Phase 1 Report of the IEA Study on High Volume Plasma-Based Neutron Source (HVPNS)

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Abstract

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. It is found that the feasibility of blanket concepts can not be established prior to extensive testing in the fusion environment. Comprehensive analysis shows that the fusion testing requirements are: 1-2 MW/m^2 neutron wall load, steady state plasma operation, > 10 m^2 test area, and a fluence of > 6 $MW{\bullet}y/m^2$. This testing fluence includes 1-3 $MW{\bullet}y/m^2$ for concept performance verification and > 4-6 $MW{\bullet}y/m^2$ for component engineering development and reliability growth. Reliability and availability analyses reveal critical concerns in fusion power development. For a DEMO 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 year. For a blanket that has 80 modules, the corresponding MTBF per module is 80 years. These are very ambitious goals. A number of scenarios for fusion facilities were evaluated using a cost/benefit/risk analysis approach. Blanket tests in ITER alone with a fluence of 1 $MW{\bullet}y/m^2$, cannot demonstrate a DEMO availability higher than 4%. An effective path to fusion DEMO involves two parallel facilities: 1) ITER to provide data on plasma performance, plasma support technology, and system integration, and 2) a High-Volume Plasma-based Neutron Source (HVPNS) dedicated to testing, developing, and qualifying fusion nuclear components and materials combinations for DEMO. HVPNS has a highly driven plasma and produces fusion power of < 150 MW. A testing strategy employing HVPNS would decisively reduce the very high risk of unsuccessful blanket operation in DEMO, and would make it possible - if operated in parallel to the ITER Basic Performance Phase - to meet the goal of DEMO operation by the year 2025. A scenario with HVPNS parallel to ITER provides substantial savings in the overall R & D cost towards DEMO compared to 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.

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1. Introduction and Background

In early 1994, the International Energy Agency's Fusion Power Coordinating Committee (FPCC) requested that action be taken on a High Volume Plasma-Based Neutron Source (HVPNS) Concept Definition Effort by the Provisional Executive Committee (PEC) of the Implementing Agreement on Nuclear Technology of Fusion Reactors. The FPCC charter for the HVPNS Concept Definition Effort specified that the first task should be to review the mission of and the technical approaches to a HVPNS.

In March 1994, the PEC prepared a work outline for a HVPNS Concept Definition Effort, which specified that the following two phases would be conducted in series:

Phase 1: Assess the needs for HVPNS from the viewpoint of fusion nuclear technology developers, reach a consensus on the mission for HVPNS, and determine the general testing capabilities, design features, and operating parameters required for a HVPNS to accomplish its mission.

Phase 2: Identify candidate plasma-based device concepts that can potentially meet the HVPNS requirements and assess the technical and economic feasibility of the leading concepts.

The PEC specified that Phase 1 be conducted as a user community assessment by specialists in fusion nuclear technology development, with the following factors being considered:

- Definition of DEMO and long-range fusion development strategies leading to DEMO
- Nuclear technology data base needs for DEMO
- The testing capabilities of ITER and non-fusion facilities for meeting the nuclear technology data base needs of DEMO
- Potential for arriving at a DEMO with too high of a technological risk (e.g., too low a probability of achieving target availability levels) based on the nuclear technology data bases from testing only in ITER and non-fusion facilities
- The DEMO technological risk reduction benefits derived from HVPNS nuclear component and materials combination testing that complements nuclear testing in ITER and non-fusion facilities
- Optimal strategies for phasing of ITER, HVPNS, and non-fusion facility operation and testing programs

At the PEC's invitation, Professor Mohamed Abdou of the University of California, Los Angeles led the Phase 1 effort.

Three Phase 1 workshops were held in April, July and October, 1994 with participants from the European Union, Japan, and the USA. The Russian Federation participated informally pending formalization of their participation as an Associate Contracting Party to the Implementing Agreement on Nuclear Technology of Fusion Reactors.

The Phase 1 effort concluded in late 1994 with successful completion of all Phase 1 activities requested by the PEC. This report synthesizes the results of the technical study performed by the participants.

The report is organized to correspond to the technical tasks identified early in the study. Section 2 identifies and compares the DEMO goals in EU, Japan, USA and RF. Section 3 briefly reviews the technical issues for fusion nuclear technology with the major focus being the blanket/first wall systems, which determines the critical path in FNT development. Section 4 evaluates the role and limitations of non-fusion facilities. Extensive testing in fusion facilities is found to be necessary; Sec. 5 quantifies the FNT requirements on such fusion testing. Section 6 discusses the possible mission, objectives and design guidelines for HVPNS (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 cost/benefit/risk analysis approach in Sec. 7. Sections 8 and 9 present the study conclusions and recommendations, respectively.

2. DEMO Goals

2.1 Introduction

DEMO refers to a thermonuclear fusion demonstration power plant based on magnetic confinement of plasmas with a deuterium-tritium fuel cycle.

The results of DEMO and DEMO-relevant planning and study activities by the fusion programs of the European Union (EU), Japan, the Russian Federation (RF), and the United States (US) 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 report and do not represent official positions on DEMO definition by any of the fusion programs.

2.2 Fusion Program Perspectives

2.2.1 EU Perspective

The most recently published document expressing general EU views on DEMO are contained in the July 1990 report for the Commission of the European Communities by the Fusion Programme Evaluation Board, which was chaired by Professor Colombo (hereafter referred to as the Colombo report).

The EU foresees a stepwise strategy toward a prototype commercial power plant involving, after JET, a next-step experimental device (e.g., ITER), and then a DEMO. With regard to the time scale for the start of DEMO operation, the date 2025 appears in the Colombo report for DEMO start-up. However, this date will be reexamined by the EU to fit the likely timetable for ITER construction and operation.

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, the Colombo report indicated 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 around 5 MW•y/m^2, while the long-term goal would be > 10 MW•y/m^2.

There is currently no planned EU effort to further define the mission and objectives or the major parameters and features of DEMO. Table 1 summarizes historical positions regarding the major parameters and features for DEMO.

2.2.2 Japan Perspective

In the Third Stage Fusion R&D Plan issued in June 1993 by the Fusion Council under the Atomic Energy Commission, it is indicated that fusion will be developed with the aim of contributing to the energy supply in the latter half of the next century. According to this plan, if the 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. In a recent draft pamphlet showing the annual progress of fusion research in the Japan Atomic Energy Research Institute, the operation of DEMO is shown to be around 2030.

The Third Stage Fusion R&D Plan indicates that the mission of the DEMO phase of fusion R&D is to demonstrate, in a plant scale, the technological feasibility to realize a high energy multiplication steady-state plasma, to extract energy generated from the plasma, and to convert 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. 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., 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 1 provides examples of DEMO major parameters and features based on the above.

2.2.3 RF Perspective

The RF 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 data base from ITER, of performing 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 1 summarizes basic technical requirements for DEMO.

2.2.4 US Perspective

The September 1990 final report of the Department of Energy's Fusion Policy Advisory Committee recommended that the US fusion program become energy oriented, with the goals of an operating DEMO by 2025 and an operating commercial power plant by 2040. It was acknowledged that achieving the 2025 goal for DEMO operation would require substantial increases in US 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 US strategy for magnetic fusion energy development, DEMO follows immediately after ITER and is based primarily on the physics and technology data bases derived from operation of and testing programs in ITER, a materials test facility, and the Tokamak Physics Experiment. 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 pre-conceptual 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 US utilities, industry, and regulatory agencies. The first workshop on the subject of DEMO reached the following conclusions about DEMO characteristics: shows for the first time all systems working as a full-scale integrated unit; addresses issues of dependability and reliability and is large enough that the step to a prototype commercial plant leaves no open questions about scalability; establishes the licensing procedures and rules for a fusion power plant; demonstrates public acceptability and cost viability; demonstrates feasibility and acceptable costs for decontamination and decommissioning; demonstrates that the industrial infrastructure exists to serve the needs of the end-users; and 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 1 provides judgments on guidelines until STARLITE has completed its work on these matters.

Table 1. 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 Tritium Breeding Ratio (TBR)	Self-sufficient TBR > 1.0	Self-sufficient TBR > 1.0	Self-sufficient TBR: 1.05-1.1	Self-sufficient TBR > 1.0+ addition for doubling time
Power Output	Significant amounts of electricity	3 GW fusion power	< 1.5 GW electric	Hundreds of MW electric
Neutron Wall Loading in MW/m^2	2-3	up to 5.0	2-3	2-3 average 3-4 peak
Availability	Depends on DEMO mission; could be > 50% for reactor island	70%	> 60 %	50% net plant goal [a]
Thermal Efficiency	Unspecified	30-40% net	> 40%	> 30% net
Blanket Lifetime Goal in $MW{\bullet}y/m^2$	Depends on specific DEMO goals; could be 5 for first blanket and > 10 long-term	up to 7	15-20	10-20
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.

3. Technical Issues and Types of Testing

Fusion nuclear technology (FNT) 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 so as to close the fuel cycle,
3) provide the vacuum boundary for the plasma-containing chamber,
4) provide radiation protection to components and personnel.

3.1 Technical (Testing) Issues

Table 2 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 3. 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-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, V alloys, and SiC composites.

Fusion nuclear technology testing issues have been identified and characterized in pervious studies (e.g. [1-9]). These issues include feasibility issues and attractiveness issues. Feasibility issues are those whose negative resolution will have the following impact:

a. may close the design window
b. may result in unacceptable safety risk
c. may result in unacceptable reliability, availability of lifetime.

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

a. reduced system performance
b. reduced component lifetime
c. increased system cost

d. less desirable safety or environmental implications.

Table 2. Fusion Nuclear Technology Components and Other Components Affected by the Nuclear Environment

1. Blanket/ First Wall*
2. Plasma Interactive and High Heat Flux Subsystems -Divertor, limiter -rf antennas, launchers, wave guides
3. Shield
4. Tritium Processing Systems
5. Instrumentation and Control
6. Remote Maintenance
7. Heat transport and power conversion

*The blanket determines the critical path to fusion nuclear technology development

A summary of the testing issues for the blanket/first wall system is shown in Table 4. 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 below.

For solid breeder blankets, the major classes of issues include:

- Tritium self sufficiency
- Breeder/multiplier/structure interactive effects under nuclear heating and irradiation
- Tritium inventory, recovery and control; development of tritium permeation barriers
- Thermal control
- Allowable operating temperature window for breeder
- Failure modes, effects, and rates
- Mass transfer
- Temperature limits for structural materials and coolants
- Mechanical loads caused by major plasma disruption
- Response to off-normal conditions

Table 3. Worldwide Blanket Options for DEMO

Breeder	Coolant	Structural Material
A. <u>Solid Breeders</u> Li_2O, Li_4SiO_4, Li_2ZrO_3, Li_2TiO_3	He <u>or</u> H_2O	FS†, V alloy, SiC Composites
B. <u>Self Cooled Liquid Metal Breeders</u> Li, LiPb	Li, LiPb	FS, V alloy with Electric Insulator, SiC Composites with LiPb only
C. <u>Separately Cooled Liquid Metal Breeders</u> Li LiPb	He He <u>or</u> H_2O	FS, V alloy FS, V alloy, SiC Composites

* almost all concepts use beryllium as neutron multiplier
† FS = Ferritic Steel

For self-cooled liquid metal blankets, including concepts with 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 concern, especially in the case of a liquid metal spill, because of the production of the α-emitter Po-210. This may require an on-line bismuth-removal technique. Corrosion and mass transfer is an issue 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 bi-metallic loops is a key concern. Large stored chemical reactivity of lithium is a serious issue if water cannot be excluded from the system.

Table 4 . List of Blanket/First Wall Testing Issues

A. Structure
1. Changes in properties and behavior of materials
2. Deformation and/or breach of components
   a. Effect of first-wall heat flux and cycling on fatigue or crack growth-related failure
   b. Magnetic forces within the structure (including disruptions)
   c. Premature failure at welds and discontinuities
   d. Failures due to hot spots
   e. Interaction of primary and secondary stresses and deformation
   f. Effect of swelling, creep, and thermal gradients on stress concentrations
   g. Failure due to shutdown residual stress
   h. Interaction between surface effects and first-wall failures
   i. Self-welding of similar and dissimilar metals
3. Tritium permeation through the structure
   a. Effectiveness of tritium permeation barriers
   b. Effect of radiation on tritium permeation
4. Structural activation product inventory and volatility
5. Hermiticity of SiC

B. Liquid Metal Coolant
1. MHD pressure drop and pressure stresses
2. MHD and geometric effects on flow distribution
3. MHD insulating coating fabrication, integrity, and in-situ self-healing
4. Stability/kinetics of tritium oxidation in the coolant
5. Coolant/purge stream containment and leakage
6. Activation products in Pb-Li
7. Liquid metal purification

C. Breeder and Purge
1. Tritium recovery and inventory in solid breeder materials
2. Liquid breeder tritium extraction
3. Temperature limits and variability in solid breeder materials
   a. Dimensional stability under irradiation
   b. Thermal conductivity changes under irradiation
   c. Effect of cracking
   d. Effect of LiOT mass transfer
4. Breeder behavior at high burn-up/high dpa

D. Coolant/structure interactions
1. Mechanical and materials interactions
   a. Corrosion
   b. Mechanical wear and fatigue from flow-induced vibrations
   c. Failure of coolant wall due to stress corrosion cracking
   d. Failure of coolant wall due to liquid metal embrittlement
   e. Bi-metallic loop impurities transfer
2. Thermal Interactions
   a. MHD effects on first-wall cooling and hot spots
   b. Response to cooling system transients
   c. Flow sensitivity to dimensional changes
3. Coolant/Coatings/Structure interactions

E. Solid breeder/multiplier/structure interactions
1. Solid breeder mechanical and materials interactions
   a. Clad corrosion from breeder burnup products
   b. Strain accommodation by creep and plastic flow
   c. Swelling driving force
   d. Stress concentrations at cracks and discontinuities
   e. Thermal expansion driving force
2. Neutron multiplier mechanical interactions
   a. Beryllium swelling (Swelling driving force in Be)
   b. Strain accommodation by creep in beryllium
   c. Mechanical integrity of unclad beryllium
3. Thermal interactions
   a. Breeder/structure and multiplier/structure interface heat transfer (gap conductance)

F. General blanket
1. D-T fuel self-sufficiency
   a. Uncertainties in achievable breeding ratio
   b. Uncertainties in required breeding ratio
2. Tritium permeation
   a. Permeation from breeder to blanket coolant
   b. Permeation from beryllium to coolant
   c. Permeation characteristics at low pressure
3. Chemical reactions
4. Tritium inventory
5. Failure modes and frequencies
6. Nuclear heating rate predictions
7. Time constant for magnetic field penetration for plasma control
8. Blanket response to near blanket failures
9. Assembly and fabrication of blankets
10. Recycling of irradiated lithium and beryllium
11. Prediction and control of normal effluents associated with fluid radioactivity
12. Liquid-metal blanket insulator fabrication, effectiveness, and lifetime
13. Tritium trapping in beryllium

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 fusion nuclear technology, which stresses the key functional aspects of the fusion reactor that must be resolved through testing, is given in Table 5.

Table 5. Summary of Critical R&D Issues for Fusion Nuclear Technology

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

3.2 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 have also been addressed in previous studies (e.g. references 1-8). However, these studies focused more on testing in non-fusion 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 are adequately simulated. The key fusion environmental conditions are indicated in Table 6. 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 in order to reduce cost.

Table 6. Key Fusion Environmental Conditions for Testing Fusion Nuclear Components

• **Neutrons** (fluence, spectrum, 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

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 7 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 7. Test Categories for Blanket R&D

Basic test
- Basic or intrinsic property data
- Single material specimen
- Examples: thermal conductivity; neutron absorption cross section

Single-effect test
- Explore a single effect, a single phenomenon, or the interaction of a limited number of phenomena, in order 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; (c) tritium production rate 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 in order 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; 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 condition to permit large cost savings
- All key physical elements of the component; 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; (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 in 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

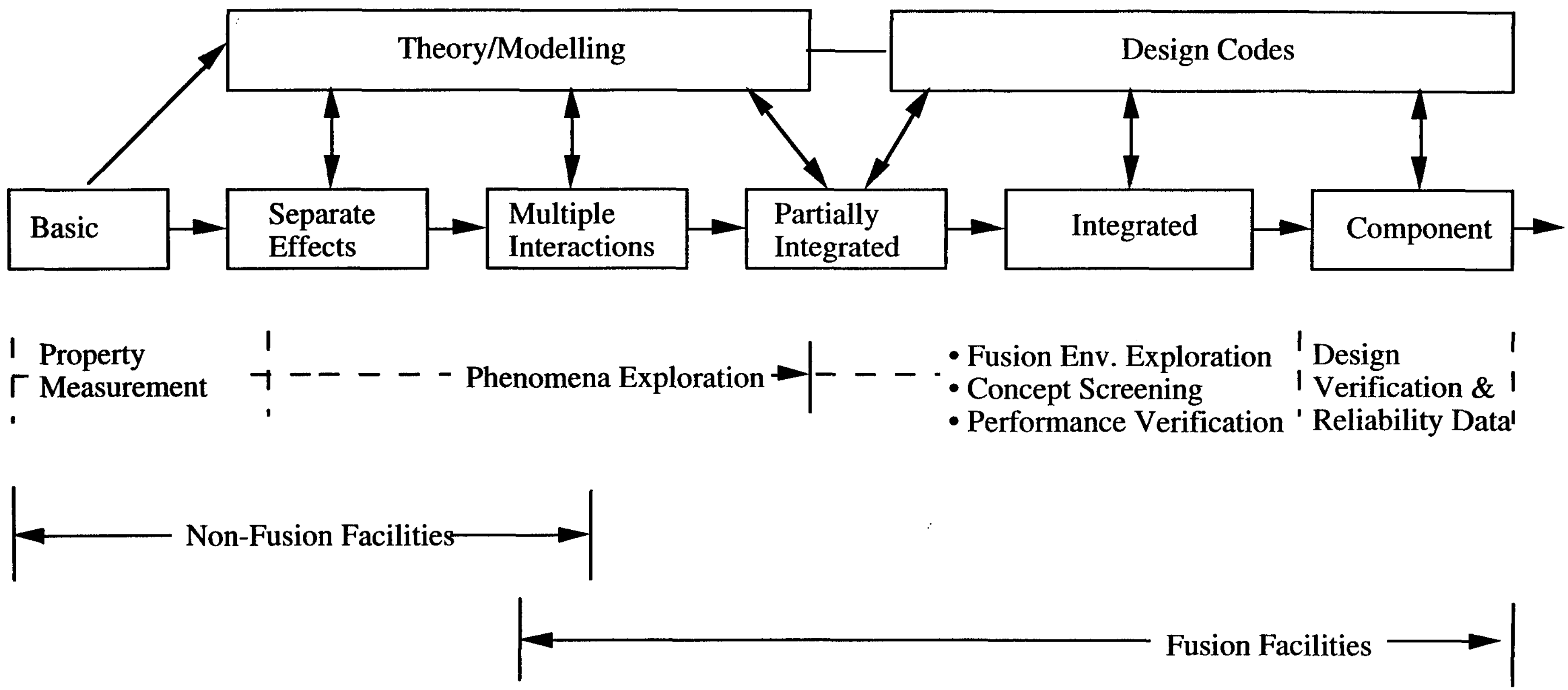

Figure 1. Types and roles of experiments and facilities for fusion nuclear technology

of the blanket to examine a particular group of multiple effects can be designed for testing in the fusion environment.

4. Role and Limitations of Non-Fusion Facilities

Non-fusion facilities can and should play a role in FNT R&D because of availability and low cost. Information from testing in non-fusion 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 non-fusion facilities have very serious limitations. Blanket concepts can not be verified in non-fusion facilities, not to mention component engineering development and reliability growth. Non-fusion facilities tests can not replace the need for a comprehensive testing program in fusion facilities; they can only help reduce the costs and risks of the early stages of this program. Non-fusion facilities can be classified into: a) non-neutron test stands, b) fission reactors, and c) point neutron sources. Each of these is discussed briefly below.

4.1 Non-Neutron Test Stands

The role of non-neutron test stands is in the area of basic property data, single-effect experiments, and 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 as well as radiation effects, the value of non-neutron test stands is limited. Studies in the early 1980's have assumed that 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. One of the fundamental feasibility issues relates to the imperfections in such coatings: a) how fast do they occur, b) how fast do 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. Therefore, an experiment with a prototypical test section in an environment that combines neutrons and a magnetic field is 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 the next sub-sections.

The above examples do not argue against tests in a non-neutron environment. They only emphasize the fact that feasibility of blanket concepts can not be established prior to testing in the fusion environment. Experiments in non-neutron 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.

4.2 Fission Reactors

Fission reactors provide neutrons in a limited volume and are thus suited to some FNT experiments. Table 8 summarizes the capabilities of fission reactors available in the USA, Canada, Russia, and Europe for blanket tests. Testing in fission reactors suffers from serious limitations which are listed in Table 9. Most serious is the small test volume. For example, there is no fission reactor operating now anywhere in the world that can provide a test location with ≥ 15 cm equivalent circular diameter at a fast neutron flux equivalent to 1 MW/m^2 wall loading ($\geq 1 \times 10^{15}$ $n \cdot cm^{-2} \cdot s^{-1}$). This limitation, together with some safety aspects of fission reactors, also makes the simulation of non-nuclear 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.

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 non-nuclear 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.

4.3 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 1 MW/m^2 wall load. Furthermore, the life of the target is limited to < 100 hour irradiation. Therefore, the usefulness of DT point neutron sources is limited to neutronics experiments, e.g. measurements of tritium production rates.

Table 8. Capabilities of Available Fission Reactors for Blanket Tests

Reactor	Location	Reactor Power (MW)	Fast Flux (n/cm^2s)	Thermal Flux (n/cm^2s)	Dimension of Irradiation Channel (cm)	Effective Core Height (cm)
ATR	US	250	$3.7x10^{13}$ $5.25x10^{14}$ $1.9x10^{14}$	$2.48x10^{14}$ $4.64x10^{14}$ $8.8x10^{14}$	3.81 (circular) 1.59 (circular) 6.05 (7 flux traps)	122
HFIR	US	100	$1.5x10^{15}$	$2.3x10^{15}$	3.7 (circular)	51
EBR-II	US	62	$2.0x10^{15}$		7.4 (circular)	36
BOR-60	Russia	16	$2x10^{15}$	$2.4x10^{15}$	4.4 (hexagonal)	45
IVV-2	Russia	15	$6.6\text{-}14x10^{13}$	$1.5\text{-}6x10^{14}$	3-6 (dia.)	50
SM-3	Russia	100	$1.0x10^{15}$	$1.5x10^{14}$	4-7 (circular)	25
OSIRIS	France	70	$5.0x10^{14}$	$1x10^{14}$	8.4 (circular)	60
SILOE	France	35	$5.0x10^{14}$	$4.0x10^{14}$	8.0 (circular)	60
BR-2	Belgium	60	$6.0x10^{14}$	$1.0x10^{15}$	20 (circular)	96
HFR	Netherlands	20	$5.0x10^{14}$		14.5 (circular)	60
PHENIX	France	250	$1.3x10^{15}$		12.67 (hexagonal)	85
JRR-2	Japan	10	$1.0x10^{14}$	$1.0x10^{14}$		
NRU	Canada	125	$4x10^{13}$	$2.4x10^{14}$	10 (circular)	300

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 DT point neutron source is the FNS facility in Japan [10]. The capabilities of FNS are compared in Table 10 to those from recent DT shots in TFTR [11]. JET [12] 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 DT point neutron sources. The key problem with 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 (D,Li) source in which neutrons are produced by bombarding a flowing lithium target with energetic (~30-40 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, $^{7}Li\,(d,np)\,^{7}\overset{*}{Li} \rightarrow T + \alpha$, $^{7}Li\,(d,2n)\,^{7}\overset{*}{Be} \rightarrow {}^{3}He + \alpha$, $^{7}Li\,(d,n)\,^{8}Be$, $^{7}Li\,(d,3n)\,^{6}Be$ and other reactions.

Table 9. Key Limitations of Fission Reactors

1. Small test volume
   a. small size per location
   b. small number of existing locations

2. Lack of fusion-related (non-neutron) conditions
   a. magnetic field
   b. surface heat
   c. particle flux
   d. mechanical forces

3. Different radiation damage simulation
   a. neutron spectra
   b. He/dpa ratio
   c. types and rates

4. Power density
   a. magnitude
   b. spatial profile

5. Lithium burn up rate
   a. magnitude
   b. spatial profile

6. Reactivity considerations limits on size and type of experiments

7. Availability of fission test reactors for testing (rapid downward trend)

Design of a D-Li source (FMIT) was started [13, 14] in the late 1970's in the US and was later terminated during construction due to a combination of funding problems and technological issues. Recently, an international activity under the auspices of the International Energy Agency (IEA) was started [15-17] to examine the need and issues for a D-Li source called IFMIF (International Fusion Material Irradiation Facility). Examples of analysis of neutronics characteristics of IFMIF-type facilities are given in Refs. 17-19.

One advantage of such a source is the existing experience with accelerators. Another potential advantage is the possibility to perform 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: 1) neutron spectrum, 2) steep flux gradient, and 3) the surface area and volume available for testing.

The D-Li neutron source produces neutrons with energies from the eV's up to ~50MeV. This is compared to the fusion D-T reaction where neutrons are produced within a narrow energy range around 14 MeV. The neutron spectrum from the D-Li reaction varies with the incident deuteron energy. As shown in Table 11 (see Ref. 18), the fraction of the neutrons above 15 MeV increases from 8% to 15.7% when the incident deuteron energy is increased from 30 MeV to 40 MeV. The average neutron energy is about 6 MeV for 35 MeV deuterium beam. The low energy component of the D-Li source may be able to simulate qualitatively the neutron spectrum created by back scattering 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 threshold 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. Available models to correlate neutronics parameters such as dpa, helium production rates, etc., to observe macroscopic behavior of materials are not reliable.

Table 10. Comparison of Present DT 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 cycles per day	~ 10 hours per day
Neutron Flux, n/cm^2•s	<u>at the first wall</u> 2×10^{12}	<u>at 5 cm from target</u> 6.4×10^9 <u>at 1 m from target</u> 1.6×10^7

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. Fig. 2 from Gomes work [19] shows the 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 are typically less than 0.1% per cm. The flux gradient at the test samples with 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 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 treated briefly below.

Table 11. 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 D-beam (neutrons/sec)	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:			
0-15 MeV	91.9	88.1	84.3
15-50 MeV	8.1	11.9	15.7

Optimization studies for the D-Li source suggest a 250 mA beam with 35 MeV deuterons. Fig. 2 (from ref. 19) shows the dpa rate per full power year in the direction perpendicular to the beam. Table 12 shows a- the test area, b- test volume obtainable with 35 MeV, 250 mA D-Li source. Table 12.a shows the maximum surface area available for testing with

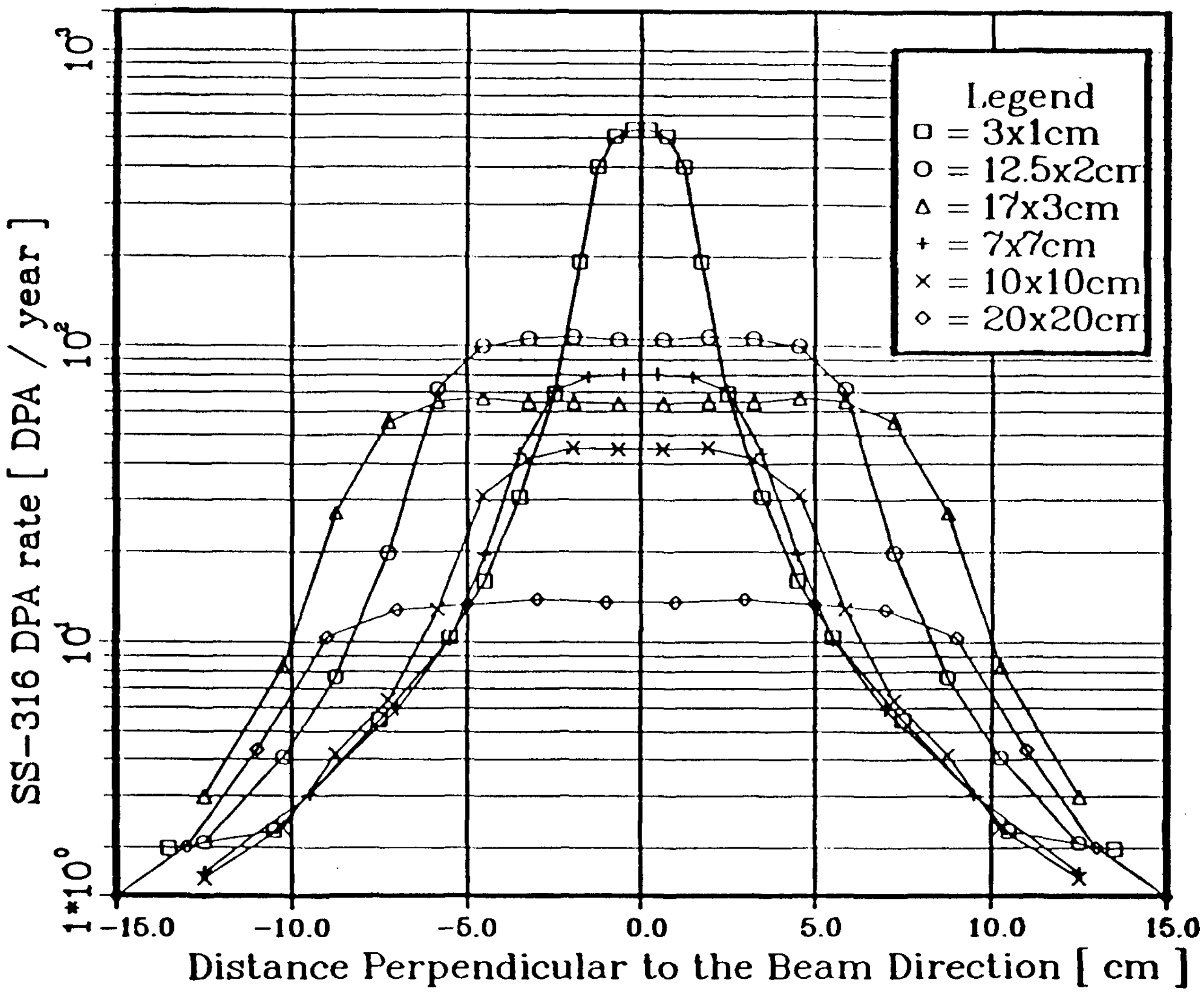

Figure 2. Gradient of the SS-316 DPA rate perpendicular to beam
Beam Current = 250 mA, Deuteron Energy = 35 MeV

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. Results show that the maximum surface area available for testing is 200 cm^2 (obtainable with beam spot area 20 cm x 20 cm) at an equivalent neutron wall load of 1 MW/m^2. The maximum test area with equivalent neutron wall load of 3 MW/m^2 is only 50 cm^2 (obtainable with beam spot area of 10 cm x 10 cm).

Table 12. Test Area and Test Volume Available with D-Li Neutron Source (35 MeV, 250 mA deuteron beam).

a. Surface Area Available for Testing with D-Li Neutron Source (35 MeV, 250 mA deuteron beam) to Simulate First Wall Conditions of a Fusion Reactor

(Equivalent) Neutron Wall Load	Maximum Surface Area Available for Testing[a]	Comments
1 MW/m^2	200 cm^2	b
3 MW/m^2	50 cm^2	c

a- Area perpendicular to beam direction
b- Possible with beam spot area 20cm x 20cm
c- Possible with beam spot area 10cm x 10cm

b. 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 x cm)	
dpa / yr.*	10 x 10	20 x 20
30	10cm^3	0
20	100cm^3	0
10	300cm^3	7cm^3

* Assuming a plant factor of 70% and stainless steel as typical material

Clearly, such test area is not suitable for module or even submodule testing. Therefore, D-Li sources can not play a 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 13 summarizes the space requirements for material specimen tests. Information in this table was first developed in INTOR [20] and subsequently improved in FINESSE [1, 2] and ITER-CDA studies [21-22]. The table is limited to structural materials and assumes four candidate metallic alloys are to be investigated. Some observations are in order here. First, non-structural 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 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 this table. 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.

Based on the above points, one concludes that the test volume defined in Table 13 is a minimum for the material science irradiation specimen matrix. Table 13 shows that more than 30,000 specimens are needed with a volume greater than 2000 cm^3. This volume does not include the additional space needed for cooling, support, instrumentation, and other functions.

Table 13. 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 Temp.	Fluence	Flux	Stress	Multiplicity	Total Specimens	Vol./Spec. (cm^3)	Total Vol. (cm^3)
Charpy-v	1/3 CVN	(3.3x3.3x23.6)	4	7	4	1	0	8	896	0.26	235
Tensile	Flat	(0.76x25.4x5.0)	4	7	4	1	0	15	1680	0.10	162
Creep	Tube	(4.57 dia.x23.0)	4	7	1	1	6	1	168	0.38	63
Swelling	Disc	(3.18 dia.x0.25)	144	7	4	1	0	6	24192	0.002	48
Fracture Toughness	Compact Tension	(16 dia.x2.5)	4	7	4	1	0	16	1792	0.5	901
Stress Corrosion Cracking	CERT SS-3	(0.76x25.4x5.0)	4	7	4	1	0	24	2688	0.10	259
Fatigue	Constant Amplitude/ High Cycle	(6.35 dia.x38)	4	7	4	1	0	3	336	1.2	404
Total for all Specimens (does not include volume for coolant and support)								----	31752	----	2072

Table 12.b shows the test volume available with dpa rate greater than a specified threshold for a D-Li source. With 20 cm x 20 cm beam focus, only 7 cm^3 is available with dpa rate of 10 per year. With 10 cm x 10 cm beam focus area, higher dpa rates are possible but still at very small test volume. The maximum test volumes at 30, 20, and 10 dpa per year are 10, 100, and 300 cm^3, respectively. These volumes are to be compared to the requirements of >2000 cm^3 in Table 13 for 4 candidate structural material science specimen irradiation. Note that the DEMO has 2-3 MW/m^2 as discussed earlier, while commercial fusion reactors will need ~3-5 MW/m^2. Also note that the dpa rate per full power year of operation at a wall load of 1 MW/m^2 in a tokamak first wall is ~11-12 for typical candidate structural materials. Assuming a 70% plant availability for D-Li source, dpa rates much greater than 50 dpa per full power year are needed if "accelerated" testing is to be possible.

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 wall load is comparable only to ITER and is about 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 non-structural 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 to <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 test space - test time matrix need to be carefully analyzed. Some of the material-science specimens tests will need to be performed in fusion testing facilities, and 5) results from specimen irradiation tests are generally meaningful only if performed in 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.

4. 4 Summary of Role and Limitations of Non-Fusion Facilities

It is important to assess the overall contribution of non-fusion facilities to the development of fusion nuclear technology. Table 14 summarizes the capabilities of non-fusion facilities for simulation of key conditions for fusion nuclear component experiments. The most important conditions are: 1) neutron effects (radiation damage, tritium and helium production), 2) bulk heating (nuclear heating in a significant volume), 3) non-nuclear conditions (e.g. magnetic field, surface heat flux, particle flux, mechanical forces), 4) conditions for simulating thermal-mechanical-chemical-electrical interactions, and 5) conditions for integrated tests and synergistic effects. A very important conclusion is that non-fusion 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 2. Table 15 shows the contribution of non-fusion 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 non-fusion facilities alone. The second most striking conclusion is that there are critical issues for which no significant information can be obtained from testing in non-fusion facilities. An example is identification and characterization of failure modes, effects and rates. Therefore, the feasibility of blanket concepts can not be established prior to testing in fusion facilities.

Table 14. Capabilities of Non-fusion Facilities for Simulation of Key Conditions for Fusion Nuclear Components Experiments

	Neutron Effects(1)	Bulk Heating(2)	Non-Nuclear(3)	Thermal/ Mechanical/ Chemical/ Electrical(4)	Integrated Synergistic
Non-Neutron Test Stands	no	no	partial	no	no
Fission Reactor	partial	partial	no	no	no
Accelerator-Based Neutron Source	partial	no	no	no	no

(1) radiation damage, tritium and helium production
(2) nuclear heating in a significant volume
(3) magnetic field, surface heat flux, particle flux, mechanical forces
(4) thermal-mechanical-chemical-electrical interactions (normal and off normal)

The word "partial" in Table 15 designates a contribution which 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 15. Contribution of Nonfusion Facilities to Resolving Critical Issues for Fusion Nuclear Technology

Critical Issue	Non-neutron Test Stands	Fission Reactors	Accelerator Based Neutron Sources	
			D-T	D-Li
1. D-T fuel cycle self sufficiency	none	none	partial[a]	none
2. Thermomechanical loadings and response of blanket components under normal and off-normal operation	small	small	none	none
3. Materials compatibility	some	some	none	small
4. Identification and characterizations of failure modes, effects and rates	none	none	none	none
5. Effect of imperfections in electric (MHD) insulators in self cooled liquid metal blanket under thermal/mechanical/electrical/nuclear loading	small	small	none	none
6. Tritium inventory and recovery in the solid breeder under actual operating conditions	none	partial	none	none
7. Tritium permeation and inventory in the structure	some	partial	none	none
8. Radiation shielding: accuracy of prediction and quantification of radiation protection requirements	none	small	partial	small
9. Plasma-facing component thermomechanical response and lifetime	some	some	none	some
10. Lifetime of first wall and blanket components	none	partial	none	partial[a]
11. Remote Maintenance with acceptable shutdown time	none	none	none	none

[a] - Partial: substantial contribution when supplemented by fusion test; not meaningful in the absence of fusion tests

It should be emphasized here once again that the above conclusions do not suggest that non-fusion facilities should not be used. They only suggest that their usefulness in resolving the critical issues is severely limited. Non-fusion facilities can and should be used to narrow materials 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 non-fusion 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 from here is that fusion nuclear technology development does require fusion testing facilities.

5. FNT Requirements For Testing In Fusion Facilities

The preceding section has shown that non-fusion facilities, albeit useful, are severely limited in simulating the key conditions for fusion nuclear component experiments and development (see Table 14). Based on results from Table 15, it is clear that non-fusion 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: 1) how should the tests in the fusion environment be structured to effectively develop and qualify FNT components for DEMO?, and 2) what are the requirements of FNT tests on the major parameters and characteristics of suitable fusion test facilities?

5.1 Testing Stages and Framework

Figure 1, shown earlier, illustrated a loose chronological order of tests for major nuclear components such as the blanket. Tests in non-fusion 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: 1) initial fusion "break-in" in the fusion environment, 2) concept performance verification, and 3) 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 caliberation and exploration of the fusion environment as well as testing and development of experimental techniques and diagnostic tools (for example, how do we measure, collect data, and interpret and extrapolate results; and 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

FIG. 3: STAGES OF FNT TESTING IN FUSION FACILITIES

	Fusion "Break–in"	Concept Performance Verification	Component Engineering Development & Reliability Growth	DEMO
Stage:	I	II	III	
Required Fluence ($MW{-}y/m^2$)	~ 0.3	1 – 3	>4 – 6	
Size of Test Article	**Sub–modules**	**Modules**	**Modules/Sectors**	
	■ Initial exploration of performance in a fusion environment ■ Calibrate non–fusion tests ■ Effects of rapid changes in properties in early life ■ Initial check of codes and data ■ Develop experimental techniques and test instrumentation ■ Narrow material combination and design concepts ■ 10–20 test campaigns, each is 1–2 weeks	■ Verify performance beyond beginning of life and until changes in properties become small (changes in structure mechanical properties are substantial up to about 1–2 $MW{\cdot}y/m^2$) ■ Data on initial failure modes and effects ■ Establish engineering feasibility of blankets (10 to 20% of lifetime) ■ Select 2 or 3 concepts for further development	■ Identify failure modes and effects ■ Iterative design/test/fix programs aimed at improving reliability and safety ■ Failure rate data: Develop a data base sufficient to predict mean–time–between–failure with sufficient confidence ■ Obtain data to predict mean–time–to–replace (MTTR) for both planned outage and random failure ■ Develop a data base to predict overall availability of FNT components in DEMO	

Stage III tests are complex, costly, and time consuming, the number of concepts should not exceed 3. 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 a latter section, 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 the next section. 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.

5.2 Testing Requirements on Major Parameters of Fusion Facilities

Satisfactory testing of the blanket in the fusion environment imposes important requirements on the design of the fusion testing facility in at least two areas: 1) major parameters, and 2) engineering design. The major parameters of concern are those that have major impact on both the usefulness of the tests and the cost of the device. The requirements on 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 on the major parameters for fusion facilities have been analyzed in several major studies [2-3, 21-28]. International workshops have also helped 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 EDA (Engineering Design Activity) have made it necessary to investigate in more detail the FNT testing requirements. A summary of our results for the FNT requirements on major parameters for testing in fusion facilities is given in Table 16. The requirements given in Table 16 are driven by the goal of providing the data base necessary to construct the blanket for DEMO.

There are other important requirements that are not given in Table 16, such as the value of the magnetic field in the blanket test region (e.g. to test liquid metal blankets or effects of ferritic steel in magnetic performance), surface heat flux, and minimum test area per module. We limited Table 16 to those requirements that appear to be a major discriminating factor in the selection among options for fusion testing facilities. Other parameters not given in Table 16 are either implied or can be deduced from those already given or do not appear to be a crucial discriminating factor in the selection among options for fusion testing facilities. The technical basis for the values given in Table 16 are briefly summarized in the following subsections.

Table 16. FNT Requirements on Major Parameters for Testing in Fusion Facilities, with Emphasis on Testing Needs to Construct DEMO Blanket

Parameter	Value
Neutron Wall Load, MW/m^2	1-2
Plasma Mode of Operation	Steady State*
Minimum Continuous Operating Time, Weeks	1-2
Neutron Fluence ($MW{\bullet}y/m^2$) at Test Module	
Stage I: Initial Fusion "Break-in"	0.3
Stage II: Concept Performance Verification	1-3
Stage III: Component Engineering Development and Reliability Growth	4-6
Total Neutron Fluence for Test Device, $MW{\bullet}y/m^2$	>6
Total Test Area, m^2	>10

* if steady state is unattainable, the alternative is long plasma burn with plasma duty cycle > 80%.

5.2.1 Neutron Wall Load

The minimum acceptable neutron wall load is derived from two factors: 1) engineering scaling considerations, and 2) tradeoffs 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 (~ 4-5 MW/m^2), engineering scaling considerations [2, 3, 27] are crucial. Useful testing at 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 overall geometry that is much less representative than a real DEMO/commercial blanket. It is found that engineering scaling techniques are useful, particularly in simulating individual effects. However, two important conclusions are reached: 1) 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, 2) the confidence in 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 more than 2-3 relative to that in DEMO/commercial power plants. Therefore, a neutron wall load of 1-2 MW/m^2 is necessary in the fusion test facility. 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 on 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 about 6 MW•y/m^2 in 12 calendar years. Table 17 shows the relationship between wall load and availability. The device availability required at wall load of 1 and 2 MW/m^2 is 50% and 25%, respectively. The present ITER-EDA design [29] plans on achieving <10% availability. Consequently, a higher wall load is needed. However, since the fusion device size and cost increases with wall load, improvements in achievable device availability are also necessary.

5.2.2 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:

- Time required to perform basic and multiple effects 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, thermomechanical interations.
- Time required to observe integrated behavior past the beginning of life and during periods of significant radiation-induced changes in material properties and component behavior.
- 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.
- Time required to obtain data on failure modes, effects and rates.
- Time required to perform the three stages of 1) initial fusion "break-in", 2) concept verification, and 3) component engineering development and reliability growth tests. The reliability growth testing phase is the most demanding on fluence requirements.

Table 17. Neutron Wall Load and Availability Required to Reach 6 MW•y/m^2 Goal Fluence in 12 Calendar Years

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

* 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.

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:

$$I_d = P_{nw} \bullet A_d \bullet t_d$$

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 (years)

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

$$I_m = P_{nw} \bullet A_m \bullet t_m \bullet T$$

where

P_{nw} = same as above

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 expressions above 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 ($MW \bullet y/m^2$ for the time-integrated wall load, I, and neutron/m^2 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 1-2 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 effects tests at the beginning of life. Examples include thermomechanical and tritium release tests. Rapid changes occur in the beginning of life under irradiation in the range of 0-0.3 $MW \bullet y/m^2$ (beyond this fluence, important changes still occur, but at a slower rate). This is one reason for selecting 0.3 MW•y/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 about 1-2 weeks. Therefore, each test campaign must be performed with continuous machine operation (100% availability) for ~ 1-2 weeks. About 10 test campaigns are needed to perform tests under different conditions (temperature, flow rates, chemistry, etc.) to fully explore relevant phenomena and submodule behavior. Assuming P_{nw} in the fusion testing facility to be ~ 1-2 MW/m^2, the initial fusion "break-in" phase requires a fluence in the range of 0.2-0.7 MW•y/m^2. Consequently, the 0.3 MW•y/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 the beginning of life 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 is 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 18 presents a summary of expected radiation induced effects in blankets in the 0 to 3 MW•y/m^2 fluence. Changes in mechanical properties of structural materials start to saturate around 2 MW•y/m^2. During the concept verification stage, it is not necessary nor practical to test components to their design end of life. However, it is desirable to test for a sufficiently long time, e.g. one third to one half the projected life in order to provide confidence in concept selection. Therefore, a fluence of 1-3 MW•y/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 2 or 3.

Table 18. Summary of Expected Radiation-Induced Effects on Blankets in the 0-3 MW•y/m^2 Fluence Range (long-term radiation effects not included)

- **0-0.1 MW•y/m^2** (at test module)
  Some changes in thermophysical properties of non-metals occur below 0.1 MW•y/m^2 (e.g., thermal conductivity)

- **0.1-1 MW•y/m^2** (at test module)
  Several important effects become activated in the range of 0.1-1 MW•y/m^2

  - Radiation creep relaxation
  - Solid breeder sintering and cracking
  - possible onset of breeder/multiplier swelling
  - He embrittlement
  - Changes in DBTT

  Correlation of materials data with fission reactors can probably be done with about 1 MW•y/m^2

- **1-3 MW•y/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
  - He embrittlement
  - Breeder and burnup effects
  - Breeder and multiplier swelling
  - Breeder/clad interactions

The required fluence during the CEDAR stage can be derived in several ways:

A. 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 (see Appendix B). This corresponds in fusion to 24 m^2 blanket test area for ~4 MW•y/m^2 (assuming 1200 m^2 first wall area and 10 MW•y/m^2 lifetime in DEMO.

However, the reliability of the fusion blanket modules must be higher than that of fuel pins in fission reactors because: 1) the fission reactor can tolerate a rather large number of defective fuel pins (e.g. water purification systems in PWR are designed to cope with 1%

defective fuel rods), and 2) in contrast, the fusion reactor has to be shutdown 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 (MTTR) has been estimated to be at least one month for a machine designed for a fast blanket exchange. Therefore, the required mean time between failures (MTBF) of blanket modules has to 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 years 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.

B. 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. 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 not 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 fusion nuclear technology because: 1) the mean time to recover from a failure is relatively long, 2) the surface area of the first wall is relatively large, and 3) 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 alternatively, the mean time between failures (MTBF) be very long. Therefore, reliability growth and demonstration testing is extremely important for blanket development.

The results on 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 DEMO reactor availability of 60% which implies blanket system availability of >90% requires >20 MW•y/m^2 testing fluence. Such a high testing fluence is practically unattainable because 1) it greatly exceeds the estimated lifetime expected for any blanket to be developed in the time frame of interest, and 2) it cannot be achieved in a reasonable time with a fusion testing device that has 1-2 MW•y/m^2 wall load and 30% availability. The results also show that benefits increase with neutron fluence at a relatively high rate up to a testing fluence of ~5 MW•y/m^2. Beyond this fluence, the rate of increase in benefits becomes much slower. Therefore, we have selected ~4-6 MW•y/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. With 4-6 MW•y/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.

It is noted 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 non-fusion 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 non-fusion facilities. However, the concept of "enhanced testing" needs to be addressed in future studies.

Considering the fluence requirements of the three stages of testing, i.e. initial fusion “break-in” (0.3 MW•y/m^2), concept performance verification (1-3 MW•y/m^2), and reliability growth (4-6 MW•y/m^2), the total fluence required for FNT testing is > 6 MW•y/m^2.

The minimum surface area at the first wall for a test module is ~0.36 m^2 (60 cm x 60 cm) based on engineering scaling considerations. Some blanket concepts, e.g. those with self-cooled liquid metal breeders or ceramic matrix composite structure might require a larger test module area. Assuming 2 to 3 blanket concepts to be tested in parallel during the reliability testing stage and 12 test modules per concept, the total testing area required at the first wall is >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, 4 to 6 concepts may be tested but the number of modules per concept can be only 4-5.

5.2.3 Plasma Cycle Parameters and COT

There are two areas of time-related parameters that have major impact on testing. The first is the plasma mode of operation, specifically the plasma burn time and dwell time. 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 [8, 30, 31] and has a large negative impact on reactor component reliability and failure rate. Therefore, steady state plasma operation is desirable for FNT testing in order to simulate well the DEMO reactor environment. However, devices such as ITER are based on pulsed plasma mode of operation. We have 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 which are not representative of equilibrium conditions, and,
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 Table 19 and 20. 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 operation (the number of cycles required to reach quasi-equilibrium is about 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 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 non-dimensional response normalized to the equilibrium value and τ_c the characteristic time constant. The allowable variation in a response during a specific test should not be no greater than 5% because small changes in some fundamental quantities result in large changes in important phenomena, i.e. 5% change in 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 aforementioned equations suggest 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 seconds for a front zone of solid breeder blanket designs in order to maintain the temperature variation to within 5% of equilibrium value. Moreover, the goal of a test is not just reaching equilibrium but it is to stay at equilibrium long enough to observe behavior. This has led to a consideration of burn time requirements approaching to $3\tau_c$.

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

Table 19. Characteristic Time Constants in Solid Breeder Blankets

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

Table 20. Characteristic Time Constants in Liquid Metal Breeder Blankets

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

Tables 19 and 20 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 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=1200s with plasma start up and shutdown times of 50s and 100s, respectively [32]. The figure 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 interpretation of tritium release very difficult and could lead to occurrence of phenomena not otherwise accessible in steady state operation.

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 continuous operating time (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 ~1-2 weeks. Based on the time constants shown earlier, 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 in order to achieve equilibrium for the most important processes.

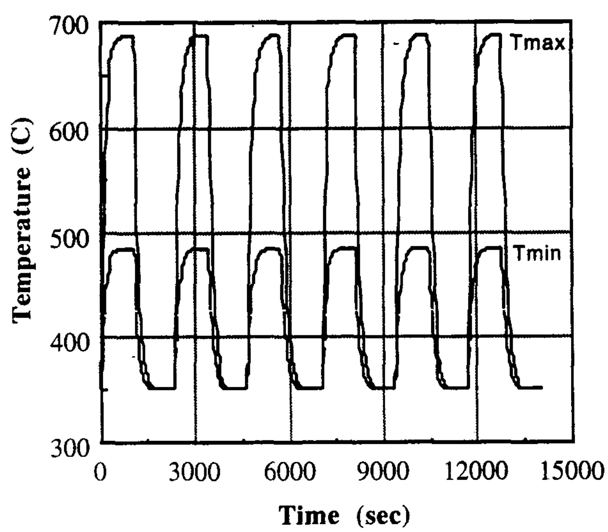

Figure 4. Scaled-up Li_2O breeder temperature response to 1 MW/m^2 pulsed wall load (blanket front position, q'''=9.4 MW/m^3), Burn time = 1000 seconds, Dwell time = 1200 seconds

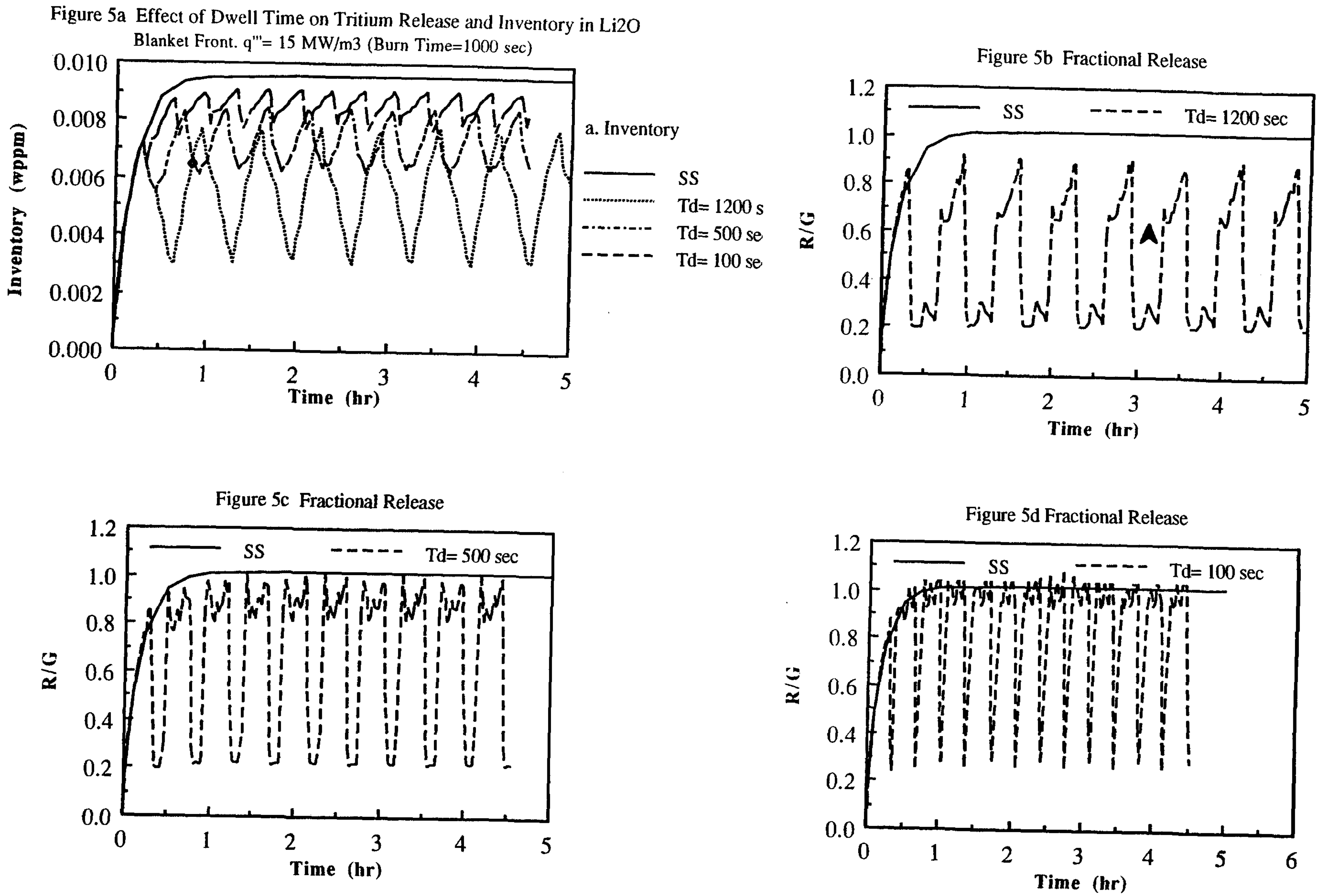

Figure 5. Effect of dwell time of tritium release and inventory in Li_2O

6. Need for VNS and Definition of Objectives and Design Guidelines

The preceding sections have clearly shown that testing in non-fusion facilities, albeit useful, can not resolve the critical issues for fusion nuclear technology. 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 16 and Fig. 3 summarized the FNT primary requirements on the major parameters for testing in fusion facilities. The key requirements are: 1-2 MW/m^2 neutron wall load, steady state plasma operation, many periods of continuous operation (100% availability) with each period 1-2 weeks, at least 6 $MW•y/m^2$ of neutron fluence, and >10m^2 of test area at the first wall.

6.1 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 Engineering Design Activity (EDA) phase, it is prudent to examine first whether ITER can satisfy the FNT testing needs. Parameters of ITER [29] are compared to those of present devices TFTR and DEMO in Table 21.

Table 21. 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 - 3
Plasma burn length, s	1	1000	Steady state (or hours)
Plasma dwell time, s	Very long	1200	0 (or < 100 s)
Fuel cycle	None	Partial (fuel consumer)	Complete, Self Sufficient
Thermal conversion efficiency	0	0	> 30%
Net plant availability	< 1%	1 % - 10%	> 50%
Fluence, $MW•y/m^2$	~ 10^{-4}	0.1 BPP* 1.0 EPP†	10 - 20

*BPP= Basic Performance Phase †EPP= Extended Performance Phase

Table 22 summarizes the major R&D tasks to be accomplished prior to DEMO: 1) plasma performance, 2) system integration, 3) plasma support systems, and 4) materials and FNT components performance and reliability and change out cycle. ITER as designed in EDA [29] will accomplish tasks 1, 2 and 3 with the possible exception of non-inductive 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 16 to the ITER parameters listed in Table 21. The primary reasons ITER can not 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

Table 22. 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) **Fusion Nuclear Technology Components and Materials**
   [Blanket, First Wall, High Performance Divertors]
   - Materials 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

ITER will address most of 1, 2 and 3

Fusion Nuclear Technology (FNT) components and materials development require dedicated fusion-relevant facilities parallel to ITER

As shown in Sec. 5, FNT testing requires steady state plasma operation; and if this can not be realized, the plasma duty cycle must be > 80%. From Table 21, ITER has a burn length of 1000s, dwell time of 1200s and a plasma duty cycle of ~ 45%. Therefore, based on the analysis in Sec. 5, 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•y/m^2 during 12 years of a Basic Performance Phase (BPP) and 1 MW•y/m^2 during an additional 12 year Extended Performance Phase (EPP). Therefore, ITER fluence is 1.1 MW•y/m^2 compared to ~6 MW•y/m^2 required for FNT testing (see Table 16). Consequently, ITER alone can not provide a database sufficient enough for 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 the next section.

FNT requires many (~100) periods of Continuous Operation Time (COT), i.e. at 100% availability, each period is 1-2 weeks. In ITER, the 0.1 MW•y/m^2 during the 12 year of BPP means that the total operating time is less than 5 weeks, i.e. only about <u>3 full power days per year.</u>

The next section will quantify the technological risks to DEMO from relying only on the database from ITER. We will also quantify the impact of long time delays in schedule to DEMO. The main point here is that ITER alone can not provide a database sufficient to construct FNT components for DEMO.

6.2 HVPNS Mission, Objectives, and Design Guidelines

Results in the previous sections strongly indicate that there is a definite need for a fusion facility to test, develop and qualify fusion nuclear technology components and material combinations for DEMO. We will call such a facility the High Volume Plasma-based Neutron Source (HVPNS). Such a facility must be a fusion facility to provide prototypical environment and since plasma-based neutron sources are the only ones capable of providing neutrons in an appropriate test volume as discussed in Sec. 4. We will occasionally abbreviate HVPNS as VNS

The HVPNS mission is to complement ITER as a dedicated fusion facility to test, develop and qualify fusion nuclear technology components and materials 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 testing strategy for VNS can be defined as follows:

- Stage I: Initial Fusion "Break-In"
  - Initial exploration of performance in the fusion environment
  - Calibrate non-fusion tests against performance in the fusion environment
  - Observe effects of rapid changes in properties in early life
  - Initial check on codes and data
  - Test and develop experimental techniques and instrumentation
  - Narrow material combinations and design concepts in the fusion environment

- Stage II: Concept Performance Verification
  - Verify performance beyond beginning of life and until changes in properties become small (changes in structure mechanical properties are substantial to ~2 MW•y/m^2)
  - Data on performance under normal operating conditions (temperature, stress, pressure drop, etc.) and under off-normal conditions (e.g. plasma disruption)
  - Data on initial failure modes and effects
  - Establish engineering feasibility of blankets (up to ~10 to 20% of lifetime)
  - Select 2 or 3 concepts for further development

- Stage III: Component Engineering Development and Reliability Growth
  - Identify failure modes and effects
  - Iterative design/test/fix programs aimed at improving reliability and safety
  - Failure rate data: Obtain a data base sufficient to predict mean time between failure (MTBF) with sufficient confidence.
  - Obtain data to predict mean time to replace (MTTR) for both planned outage and random failure
  - Obtain a data base 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. From Table 16, it is clear that VNS must have the following parameters in order to meet FNT development requirements:

1. Neutron wall load: 1-2 MW/m^2

2. Steady state plasma operation
3. COT of 1-2 weeks
4. Total Neutron Fluence of ≥6 MW•y/m^2 (0.3, 1-3, and 4-6 MW•y/m^2 for Stages I, II, and III, respectively)
5. Total test area at the first wall of >10m^2

One observation that can be made here is that FNT testing requires ~10m^2 of test area at 1-2 MW/m^2 neutron wall load, i.e. total fusion power of only about 20 MW. In contrast, plasma ignition in tokamaks requires >1500 MW of fusion power. In Table 23, the plasma ignition physics in tokamaks and FNT testing requirements are compared. Plasma ignition physics requires ~1500 MW fusion power with total integrated burn time of ~15 days. The tritium consumption, and hence the tritium supply requirement, for ignition physics is only about ~3.5 kg. In contrast, FNT testing requires only about 20 MW of fusion power but a long test time of about 5 full power years. Because of the low fusion power, the tritium supply required for 5 FPY of FNT testing remains modest, ~5.6 kg. If one combines the missions of: A) plasma ignition testing and B) FNT testing in one facility, this leads to combining the large power requirements of A) with the long test time of B), and therefore the tritium supply requirement becomes very large, about 420 kg. To put the magnitude of this tritium supply in perspective, consider the cost. At today's price of $20 million per 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 USA and Russia, and since the half-life for tritium radioactive decay is only 12.3 years, it is reasonable to deduce that no supply will be available from such a source in the 2006 to 2020 time frame. The only known supply is from operation of heavy water moderated CANDU reactors in Canada. This supply is estimated [33] 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 data base. The technical logic in such a scenario is flawed.

Table 23. Comparison of Physics and Nuclear Technology Requirements for Testing and Impact on Required Tritium Supply

Scenario	Fusion Power†	Integrated Burn Time†	Tritium Consumption
A. Separate Facility for Plasma Ignition	1500 MW	15 days	3.5 kg
B. Separate Facility for Fusion Nuclear Technology	20 MW	5 years	5.6 kg
Combined * (A + B) in one facility	1500 MW	5 years	420 kg

† Physics and FNT requirements are very dissimilar

* Combining large power and high fluence leads to large tritium consumption requirements

The above discussion leads to the following points:

1. Although we derived the need for VNS from detailed examination of FNT technical issues and evaluation of facilities' capabilities, there is another way to arrive at the need for VNS. This is based on 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 work [1,2,34] 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 less than 150 MW to keep the annual tritium consumption to ≤ 2 kg/yr., assuming 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. In order to limit the fusion power in VNS to 150 MW or less, the plasma in VNS must be in a driven mode with Q ~ 1-3 (Q = fusion power output/drive power in).

Designing for maintainability and high availability is both an objective and a requirement on VNS. To achieve the required testing fluence of ~6 MW•y/m^2 in 12 years with wall loads in the range of 1.5-2 MW/m^2, the device availability must be in the range of 25 to 30%

(see Table 17). 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 in order to reduce the device downtime.

Table 24 summarizes the ground rules suggested for evolving VNS design concepts.

6.3 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 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 ~0.5-0.75 m^2. Thus, a single GDT can not provide the surface area required for FNT testing (>10 m^2; see Table 16). 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 ~1-3, it can be shown that a Tokamak with TFTR/JET type of devices supplemented by non-inductive current drive and a divertor can satisfy FNT requirements and provide a VNS at a relatively low cost.

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

Table 24. Guidelines for Evolving HVPNS Design

HVPNS Mission

To serve as a test facility for fusion nuclear technology and to provide a database sufficient to construct FNT components for DEMO.

Testing Requirements

HVPNS must satisfy the following FNT testing requirements:

Wall Load:	1-2 MW/m^2
Neutron Fluence:	≥ 6 $MW{\bullet}y/m^2$
Plasma Mode of Operation:	steady state, or long plasma burn with duty cycle > 80%
Minimum Test Area per Test Article:	0.5 m^2
Total Test Area: (however, test devices that can satisfy part of the total testing area requirements should be considered in a cost/benefit/risk analysis)	> $10m^2$ (up to ~ $20m^2$)
Device Availability:	> 25%
Minimum Continuous Operating Time: (periods with 100% availability)	1-2 weeks
Magnetic Field at the Test Region:	> 2T

Design Features/Constraints

HVPNS design should be consistent with the following features/constraints:

- Configuration, remote maintenance and other design features must emphasize 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., less than 25% of that for ITER).
- The power consumption of the HVPNS site (e.g., from normal copper coils, current drive, etc.) should be kept reasonable low (e.g., < 700 MW).

7. 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 on 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 on these scenarios are possible. However, we used experience and knowledge to limit the scenarios to the ones essential to understanding the impact of adding or eliminating VNS. The scenarios are as follows.

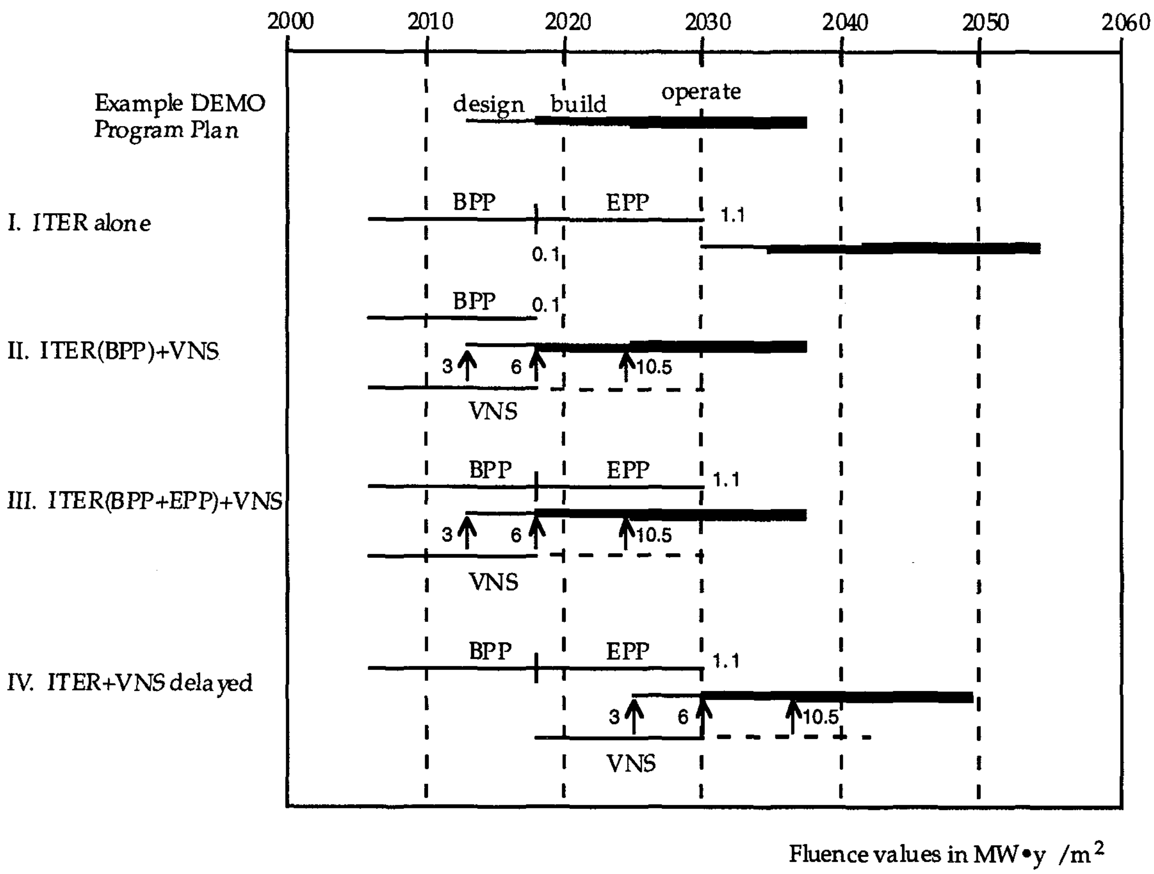

Figure 6. Scenarios for major fusion devices leading to a DEMO

7.1 Fusion Facilities 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 years. We did not explicitly account for the ITER downtime between BPP and EPP. If this period is 2 years, for example, the EPP duration can be shortened to 10 years at the same total fluence with little impact on the comparative resulted derived here.

Scenario II: ITER (BPP only) + VNS

In this scenario VNS operates in 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). VNS operation can also extend beyond the initial 12 years.

Scenario IV: ITER + VNS Delayed

In this scenario ITER operates for the two phases BPP and EPP as in scenarios I and III. However, 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 $MW\bullet y/m^2$ during BPP, and 3 $MW\bullet y/m^2$ during EPP.

In addition to the above scenarios, another interesting and promising scenario will be discussed later. In this scenario, VNS is operated parallel to ITER BPP for 12 years, beyond which data from VNS is used to construct hot DEMO-type blanket on ITER, which makes ITER operation during the second phase "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:

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?

7.2 Time Schedule

The date for the beginning of fusion DEMO operation is important from technical and programmatic viewpoints. In a "roll backward" technical planning, the date of the DEMO is a key factor in determining the pace of R&D to DEMO. In a "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 a worldwide spending of ~ $1.2 B/year on fusion R&D, tens of years 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 years of construction and 3 years of final engineering design. For the example case, the final design begins the year 2015 and construction begins the year 2018. From results in the previous sections, ~ 3 MW•y/m^2 and 6 MW•y/m^2 of testing for DEMO nuclear components must be available by the beginning of final design, and at the beginning of construction, respectively.

The scenarios considered here (I-IV, and Ib), illustrated in Fig. 6, vary considerably in the achievable time schedule to DEMO. Scenario I, ITER alone, achieves only 1 MW•y/m^2 by the end of EPP, i.e., by the year 2030. The clear conclusion here is that ITER alone can not provide a sufficient data base 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 ITER alone scenario is the year 2042, as illustrated in Fig. 6. Therefore, the conclusion here is that ITER alone strategy results an unacceptably high level of risk, and even with such a risk the DEMO operation is delayed by 17 years.

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 25 provides for all scenarios considered here, the DEMO start date, 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 the next subsection.

Scenario III is the same as Scenario II, with VNS operating in parallel to ITER BPP, except Scenario III assumes that ITER will also continue operation into the EPP phase. From Fig. 6 and Table 25, it is clear that adding EPP to ITER has very little effect on the DEMO start date or the confidence level in 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 so much (~ \$500 M per year) and achieves little benefits, compared to VNS.

Scenario IV is similar to Scenario III except 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 ITER EPP phase into "Pre-DEMO" with full hot reactor relevant blankets.

7.3 Technical Risk

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 fission industry. Current fission technology is replete with examples of poorly characterized data sets, often generated with non-prototypical 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 [35].

Table 25. DEMO Start Date and Testing Fluence Achieved for Various Scenarios

Scenario	DEMO Operation Start Date	Fluence at Start of DEMO Construction ($MW\text{-}yr/m^2$)	Fluence at Start of DEMO Operation ($MW\text{-}yr/m^2$)	Confidence Level in DEMO*
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

* Confidence level is quantified under Technical Risk in the next subsection.

As discussed previously, one of the important requirements set by industry and utility for DEMO is demonstration of dependability and reliability. As discussed in Sec. 2, 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):

$$\text{DEMO Reactor Availability} = A_R = \frac{1}{1 + \sum_i (\text{outage risk})_i}$$

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

$$\text{outage risk}_i = \text{MTTR}_i \text{ x failure rate}_i$$

$$= \text{MTTR}_i/\text{MTBF}_i$$

where MTTR_i is the mean down time to recover from a failure in component i and MTBF_i is the mean time between failures for component i.

To achieve a low outage rate operation requires that a high reliability of 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 mean time between failure (MTBF). MTBF and MTTR are the parameters which directly affect the percentage of time that a system is available for use. Notice that a number of combinations may be possible for achieving 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 B, MTTR for tokamaks is predicted to be long, which necessitates that MTBF be long to achieve the desired availability.

We can not be assured with a high degree of confidence that the reliability of the blanket concept selected will be adequate for DEMO. Indeed, it is clear at the outset that 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.1, 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. The Possion model might not really describe the failure behavior of the fusion nuclear technology components (in particular, material properties changes with time due to irradiation effects), it is widely used, and often 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 close enough. Figure 7 shows the upper statistical confidence level as a function of the test time in MTBF multiples and the number of failures which have occurred during the test [24]. 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 about 30 percent. 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 component is higher than its required MTBF.

7.3.1 Results of Approach I

The results for Approach I are summarized in Table 26 and Figures 8 and 9 (with the computational method described in Appendix A.3). 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 $MW{\bullet}y/m^2$ 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 results show that testing in the ITER alone scenario could only confirm with 80% confidence level for achievement of a DEMO reactor availability of ~ 7.1% for MTTR = 1 week by the year 2025. This reduces to about 1.8% if MTTR equals to 1 month . In contrast, with scenario II (i.e. VNS operating parallel to ITER BPP), it is possible to confirm with 80% confidence level the achievable DEMO reactor availability of about 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 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.

7.3.2 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 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 ITER alone strategy. However, notice that confirming a 80% confidence

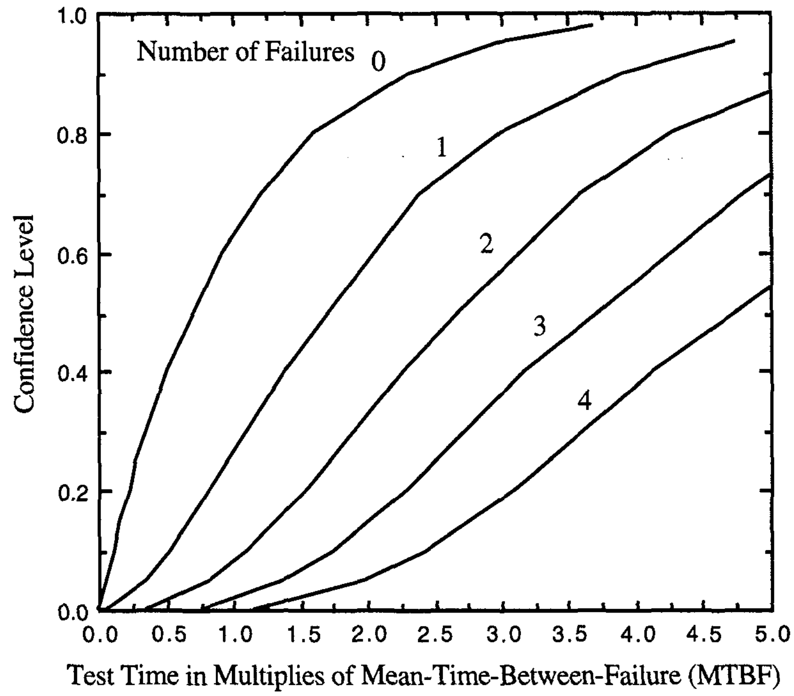

Figure 7. Upper statistical confidence level as a function of test time in multiples of MTBF for time terminated reliability tests (Poisson distribution). Results are given for different numbers of failures.

Table 26. Summary of DEMO Reactor Availability (%) Obtainable with 80% Confidence vs. Calendar Year in the Various Scenarios

Scenario	MTTR = 1 week			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

12 test modules, 1 failure during the test, experience factor = 0.8
MTTR = meant time to repair

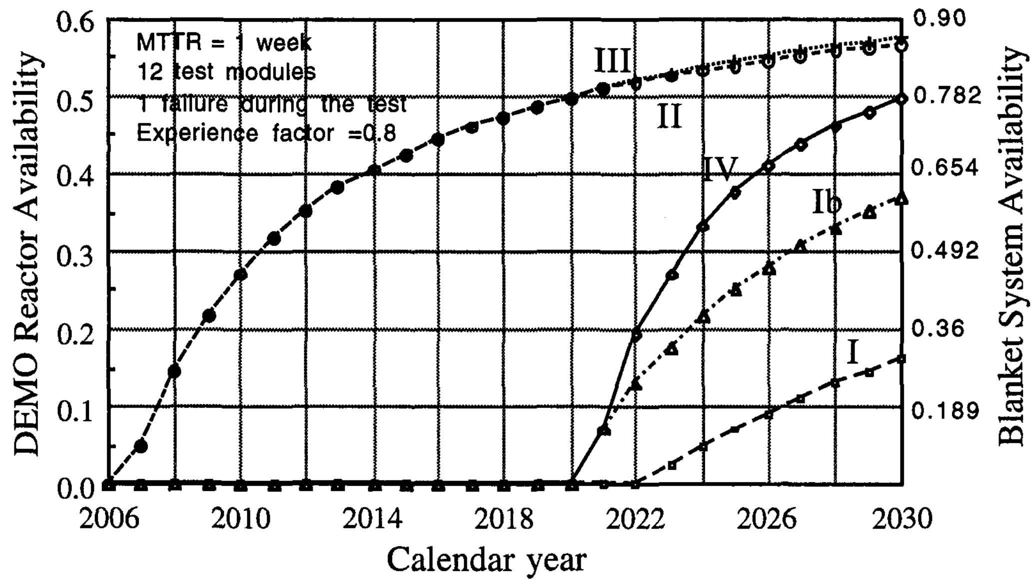

Figure 8. DEMO reactor availabilities obtainable with 80% confidence for different testing scenarios, MTTR = 1 week (I: ITER only; II: ITER BPP +VNS; III: ITER +VNS; IV: ITER + delayed VNS; Ib: ITER only, high fluence)

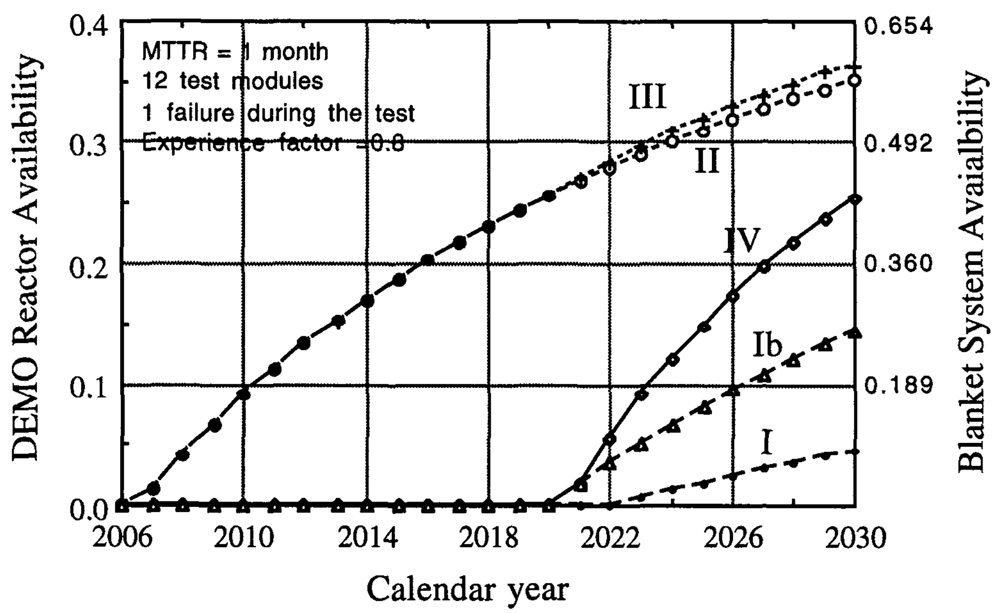

Figure 9. DEMO reactor availabilities obtainable with 80% confidence for different testing scenarios, MTTR = 1 month (I: ITER only; II: ITER BPP +VNS; III: ITER +VNS; IV: ITER + delayed VNS; Ib: ITER only, high fluence)

(or greater) in the achievable parameters is generally required for major and critical projects such as DEMO. In order to ascertain that the risk associated with the achievable reactor availability goal can be acceptable, reactor availabilities of 50% in addition to 30% are examined. Fig. 10 shows the confidence level in achieving DEMO reactor availability of 50% for different FNT testing scenarios, assuming 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 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 only in a confidence of 14 % for DEMO availability of 50% at the year 2025. If the more realistic vale of MTTR = 1 month is considered, the confidence level with 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 availability of 50% achievable by the year 2025, as shown in Table 27 and Figure 11. The increment in the confidence level by employing the ITER EPP nuclear testing is negligible.

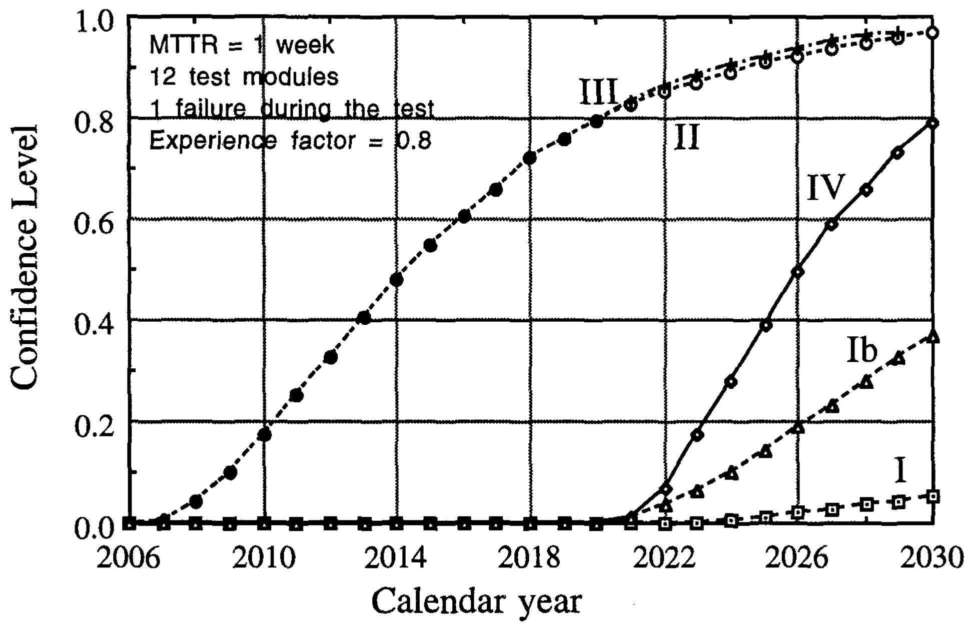

Figure 10. Confidence levels of confirming DEMO reactor availability of 50% for different testing scenarios, MTTR = 1 week (I: ITER only; II: ITER BPP +VNS; III: ITER +VNS; IV: ITER + delayed VNS; Ib: ITER only, high fluence)

Table 27 Summary of Confidence Level (%) Obtainable in DEMO Reactor Availability

Scenario	I: ITER alone		II: ITER (BPP) +VNS		III:ITER (BPP+ EPP) +VNS		IV: ITER + VNS delayed		Ib: ITER alone (high fluence)	
DEMO availability	30%	50%	30%	50%	30%	50%	30%	50%	30%	50%
MTTR = 1 w 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 w 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 m 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

1 failure during the test, experience factor =0.8
MTTR = mean time to repair
1 w = 1week
1 m = 1 month

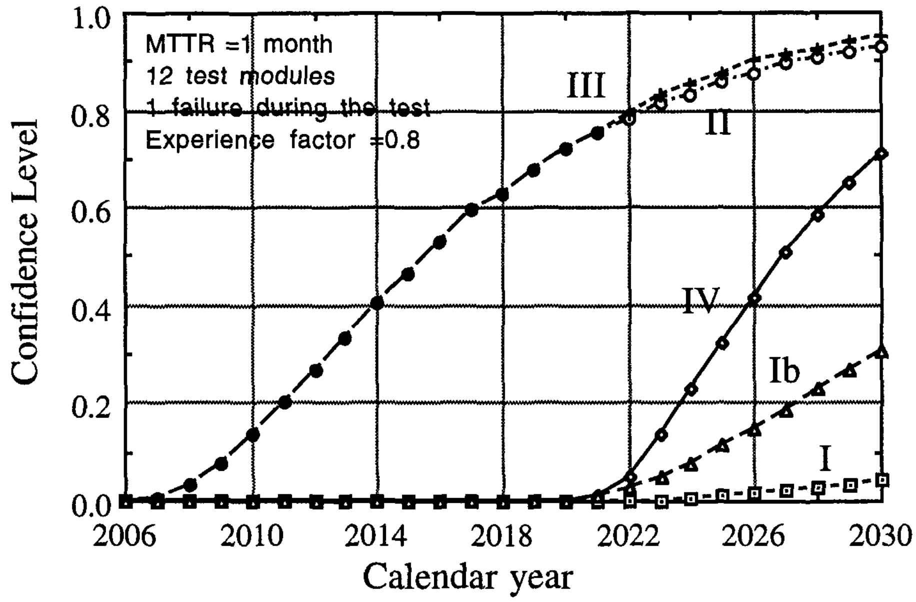

Figure 11. Confidence levels of confirming DEMO reactor availability of 30% for different testing scenarios, MTTR = 1 month (I: ITER only; II: ITER BPP +VNS; III: ITER +VNS; IV: ITER + delayed VNS; Ib: ITER only, high fluence)

7.3.3 "Pre-DEMO" Scenario

The above analyses suggests that the blanket (and other FNT components) tests in ITER alone can not 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."

This raises the possibility of "Pre-DEMO". A "Pre-DEMO" device would be a device between ITER and DEMO. If such a device were of the same type of ITER (size, fusion power, etc.), it will not improve the situation because of several reasons: 1) ITER would have exhausted the external tritium supply and there will be no tritium left to operate the Pre-DEMO device, 2) the testing information from ITER will not allow a full hot breeding blanket to be constructed with acceptable confidence in the Pre-DEMO device, and 3) a Pre-DEMO device will add a very substantial burden to the total cost of fusion R&D and will further delay the DEMO operation to the point that fusion will not play a role in world energy production in the 21st century.

Alternatively, 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 heat- and tritium extraction.

This scenario is illustrated in Fig. 12. If VNS operates parallel to ITER BPP and the results from ITER and VNS are good in all technical areas, then there will be sufficient data to construct full hot breeding blanket on ITER, which allows 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.

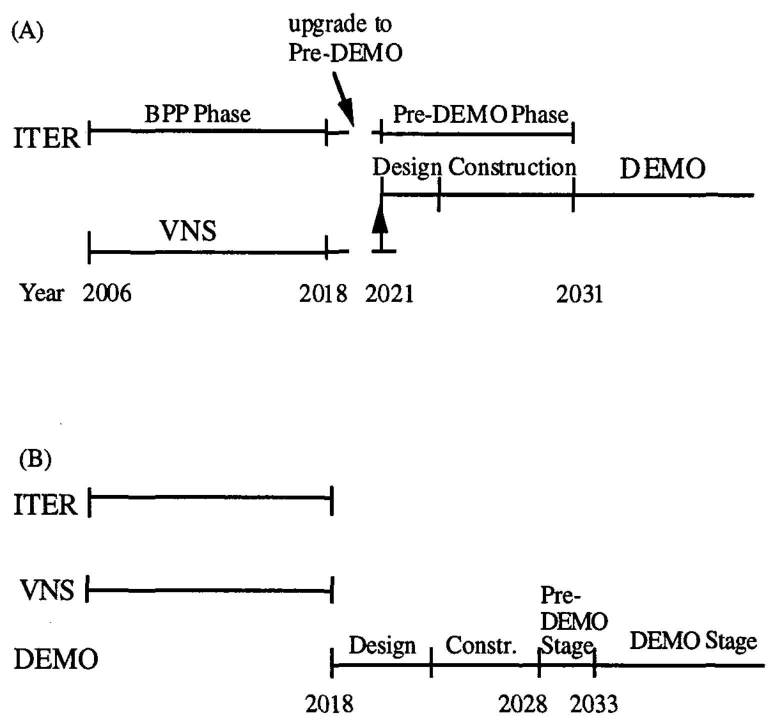

Figure 12. Promising scenario of (A) VNS parallel to ITER BPP and ITER second phase becomes a Pre-DEMO with full hot breeding blanket and (B) VNS parallel to ITER BPP and operating DEMO for five years as "Pre-DEMO".

As shown in Figure 13, an achievable DEMO reactor availability of 41% can be confirmed with 80% confidence after 10 years 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 DEMO neutron wall load of 3 MW/m^2.

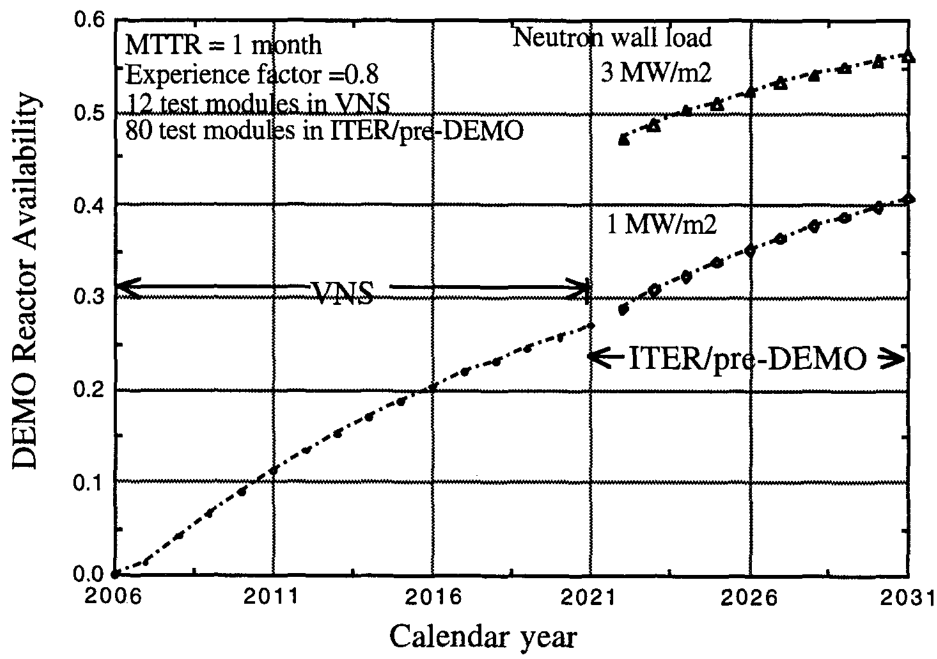

Figure 13. DEMO reactor availabilities achievable with 80% confidence with the scenario in which VNS operates parallel to ITER BPP, and the second phase of ITER is upgraded to Pre-DEMO using VNS testing data (Results are for MTTR =1 month).

7.3.4 Summary of Technical Risk

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

- Blanket tests in ITER can not demonstrate 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.
- 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.

- The contribution of blanket tests in the presently envisaged ITER EPP to the reliability testing is negligible.
- In optimizing system design, maintainability requirements (the length of MTTR) and maintainability design criteria are critical. If MTTR is greater than 1 week, the reliability improvement in blanket components (increase in MTBF) must be more substantial.
- With ITER alone strategy, the problem of high technical risk to DEMO can not even be credibly resolved by assuming another Pre-DEMO device between ITER and the DEMO.
- Operating VNS parallel to ITER BPP makes it possible to envision a credible scenario in which the second phase of ITER is upgraded to "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.

7.4 Costs

All of the above 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 which 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: 1) the total cost of fusion R&D from now until DEMO, and 2) 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 28 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 ~ \$8B 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, 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%. ITER has an estimated operating cost of ~\$400 M/yr. Relative to this, we estimate VNS operating cost to be ~\$200 M/yr including the power consumption cost. The tritium supply cost is calculated at \$20M/kg.

Table 28. Costing Assumptions for Scenario Evaluations

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

7.4.1 Total R&D Cost

The results in Table 29 show that the total capital cost, operating and tritium supply costs for 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 year reduction in the period from now to DEMO. At the present, the world expenditure on fusion R&D of $1.2 B, this shortening of time to DEMO made possible by VNS provides additional savings of ~ $20B. This cost saving 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 to 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 results in only the same confidence level as that achievable with VNS for a DEMO by the year 2025.

7.4.2 Near Term Cost

Another point on cost is whether constructing and operating VNS in 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: 1) ITER and VNS will be sited in two different countries, instead of in the same country, 2) 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 29. Total Cost to Start of DEMO Construction ($B)

	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

Table 30 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 $4.96B of which only $0.33 B, i.e. < 10%, is the additional burden due to VNS. The VNS host Y will pay a total cost of $2.96 B 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.

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.

Table 30. Construction and Annual Operating Costs by Party for ITER and VNS

	Total Cost	Cost to party that hosts ITER	Cost to party that hosts VNS	Cost to party with no site
Construction Cost ($B)				
ITER	8	4	1.33	1.33
VNS	2	0.33	1	0.33
IFMIF	0.8	0.13	0.13	0.13
Other	2	0.5	0.5	0.5
Total Construction Cost ($B)	*12.8*	*4.96*	*2.96*	*2.29*

8. Conclusions

FNT Issues and Testing Needs

- Fusion Nuclear Technology (FNT) development has many of the remaining feasibility and attractiveness issues for realizing fusion power. A serious R&D program with clear strategy and goals for FNT development must now be a high priority for the world's fusion energy development programs.

- 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.

- 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.

- Adequate performance verification and engineering development require prototypical test articles (e.g. materials, configurations, 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, vacuum) represent the major testing issues.

Role of Non-Fusion Facilities

- The feasibility of blanket concepts cannot be established prior to extensive testing in the fusion environment. None of the critical issues can be fully resolved by testing in non-fusion facilities. Non-neutron 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. However, non-fusion facilities can and should play a role in blanket R&D because of availability and low cost and in order to reduce the cost and risk of the more complex fusion experiments.

Fusion Testing Requirements

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

Blanket Failures and Demo Availability

- Availability analysis reveals critical concerns in fusion power development, some of which can be addressed by changes in blanket and machine design, but most must be addressed by extensive testing, in order 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 needs to be about 80%. The mean time to replace (MTTR) or recover from a failure and the mean time between failures (MTBF) are the parameters which directly affect availability. For MTTR = 3 months, the blanket MTBF must be greater than 1.0 FPY, i.e., only one failure anywhere in the blanket is allowed about every one year of operation. For a blanket that has 80 modules, the corresponding MTBF per module is 80 FPY. These are very ambitious goals. Experience from non-fusion technologies shows that achieving such long MTBF's requires very extensive testing and development.

- Some of the important conclusions regarding failure modes, failure rates and reliability growth testing are:

- a) Capability to replace first wall/blanket 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, total tube length).
- c) A serious reliability / availability analysis must be an integral part of the design process.

- d) R&D programs must be based on quantitative goals for reliability (type and number of tests, test duration, prototypicality).
- e) Reliability growth testing in fusion devices will be the most demanding, particularly on number of tests and time duration of tests (> 10 m^2 and ~ 6 MW•y/m^2 for blankets).
- f) Reliability testing should include: 1) identification of failure modes and effects, 2) aggressive iterative design/test/analyze/fix programs aimed at improving reliability, and 3) obtaining failure rate data sufficient to predict MTBF.

ITER Alone Scenario

- ITER alone can not satisfy the FNT fusion testing requirements because of 1) pulsed operation with low duty cycle, 2) low fluence, 3) short continuous operating time, 4) low device availability, and 5) small number of blanket testing ports.

- For the presently envisaged ITER strategy based on an Extended Performance Phase with a fluence of 1 MW•y/m^2, blanket tests in ITER alone cannot demonstrate a blanket system availability in DEMO higher than 4%.

- 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.

Scenarios with HVPNS

- A DEMO availability of > 30% can be demonstrated by adding blanket tests in a HVPNS characterized by the following parameters:
  - average neutron wall load 1-2 MW/m^2
  - maximum neutron fluence ≥ 6MW•y/m^2
  - testing space at the first wall ≥ 10m^2
  - device availability > 25%

- Presentations made during the Phase 1 effort on candidate HVPNS concepts show that an attractive design envelope for HVPNS exists. A small size (R < 2m) tokamak with normal conducting TF coils and driven (Q ~ 2-3) steady state plasma meets the FNT testing requirements with capital cost expected to be < 25% that of ITER. (Design of HVPNS was outside the scope of Phase I. Presentations were made by volunteers from USA, EU and RF. The study participants did not address the specifics of any design. See Recommendations.)

- An effective path to fusion DEMO involves two parallel fusion facilities: 1) ITER to provide data on plasma performance, plasma support technology, and system integration, and 2) HVPNS to test, develop and qualify fusion nuclear components and materials combinations and to demonstrate acceptable MTTR for DEMO.

- A testing strategy employing such a HVPNS would decisively reduce the very high risk of unsuccessful blanket operation in DEMO, and would make it possible - if operated in parallel to ITER Basic Performance Phase - to meet the goal of DEMO operation by the year 2025.

- 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
    - testing large blanket modules up to the size of a segment at low fluence

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

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

- A scenario with HVPNS parallel to ITER provides cost savings in the overall R&D towards DEMO compared to 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.

9. Recommendations

9.1 Overall Recommendation

Blanket tests in ITER alone can not demonstrate a DEMO availability higher than 4%. A fusion test facility of the HVPNS type is required to test and develop blankets for DEMO and to demonstrate the performance and availability levels required prior to committing to the DEMO construction. Therefore, it is recommended that Phase 2 of the IEA-HVPNS study be initiated as soon as possible to explore design options for HVPNS. This Design Exploration Study should have the following objectives:

1. Investigate the range of parameters and design options and determine the design envelope for an attractive HVPNS.
2. Identify the technical issues and R&D for HVPNS.
3. Develop the key elements of FNT test program on HVPNS. Identify the design, safety, and operational issues associated with conducting such a test program.

The goal of the Design Exploration Study (DES) is to develop information sufficient to judge whether to proceed with a more detailed Conceptual Design Activity for HVPNS. The DES should be completed by the fall of 1996. The level of resources recommended for the DES is ~ 20 PMY, or higher.

9.2 Additional Guidelines for HVPNS Design Exploration Study

<u>HVPNS Mission</u>

To serve as a test facility for fusion nuclear technology and to provide a database sufficient to construct FNT components for DEMO.

<u>Testing Requirements</u>

HVPNS must satisfy the following FNT testing requirements:

Wall Load:	1-2 MW/m^2
Neutron Fluence:	$\geq$ 6 $MW{\bullet}y/m^2$
Plasma Mode of Operation:	steady state, or long plasma burn with duty cycle > 80%
Minimum Test Area per Test Article:	0.5 m^2
Total Test Area:	> $10m^2$ (up to ~ $20m^2$)

(however, test devices that can satisfy part of the total testing area requirements should be considered in a cost/benefit/risk analysis)

Device Availability:	> 25%
Minimum Continuous Operating Time: (periods with 100% availability)	1-2 weeks
Magnetic Field at the Test Region:	> 2T

Design Features/Constraints

HVPNS design should be consistent with the following features/constraints:

- Configuration, remote maintenance and other design features must emphasize 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. less than 25% of that for ITER).

- The power consumption of the HVPNS site (e.g. from normal copper coils, current drive, etc.) should be kept reasonable low (e.g. < 700 MW).

Figures of Merit

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

- extent of meeting FNT requirements (wall load, fluence, test area, etc.)
- total capital and operating costs
- contribution to nuclear testing for DEMO components
- additional contributions to satisfying DEMO database requirements other than testing
- minimal R&D to construct HVPNS

- confidence in achieving HVPNS goals
- contributions to ITER (e.g. reduced technological burden and possible cost savings)
- contributions to improvements in the development schedule to DEMO

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Appendix A: Failures and Reliability Testing in Fusion Facilities

One of the most serious concerns in the engineering development of a component, particularly for 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
- Random occurrence despite available knowledge and experience

Experience from other technologies show [A-1,A-2] that the failure rate, λ, during the lifetime of a component for 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 it reaches a "steady state" value, λ_b, at the "bottom of the bathtub." 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 fusion nuclear technology development is the value of λ_b for the blanket; what is the goal value for λ_b and how to achieve it through testing. Experience shows that the value of λ_b for 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 mean time between failure ($MTBF = \frac{1}{\lambda}$) 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 demonstration of satisfactory performance [A-3-A-5]. While the component lifetime is mainly determined by the fluence limitation (i.e. damage level) which leads to performance degradation, the mean time between failures (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.

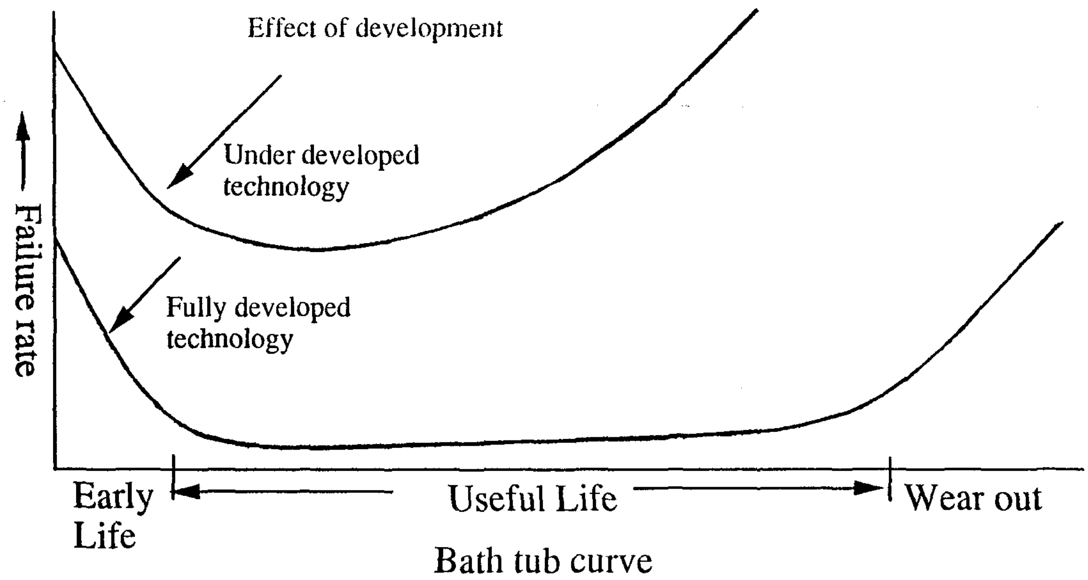

Figure A-1. Failure rate characteristics for developed and under-developed technologies

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 mean time between failure (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.1 Goal MTBF (and MTTR) for DEMO Blanket

The availability allocation among components of a fusion reactor system so that a target availability could be achieved have been performed in INTOR [A-6], STARFIRE[A-7], and NET[A-8]. These studies have shown that blanket system availabilties of about 97% and about 99% are required to meet reactor availability goals of 60% [A-8] and 77% [A-6], respectively. Notice that these values are estimated based largely on the expert opinion and the data obtained from the experience of non-fusion 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 in order 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:

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

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, it is further assumed that the reactor availability goal is determined by the six major components (i.e., blanket/first wall 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 first wall/blanket system availability of 90% (or first wall/blanket system outage risk of 0.111) for achieving a DEMO reactor availability of 60%. The impact of lack of fusion nuclear technology component development is then evaluated by the higher outage risks and consequently by the lower reactor availabilities, as shown in Table A-1. In this case, a blanket system availability of 49% is projected in order to achieve a DEMO reactor availability of 30%.

The blanket system availability goals, such as those shown in Table A-1, 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:

1. the failure of any blanket module causes failure of the blanket system, and
2. failure of any blanket module is entirely independent of the failure of any other blanket module.

Table A-1 Requirements on Blanket System Availability as a Function of Reactor Availability

DEMO Reactor Availability	First Wall/Blanket System Availability
60%	90%
51.4%	80%
50%	78%
43.5%	70%
36%	60%
30%	49%
23%	30%
10%	20%
5%	10%

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:

$$A_{BS} = \frac{\text{uptime}}{\text{uptime} + \text{downtime}} = \frac{MTBF_{BS}}{MTBF_{BS} + MTTR_{BS}} = \frac{1}{1 + \lambda_{BS} MTTR_{BS}}$$

where $MTBF_{BS}$ is the mean time between failure 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 down time of the reactor to replace, or fix, the failed portion of the blanket system. In order 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:

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

where A_n is the blanket module availability defined as:

$$A_n = \frac{1}{1 + \lambda_n MTTR_n}$$

The target MTBF per module is shown in Table A-2 for MTTR values of 1 week, 2 weeks, 1 month and 2 months. For the 30% DEMO reactor availability, the MTBF (module) varies from 1.5 to 12.65 full power years for MTTR =1 week to MTTR = 2 months. For 60% DEMO reactor availability, the MTBF (module) is ~14 FPY for MTTR = 1 week and becomes ~119 FPY for MTTR = 2 months.

Table A-2. Required DEMO Blanket Module Mean Time Between Failure ($MTBF_n$) 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

The results in Table A-2 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 blanker module is much longer than the design life of the blanket (10-20 $MY.y/m^2$ which is ~ 3-7 FPY at P_{nw} ~ $3M/m^2$). 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 to 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 to the state of the art discussed in the next subsection. As also shown later, the testing requirements to demonstrate 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. 2, which assumes the DEMO will have two stages. The first has initial target availability of 30% and it reaches 60% only in the second stage.

One additional observation on the results of Table A-2 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 two months. 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, disconnects, removal, insertion, reconnect, etc.) In addition, when a module fails, one needs to identify the failure consequences (e.g. the distortion of module geometry) on the maintenance operation. There are also many safety related precautions and operations. Therefore, one week appears a low value for MTTR. However, values of 1-2 months have very serious impact on the required MTBF and achievable availability. The results here and in other sections 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 needs to be obtained from fusion test facilities.

A.2 Estimates of Failure Rates

Given the target MTBF values for blanket DEMO in the previous subsection, a key question is: what do we expect the failure rate to be based on current knowledge? Unfortunately, our current data base 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. [A-9] in assessing 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 [A-10], with first wall surface area of ~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-3 shows the estimated failure rates using data compiled by Bünde [A-9] 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-3. The estimated length and number of elements per blanket module are also given in the table. The overall failure rate per blanket module is estimated to be in the range of 7×10^{-6} to 1×10^{-4} per hour. Thus, the MTBF (module) is in the range of 1-16 years and the MTBF for the overall blanket systems 0.01-0.2 years, 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-4 presents a comparison of what is <u>expected</u> versus what is <u>required</u> for the blanket mean time between failure. The MTBF values are shown for the blanket module and the blanket

Table A. 3 Estimated Failure Rate for Typical Blanket Based on Data from Non-fusion Technologies. Failure rates given here do not include fusion-specific failure modes.

Blanket Element	No. or Length of Elements per Blanket Module	(Unit) Failure Rate[1]		Failure Rate per Blanket Module (1/h)	
		Mean	High	Mean	High
Longitudinal Welds	66 m	5.0e-8 /h-m	5.0 e-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)		$7x10^{-6}$ - $1x10^{-4}$			
Calculated MTBF per module (years)		1 - 16			
Calculated MTBF for blanket system (years)		0.01 - 0.2			

(1) R. Bünde, et. al., 16(1991) 59-72 (Fus. Eng. & Design 1991) [failure rates are based on experience from non-fusion technologies]

system, which consists of 80 modules. The expected MTBF is based on results in Table A-3, i.e. based on those failure modes and failure rates we know from the mature technologies of steam generators and fission reactors that are likely to exist in fusion DEMO blankets. The expected MTBF values in Table A-4 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-4 are those that must be achieved in order to meet certain availability goals for the blanket. We show required MTBF in Table A-4 for two cases of DEMO reactor availability: 30% and 60%. For each case, MTBF values are given for different values of the MTTR, i.e. the downtime to recover from a blanket failure.

The results in Table A-4 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 years for MTTR in the range of 1 week to 2 months for the case of DEMO reactor availability of 30%. These are within the range of expected values, which is 1-16 years. For the DEMO reactor availability goal of 60%, the MTBF per blanket module with the shortest time estimated for MTTR of 1 week falls in the range of expected values. However, the MTBF per blanket module increases to 60 and 119 years at MTTR = 1 month and 2 months, respectively. These values are much greater than the 1-16 year range of expected values. In other words, assuming a realistic time for MTTR of 1 month, the MTBF value required to achieve DEMO reactor availability of 60% is much longer than those expected to be achievable. This suggests that a blanket with low enough failure rate to achieve a DEMO reactor availability goal of 60% appears to be an ambitious goal.

It should be noted that the expected values derived here 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 can not 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 selection among blanket concepts.

Another serious concern is that the failure rates in Table A-3 account only for the very limited number of known failure modes. Very little work has been done to date to identify failure modes in first wall / blanket systems. Table A-5 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.

Table A-4. Comparison of Expected Blanket MTBF to that Required in DEMO

		MTBF (years)	
		Blanket Module	**Blanket System**
Expected	EXPECTED (for fully developed technology based on steam generators & fission reactors data)[a]	1-16	0.01-0.2
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= 1week	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

It is reasonable to ask whether the failure rate in fusion blanket systems can be expected to be lower or higher than steam generators and fission reactors. A quantitative answer is beyond the scope of this work but should be seriously addresses in the future, most importantly by generating a data base 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 steam generators and the core of fission reactors because of the following points:

- Larger numbers of subcomponents and interactions (tubes, welds, breeder, multiplier, coolant, structure, insulators, tritium recovery, etc.).
- More damaging, higher energy neutrons.

- Other environmental conditions: magnetic field, vacuum, tritium, etc. (for example, a leak from the first wall or blanket module walls into the vacuum system results in failure, while in steam generators and fission reactors, continued operation with leaks is often possible).
- Reactor components must penetrate each other; many penetrations have to be provided through the blanket for plasma heating, fueling, exhaust, etc.
- Ability to have redundancy inside the blanket / first wall system is practically impossible.

Table A-5. Some possible Failure Modes in Blanket/First Wall (for solid and liquid breeder blanket concepts)

• Cracking around a discontinuity/weld
• Crack on shutdown (with cooling)
• Breeder (solid) disintegrates/cracks
• 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 of coolant tubes
• Cracks in electrical connections between modules

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

- 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.
- 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, tube length).
- A serious reliability and availability analysis must be an integral part of the design process.
- R&D program must be based on quantitative goals for reliability (type of tests, prototypicality of test, number of tests, test duration).

- Reliability growth testing in fusion devices will be the most demanding (particularly on 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
  - Obtain failure rate data sufficient to predict MTBF

A.3 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 test for a pass decision or enough failures have occurred to make a fail decision. Based on a study of sequential test plans[A-11], the INTOR critical issues study concluded that the achievement of 80 percent confidence in a given mean time between failures (MTBF) in the constant failure rate regime of operation would typically require a cumulative test period of 3.5 times the MTBF[A-6]. 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 [A-12].) As shown in Figure 7 (see section 7), 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 [A-13]. The primary objectives here are, by utilizing this test plan, to determine the blanket test time and test area in fusion facilities which are required to meet certain goals for MTBF.

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

$$T = \frac{N\ t\ x\ N^{1-\alpha}}{\Phi_{NW}}$$

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[A-13]. 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-1 and A-2 that correlate reactor availability, blanket availability, MTBF, and MTTR. In all cases, we assumed the number of blanket modules in DEMO to be 80.

Fig. A.2 shows the DEMO reactor availability achievable with 80% confidence, and assuming one failure during the test, as a function of fluence on test modules. Results are shown for 2 cases of 6 and 12 test modules and for 2 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 below 40% even for a fluence of 10 MW•y/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. From Fig. A-2, it is clear that the achievable DEMO blanket availability, and hence the DEMO reactor availability, increases substantially with testing fluence. For MTTR = 1 week, increasing the testing fluence from 1 to 6 MW•y/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 MY•y/m^2 leads to reactor availability of only 5.6% with 12 test modules; but increasing the testing fluence to 6 MY•y/m^2 increases the DEMO reactor availability to 24%.

Notice that as the test fluence increases beyond ~6 MW•y/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•y/m^2. Since the blanket design lifetime may be limited to about 10 MW•y/m^2, testing will become difficult at such high fluence.

A number of key conclusions are important from the results here: 1) achieving a fluence of ~6 MW•y/m^2 at the test modules with ~6-12 test modules is crucial to achieving DEMO reactor availability in the 40% to 50% range with 80% confidence, 2) achieving DEMO reactor availability of 60% may not be possible with 80% confidence for any practical blanket test program, and 3) the length of down time to recover from random failures must be by itself one of the critical objectives for testing in fusion facilities.

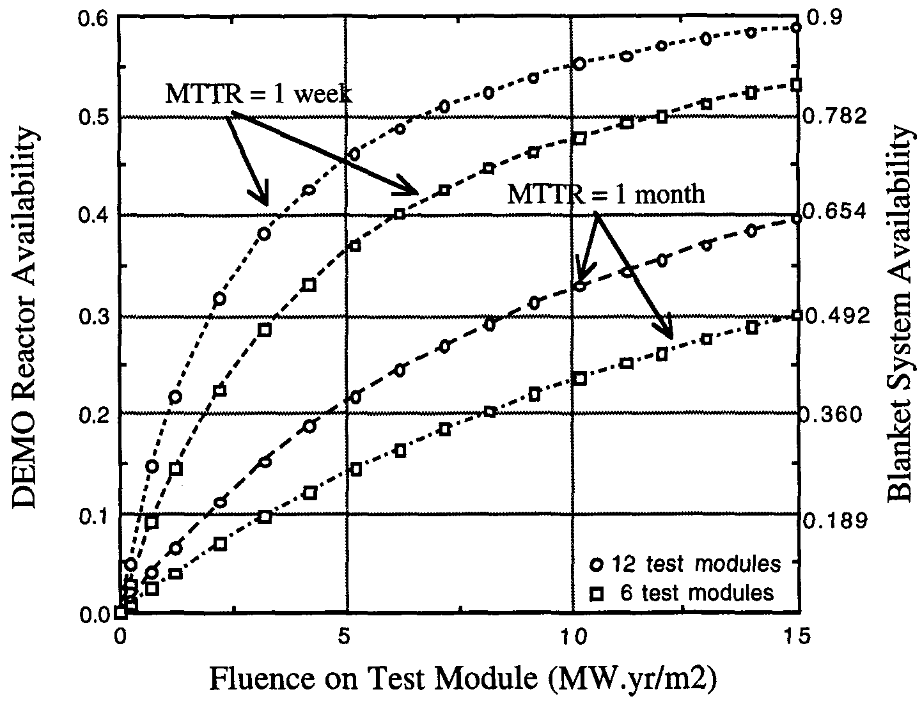

Figure A-2. DEMO reactor and blanket system availabilities obtainable at 80% confidence as a function of testing fluence on the blanket test module for MTTR = 1week and 1 month

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Appendix B: Experience from Fission Reactors

In the development of fast breeder reactors in Germany, the following essentials in regard to fuel rod testing had been specified as a prerequisite for the decision to start construction of a DEMO-reactor.

a) Sodium cooled fast breeder reactor:
30 prototypical fuel pins have to be irradiated in prototypical boundary conditions (neutron spectrum, power density, temperatures, sodium environment) up to a burn up of 50,000 MWd/t;

b) Steam cooled fast breeder reactor:
500 prototypical fuel pins have to be irradiated up to 30,000 MWd/t in fast neutron flux, superheated steam cooling and prototypical conditions (power density, temperature and pressure difference at fuel pin cladding).

The difference in the required number of fuel pins can be explained by the fact, that there were already results available for sodium cooled fuel rods (EBR, DfR) but no irradiation tests at all for steam cooled fuel pins.

Both DEMO reactors were composed of roughly 25,000 fuel pins with a goal maximum burn up of 85,000 MWD/t. If we extrapolate these essentials to a fusion DEMO reactor, the relevant comparison is the steam cooled reactor because this was, similar to fusion breeding blankets, the first of its kind.

Irradiation of 2% of the total number of fuel pins up to roughly 40% of the maximum goal lifetime fluence corresponds roughly to 2 blanket segments up to 4 MWy/m^2 (assuming 10 MWy/m^2 as a fluence goal). However, the reliability of the fusion blanket segments must be much higher than the one of fuel pins in a fission reactor because: 1) the fission reactor can tolerate a rather large number of defective fuel pins (e.g. water purification system in PWR are designed to cope with 1% of defective fuel rods); and 2) the fusion reactor, however, has to be shut down immediately if one of the blanket segments leaks or if there is a local malfunction in the cooling system. Consequently, a malfunction in one of the blanket segments requires a blanket exchange since an in situ repair is not possible. The time for this exchange (MTTR) has been estimated to be at least one month even for a machine designed for a fast blanket exchange. Therefore, the required mean time between failure (MTBF) of the blanket segments has to be exceptionally long (much longer that the life time as limited by the neutron fluence).

The demonstration of a higher reliability requires a larger number of test specimen and/or a longer testing time.

Appendix C. Tokamak VNS Design Envelope

In this appendix we summarize the performance guidelines established in the preceding sections, identify the physics and engineering assumptions for the VNS consistent with the present tokamak data base, and determine the range of VNS device parameters. A number of design options are considered, based on the use of superconducting or normal conducting toroidal field coils (TFCs). An updated version of the SuperCode [C-1] will be utilized for this purpose.

The basic variations of tokamak VNS designs include:

1. Superconducting (S/C) TFCs and adequate inboard radiation shield to protect the S/C magnets,
2. Multiturn normal conducting (M-T N/C) TFCs and adequate inboard radiation shield to limit damage to TFC insulators and normal conductor requiring standard aspect ratios ($R_0/a \geq 2.5$), and
3. Single-turn normal conducting (S-T N/C) TFCs and essentially no inboard nuclear shielding, permitting $R_0/a \leq 2$.

These design options have been considered recently [C-2-4] 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.

C.1 Performance Guidelines

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

1. Fusion power < 150 MW (In some cases, fusion power up to 400 MW were considered to assess the impact on design).
2. Neutron wall load = 1.0-2.0 $MW{\cdot}m^{-2}$.
3. Device fusion neutron fluence at the first wall $\geq$ 6 $MW{\cdot}y{\cdot}m^{-2}$.
4. Maximum site power requirement $\leq$ 700 MW.
5. Steady state (or long plasma burn with duty cycle $\geq$ 80%
6. Device load factor (duty cycle • availability) $\geq$ 30%.
7. Surface area (at the first wall) for test module $\geq$ 10 m^2.
8. First wall/blanket/shield thickness:

a. Inboard distance from first wall to inboard leg of TF coils: Δ_I = 83 cm, 44 cm, or 4 cm for the three design options.

b. Outboard thickness: $\Delta_O \geq 1.0$ 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 magnet sizes.

C.2 Physics Assumptions

The plasma physics assumptions and operating conditions for the tokamak VNS are summarized in Table C-I, and compared with available information on ITER [C-5]. A lower bound of 0.6 m in 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 [C-6]. *The subscript "95" refers to the surface containing 95% poloidal flux in the plasma.* Somewhat higher elongations (κ_{95} = 2.1), safety factors ($q_{95} \geq$ 3.5-4.5), and plasma Troyon beta factor ($g_T \leq 3.5$, with β defined relative to the average magnetic field in the plasma [C-7]) are assumed [C-8,C-9] for the VNS, than those assumed for ITER. The confinement improvement factor H_f (relative to the ITER-89P scaling [C-10]) assumed for VNS is £ 2.5, as indicated in present experiments for H-mode plasmas [C-11].

Steady-state non-inductive 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 non-inductive current drive efficiency required for VNS is assumed to be similar to those already achieved to date using neutral beam and/or rf injection ($I_{CD}/P_{CD} \leq 0.3 \bullet 10^{20} \cdot A \cdot W^{-1} \cdot m^{-2}$) [C-12, C-13]. Minimum-size tokamaks are obtained when the solenoid is eliminated. This will lead to the requirement for non-inductive initiation and ramp-up of the plasma current. There is a significant data base for such operations [C-14,C-15], suggesting a viable and potentially low cost option for consideration in future VNS studies.

The toroidal field ripple at the outboard plasma edge will affect the confinement of supra-thermal 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 [C-16]. A large fraction of such ions are expected in VNS due to the steady-state drive powers. A somewhat more stringent limit than the ITER assumption is therefore required.

Table C-I. Plasma Assumptions and Operating Conditions for Tokamak VNS

	Tokamak VNS	ITER*
Plasma minor radius, a (m)	≥0.6	2.8
Plasma elongation, k_{95}	≥1.6-2.1	≥1.65
Plasma triangularity, δ_{95}	≥0.2-0.3	0.3
Plasma edge safety factor, q_{95}	≤3.5-4.5	2.9
Normalized plasma beta, g_T (%•m•T•MA^{-1})	≤3.5	2.0
Confinement H_f relative to ITER-89P scaling	≤2.5	2.5
Current drive coefficient (10^{20}•A•W^{-1}•m^{-2})	≤0.3	N/A
Toroidal field ripple at outboard plasma edge (±%)	≤0.5	≤1.0
Core ash build-up factor, $t_a/t_{burn-up}$	≤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
x-point to divertor distance in elevation plane (m)	1.0	2.0

*For nominal operation of the Basic Performance Phase of ITER [C-5].

The ratio of the confinement time for fusion ash (thermalized fusion alpha particle) to the D-T fuel burn-up time (ash production time), $t_a/t_{burn-up}$, controls the ash concentration in the plasma core [C-17]. High ash concentration increases the size and cost of a fusion device in order to maintain constant performance. For VNS, t_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 about 20% is presently estimated for ITER ($t_a \geq$

$15t_E \sim 50$ s, $t_{burn-up} \sim 150$ s) for a He recycling coefficient of 0.95-0.98 [C-17], where t_E is the plasma core energy confinement time. An ash concentration of 3-10% is therefore expected for VNS ($t_a \sim 15t_E \sim 6$-20 s, $t_{burn} \sim 150$ 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 two of those anticipated for ITER. For plasmas with successful divertors, the loading on the first wall is expected to be primarily due to radiation (Bremmstrahlung, impurity line, and synchrotron), which is assumed to be up to about 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 10-15%, 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}:

$$F_{div} \mu f_{div} \int P_{heat} B^g / R_0^{1.5} q_{95}^{0.5},$$

where P_{heat} is the total kinetic power entering the divertor chamber and g 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 [C-18], making g = 0.5. However, if the cross-field diffusion remains constant among all tokamaks (such as the 2-3 m^2/s assumed in ITER), making g = 0, the divertor heat flux factor f_{div} would be higher for the smaller lower-field tokamaks. The physical distance between the x-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, non-inductive 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 data base. 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.

C.3 Design Configuration and Engineering Features

Tokamak VNS design configuration and engineering features are driven by the nuclear testing requirements discussed in the preceding sections. These features are summarized in Table C-II and depicted in Figures C-1 and C-2 for the VNS options with normal conducting toroidal field coils. To ensure capability for achieving high fluence (≥ 6 MW•yr•m^{-2}) and high load factor ($\geq 30\%$), critical components in the VNS toroidal chamber will require ready access for repair or replacement. These components include divertor plates, first wall protection tiles, and nuclear test modules.

Table C-II. Key Design Configuration and Engineering Features for Tokamak VNS

Configuration/Features	S/C	M-T N/C	S-T N/C
Total inboard shield material thickness (cm)	72	23	3
Total outboard shield/blanket thickness (cm)	100	100	100
Number of outboard TF coil legs	12	8	8
Number of removable divertor modules	12	8	8
Elevation of outboard poloidal field coils	x-point	x-point	x-point
Jointed demountable TF coils	no	yes	yes
Average TF inner winding current density (kA•cm^{-2})	3.7	3.0	1.9-2.1*
Average TF outer winding current density (kA•cm^{-2})	3.7	1.0	1.0
TF coil load path through radiation shield	no	yes	yes
Poloidal field coil location vs. TF coil bore	external	internal	internal

*averaged over the entire center leg, which is hour-glass shaped at the mid-section (Figure C-2).

Features common to all neutron producing tokamaks include inboard shielding to protect magnets with electrical insulation, outboard shielding to minimize reactor hall activation and ensure personnel safety and access, accessible and removable blanket test modules at the outboard mid-plane, and removable divertor cassettes between the toroidal field coils [C-19].

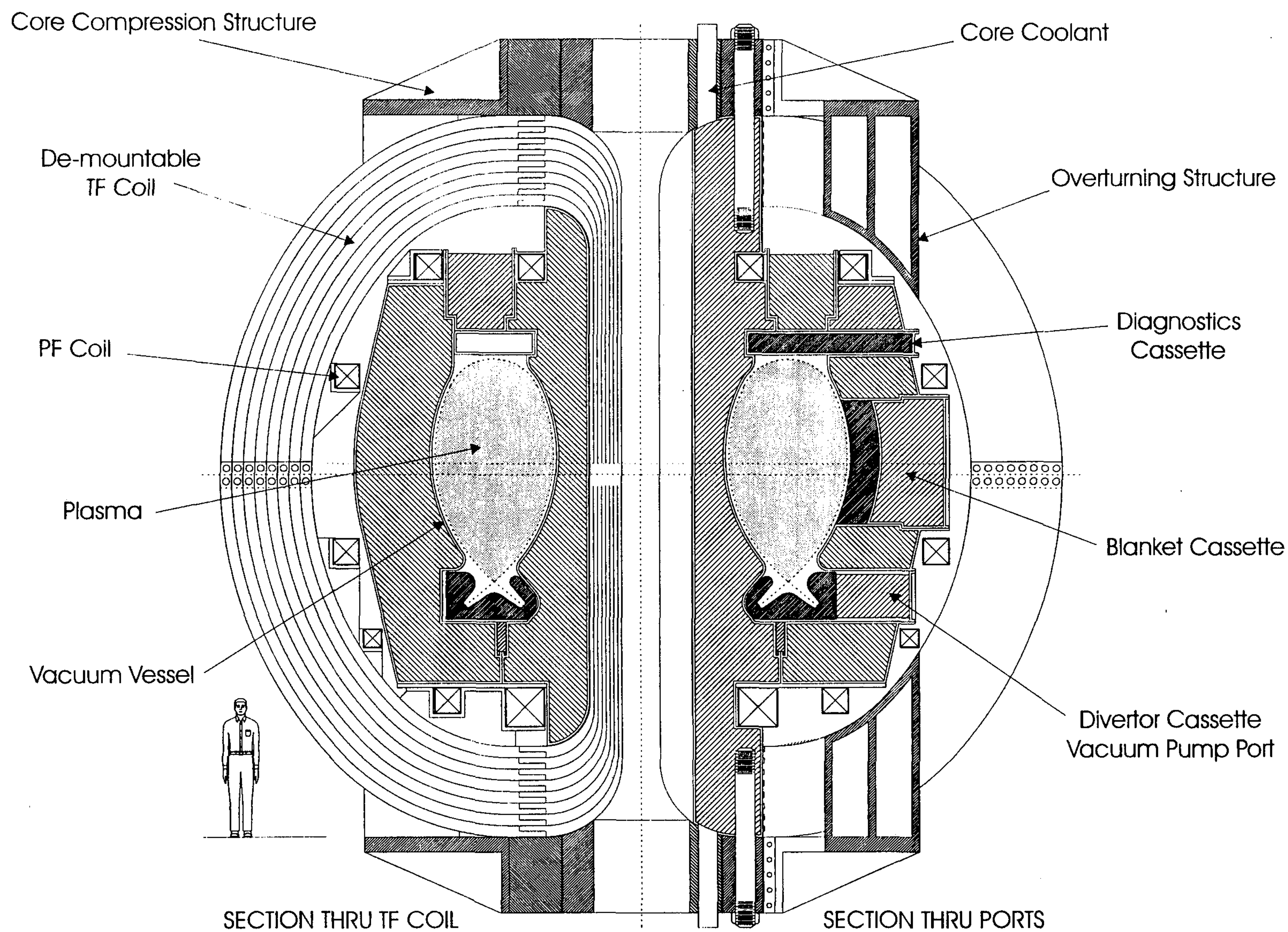

Figure C-1. Elevation View Depicting a VNS Using Multi-Turn Normal Conducting Toroidal Field Magnets That Require Some Inboard Shielding.

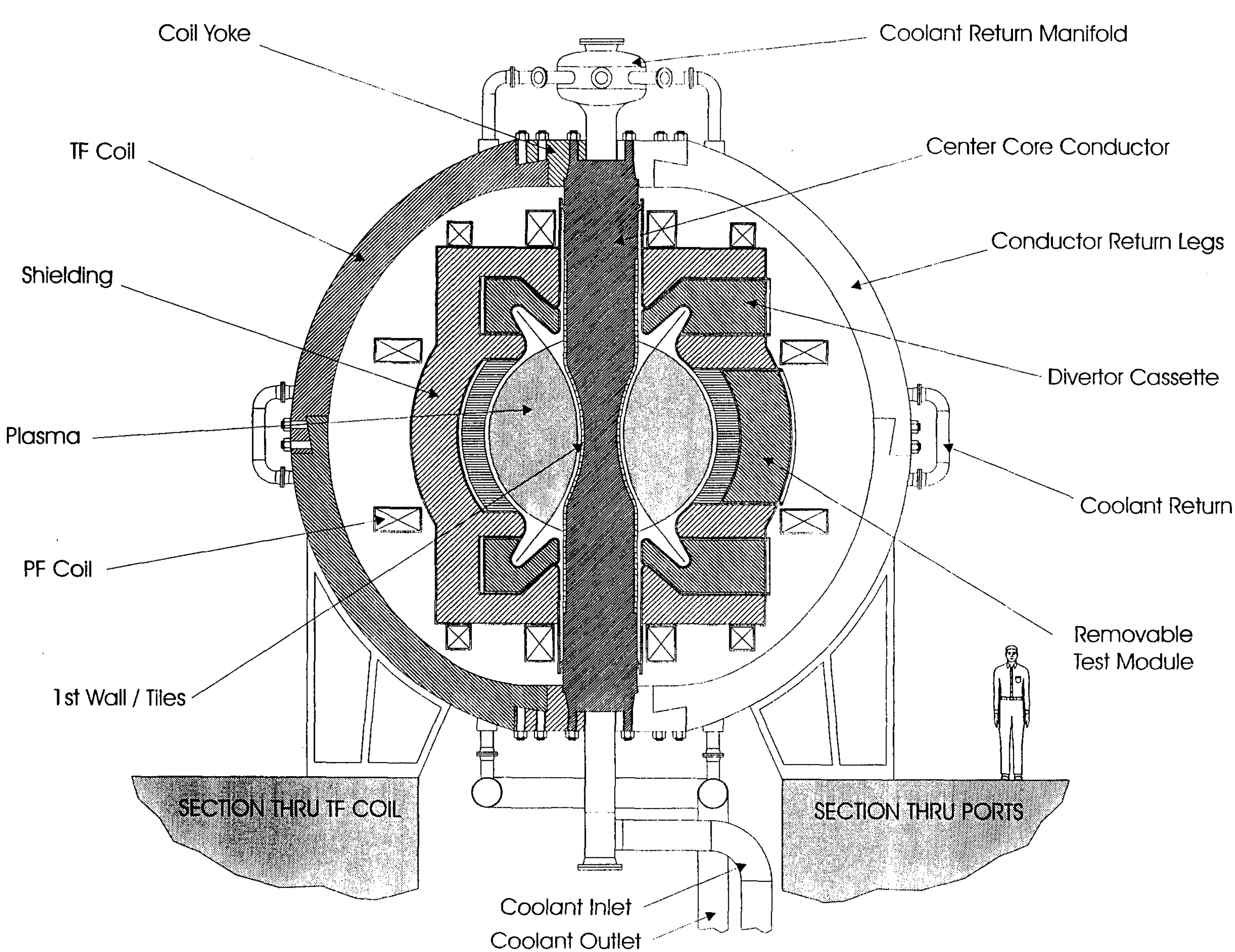

Figure C-2. Elevation View Depicting a VNS Using Single-Turn Normal Conducting Toroidal Field Magnets That Require No Inboard Shielding.

As shown in Figures C-1 and C-2, jointed demountable TF coils are used to ensure ability to disassemble and replace all key components of the VNS tokamak with normal conducting TF coils [C-3, C-20]. This also permit placement of super-conducting poloidal field coils internal to the TF coil enclosure. To maximize the wall area available for nuclear test modules in all options, the outboard poloidal field coils are placed at the maximum possible distance from the midplane permitted by the removable divertor modules. The average current density for the inner and outer legs of the TF coils are limited to 3.0 $kA \cdot m^{-2}$ and 1.0 $kA \cdot m^{-2}$, respectively, subject to temperature rises up to 150 °C in separate pressurized coolant channels.

To ensure adequate rigidity of the multiturn joints in the toroidal field coils (Figure C-1), the inboard shielding is configured to carry a fraction of the out-of-plane loads, a feature similar to that used in the SFDP concept [C-3]. In the case of single-turn toroidal field magnet (Figure C-2), the outboard shield is configured to carry essentially all of the out-of-plane loads.

The configuration for VNS using superconducting toroidal field coils is expected to be roughly similar to that of ITER, though much smaller in size and plasma current. Key differences in contrast with the normal conducting cases include: significantly thicker inboard shield (72 cm in thickness plus spacing), 12 TF coils, separate load paths for the TF coils and the shield structure, and poloidal field coils external to the TF coils. The current densities in the superconducting windings are assumed to be close to those assumed in the ITER design concept [C-5].

C.4 VNS Design Envelope

The preceding assumptions on the VNS plasma and device concepts define the constraints within which desirable design parameters can be estimated. We use the SuperCode [C-1], modified to account also for the regime of low plasma aspect ratios ($R_0/a \leq 2$) [C-4] and the use of normal conducting TF coils. The physics and engineering models in the code are up-to-date and consistent with the ITER design assumptions [C-5]. 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 above.

The key results for typical VNS designs providing a neutron wall loading of 1-2 $MW \cdot m^{-2}$ are summarized in Table C-III together with the ITER parameters for comparison. The radial build calculated for these designs for the lower wall loading are given in Figure C-3, consistent with Figures C-1 and C-2 in the gaps and thicknesses of the indicated elements. The S/C VNS option mimics the build-up configuration concept of ITER.

Table C-III. Key Parameters for VNS with Superconducting (S/C), MultiTurn Normal Conducting (M-T N/C), and Single-Turn Normal Conducting (S-T N/C) Toroidal Field Magnets, Compared with ITER.

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

*Parameters chosen for the Basic Performance Phase of the ITER outline design [C-5].
†Double-null poloidal divertors assumed.

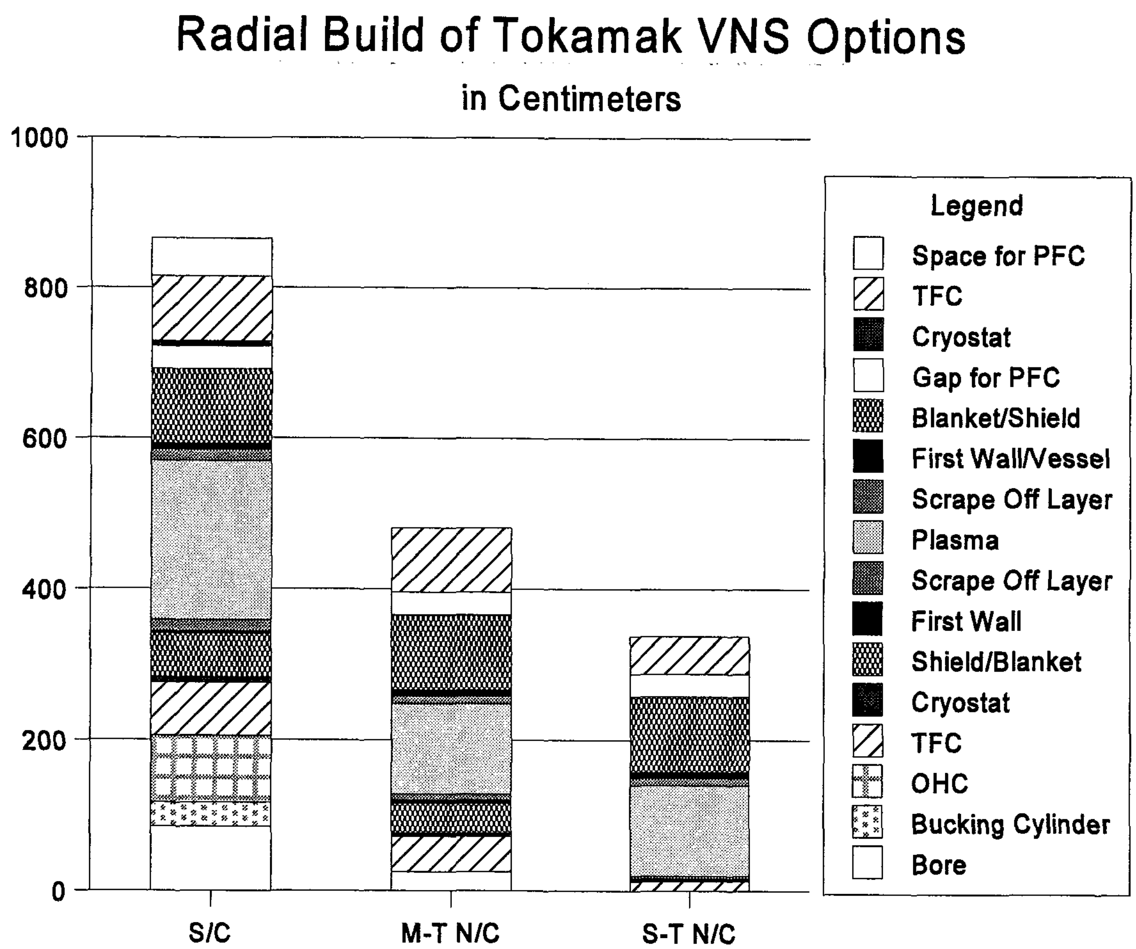

Figure C-3. Radial build of tokamak VNS options with superconucting (S/C), multi-turn normal conducting (M-T N/C), and single turn normal conducting (S-T N/C) toroidal field coils for an average neutron wall load of 1 MW/m^2.

Relative to ITER, the VNS with S/C TF coils has typically about 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 the 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 (about 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 $\langle T\rangle_n\langle n_e\rangle\tau_E \approx 6.6\times10^{20}$ keV$\bullet$ m$^{-3}\bullet$s, which is about a factor of 3 below that required for ignition. Relative to ITER, this VNS is about 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 about 17 MW$\bullet$T$^{0.5}\bullet$ m$^{-1.5}$, comparable to the ITER value. The total wall area accessible from outboard between the outer TFC legs and the outboard poloidal field coils is estimated to be about 56 m^2.

A significant reduction in device linear size (to about 10 m over all) from the S/C option is obtained by using multiturn normal conducting (M-T N/C) TFCs, in spite of the doubled wall loading. The latter permit a reduction in the inboard radiation shield (from 83 cm to 44 cm). The values for the plasma current and density remain similar to those for the S/C option. Reductions in plasma drive power (to 51-60 MW) and fusion power (to 109-231 MW) are significant, without leading to a significant change in the ignition parameter $\langle T\rangle_n\langle n_e\rangle\tau_E$ ($\approx$5.1-8.4$\times10^{20}$ keV$\bullet$ m$^{-3}\bullet$s). The (double-null) divertor heat flux factor f_{div}, assuming anomalous diffusion in the SOL ($\gamma = 0.5$), remain similar to the S/C 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 5.3-6.4 T at the major radius of 1.9-2.0 m. A wall area of about 35 m^2 is accessible from the outboard side in this device.

The use of a single-turn normal conducting (S-T N/C) inner leg for the TFCs permits the elimination of the inboard radiation shield, leading to a further reduction in device size for constant neutron wall loading. The device is reduced to about 7 m overall in linear size, the major radius being about 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 2.0-2.4 T) is seen and is a result of the low aspect ratio ($R_0/a \sim 1.3$) [C-8]. Further reductions in plasma drive power (to 19-29 MW) and fusion power (to

32-65 MW) are obtained, now with a reduced ignition parameter ($\langle T\rangle_n\langle n_e\rangle\tau_E \approx$ 4.4-5.5×10^{20} keV•m^{-3}•s). The relatively small change in the fusion amplification Q (to 1.7-2.2) 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 about 40% of the plasma pressure. Neutral beam injection heating and current drive at about 0.5 MeV [C-20] can be used to achieve this condition, which is similar to those achieved or simulated recently in TFTR [C-21] and JET [C-22, C-23], respectively. The power consumption for this case amounts to about 130-180 MW, half of which is supplied to the TF coils. An outboard wall area of about 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.

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 1-2 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 Figure C-4. The parameters determined so far point to smaller fractions of the ITER cost.

C.5 Discussion

Our results suggest that tokamak VNS with normal conducting toroidal field coils can be similar to present-day D-T tokamaks, such as TFTR [C-21] and JET [C-22,C-23], in several important parameters.

The former do not exceed these experimental devices in major and minor radii, toroidal field, peak ion temperature, plasma volume, and plasma surface area. However, important differences exist. These include about twice the plasma current, 3-4 times the density, three times the divertor heat flux factor, 3-4 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 around 1.0×10^{20} m^{-3},
2. Steady-state plasma particle and power handling at divertor heat flux factors f_{div} around 14 $MW \cdot T^{0.5} \cdot m^{-1.5}$,
3. Steady-state neutral beam operation at energies around 500 keV [C-24], or rf operation with equivalent performance, and
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 toroidal field coils. In the case of the low-aspect-ratio option, a key difference is the use of a single-term, demountable center leg for the toroidal field coils. 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 development step in fusion energy research.

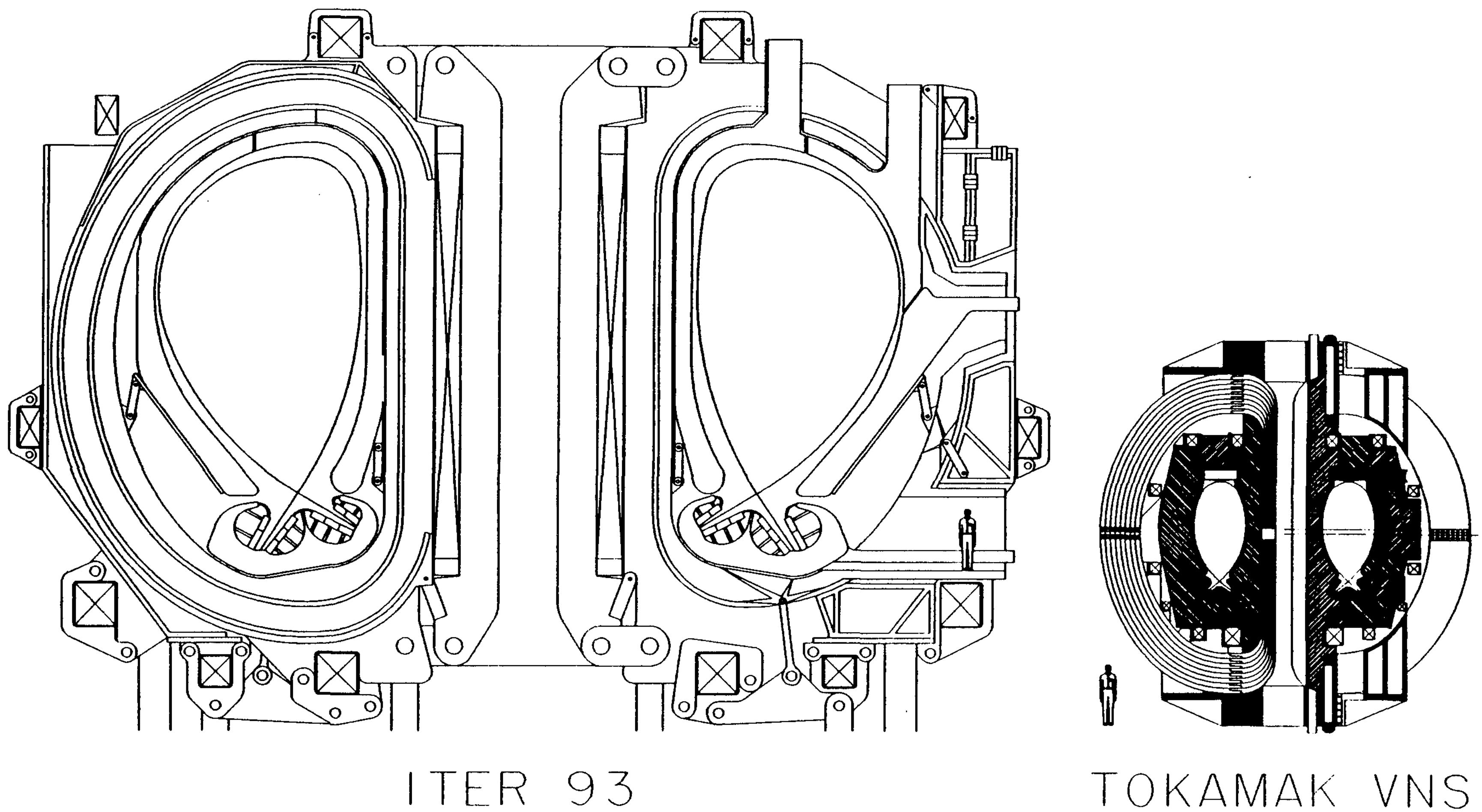

Figure C-4. Elevation views for ITER [C-5] and a typical tokamak VNS with multiturn normal conducting toroidal field coils (M-T N/C, Figure C-1), depicted in the same scale.

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