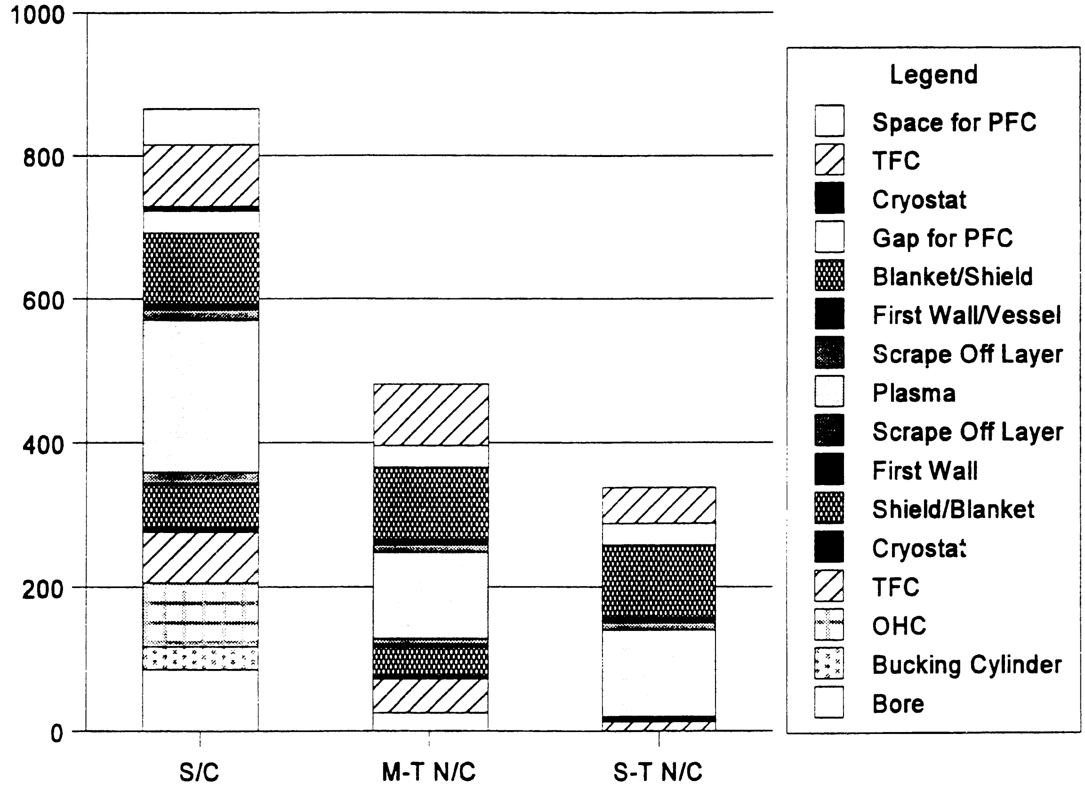

Fig. B.3. Radial build of tokamak VNS options (in centimetres) with superconducting, multiturn normal-conducting, and single-turn normal-conducting TFCs, producing an average wall loading of 1 MW/m^2.

and current drive at ~0.5 MeV (Ref. 71) can be used to achieve this condition, which is similar to those achieved or simulated recently in TFTR (Ref. 72) and JET (Refs. 73 and 74), respectively. The power consumption for this case amounts to ~130 to 180 MW, one-half of which is supplied to the TFCs. An outboard wall area of ~20 m^2 is accessible. Finally, the (double-null) divertor heat flux factor f_{div}, assuming anomalous diffusion in the SOL ($\gamma = 0.5$), remains unchanged. A key issue for this approach is the survivability and design of this single-turn, normal-conducting inner leg for the TFC.

The results for these representative VNS parameters with varying TF magnet approaches show a wide design envelope in size, field strength, drive power, fusion power, and electric power consumption. Over this range, similar values in plasma current, density, temperature, and divertor heat flux (assuming anomalous cross-field transport in the SOL) are achieved in producing a constant neutron wall loading of 1 to 2 MW/m^2 in the present study. Relative to the present ITER design, these VNS parameters are drastically smaller: the plasma volume by one to two orders of magnitude and the plasma surface area by factors between 5 to 50. This comparison is depicted in Fig. B.4. The parameters determined so far point to smaller fractions of the ITER cost.

B.V. DISCUSSION

Our results suggest that tokamak VNS with normal-conducting TFCs can be similar to present-day D-T tokamaks, such as TFTR (Ref. 72) and JET (Refs. 73 and 74), in several important parameters. The former do not exceed these experimental devices in major and minor radii, toroidal field, peak ion temperature, plasma

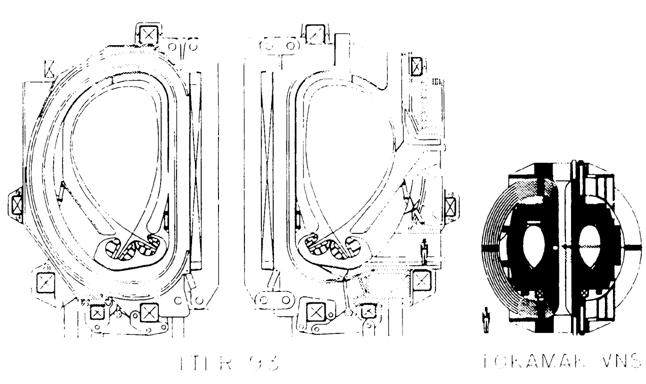

Fig. B.4. Elevation views for ITER (Ref. 58) and a typical tokamak VNS with multiturn normal-conducting TFCs (multiturn normal conducting, Fig. B.1) depicted in the same scale.

volume, and plasma surface area. However, important differences exist. These include about twice the plasma current, three to four times the density, three times the divertor heat flux factor, three to four times the neutral beam energy, more than three times the fusion power, and about three orders of magnitude the plasma duration.

The most important physics issues for a D-T-fueled VNS relative to TFTR and JET can therefore be identified. They include

1. steady-state current drive at densities of $\sim 1.0 \times 10^{20}\ m^{-3}$
2. steady-state plasma particle and power handling at divertor heat flux factors f_{div} of $\sim 14\ MW \cdot T^{0.5}/m^{1.5}$
3. steady-state neutral beam operation at energies of ~500 keV (Ref. 75) or rf operation with equivalent performance
4. for the low-aspect-ratio VNS option, tokamak-like plasma behavior or better in these areas.

In the engineering concept, the VNS tokamaks are different from TFTR and JET in the use of steady-state, demountable, jointed TFCs. In the case of the low-aspect-ratio option, a key difference is the use of a single-term, demountable center leg for the TFCs. The feasibility of these options should be estimated by engineering design studies.

The magnitude of these advances needed for VNS, however, is well within the capability of the present fusion research activities in the world. If the cost of such a VNS device and facility can be drastically below that anticipated for ITER, the former can become a highly attractive developmental step in fusion energy research.

ACKNOWLEDGMENTS

A number of distinguished scientists and engineers provided advice on some aspects of this study, contributed to discussions during the study workshop meetings, and/or made important comments on early drafts of this report. All such advice, comments, and contributions are gratefully acknowledged. Special acknowledgments are due K. Miya (University of Tokyo), S. Shimamoto (JAERI), J. Gilleland (Bechtel), M. Eid (Commissariat à l'Energie Atomique), I. Cook [United Kingdom Atomic Energy Authority (UKAEA)], T. Hender (UKAEA), M. Tillack (University of California, San Diego), O. G. Filatov (Efremov), A. B. Mineev (Efremov), Yu. Sokolov (Kurchatov Institute of Atomic Energy), and A. Ivanov (Budker Institute).

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