Chapter XIV

MACHINE OPERATION AND TEST PROGRAMME

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1. ROLE OF INTOR TESTS

INTOR [1] is considered to be the major device in the fusion reactor development programme between the next generation of large tokamaks (TFTR, JET, JT-60, T-15) and a demonstration tokamak reactor (DEMO). Thus, whilst the design of INTOR must build on the knowledge gained in the existing research and development programmes in plasma physics and reactor technology, it is equally true that the construction of INTOR and the results of its test programme must provide a major part of the foundation on which DEMO is to be designed. It is therefore necessary to relate the design of INTOR and the specifications of its test programme to the needs of DEMO and to other complementary development programmes which may run in parallel.

DEMO is taken to be a tokamak reactor operating with plasma and blanket parameters extrapolated to a commercial fusion reactor and which will allow a reasonable estimate to be made of the commercial prospects of fusion power. It follows that DEMO will generate several hundred megawatts of electricity and produce net electrical power; it will also have a tritium-breeding blanket with a net breeding ratio greater than unity. DEMO must also demonstrate component and system reliability, availability and lifetime, as well as safe and environmentally acceptable operation under conditions approaching those required of a commercial reactor.

Table XIV-1 compares the anticipated range of operating conditions for INTOR, DEMO and commercial reactors. The neutron and surface wall loadings possible with INTOR are less than those anticipated for DEMO by approximately a factor of two, and this has been a driving factor in the design of INTOR, since it is thought that a lower wall loading in INTOR would strongly reduce the value of blanket engineering tests. On the other hand, there are strong incentives for seeking steady-state operation of a commercial reactor.

TABLE XIV-1. ANTICIPATED REACTOR OPERATING CONDITIONS

	INTOR	DEMO	COMMERCIAL
Thermal power (MW)	620	$\approx$ 1000	4000
Gross electrical power (MW)	10	$\approx$ 100	1000
Neutron wall loading (MW/m^2)	1.3	2–3	3–6
Surface wall loading (MW/m^2)	0.1	0.2–0.8[a]	0.3–1.5[a]
Burn time (s)	$\gtrsim$ 200	$\gtrsim$ 1000	$\gtrsim$ 1000
Number of cycles	7×10^5	$\approx 10^5$	$\approx 10^5$
Integrated wall load ($MW \cdot a/m^2$)	6.6	10–20	15–30

[a] Depending on whether or not a divertor is necessary.

The tasks which must be accomplished by INTOR and the complementary development programmes are as follows:

(a) Demonstration of a plasma physics performance which can be extrapolated to DEMO conditions, and in particular the containment of a controlled D-T plasma for long pulse-lengths at optimum plasma parameters
(b) Testing and development of reactor materials and components, and demonstration of their operation at high availability and reliability under conditions approaching those required for DEMO
(c) Demonstration of the integration of all necessary components into an overall reactor system which can be safely and remotely maintained
(d) Investigation of electricity generation and tritium breeding in INTOR in a local structure which is prototypical of DEMO

These requirements lead to a schedule for staged operation, described in Section 2, in which the successive stages are concerned with: (1) the achievement of the necessary plasma conditions and the operation of the basic INTOR device, (2) the testing and development of candidate first-wall and blanket materials and structures in test channels and modules, and (3) a more extensive testing of certain materials and components to neutron fluences approaching half of that expected in DEMO, together with a limited demonstration of electricity production. The major features of this test programme are described in Section 3, and further details are given in Sections 4 to 13. Since the implementation of such a test programme must also influence the design of INTOR and its associated facilities, the remaining sections describe the necessary support facilities and the operational requirements of the test programme.

Whilst INTOR will be a major element in the programme leading to a DEMO reactor, there will certainly be complementary programmes in both plasma physics and technology. These are discussed in the next two subsections.

1.1. Plasma physics experiments in INTOR and complementary programme

An extensive experimental and theoretical plasma physics programme should support the design and construction of INTOR and will supplement INTOR in providing the physics base necessary for DEMO. The present and next generation of tokamaks (JET, T-15, JT-60, TFTR) will provide preliminary information on the behaviour of reacting plasmas, plasma shaping, profile control, and alternative methods for auxiliary heating. A primary physics objective of INTOR is to investigate the operation of an ignited D-T plasma and to achieve long, controlled and reproducible burn with optimized plasma parameters. INTOR will also be used for performing certain plasma physics studies. These generally should be limited to experiments related to learning how to run INTOR as well as such investigations that require the unique capabilities of INTOR. Other physics experiments should be made in other plasma physics devices.

1.2. Technology testing in INTOR and complementary programme

A technology development programme for DEMO requires testing in both INTOR and simulation test facilities. Extensive screening of candidate materials and component design concepts will be carried out in test facilities which partially simulate the fusion environment prior to operation of INTOR. INTOR will then serve principally to provide component design verification tests with all synergistic effects in a true fusion environment. Testing in INTOR must be supplemented with tests in other facilities to provide the basis for design and construction of DEMO. Such a complementary test programme is necessary because specific types of tests can be simulated more readily in test facilities than in INTOR; in some cases they can only be done in test facilities.

The relation of INTOR to supporting and complementary technology and component test facilities is examined on a technology-by-technology basis in the following subsections.

1.2.1. Materials

The basic properties of structural, breeding, insulating and other materials will be determined in test facilities. The behaviour of materials under irradiation will be investigated with accelerator-type neutron sources and in fission reactors, both of which will be able to achieve component end-of-life fluence levels, but not in a fusion radiation environment. Accelerator neutron sources suffer from limitations on the number and size of samples, while fission reactor neutron spectra lack the important high-energy component associated with 14 MeV fusion neutrons. These test facilities will continue to be used to screen candidate materials for DEMO and commercial reactors. They will also be used to irradiate the few primary candidate materials to high fluences approaching those projected for DEMO. INTOR will then complement these test facilities by performing component element tests in a true fusion reactor environment and by irradiating a large number of samples of the primary candidate materials to fluence levels (>5 MW·a/m^2) approaching 1/3 to 1/2 of the end-of-life fluence for components in DEMO. These tests in INTOR will thus provide a benchmark for some interpreting and validating of the non-fusion irradiation data, and will contribute directly to the data base required for DEMO and subsequent commercial reactors.

1.2.2. First wall/blanket

The thermomechanical and electromagnetic performance of first-wall/blanket elements can be tested in "separate effect" test stands to examine the response to normal surface and bulk heating loads, and to simulate off-normal conditions such as plasma disruption thermal and electromagnetic loads. A "multiple effect" facility can be constructed to investigate the thermomechanical and electromagnetic response to several simultaneous effects in a non-radiation environment. Tritium production capabilities of candidate tritium-breeding materials can be examined in simplified geometries in a 14 MeV neutron source facility. Tritium recovery can be investigated in special test stands and in fission reactors. One of the main objectives of INTOR is to provide simultaneous testing of thermomechanical and electromagnetic response, and tritium production and recovery for blanket assemblies in an actual fusion environment. Design verification and long-term reliability demonstration tests of first-wall/blanket elements and modules will provide an engineering data base for the design and construction of DEMO.

1.2.3. Plasma heating

An integrated neutral beam (or RF) heating system and its reliability under sustained operation in a non-radiation environment can be demonstrated in test facilities. The integration of the heating system into a tokamak reactor system is an objective of INTOR. Information on neutron and plasma radiation effects upon components can be obtained from irradiation experiments in fission reactors and accelerators. Testing of advanced components can be carried out in INTOR. Also, the investigation of the performance of integrated heating systems under sustained operation with high availability in a fusion reactor radiation environment is an objective of INTOR.

1.2.4. Magnetics

Proof-of-principles testing of superconducting toroidal and poloidal field coils can be carried out in test facilities such as the LCP test facility and the planned 100 MJ pulsed coil test facility in the USA, and in the SC magnet test stand (SIMS) in the USSR. Experience with superconducting coils on tokamaks will be obtained from T-7, T-15 and Torus II. Beyond this level of testing, it is necessary to demonstrate: (1) an integrated magnetic system that functions in a tokamak fusion reactor environment, and (2) the reliability of that system under sustained operation. The investigation of these two items is an objective of INTOR. Since radiation effects upon materials properties can be studied separately, preliminary investigations could, in principle, be performed on a modified LCP test facility incorporating superconducting toroidal and poloidal coils and normal poloidal coils, a vacuum vessel/blanket/shield/structural system, and a means of simulating the electromagnetic and mechanical effects of plasma disruptions. However, such a modified LCP test facility would be complicated and costly. It is probably more feasible to study separate effects (e.g. fatigue and crack growth limits on large structural elements) in separate, single-purpose test facilities and then to rely upon INTOR for an integrated test of the complete magnetics system.

1.2.5. Tritium processing and containment

Most of the key issues involved in constructing and operating an integrated tritium processing system (excluding tritium recovery from the blanket) can be examined in facilities such as the Tritium Systems Test Assembly (TSTA). Such a facility can also deal with important aspects of tritium containment. INTOR will investigate the integration of the tritium processing system into a tokamak reactor system, including plasma exhaust, refuelling and heating, and blanket tritium recovery.

1.2.6. Radiation shielding

The methods and basic nuclear data for radiation shielding can be validated by performing integral shield experiments with 14 MeV neutron sources. INTOR will verify the prediction capability for the design of an integrated radiation shield system and will provide engineering safety factors for the design of DEMO and subsequent commercial reactors.

1.2.7. Divertor collector plates, limiters

The thermomechanical and electromagnetic performance of integrated systems can be demonstrated under sustained operation in a non-radiation environment in test facilities. Neutron radiation damage effects on these components can be studied in fission reactors, but not with the important 14 MeV neutron component in the spectrum and not under the appropriate thermal and mechanical loadings. Erosion due to plasma interactions can be studied in plasma physics experiments and in accelerator test facilities. Testing of several design concepts can be carried out in INTOR. The integration of these systems into a tokamak reactor system is an objective of INTOR, as is the investigation of the performance of these integrated components under sustained operation in a fusion reactor environment.

1.2.8. Remote assembly/disassembly and maintenance

The tools and techniques for remote assembly/disassembly and maintenance can be developed and tested in test facilities, including a mock-up facility for INTOR. The investigation of remote maintainability of a tokamak in a radioactive environment is an objective of INTOR.

1.2.9. Diagnostics, data acquisition and control

Diagnostics, data acquisition and control components can be tested in plasma experiments, in fission reactors and in other test facilities. The investigation of an integrated diagnostics, data acquisition and control system for a tokamak reactor and of the integration of such a system into a tokamak reactor are objectives of INTOR. Testing of advanced components can be done in INTOR. The investigation of the performance of such a system under sustained operation in a reactor environment is also an objective of INTOR.

2. STAGED OPERATION SCHEDULE

It is convenient to define three different stages of operation for INTOR, with emphasis on different aspects of the utilization of the device in each stage. During the first stage, the emphasis will be upon learning how to operate the device so as to obtain optimum performance. This will entail about one year of hydrogen plasma operation and engineering check-out, followed by about two years of D-T plasma operation. Calibration testing and some preliminary engineering testing may be performed.

The second stage will be devoted to engineering testing, with emphasis on a flexible test programme and with the objective of 25% availability. The duration of this stage will depend upon the exact test programme that will be developed ultimately for INTOR. At the present time, four years are anticipated for this stage of operation. This period would allow for initial testing and modification of a primary and back-up blanket concept by each of the four participating partners.

The third stage of operation differs from the second stage in that the emphasis is upon maximization of availability and fluence accumulation for performance testing of components. The objective of this stage is to obtain about 5 $MW \cdot a/m^2$ of 14 MeV neutron fluence in not more than ten years after the end of the second stage.

A representative schedule for staged operation is given in Table XIV-2. The different INTOR testing objectives to be achieved during the various stages of operation are indicated in Table XIV-3.

3. MAJOR FEATURES OF THE TEST PROGRAMME

A preliminary test plan has been developed to provide insight into the design and operational requirements that must be imposed on INTOR. This initial plan has been developed using judgements as to where INTOR fits into an international fusion development plan, as discussed in Section 1, and is considered an essential part of the development of a demonstration reactor.

The test plan is based on three stages of reactor operation, and the testing sequence is arranged to permit timely collection of data, as shown in Fig. XIV-1. During Stage I, emphasis is placed on plasma physics tests whose objective is the provision of a basis for the physics required for the DEMO reactor. In addition, these tests will provide the necessary experience and understanding of the INTOR plasma so as to permit long-pulse operation during Stages II and III when other tests requiring large fluences will be conducted. Most tests requiring frequent change-out will be performed during Stage II. This will provide a maximum amount of data early in the programme and will permit the use of a high-duty cycle during Stage III.

All material and module tests in INTOR can be fulfilled in about 12 m^2 of test area. A standard size of 1 m × 1 m has been defined for test channels and test modules. Two sectors of the machine incorporate six test pockets of 1 m^2 each. Some of the pockets (two or three) have to be opened to the plasma for surface materials tests. Other test units will be located behind the first wall and will not interact with the plasma. Test modules and test channels have equal dimensions, which facilitates interchange of test locations. Horizontal module and channel installation was selected as an approach that could provide the largest test area and minimum interference with other reactor components. A third sector is provided for plasma engineering and electricity generation tests.

Text continued on page 590

TABLE XIV-2. INTOR STAGED OPERATION SCHEDULE

Stage	Years	Emphasis	Availability	Annual tritium consumption[a] (kg)	Annual total (0–14 MeV) neutron fluence[a,b] ($n/m^2 \cdot a$)	Annual 14 MeV neutron fluence[a,c] ($MW \cdot a/m^2$)	Annual number of shots[d]	Total number of shots
IA	1	Hydrogen plasma operation Engineering check-out	10%	–	–		2.2×10^4	2.2×10^4
IB	2	D-T plasma operation	15%	3.6	1.8×10^{25}	0.16	3.3×10^4	6.6×10^4
II	4	Engineering testing	25%	6.9	3.2×10^{25}	0.31	3.1×10^4	1.2×10^5
III	8	Upgraded engineering testing[d]	50%	13.8	6.4×10^{25}	0.26	6.2×10^4	5.0×10^5

[a] Based on 200 s shots and 80% duty cycle (100 s shots and 70% duty cycle in Stage I), 620 MW(th) flat-top power, 1.3 MW/m^2 average neutron wall loading and the indicated availability.

[b] At the outboard location of the test modules. These values are representative; the actual values depend upon the design details.

[c] Based on a geometric peaking factor of 1.2 at the outboard location of the test modules.

[d] The requirement on Stage III is to accumulate $\sim 5\ MW \cdot a/m^2$ within $\leqslant 10$ years after the end of Stage II. This could be achieved in several ways; the case given here is only a representative one.

TABLE XIV-3. INTOR TESTING OBJECTIVES

	Stage I	Stage II	Stage III
PHYSICS			
Ignition physics investigation	x		
Achievement of long, controlled, reproducible burns with optimized parameters	x		
High (> 70%) duty cycle	x		
Plasma physics experiments	x	x	
MATERIALS			
Materials bulk property investigation		x	x
Radiation damage investigation			x
Surface effects investigation	x	x	x
MAGNETICS			
Integrated magnetic system investigation	x		
Integration of magnetic system into tokamak reactor system	x		
Performance of magnetic system under sustained operation		x	x
PLASMA HEATING			
Integration of NBI system into a tokamak reactor system	x		
Testing of NBI components in fusion radiation environment	x	x	x
Testing of RF components in fusion radiation environment		x	x
Performance of NBI system under sustained operation		x	x
PLASMA FUELLING			
Integration of pellet injector into tokamak reactor system	x		
Testing of pellet injector components in fusion radiation environment	x	x	x
Performance of pellet injector system under sustained operation		x	x

TABLE XIV-3 (cont.)

	Stage I	Stage II	Stage III
TRITIUM			
Integration of tritium-breeding system into tokamak reactor	x		
Investigation of tritium containment in tokamak reactor	x	x	
VACUUM			
Integration of vacuum pumping system into tokamak reactor system	x		
Performance of vacuum system under sustained operation in tokamak reactor		x	x
Component testing		x	x
BLANKET			
Integration of blanket system into tokamak reactor system	x	x	
Basic engineering data		x	
Tritium production and extraction		x	x
Electricity production		x	
DEMO prototypical blanket segment testing			x
Performance of blanket system under sustained operation in fusion reactor			x
Design concept testing		x	x
FIRST WALL, LIMITER, DIVERTOR			
Integration into tokamak reactor system	x		
Basic engineering data	x	x	
Performance under sustained operation in fusion reactor		x	x
Design concept testing		x	x
SHIELD			
Integration of shielding into tokamak reactor system	x		

TABLE XIV-3 (cont.)

	Stage I	Stage II	Stage III
REMOTE MAINTENANCE			
Investigation of remote maintainability of tokamak reactor system	x	x	x
DIAGNOSTICS, DATA ACQUISITION, CONTROL			
Integration into tokamak reactor system	x		
Basic engineering data	x	x	
Investigation of integrated system under sustained operation	x	x	x
Component testing		x	

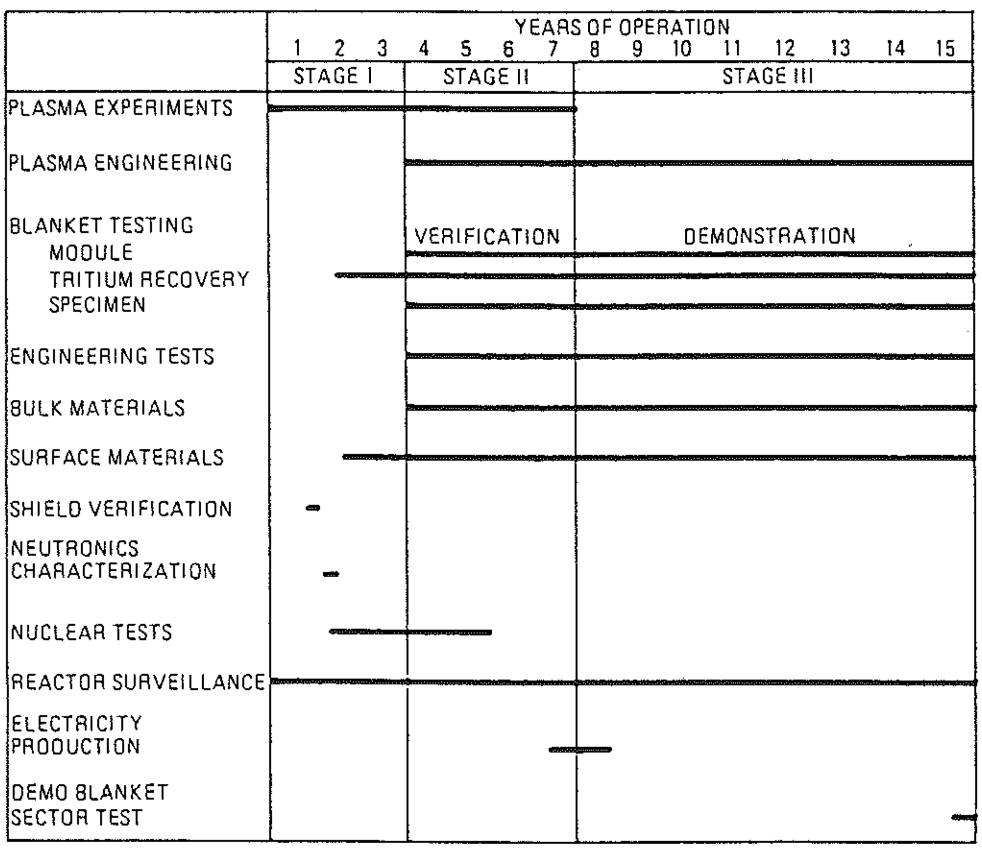

FIG.XIV-1. INTOR test schedule.

Plasma physics tests will emphasize those studies that cannot be performed in other experiments and for which INTOR represents a unique test bed, e.g. the study of specific aspects of long burn pulses. In addition, a variety of tests will be performed as part of the process of achieving optimized plasma performance of INTOR, e.g. profile control and burn temperature control.

Plasma engineering tests include the testing of plasma heating and confinement hardware that interacts directly with the plasma. Testing of radiofrequency launchers in a radiation environment is one example.

Blanket engineering tests will be used to confirm the results predicted from ex-machine tests. The tests will include 1 m^2 modules, tritium recovery capsules and critical life-limiting elements of the blanket. Tritium recovery tests will be used to confirm ex-machine tritium breeding and extraction tests. The critical element tests are used to provide a closer simulation of DEMO reactor conditions and to permit the use of accelerated testing. At least four long-term blanket demonstration module tests are planned.

Demonstration tests are expected to be used to correlate the testing results and analytic predictions for combined materials and synergistic effects, and to provide information on performance changes with irradiation. The test modules will be left in the reactor until the end of Stage III. Testing to within a factor of 2 or 3 of the design lifetime for DEMO is considered to be a requirement of INTOR. If failures should occur, failure analysis could provide information on failure modes and could direct the tests for DEMO design improvement efforts.

Bulk materials property tests will provide information on: (1) primary and back-up structural materials, (2) high-heat-flux materials, (3) insulators, (4) breeders, and (5) multipliers. Two 1 m^2 test modules provide adequate space for the materials specimens. Standard capsules of 5 cm dia. × 15 cm length are used to contain varying numbers of specimens. Each capsule can contain as many as 1440 swelling and phase stability specimens. Only one in-situ cyclic fatigue specimen could be fit in a single capsule. The design permits as many as 153 capsules in the reactor at one time. At each change-out interval, 60 capsules are removed and replaced. The temperature of each capsule can be controlled and kept at a specified level between 50°C and 700°C. Individual capsules can be removed without having to disconnect the services.

The test duration was established by considering the fluence requirements of the DEMO first wall and blanket. Extrapolation by a factor of two or less is considered acceptable. To provide data over a range, it was decided to remove samples at 0.26, 0.66, 2.0, 3.3 and 6.6 $MW \cdot a/m^2$. The tests have to be started at the beginning of Stage-II INTOR operation and the last specimen must be removed at the end of Stage-III INTOR operation.

Surface tests will provide data on the surface effects of the divertor plate, armour, limiter and first-wall materials. The plan requires the use of ~5000 material specimens. Most specimens are 1 cm × 1 cm, but some larger samples will be required. The 1 cm × 1 cm samples can be tested in a 1 m^2 test area, but the larger samples will have to be included in other areas, possibly as the first wall of other test modules. Specimen locations, cleaning method, temperature, material and fluence levels are varied in the test programme.

Nuclear testing is planned for Stage-I operation on a non-interference basis with the physics testing. These tests will provide data to determine the environment needed for subsequent tests. Nuclear tests are planned to investigate the basic machine neutronic performance such as source intensities and spatial distributions, to determine the radiation environment at test sites and around sensitive components, to determine radiation streaming characteristics of the reactor shield, and to provide a basis for code and data validation and improvement. Some of these tests must be done during early D-T shots, before background radiation levels build up, and will require reactor operation at low power levels to permit the use of accurate (~5%) direct measuring techniques. For some of the special nuclear tests, two months are required before the start of Stage II.

Electricity production. Electric power generation at the end of Stage II or at the beginning of Stage III can be accomplished using the outboard regions of two tritium-breeding segments which have been installed during the initial construction of INTOR. Simultaneous tritium and electrical power generation in a prototypical DEMO blanket sector could be performed during the second half of Stage III.

4. PLASMA OPERATION IN STAGE I AND PLASMA EXPERIMENTS

4.1. Objectives

During the initial stage of INTOR operation, the major objectives of the device will be the demonstration of plasma physics required for DEMO, as well as the demonstration of a number of the intrinsic reactor-relevant technologies in a fusion environment. Besides these specific programmatic objectives, however, the activities in Stage I are intended to prepare for the Stage-II engineering tests by establishing reliable operation of the device (25–50% availability) at the design parameters ($\langle\beta\rangle \sim 5\%$, $\geqslant 100$ s burn pulse at 70% duty factor, and a neutron wall loading of 1.3 MW/m^2). It is crucial, therefore, that the Stage-I schedule provide for the optimization of the tokamak, heating, fuelling and diagnostic systems as well as for remote maintenance capability.

TABLE XIV-4. INTOR STAGE-I OPERATION PLAN

Year	1		2			3
Operational mode	Vacuum vessel conditions	Ohmic heating start-up	Neutral beam start-up			Long-pulse ignition experiments
			$H^0 \rightarrow H^+$	$D^0 \rightarrow H^+$	$D^0 \rightarrow (H^+, D^+, T^+)$	
Operational objectives	GDC[a] PDC[b]	Feedback control: plasma current plasma position plasma density Plasma start-up Divertor operation Disruption behaviour Ohmic heating power balance, energy deposition on divertor plates Maximization of plasma current and pulse length Diagnostic check-out	Neutral beam operation up to maximum power and pulse length Pellet injection Tritium handling NB power balance and energy deposition on divertor plates D-T diagnostic check-out Neutron diagnostic calibration Determination of erosion/re-deposition for first wall and divertor plates and disruption behaviour			Ignition Increase of pulse length to maximum Check of power balance and energy deposition on first wall and divertor plates Determination of erosion/re-deposition for first wall and divertor plates Attainment of 70% duty factor Surface effect material tests Plasma physics experiments

Maximum parameters		$I_p \geqslant 6$ MA	$P_{NB} = 75$ MW (6 s)	$N_{D/T} = 1.3 \times 10^{20}\ m^2$ Pulse length $\geqslant 100$ s $T_i = 10$ keV Duty factor = 70% $\langle\beta\rangle = 5\%$ Availability = 25% $P_{D/T} = 620$ MW ($P_N = 1.3$ MW/m^2)
Reactor-relevant technologies		Vacuum system Plasma composition control (divertor) Plasma fuelling (puffing)	NB heating Plasma fuelling (pellet) Tritium fuel cycle Remote maintenance	Plasma power balance at ignition Tritium fuel cycle Remote maintenance 25% availability Superconducting toroidal and poloidal coils in a fusion environment First wall and divertor collector plate performance

[a] GDC = glow discharge cleaning.
[b] PDC = pulse discharge cleaning.

In addition to the demonstration of DEMO characteristics, it is imperative that INTOR verify and expand the plasma physics data base for the ignition regime. This is important for establishing more detailed models to predict the plasma behaviour in future devices and thus to ensure reliable, optimized operation. Plasma physics experiments which are intended to provide such information will be performed during the final phase of Stage-I operation.

4.2. Operation plan

The operation plan for the three years of Stage I is given in Table XIV-4. It comprises four sequential phases: (1) vacuum vessel conditioning; (2) Ohmic heating start-up; (3) neutral beam start-up; and (4) long-pulse ignition operating/ plasma physics experiments. The operational objectives are listed for each phase and are described in more detail in the following sections. Maximum values of relevant machine parameters are indicated for phases (2)–(4), together with the reactor-relevant technologies demonstrated. The plan places strong emphasis on the long-pulse ignition experiments and on achieving a 25% availability in this regime.

It is assumed that all necessary engineering tests on the power supplies, coils, vacuum and gas handling systems, beams, etc. have been performed before Stage I, although some overlap of such tests with the conditioning period is possible. Also, confirmatory measurements of stresses at maximum plasma current and during disruptions will probably be required. It is further anticipated that modifications will be made, especially for diagnostics, whenever the vacuum is opened during Stage I. It is, in fact, essential that the remote maintenance system be thoroughly exercised before Stages II and III.

5. PLASMA ENGINEERING TESTS

Plasma engineering tests for INTOR have not been defined yet. A test plan will have to include the information to be gathered from experiments on large tokamak machines now under construction and from the operation of INTOR during Stages I and II. Plasma engineering tests of plasma-wall interaction and long-pulse operation as well as plasma heating will be required for INTOR design. For a reliable operation of DEMO and of future power reactors, plasma control and operation should be as simple as possible. Examples of possible plasma engineering tests on INTOR are:

(a) Tests of new impurity control and exhaust concepts, such as a mechanical divertor, under burning plasma conditions
(b) Experiments for studying non-uniform energy deposition (during disruptions, due to shine-through, etc.) on the divertor plates and the first wall

RF launcher tests
Optimization of burn control.

These tests may be perfprmed at available experimental ports with a total a of about 3 to 5 m^2 or at neutral beam injection ports. Plasma engineering ts which will require a drastic change of the INTOR configuration are acceptable; however, some tests such as the application of helical winding stability control coils may be feasible. Nonetheless, this kind of experiment ould be performed by hydrogen tokamaks. Plasma engineering tests that will ıbstantially alter the plasma operating characteristics must be scheduled during tage I. Only a limited number of such tests could possibly be performed during tages II and III, since the engineering tests (e.g. for blanket, materials) require table, reproducible and well-controlled plasma operation. A full 1/12th sector)f the first wall/shield has been allocated to plasma engineering tests.

6. BLANKET ENGINEERING TESTS

The primary purpose of blanket module tests in INTOR is to provide operating experience and design verification for extrapolation to a DEMO blanket design. The programme of blanket testing will provide basic engineering data in a fusion environment and deal with the following three main issues which cannot be otherwise evaluated:

(a) The capability of each of the blanket modules designed for INTOR testing to function in a fusion environment
(b) The response of the blanket modules to extreme conditions such as plasma disruptions and various failure/accident scenarios
(c) The selection of the best blanket design(s) for DEMO.

In addition, certain data required for the success of INTOR will be obtained in the early part of the blanket engineering test programme.

Since INTOR will not be able to simulate completely the DEMO operating conditions or to provide for large numbers of blanket module tests, it is important to optimize the test modules in order to maximize the relevance of the test results to DEMO application.

The INTOR testing of blanket modules will be of two general types. One will establish a data base for operation under normal operating conditions, the other will examine module responses under off-normal operating conditions. Both kinds of testing will have two essential features. First, to minimize time requirements on INTOR and overall costs, the testing will be of the benchmark or proof type, confirming or validating the results obtained in non-fusion-environment testing. Second, tests planned on INTOR will be configured so as

TABLE XIV-5. SUMMARY OF BLANKET ENGINEERING TESTS

Blanket test category	Demonstration	Tritium recovery	Specialized engineering
Objectives	Design-specific engineering issues (mechanical behaviour, materials response, neutronics, thermohydraulics), verification of design, demonstration of performance	Tritium recovery process, containment, breeder performance, extreme condition tests	General engineering issues (piece part behaviour, joints, seals, etc.), extreme condition tests
Character	DEMO LOOK-ALIKE	DEMO ACT-ALIKE	DEMO ACT-ALIKE
Number of test stations	4	1	1
Number of modules	4 – 8	1 – 2	8 – 20
Test duration (stages)	12 years (II, III)	0.5 – 2 years (I, II, III)	0.5 – 2 years (I, II, III)
Installation	unpocketed	pocketed	pocketed
Module size	1 – 3 m^2	1 m^2	1 m^2

to minimize the probability and adverse consequences of a blanket module failure which may be detrimental to INTOR.

Table XIV-1 (see Section 1) compares the anticipated operating conditions of INTOR, of a DEMO and of a commercial fusion reactor. Note that heating and neutron loading are much lower for INTOR as compared with DEMO or commercial reactors, the number of power cycles being higher in INTOR. The INTOR blanket engineering test programme attempts to accommodate and perhaps take advantage of these differences.

Two basic module design approaches have been considered:

DEMO LOOK-ALIKE
DEMO ACT-ALIKE

The DEMO LOOK-ALIKE approach would simply utilize the DEMO blanket design as specified and insert it in INTOR. Differences in INTOR and DEMO operating conditions may limit the value of these tests. For example, the lower nuclear heating rates may reduce the generally life-limiting cyclic thermal stresses by a factor of two to three. This difference, coupled with others, would greatly decrease the value of the tests unless appropriate scaling is used in test design.

The DEMO ACT-ALIKE tests could be developed by identifying failure modes for a specific blanket design and developing a special test to simulate each failure mode. For example: (1) the coolant flow may be slowed down and the pressure increased to simulate temperatures and stresses which are important to an irradiation creep failure mode, and (2) the structural temperature may be increased to enhance swelling in order to simulate a creep failure mode. It is unlikely that all failure modes can be simulated simultaneously in a single large test module (1 m $\times$ 1 m). For this reason, an approach to testing would be pursued whereby a number of smaller elements of a given blanket are tested independently.

The blanket module test programme incorporates several different blanket module configurations which represent different aspects of breeding blanket technology. They are of the DEMO ACT-ALIKE type in the sense that their behaviour characteristics and failure modes are expected to be similar to those of a DEMO blanket module using the same technology.

Three kinds of blanket engineering test modules have been considered and are summarized in Table XIV-5. The first is a large blanket module with the overall configuration and components of a DEMO blanket. The second kind treats the problems of tritium breeding and recovery, and the third one is used for specialized engineering tests examining components and sub-elements of particular blanket designs. In all cases considered, except for the hybrid blanket type, a module size of 1 m^2 or smaller was found to be adequate to evaluate the effects of a fusion environment.

6.1. Description of blanket concepts

Several blanket module designs of the DEMO LOOK-ALIKE type were considered as an aid in defining the INTOR blanket module test approach; however, these were not necessarily the designs actually considered. The following blanket configurations were considered in Refs [2–5]:

Helium-cooled liquid lithium
Liquid lithium
H_2O-cooled solid breeder with graphite moderator
Helium-cooled solid breeder
H_2O-cooled hybrid blanket.

The final designs to be tested will be selected later in the INTOR programme, after a more comprehensive international fusion development plan is defined.

Two modes of blanket test module installation were considered. In the "pocketed" mode of installation the test modules are placed in pockets or recesses in the shield structure. These isolate the experiments from first-wall effects, but allow neutron and gamma radiation to stream into the test module. This installation has the principal advantage of maintaining vacuum integrity and associated cleanliness of the plasma chamber during test change-out. The second or "unpocketed" mode of installation is to simply leave ports or apertures in the first wall/shield into which the test modules would be placed. This configuration allows the demonstration test modules to be fully exposed to the plasma, whereby the concomitant heating and erosion effects provide a more realistic test environment.

6.2. Design verification tests

The first tests conducted on the demonstration test modules will have three functions: (1) to proof-test module designs in a fusion environment; (2) to explore blanket module design response to off-normal conditions (depending on module type, such test conditions will include loss of flow, loss of coolant, liquid-metal heating system failure and induced leaks; and extremes of temperature, pressure, surface heat load, etc., caused by off-normal reactor conditions); and (3) to proof-test module designs on which changes or alterations have been made.

An important characteristic of design verification testing is that frequent change-out of test modules will be required. Likewise, there will be frequent and extensive post-test examination and analysis. In addition, especially in off-normal condition testing, there will be a somewhat greater probability of test item failure than in the demonstration tests. For these reasons, design verification testing will be conducted in the "pocketed" configuration, i.e. there will be a fixed barrier between the test modules and the interior of the reactor. This will significantly speed up the change-out process and minimize risk to the INTOR facility.

The specific details of test configurations and modes will be determined in the process of blanket module development.

6.3. Demonstration blanket tests

The most comprehensive testing in the blanket engineering test series will be the demonstration tests. These will demonstrate DEMO test blanket module performance as indicated by analyses and by simulation tests in other facilities. The required test conditions include power excursions at high and low levels, long and short burns, routine interruptions of operation, and such other features as may be expected in the course of DEMO operation.

It appears necessary to conduct demonstration tests in the "unpocketed" configuration, with the first wall exposed to the plasma. First-wall heating and particle erosion effects from the plasma are significant to the overall functioning of a breeding blanket, particularly in designs where first-wall cooling is integral with module bulk cooling.

The demonstration tests will serve to verify assumptions made in test module design and/or generate more realistic data for the analytical assessment of module behaviour. The objectives of these tests are discussed below.

Mechanical behaviour. The goal is to achieve a global test of the structural integrity of the module as a whole. Particular aspects include thermal cycling effects such as fatigue and thermal ratcheting, creep, weld behaviour, seal integrity, and response to transient loads.

Materials response. Issues of concern include radiation effects on materials, such as swelling, embrittlement, sintering and transmutations. Synergisms between these effects and structural integrity and module performance will also be studied. Additional issues include chemical interactions between structure, breeding material and coolant in the presence of impurities and radiation, corrosion, tritium release and transport.

Neutronics. One primary focus of these tests will be the comparison of calculated values of neutronic behaviour with experimentally determined values obtained under actual fusion conditions, thus verifying methods and data used in design and analysis. Nuclear heating profiles, breeding profiles, activation and afterheat, and neutron and gamma-ray spectra will be studied.

Thermohydraulics. These tests will allow the comparison of calculated results and data obtained from ex-machine testing with experimental values obtained under INTOR test conditions. Parameters of interest include temperature and flow distributions, pressure distributions, thermal and pressure stabilities, control effectiveness, and overall thermal performance.

Demonstration testing will be initiated in Stage II of INTOR operation and continued in Stage III. An objective in these tests is to expose the demonstration test modules to fluences that are within a factor of two or three of the DEMO lifetime fluence for the blanket and to gain operating experience on these modules. While it will not be possible to fully qualify these modules for use on DEMO, enough experience on the designs should be obtained to give confidence that they will function properly, at least for the early stages of DEMO operation.

Special requirements are indicated for test item removal or replacement. It is planned that, once a demonstration test module is installed in INTOR, it will not be necessary to remove it until the test is completed, except when unfavourable test data or a possible failure would require removal. It will be necessary to remove periodically certain items from the experimental modules, such as dosimetry and breeding capsules. This is planned to be done during routine down-times. While manual removal would be possible if appropriate safeguards were employed, it is planned to use remote manipulators. Not more than one day should be required to remove and replace capsules. Removal and replacement of the blanket test modules will be a much more complex operation. It must be done remotely, using special equipment designed for this operation. It will probably require several (1–4) weeks to complete the removal and replacement of demonstration test modules, even if special care is taken in the design of the experimental hardware. All items removed from the reactor will require a hot cell for storage and examination.

6.4. Tritium recovery tests

The demonstration of tritium breeding and recovery capability in a fusion environment is critical to the success of the INTOR mission of preparing for DEMO. The testing goals for tritium breeder processing include the following:

(a) Demonstration of steady-state tritium recovery from candidate breeder blankets under credible conditions that would yield an acceptable blanket tritium inventory (e.g. $\leqslant 1$ kg)

(b) Demonstration of an acceptable level of tritium containment in breeder blanket systems that are operating under reactor-typical thermohydraulic conditions (e.g. a loss rate of $\leqslant 1$ Ci/day/1000 MW(th)).

Several aspects of tritium breeding and recovery technology will require demonstration before blanket testing can begin. These specialized tritium breeding and recovery tests are intended to measure specific, separate effects related to the production and recovery technologies. They will serve to verify in the fusion environment of INTOR the results obtained in ex-machine simulation testing and in laboratory simulations.

Several kinds of data requirements for these specialized tests have been identified. They include proof-testing for the large blanket modules of the demonstration tests, testing of alternative breeder concepts for tritium breeding and recovery, testing at extreme conditions, and certain post-test analyses to study the effects of tritium production on breeder materials. There will be a need for data such as heat transfer effects in liquid metals subjected to magnetic fields, effects of sintering on tritium release, effects of tritium-formation energy release on the microstructure of ceramic breeders, and tritium migration and containment effectiveness.

The tritium recovery tests will deal with INTOR-relevant and DEMO-relevant issues. Many of the issues discussed above relate to testing in INTOR for which data will be required before the beginning of demonstration tests. Hence, the specialized tritium breeding and recovery tests for experiments in INTOR must be conducted late in Stage I or very early in Stage II. The DEMO-relevant work will be done in Stage III and will have the principal objective of providing DEMO designers with the data needed to ensure its proper functioning.

6.5. Specialized engineering tests

The specialized engineering tests will treat specific, component-configuration-dependent issues which cannot be evaluated properly in simulation facilities. Some specific investigations which may be included in these tests are: temperature rise times; materials integrity; welding performance; phase, structural or compositional changes in breeder and/or coolant materials; induced oscillations; and pressure drops or rises. Also failure modes may be observed.

The response of blanket demonstration test modules to extreme conditions, and the failure modes and effects of these test items must be examined experimentally before the start of the demonstration tests. The design verification tests in this group will be conducted in the latter part of Stage I of INTOR operation. These tests will be conducted on a non-interference basis with other ongoing work related to machine operation and control.

The other group of specialized engineering tests will be structured to answer questions raised by the demonstration tests about the readiness of blanket designs for operation at DEMO. This type of testing will be useful in those instances where INTOR will provide a DEMO-relevant test environment which cannot be provided in other test facilities or where the answers will not be readily obtained from the demonstration tests.

The DEMO-relevant specialized engineering tests will be conducted in the latter part of INTOR Stages II and III. These tests will occupy the "pockets" used in Stage I for the INTOR-relevant work.

6.6. Special requirements

One important item with regard to DEMO-relevant tests is the limitation of available neutron fluence in INTOR. Most of the information now lacking in blanket design is related to radiation and its influence on material performance. Life-limiting effects are frequently quantified in terms of atomic displacement or other parameters related to neutron fluence. Without sufficient neutron fluence, it will not be possible to determine accurately what the lifetime of DEMO blanket components will be.

INTOR testing is desirable to obtain all attainable information from a fusion environment before the construction of DEMO. It will not be possible to develop accurate lifetime estimates or reliability information on components, but it may be possible to verify that the probability of failure in the early part of their lifetimes is low.

The entire test programme will be very carefully integrated before testing begins. Wherever there are plans to change one test item for another in the course of the INTOR mission, the details of the hardware interface will be firmly fixed and optimized for ease of installation, using remotely operated tools and the most advanced methods of remote handling technology. Particular attention will be given to shielding and neutron streaming paths in the design of the experiments.

In scheduling the various tests, it will be planned to use every experimental port or pocket when the reactor is in operation. If a test port is not needed for blanket or materials testing, it will be filled with a shield plug. Frequent changing of test pieces will be easier during Stage I when the machine availability will be low. Frequent changes later in the programme may compromise machine availability objectives.

In summary, it is clear that INTOR will provide a good test bed for blanket modules and components as compared with ex-machine testing. An extrapolation by a factor of 2–3 will still be required for DEMO design, which is probably acceptable.

7. MATERIALS TESTING – BULK PROPERTIES

Materials tests in INTOR will provide information on radiation effects at low and moderate fluences and on surface effects (plasma-materials interactions) under reactor-relevant plasma conditions. INTOR is useful as a test facility for both areas. The relatively large volume available in INTOR compared with accelerator-based sources and even with fission reactors is consistent with the amount of testing that will be needed to characterize the performance of materials in a true fusion environment.

INTOR data on bulk properties at low to moderate fluences will be compared with, and will hopefully verify, trends and correlations of properties that have been derived from other sources, primarily from fission reactors and neutron spallation sources such as FMIT in the USA. Since INTOR will not produce data at high fluences (greater than ~5 MW·a/m^2 equivalent exposure for the first wall), good correlations between INTOR data and data obtained at similar fluences from other sources will give confidence in predictive correlations.

As testing in INTOR progresses, there will presumably be a trend from general verification of correlations to more intensive testing where critical changes in properties are identified. INTOR is not a test bed for materials selection but rather for obtaining critical engineering data after a primary material and one or two back-up materials have been selected. The general data requirements indicate a variety of properties of interest which are appropriate for an initial phase of testing and correlations. The subsequent test programme will concentrate on greater numbers of specimens and a more limited range of variations among the parameters. Testing will provide bulk properties of materials for a variety of applications. Materials to be included are structural, high-heat-flux and heat-sink materials; breeders, multipliers and insulation. Emphasis will be placed on obtaining properties of those materials nearest to the plasma which are subjected to high fluences.

The test matrix for materials irradiations in INTOR was developed to specify the types of materials tests, with details on specimen numbers, irradiation temperatures, fluences, stress, change-out frequencies and post-irradiation testing conditions. The test matrix provides a basis for estimating the volume needed for materials testing in INTOR. Also the frequency of specimen change-out must be accommodated by the channel test design with minimal impact on reactor availability.

The bulk material test matrix is given in Table XIV-6. It is based on the assumption that one or two candidate materials have been identified (through extensive fission reactor and accelerator source screening programmes) for the various applications (first wall, armour, etc.) and that the extent of testing is determined by the number of specimens needed to determine quantitatively the critical properties of the irradiated materials.

The first portion of the test matrix was developed for candidate first-wall materials and other structural components receiving high radiation doses. The second portion of the test matrix was developed for non-structural materials subjected to high radiation, such as limiters, divertor plates, armours and RF components. It can be seen from the test matrices why large numbers of specimens are needed to provide design data.

The first column of Table XIV-6 lists the types of tests and the headings of columns (2)–(5) indicate important test parameters. The first parameter is "materials times variations" and refers to the number of base (candidate) materials and to the number of variations in the starting conditions to be tested, such as

TABLE XIV-6. INTOR BULK MATERIAL TEST MATRIX

(1) Test	(2) MAT[a]	(3) DUP[b]	(4) T[c]	(5) F[d]	(6) Conditions	(7) Total
Structural materials						
Tensile	5	3	6	5	6 rate/temp.	2 700
Fatigue (high cycle)	8	3	2	4	4 stress levels	768
Crack growth	6	2	4	4	3 stress levels	576
Fracture toughness	4	3	4	2	2 temp.	192
Swelling	10	5	10	5	4 post-irradiation tests	10 000
Creep and stress relaxation	6	4	6	–	5 stresses	720
Creep rupture	6	2	6	–	5 stresses	360
Corrosion	–	–	–	–	–	–
In-situ cyclic sputtering of first-wall and divertor material	4	2	2	–	4 stress levels	64
Other materials (ceramics, electrical and heat sink materials)						
Fatigue life	10	3	4	4	2 strain range	960
Tensile properties	15	3	6	4	4 rate/temp.	4 320
Swelling and phase stability	15	5	6	4	4 post-irradiation tests	7 200
Irradiation creep	10	2	6	–	4 stresses	480
In-situ cyclic loading	6	2	2	–	4 stresses	96
Fracture (DBTT)	3	2	2	2	6 temp.	144
Electrical properties	6	3	2	4	3 tests	132
Thermal conductivity	6	3	4	4	–	288
						~ 29 000

[a] Materials times variations.
[b] Duplication of specimens.
[c] Irradiation temperatures.
[d] Fluence levels.

heat treatment and weldment. Column (3) refers to duplication of specimens, primarily for verification of data reproducibility. In some cases, such as swelling and phase stability studies, the duplication also allows for possible losses in the post-irradiation preparation of samples. Columns (4) and (5) refer to the numbers of separate irradiation temperatures and goal fluence levels, respectively.

Typically, parallel tests are performed at several irradiation temperatures in order to evaluate the expected temperature-dependent behaviour over the range of interest. Usually, a linear dependence of materials properties on temperature is not expected, especially near ceiling temperatures (above which useful service is not expected). In the test matrices for INTOR, the number of irradiation temperatures varies from four temperatures for several mechanical property tests to up to ten temperatures for observations of swelling and phase stability.

The irradiation experiments typically include tests at a goal fluence level, representing end-of-life conditions, if possible, and several interim fluences. Preliminary predictions of material behaviour, based on low-exposure data, are used for re-directing the experimental objectives and for preliminary design data, pending corroboration with data for high fluences. However, the extrapolation of low-fluence data to predict properties at high fluences can be grossly misleading. As a rule of thumb, an extrapolation by less than a factor of two in fluence (for moderate fluence levels, i.e. exceeding incubation doses for property changes) is acceptable. Data at several values of moderate to high fluences are desirable to define trends, especially where the properties after irradiation are only marginally acceptable. Goal fluences of 0.6, 0.66, 2.0, 3.3 and 6.6 $MW \cdot a/m^2$ are appropriate for tests in INTOR on structural materials. Components using "other" materials were assumed to be more easily replaceable than the first wall and able to withstand lower fluences.

Column (6) indicates test conditions other than temperatures and fluences. Most of the conditions, for example the strain ranges in fatigue tests, are those in post-irradiation tests performed after the specimens are removed from INTOR. Another example of post-irradiation test conditions is that of tensile tests where both the strain rate and the post-irradiation test temperature are variables. Typically, post-irradiation tensile tests are performed at the irradiation temperatures, at a temperature slightly above the irradiation temperature and at the temperature expected in the material during refuelling or repair. Designers use these data to predict the behaviour during operation and in fault conditions such as overheating and seismic response.

In the case of creep tests, the other "condition" is stress. One of the current techniques for in-reactor creep tests utilizes sealed tubes pressurized to different stress levels. The diameters of the tubes are measured at several fluence intervals, after which the specimens are again placed in the reactor for further exposure.

The last column of Table XIV-6 gives the products of the previous columns. For example, the fatigue tests will require 8 material conditions with a duplication

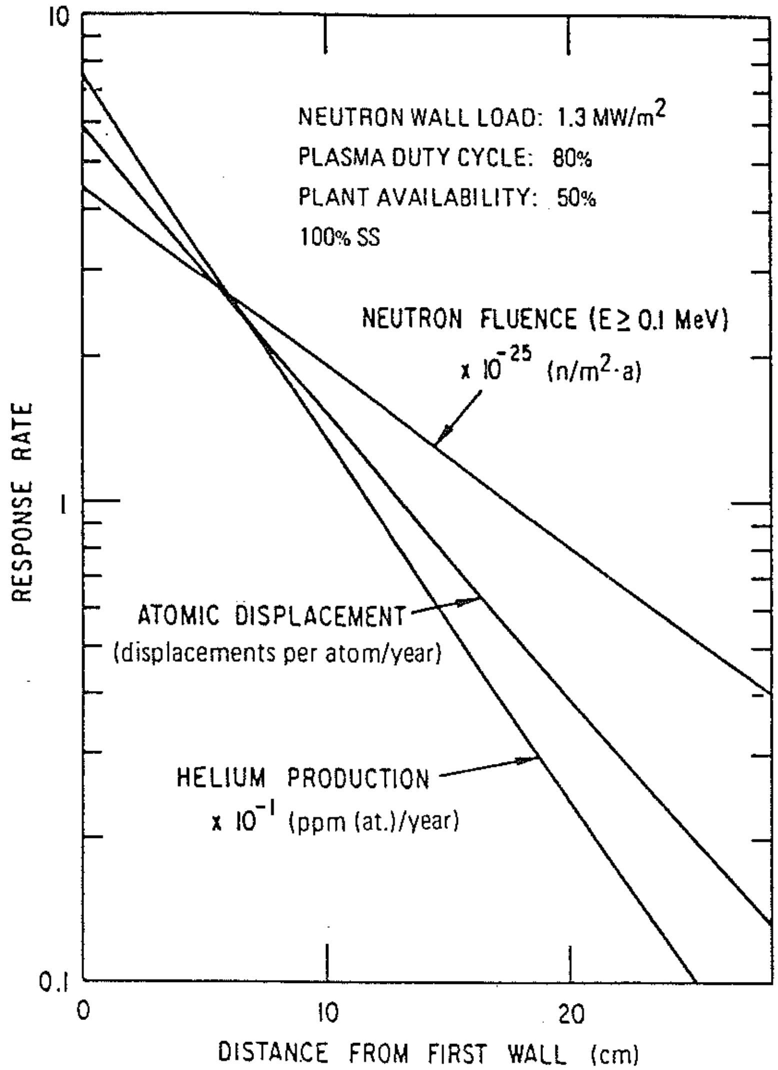

FIG.XIV-2. Nuclear response rates in experimental module: displacements per atom versus distance into specimen capsule.

of 3 specimens, each at 2 temperatures and 4 fluences, and for 4 stress levels in the post-irradiation tests. The product is 768 specimens – a fairly large number for a seemingly modest set of requirements. Thus, the number of specimens needed to provide reliable data for design is much greater than might be imagined.

A detailed description of the test specimens is given in Ref. [4]. Detailed designs for material test channels were developed to determine how much space was required for materials testing and to evaluate the approximate fluence that could be obtained in the test specimens. It was shown that 1 m^2 could accommodate the tests listed in Table XIV-6. In addition, it was recognized that more extensive tests should be performed to develop an understanding of single-variable response to combinations of flux, temperature and cyclic effects. For

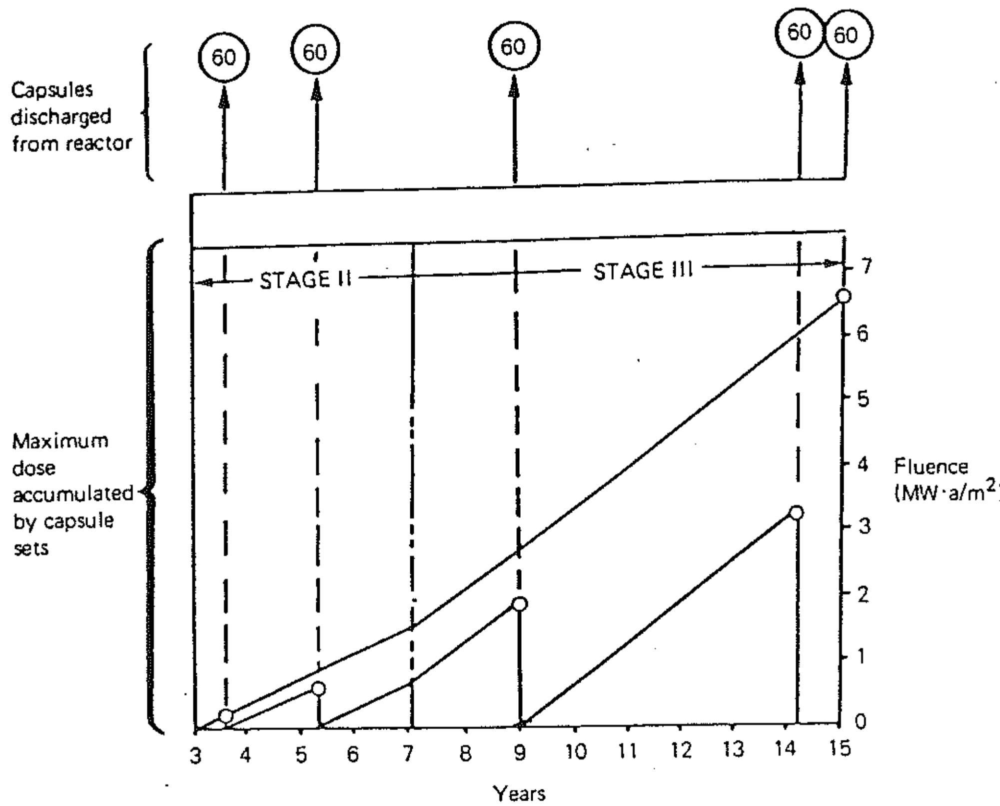

FIG.XIV-3. Capsule (circles) irradiation sequence and maximum accumulated dose as a function of time.

example, a typical test might be to hold a single variable (temperature) constant, and to vary the specimen stress during irradiation. A second 1 m^2 test channel was provided for these tests.

Displacement damage was determined as a function of distance from the front of the test channel and is shown in Fig.XIV-2. A maximum of approximately six displacements per atom per year can be attained. Specimen arrangement and change-out sequences must accommodate the damage gradients.

To achieve the desired neutron exposures corresponding to 0.2, 1.5, 2.5 and 5 MW·a/m^2 of first-wall exposure, a total of 300 capsules containing ~30 000 test specimens are required. A schedule for capsule insertion and removal is shown in Fig.XIV-3. Two capsule change-outs occur during Stage II and three occur during Stage-III operation.

7.1. Utilization of INTOR data

The materials data generated in INTOR will be necessary for the confident design of a demonstration fusion reactor, even though this reactor may have different wall loading and burn cycle parameters. The justification of this statement requires that some background be provided on the parametric sensitivity of the mechanisms involved in dimensional and mechanical property changes, and also on the steps required to develop fusion design equations from fission-derived data.

The changes induced by irradiation in the behaviour of some alloys can be directly correlated to two concurrent microscopic evolutions, one involving substantial changes in the dislocation and cavity microstructure and the other involving an extensive separation of various elements between the alloy matrix and precipitate phases. Both evolutions are sensitive to many variables. The most consequential variations in the 300-series stainless steels reside in the microchemical evolution, with displacement rate, temperature history and stress state being the most sensitive. In a fusion device, these three variables will be cycled through large ranges simultaneously. Since single-variable experiments on the influence of each of these parameters have shown pronounced effects, it is expected that strong synergism will exist when these parameters are simultaneously varied. In addition, the chemical nature of this evolution means that transmutation processes may have a pronounced effect.

It is important to recognize that the overwhelming majority of the materials data base relevant to fusion reactor design has been and will be developed in thermal and fast breeder fission reactors. Compared with fusion devices, these reactors generate different rates of transmutation, different helium/displacement ratios, and large differences in the number and possibly in the distribution of point defects per unit neutron energy. In the available thermal reactors, large test volumes and high displacement rates generally cannot be found in the same facility. The utilization of fission data for fusion reactor design therefore requires that correlations be developed between the material response in fission environments and that in fusion environments.

The only experimental devices producing 14 MeV neutrons do so at very low fluxes. In the mid-1980s the FMIT (Fusion Materials Irradiation Test) facility will be available, which will produce neutrons in a broad energy range with a mean near 14 MeV. This facility will not only provide the closest simulation of the required helium/displacement and transmutation/displacement ratios but will also operate at the required displacement rates. The irradiation volume in FMIT is very limited, however.

The available irradiation facilities suffer from three deficiencies which INTOR does not have. First, each has a limited irradiation volume relative to INTOR. Second, none of these facilities provides an exact simulation of the expected environment, matching simultaneously the displacement rate, helium/displacement ratio, point defect distribution and transmutation characteristics. Third, and possibly most important, all of these devices operate in essentially steady-state modes. As mentioned previously, transients of displacement rate, temperature and stress have been shown to have a significant and major effect on the microstructural/microchemical evolution. Of course it is anticipated that a demonstration plant would have wall loading and burn cycle characteristics different from those of INTOR, but the magnitude of the differences is small compared with the extrapolations from the fission data base. The details of the burn cycle are not really very important for microstructural development if the power-on times are long compared with the transients in vacancy and interstitial populations. The anticipated flux difference between INTOR and commercial plants is perhaps a factor of 2–4. This relatively small flux range has already been bridged in the design of second-generation breeder reactors. There are some properties such as fatigue crack growth which appear to be relatively insensitive to radiation-induced microstructural development, but which are sometimes sensitive to hold-time and frequency effects. For the 300-series stainless steels in vacuum or sodium environments, the data derived from the breeder programmes show a dependence on maximum stress and total number of cycles, but no effect of hold-time or cycle frequency. The major contribution of INTOR data to the development of fission-fusion correlations will be in the determination of the influence of temperature and stress cycling on the radiation-induced evolution.

Other irradiation programmes have shown that comparative experiments directed toward identification of single-variable response are the best route on which to proceed. Therefore, the types of tests envisioned fall into the following categories:

(a) Encapsulated experiments which allow the flux and temperature to cycle simultaneously on unstressed specimens
(b) Encapsulated experiments which do not allow the temperature to cycle with the flux on unstressed specimens
(c) Dynamic tests where stress is applied and allowed to cycle with flux and temperature.

8. SURFACE MATERIALS TESTS

8.1. Objectives of surface materials tests

Plasma-wall interaction is of great importance as it substantially affects the performance of the plasma and the integrity of the wall surfaces. INTOR studies have shown that the first-wall components and the divertor plates are eroded rather rapidly by plasma-wall interactions during the operation of the reactor, and that this erosion not only may be a problem for plasma operation but also is a life-limiting factor for components whose surfaces are exposed to the plasma.

Plasma-wall interaction is expected to continue to be an important aspect of the design for DEMO and other devices that will follow INTOR. An extensive programme for surface testing in INTOR has been developed and is described in this section.

The objectives of the surface materials test programme are:

(a) To obtain information that will enhance the data base needed for the design of DEMO and other reactors that will follow INTOR
(b) To provide proof-of-concept tests for selected surface structures
(c) To examine the performance reliability of selected surface structures.

INTOR will permit tests to be made in a fusion environment on the influence of synergistic effects on sample surface performance, to an extent that could not be achieved elsewhere. The synergistic effects are caused by the simultaneous irradiation of surfaces with neutrons, atoms, ions, electrons and photons. The synergistic effects may influence the sample surface performance with respect to particle retention/re-emission, plasma impurity release and surface erosion/impurity deposition.

INTOR will not be a substitute for simulation experiments, since the range of operating conditions is limited and therefore the results will be valid only for a narrow range of parameters which may not correspond to the expected DEMO conditions.

The most important surfaces are those of the first wall and of impurity control components such as divertor collector plates and limiters.

8.2. Test matrix

The three major types of radiation conditions in INTOR to which surface structures will be exposed and in which tests are possible are: (1) discharge cleaning conditions (e.g. glow discharge or plasma discharge); (2) normal plasma radiation, including neutrons, alpha particles and charge-exchange neutrals; and (3) plasma disruptions and other plasma fault conditions. The tests and observations that can be executed in INTOR are as follows:

(a) *Fuel and fusion reaction product retention and re-emission from the surface.* The particles leaving the plasma may penetrate into the confining surfaces because of their high energy and form trapped gases that are partially released after a certain time. Changes of the plasma composition are a consequence of the differences in the trapping and release behaviour.
(b) *Tritium permeation.* The permeation of tritium to the coolant is affected by the retention of the fuel.
(c) *Plasma impurity release from the surface.* The release of any material from the surface will have some influence on the vacuum in the start-up phase and

on the plasma during the operation phase. Information on the composition and fluxes of these materials will give the possibility to improve the operation of the machine. Certain in-situ diagnostics must be applied in these tests, an example being laser fluorescence spectroscopy.

(d) *Surface erosion* has been identified during the INTOR study as potentially life-limiting for many components. The experiments will give the opportunity to measure the erosion due to sputtering and disruptions, and to follow also the re-deposition and behaviour of the eroded material. It would be necessary to extend the range of conditions by defining test locations at different places (e.g. first wall, divertor and divertor chamber).

(e) *Surface microstructural changes.* A combination of many damaging mechanisms (e.g. preferential sputtering, segregation of alloying elements, bubble formation) will alter the properties of the surface material.

The materials selected for surface tests will generally be limited to one primary candidate material and one or two back-up materials for each specific component (e.g. first wall, divertor plate). These materials could be homogeneous alloys and composite materials consisting of coatings or clad (or protective) material and a structural material alloy. However, it should be possible to insert in the test programme other materials from further DEMO studies which may need surface testing.

The sample size will be restricted in many types of tests to a coupon size of a few cm^2. Furthermore, it is necessary in some cases to test the surfaces of full-size components. Such tests will likely alter the plasma operating characteristics and will generally be limited to testing during Stage I. Post-irradiation analysis of components removed for maintenance purposes will provide information on INTOR materials that will reduce the need for many full-size component tests.

The present INTOR operating procedures provide for reactor shut-down only once each month. For some types of surface tests, the coupon-size samples need to be removed more frequently, in some cases every two days. Preliminary considerations were given to the design of a remote maintenance system that would allow for removal and insertion of samples without personnel access into the reactor building. Table XIV-7 gives an example for a possible test matrix.

8.3. Test area and test location requirements

An area of about 1 m^2 has been allocated for surface tests in INTOR. This area is adequate for the testing of the coupon-size samples, but it cannot accommodate full-sized component tests. The difficulties with the full-sized component tests can be resolved by performing these tests during parts of Stage I (when substitution of INTOR surface-related components by test-type components is permissible) and by using information obtained from post-irradiation examination of components removed for maintenance purposes. In addition, one divertor

TABLE XIV-7. TEST MATRIX FOR SURFACE EFFECTS OF FIRST-WALL MATERIALS

Type of test	Type of radiation environment	Materials times variation	Size of samples	Test sample duplication, number of samples for each material	Position (toroidal, poloidal)	Fluence	Minimum flux	Temp.	Total number of samples
Fuel and fusion reaction product retention/re-emission characteristics of surfaces	B_1 C_1 & C_2	i & ii j & jj	coupons (1 cm X 1 cm)	4	2 toroidal and 2 poloidal	$> 10^{19}$ cm^{-2} (D,T) $> 10^{18}$ cm^{-2} (α,n)	2×10^{16} cm$^{-2}\cdot$s^{-1} for D 5×10^{13} cm$^{-2}\cdot$s^{-1} for α,n	100°C 300°C 500°C	576
Plasma impurity release characteristics of surfaces (tests[a])	A_1 & A_2 B_1 C_1 & C_2	i & ii j & jj	coupons (1 cm X1 cm) large panels (size unspecified)	4	2 toroidal and 2 poloidal	$> 10^{19}$ cm^{-2} (D,T) $> 10^{18}$ cm^{-2} (α,n)	2×10^{16} cm$^{-2}\cdot$s^{-1} for D 5×10^{13} cm$^{-2}\cdot$s^{-1} for α, n	100°C 300°C 500°C	960
Surface erosion/impurity re-deposition characteristics	A_1 & A_2 B_1 C_1 & C_2	i & ii j & jj	coupons (2 cmX2 cm)	4	2 toroidal and 2 poloidal	$> 10^{20}$ cm^{-2} (D,T) $> 10^{18}$ cm^{-2} (α,n)	2×10^{16} cm$^{-2}\cdot$s^{-1} for D 5×10^{13} cm$^{-2}\cdot$s^{-1} for α,n	100°C 300°C 500°C	960

Surface microstructural changes	A_1 & A_2 B_1 C_1 & C_2	i & ii j & jj	coupons (2 cm X 2 cm)	4		2 toroidal and 2 poloidal	$> 10^{20} cm^{-2}$ (D,T) $> 10^{18} cm^{-2}$ (α,n)	$2 \times 10^{16}\ cm^{-2} \cdot s^{-1}$ for D $5 \times 10^{13}\ cm^{-2} \cdot s^{-1}$ for α,n	100° C 300° C 500° C	960
Changes of mechanical and physical properties	A_1 B_1 C_1 & C_2	i & ii	coupons (size unspecified)	4		2 toroidal and 2 poloidal	$> 10^{20} cm^{-2}$ (D,T) $> 10^{18} cm^{-2}$ (α,n)	$2 \times 10^{16} cm^{-2} \cdot s^{-1}$ for D $5 \times 10^{13} cm^{-2} \cdot s^{-1}$ for α,n	100° C 300° C 500° C	768

A – Discharge cleaning (A_1 – glow discharge, A_2 – pulse discharge).

B – "Normal" plasma radiation, including neutrons, alpha particles, charge-exchange neutrals (B_1 – including radiations typical for start-up (e.g. runaway electrons), B_2 – neutral beam shine-through).

C – Plasma fault condition (C_1 – plasma disruption, C_2 – disruption and runaway electrons, C_3 – full neutral beam power plus disruption).

i – First-wall candidate material; ii – first-wall back-up material.

j – Liner candidate material; jj – liner back-up material.

[a] Small-sample tests are only meaningful with in-situ diagnostics such as laser fluorescence spectroscopy.

channel will be allocated for proof-of-concept and reliability tests of surfaces of the divertor collector plates. The possibility of coupon tests at the divertor location is desirable.

While most of the surface tests will be performed in the 1 m^2 test pocket, a strong need has been identified for some surface tests in other locations in the plasma chamber. For example, plasma disruptions cause non-uniform deposition of power on first-wall components, resulting in localized surface damage and erosion. The corresponding test location has to be chosen in the region which is expected to receive the highest heat load (inboard first wall).

Another example is the re-deposition of eroded materials, which appears to be strongly position dependent. Therefore, a number of additional surface tests will be performed in different locations. Plasma physics considerations lead to considerable differences in particle fluxes and energies on the wall of the divertor chamber with respect to the first wall. In order to extend the range of test parameters, the possibility to do tests in this region should be envisaged.

For the application of certain in-situ diagnostics for plasma impurity releases from surfaces (e.g. laser fluorescence spectroscopy) and for surface erosion, the availability of ports permitting the viewing of irradiated surfaces (e.g. through the use of fibre optics) appears highly desirable. The problems involved in providing such "windows" in a radiation environment should be assessed in future work.

Characterization of particle and energy fluxes at test locations is an important aspect of surface testing.

9. SURVEILLANCE TESTS

9.1. Tasks of surveillance tests

The working conditions in INTOR for materials and structures will be new. Irradiation by 14 MeV neutrons, pulsed operation with high numbers of pulses, and the exposure of large complicated structures to a high radiation environment are situations for which only limited experience exists. Even when all available information is used, evaluations of the lifetime and reliability of many components will contain many uncertainties. Therefore, a well-devised plan for surveillance activities that encompasses all reactor materials and components is necessary.

Two sources of information will exist:

(a) Data on the performance of INTOR materials and components in the test facilities which are obtained by non-destructive examinations carried out specifically for testing the construction materials of INTOR.
(b) Very extensive analyses of any parts of the machine which will be exchanged according to schedule or owing to failure.

Examples of problem areas still subject to uncertainties are:

- Radiation damage by 14 MeV neutrons and its effects on the mechanical properties of the construction material and its weldments
- Swelling due to neutron irradiation
- Sputtering rates at the first wall
- Sputtering of the limiters
- Behaviour of the first wall under plasma disruptions
- Sputtering of the divertor plate and the divertor channel
- Fatigue damage of the construction material, components and structures
- Corrosion
- Deposition of the sputtered material
- Radiation damage of the superconducting magnets.

Part of the above-mentioned tests is discussed in Sections 7 and 8. The surveillance tests should include not only specimen tests, but also tests on typical elements of INTOR, such as, for example, tubes of the diameter used in INTOR, certain complicated weldments and other structures.

9.2. Pre-service qualification

The main task of the surveillance programme is to follow the change in performance of machine components under service conditions and to compare the results with the predictions of design evaluations. In order to perform this task, it is necessary to define very carefully the starting conditions. This has to be done by non-destructive testing on the built-in structures and components, and by characterizing the base material.

9.2.1. Non-destructive control of components

Given the complexity of INTOR, the design has to consider specifically the possibility of access for pre-service and in-service non-destructive examinations. Without a suitable design, many tests which might assist in system maintenance will not be possible. Non-destructive examinations should be foreseen in all areas which during design have been identified as being critical. An in-service inspection plan has to be developed on the basis of the predicted lifetime of the different components. Special examination procedures and methods need to be developed. Most of these methods have to be adapted to remote application. Techniques such as radiographic examination, ultrasonic control, visual inspection and acoustic emission will be required. The features to be controlled will be the dimensions of structural components, defect size and subcritical crack growth, and damage state of the material.

TABLE XIV-8. SURVEILLANCE TEST MATRIX

Test	MAT[a]	DUP[b]	T[c]	F[d]	Conditions	Total
Tensile	3	2	2	2	2 rates/temp.	48
Fatigue	3	2	2	2	4 stress levels	96
Crack growth	2	2	2	3	3 stress levels	72
Fracture toughness	2	2	2	2	2 temperatures	32
Sputtering	2	2	2	3	2 positions (in pile)	48
Depositioning	2	3	2	3	1	36
Outgassing	2	3	2	3	1	36
Gas retention	2	3	2	2	in pile	24
Tritium permeation	2	1	2	2	in pile	8
Disruption	2	3	1	2	1	12

[a] Materials times variations.
[b] Duplication of specimens.
[c] Irradiation temperatures.
[d] Fluence levels.

9.3. Base materials tests

In the design phase, the development of defects in the structure of different reactor components is predicted. On the basis of these predictions, the lifetime of critical components has been determined and a replacement schedule will be established if the components cannot withstand the damage during the whole reactor life. The predictions contain uncertainties. It is necessary to use INTOR itself to confirm or to correct these predictions. For this reason, a number of materials tests will be foreseen for the materials used in INTOR. These tests will be done in the test facilities described in Sections 2 and 8. They should be given high priority and are expected to give preliminary results already at an early stage of operation in order to help evaluate correctly the lifetime of important parts in the reactor structure. Tests of primary importance are:

- erosion rates at the first wall
- synergistic effects of 14 MeV neutron irradiation and fatigue
- embrittlement of base material and welds under the reactor conditions
- radiation damage of the insulators in the superconducting magnets and its influence on their mechanical and electrical properties.

Table XIV-8 is an example of a tentative test matrix for some of the materials properties.

9.4. In-service inspection

As indicated in Section 9.2.1, a great number of non-destructive examinations are done in order to define as precisely as possible the conditions of the machine at the beginning of operation. Parts of this inspection will be repeated at regular intervals during the shut-down periods. Such inspection will concern the first-wall thickness, the effects of disruptions at the inboard first wall, the shine-through regions, components containing subcritical flows, and dimensions of parts which might be subjected to swelling and creep.

One means of in-service control should be visual inspection, which will be particularly important for handling the problem of transport of eroded material. It will also be helpful to inspect the divertor plate, where the protective tiles may be easily detached. Strong damage by disruptions could be revealed by visual inspection. All in-service inspection has to be performed by remote control. It is therefore necessary to make detailed plans, to prepare the necessary means, and to provide for the most adequate procedures already during the design phase.

9.5. Operational history

The surveillance programme has first of all to furnish the data necessary for safe operation of INTOR. The unique experience offered by the device will be extremely helpful for the further development of fusion reactors. The test programme contains an extensive number of experiments which can contribute to further development only if the conditions in the reactor are well known. Examples of this dependence are:

- For an evaluation of radiation damage, it is essential to have a knowledge of the neutron flux and of the spectrum around the plasma and within the blanket.
- Temperature measurements are important, since the radiation damage in the construction material is temperature dependent. It is also important to determine the temperature variations during plasma pulses since they affect thermal stresses.
- Information on the effects of ion and particle flux to the surface is essential. Erosion processes and tritium permeation depend strongly on the situation at the first wall and the divertor.

9.6. Lifetime predictions

The purpose of the surveillance programme is to ensure satisfactory operation of the machine. The design aims at building the reactor and the different components in such a way that they will last for the projected lifetime. Some of the projections made include uncertainties. The results obtained during the surveillance programme will help to ensure that the reliability of the predictions is upgraded during the operation of INTOR. The surveillance programme can be considered successful when it is capable of avoiding unexpected failures and of indicating the timely replacement of components whose function is degraded to such an extent that failure has to be expected.

10. NUCLEAR TESTS

10.1. General description

10.1.1. Objectives and required neutronics information

INTOR will offer one of the first opportunities to perform important neutronics experiments in a radiation environment that is prototypical of a demonstration tokamak fusion reactor and in realistic blanket modules. INTOR will have not only higher fluxes than previous machines but also a significantly longer pulse time and a much higher duty factor. For planning purposes, this means that the neutron source can be considered to be almost steady state.

Neutronics information is required throughout the INTOR testing programme:

(a) Testing of neutronics is required especially in the early stages of D-T burning for characterization of the basic machine performance. For example, the source intensities and the spatial variation of the source must be known. Traditionally, this area has been dealt with as part of the plasma-diagnostics studies.
(b) The INTOR shield should be verified also in the early stages of D-T burning, at a low power level.
(c) Neutronics information is required in proof-testing of various tritium-breeding blanket concepts. These studies include the determination of tritium breeding and nuclear heating.
(d) Neutronics information is needed in providing important engineering data, such as: the radiation environment at sites where materials damage is studied, around sensitive structural components and at the magnets; the activation of various reactor components, especially the first wall and blankets; and radiation streaming through the shields and around various penetrations.
(e) Neutronics information is needed to link the INTOR experiments with accelerator-based experiments, such as those using RTNS-II and FMIT, so that the whole body of information will be useful for the design of demonstration fusion breeders.
(f) Neutronics experiments can provide important data for validating neutronics codes and various modelling approaches.

10.1.2. Experimental methods

In general, in a discussion of experimental methods, it is convenient to distinguish between those that may be used in high-power environments and those that are limited to low-power environments. For both of these environments, active and passive techniques are considered. While active techniques are usually more desirable because they can provide a time-dependent history and typically have better accuracy, they are not always possible because of inherent limitations. Moreover, passive techniques can extend the utility of the irradiation facility by significantly increasing the possible number of concurrent measurements. Passive techniques are also more tolerant of harsh environments.

Critical facilities where experiments are conducted at low power have been the primary test bed for obtaining neutronics information for fast fission reactors. Many experiments of interest cannot be carried out in high-power reactors because of the lack of suitable experimental techniques. The use of critical assemblies has considerable advantages: they afford convenient access, an optimum experimental environment, and the capability of mocking up different material arrangements without the need of providing elaborately cooled assemblies and facilities for remotely handling these assemblies because of the high activation.

The impossibility of carrying out certain experiments in a high-power-density fusion environment results from the following inherent limitations:

(a) Count-rate limitation
(b) Radiation damage of detectors and/or electronic components
(c) Sensitivity to background radiation
(d) Temperature sensitivity of detectors and/or electronic components.

These points have a direct bearing on the neutronics programme of INTOR. It is proposed to carry out many of the neutronics studies at low power, as discussed below.

10.2. Nuclear experiments

Experimental methods that are available or that should be developed for carrying out the neutronics measurements in a fusion reactor environment have been assessed, especially methods for measuring tritium production, nuclear heating, reaction rates, and neutron and gamma-ray spectra.

Table XIV-9, taken from Ref. [4], lists the techniques for each integral parameter and indicates the accuracies and the mode of machine operation for which the technique may be used. It is seen that if good accuracy is required, it is necessary to use some active techniques. This is especially the case for neutron spectroscopy. Furthermore, it must be emphasized that most of the active techniques require that the machine be operated at low power and that

TABLE XIV-9. EXPERIMENTAL TECHNIQUES FOR NEUTRONICS

Integral parameter	Measurement technique	Machine operation			Accuracy (%)
		Low power		Full power	
		Low intensity	Low number of cycles		
Tritium production	Radiochemistry		x	x	5
	Mass spectroscopy		x	x	5
	Track recorder		x		–
	Proportional counter	x			–
Nuclear heating	Calorimeter		x	x	10
	TLD		x		10
	Proportional counter	x			–
Reaction rates	Foil activation		x	x	5
	Mass spectroscopy		x	x	5
Neutron spectroscopy	Foil unfolding		x	x	15
	Proportional counter	x			5
	NE-213[a]			x	10
Gamma-ray spectroscopy	Compton recoil	x			10
	NaI[a]			x	15

[a] Must be used outside the reactor.

the measurements be made before there is significant activation. Several of the techniques require further development and all of them must be thoroughly tested before they are used in the INTOR programme. No attempt has been made at this time to determine the effect of the fusion reactor electromagnetic environment on the techniques; this problem will have to be considered.

10.3. Other experiments

10.3.1. Machine operation

It is proposed that the majority of the neutronics experiments be performed at low power. The two main reasons for this are: (1) the ease and economy with which different configurations can be constructed, and (2) the broader range of quality experimental techniques that are available only under this condition.

Performance of experiments at low power involves two distinct modes of machine operation. The first mode is operation at the normal high plasma duty factor and normal plasma burn length, but with reduced intensities. The lowest intensities would be required for neutron and gamma-ray spectroscopy, for which the neutron flux at the measurement position would be of the order of $10^7\,\mathrm{n/cm^2 \cdot s}$ per unit lethargy in the peak. This will require that the plasma be driven by neutral beams (since ignition is not possible at such low fusion reaction rates). The details of the problem have to be worked out in the future. Measurements using this mode would be made in Stage I.

The second mode is operation at moderately high intensities, but with a low duty factor. Most of the tritium breeding, nuclear heating, activation, and neutron spectrum and spectral characterization measurements with foil activation or track recorders would fall into this category. The important features of these measurements are that simple test modules could be used and rapid access to the modules would be possible because the background radiation is low. Measurements using this mode would be made in Stage II.

10.3.2. Blanket experiments

The most important nuclear tests on INTOR will be those associated with the blanket. Two types of blanket modules are being used: low-power and full-power. Low-power demonstration blanket modules will preserve the geometrical complexities of the full-power demonstration blanket that they are mocking up.

The parameter of most interest for the blanket module is probably the tritium-breeding ratio. However, both the experimental determination and the interpretation of this parameter for an isolated blanket module is fraught with difficulties. Real difficulties are encountered in attempting to measure a reference tritium-breeding ratio for the blanket module with an accuracy of 5% because the differences in neutron spectrum and boundary conditions introduce significant biases.

The following programme in INTOR for investigating tritium breeding is proposed. Tritium production will be measured along the central axis, which extends from the front face to the reflector of the blanket module. In addition, a limited number of measurements along the radius will be made. Tritium production will be determined by radiochemistry and mass spectroscopy, with an accuracy of at least 5%. At two or three locations in the blanket, gas-filled counters will be used to accurately measure the neutron spectrum. At these locations, foil activation neutron spectroscopy packages will be exposed. The above measurements will be compared with detailed three-dimensional analyses incorporating the particular blanket module and the details of the rest of the machine when the module is in place.

In addition to the tritium-breeding investigations, nuclear heating will be determined by calorimeters at a few representative locations, giving the total heating. The gamma-ray component will be mapped with TLDs. Gas-filled counters may be used for measurements at the locations of the calorimeter measurements to determine the neutron component. Reaction rates will also be measured in the blanket. Gas production will be determined near the front face with mass spectroscopy and foil counting. Activation rates will be determined for the appropriate structural components of the blanket.

In general, a set of low-power experiments are proposed, analogous to the fast-reactor critical experiments during the early part of Stage-I operation.

10.3.3. Engineering experiments

While some of the engineering experiments can be carried out at low power, most of them will require full-power operation of the machine. For example, measurements of gas production rates and activation rates in the first wall can be carried out at a low power. However, measurements of activation rates and nuclear heating at the magnet will require full-power operation. Radiation flux measurements for leakage through the shield and around penetrations will probably require full-power operation. Some scoping measurements should be done at low power early in Stage I in order to see if there are any unexpected hot spots.

11. ELECTRICITY GENERATION

One of the objectives of INTOR is to demonstrate the generation of electricity under reactor-relevant conditions. This does not necessitate the generation of net electricity, but can best be achieved by the production of 5 to 10 MW of electricity in an accessible outboard region of the reactor for a period of several months.

The main reason for producing electricity in INTOR is the verification that the necessary technology is available. Since the technology for producing electric power from heated steam is mature, the major technological problem is the recovery of heat from a blanket which will breed and safely contain tritium. To maximize the thermal efficiency it is desirable to run the primary coolant at the highest temperature consistent with material constraints, which is currently accepted as being in the range of 300 to 500°C. (The tritium-producing blanket in INTOR will operate with a coolant temperature of about 100°C.) Once the steam has been generated in the secondary system, present power plant technology applies and there is virtually no technical development risk.

Another technological step of some concern to the generation of electricity from tokamaks relates to the pulsed nature of their operation. Commercial power systems must deliver a steady output. During the period between tokamak pulses no fusion power is produced, but output must be maintained. Hence thermal storage is required. The blanket system itself provides some thermal storage, and the requirements for additional storage are determined by the thermal characteristics of the blanket. If the blanket has a thermal time constant which is long compared with the period between tokamak pulses, it can continue to provide heat to the power system with very little degradation. The penalty for this is an extended warm-up period at the beginning of operation before electrical power production can begin. Blanket systems with short time constants would require extensive storage external to the reactor to provide heat during recovery. Various options exist for this storage, such as latent heat of melting, chemical heat of formation and sensible heat in a storage medium. Some present designs for tritium-breeding blankets appear to have thermal time constants of the order of hundreds of seconds, but others have very short thermal time constants. A demonstration goal for INTOR is therefore to provide uniform electrical power to a load.

The reason for demonstrating the generation of electricity on INTOR rather than waiting for DEMO is two-fold. First, as with any emergent technology, there will be not only problems which are now recognized as requiring solutions, but also unforeseen problems which will arise during development. By attacking the development of the technology early, the time available to resolve problems of both kinds is increased. This leads to the second consideration, which is cost. Although INTOR will be an expensive project, it will be much less expensive than DEMO, both in capital investment and in operating costs. Problems encountered with technological development on INTOR will therefore be less costly to solve than will the same problems encountered on DEMO. These factors suggest that it will be wise to take as large a step in advancing technology with INTOR as is possible within present knowledge and budgetary constraints.

With these considerations in mind and a commitment to generate electrical power from fusion, it seems clear that there is great merit in generating electrical power on INTOR using a high-temperature tritium-breeding blanket. There may also be the intermediate objective of producing electrical power in INTOR using other technologies which do not include producing and recovering tritium. These options are considered in further detail in the following sections.

11.1. Electricity generation without tritium breeding

If it is decided that there is sufficient merit in generating electricity during Stage II or Stage III without the simultaneous breeding of tritium, there are three approaches which may be pursued. One or two complete segments of the outboard breeding blanket might be replaced by high-temperature non-breeding segments; or low-temperature heat may be extracted from the permanent non-breeding inboard region of the blanket at a temperature high enough for electricity generation; or low-temperature heat may be extracted from the permanent outboard breeding blanket. This last option is included here since the permanent breeding blanket gives only partial breeding and the operating temperature would be kept low enough to prevent tritium diffusion into the coolant, so that its construction and operation could not be considered reactor relevant. The use of several test modules is also a possibility, but the total surface area available in modules will not exceed 12 m^2, so that the maximum power level that could be generated would be 3 MW(e) compared with the objective of 5 to 10 MW(e).

The use of one or two non-breeding replacement segments for electricity generation would allow the highest coolant outlet temperature of the three alternatives, but it requires the greatest modification of the tokamak and would result in a significant loss of breeding area. The design and construction of such non-breeding segments would pose fewer problems from a technological point of view relative to breeding segments, but consequently their operation would yield much less new design experience. The installation of the segments would require the use of remote maintenance equipment. There would be no additional cost for this remote equipment since it must be provided in any case, but the time taken to install and remove the segments would be considerable.

The use of lower-temperature coolant from the inboard shielding or the permanent outboard breeding blanket is attractive in so far as it minimizes the disturbance to the reactor structure and to the operational schedule, but it is technically uninteresting. Coolant temperatures up to 200°C would allow thermal efficiencies of up to about 25% to be obtained, so that a surface area in excess of 40 m^2 would be required to generate 10 MW(e). Advantages of these approaches are that they avoid the problems of remote installation and the associated delays, that negligible additional development is required, and that capital costs are minimized because no additional tokamak components are necessary. On the other hand, not only is the technology not reactor relevant but these low-temperature systems may be more expensive than high-temperature systems because of the greater flow rates involved. Furthermore, the same steam generators and turbines would not be usable during a subsequent demonstration of electricity generation with tritium breeding in which higher operating temperatures would be used.

The purpose of generating electricity without tritium breeding would be to demonstrate the continuous operation of INTOR at high availability for an extended period and to demonstrate that continuous electrical power can be generated from a pulsed reactor. The first of these demonstrations does not actually require that electricity be generated, and the second could be made with a non-nuclear heat source. The first successful generation of several megawatts of electricity from a fusion reactor could certainly be publicly claimed as an important milestone.

11.2. Electricity generation with tritium breeding

The demonstration of the simultaneous generation of electricity and breeding of tritium is one of the declared objectives of INTOR, and an essential step on the way to the design and construction of a DEMO reactor. Such a demonstration would take place during Stage III of INTOR operation and might last for two or three months. Two alternatives have been proposed, involving the use of either the existing permanent breeding blanket or a new blanket segment.

The simplest approach to the required demonstration would be to utilize a segment of the permanent breeding blanket, specifically constructed to operate at higher coolant temperatures and pressures, instead of the standard low-temperature segments. This modified segment would be operated at low temperature during Stages I and II and then, provided its performance had been satisfactory, it could be operated at a higher temperature during part of Stage III. This approach has the advantage that the special segment would have been installed before activation of the structure, at minimum cost or loss of operating time. It should be noted that the choice of the solid breeder material, lithium silicate, with water coolant contained in a small-diameter tube, allows the blanket design to be easily modified for high-temperature and high-pressure operation, and is one possible option for a DEMO reactor.

The great disadvantage of this simple approach is that the choice of breeder material and coolant, and the design of the blanket will have been fixed during the original design of INTOR, more than ten years before the beginning of Stage-III operation. In this case, no advantage could be obtained from the experiments in test channels and modules in INTOR during Stage II or from parallel development programmes. These experiments and programmes will be specifically designed to test alternative breeding blanket concepts and materials, both in and out of a reactor environment, in order to determine which blanket design is best suited for use in a DEMO reactor. It seems very likely that such tests over a period of ten years would lead to a design different from the one adopted for the cold INTOR blanket, in which case it would be desirable to install such a blanket in INTOR for testing in Stage III.

The alternative system for the demonstration of electricity generation with tritium breeding, therefore, involves the installation of one or two new segments in INTOR at the end of Stage III. The choice of materials and design would depend on tests undertaken in Stage II, and the objective of installing the segments would be to demonstrate the simultaneous operation of all necessary functions of a prototypical, full-scale blanket suitable for use in a DEMO reactor.

The extent to which the segments will represent a prototype DEMO blanket cannot be decided yet, but it should be noted that the most comprehensive testing of a segment might include useful heat recovery for electricity generation from the first wall and divertor targets as well as the breeding blanket. The time taken to fit the new segments and the reliability of their operation should be such that they do not adversely affect the achievement of 5 $MW \cdot a/m^2$ during ten years of Stage-III operation in order that adequate blanket and other component reliability testing and materials testing can be undertaken in test modules.

The satisfactory completion of both small-scale materials tests in test channels and modules and large-scale engineering tests in the electricity-generating segments are essential stages in the development of the necessary technology for a DEMO reactor.

12. TEST MODULE INSTALLATION

The INTOR approach of test module installation was selected to minimize the effects of the test programme on the overall operation of INTOR. The complexity of past tokamak experiments has greatly limited the access to the reactor and constrained machine maintenance. To alleviate this problem, INTOR design has selected to radially extract shield/blanket sectors between magnets. This approach requires that open access be available around the reactor for transfer of NBIs, shield sectors, divertors and other components. The INTOR test module installation approach has been developed to be compatible with the reactor design. The major features of the approach are:

- Three dedicated test sectors
- Standardized test pockets (1 m^2)
- Horizontal test module installation
- Independent replacement of test modules
- Isolation of most testing from the plasma vacuum.

Dedicated test sectors are used to permit direct access to the test modules without having to remove major reactor components or having to use special equipment to work around other components for test module installation or removal. This approach permits the allocation of a test "zone" where equipment can be installed and modified without interfering with other normal reactor operations. As the INTOR programme evolves, it is likely that some small test ports will be required in other areas around the reactor. These ports will probably be used for diagnostics and specialized tests for which specific locations are required. Every effort should be made to minimize the number of these ports so that machine operation and maintenance are not adversely affected.

The test module size is standardized in order to permit the testing programme to incorporate changes more readily. Also, if a particular test indicates an anomaly, a parallel test of an identical module can be initiated without removing the initial test module, while observations continue on the original test module. On completion of any test, the next test module can be installed without having to wait for a particular test space to be cleared. The use of standardized test modules will result in a loss of packaging efficiency for some testing; however, the increase in overall efficiency of the test programme will offset this penalty. As a better overall definition of the tests is developed, it may become desirable to include a second, smaller, standard-size test module.

Horizontal test module installation was selected as the approach that could provide the largest test area without interfering with normal reactor operation or requiring removal of a reactor component such as a neutral beam system to gain access. Coolant tube routing to permit independent module replacement allows tested modules to be changed without disturbing other tests. Horizontal installation permits simultaneous test module replacement at the two adjacent test sectors by duplicate sets of maintenance equipment, whereas vertical installation would be restricted by the overhead crane. The total area available for testing in a single sector is approximately 22 m^2. The total test area is limited primarily by the necessary access for module removal and cooling. Currently, 6 m^2 of test area per sector is used; careful module design might permit using up to 12 m^2 per sector.

Both pocketed and unpocketed test ports are used for test operations. The two module installation methods are shown in Fig. XIV-4. Unpocketed test ports are used for tests of plasma surface effects, first-wall tests and plasma experimental tests; pocketed test ports are used for all other tests. The primary advantages of pocketed tests are that the module can be exchanged without a breach of vacuum and that a test-module failure will not affect the plasma chamber. The disadvantage of pocketed tests is that approximately 0.75 cm of additional metal is added in front of the test specimens, which reduces neutron effects in the test module.

Figure XIV-5 shows an isometric sketch of a dedicated test sector incorporating six test pockets of 1 m × 1 m in the outer wall. These test pockets provide access to an area of 1 m × 1 m at the first wall. If the test-module pockets are not used during parts of the INTOR programme, temporary shield plugs will be installed.

A plan view of the test locations is shown in Fig. XIV-6, indicating where the tests are performed. Sector 12 has been allocated for as yet undefined plasma

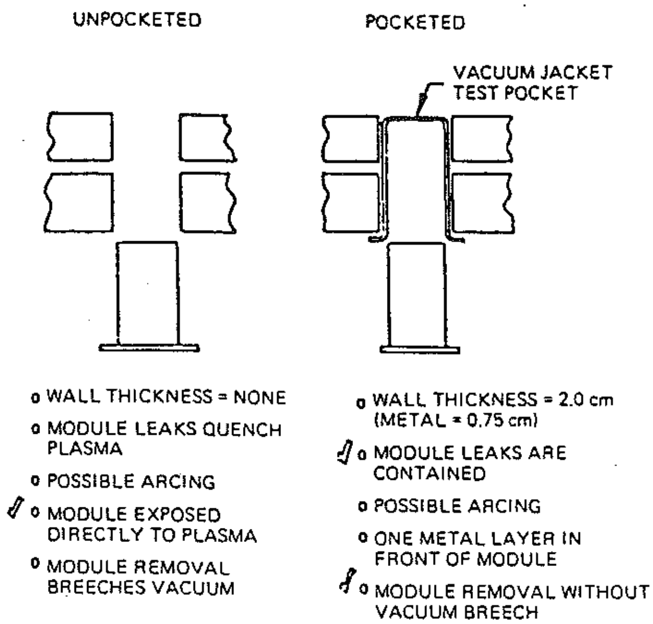

FIG.XIV-4. Test port design.

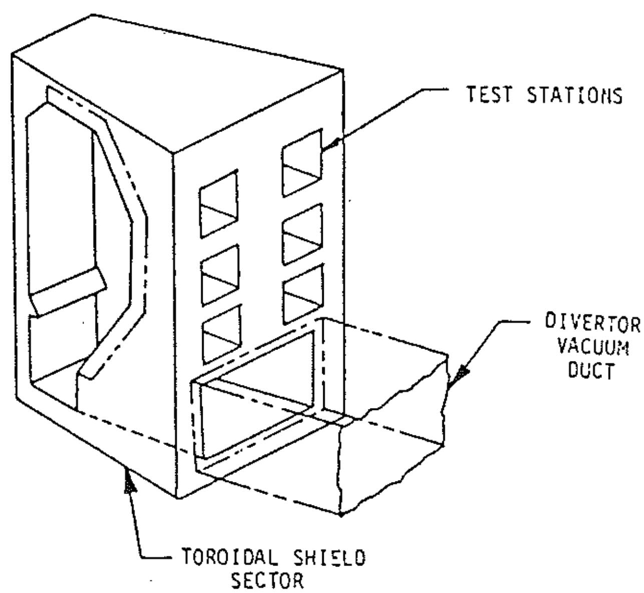

FIG.XIV-5. Dedicated test sector configuration.

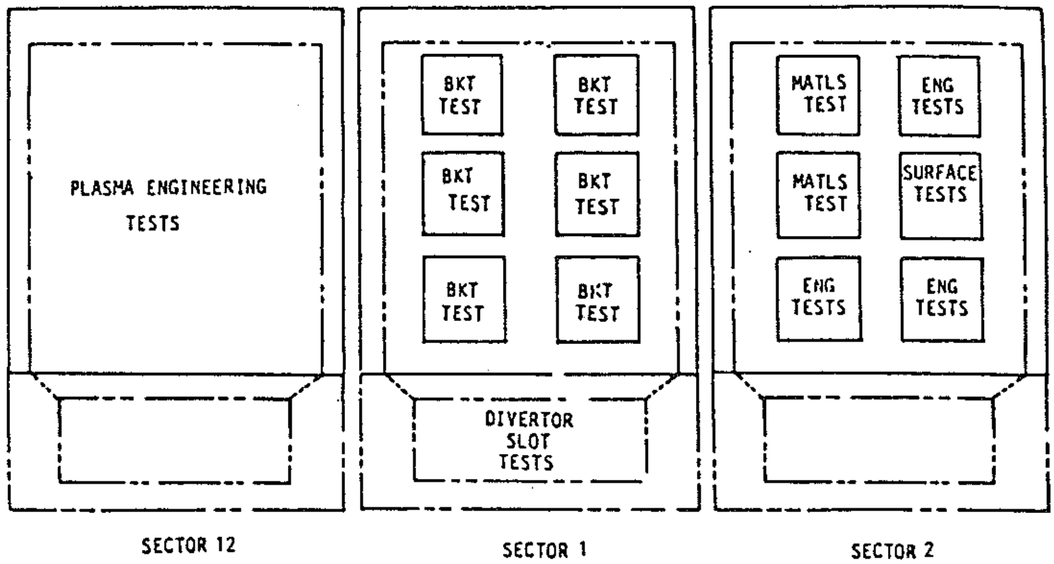

FIG.XIV-6. Plan view of test locations.

engineering tests. Six test pockets have been allocated in Sector 2 for first-wall/ blanket tests, four pockets being planned for long-term tests and two for short-term tests. Sector 1 includes all materials and engineering testing. The surface effects test and a bulk materials test are located in test pockets at the mid-plane, in close proximity to the plasma, permitting surface specimens to be installed normal to the plasma.

13. SUPPORT FACILITIES

13.1. Objectives and requirements for support facilities

Each test component inserted in the reactor will require support facilities whose size can be large. Therefore, scoping design studies are useful in assessing the impact of the test programme requirements on the INTOR reactor design. The support facilities can be located in the reactor building outside the magnets. However, in cases requiring large floor space, consideration should be given to locating the support facilities in the basement below the test area.

The test facility layout has been developed with the following aims:

(a) To permit access for independent removal and installation of test modules
(b) To provide clearance around the reactor for normal reactor maintenance equipment
(c) To permit rearrangement of the modules within the test sectors without major revision of the support system components.

TABLE XIV-10. HORIZONTAL VERSUS VERTICAL TEST MODULE INSTALLATION

Installation direction \ Test area	Two dedicated test sectors	Open-area installation
Horizontal	o Clear access o Test area = 43.4 m^2 o Use of sector handling approach	o Work around reactor components or their removal is required o Test area > 100 m^2 o Special equipment or significant reactor disassembly is required
Vertical	o Clear access o Test area < 10 m^2 (assumes that inner coil structure blocks 50%) o Existing crane o Inaccessible seals o Module removal is required for sector removal	o Clear access o Test area < 37 m^2 (assumes that inner coil structure blocks 50%) o Existing crane o Inaccessible seals o Module removal is required for sector removal

13.2. Access of support facilities to the tokamak

The complexity of past tokamak experiments has greatly limited the access to the reactor and constrained machine maintenance. Considerable effort was made to enhance INTOR maintainability and it has been decided to radially extract shield/blanket sectors between magnets. A successful implementation of this approach requires that open access be available around the reactor and that ready access to the test modules be provided.

A comparison of horizontal versus vertical installation of test modules (see Table XIV-10) has indicated that horizontal installation has significant advantages.

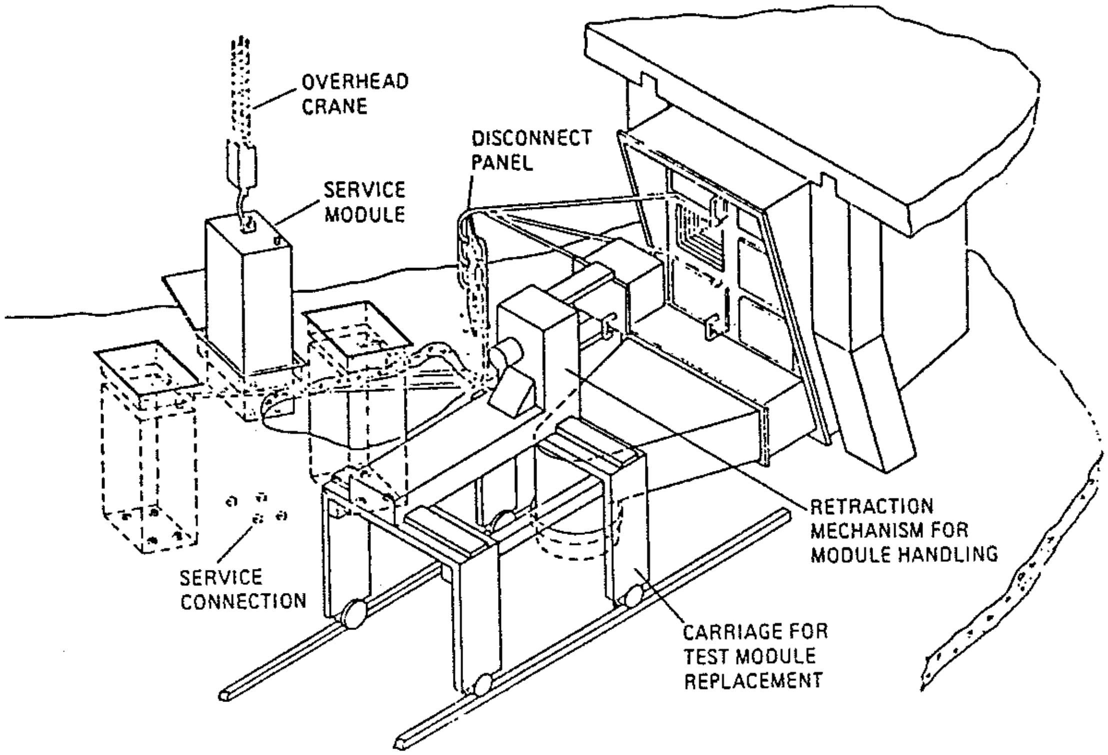

FIG.XIV-7. General arrangement of support facility.

The comparison was made for conditions where two dedicated sectors are used and the test modules are installed at any "open area" around the reactor. Open-area horizontal installation would require that the modules be installed above the neutral beams and fuelling assemblies in areas where access is quite restricted.

Vertical installation of test modules between TF coils would require a modification of the common cryostat. A 30 cm vacuum jacket was assumed to be added around the 1.25-m-wide magnet. A seal at the vacuum boundary would have limited access for inspection and repair.

Access from the top of the reactor to the first wall is also restricted by structure between the TF coils. When dedicated sectors with horizontal access are used, a total of about 40 m^2 accessible area is available. Both horizontal and vertical installation of test modules will result in a loss of efficiency in packaging; perhaps less than one half of the accessible area will be available for actual testing [4].

13.3. Layout of support facilities

The general arrangement of support facilities is shown in Fig.XIV-7. Test modules can be installed in and removed from the sector while it is located in the reactor, with the divertor vacuum duct in place. The coolant lines and instrumentation leads will be routed in such a manner as to provide direct access to each test module without disturbing the adjacent ones so that independent module replacement can be accomplished. Hard line connections are routed through a 180° bend to a disconnect panel in order to permit module removal without removing the lines. Flexible lines and instrumentation leads will be routed to the module from below, if possible, so that after being disconnected they are out of the way. The vertical clearance between modules is ~0.4 m and the horizontal clearance is ~1 m.

Test equipment must be located away from normal reactor maintenance equipment, preferably in the reactor building basement, so as not to interfere with normal reactor maintenance operations. It is anticipated that some instrumentation and test equipment such as optical devices will eventually have to be located close to the reactor. The individual coolant service modules will be located in the reactor basement near the test sector so that other instrumentation can be placed nearer to the reactor. Each service module is expected to be ~2 m × ~2 m × ~7 m and to be installed under a removable floor panel. Space is provided for 12 service modules (one for each test module) which can be removed by the overhead crane. Cooling water and power inputs will be provided for each service module at its base. Quick-disconnect fittings will be used to aid maintenance.

Remote maintenance facilities will be used for test module handling (see Section IV-6).

14. POST-IRRADIATION EXAMINATION (PIE) FACILITY

Although some useful information can be obtained from test instruments during irradiation, most of the technologically important data on materials properties can be obtained only through post-irradiation examination (PIE) of the experiments.

The examination facilities (for the early PIE stages at least) must be located at the INTOR site in order to efficiently service the reactor, to minimize the time for change-out of modules and samples, and to reduce time delays and costs for shipping large irradiated components. Later post-irradiation examinations, especially those requiring very sophisticated and expensive equipment (such as electron microprobes or microscopes) or highly trained operators, could be done away from the INTOR site if appropriate facilities exist which can accommodate the workload.

14.1. Experiments requiring PIE

The three main types of experiments requiring post-irradiation examination are the breeder blanket modules, bulk-material capsules and surface-effect channels.

14.1.1. Breeder blanket modules

Breeder blanket modules consist of solid or liquid breeder materials, cooling lines, multiplier materials, walls subject to plasma interaction, and other components subject to changes or damage during reactor service. They are the largest components (about 1 m^3 volume) to be routinely received and handled in the PIE facilities. Large shielding casks for transporting these modules, and large ports for access into the receiving, handling, disassembly and assembly (RHDA) cell are required. The RHDA cell walls must be made of high-density concrete, about 1.3 m thick, to reduce radiation from the modules to 0.1 mR/h at the operating face of the cell.

Sophisticated, highly flexible, remotely operated cutting equipment will be required for disassembling the blanket modules.

14.1.2. Bulk-material capsules and samples

This programme will require a shielded-cask system capable of removing and transferring 60 capsules from the reactor to the PIE facilities within 30 days or less. The capsules will be transferred to the RHDA cell for disassembly, and for cleaning, identification and transfer of the samples to the test facility at the INTOR site or elsewhere. Disassembly and reassembly of the capsules and handling of the samples will require capabilities for machining, for removing and handling liquid metals, and for storing, packaging and shipping of the irradiated specimens. Good viewing systems, including magnified TV or periscope imaging, will be essential.

14.1.3. Surface-effect (first-wall) channel tests

The requirements for receiving and disassembling surface-effect tests are similar to those for the bulk-material samples, but the channels are larger than the capsules and contain only one sample each. Examinations of surface-effect samples will be primarily visual and metallographic (including swelling and erosion determination). Some mechanical tests may also be required; in this case, the test specimens would be machined away from the bulk sample or from the channel itself.

14.2. Functional divisions of the PIE facility

PIE support for INTOR test operations will be divided and assigned to individual divisions, according to the specialities and functions involved. The facilities for the divisions may be adjacent to each other, or separated, as best suits the overall design. Some of the facilities should be located at the INTOR site; others could be located elsewhere, if necessary.

14.2.1. PIE divisions that should be located at the INTOR site

The following PIE divisions must, or should if possible, be located at the INTOR site:

(a) Receipt, handling, disassembly and assembly (RHDA) cell

The RHDA cell must be able to unload the transfer casks and to perform initial disassembly operations, final reassembly operations, and all necessary support functions (sample cleaning and identification, packaging, storing, waste disposal and transfer). The cell might also be designed to provide remote maintenance functions for the reactor, but it would be difficult to avoid compromising either the reactor-support functions or the experiment examinations.

The RHDA cell will be large (about 20 m by 13 m) and open, and will provide about eight to ten work-stations. (If reactor maintenance functions are included, the cell might be up to twice as large.) The largest irradiated test components will be handled and dismantled in this cell, so it requires the best shielding of all the INTOR PIE facilities (about 1.3 m of high-density concrete, or equivalent).

(b) Equipment decontamination cell

This small (one or two work-stations), air-atmosphere decontamination cell will provide essential support for the examination cells of the PIE facility. It will receive contaminated equipment from the other cells and, using dry and wet methods as required, reduce contamination so that the equipment can be repaired by contact maintenance in the connected equipment repair area.

(c) Contaminated equipment repair area

This facility will be designed and equipped specifically for repairing contaminated equipment from the PIE cells. It will be connected directly to the decontamination cell so that equipment may be transferred easily.

(d) Cell equipment mock-up area

The special requirements of hot-cell service make it essential that equipment be mocked up, operated and modified as necessary before it is installed in the cell.

(e) Analytical radiochemistry cell

INTOR reactor operations and tests will generate many radioactive samples that must be chemically analysed. The analyses will include dosimetry measurements; determination of material purity, gas composition and breeder-material burn-up; and isotopic identification of contaminants.

14.2.2. PIE divisions that can be remote from the INTOR site

Although close proximity would be advantageous, the following PIE facilities need not be located at the INTOR site: material test cell, metallography cell, and radioactive sample test laboratory. If adequately staffed to handle the INTOR workload, existing facilities might be used to save costs. Any off-site facilities to be used for PIE should be identified, and binding commitments should be concluded as early as possible.

14.3. Equipment

The equipment needed for the INTOR PIE facilities is classified and briefly described below. Each item will be specially designed or modified for remote operation and maintenance, and must be thoroughly checked in mock-up operations before being used.

14.3.1. Shielded transfer casks

Shielded casks will be required for transferring components between the reactor and the RHDA cell. The cask shielding must be sufficient to limit surface radiation levels to 10 mR/h or below. Separate casks may be necessary for handling of modules, materials-test capsules and surface-effect channels.

14.3.2. General-purpose equipment

General-purpose equipment in the PIE cells will include cranes (up to 5 t capacity), electromechanical manipulators, master-slave manipulators, lights, viewing systems (periscopes and TV), repair hoists, transfer locks, pneumatic transfer tubes, welding equipment, decontamination equipment, and other items.

14.3.3. Special-purpose equipment

Special-purpose equipment in the PIE cells will include disassembly and machining devices, materials-test machines, measuring machines, gamma-detection equipment, metallography equipment, special test equipment (for eddy-current, ultrasonic, and acoustic emission testing), and miscellaneous handling tools and fixtures.

15. OPERATIONAL REQUIREMENTS

The impact of the test programme on INTOR operation can be classified into two types:

(a) The impact of the testing requirements on the selection of operating and design parameters for INTOR. These parameters must be selected to ensure that meaningful tests relevant to the fusion demonstration plant (DEMO) can be performed satisfactorily.

(b) The impact of tests on INTOR operation; for example, some tests (e.g. plasma engineering tests) might alter the plasma characteristics, while others may cause an interruption of INTOR operation, either for test module replacement or because of an unexpected failure of a test module.

These two types are discussed in the following subsections.

15.1. Impact of tests on the selection of operating and design parameters

A major objective of INTOR is to provide test information relevant to the construction of DEMO. Therefore, the selection of some key INTOR parameters was influenced by the test requirements.

(a) Wall load and power density

The power density in the blanket greatly influences the temperature distribution, thermal stresses, radiation damage rate, and other key operating parameters. Possible failure of the blanket is strongly dependent on the power density. For example, a change of a factor of two in power density and stresses can lead to a change of more than an order of magnitude in the fatigue life. Therefore, it is important to select for INTOR a high power density, as close as possible to that of DEMO. However, there are limitations, due to the achievable power density of the plasma and to the desire to minimize the capital cost of the device. The power density of the plasma is limited by technological constraints on the achievable magnetic field and by plasma physics constraints on the maximum stable beta. Careful study of these considerations led to INTOR specifications of a neutron wall load of 1.3 MW/m^2. Since the neutron wall load for DEMO is expected to be ~2–3 MW/m^2, the power density in the INTOR blanket is high enough to provide meaningful test information. However, blanket testing in INTOR will still fall short of complete verification of DEMO blanket performance.

(b) Plasma burn length

One of the important parameters that affect material performance under irradiation is the ratio of the fluence to the number of cycles in a given period. This ratio is governed by the neutron wall load (neutron flux) and the plasma burn length. Ideally, the plasma burn length in INTOR should be shorter than that of DEMO by only a factor of 2 or 3 (same ratio as the neutron wall load). Since DEMO is projected to have a long plasma burn, possibly approaching steady state, the plasma burn length in INTOR should be maximized. A 200 s plasma burn length was specified as the base design for INTOR during Stages II and III.

(c) INTOR lifetime

Long-term demonstration tests of blanket modules, of other reactor components and of materials require irradiation to high fluences. The integrated neutron wall load for DEMO has been projected to be ~10 MW·a/m². Although it is very desirable to achieve such levels in INTOR testing, the requirements imposed on the lifetime and reliability of basic INTOR components are severe. Therefore, INTOR was designed for operation of up to ~5 MW·a/m² during Stage III. Thus, an extrapolation of a factor of two for fluence-dependent tests from INTOR to DEMO will be required. This is considered to be acceptable; it permits data for sufficiently high fluence to be obtained in the fusion environment to calibrate irradiation data for fission reactors.

(d) Availability factor

An availability factor of ~50% is a goal for INTOR during the dedicated testing period of Stage III in order to achieve 5 MW·a/m² in 10 years of operation so that data can be provided for DEMO on a time scale that is compatible with DEMO construction in the period from 2010 to 2020.

(e) Minimum duty cycle

Tests such as those for thermomechanical response and tritium recovery are sensitive to the duty cycle. A duty cycle of 50% has been defined as a minimum. A higher duty cycle is desirable to maximize the test information obtained in a given period.

(f) Continuous operation

Some tests require a minimum period of continuous operation. For example, starting from room temperature, the temperature profile during each cycle reaches equilibrium after approximately 10 cycles. Therefore, tests at equilibrium conditions will require more than 50 consecutive cycles. Tests for tritium recovery at equilibrium conditions may require about 1 week to 1 month of continuous operation.

(g) Test duration and replacement times

Estimates of test duration must be based on a definition of the tests and the maintenance equipment. For many of the tests (thermohydraulics) to be

TABLE XIV-11. SOME OPERATIONAL REQUIREMENTS OF INTOR

Test	Minimum operating time (back-to-back cycles)	Change-out frequency/ test duration
Blanket thermohydraulics	> 50 cycles	monthly
Tritium recovery	> 1 month	monthly
Bulk materials	not specified	0.7, 2.4, 4.0, 6.6, 14 years
Surface materials	not specified	monthly
Nuclear testing	not specified	2 months
Electricity production	~ 1 week	6 months
Demonstration breeding blanket sector	> 1 month	6 months

performed in the reactor, only a few hours will be required to collect all the necessary data; however, it is assumed that these tests will be left in the reactor until the next scheduled outage or forced shut-down. No test change-outs are planned more frequently than once per month.

Preliminary estimates of change-out times indicate at least one week, considering that any replacement will require a radiation cool-down time of 24 hours. Start-up and verification tests are also assumed to take 24 hours. This leaves 3–5 days for change-out of the surface test specimens and blanket modules, a procedure which will require highly sophisticated maintenance equipment. Electricity production and plasma engineering tests in a special sector are likely to require a change-out of the complete sector and are estimated to take approximately 1 month, assuming that a significant effort will be undertaken to develop the maintenance equipment. Figure XIV-1 indicates the testing period required for various types of tests.

(h) Other parameters

The selection of several other INTOR parameters was influenced by the test programme requirements as discussed in various chapters of this report. Table XIV-11 gives a summary of some operational requirements of INTOR.

15.2. Impact of tests on INTOR operation

Once INTOR is operating, some tests will significantly affect INTOR operation; others will have a minimal impact.

The tests with significant operational impact will generally require that the machine be operated in a particular manner or will affect the plasma burn characteristics, whereas the tests without significant impact are generally modular and can be installed in the reactor, tested, and left in place until change-out is convenient. Failure testing of these modules is not planned; hence, reactor shut-down as a result of these tests should be infrequent. Generally, reactor shut-down for test module replacement will not be required more frequently than once per month.

15.2.1. Tests with severe operational constraints

These tests require frequent start-up, shut-down, or altered plasma operation.

(a) Plasma experiments

Plasma experiments are planned for the full three years of Stage-I operation. During the first two years of INTOR operation the experiments will involve exploratory tests to permit mapping of stable plasma ignition regimes. These tests will lead to numerous shut-downs and disruptions as the bounds of operation are explored and will probably result in several openings of the plasma chamber to inspect first-wall components. The third year of Stage-I operation will be dedicated to long-pulse experiments in order to ensure that the machine can be consistently operated with long plasma burn pulses during Stages II and III where fluence accumulation is needed. A 25% availability goal has been set for the third year of operation.

(b) Plasma engineering

Plasma engineering experiments, such as tests of hardware for impurity control and RF heating, have not been defined to date, but a one-twelfth sector of the first wall and shield has been allocated for these tests. It is unlikely that tests requiring major configurational changes will be allowed in INTOR, otherwise the time lost would greatly hamper other testing. Most of the plasma engineering tests are scheduled for Stage I since they considerably affect plasma operation, which would be inconsistent with engineering (e.g. blanket) tests.

(c) Nuclear tests

Nuclear tests are required for: (1) characterization of basic machine performance; (2) proof testing of tritium-breeding blanket concepts; and (3) engineering experiments, providing radiation mapping, streaming and activation information. These tests require the machine to be operated in different modes to attain adequate data. During Stage I, low-power tests are required before substantial background radiation builds up to permit the use of more accurate active measuring techniques. Early neutron and gamma-ray spectroscopy tests will be performed with full-length plasma burns but with a reduction in plasma intensity of five orders of magnitude. This will require that the plasma be driven with neutral beams. Nuclear tests can be accomplished within about 1 month if 10% of the machine time is dedicated to them. The tests that follow will provide spectral characteristics, using short plasma burns at moderate plasma intensities. Only a few burn cycles and frequent removal of passive monitoring devices will be required.

Engineering experiments will generally require full-power operation of the reactor for examination of neutron streaming, nuclear heating, and activation rates. Activation rates will be monitored throughout reactor operation.

(d) Surface materials tests

Surface materials tests have to be closely co-ordinated with plasma experiments because of plasma-wall interaction. The total amount of testing of materials different from those used in the basic reactor will be severely limited by how tolerant the plasma is to contamination by test specimens. Additionally, it is desirable to remove some surface material specimens frequently (for example, once every two days). Special-purpose maintenance equipment is needed for this and vacuum locks must be used to avoid breaking the plasma vacuum, but reactor shut-down will still be necessary. Significant testing of surface effects during Stage I, when frequent shut-downs are expected, could alleviate this problem if the times for surface effect testing are controlled by physics testing.

15.2.2. Tests with minor operational constraints

The blanket, bulk materials, electricity production and reactor surveillance tests have greatly influenced the definition of the INTOR basic operation schedule. However, these tests will not adversely affect normal operation, since frequent removal is not required and they are designed to be left in the machine until the reactor is shut down for periodic test-module change-out. Furthermore, none of these tests interacts with the plasma to any extent that would alter the plasma operation characteristics.

(a) Blanket tests

Six test pockets are allocated to demonstration, tritium recovery and specialized engineering tests. Some tritium recovery and specialized engineering tests will be performed in two test pockets during the latter portions of Stage I and during Stages II and III. These tests will generally require 1/2 to 2 years of test time and will be exchanged when the reactor is shut down for maintenance. Blanket module demonstration tests will be installed in four test pockets at the start of Stage II and will remain in place through Stage III to permit accumulation of fluence and operation time. Periodic removal of dosimetry and breeding capsules will also be required.

(b) Bulk materials tests

Bulk materials tests will be installed in two test pockets at the start of Stage II. The basic test modules will contain capsules of test specimens which will be exchanged periodically. Exchanges are planned at about 0.5, 2.3, 6, 11.2 and 12 years after the start of Stage-II operations.

REFERENCES TO CHAPTER XIV

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