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ITER OPERATIONS AND RESEARCH PROGRAMME

INTERNATIONAL ATOMIC ENERGY AGENCY, VIENNA, 1991

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G.W. PACHER, D. POST, E. SALPIETRO, G. SHATALOV,
Y. SHIMOMURA, M. ABDOU, C. BAKER, F. CASCI, G. CASINI,
M. CHAZALON, M. DALLE-DONNE, W. DAENNER, J. DOGGET,
F. ENGELMANN, N. FUJISAWA, Y. GOHAR, J. HOGAN,
T. KURODA, V. KOVALENKO, I. KIRILLOV, S. MALANG,
R. MATTAS, S. MORI, V. MUKHOVATOV, E. MURAVIEV,
H.D. PACHER, L.J. PERKINS, A.R. RAFFRAY, D. SMITH,
YU. STREBKOV, M. SUGIHARA, M. TILLAK, T. TSUNEMATSU,
J. WESLEY, H. YOSHIDA, K. YOUNG

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FOREWORD

Development of nuclear fusion as a practical energy source could provide great benefits. This fact has been widely recognized and fusion research has enjoyed a level of international co-operation unusual in other scientific areas. From its inception, the International Atomic Energy Agency has actively promoted the international exchange of fusion information.

In this context, the IAEA responded in 1986 to calls for expansion of international co-operation in fusion energy development expressed at summit meetings of governmental leaders. At the invitation of the Director General there was a series of meetings in Vienna during 1987, at which representatives of the world's four major fusion programmes developed a detailed proposal for a joint venture called International Thermonuclear Experimental Reactor (ITER) Conceptual Design Activities (CDA). The Director General then invited each interested party to co-operate in the CDA in accordance with the Terms of Reference that had been worked out. All four Parties accepted this invitation.

The ITER CDA, under the auspices of the IAEA, began in April 1988 and were successfully completed in December 1990. This work included two phases, the definition phase and the design phase. In 1988 the first phase produced a concept with a consistent set of technical characteristics and preliminary plans for co-ordinated R&D in support of ITER. The design phase produced a conceptual design, a description of site requirements, and preliminary construction schedule and cost estimate, as well as an ITER R&D plan.

The information produced within the CDA has been made available for the ITER Parties to use either in their own programme or as part of an international collaboration.

As part of its support of ITER, the IAEA is pleased to publish the documents that summarize the results of the Conceptual Design Activities.

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1. GOALS

The objectives of ITER are stated in the terms of reference [1.1]. In an abbreviated form, they are:

1. to demonstrate controlled ignition and extended burn of deuterium-tritium plasmas, with steady state as an ultimate goal,
2. to validate design concepts, qualify engineering components, and demonstrate the potential of a fusion power reactor, and
3. to serve as a test facility for neutronics, blanket modules, tritium production, and advanced plasma technologies.

In summary, "ITER should provide the database in physics and technology necessary for the design and construction of a demonstration fusion power plant".

ITER will be operated in two phases, a physics phase lasting about six years, and a technology phase lasting about 12 years. The physics studies will be primarily conducted during the physics phase with low neutron fluences. Initial studies for the technology issues will begin during the physics phase, and will be intensified during the technology phase. Engineering testing will be done with total neutron fluences in the range of 1-3 MWa/m^2. Successful ITER operation, in itself, will mark a major milestone in the development of fusion energy because all of the major components of a reactor will have been successfully operated as an integrated reactor system.

Operational scenarios have been developed for both ignited operation and for technology testing operation with long pulses. These operational scenarios are consistent with the physics and engineering design constraints. However, steady-state operation places severe demands on the power exhaust system and improvements in the projected performance of the divertor or new divertor concepts are needed for the steady-state operating scenarios.

Operational flexibility is an important part of the design of the tokamak and tokamak systems to provide margin to ensure that the goals of the program can be met. This flexibility includes non-inductive current drive systems for control of the plasma current profile, fuelling systems to affect the density profile, three types of auxiliary heating systems to provide flexibility in the heating profile, a flexible poloidal field system to provide plasma shaping and control, divertor systems to control the impurity levels and provide control of recycling, and the capability in the PF and non-inductive current drive systems to increase the maximum current from 22 MA to 28 MA for short pulses (about 50 s) to provide margin for unfavorable scaling of plasma energy confinement.

1.1. Physics Objectives

The major goal of the Physics program of ITER is to establish the physics basis for the design of a Tokamak-Based Demonstration Power Plant - DEMO. This means that ITER must study long-pulse ignited plasmas. "Long pulses" in this context means durations greater than 200s long, since this is long compared with all the relevant plasma physics time scales except the time required to establish an equilibrium current profile (which takes 1000-2000 s at ITER parameters). The DEMO plasma physics issues include: confinement, operational limits, plasma

TABLE 1.1. POSSIBLE OPERATING PHASES FOR ITER

Physics Phase						**Technology Phase**							
		Pre-Technology Phase											
OPERATIONAL CONDITIONS		PLASMA OPTIMIZATION		IGNITION & DRIVEN OPERATION		CONCEPT PERFORMANCE TESTS		LONG-TERM CONCEPT VERIFICATION TESTS					
year 1	year 2	year 3	year 4	year 5	year 6	year 7	year 8	year 9	year 10	year 11	year 12	year 13	years 14 - 18
H/He	H/He	D/D/He	D/T	D/T	D/T	D/T	D/T	D/T	D/T	D/T	D/T	D/T	D/T
Initial ohmic operation; Full-field and heating systems tests		Physics studies; Steady-state studies; He-pumping; Current drive studies and profile optimization;		"Flash" ignition studies; α-particle studies; He ash removal; 200 s controlled burn; Driven operation		Short-term blanket sub-module tests including T-extraction; Surveillance tests of machine components;		Engineering equipment reliability data; Component reliability data; Blanket module performance tests; Fuel processing reliability data; Material irradaiation experiments; Transient safety tests; Continued surveillance testing;				Segment tests This phase can continue further if necessary	
MHD tests for liquid metal blankets				Driver blanket & fuel processing system;									
		Divertor tests; Wall material study; Test blanket check-out; Neutronic tests											
Hands-on maintenance			Remote maintenance			Availability 5-10%		Availability 15-20%					
						Individual campaigns will usually last 1-2 years		Experimental campaigns include combinations of long-term tests (several years each) and short-term tests (~1 year)					

control, heating, and non-inductive current drive. All of these will be addressed by the ITER investigations.

ITER will study both the conditions necessary to produce ignited and high-Q plasmas and the properties of such plasmas. [1-2]. These include the achievement and study of:

1. Sufficient energy confinement for ignited and high-Q operation ($n\tau_E T$=4 to 8×10^{21} keV s/m^3);
2. Adequate level of MHD stability to minimize the frequency of plasma disruptions for the relatively high-β plasmas needed to produce fusion powers near 1 GW;
3. Exhaust of high levels of thermal power without producing excessive plasma contamination by impurities;
4. Production and control of a highly elongated, high-β plasma; and
5. Successful heating of a plasma to ignited conditions with auxiliary heating systems and alpha particles.

In addition to the above, long pulse operation requires the achievement and study of:

6. Control of the helium "ash" concentration during long burn times in order to sustain the fusion process.
7. Fuelling of the plasma to replace the D-T burned by the fusion reactions and pumped by the particle exhaust system;
8. Extension of the pulse length and control of the current profile with non-inductive current drive;
9. Production and control of highly elongated plasmas for time scales comparable to or larger than the resistive skin time; and
10. Control of MHD stability during pulse lengths long compared with the time required for the current profile to reach equilibrium in order to assure adequately low disruption frequency;

Most of the physics studies will be conducted in D-T plasmas after initial commissioning and physics confirmation in hydrogen, helium, and deuterium discharges. The physics phase of ITER is expected to last for six years, as indicated in Table 1.1. The operational plan for the physics phase is shown in more detail in Table 2.1.

1.2. Technology Goals.

ITER has two major technology goals. First, ITER has to conduct extensive tests of the machine components to demonstrate the high reliability of its components so that the operation requirements of the basic machine for Physics and Technology Phases and the availability goals of ITER itself can be met. These tests should also provide a basis for extrapolating the design of the major components to DEMO conditions and requirements. Second, ITER have to serve as a test bed for DEMO-like components (e.g. blanket, first wall and divertor target designs) to be tested in test modules or segments.

In order to meet the first goal, engineering tests will be carried out, whose purpose it is to validate the process followed in designing, building, and operating ITER as a prototype of a fusion reactor. In particular the design codes, the technological processes, the manufacturing procedures, the modelling of components and of the plasma will be validated. This engineering data collected during the operation of ITER will be used to establish the design, and lifetime, as well as reliability and availability estimates for DEMO and future fusion reactors. Tests will notably be carried out on the vacuum vessel, cryogenic system, TF and PF coils, pumping and fuelling systems, tritium recovery system, heating and current drive systems, and diagnostics and maintenance equipment.

For the second goal, ITER testing objectives contained in ANNEX 1 to the Terms of Reference state that ITER will provide the date base "necessary for the design and construction of a demonstration fusion power plant". To do so ITER will serve as a test facility for blanket modules, tritium production, neutronics studies and testing advanced plasma technologies, including high-heat-flux components. An important objective will be the extraction of high-grade heat from reactor-relevant blanket modules and testing of reactor-relevant materials in a fusion environment, including advanced low activation and radiation resistant materials. However, blanket designs and, partly, materials proposed for DEMO and fusion power reactors differ from those of the ITER driver blanket. Major differences arise from the driver blanket operation at lower parameters (e.g. coolant temperature and pressure). Reactor-relevant blankets should demonstrate the potential for electricity production and the environmental and economic attractiveness of fusion. Accordingly they have to operate at higher temperatures and pressure, use advanced materials and operate at higher specific heat loads. Because the majority of ITER parameters will most likely be lower than those DEMO and commercial reactors, the modules for testing have been designed using engineering scaling to preserve important phenomena [1.3]. Engineering scaling involves altering physical dimensions (e.g., increasing the thickness of a solid breeder plate in a blanket to increase temperature differences and thermal stress) and changes in operating conditions (e.g. reducing the mass flow rate of the coolant to maintain coolant temperature rise). Data from tests at these "scaled down" conditions can then be extrapolated to reactor conditions and thereby fulfil the second goal of the technology testing mission of ITER.

REFERENCES:

[1.1] *Establishment of ITER: Relevant Documents,* IAEA/ITER/DS/1, ITER Documentation Series, IAEA, Vienna (1988).

[1.2] *ITER Physics,* IAEA/ITER/DS/21, ITER Documentation Series, IAEA, Vienna (1991).

[1.3] *ITER Test Program,* IAEA/ITER/DS24, ITER Documentation Series, IAEA, Vienna (1991).

2. OPERATIONAL PLANS

ITER operations will be carried out in the two phases illustrated on table 1.1: a physics phase devoted mainly to attaining the plasma physics objectives, and a technology phase devoted to fulfilling the engineering objectives and completing the testing program.

Before the initial operation in the physics phase, extensive commissioning tests of the components of the tokamak and the necessary support systems will have been carried out. This includes tests of the magnet systems with the associated power supplies and protections, gas handling systems, plasma heating, plasma control, and plasma diagnostic systems, as well as the reactor control systems and instrumentation.

In the physics phase, ITER will initially aim to demonstrate controlled burn of more than 200 s duration in inductive operation. The burn pulse will be extended towards longer burn duration, possibly up to steady-state, using non-inductive current drive, both to obtain the physics data base required for future reactors and to optimize the long-burn mode for the technology phase. The test of feedback-control of the reactor systems is to be demonstrated in this phase of machine operation.

Neutron fluence is neither a requirement nor a constraint in the physics phase. The purpose of this phase is rather to demonstrate the validity of the physics basis for the design, by attaining ignited burn conditions, and to develop the long-pulse modes required for the technology phase operation.

The technology phase is characterized by a specific fluence goal, required for attaining the testing objectives of the device. Nuclear and blanket testing sets a minimum requirement for neutron wall loads (>0.8 MW/m^2 average, corresponding to 1.2 MW/m^2) and pulse lengths (~1000 s), and also an upper limit to the maximum time between pulses (dwell time). Operation in the technology phase will therefore be optimized to satisfy these requirements concurrently with stress and fatigue limits of the machine. The determination of this optimal operation scenario is one of the objectives of the physics phase which precedes the technology phase.

2.1. Physics Phase Experimental Plan

Since many of the preparatory experiments can be carried out with low neutron fluxes, the physics phase of ITER can be subdivided according to the activation of in-vessel components into: (i) a zero activation phase, i.e. operation in hydrogen, (ii) a low activation phase, in which deuterium and helium are used; and (iii) a high activation phase, with deuterium-tritium operation and ignition. These phases are indicated on table 2.1.

2.1.1. Zero-activation phase

The zero-activation phase serves to commission the device, its subsystems, a large part of the plasma diagnostics, and the plasma control system in operation with plasma. The operational space of the machine will be explored with ohmic heating,

TABLE 2.1. ITER PHYSICS PHASE OPERATIONAL PLAN

ZERO ACTIVATION	LOW ACTIVATION	HIGH ACTIVATION
OPERATIONAL CONDITIONS	PLASMA OPTIMIZATION	IGNITION AND DRIVEN OPERATION
6 000 Shots	2 000 Shots	7 000 Shots

I---Year 1---I----Year 2----I----Year 3-----I----Year 4----I----Year 5-----I----Year 6-----

H/He	H/He	H/He	H/He	H/He/D	D/T	D/T	D/T
Initial Ohmic Operation	Full Field & Heating System Tests	Physics Studies Wall Material Studies	Physics Studies Steady State Studies	He-Pumping Studies; Pre-Ignition Optimization	"Flash" Ignition Alpha Particle Studies	200 s Burn He-ash Removal	Final Parameters Definition Driven Operation
1500	500	2000	2000	2000	3000	2000	2000 shots

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3.5 T	5 T	$\tau_E \geq 2$ s	$\Delta t \geq 1000$ s	$\tau_E \geq 3$ s	$\Delta t \geq 20$ s	$\Delta t \geq 200$ s	$\Delta t \geq 1000$ s
15 MA	22 MA	$\geq 10^{20} m^{-3}$	$< 5.10^{19} m^{-3}$	$\geq 10^{20} m^{-3}$	$Q \geq 30$	$Q \geq 30$	$Q \geq 5$
10 MW	100 MW	100 MW	100 MW	100 MW	1 GW	1 GW	1 MW/m^2

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Hands-on Maintenance Test of Remote Maintenance	Internal Changes Remote Maintenance Tests	Internal Changes for DT Physics Studies by Remote Maintenance	Transition to Technology Phase

auxiliary heating, and non-inductive current drive. Since a relatively high frequency of disruptions is likely in this exploratory phase, the investigations of operational limits will be carried out at reduced field and plasma current whenever possible so as to reduce the demands on the machine. At the end of this phase, full-performance shots in hydrogen with optimized heating should demonstrate that ITER can attain the plasma parameters necessary for ignition. This must be confirmed in the following, low-activation, phase. Operation with non-inductive current drive will indicate the operating scenario that is desirable for the long-pulse or steady-state mode.

Approximately six thousand discharges are expected to be required for the zero-activation phase (see Table 2.1), and a time span of three years is expected to suffice for this phase and the following low-activation phase.

2.1.2. Low-activation phase

Operation with deuterium (and some experiments with an admixture of helium-3) in the low-activation phase allows a confirmation of the ion mass effect on plasma confinement to be obtained with much reduced activation of the structures. Radiation shielding calculations can be validated and the extrapolation to thermonuclear conditions can be confirmed with the moderate neutron yield from D-D fusion reactions. The performance of the diagnostic system in a moderate radiation environment will be verified. Helium injection into the plasma will be used to demonstrate helium exhaust and pumping from the divertor. Tests of control strategies and emergency shutdown procedures will be conducted to provide documentation for the licensing for full-power DT operation.

Approximately two thousand discharges are estimated for this stage.

2.1.3. High-activation phase

The purpose of the high-activation phase, with deuterium/tritium plasmas, is twofold: to demonstrate controlled, ignited burn in a D/T plasma, and to define and optimize the long-pulse mode required for the subsequent technology testing program.

In the ignited burn condition, the alpha particle heating will eventually dominate the auxiliary heating input to the plasma by over an order of magnitude, so that significant, previously unexplored, physical effects can be expected. The development of a controlled burn includes both ensuring stationary plasma conditions at a given fusion power and thermal stability of the discharge. Operation at high Q (~40) with external power controlled by feedback is a scheme that has been proposed and must be tested.

Controlled burn at high Q will be a new experimental situation. It is therefore reasonable to foresee an extended period of experimentation in order to develop and optimize the control until extended burn periods, of the order of 200 s, are achieved and helium ash accumulation becomes significant. At that time, the problem of helium exhaust from the main plasma to the divertor, and from the divertor to the pumps, will become of critical importance to keep the helium fraction in the discharge low enough (below 10%) to maintain ignited burn. The study of controlled burn conditions is therefore necessary to optimize the helium pumping and the divertor and first wall heat loads for the technology phase operation.

During this stage, the physics of a burning plasma with an appreciable hot alpha particle component will be investigated. This study will address energy and particle transport, MHD effects, changes in operational limits, and disruption control under these conditions. As the burn time becomes longer, current profile control (by non-inductive means) must be implemented to assure the MHD stability of the discharge and avoid disruption.

The further major goal of the high-activation phase is the development of the long-pulse scenario suitable for the technology phase, using non-inductive current drive. This includes experiments to maximize the inductive burn duration by reducing volt-second consumption during current ramp-up, long burn experiments (hybrid operation) in which non-inductive current drive reduces the consumption of inductive volt-seconds during the burn, and experiments aiming towards a true steady-state in which the compatibility of a current driven by purely non-inductive means during the burn with acceptable divertor conditions is examined. An essential part of the program is the demonstration of plasma diagnostic capability over extended periods in full reactor conditions. The optimal mode of operation for the technology phase will be determined from the results of these experiments.

All of these investigations in deuterium/tritium plasma are expected to require seven thousand discharges over a span of three years. The integrated burn time in the physics phase is expected to be 1.5×10^6 seconds for a total first-wall fluence of .05 MWa/m^2.

2.2. Technology Phase Operational Plan

Operation in the technology phase is subject to constraints different from those in the physics phase. Testing of blanket concepts is taken to require an average neutron wall loading of at least 0.8 MW/m^2 with a minimum burn pulse length of the order of 1000 s to achieve thermal equilibrium in the blanket. The off-burn time ("dwell time") should not be too long, typically a few hundred seconds. For materials testing in ITER, the accumulated fluence goal is taken to be 1-3 MWa/m^2.

Because of these requirements, which imply an integrated burn time of more than one year (and therefore 5 to 10 years of operation in an experimental context), the reduction of fatigue effects, both thermal and mechanical, is important. Clearly, a continuous mode of operation, using steady-state current drive, is the preferred operating mode if all other constraints can be met. If the conditions for true steady-state operation can not be satisfied (notably because of excessive divertor heat loads and temperatures), it is desirable to maximize the burn length by non-inductive ramp-up assist and hybrid burn (lengthening the inductive burn duration by non-inductive current drive assist) in order to reduce the number of cycles required to attain a given fluence. The precise optimization of operation in the technology phase will be performed towards the end of the physics phase in the light of these considerations.

Initial operation in the technology phase will concentrate on the verification and selection of DEMO-relevant blanket concepts and material testing. This stage is expected to last for approximately three years, and will require a fluence of the order of 0.1 MWa/m^2. It will be followed by the long-term testing stage, in which the endurance tests of the chosen concepts at neutron fluences in the range of 1 MWa/m^2 will be performed.

2.3. Remote-Site Participation in ITER Operations

A partial decentralization of ITER operations will make it possible to increase the level of participation of home fusion groups in the scientific research program and testing activities of ITER, and to increase the resources available above those available at the central site. Even though the considerations are still at a preliminary stage [2.1] and must be considerably expanded in the course of the Engineering Design, the following points have emerged.

Certain activities must remain fully at the central site, especially the decisions as regards safety, direct technical operation of the machine and its auxiliary systems, and coordination of the experimental and testing program. Other tasks, such as the definition of the experiment or test, and the data evaluation and analysis, can be decentralized around a strong nucleus at a central site. The degree to which this is feasible depends on the number of external sites collaborating in a specific program, the technical quality of communication links, and the degree of experience the external collaborators have at the central site.

For certain tests, especially for the technology phase, the object to be tested is a self-contained unit, apart from site services. Similar to experimental operation of accelerators, ITER would be a user-facility for such tests, and the preparation and evaluation of the test could easily be carried out elsewhere than at the central site. In analogy to accelerators, the experimental or testing staff for these self-contained experiments would be at the central site only for installation of the unit, and some fraction of the actual testing time. A good part of the test evaluation would be carried out at the home laboratory. This is clearly especially attractive if only one partner or laboratory is involved, and becomes more difficult when the number of external sites directly involved is large.

The situation is more complex for experiments concerning all partners such as, for example, the optimization of ITER operational characteristics. The degree to which remote participation in such experiments is feasible depends on the development of highly specific experimental plans. Activities at a central location demanding strong participation from all involved parties would continue to include planning, coordination, and execution of the experimental run. Even though analysis can be carried out off-site, access to and release of data will need to be controlled. Co-ordination and validation of the results would therefore be essential activities of the central team.

Off-site activity could be centered around remote "annexes" of the operation centre, connected to ITER via videoconferencing and high-speed transmission links to allow interactive participation in operations and to ensure rapid data transfer to remote locations for analysis and evaluation. The installation of remote "annexes" is presently being considered for several large experiments. At the present time, a discussion of specific technical solutions is considered to be premature, because, with the rapid advance of technology in high-speed data transmission, the feasibility of this approach is expected to be greatly enhanced in the future.

Active and effective participation in the experimental and test program depends also on the personal knowledge off-site collaborators have of the conditions and personnel at the central site. An extended period (~year) on-site for most off-site personnel is likely to be necessary for developing this knowledge, and should be a part of the implementation of remote-site collaboration.

At the present time, the TFTR experiment is beginning to implement remote participation in experimental runs at the University of Wisconsin, and is actively considering other collaborations. First-hand experience with off-site collaboration on large fusion devices will therefore become available during the Engineering Design Phase. Along with the experience from accelerator experiments, this will be used to develop a strategy for this aspect of the experimental and test program for ITER.

REFERENCES:

[2.1] D.E. POST and J. HOGAN, "Proposal for De-centralizing the Operation of ITER", ITER-IL-PH-15-0-U-1, 1990

3. OPERATIONS AND RESEARCH PROGRAM - PHYSICS PHASE

3.1. Reference Physics Operational Modes

The physics phase of operation is intended to confirm the extrapolation of the physics data base from present experiments to ITER, to demonstrate controlled burn, to investigate steady-state operation, and to develop the operating mode for technology testing. These goals have determined the reference operating points described in this section. The parameters of the operating points to be developed in the physics phase are listed on table 3.1. For the reference operating points the plasma major radius is 6m, minor radius 2.15 m, elongation at the 95% flux surface is 1.98, and triangularity for the same surface is 0.4. The normalized internal inductance of the plasma has a reference value $l_i(3)=0.65$. In order to assure an adequately low frequency of disruptions, the minimum safety factor q for the reference scenarios was chosen to be 3. The reference operating points are described in more detail in refs. 3.1 and 3.2.

The reference operating points of table 3.1 are chosen according to the following criteria. The fusion power should be approximately 1 GW, assuring an average neutron wall load of 1 MW/m^2. The burn time for ignition experiments should be in excess of 200 s, and for technology tests in excess of 1000 s. The power conducted to the divertor plates should be less than ~120 MW (which corresponds to static peak power loads on the divertor plates of ~20 MW/m^2 according to present 2-D modelling with physics peaking factors - Table 3.1 and section 3.4 in [3.1]). If injection of additional impurities ("impurity seeding") is used to reduce this heat load, the concentration injected should be modest (<0.1% of medium-Z impurity). The maximum additional heating and current drive power is 115 MW, and, in long-pulse operation, more than 30% of the plasma current is to be driven non-inductively in order to assure current profile control. The maximum required confinement enhancement over L-mode should be 2.2, and the operating points must satisfy density and beta limits (g_{Troyon} <2.5 for inductive, and <3 for current-driven operation).

The reference ignition scenario A1 (column 1, Table 3.1) aims to produce a fusion power of 1100 MW with a purely inductively driven plasma current of 22 MA, which corresponds to an edge q of 3 at the 95% flux surface. For these parameters, L-mode scaling according to the ITER power law [3.1] predicts a confinement time of 1.9 seconds. An H-mode enhancement factor of 2 is required to assure ignition with 10% thermal alpha particle fraction. ITER H-mode scaling [3.1] predicts a confinement time of 5.9 seconds for these parameters. The expected degradation of confinement by ELM's, which are required to prevent helium and impurity accumulation in the plasma core, is of the order of 25%. The margin in confinement time taking into account ELM's is therefore almost 20%. The average plasma density ($1.2\times10^{20}m^{-3}$) exceeds the L-mode density limit slightly for the inner divertor channel according to the present models for edge density limits. However, the density limit for H-mode operation is expected to be somewhat higher and more power is expected to flow into the inner divertor channel, thus raising the limiting density. The plasma beta required corresponds to a Troyon g-factor of 2, leaving ample margin from the

TABLE 3.1. REFERENCE AND NOMINAL OPERATING MODES

code	A1	A1"	B1	B6	
description	reference ignited burn**	short ignition (0% alphas)	reference long pulse	nominal steady state	steady-state physics tests
I (MA)	22	22	15.4	19	10
Wall load (MW/m^2)	1.0	1.0	0.8	0.7	0.27
Q	-	-	8	6.7	2.6
Burn time (s)	400	short	2500	-	-
I_{BS}/I	0.14	0.12	0.3	0.3	0.5
I_{CD}/I	0	0	0.3	0.7	0.5
Loop voltage (V)	0.12	0.11	0.045	-	-
Z_{eff}	1.66	1.56	2.2	2.2	1.9
q(95%)	3	3	4.4	3.5	6.5
g-Troyon	2	1.75	2.7	3	2.4
beta (%)	4.2	3.7	4	5.4	2.3
beta-p	0.62	0.54	1.4	1.1	1.6
$\langle n_e \rangle (10^{20} m^{-3})$	1.2	1	1.1	0.64	0.76
$\langle T_e \rangle$ (keV)	10	10	11	20	8.5
$\tau_E^{ITER89-P}$ (s)	1.9	1.7	1.2	1.3	1.0
τ_E requ'd. (s)	3.8	2.8	2.6	2.7	2.2
τ_E^{H-mode} (s)***	4.4***	4.1***	2.8***	3.1***	2.3***
P_{fus} (MW)	1100	1100	900	750	300
P_{CD} (MW)	0	0	110	115	115
$Prad_{core}$ (MW)	67	43	90	49	36
$Prad_{edge}$ (MW)	35	29	95	27	27
P_{div} (MW)	116	146	105	189	112
static H_{DIV}* (MW/m^2)	20*a	21*b	14*a	75*a	17*b

* static peak power load on divertor plates from [a]present 2-D modelling [b]simpler models, multiplied by a physics safety factor of 3.4 to account for non-uniformities in power flow and physical effects not yet included in the models.

** true ignition; long-pulse operation may require burn control (~25MW control power)

*** ITER H-mode scaling, derated to 75% to take into account the effect of ELM's beta limit. With the expected impurity concentration, Z_{eff} equals 1.66, which, with 14% of the plasma current driven by bootstrap effect, gives a loop voltage of 0.12 volts at 22 MA. Inductive current drive therefore suffices for a burn time of 400 seconds for the reference poloidal field configuration (326 V-s) and the reference plasma internal inductance ($l_i(3)$=0.65). For more peaked current profiles (l_i=0.75), the burn time would be approximately 100 seconds less. In the reference ignition case, the plasma density is maximized, subject to the other constraints, in order to alleviate the divertor conditions. The conditions for this operating mode lead to a total power of 116 MW going to the divertor, to be distributed over top and bottom inner and outer divertor channels. According to present 2D divertor models, this can result in a peak static heat load of 20 MW/m^2. Included in this peak power load is a physics safety factor of 3.4 to take account of asymmetries in power load and model uncertainties.

In the physics phase, initial studies on ignited plasmas can be carried out during periods much shorter than the 400 seconds burn time attainable in the reference ignition scenario. For short periods, before the alpha particle concentration builds up, the confinement requirement can therefore be reduced. This type of ignited operation (e.g. column 2, Table 3.1), at a density of 10^{20}m^{-3} and a helium concentration of 0% requires a confinement time enhanced by only a factor of 1.6 over ITER L-mode scaling and only 50% of that predicted by ITER ELM-free H-mode scaling for these parameters.

Since the density has been maximized for the preceding operating scenarios, the temperature is low, 10 keV, which may represent an unstable operating point (section 2.4.4 of ref. 3.1). For long pulses, the operation mode chosen will therefore be a sub-ignited mode with Q near 40. This operating point will have some slightly different parameters from the true ignition points, and will notably require an average auxiliary heating power of about 25 MW.

The reference long pulse scenario B1 (column 3, Table 3.1) is intended to satisfy minimum wall load requirements for technology phase testing. Clearly, during the physics phase, variants of this scenario are possible with shorter burn times. In the reference long-pulse scenario, the plasma current is only slightly higher than 15 MA, and, for Q=8, an enhancement factor over L-mode of 2.2 is required. The burn duration is lengthened to 2500 seconds, in good part due to the volt-seconds made available because of the low plasma current. In addition, 30% of the current will be driven by non-inductive means, using 115 MW of injected power, and another 30% is expected to result from the bootstrap effect. In order to reduce the power to the divertor plates to the order of 100 MW, additional radiation is provided by the introduction of a modest amount (.07%) of medium-Z impurity. In this reference long-pulse scenario, the plasma current ramp-up is accomplished solely by inductive means. Using non-inductive ramp-up assist, with lower hybrid current drive, higher currents to improve the confinement or higher densities during the burn phase to reduce divertor loading could be attained.

Presently, no steady state scenario can fulfill the requirements for technology testing, because of the conflicting requirements of divertor heat loads, requiring high densities, and of efficient non-inductive current drive, requiring low densities. Nevertheless, a nominal steady state scenario has been defined (column 4, Table 3.1), which almost satisfies the wall load requirements of the technology phase, as a goal for the physics phase investigations. The feasibility of this scenario with a wall load of 0.7 MW/m^2 depends on reducing the peak divertor heat load, increasing the peak power handling capacity by innovative divertor concepts, by a more favorable extrapolation from present experiments, or by a further increase in radiated power. The total power to the divertors is 190 MW, almost twice that which can be handled with confidence. In addition, the peak power load and the plasma temperature in the scrape-off layer is very high because of the low operating density necessary when 70% of the plasma current is to be driven by 115 MW of current drive power. The plasma current for this scenario is 19 MA, requiring a confinement enhancement factor of 2.1 over L-mode for Q=6.7. The plasma beta for this case is 5.4%, i.e. a Troyon factor of 3, which is consistent with the current profile control possible when a large fraction of the current is driven non inductively.

As has been stated, the steady state operation with high neutron wall load described in the preceding paragraph is considered an ultimate goal. Nevertheless, lower neutron wall load scenarios for physics investigations of steady-state operation can be devised which satisfy presently known limits (such as column 5, Table 3.1). A scenario which does not employ impurity seeding to reduce divertor loads with present models is limited to a wall load below 0.3 MW/m^2. At a plasma current of 10 MA, with 50% of the current driven by bootstrap effect, the total power load to the divertors is expected to be about 110 MW. The confinement enhancement necessary for a Q of 2.6 in this case is 2.2, and the ELM-free H-mode confinement scaling still predicts a confinement time almost 40% larger than that required. This operating point is well away from beta and density limits, and represents a typical operating mode for investigating steady-state operation.

3.2. Operational Flexibility

Performance flexibility is essential to enhance the capability of the ITER machine, to provide access for the introduction of advanced features and new capabilities, to accommodate physics uncertainties and to optimize the plasma performance of the device itself during the physics phase. This will be done both for ignited burn and for burn in a driven regime where the plasma current is partially or fully maintained by external power. Reaching sufficiently good energy confinement as well as long and stable plasma operation and reducing the frequency of disruptions to a minimum are the major issues in this context. Therefore the machine must be designed to accommodate modifications for future optimization and to have a capability for extended and flexible operation beyond the standard operating regime.

ITER operation is characterized by its high plasma current and high heating and fusion power with very long pulses. Operational flexibility is therefore especially important as regards the poloidal field system, the plasma facing components and current drive and heating systems. The poloidal coil system is designed to provide not only the reference double null configuration with 22 MA but also a single null configuration with 22 MA as well as the option of higher current operation (up to 28 MA with non-inductive ramp-up assist). Plasma facing components and the divertor throat configuration will be optimized during ITER operation. Scheduled replacement of these components has therefore been included in the design. High fusion power operation up to 2GW is also to be investigated with short pulses.

The machine capability and the optimal range for D/T operation can be determined in the favorable environment of the H/D phase, which eases diagnostics and maintainability. For these investigations, it is essential to have the full current-drive and heating system and pumping system available from the beginning of ITER operation.

3.2.1. Overview of options

The options possible in ITER for plasma operation in the Physics Phase are summarized below:

A. BASELINE:

-maximum plasma current 22 MA,
-maximum fusion power = about 1 GW,
-R=6.0 m,a=2.15m,q=3,k=2 at the 95 % flux surface,
-double null poloidal divertor,
-distance between separatrix and plasma-facing components at inboard/outboard midplane: 14/15 cm,
-distance between X-point and strike point on divertor plate: 1.4 m for outer plate and 0.6 m for inner plate,
-poloidal beta<0.7, l_i=0.55-0.75.

A1. inductive + non-inductive operation:

-pure inductive operation with about 300 s burn,
-steady-state and hybrid operations, Volt-second saving, during ramp-up, very long pulse, and sub-ignition with current profile control.

A2 optimization of power and particle exhaust to minimize damage of plasma-facing components:

-high density/low temperature operation, and/or controlled medium-Z impurity injection to enhance radiative cooling mainly of the plasma edge and scrape-off layer,
-high-Z material for the divertor plates, if viable.

B. FLEXIBILITY AND EXTENDED PERFORMANCE

B1. flexibility

-wide range of operation parameters including:
double, single and semi-double null divertors,
distance between separatrix and first wall,
elongation, triangularity, aspect ratio,
safety factor, and profiles.
-innovative concepts for optimizing power and particle exhaust and minimizing the damage of plasma-facing components:
radiative cooling for reducing the power to the divertor by injecting medium-Z impurity,
-low-Z or high-Z material for divertor plates,
-SN divertor operation to be studied for efficient He exhaust (power exhaust may be problematic).

B2. operations in extended performance:(limited number of pulses with limiting conditions)

B2.1. high current with large plasma volume:

-25 MA,R=6 m, a=2.2 m, k=2 with several tens of seconds with pure inductive drive and a few hundred seconds with non-inductive assist.

-28 MA,R=6 m, a=2.2 m, k=2 with 20 volt-seconds saving with non-inductive assist. (Disruption load is serious.)

B2.2. short burn pulses with higher fusion power for enhancing confinement, study of burn control and power handling: e.g. 2 GW for 10 s (Advances in divertor operation are required.)

C. DEVICE MODIFICATIONS

The possibility of machine modifications enhances the flexibility, especially in the H/D phase before activation. Provision for modification of the plasma-facing components by remote maintenance has been included in the design, so that this remains possible throughout the D/T phase.

In the technology phase, the ITER machine and its operation will be configured for engineering tests The optimized technology phase scenario will be definitely clarified only in the physics phase. The machine will be designed to allow modification of in-vessel components and the plasma configuration after the Physics Phase if necessary. The baseline technology operation has the same configuration as the baseline physics case. Lower plasma current will be employed, and the burn pulse will be lengthened by non-inductive current drive (hybrid operation).

3.2.2. Machine design with flexibility

It is very important to clarify the impact of flexibility on the design of the device and to find a solution in which a reasonably small additional effort ensures the flexibility needed. Major impact on the design is expected if the configuration of the device or the poloidal field system must be modified.

FLEXIBILITY OF BASIC DEVICE CONFIGURATION

Flexibility has been designed into ITER from the outset. For this reason, ITER is divided into: (1) the basic machine whose main systems, the TF and PF coils and the vacuum vessel, are semi-permanent and are not intended to be changed during the life of the plant and (2) the removable in-vessel components, i.e. the first wall, the inboard and outboard blanket segments, and the divertor. These components are segmented for relatively easy replacement. The possibility of in-situ repair (e.g. by plasma-spray) will also be investigated in the EDA.

FLEXIBILITY OF POLOIDAL FIELD SYSTEM

Proper design of the poloidal field system is the key to realizing operational flexibility. The performance of a few coils must be enhanced with respect to the baseline 22 MA design values, but this enhancement is small. The following operational conditions will be realized by the PF system [3.3] and the maximum currents required are listed in Table 3.2:

TABLE 3.2. MAXIMUM PF CURRENTS AND VOLTAGES

Coil No.	Maximum current(MAT) 22 MA operation	extended operation*	Maximum voltage (kV)
PF1 U/L	22.7	22.8	10
PF2 U/L	22.7	22.8	10
PF3 U/L	20.3	22.8	20
PF4 U/L	20.3	22.8	20
PF5 U/L	18.4	18.5(19.4*)	20
PF6 U/L	14.5	16.5	20
PF7 U/L	8.2	9.7	20

*including 0.9 MA induced by disruptions at 28 MA plasma current

-22 MA operation

li=0.55-0.75, β_p=0-1.0,

burn duration=200-400s with pure inductive current drive.

-25 MA

li=0.55-0.75, β_p=0.4-0.8,

burn duration = 50 s without non-inductive assist.

-28 MA

li=0.6-0.7, β_p=0.4-0.6,

burn duration = 50 s with 20 volt-seconds savings by non-inductive ramp-up assist.

-Single and double null divertor configurations.

-Seven independent coils in a half plane have independent power supply system in order to optimize volt-seconds and magnetic forces. The maximum allowable additional force of the central solenoid is estimated to be 150 MN which corresponds to about 60 Vs saving at 22 MA plasma current. 20 Vs saving is needed to give a current flat-top in 28 MA operation.

MAGNET

From the magnet design point of view, there are operating limits on the mechanical, electrical and nuclear loads. The critical limit of normal operation is mainly related to the mechanical design considerations such as fatigue characteristics of the structure and superconductor of the PF magnets, and shear transmission due to the out-of-plane forces [3.4]. The stress level in nominal operation is close to the allowable stresses. Accordingly, operation in the extended condition has to be considered as a limiting loading condition with a limited number of cycles and must be exhaustively analyzed in the engineering design phase.

As regards the PF magnets, the plasmas in extended operation are larger and have a higher plasma current, which is partially offset by reduced equilibrium requirements. The central solenoid works at its allowable limits in both physics and technology phases and is not influenced by the extended mode of operation, unless non-inductive volt-second saving is required. Some extra current capacity in the outer PF magnets has been designed for as discussed in the previous sub-section.

A critical issue is the allowable heat flux on the TF magnets due to the nuclear heating and ac losses. This is expected to lie between 1 mW/cc (20 kW total heat) and 5 mW/cc (100 kW total heat, which is near the limit of the present design). Therefore extended operation with higher fusion power is possible only for short durations with a low duty cycle. Separatrix sweeping increases the ac losses in the coils. A higher range of separatrix sweep is therefore possible at lower fusion powers, corresponding to the steady-state scenarios, than at the high fusion power of ignition scenarios.

FLEXIBILITY OF DIVERTOR OPERATION

Because of the high heat flux and large uncertainties of the scrape-off layer, the plasma facing components, especially the divertor plates, have to be designed for the best possible performance. Limits are placed on the heat flux, on the erosion by normal operation and by disruptions, on the forces exerted on the plates especially by disruptions, and on thermal and mechanical cycling stresses. During the physics phase with short integral operation time and less emphasis on the lifetime of the components, the permitted heat loads and hence the neutron wall load could be allowed to be somewhat higher than in the technology phase. The critical range for static peak heat load is around 15 MW/m^2 [3.5]. After studying various operational condition in the physics phase, the machine will be optimized for the technology phase. Major modifications of plasma-facing components may then be carried out, such as changing the divertor plate material or modifying the divertor configuration (e.g. divertor throat, distance between X-point and striking point, shape of the plate, the angle between plates and magnetic surfaces etc.).

Uncertainties in scrape-off layer physics require flexible divertor operation. As an example, present 2-D modelling (with a perpendicular heat transport coefficient of $2m^2/s$ and no inward pinch) predicts a half-width for radial power flow at the midplane of 5 mm for the reference ignition scenario with 1 GW fusion power. On the inclined divertor plate (at 15 degrees to the magnetic field) the ideal peak power load corresponding to this case is calculated to be 6 MW/m^2, which is to be multiplied by a physics peaking and safety factor of 3.4 to obtain the peak static heat loads quoted in Table 3.1. This factor, reasonable for the standard ignition/high-Q cases, is expected to be too high when the predicted scrape-off layer is very narrow, as in the lower-density steady-state cases. Even in the ignition case, however, the possible heat loads may exceed the practicable limit for the static heat load. The effect of the large heat load can be reduced by active means to increase the effective half-width of power scrape-off and active means to decrease the asymmetries inherent in the physics peaking factor. The following schemes will be employed in ITER:

a)Toroidal asymmetry in an actual device can cause $\gtrsim$ 1mm broadening of the midplane. If necessary, an additional ergodic layer can be applied with the in-vessel copper coils set for vertical plasma position control, i.e. about 6 cm/20 kAT on the outer divertor plate. In this case, the copper coils have to be divided into about 4 groups in the toroidal direction so that a toroidally rotating field with toroidal mode number of 1 can be produced.

b)Separatrix sweep with SC coils will be employed to produce a possible factor of two improvement in acceptable static peak heat flux. Presently, a sweep amplitude of +/-15 cm on the outer divertor plate is envisaged at 0.2 Hz for

TABLE 3.3. ITER DIAGNOSTIC SYSTEMS

Diagnostics for the physics and technology phases	
	Magnetic diagnostics
	Fusion product diagnostics
	Interferometry and polarimetery
	Bolometer arrays
	ECE diagnostics
	Langmuir (fixes and movable) and calorimeter probes
	Tile markers
	Infrared and visible inspection periscopes
	IR thermometers
	Plasma facing component thermocouples
	Pressure gauges and residual gas analyzers
	Thomson scattering system
	Collective Thomson scattering system
	Microwave reflectometry
Additional diagnostics for the physics phase	
	Spectroscopy (visible, XUV/VUV, X-ray)
	CHERS
	Motional Stark effect
	Neutral particle analysis
	Photo-electric detectors
	Synchrotron radiation analysis (runaway electrons)
	Ion cyclotron emission probes
	Pellet measurements
	Blanket diagnostic system

the reference ignition scenario, Under steady state operation, the fusion power is relatively low so that a higher heat load due to separatrix sweep can be accepted without increasing total heat load on the SC coil system.

c)Up/down asymmetry is expected because of vertical movement and drifts. A control accuracy of +/-5 mm or better of vertical position is necessary and appears feasible. If the up/down asymmetry is serious, the plasma can be artificially moved up and down, e.g. +/-1 cm, so that the time averaged heat flux is distributed equally to the top and the bottom plates by optimizing the cycle fraction.

d)Radiative cooling in the peripheral region of the main plasma and the divertor region mitigates divertor heat load. Radiation of the order of 10 MW from one divertor channel is expected.

e)Maximizing the density in the scrape-off layer by producing a density profile as flat as possible in the main chamber and/or by gas feeding in the divertor. The higher density acts strongly to widen the scrape-off layer.

If high heat flux handling is achieved, a wide range of plasma operation is accessible to study. Although narrowly localized erosion can be eliminated easily by slowly sweeping the strike point on the divertor plate, additional schemes will have to be developed to reduce impurity contamination. These problems will have to be intensively studied in the engineering design and construction phases and finally during ITER operation.

3.3. Diagnostic capability

The ITER objectives of establishing the physics and technology data base for designing a demonstration fusion reactor, require that very reliable and detailed measurements of the plasma behavior for all phases of operation should be achieved. This demands an extensive and well coordinated set of diagnostics providing appropriate spatial and temporal resolution. During the physics phase these diagnostics will be needed to provide the data for the exploration and optimization of various modes of operation. The goal will be to reach ignition in one or more of these modes, and subsequently to explore the ignited regime, both to establish the limits to the operation and to advance the understanding of the physics of an ignited plasma. In addition to providing information for the physics understanding of the plasma behavior, the plasma diagnostics on ITER must provide feedback control of plasma parameters. All these diagnostics must be available from first low power ohmic discharges through to the full burn condition with alpha-particle heating.

Several types of abnormal behavior of the discharge can be expected in ITER that may result in damage to the machine (e.g., uncontrolled rise of the fusion power, generation of a large amount of high-energy runaway electrons during disruptions, generation of high-poloidal currents in first wall elements during the vertical displacement event, local increase in the heat load on divertor plates). Development of safety diagnostics which can detect the approach of these events and creation of rapid control techniques to cope with them are of primary importance for ITER. At this time, however, the detailed control requirements and response to off-normal events are not well specified. The Physics R&D program on operational tokamaks must improve the definition of the control requirements.

The diagnostics proposed [3.6] for safety, control and plasma performance evaluation are listed in Table 3.3. Diagnostics for control include (i) magnetic loops for plasma current, plasma position and shape, (ii) interferometry for electron density, (iii) neutron spectrometry for ion temperature and for the fuel composition (i.e., n_D/n_T ratio), (iv) bolometers for radiative loss, (v) electron cyclotron emission (ECE) and magnetic loops for disruption precursors and, (vi) infra-red detectors and thermocouples for divertor and first wall temperatures. All these diagnostics must be radiation-insensitive, extremely reliable and remotely maintainable.

The ITER diagnostic system has to be capable to provide detailed information on stability and confinement of ignited plasma. A substantial amount of data will be provided by the control diagnostics. However, additional information will be necessary for physics understanding and plasma optimization. Radial profiles of electron temperature and density will be measured by a multipulse Thomson scattering (LIDAR) system along three tangential chords. The line-averaged electron density measured along the same chords by laser interferometers will be used for density control and for LIDAR data calibration. Since ITER will probably be the major device with the mission of broad physics and engineering studies of ignited DT plasma, it has to be equipped with a complete set of the fusion product diagnostics. This set will consist of 8 systems which allow to determine (i) the absolute and time dependent neutron yield (i.e., fusion power), (ii) radial neutron intensity distributions in two directions, (iii) 2-D distribution of gamma-ray intensity (for fusion reacting rate in D-He plasma), (iv) neutron energy spectra profiles, (v) the ratio of 12-MeV to 2.5 MeV neutron fluxes (for n_T/n_D ratio), and (vi) the energy spectra of slowing-down alpha-particles (by collective Thomson scattering and, possibly, via double charge exchange with injected 100-keV He atoms). Measurement of electron cyclotron emission (ECE) will provide the electron temperature with good spatial and temporal resolutions. This technique can be used to diagnose the core and the edge, and in principle can provide information on the divertor plasma and suprathermal electron populations. The Langmuir and calorimeter probes and spectroscopic markers will be used for real-time measuring of the electron density and temperature in the scrape-off layer and the power load and rate of erosion at selected points on the first wall. In addition, methods for determining the profile of the current density and the helium concentration in the core and divertor regions must be provided.

It should be noted, however, that the possibilities offered by the reference design do not presently match all of the requirements for physics implementation of the diagnostics at the tokamak. In particular, no satisfactory solution was found for real-time monitoring of the divertor plate surface which is a particularly critical issue. Hence, changes in design of the horizontal ports, of the top and pumping duct access, and of the blanket and divertor structures at a number of locations will be necessary during the Engineering Design Activity.

3.4. Commissioning and Design Code Validation Tests

For commissioning and initial physics operation, ITER will not yet be activated. This is followed by a short low-activation phase, and finally a high-activation phase with D/T operation. The engineering test program is developed in accordance with the activation level of the vacuum vessel and in-vessel components. The commissioning tests for ITER preceding the exploratory physics operation will establish the adequacy of the design to meet the operating requirements for the physics and technology phase. Additional testing of the machine components will be used as a method of validating and calibrating the design codes that were used in the engineering design. In order to effectively validate the accuracy of the codes physics data that corresponds to the true plasma, electromagnetic and radiation loads will be needed, as input. The other needed input are the measured responses to the loads by the ITER systems. Actual physics data combined with measured thermal and mechanical responses of the engineered systems will allow calibrations of design codes based on actual machine performance.

Based on calibration of the design codes better predictions of the lifetime performance of ITER are possible. Increase in performance may then be permitted by using more of its actual design margin or by making minor improvements in hardware or operational scenarios. The calibration and subsequent modification of the design codes will also improve the predictions of the performance of experiments beyond ITER such as the DEMO reactor.

During the first phase (zero activation) the various engineering systems will be tested. The engineering tests will serve the purpose of validating the process followed to design, build, and operate ITER as a prototype of a fusion reactor. In particular the design codes, the technological processes, the manufacturing procedures, the modelling of components and plasma will be validated. As a result, the engineering data collected during the operation of ITER will be of great value for establishing the design, and lifetime, as well as reliability and availability estimates for DEMO and future fusion reactors.

VACUUM VESSEL

The Vacuum Vessel has three major requirements to fulfill:

- To maintain the high grade vacuum necessary for plasma operations
- To act as primary containment of the tritium during normal operations and in case of accident in the plasma facing components
- To contribute to the plasma vertical stabilization
- The normal operating conditions of this component include baking and plasma disruptions.

The first bakeout will require a longer time than the design value so that the behavior of the whole system can be tested and the homogeneity of the heating and the stresses induced in the structure verified. All these data will be necessary to validate the calculation models used during the design.

During plasma disruptions the response of the complete structure will be verified. The induced currents,forces,displacements and stresses (in both rigid sectors and resistive elements) and the voltages across the gaps and the electrical break will be measured and compared with the calculated values.

The coolant flow, temperature, and pressure will be monitored in all operating conditions.

The outgassing of the vessel will be checked by monitoring both baking and pumpdown performances. This will enable a more accurate estimate of the chemical species retention in walls, insulators, and gaps to be made and a comparison with the predicted values to be performed.

The behavior of the insulation inside the parallel segment will be checked during plasma disruptions as well as the changes in performance due to deterioration caused by mechanical and nuclear loads.

The tests to be carried out on the vacuum vessel also include the interfaces with the adjacent systems. In this case the most interesting information will come from the measurement of the forces transmitted by the in-vessel components to the vacuum vessel by attachments and support points.

CRYOGENIC SYSTEM

The behavior of the whole system (i.e.stresses, displacements) during the cooldown will be established. The homogeneity of cooling will be verified as well as the existence of local hot spots which could cause an increase of temperature of the coil and therefore a premature thermal quench.

The overall value of thermal load on the coil will also provide information on the performance of the refrigerated shield (i.e. overlapping of the thermal barriers, conductivity of the supports) on the cold structures and at the magnets supports.

TF AND PF COILS

The major uncertainties on TF and PF coils are linked to their manufacturing features and procedures. Therefore the most critical questions will be answered before (on prototypes) and during the manufacturing of the coils. The main questions to be addressed will concern the control of the manufacturing procedure to permit the fabrication of hundreds of kilometers of high quality superconducting cable. Materials problems will also be addressed,mainly those concerning the insulator.

Some questions will still require a more precise quantification from the operation of the machine. One is the evaluation of nuclear radiation effects on the coil electrical insulation and the overall behavior of a superconductor cryogenic cooled cable in a nuclear environment.

Additional information on the TF coils will include AC losses, out of plane forces and displacements due to the pulsed PF fields. In the case of the PF coils,tests will be carried out on the effect caused by plasma disruptions (i.e.induced currents).

An important number of tests will concern the protection systems and the power supplies. Once the systems are cooled down,first a low current will be put into the coils to test all detection and protection systems at low energy.

A simulation of electrical black out involving also the cooling supply will be carried out,including the coil energy being dumped into emergency resistors.

The PF and TF power supplies will be tested for normal operation and protection in case of fault conditions.

PUMPING SYSTEM

Verifications will be carried out on the He pumping efficiency and on the DT co-pumping. The exhaust species impurity composition will be monitored in order to validate the calculation models and also because of its impact on the fuel purification system.

Dust transport and its impact on pump and valve operations will be checked.

During machine operations the modularity of the system will be tested by switching off some of the vacuum pumps to reduce the pumping rate.

FUELLING SYSTEM

The adequacy of the gas puffing response time and its control system will be tested.

The adequacy of pellet velocity and repetition rates required both during ramp-up and burn will be checked. Reliability and availability of the system will be assessed.

BLANKET TRITIUM RECOVERY

An important task will be to check the purge gas flow rate and the hydrogen content required for an efficient tritium recovery.

The tritiated water concentration and quantity to be processed is also a value to be verified.

HEATING AND CURRENT DRIVE SYSTEMS

The tests of the auxiliary heating system for hydrogen plasmas will be done with gradual increases of heating power up to the design level. Also the experiments with non-inductive ramp-up will be done in this period.

Heating input at different plasma radii will be explored in both electron cyclotron (change in intake angle) and lower hybrid (change in the radial stroke of the launcher). Therefore, coupling studies between rf waves and plasma can be performed based on these parameters. Optimization of parameters for different plasma configurations will be checked and quantified.

Experiments on modulation of the injected heating power will be carried out in conjunction with different start-up scenarios.

In experiments with DT plasmas,physics studies will continue with non-inductive current drive experiments (hydrogen plasma,approx. 6 months). The main objectives of the CD/heating systems in this period are as follows:

(1) performance of RF/NBI systems under initial steady state operation

(2) heating/CD system tests and operation in D-T plasma

(3) testing of RF/NBI components under fusion radiation conditions (with tritium plasma).

DIAGNOSTICS

The diagnostics will be checked for correct operation under reactor operating conditions and the final adjustments will be done during the first shots. The data acquisition system will also be checked. Special emphasis will be given to verifying the reliability of the diagnostics required for safety and machine control.

MAINTENANCE

The procedures and the tasks for both ex-vessel and in-vessel maintenance operations will have been tested, prior to the start of operations, during the final assembly, by installing one sector of the reactor remotely.

Experience will have to be gained for all the major maintenance operations involving both plasma facing components (i.e.diverter removal and handling,FW tiles replacement) and semi-permanent components (i.e. blanket segment removal,vacuum vessel segments and TF coil replacement). Important data will come from remote operations (welding and cutting) on the structural welds between vacuum vessel segments and, more generally,from the re-welding of irradiated materials.

Alignment systems for large components and the impact of the achieved accuracy on the their performances will be tested and analyzed in the real environment.

Intervention and operations in some areas will be simulated in order to check the access and the procedures to reach some components in a real reactor assembly.

The In-Vessel Viewing System will also be tested and operated between shots in order to check procedures and features in the real working environment.

PLASMA ENGINEERING

Plasma engineering will be the system most extensively tested once the machine begins to operate. The codes used for elaborating different plasma configurations and scenarios will be checked and validated during operation.

The predicted plasma control parameters will be checked for consistency and the plasma tarsnient behavior will be compared with that predicted by the codes.

Different plasma positions at start-up and all possible plasma configurations (i.e.circular,SN,DN) will be tested. Different start-up scenarios will be performed with and without the assistance of RF. Separatrix sweeping tests will be also carried out at low heat loads.

The parameters will be optimized to achieve disruption-free operation as far as feasible. Whenever disruptions occur nevertheless, their influence on the whole machine and theexistence of the predicetd eddy currents will be verified.

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4. OPERATIONS AND RESEARCH PROGRAM - TECHNOLOGY PHASE

4.1. Testing Requirements

ITER will play a critical role in the development of components and systems for fusion reactors. One of the major programme objectives of ITER is testing of neutronics, materials, tritium production/extraction, blanket modules and sectors and advanced plasma technologies. Also an important objective of testing is extraction of high grade heat from reactor relevant modules and sectors. To obtain test information suitable for fusion demonstration reactors requires appropriate scaling of the testing conditions. The minimum neutron first wall load and fluence for DEMO are ~2 MW/m^2 and ~10 MWa/m^2, respectively. Scaling by about a factor of two in power density and, possibly, by a higher factor in neutron fluence is reasonable. The majority of ITER parameters will most likely be lower than those DEMO and commercial reactors. If the test modules are designed to "look like" components in DEMO and commercial reactors, temperatures, stresses and other operating parameters are reduced and information from the tests is generally not useful. Therefore, "act-alike" modules have been designed using engineering scaling to preserve important phenomena so that data from tests at "scaled down" conditions can be extrapolated to reactor conditions. Engineering scaling involves altering physical dimensions (e.g., increasing the thickness of a solid breeder plate in a blanket to increase temperature differences and thermal stress) and changes in operating conditions (e.g. reducing the mass flow rate of the coolant to maintain coolant temperature rise). However, there are limits to engineering scaling. Therefore, there are minimum values for the major device parameters below which the test information is not useful, because results can not be extrapolated to reactor conditions.

Engineering scaling requirements are different for the various issues. In general, it is found that it is nearly impossible to design an "act-alike" module that can simultaneously provide testing for all the issues. Thus, several "act-alike" modules are generally required, with each one properly scaled to obtain useful test information for a subset of the technical issues.

Test requirements have been considered in two main categories: requirements on parameters and requirements on engineering. The latter category includes concerns such as testing space (total space, configuration of space), ancillary systems, and maintenance and handling requirements. These are treated in the section 4.2 in this report. Test requirements on the major device parameters are the subject of this section.

The most important parameters which affect the value of testing are neutron wall load, neutron fluence, and time-related parameters (burn time, dwell time and continuous operating time). The requirements are based on extensive analysis of the behaviour of nuclear components as a function of these parameters. Table 4.1.1. summarizes the recommendations. Minimum values are determined primarily from analysis of the important blanket phenomena under scaled conditions. One can show that test device parameters below the minimum value in any category will seriously limit the usefulness of nuclear testing for at least one identifiable phenomenon. There is a high probability that results could not be extrapolated to reactor conditions under

TABLE 4.1.1. NUCLEAR TESTING REQUIREMENTS - SUMMARY OF RECOMMENDATIONS AND REFERENCE VALUES

Device Parameter	Minimum needed for scalable tests	ITER Conceptual Design Parameters (Technology Phase)	Desirable Extensions
Average neutron wall load at the test module, MW/m^2	≥1	1.2	2
Number of ports	5	5	7 (plus segement or sector)
Minimum port size	2-3 m^2	3.74	segment or sector
Total test area	10 m^2	18.7m^2	20-30m^2
Plasma burn time	≥1000 s	2500 s**	1-3 hrs (to steady-state)
Dwell time	*	200-400 s	*
"Continuous" test duration	≥1 week		2 wks
Number of "continuous" tests per year	2-3		~5
Average availability	10-15%	18%	25-30%
Annual neutron fluence (at the test module), $MW\text{-}yr/m^2$	0.1	0.19	0.4
Total neutron fluence (at the test module), $MW\text{-}yr/m^2$	≥1	1.53	2-4

* Minimum acceptable dwell time is highly dependent on the design concept, and is difficult to specify. Further analysis in this area is recommended.

** Alternate plasma scenario B6 (col. 4, Table 3.1) provides for steady operation.

these circumstances. Conversely, the desirable ranges provide values above which there is confidence that the extrapolation to reactor conditions is straightforward. The desirable values are determined partly from analysis and partly from engineering judgement. While testing at prototypical values is clearly more beneficial, most nuclear issues can be resolved if the desirable parameters are achieved in ITER and many can be adequately investigated if the minimum parameters are exceeded.

Table 4.1.1 also shows the reference parameters for ITER in the long-burn hybrid operating scenario (B1). In every case, the design of ITER meets or exceeds the minimum values.

TIME-RELATED PARAMETERS: BURN, DWELL, AND CONTINUOUS OPERATING TIME

Steady-state operation is a highly desirable ultimate goal for ITER during the technology testing phase. Pulsing has several negative effects on testing, creating difficulty in obtaining and sustaining equilibrium conditions. The current pulsing options for ITER indicate that dwell times (times between burn pulses) will be long enough in all cases to drop the test module conditions substantially away from equilibrium. Therefore, attaining a "true" equilibrium through a series of sequential pulses is difficult. In that case, it is more important to extend the burn time than to attempt to maintain a short dwell time. Longer burn times also lead to fewer net cycles, which is advantageous for extending the machine lifetime. However, very long dwell times may reduce the duty cycle, leading to an unacceptably low rate of fluence accumulation. If the dwell time is large, then the burn time should be much longer than 1000 s, and should approach or exceed 3000 s. Continuous test periods with high availability of 1-2 weeks have been shown to be desirable and practical to achieve with the assumed device availability goal. Most important tests can be completed within this amount of time.

NEUTRON FLUENCE

The fluence recommendation is based on a combination of the need to perform a sequence of concept performance tests, which take roughly 3-6 years at full power and high availability (~25%), resulting in 1-2 MWa/m^2 of fluence, and the desire to perform concept verification tests, which require activation of fluence-related phenomena, resulting in 3-5 MWa/m^2 of fluence. Table 4.1.2. provides a summary of fluence effects on blankets. Also, the current test schedule provides for extensive use of sequential testing. Most tests have to be inserted and removed over periods ranging from 1-3 years.

NEUTRON WALL LOAD

The minimum acceptable wall load depends primarily upon two factors:

(1) Heat sources are directly proportional to the wall load. Most thermomechanical and tritium-related phenomena in nuclear components strongly depend on temperature profiles, which in turn are determined by the heat sources.

(2) The ability to achieve adequate fluence exposure to test modules in a reasonable amount of time requires relatively high wall load and high availability.

Past studies suggest that a wall load in the range of 1-2 MW/m^2 is adequate for thermomechanical and tritium testing. Useful testing at reduced wall load (relative

TABLE 4.1.2. SUMMARY OF FLUENCE EFFECTS ON BLANKETS

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

- **0.1-1 MW-yr/m^2** (at test module)
  Several important effects become activated in the range of 0.1-1 MW-yr/m^2
  - Radiation creep relaxation
  - Solid breeder sintering and cracking
  - Possible onset of breeder/multiplier swelling
  - He embrittlement

  Correlation of materials data with fission reactors and 14 MeV sources can be done with 1 MW-yr/m^2

- **1-3 MW-yr/m^2** (at test module)

  Numerous individual effects and component (element) interactions occur here, particularly for metals, e.g.:
  - Changes in DBTT
  - Changes in fracture toughness
  - He embrittlement
  - Breeder burnup effects
  - Breeder swelling
  - Breeder/clad interactions

---

to DEMO) is made possible by altering the design and operating parameters of the test modules. Generally, bulk average temperatures are easy to maintain by varying the coolant speed and controlling the amount of heat removed through the heat exchanger. Temperature gradients within components are much more difficult to maintain. Some control over temperature gradients can be obtained by changing the thickness of blanket elements. However, if sizes are changed by more than a factor of 2-3, new effects may arise and the overall geometry may become less representative of a real reactor component. Surface heating is an important aspect of thermomechanical performance, and care must be exercised to maintain prototypical ratios of surface to bulk heating.

4.2. Nuclear Component Technology Tests

4.2.1. Test program description

This section describes the test program developed for ITER. The nuclear component tests considered to date are mostly related to the blanket and materials. Surveillance testing of plasma facing materials and blanket/first wall materials of the basic machine is planned in the Physics and Technology phases. Other tests such as those for high heat flux components are equally important and will be considered in more details in the future.

ITER has been designed to operate in two phases.The Physics phase which lasts for 6 years, is devoted to the machine checkout and physics testing. Some useful technology tests are also planned during the Physics phase.The technology phase lasts for 8 years and is devoted primary to the technology nuclear and engineering tests. There are a number of especially designed ports on ITER that are allocated exclusively for technology testing. The numbers of ports available are 3 during the Physics phase and 5 during the Technology phase. The allocation of the 3 ports during the physics phase is as follows: one port for neutronics tests (including possible sharing with some materials tests), one port for liquid metal blankets (both self cooled and separately cooled), and the third port for all types of solid breeder blankets, (gas cooled and water cooled).

During the technology phase the 5 ports are allocated as follows:

1) one port for solid breeders, gas cooled,
2) one port for solid breeders, water cooled
3) one port for self cooled liquid metals,
4) one port for separately cooled liquid metals, and
5) one port for material and other types of tests.

A strategy for allocation of these ports among parties has been developed. It involves a combination of collaboration on some tests, and allocation of testing space and time by party.

TEST SCHEDULE

Device parameters for the Technology Phase are based on the reference long-pulse hybrid operating scenario, and are listed in Table 4.2.1. The wall load at the test port has an average value of ~1.2 MW/m^2. The minimum achievable dwell time is ~200 s, and may be as high as 400 s or more. The reference plasma burn time is 2500 s, but other options for plasma steady state operation during the Technology Phase are being explored. The availability of 18% is a minimum value which is required to achieve the goal fluence within the 8-year technology phase, and is based on operation only at the reference conditions

The overall test schedule is shown in Fig.4.2.1. During the physics phase, liquid metal blanket tests, solid breeder blanket tests, neutronics and material tests will be performed. The tests in the liquid metal port are focussed on MHD and MHD/thermalhydraulics due to the effect of the ITER magnetic field on liquid metal flow patterns and heat transfer. In addition, time is allocated for ancillary system check-out in preparation for the technology phase. The solid breeder port is to be used for system check-out and environment characterization for both water cooled and

TABLE 4.2.1. TECHNOLOGY PHASE REFERENCE PARAMETERS

Device parameter	Reference Value
Average neutron wall load, MW/m^2	
- device average	0.8
- at the test module	1.2
Number of ports	5
Port size	3.74 m^2
Total testing area	18.7 m^2
Plasma burn time	2530 s
Dwell time	>200 s
Technology testing phase duration	8 years
Average availability required	18%
Total neutron fluence, $MW\text{-}a/m^2$	
- device average	1.02
- at the test module	1.5

helium cooled concepts. The neutronics port will be used for tests of shielding performance and tritium breeding for all types of blankets. A small portion of this port will be allocated for plasma exposure of material test samples.

During the Technology Phase, five ports will be available for blanket and material testing. The general approach to blanket testing is to first perform screening tests or short-term performance tests of several designs using sub-modules. The lead designs would then be selected for extended performance testing to determine their potential for use in advanced reactors. Finally, DEMO candidate blankets would be tested using full segments. The materials tests would consist of irradiation of many small samples in a well characterized environment. The tests would be conducted at different temperatures and neutron fluences, and the samples would be removed or replaced at relatively frequent intervals.

NEUTRONIC TESTS.

Neutronics issues include (1) the demonstration of tritium self-sufficiency for the various test blankets, (2) verification of the adequacy of current neutron transport codes and nuclear data in predicting key parameters such as tritium production rate, heating rate, gas production and activation, (3) verification of adequate radiation protection of machine components, as well as adequate protection to personnel, and (4) confirming the safety factors implemented in the design of the shield system to account for streaming through gaps and penetrations.A number of these tests will be carried out prior to the introduction of the test blankets in the machine.

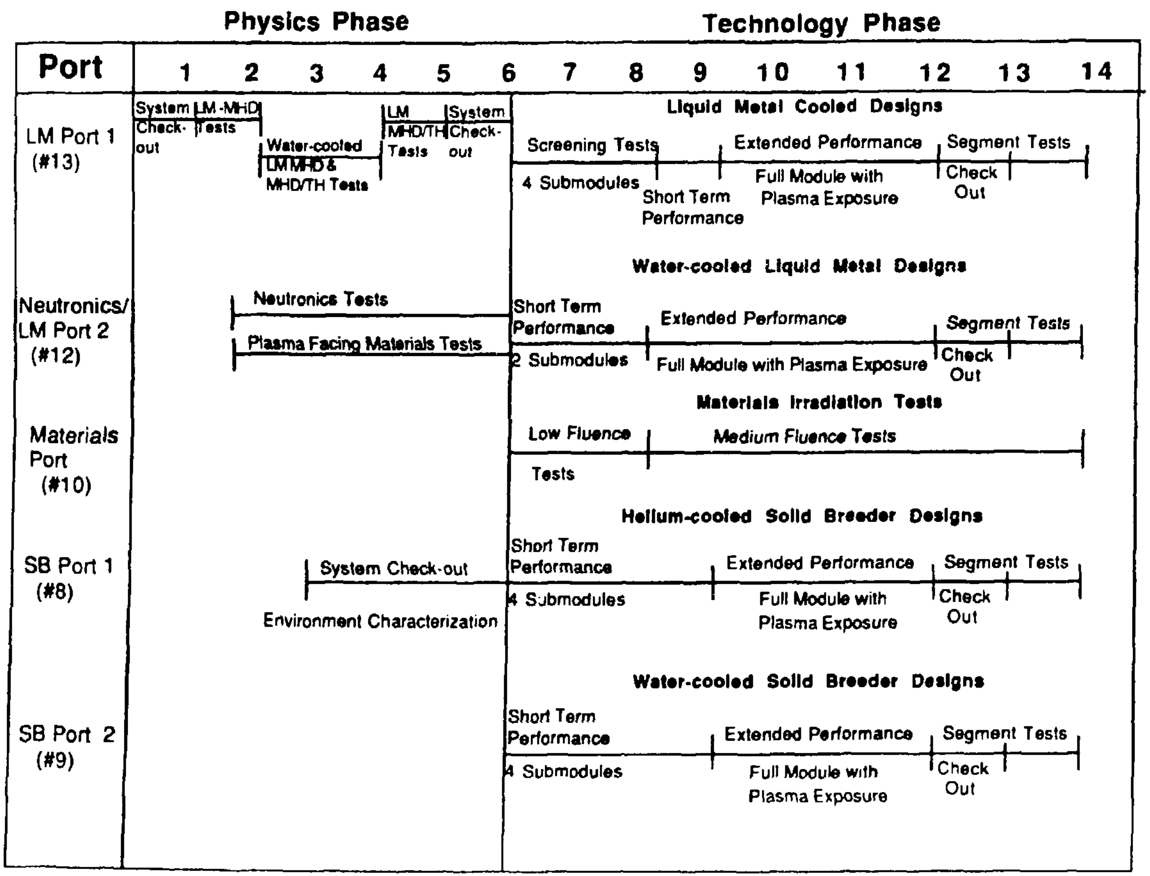

FIG. 4.2.1 - Blanket Test Schedule

Neutronics tests needed to resolve these issues can be classified into three categories.

Dedicated Neutronics Tests.These tests aim at examining the accuracy in predicting key neutronic parameters in the fusion environment. The goal is to identify the source of discrepancies between the analytical predictions and the experimental data related to parameters as tritium production rate, nuclear heating, induced radioactivity and decay heat.While a small size submodule (~ 0.3m x 0.3m) could be used for predictive capability verification, a full test module (3.4m x 1.1m x 1.5m) is preferred.These measurements will require, in general, very low fluence, from 1 W•sec/m^2 to 1 MW•sec/m^2, and thus are suited for early stages of reactor operation. Exceptions are the gas production rate and activation rate measurements, which require larger fluence, up to 0.2 MW•a/m^2.

Tritium Self-Sufficiency Tests. It will be necessary in ITER to rely on indirect demonstration of tritium self-sufficiency through extrapolation from the operation of the driver blanket and test elements.A segment test, rather than only a module test, will be useful because there are strong poloidal variations in tritium production rates. One main parameter to be measured will be the tritium contained in the purge gas in solid breeder blankets or from the liquid metal traps in the self-cooled liquid metal concept after saturation is reached.

Neutronics Measurements for Engineering Performance Tests. These measurements are intended to provide the source terms for non-neutronic tests, in particular those related to the tritium recovery and thermo-mechanics tests foreseen in the test ports used for the various test blanket investigations.Time requirements are the same as for dedicated tests.

Neutronics Measurements for the Basic Machine. As mentioned a number of measurements described will have to be carried out already during the commissioning of ITER for full D-T operation. They include measurement of the afterheat level, accumulated activation level and personnel exposure level behind the shield. Of interest are the measurements of neutron and gamma fluxes, energy deposition and leakages across and behind the driver blanket and shield. Tritium self-sufficiency measurements of the driver blanket are also foreseen.

One port will be allocated mostly for neutronics tests during the last years of operation in the Physics Phase (D-T operation).The possibility to exploit the test modules foreseen in the other ports for specific neutronics measurements should be envisaged.The tests for tritium self-sufficiency will require a full segment test. These measurements will be done at the end of the Technology Phase for the test blankets.

LIQUID METAL BLANKET TEST PROGRAM.

The liquid metal blanket test program in ITER includes self-cooled, separately cooled, and water cooled concepts. The self-cooled designs use either pure Li or Pb-17Li as the coolant/breeder, the separately cooled design uses liquid Pb as the coolant with a Li breeder, while only Pb-17Li has been proposed for the water cooled concept. The approach to blanket testing results from the limited amount of test space and test time that is available in ITER, along with the desire to screen several different designs. The following approach is proposed:

-Make extensive use of all testing possibilities outside of ITER. Nearly all separate effects relevant to a blanket design can and should be investigated in the test loops.

-Perform as many tests as possible during the ITER physics phase. This phase is especially suited for MHD tests and MHD-thermalhydraulics tests since these tests do not require long, repeated burns. Some of the tests can be performed even without a burning plasma. For availability reasons, all these tests will be performed with modules which are not exposed to the plasma. It is proposed that electric heaters be installed at the front surface of the modules in order to simulate plasma surface heating.

-Divide the test port into parts in order to conduct parallel tests during the first years of the technology phase. It is proposed to test four submodules for liquid metal cooled blankets and two submodules for water cooled blankets during the first couple of years of the technology phase to screen a number of designs using either Li or Pb-17Li.

-Perform sequential tests with full size modules during the second half of the technology phase. It is expected that the screening tests will reduce the number of concepts to be tested. Therefore, two to three designs for the liquid metal cooled and one or two designs for the water-cooled concepts will be tested using full size modules.

-Test segments of one or two liquid metal blanket designs towards the end of the technology phase. These tests are highly recommended to test the integrated performance of the blankets that could be installed in a DEMO reactor.

The test schedule for liquid metal modules testing is shown in Fig. 4.2.2. During the Physics phase a single test port will be shared between the self-cooled, separately cooled, and water-cooled concepts. The type of tests to be conducted in the physics phase are system check-out tests, MHD tests and MHD-thermalhydraulic tests. During the first two years the self-cooled blanket will be tested, with the first year devoted to check-out tests, and the second year devoted to MHD tests. For the next two years, the water cooled blanket will be subjected to similar tests as well as neutronics and thermal hydraulic tests. During the fifth year of the physics phase, a self-cooled module will be installed to conduct MHD-thermalhydraulic tests. For all these tests, full modules (1x3m) will be used, and they will be completely enclosed in a separately cooled shell with no first wall exposure to the plasma. In the case of the liquid metal cooled concepts, the modules may operate at low temperatures and pressures and use NaK for the liquid metal. During the last year of the physics phase, the systems required for advanced module and sub-module testing will be installed and checked-out. Six sets of ancillary systems (4 sets for self-cooled/separately-cooled and 2 sets for water cooled), each capable of testing a full module, are to be installed. No further replacement of ancillary systems should be required until full segment tests begin.

During the technology phase, tests for self-cooled/separately-cooled and water-cooled concepts will be conducted in parallel. In both cases, the first two years of the Technology Phase are devoted to screening tests using sub-modules. Two to four sub-modules will be tested in parallel in each port. These tests will be conducted at high temperatures using the actual materials (coolant, breeder, and structural

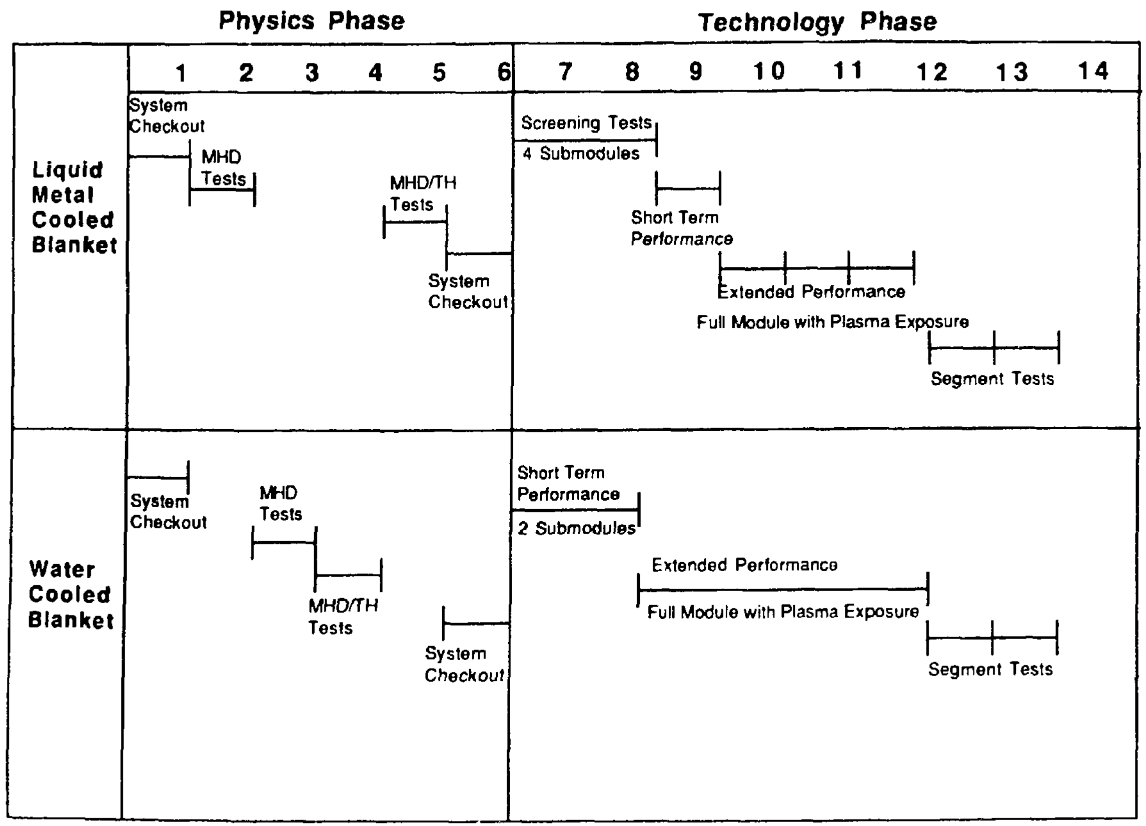

FIG. 4.2.2 - Liquid Metal Test Schedule

material) for advanced blankets. Short time tests are to be performed to examine the overall performance of different designs.

Following the screening tests, the lead designs will be selected for extended performance testing. For the water-cooled designs, full module testing with the first wall exposed directly to the plasma will begin immediately. In the case of the self-cooled designs, there is an intermediate test period with 1/2 size modules. These tests will be conducted with the first wall exposed to the plasma, and they will take place over a one year period. The self-cooled test program then moves to tests of full modules. The aim of the extended performance tests is to select the best candidate design for possible use as a DEMO blanket.

During the last year of the technology phase, segment tests are scheduled. Detailed design description of the test module is presented in [4.1].

<u>SOLID BREEDER BLANKET TEST PROGRAM</u>

Among the ITER participants, the European Community (EC), Japan and the USA are active in the development of a Demo or power reactor relevant blanket with a helium cooled solid breeder, while the water cooled version is investigated by Japan, the USA and the USSR.

The main technical issues for the development of these blankets are summarized below:

a) <u>ceramic breeder material</u> (Li_2O, Li_4SiO_4, $LiAlO_2$ either as pebbles or as pellets):
- tritium transport (tritium residence time)
- lithium transport in presence of temperature gradients
- mechanical stability
- compatibility with beryllium and structural material
- thermal conductivity

b) <u>Structural material</u> (austenitic and ferritic steels, molybdenum alloy, SiC):
- mechanical stability (embrittlement, swelling)
- compatibility with beryllium and ceramic breeder

c) <u>Beryllium multiplier</u>
- mechanical integrity (embrittlement swelling)
- compatibility with structural material and ceramic breeder

d) Behaviour of the <u>blanket structure</u> under high neutron fluences and temperatures, stationary and cycling thermal stresses and other stresses.

The overall R&D testing strategy including testing in ITER for the solution of these problems is based on the following approach:

a) out-of-pile and in-fission-reactor measurements of the relevant basic properties of the materials. Especially for the metallic structural materials and the beryllium it is necessary that experiments with a high intensity dedicated neutron source are carried out to investigate the effects of the 14 MeV neutrons.

b) exhaustive out-of-pile testing (thermal cycling, flow distribution and others) of the blanket structures, starting from small modules and going to more complex ones.

TABLE 4.2.2 - MAJOR DESIGN PARAMETERS

Type of Blanket	EC-BOT-He	EC-BIT-He	JPN-BOT-He
Thermal Power (MW)	2.4-4	2.4-4	5.9
Neutron Wall Loading (MW/m2)	1.2	1.2	1.2
Surface Heat Flux (MW/M2)	0.15	0.15	0.15
Structural Material	MANET (316SS)*	MANET (316SS)*	Ferritic/martensitic steel (Mo-alloy)**
Module Configuration	BOT	BIT	BOT
Breeder	Li4SiO4	LiAlO2	Li2O
6Li Enrichment (%)	90	90	30
Form	Pebbles (0.35-0.6 mm dia.)	Annular Pellets	Pebbles (<1 mm dia.)
Temperature Control minimum	Coolant Inlet Temperature	Coolant Inlet Temperature	Coolant Inlet Temperature
maximum	Breeder Pebble Layer Thickness	Pellet Thickness	Cooling Tube Arrangement
Operating Temperature (°C)	380-720	420-590	450-600
Multiplier	Beryllium	Beryllium	Beryllium
Form	Plates	Blocks	Pebbles (<1 mm dia.)
Local Tritium Breeding Ratio	1.5	1.4	1.6
Tritium Recovery	He Purge	He Purge	He Purge
Coolant	He	He	He
Pressure (MPa)	6-8	6	9
Inlet/Outlet Temperature (°C)	250***/450	250*** /520	360***/480

* : used for preliminary tests
** : used in later stage of tests depending on the material development
*** : Coolant will remove the heat from the first wall and raise its temperature before flowing into the blanket region.

c) tests in ITER of submodules, modules and possibly segments of the blankets. ITER tests are necessary, because in out-of-pile tests it is not possible to obtain the correct power and temperature distribution, while in-fission-reactor experiments allow only too small test-samples.

The test program in ITER for the solid breeder blanket foresees that initially there will be 3 concepts for each coolant.The test program will be implemented as follows:

1. During the Physics Phase, a horizontal port will be allocated to the solid breeder blanket. The purpose of the tests in this phase is to characterize the neutronic environment, to check out blanket systems, and to check out instrumentation.

FOR TEST MODULES OF SOLID BREEDER BLANKETS

USA-BIT-He	USA-BOT-He	JPN-BOT-H2O	USSR-H2O	USA-BOT-H2O
		5.8	~6	
1.2	1.2	1.2	1.2	1.2
0.15	0.15	0.15	0.15	0.15
HT-9	SiC-Composite	316SS	Austenitic Stainless Steel	
BIT	BOT	BOT	BOT or BIT	BOT
Li4SiO4	Li2ZrO3	Li2O	Li4SiO4	Li4SiO4
90	80	30	50	90
Rods (38 mm dia.)	Pebbles	Pebbles (<1 mm dia.)	Pebbles/Rods or Pellets	Pebbles (binary)
Beryllium Sphere Pack Thickness	Coolant Inlet Temperature	Thermal Insulator around Cooling Tube	Vacuum Gap, Be Block Thickness or Layer of SS tubes	Beryllium Pebble Layer Thickness
Breeder Rod Diameter	Pebble Layer Thickness	Cooling Tube Arrangement	Pebble Layer Thickness or Pellet Diameter	Cooling Tube Arrangement
500-700	450-950	450-600	500-700	350-1000
Beryllium	Beryllium	Beryllium	Beryllium	Beryllium
Rods + Anuular Sphere Pack	Pebbles	Pebbles (<1 mm dia.)	Pebbles or Blocks	Pebbles (binary)
1.3	1.3	1.5	1.4	1.3
He Purge	He Purge	He Purge	He Purge	He Purge
He	He	Water	Water	Water
5	10	15	6.8	15
200***/450	350***/650	280/320	260/285	280/320

2. During the Technology Phase, two ports will be allocated for solid breeder blankets: one for designs with gas cooling and one for designs with water cooling.
3. During the first four years of the Technology Phase, three submodules will be tested in each of the two ports available. In the port for gas-cooled designs, the EC, Japan, and the US shall have the lead on the design, construction and operation of one of the three modules respectively. The port for water-cooled designs is partitioned in exactly the same way. The lead for submodule design, construction and operation shall be taken by Japan, the US and the USSR.
4. During the following three years of the Technology Phase, a single module of the chosen reference solution with helium and water cooling shall be tested in each of the two ports. Alternatively, three single modules for the three different concepts can be tested successively for a period of one year each.

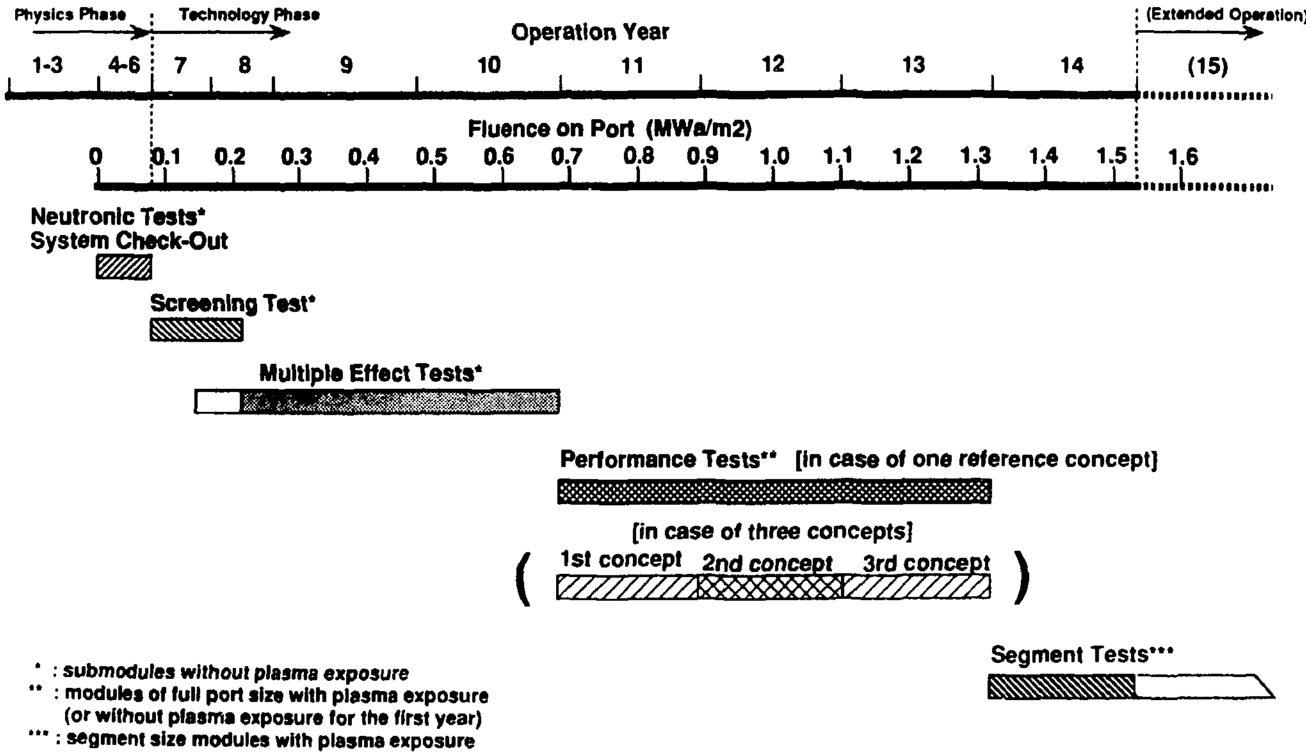

FIG. 4.2.3 - Testing Schedule for Helium- and Water-Cooled Solid Breeder Blankets

5. The tests of the submodules will be performed behind a first wall similar to the driver blanket first wall. In case of a single reference blanket, the blanket module will be tested for the first year behind a driver blanket-type first wall and with its own first wall facing the plasma for the two remaining years.
6. For various designs it may be necessary to perform tests with complete segments or even sectors during the final period of the Technology Phase (the last year) and during a possible extended phase operation.

Fig. 4.2.3 show schematically the testing schedule for the solid breeder blankets. Table 4.2.2 gives the data of the test modules.

PLASMA FACING COMPONENTS TESTING

A limited amount of work has been done to identify the test requirements for plasma facing components. The specification of the test program is complicated by the close coupling of the plasma facing components with the overall physics operation. Certainly, additional work is required to specify a comprehensive test program.There are several types of tests that can be performed in ITER, and they can generally be divided into plasma physics related tests and engineering related tests. The former will be monitored throughout the engineering testing activities.

The overall plasma performance will depend on the characteristics of the impurity control system (e.g., the type of divertor material exposed to the plasma), and initially these tests will be part of the initial ITER physics program. However, these types of tests will need to be performed whenever there is a change in the impurity control system configuration.

Engineering related tests are similar in scope to those conducted on blankets, and they include the following types:

1. Thermal- hydraulics performance. Assess the capability of efficiently removing heat from the impurity control system. These tests could be performed using different coolants and/or higher coolant temperatures than the base design.
2. Thermo-mechanical behaviour. Divertor plates subjected to high heat fluxes and a high number of cycles will be subjected to high thermal stresses.
3. Extended performance of divertors. Over a long period of time, the effects of surface erosion and neutron radiation will strongly influence the divertor lifetime. Extended tests are desirable to assess the capability for long lifetimes.
4. Transient response. The response of the PFC's to both normal and off-normal transients needs to be determined. In particular, surface erosion and electromagnetic forces during disruptions could severely reduce the operating life.
5. Alternate pumping and impurity removal systems - to be developed ex-ITER and installed if promising, e.g. by advanced approaches such as helium surface burial or He exhaust enrichment using palladium membranes to supplement standard vacuum pumping systems
6. Advanced divertor targets - to be developed ex-ITER and installed in ITER if promising and feasible - such as liquid metal droplet divertors, . Many of the engineering concerns with standard plate designs would be eliminated, but new concerns will arise, and the alternate concepts will need to be tested.The overall objectives for the tests are:

- Test the feasibility of the liquid metal target in an ITER-like magnetic field.
- Study the compatibility of the liquid metal free surface with the plasma for ITER operating conditions, including transients and disruptions.
- Study the thermal-hydraulic characteristics of the liquid metal targets under different divertor operating conditions.
- Study the extended performance of the liquid metal target including the effects of tritium extraction, radiation damage, materials corrosion, and safety.

MATERIALS TESTING

Materials testing in ITER is aimed to:

- validate results of material irradiation tests in fission reactors and simulation facilities.
- receive data on materials properties in fusion reactor environment as close to DEMO as possible, however the fluence goal is limited by the present ITER design.

Test goals include :

Surveillance testing - characterization of thermo-physical and mechanical properties so that the remaining life-time correction of heavily loaded elements of first wall, blanket, divertor welding and brazing joints in the in-vessel components can be determined.

Advanced material testing and fundamental investigations

- establishing of the correlations between properties changes and operation time, neutron fluence, temperature, environment, pulse characteristics etc.
- establish the correlations between radiation damage of fission and fusion spectra to enable database extrapolation from available fission reactor data.

Test media and parameters for different coolants are:

Coolant	Temperature range, °C
H_20	60-320
$He,C0_2$	200-650
Steam/water mixture	260-285

Investigations in liquid metal media will be carried out in the submodules of liquid eutectic water-cooled and self-cooled liquid metal module. Material testing is planned during eight years of operation in the Technology Phase with several replacements of test samples. Surveillance test of plasma facing materials in the physics phase will be performed in the submodule combined with neutronic module.

The total number of specimens for material testing is about 32000. A list of proposed materials and type of the testing is presented in Table 4.2.3.

SAFETY ASPECTS OF TEST PROGRAM

ITER will be the first significant fusion test environment with plasma, neutrons, and magnetic fields. The ITER testing value should be maximized, subject to cost penalties, risk to the public, and risk to the machine. Two basic safety issues relevant to the conceptual design should be considered in the testing program:

TABLE 4.2.3. MATERIALS PROPOSED FOR TESTING

	State & Material Employment		Damage type	Aim of testing
316 type austenitic steels	b,w,br	SM	B,S	st
Other austenitic steels	b,w,br	SM	B,S	f
Ferritic and ferritic-martensitic steels	b,w,br	SM	B,S	f
Beryllium	b,j	PFM,M	B,S	st,f
Copper and its alloys	b,br	HSM,SM	B	st,f
Molybdenum based alloys	b,w.br	HSM,SM	B,S	st,f
Nickel based alloys	b,w,br	SM	B	f
Niobium based alloys	b,w,br	SM,PFM	B,S	f
Titanium based alloys	b,w	SM,PFM	B,S	f
Tungsten and its alloys	b,br	PFM	B,S	st,f
Vanadium and its alloys	b,w	SM,PFM	B,S	f
Carbon based materials	b,br	PFM	B,S	st,f
Ceramics	b	BREEDER	B	st,f
Insulators	b		B	st,f
SiC	b	SM,PFM	B,S	f

b - base metal, w - welding joint, br - brazing joint, j - other type of joint,
SM - structural material, PFM - plasma facing material, SPM - special purpose material,
HSM - heat sink material, M - multiplier, B - bulk damage, S - surface damage,
st - surveillance testing (for ITER materials), f - fundamental investigations (for candidate materials).

1.What limitations do safety considerations place on the testing program? How can these limitations be overcome by modifying the design of test components or the basic machine?

2.What safety research value can (and should) be obtained from ITER? How can this be enhanced by design?

One area of concern for the testing program includes potential thermal interactions or incompatibilities. Pressurized-water test modules may pose steam explosion or vacuum chamber overpressure hazards; additional analyses are needed.

Also, there are chemical incompatibilities. Especially the liquid metal breeders have the potential for hydrogen production and energy release if they come in contact with water or air. Hydrogen production by a possible liquid metal-water-reaction leading (in connection with an air-inflow) to an explosive mixture limits the amount of liquid metal tolerable in the test module. At the beginning of the test program in ITER the confidence in the reliability of a single wall between liquid metal and the plasma chamber is probably not high enough to exclude a reaction between the liquid metal and the coolant of other components i.e. water-cooled divertor plates. If the premise is made that the hydrogen generated by this reaction should not be more than the one anticipated for carbon-water reaction, the volumes of lithium and lithium-lead would have to be limited to 0.1 m^3 and 0.6 m^3 respectively. Sodium-potassium (NaK) presents about the same hazard as lithium if the neutron fluence is low enough not to produce significant amount of Na-24 or Ar-41. These volume limits are too low for meaningful sized test modules.

Therefore, a dependable second barrier between test modules and plasma chamber is planned for the Physics Phase and the first two years at the Technology Phase of ITER operations. This raises the allowable liquid metal volumes approximately by a factor of 10. The test series with the modules behind this second barrier should provide the confidence in the blanket designs and the materials used to allow for an exposure of the modules to the plasma at a later stage.

Finally, as presently envisioned, ITER will provide substantial safety data, both planned and unplanned. Data will include failure rates, failure types, failure effects, behaviour of divertors, off-normal plasma behaviour, etc. As the design progresses, attention is needed to insure that full benefit be obtainable from ITER, taking advantage of its unique testing environment and role as global fusion showcase.

4.2.2. Ancillary equipment, configuration, and maintenance

TESTING SPACE

Testing will be performed primarily through horizontal access ports around the machine. The area of these ports is ~1.1m wide by 3.4 m high. Full segment tests are proposed in the last year of the Technology Phase.

The current design of ITER allows for 3 full ports for testing activities during the Physics Phase and 5 full ports during the Technology Phase. Tests have been allocated to ports according to the type of breeder and coolant. This provides the simplest arrangement for ancillary equipment and the most compatibility between submodule tests performed simultaneously within the ports. Ports numbered 8, 9, 10, 12, and 13, which are to be used for the nuclear test program, are shown schematically in Fig.4.2.4

ANCILLARY SYSTEM CONFIGURATION AND SPACE REQUIREMENTS

For each test location in the machine, a specific set of external equipment must be provided in the ITER plant, with supply lines to the test location. The main equipment required to supply and support the tests are:

- heat rejection
- tritium recovery systems and test-specific intermediate tritium processing
- chemical (impurity) control systems
- coolant and purge fluid storage, start-up, dump tanks, and volume control systems

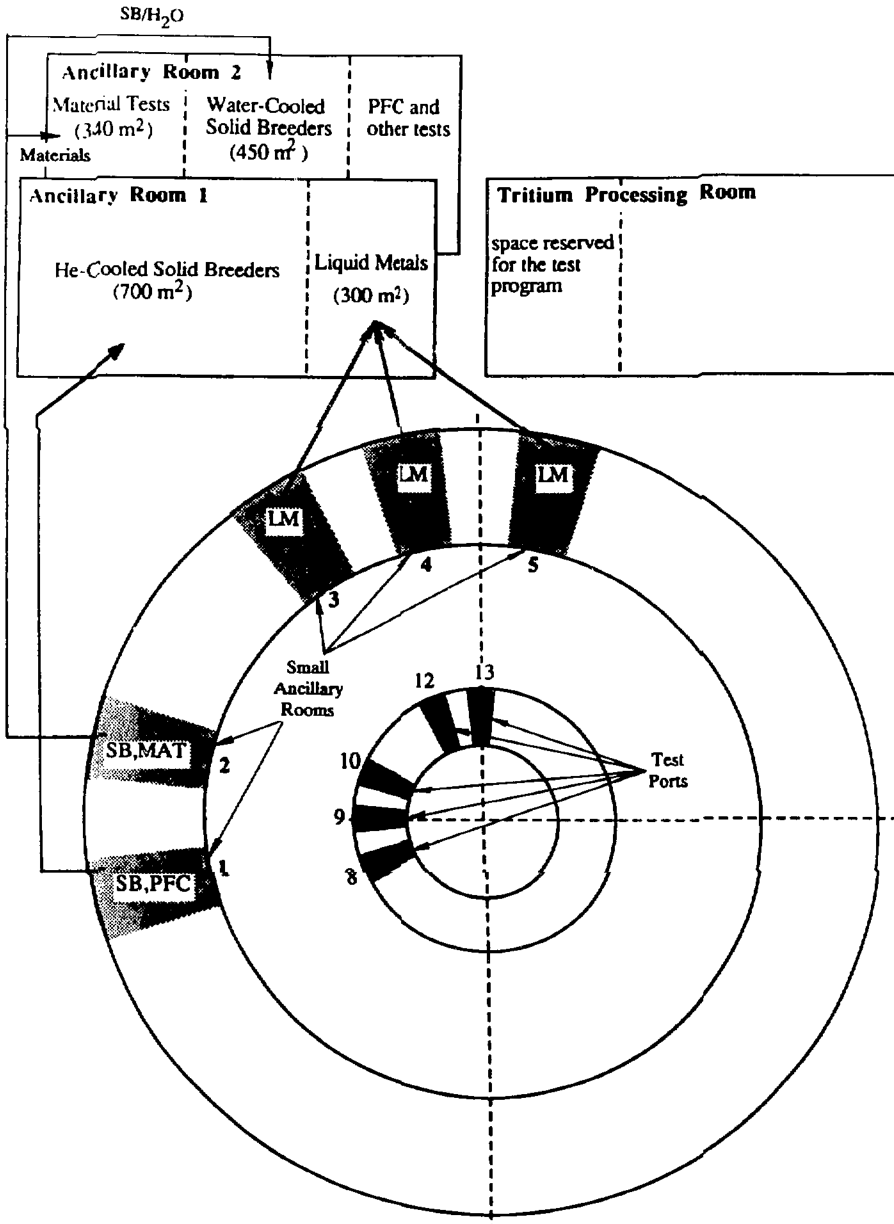

FIG. 4.2.4 - Space Allocation During the Technology Phase

- emergency and safety systems
- remote handling equipment
- test rooms and hot cells for examinations
- control and data acquisition systems

Ancillary equipment needs will change from the Physics Phase to the Technology Phase as additional ports and ancillary rooms become available.

Most of this equipment is specific to the individual tests, and cannot be integrated into the main system of the plant. Coolant and purge systems operate with fluids and conditions (temperature and chemistry) which are different from the basic machine components, and also require separate monitoring as an integral part of the testing.

Tritium extraction will take place in the ancillary equipment specific to the individual test modules and tritium will be released to the main plant tritium system (or stored in beds for future disposal) following extraction. It is envisioned that services from the basic plant will provide the test programs with at least the following tritium processing capabilities:

- He stream carrying tritium and some level of impurities (to be specified)
- water stream carrying tritium and some level of impurities
- room air detritiation

Similarly, heat removal systems are specific to the individual test modules. Generally, low-temperature water will be passed from the final heat exchanger stage of the test module ancillary equipment to the plant heat rejection systems.

Test ports will be occupied by submodules, full-port modules, and possibly full segment tests. The ancillary equipment needs depend on the test type.In some cases, equipment can be designed to handle the higher power and tritium levels and then shared among the various types of tests.

Preliminary designs of the required ancillary equipment have been performed for some of the proposed tests. Estimates of the total volumes required for ancillary equipment for all the test modules are shown in Table 4.2.4.

In general, as the size of a test object increases, the space required for ancillary equipment increases only moderately. The greatest space requirement comes from submodule tests, because the majority of the ancillary equipment can not be shared. The cooling and tritium processing systems for each submodule may be different in design, and must be allowed to operate independently.
Design of the ancillary equipment rooms must include proper tritium containment and protective measures. In addition, some parts of the ancillary systems will require accessibility for maintenance and replacement.

The distance between the test modules and ancillary systems is also very important. This concern arises due to the following concerns:

- Safety requirements constrain the total amount of potentially hazardous materials present, particularly a concern for liquid metals and tritium entrained in process lines.
- Time constants for system equilibrium depend on the volume of fluids and distance to the ancillary equipment. Time-related test requirements will increase if the distance to the ancillary equipment increases.

TABLE 4.2.4. ESTIMATE OF ANCILLARY EQUIPMENT SPACE REQUIREMENTS

Port Test Article Type	Space Requirements (Area x Height, m^2xm) behind test port	ancillary rooms	plant services*
SB/gas			
3 submodules		730 x 11	300 x 11
full module or segment		370 x 11	130 x 11
SB/H2O			
3 submodules		450 x 11	150 x 8
full module or segment		150 x 11	50 x 5
LM/self			
4 submodules	300 x 11		
full module or segment	300 x 11		
LM/H2O			
2 submodule	50 x 11	100 x 11	
full module	100 x 11	100 x 11	
segment	100 x 11	200 x 11	
Materials			
Test assembly	120 x 5**	220 x 11	525 x 11
TOTAL FLOOR AREA	400-500 m^2	1600 m^2	975 m^2

* plant services include space allocated in the main tritium processing hall and post-irradiation examination rooms (hot cells)

** pneumatic system for test specimen insertion/extraction, may be located in ancillary room

- The test ports contain highly-instrumented, complex systems. In some cases, there are several submodules within a single port, and the total number of process lines and instrumentation cables is expected to number in the hundreds. Running these lines over long distances will make the systems complicated and may lead to problems with reliability and maintenance. Some piping will require special measures (such as guard heaters for liquid metal systems) to control the process fluid conditions.
- In general, the behaviour of and interactions with ancillary systems are an integral part of tests, especially for integrated tests. The design of the ancillary systems must be prototypical, including the process lines, in order to obtain valid information from the tests.

For these reasons, it is strongly recommended that space be provided for the installation of ancillary equipment as close as possible to the torus.

In addition, the vertical location of external cooling systems relative to the test modules is an important concern. In general, the ancillary equipment should be at

TABLE 4.2.5. AVAILABLE SPACE FOR TEST PROGRAM ANCILLARY EQUIPMENT

Type	Number	Size	Location
Test cells test modules, mid-plane	5	$65m^2 x 11m$	15 m behind the floor 3.65 m below
Ancillary rooms building	2	$995m^2 x 11m$	adjacent to reactor
Plant tritium processing hall (shared) building	1	$995m^2 x 11m$	adjacent to reactor
Post-test examination rooms and hot cells			

least as high as the test modules to provide natural circulation for off-normal operation and emergency conditions.

Figure 4.2.4 shows a plan view of the mid-plane of ITER. Space for ancillary equipment is provided in several rooms, as shown in Table 4.2.5.

As mentioned above, proximity to the test ports is important for the ancillary equipment associated with most of the test types. However, since adequate space is not available directly behind the test ports, priority has been given to the liquid metal tests, for which safety and time constants are more important concerns.

Based on the space requirements and the available space, tentative allocation of space has been defined for both the Physics Phase and Technology Phase. Figure 4.2.4 shows the location of the 5 test cells and 2 ancillary rooms. Test cells 1 and 2 are reserved for diagnostics during the Physics Phase, and are released to the test program during the Technology Phase. Test cells 3-5 are permanently reserved for liquid metals. This allows the installation of at least the primary liquid breeder loops as close as possible to the test ports. An additional 300 m^2 is reserved in Ancillary Room 1 to accommodate the remaining needs for liquid metal blanket testing, which would include primarily water cooling and secondary NaK loops.

Solid breeder blanket ancillary equipment is housed primarily in the two ancillary equipment rooms. The gas-cooled solid breeder blankets have the largest space needs, and are allocated 700 m^2 in the ancillary room. An additional 320 m^2 is required for tritium recovery equipment, and it is proposed to occupy this amount of space in the main plant tritium processing hall. Water-cooled solid breeder blankets have smaller space needs, but could also benefit from space allocated in the plant tritium processing hall, provided the room is separated from the main hall. A common holding tank is desirable to isolate and combine the exhaust streams from all of the test module extraction systems before combining with the plant tritium system. This system would also be best placed in the tritium processing hall.

Support equipment for the materials test module includes cooling systems, a pneumatic system for specimen withdrawal and replacement, post-examination, and control systems. Additional space for post-irradiation examination (PIE) is required for all test objects, including material specimens, submodules, modules, and segments. Analysis of full 10-m segments could be performed in the area reserved for base blanket PIE.

MAINTENANCE AND HANDLING REQUIREMENTS

A preliminary estimate has been made of the number of insertion/extraction cycles likely to be expected for each type of test article, for examination and replacement. This is summarized in Table 4.2.6. The machine shutdown times required for these operations are likely to be significantly different depending on whether or not the primary vacuum would have to be broken.

From the number of movements estimated, it is clear that the remote handling procedures and systems must have the following capabilities:

- replacement of material specimens within a few hours without deenergizing the TF coils
- replacement of submodules within 2-3 days
- replacement of modules within a week.

The total weight of the test modules (at least 10 tonnes) must be taken into account when considering remote handling. It is important that the gripping operation necessary to remove the module be performed as close as possible to the Center of Gravity (CG) of the module in order to minimize the out-of-balance forces on the handling equipment.

The biological shield plug is a large and heavy item. It would therefore be more convenient to remove it prior to breaking the vacuum seal, i.e. before opening up the Contained Transfer Unit (CTU) housing the remote handling equipment for the test module. It would not be necessary then to house the plug within the CTU.

TABLE 4.2.6. HANDLING REQUIREMENTS FOR THE TEST OBJECTS

Test Articles	Typical Size (Weight)	Maximum # of Articles		Maximum frequency load/unload	Maximum # Loads/Unloads per year	
		with plasma exposure	without plasma exposure		with plasma exposure	without plasma exposure
Material Specimens	1-100 cc (<1 kg)		1000's	1-3/yr		1000's
Submodules	0.05-0.2 m^3 (1 T)	6-7	12	2/yr	12-14	24
Modules	0.5-2 m^3 (<10 T)	4	4	1/yr	4	4
Segments	10 m^3 (30 T)	4		0.1-1/yr	4	

The blanket module testing programme requires the use of several equatorial ports for various testing campaigns. These campaigns involve several different approaches for remote handling which are described in more detail in [4.2].

4.2.3. International aspects of the test program

The plasma-facing area required to implement the full test program of the four parties has been estimated and it was found to exceed the total available space. Subsequently, the test plans were reduced and fuller utilization was made of international combined testing in order to optimise the use of the available port space. Beside relieving limitations on testing space, international collaboration in the test program is the most cost-effective and efficient means to satisfy the testing objectives of the 4 parties.

Collaboration in the test program is fundamentally different from collaboration on the basic device. ITER itself is a single, clearly-defined machine with generic capabilities to perform testing of components and to demonstrate the physics and engineering potential of fusion. On the other hand, the test program is tightly coupled to, and in most cases plays a key role in the entire R&D plans for nuclear components.

Increased international collaboration in ITER naturally leads to a greater amount of common R&D to develop and test components. This has far-reaching implications on the design and operation of ITER and on R&D programs in fusion nuclear technology throughout the world.

International collaboration in the ITER test program can take several forms. While it is necessary to jointly plan the testing use of the machine, there are many options for implementing the test program. In the definition of the test program, the key features related to international collaboration include:

1. The amount of common testing. Test programs can be fully independent, fully in common, or some combination of joint and independent testing. Different parties can be given a lead role for testing components, with the remaining contributing at different levels.
2. The degree of design specificity. All the test space can be pre-designed for particular applications, left as generic slots, or some combination of pre-design and flexibility.
3. Allocation of available testing spaces to countries. The available testing space can be pre-allocated to separate parties, left open as a "user" facility, or principles can be established for the allocation of space based on some form of selection criteria.

In the conceptual design phase, some of these options have been clearly defined, and others have been left yet unanswered. For the Engineering Design Phase, it is important to clearly define aspects of international collaboration in the test program so that the design of the machine allows the maximum benefits from testing and so that R&D programs can proceeded in a timely way toward the development of test modules for introduction into ITER.

The single integrated test program was defined for ITER which features the sharing of test ports among parties. Test ports are allocated according to the type of tests to be performed, and all interested parties have worked together to define the test

TABLE 4.2.7. SHARING OF PORTS FOR NUCLEAR TECHNOLOGY TESTING

PORT	PARTICIPANTS			
	EC	Japan	US	USSR
Gas-cooled solid breeders	√	√	√	
Water-cooled solid breeders		√	√	(√)
Liquid-metal-cooled liquid metals	√		√	√
Water-cooled liquid metals	√			√
Materials	√	√	√	√

schedule within the ports where testing of their preferred design options will take place. During the Technology Testing Phase, 5 ports are allocated for nuclear testing, and have been assigned as shown in Table 4.2.7. (3 ports are reserved during the Physics Phase, allocated to liquid metals, solid breeders, and neutronic tests). A key element of this test plan is the intent to test several submodules simultaneously during the early years of the Technology Phase, followed by a selection process and narrowing of concepts for full-port and segment testing during the final years of the Technology Phase.

Sharing of space and joint planning of the test programs is an important aspect of collaboration, but is also desirable to enhance the amount of **joint** testing, in which a single test object is developed and tested by more than one party. Common interests in the world programs allows for a range of bilateral and multilateral collaborations on different design concepts. During the submodule testing (scoping) phase, it was decided that the ports will be divided and different parties will take the lead role in developing and testing submodules. Participations by the other parties is encouraged, but the nature of this cooperation has not been fully defined. More effort will be required in the future to move towards combining R&D programs outside of ITER and planning "true" joint testing in ITER.

Materials testing is probably the most truly-integrated test program within ITER, since mutual interests exists for a variety of materials. Testing needs are well-defined and, to some extent, independent of the component design.

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