SPECIAL TOPIC

INTERNATIONAL TOKAMAK REACTOR – PHASE I

Executive Summary of the IAEA Workshop, 1981

INTOR Group*

ABSTRACT. A conceptual design for a next-generation experimental tokamak fusion reactor, INTOR, Phase I, is summarized. The objectives and role of INTOR and the technical aspects of the design are discussed, as well as specific R and D required to confirm the design and the test programme that would be carried out.

I. INTRODUCTION

The International Tokamak Reactor (INTOR) Workshop is a collaborative effort among Euratom, Japan, the USA, and the USSR. It is conducted under the auspices of the International Atomic Energy Agency (IAEA), in terms of reference defined by the International Fusion Research Council (IFRC), an advisory body to the Director General of the IAEA. The broad objectives of the INTOR activity, as set forth by the IFRC, are to draw upon the capability that exists world-wide: (1) to identify the objectives and characteristics of the next major experiment (beyond the upcoming generation of large tokamaks) in the world tokamak programme; (2) to assess the technical data base that will exist to support the construction of such a device for operation in the early 1990s; (3) to define such an experiment through the development of a conceptual design; (4) to carry out a detailed design of the experiment; and, finally (5) to construct and operate the device on an international basis.

The INTOR activity is being carried out in phases. At the end of each phase, the participating governments review the progress of the activity and decide upon the objectives of the next phase.

The Zero Phase of the INTOR Workshop, which was conducted during 1979, addressed the first two objectives cited previously. Each of the four partners was represented by four participants who met periodically in Workshop sessions at IAEA Headquarters in Vienna to define the tasks of the Workshop, to review and discuss critically the contributions of the four partners, and to prepare the report of the Workshop. The bulk

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of the work was carried out by experts working under the guidance of the Workshop participants in their home institutions to perform the tasks that had been defined at Workshop sessions. This home country effort involved more than 100 of the leading magnetic-fusion scientists and engineers (about 15–20 man-years of effort) from each of the four partners. The participants met in Vienna four times, for a total time of 10 weeks, to define, review and discuss this work.

The broad tasks of the Zero-Phase INTOR Workshop were to define the objectives and physical characteristics of the next major experiment (after TFTR, JET, JT-60, T-15), in the world-wide tokamak programme and to assess the technical feasibility of constructing this experiment to operate in about 1990. Detailed assessments of the plasma physics and technology bases for such an INTOR experiment were developed, and physical characteristics were identified which were consistent with this technical basis and with the general objectives of the INTOR device as they evolved in this process.

Each partner submitted detailed contributions to the Zero-Phase Workshop, which were subsequently published [1–4]. These contributions underwent extensive discussions at the Workshop sessions and formed the basis for the report of the Zero-Phase Workshop [5]. This report, which represents a technical consensus of the world-wide magnetic-fusion community, concludes that the operation by the early 1990s of an ignited, deuterium-tritium burning tokamak experiment that could serve as an engineering test facility is technically feasible, provided that the supporting research and development activity is expanded immediately, as discussed in the report. This broad international consensus on the readiness of magnetic fusion to take such a major step is in itself an important milestone.

As a result of this positive conclusion, the INTOR Workshop was extended into Phase One, the Definition Phase, in early 1980, on the basis of the IFRC review and recommendation to the IAEA. The objective of the Phase One Workshop was to develop a conceptual design of the INTOR experiment.

The Phase-I INTOR conceptual design has been carried out by teams working in the home countries (20–40 man-years of effort by each partner). The starting point for the conceptual design effort was the set of reference parameters suggested by the Zero-Phase Workshop. Senior representatives (six to eight from each partner) of these design teams have met periodically at Workshop sessions in Vienna (for a total of about 13 weeks during Phase One) to define the tasks of the home design teams, to review the ongoing design and to take decisions on the evolving design. The decisions taken at each Workshop session were then incorporated into each partner's design activity, so that the four design contributions progressively converged towards a single design, at an increasingly greater degree of detail, during the course of the conceptual design activity.

The conceptual design contributions to the Phase One INTOR Workshop have been published [6–9]. These contributions form the basis for the INTOR conceptual design, which is summarized in this paper. A full report will be issued by the IAEA in early 1982.

The INTOR Workshop has played a major role in identifying and focusing the attention of the world fusion community upon the principal problems that must be addressed before the detailed design and construction of the next major experiment in the tokamak programme can be undertaken. The Workshop has also made a major contribution to developing a consensus on the most likely solutions to these problems. An important new task assigned to the Workshop in 1981 is the definition of specific R and D that could resolve the major uncertainties that have been identified in the Phase One INTOR conceptual design.

The International Fusion Research Council (IFRC) of the IAEA, which supervises the INTOR Workshop, has recommended that the Workshop be extended for the period July 1981 through June 1983.

II. SUMMARY

1. ROLE OF INTOR IN THE FUSION PROGRAMME

INTOR is viewed as the major experiment in the tokamak programme between the next generation of large tokamaks (TFTR, JET, JT-60, T-15) and the generation of demonstration reactors (DEMOs). The DEMOs will generally have the following objectives:

(a) Production of several hundred megawatts of electricity and achievement of net electrical power production;
(b) Production of tritium in the blanket, with a net breeding ratio greater than unity;
(c) Demonstration of the development and integration of full-scale components which can be extrapolated to a commercial reactor;
(d) Demonstration of component and system reliability, availability and life-time at a level that would be acceptable for a commercial reactor;

(e) Demonstration of safe and environmentally acceptable fusion reactor operation that would satisfy the requirements for a commercial reactor; and
(f) Demonstration of commercial feasibility (although the DEMO would not need to be itself economically competitive).

The role of INTOR in the fusion programme can be defined upon identifying the physics and technology prerequisites for the design and construction of the DEMOs. Then those prerequisites which can best be satisfied by INTOR and those for which complementary physics experiments and technology test facilities are needed can be distinguished.

The broad, general prerequisites for the design and construction of DEMOs are:

(a) Development of an adequate plasma physics and engineering data base for prediction of the performance of the DEMOs;
(b) Demonstration of the plasma physics performance required for the DEMOs;
(c) Development of fusion reactor components;
(d) Testing of component integration into an overall fusion reactor system;
(e) Testing of fusion reactor maintainability;
(f) Testing of component and overall reactor system reliability, at least to some significant fraction of the availability and design life-time of the DEMOs;
(g) Testing of electricity and tritium production by fusion; and
(h) Testing of the safety and environmental characteristics of a fusion reactor.

An extensive plasma physics experimental and theoretical programme will support the design and construction of INTOR and will supplement INTOR in providing the physics basis for the design and construction of DEMOs. In this context, INTOR is viewed as the maximum reasonable physics step beyond the next generation of large tokamaks towards a tokamak DEMO and is intended to demonstrate the achievement of most of the plasma conditions that will be required for tokamak DEMOs. Primary physics objectives of INTOR then are to investigate the operation of an ignited, D-T plasma and to achieve long, controlled, reproducible burn with optimized plasma parameters. Achievement of these objectives requires satisfactory impurity control, power and particle balance control, and profile control for parameter optimization. A closely related objective is the achievement of high-duty ($\geqslant 70\%$) cycle operation. INTOR may also be used to perform certain plasma physics experiments not directly related to learning how to operate INTOR, but such experiments should be carried out in other plasma physics devices if possible.

An extensive technology and component development and testing programme will be required in the development of fusion power reactors to the demonstration reactor stage. This programme will both support INTOR in providing the basis for its design and construction, and supplement INTOR in providing the basis for the design and construction of the DEMOs.

In general, it is anticipated that a thorough screening of candidate materials and component design concepts will be carried out in test facilities and that, before the final design and construction of INTOR, components will be developed and tested under conditions that at least partially simulate a fusion reactor environment. INTOR will then serve principally to:

(a) Test the compatibility of components within an integrated reactor system;
(b) Test the remote maintainability of a fusion reactor system;
(c) Test components and materials in a fusion reactor environment;
(d) Test the reliability of components under sustained operation in a fusion reactor environment i.e. to some significant fraction of the component design life-time against the limiting phenomenon (e.g. neutron damage, fatigue);
(e) Irradiate materials samples to moderate fluences in a fusion neutron spectrum;
(f) Test the production of electricity and tritium in a fusion reactor; and
(g) Test the safety and environmental acceptability of a fusion reactor.

Much will be learned in carrying out these investigations that will be utilized to improve the design of components and the overall reactor system for the DEMOs. It is then the role of the DEMOs to provide convincing demonstrations with full-size, fully developed components that can readily be extrapolated to commercial reactors.

Other magnetic-confinement concepts besides the tokamak are being developed. There is a good chance that one or more of these concepts will be developed to the commercial stage, and there is even a possibility that some other concept will supplant the tokamak as the front-runner before the DEMO stage. Thus, it is important that INTOR serve also to test technology that is required for other magnetic-fusion concepts. Fortunately, the technologies required for the principal magnetic-confinement concepts are, to a high extent, common to all systems.

2. INTOR OBJECTIVES

The objectives of INTOR follow from the foregoing considerations of its role in the fusion programme and from an assessment of the technical basis which could exist within the next several years for its design. (An assessment of this technical basis and an identification of required additional R&D was made during the Zero-Phase INTOR Workshop [5].)

The programmatic objectives for INTOR are:

(a) INTOR should be the maximum reasonable step beyond the next generation of large tokamaks (TFTR, JET, JT-60, T-15) in the world fusion programme.

(b) INTOR should demonstrate the plasma performance required for the tokamak DEMOs.

(c) INTOR should test the development and integration into a reactor system of those technologies required for the DEMOs.

(d) INTOR should serve as a test facility for the blanket, tritium production, materials and plasma engineering technology development.

(e) INTOR should test fusion reactor component reliability.

(f) INTOR should test the maintainability of a fusion reactor.

(g) INTOR should test the factors affecting the reliability, safety and environmental acceptability of a fusion reactor.

The technical objectives of INTOR have been developed to support the achievement of the programmatic objectives, while being consistent with the anticipated technical basis [5] for the design and

TABLE I. INTOR TECHNICAL OBJECTIVES

A.	**Reactor-relevant mode of operation**
(1)	Ignited D-T plasma
(2)	Controlled burn pulse of $\geqslant$ 100 s
(3)	Reactor-level particle and heat fluxes ($P_n \geqslant 1$ MW·m^{-2})
(4)	Optimized plasma performance
(5)	Duty cycle $\geqslant$ 70%
(6)	Availability 25–50%
B.	**Reactor-relevant technologies**
(1)	Superconducting toroidal and poloidal coils
(2)	Plasma composition control (e.g. divertor)
(3)	Plasma power balance control
(4)	Plasma heating and fuelling
(5)	Blanket heat removal and tritium production
(6)	Tritium fuel cycle
(7)	Remote maintenance
(8)	Vacuum
(9)	Fusion power cycle
C.	**Engineering test facility**
(1)	Testing of tritium breeding and extraction
(2)	Testing of advanced blanket concepts
(3)	Materials testing
(4)	Plasma engineering testing
(5)	Electricity production ~ 5–10 MW(e)
(6)	Fluence ~ 5 MW a·m^{-2} during Stage III for component reliability and materials irradiation testing

TABLE II. STAGED OPERATION SCHEDULE

Stage	Number of years	Emphasis	Availability	Annual 14-MeV neutron fluence (MW·a·m^{-2})[b]	Annual tritium consumption (kg)
IA	1	Hydrogen plasma operation Engineering check-out	10%	–	–
IB	2	D-T plasma operation	15%	0.16	3.6
II	4	Engineering testing	25%	0.31	6.9
III	8	Upgraded engineering testing[a]	50%	0.62	13.8

[a] The objective is to achieve ~ 5 MW·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 representative.

[b] At the outboard location of the test modules.

construction of such an experiment to initially operate in the early 1990s. These technical objectives are given in Table I.

These objectives will be achieved at different stages of INTOR operation. The staged operation schedule proposed for INTOR is shown in Table II. Stage I will be devoted to learning how to operate with an optimized D-T plasma. Most of the technical objectives in categories A and B will be achieved during this first stage. Stage II will be devoted to flexible engineering testing, and many of the technical objectives in category C will be achieved during this second stage. The high availability operation and high fluence accumulation for component reliability and materials irradiation testing objectives will be achieved during Stage III.

3. DESIGN DESCRIPTION

3.1. Overview

Based on a realistic assessment [5] of the anticipated status of plasma physics research and technology development a few years hence, a conceptual design has been developed for an INTOR device which could fulfil the objectives listed in the previous section. Emphasis has been given to developing the design self-consistently and in sufficient detail in certain important areas so that the critical problems could be identified and resolved and so that the consequences of certain major design decisions could be investigated.

The major features for the INTOR conceptual design are specified in Table III, and a perspective view of the device is shown in Fig.1.

An analysis of the magnetics, MHD equilibrium and stability, energy transport, plasma heating and impurity control has been made to support the plasma physics parameters specified for INTOR. The INTOR plasma, operating with the indicated parameters, should achieve an ignited burn with an average thermonuclear power output of 620 MW(th). The plasma current, in excess of 6 MA, should adequately confine alpha particles. The value $\langle\beta\rangle = 5.6\%$ is somewhat greater than the theoretical limit, but experimental evidence that tokamaks can operate in excess of this theoretical limit supports this choice. A divertor is used to exhaust helium and to prevent heavy impurities from reaching the plasma in order to achieve the 200 s burn time, which was set at about one-fifth the theoretical magnetic surface diffusion time. Based upon the present best estimate of plasma

TABLE III. INTOR DESIGN SPECIFICATIONS

GEOMETRY	
Chamber major radius, R	5.2 m
Chamber volume	320 m^3
Chamber surface area	380 m^2
PLASMA	
Plasma radius, a	1.2 m
Plasma elongation, K	1.6
Plasma aspect ratio, A	4.4
Burn average beta, $\langle\beta\rangle$	5.6%
Poloidal beta, β_I	2.6
Average ion temperature, $\langle T_i\rangle$	10 keV
Average ion density, $\langle n_i\rangle$	1.4×10^{20} m^{-3}
Energy confinement time, τ_E	1.4 s
Plasma current, I_p	6.4 MA
Field on chamber axis, B_T	5.5 T
Safety factor (separatrix), q_I	2.1
Thermonuclear power, P_{th}	620 MW
Neutron wall load, P_n	1.3 MW·m^{-2}
OPERATION	
Burn time, Stage I/Stages II and III	100/200 s
Duty cycle, Stage I/Stages II and III	70/80%
Number pulses life-time	7×10^5
Maximum availability goal	50%
HEATING - NEUTRAL BEAM	
Number of injectors (active/spare)	4/1
Beam power	75 MW
Beam energy	175 keV
Pulse-length capability	10 s
FUELLING	
Method	pellet injection and gas puffing
IMPURITY CONTROL	
Method	single-null poloidal divertor
Collector	W armour on SS or Cu heat sink
Power ot divertor	80 MW
FIRST WALL	
Power to first wall (excluding neutrons)	44 MW
Outboard: material	D_2O-cooled SS 316
Outboard: thickness	11.7 mm
Inboard: material	H_2O-cooled SS 316
Inboard: thickness	13.5 mm
Life-time	15 a (full)

TABLE III. (cont.)

BREEDING BLANKET	
Material	H_2O, SS 316, Li_2SiO_3 Pb, C
Breeder temperature	400–600°C
Thickness	0.5 m
Location	outboard and top
Breeding ratio	0.65
Tritium extraction	continuous He purge
TRITIUM FUEL SYSTEM	
Tritium flow rate	64 $g \cdot h^{-1}$
Annual tritium consumption at 25% availability	7 $kg \cdot a^{-1}$
Isotopic enrichment	cryogenic distillation
TRITIUM INVENTORY	
Breeding blanket	0.5–1.0 kg
Storage	2.3 kg
Plasma reprocessing system	0.2 kg
Pumps, fuelling and other systems	0.4 kg
TORUS VACUUM SYSTEM	
Initial base pressure	10^{-7} torr
Pre-shot base pressure	3×10^{-5} torr
Pumps	compound cryopumps
Pumping	through divertor chamber
TOROIDAL FIELD COILS	
Number	12
Bore	7.7 m × 10.7 m
Conductor	Nb_3Sn, NbTi
Stabilizer	Cu
Maximum field	~ 11 T
POLOIDAL FIELD COILS	
Total flux	110 V·s
Location	external to TF coils
Conductor	NbTi
Maximum allowable field	8 T
BREAKDOWN COILS	
Breakdown voltage	70 V
Location	solenoidal bore
Conductor	cryoresistive Cu
POWER SUPPLIES	
Stationary loads	241 MW
Pulsed energy storage	22.5 GJ

TABLE III. (cont.)

MECHANICAL CONFIGURATION

Twelve blanket sectors assembled with straight-line horizontal motion through windows between TF coils
Semi-permanent inboard, upper and lower shield forming primary vacuum boundary on inner surface
Final closure of primary vacuum boundary on outer boundary of removable torus sectors
Test modules inserted horizontally at mid-plane
All superconducting coils in a common cryostat
Dedicated sectors: 5 NBI
2 fuelling
3 testing
2 I & C

SHIELDING	
Inboard (non-breeding blanket and shield)	0.80 m[a]
Outboard (breeding blanket and shield)	1.65 m[b]
Neutral beam drift tube	1.0 m
NBI box	0.5–0.75

[a] Includes 3 cm gap.
[b] Includes 10 cm gap.

energy transport losses, the predicted alpha-heating power exceeds that required for ignition by a factor of about two. The 75 MW of neutral beam heating power allows the plasma to be heated to ignition for energy losses up to about twice as large as the present best estimate for heat conduction. This provides a margin for coping with radiation losses and existing uncertainties.

A single-null poloidal divertor, with the chamber at the bottom, has been chosen for impurity control. Analyses indicate that it is possible to magnetically form the divertor channels and to control the separatrix motion to within several centimetres with coils external to the toroidal field coils. A relatively short channel length is adequate because of the high-density mode of divertor operation. Two divertor collector plate designs have been developed. In one, tungsten is brazed to a copper heat sink. In the other, tungsten tiles are mechanically attached to a stainless-steel heat sink. The tungsten and copper would need to be replaced every few years; the stainless steel should last the design life-time of INTOR. The analysis in support of the single-null divertor included self-consistent treatments of the magnetics for separatrix control and

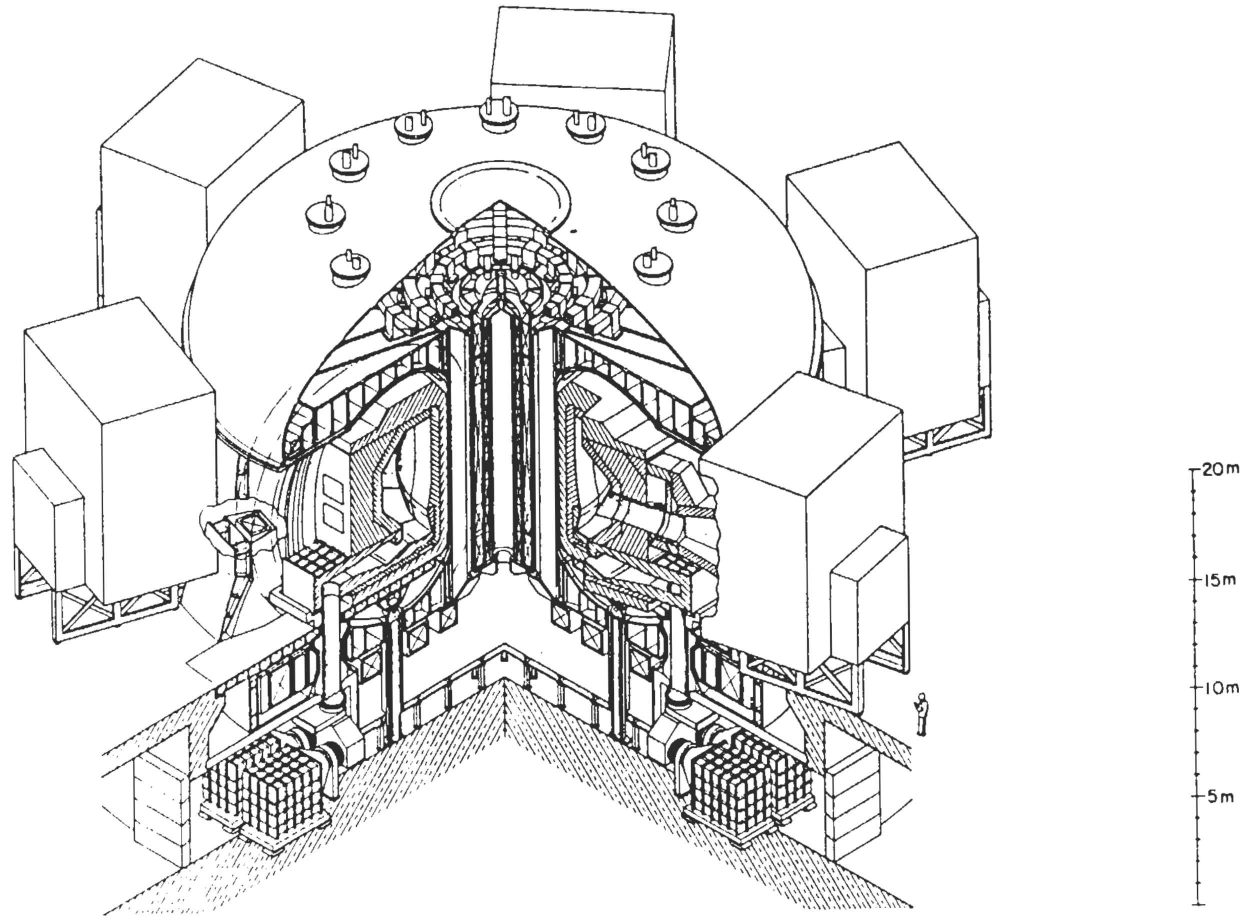


FIG.1. Perspective view of INTOR.

divertor channel formation, the plasma physics of the divertor channel and scrape-off region, the nuclear design of the divertor collector plate, and the engineering design of a maintainable divertor.

The mechanical configuration design was driven from the outset by the requirement to provide maximum access to facilitate maintenance and assembly/disassembly. A semi-permanent inboard, upper and lower shield forms the primary vacuum boundary. Twelve torus sectors fit within this semi-permanent shield. These torus sectors are partially (outboard and upper) tritium-producing blanket and partially (inboard and lower) heat-removal shield. The final closure of the vacuum boundary on the outboard is at the outer boundary of the blanket, inside of the outboard bulk shield. Once the outboard bulk shield is removed and the vacuum boundary is cut, each torus sector can be withdrawn horizontally with straight-line motion through a 'window' between adjacent toroidal field coils. The divertor channel is broken up into twelve modules which are removable with straight-line horizontal motion between the toroidal field coils. The single-null divertor was chosen over the double-null divertor in order to achieve this more simply maintainable mechanical configuration.

Semi-permanent, superconducting toroidal and poloidal field coils will be enclosed in a common, semi-permanent cryostat, thus completely separating the cold and warm structures. All poloidal field coils will be external to the toroidal field coils and superconducting, except for a set of cryoresistive coils internal to the Ohmic heating solenoid which provide the voltage pulse for plasma breakdown. Both forced-flow and pool-boiling conductor designs have been developed for the toroidal and poloidal field coils, and in addition a superfluid pool-boiling conductor design using NbTi has been developed for the toroidal-field coils. Each of these conductor concepts is under active development, and a final decision can await results from the development programmes.

The rather demanding structural requirements for the toroidal field coils are met by a combination of design strategies. Coil wedging, intercoil support structure and a bucking cylinder will be used to handle in-plane and centring forces. Gussets, intercoil support structure, a ring girder, the bucking cylinder and shear ties will be used to handle out-of-plane forces and the overturning moment. A built-up laminated structure will be used. A major accomplishment of the INTOR design effort has been to develop a credible structural design for a high-field pulsed tokamak.

Extensive analysis supports the design of the first wall, blanket and shield. A water-cooled, stainless-steel first wall with a panel-type construction is specified. This first wall is expected to last the full life-time of the device, provided that the melt layer which is predicted to form on the inboard section during a plasma disruption is stable. A tritium-producing blanket will be installed from the outset of operation in order to reduce the operational cost. A solid breeder (Li_2SiO_3) blanket that covers the out-board and upper surfaces of the plasma chamber can produce more than 60% of the tritium consumed in INTOR. Shielding for the torus and major penetrations has been determined on the basis of several one-dimensional and extensive three-dimensional radiation transport calculations. Adequate shielding is available for component protection and to allow access 24 hours after shut-down.

The availability goal for INTOR is 50% during the last stage of operation. Reliability analyses based upon component reliability estimates provided by the component developers indicate that achievement of this goal will require increased emphasis on component reliability in the component development programmes. Extrapolation of present reliability data leads to availability estimates of ~30–40%, depending upon the degree of redundancy.

It should be noted that the purpose of this report is to define the INTOR concept. Specifications for the different systems were naturally evolved in a parallel and iterative fashion, with the result that not all systems were specified to the same level of consistency and detail. The present work suffices to define the concept, and further work to achieve consistency on a detailed level is a task for a later phase.

In retrospect, a few factors can be identified which had a major influence upon the INTOR design. It is useful to review those factors which most contributed to imposing demanding engineering design requirements. The emphasis on maintainability was the major factor in determining the mechanical configuration. The requirement of a relatively simple torus and divertor assembly/disassembly procedure led to somewhat larger toroidal field coils than otherwise would be necessary, imposed certain constraints upon the toroidal field coil structural support system and led to the choice of a single-null (rather than a double-null) poloidal divertor (which imposed additional requirements upon the poloidal field coil system). The requirement of maintainability led to a choice of an all-external poloidal field coil system. The all-external PF coils and the large TF coils necessitated by the accessibility requirement, together with the high toroidal field specified, combined to produce large overturning moments on the toroidal field coils and thus led to additional requirements for the structural support system. The requirement of personnel access for maintenance at the outer boundary of the reactor led to considerably more outboard shielding than would be necessary for component protection. The objective of achieving an appreciable fraction of the design life-time neutron fluence for first-wall and blanket components in a DEMO had a significant influence upon the specification of the neutron wall load and availability goal, upon the inboard shield thickness, and upon the design of the first-wall and magnet structural system with respect to fatigue and crack growth limits. Subsequent optimization of the INTOR design should begin from these considerations.

Specific aspects of the INTOR conceptual design are described in the following sections.

3.2. Physics basis

The physics basis of INTOR, described in detail in the INTOR Phase-Zero Report [5], has been reviewed and elaborated during Phase One, with emphasis on the interface problems between physics and technology. Major points in this context are plasma confinement, beta limits, disruption characteristics, ripple-induced transport, generation and control of plasma equilibria including the requirements on the poloidal field system, the physics of the plasma edge and divertor, as well as the particle and heat loads to the walls both in the main discharge chamber and in the divertor area, and the problems related to the plasma operation (current initiation, heating to ignition, control of burn temperature, shut-down and dwell).

As far as plasma confinement is concerned, recent results have confirmed that losses due to anomalous electron heat conduction are by about 50% weaker than anticipated in Phase Zero. This implies that the confinement potential of INTOR at its working point, using a simple Alcator-like scaling for extrapolation, is now estimated to be about twice as high as required for ignition. Using alternative scaling laws which provide a somewhat better description of the available confinement data, a similar or even better confinement potential is obtained for INTOR. There is, however, still considerable uncertainty in this extrapolation. The requirements for heating to ignition could be eased if indications of a favourable dependence of confinement on plasma temperature and on major radius were substantiated. But unfavourable tendencies cannot be excluded either, particularly when beta is increased. Furthermore, the above estimate of the INTOR confinement potential does not take into account radiation losses by plasma impurities.

Theoretical predictions for the limits on the plasma beta in INTOR now stand at about 3%. Such values of beta have, however, already been reached experimentally, even in tokamaks with circular cross-section, so that the value of beta of 5.6%, as specified for INTOR, should be possible if elongation and triangularity of the plasma cross-section have the anticipated beneficial effects. In any case, an experimental verification of this expectation will be possible in the near future. In this respect, it is important to note that beta limits in present experiments are soft, i.e. there is an increase of transport losses rather than a sudden loss of discharge stability.

A specification of disruptions in INTOR, consistent with present knowledge, was developed. Although additional work on disruptions has recently been performed, there is still little knowledge of the details of the phenomenon (power deposition profile, time scales) in present devices, and extrapolation to the INTOR working regime adds further uncertainty. The disruption frequency, in the best regimes of Ohmically heated tokamaks, now is below 1%, but in the presence of intense auxiliary heating this low value is not reached. For INTOR, disruption frequencies of 5×10^{-3} and 10^{-3} are specified for initial and later operation. A characteristic time of 20 ms is taken for energy deposition and current decay. The density limit of Ohmically heated discharges has been shown to rise with plasma cleanliness. When additional heating is applied, still larger densities can be achieved.

A refined theory of ripple-induced heat transport led to the conclusion that ripple losses in INTOR are much weaker and much less temperature-dependent than anticipated earlier. Estimates of the losses of energetic particles injected during beam heating indicate that these should remain within acceptable limits.

Considerable effort has gone into developing an INTOR divertor design with consistent engineering and physics design features. A plasma equilibrium in a single-null divertor configuration can be provided, at the working point of INTOR where the poloidal beta is anticipated to be 2.6, by a poloidal field system with coils outside the toroidal coils. The sum of the current moduli needed is about 100 MA-turns. The divertor configuration is rather open in this case. A preliminary analysis indicates that this configuration can also be maintained during plasma heating from the Ohmic regime to ignition. For this, the poloidal field coil system, however, must have an even higher total current capability. Control of the horizontal position of the plasma is anticipated not to be a problem. Vertical position and shape control, on the other hand, require further study. A passive loop system adjacent to the first wall is specified for this purpose.

The modelling of the plasma in the edge and divertor regions and its interaction with the walls were improved. At the same time, important new experimental results have been reported from tokamaks with divertors. It was concluded that INTOR should be operated with a high density ($\approx 5 \times 10^{19}\ m^{-3}$) and a low temperature (≈ 100 eV) at the edge, if one allows for strong re-cycling in the divertor. Under these conditions the pumping speed for providing the exhaust is expected to be about 2×10^5 litre$\cdot s^{-1}$. Furthermore, self-sputtering of divertor plate material is minimized, and the penetration length of neutrals into the scrape-off and divertor plasma is small (a few centimetres). For these reasons, an open divertor geometry is acceptable and it is expected that a cool protecting plasma layer

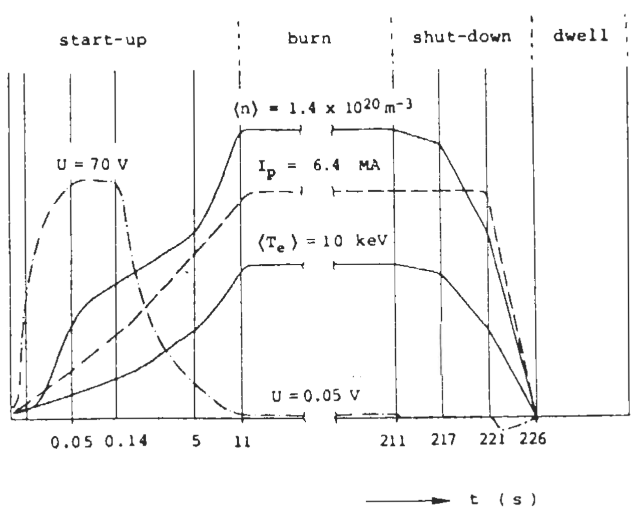


FIG.2. INTOR operation scenario.

will form in front of the first wall if the scrape-off layer width is 10 cm or larger. Although the space necessary for the scrape-off layer is less in a double-null divertor configuration, it was concluded that the related plasma-physics advantages are not strong enough to outweigh engineering arguments (see Section 3.3.5) for a single-null divertor (including its higher reactor relevance). The configuration adopted for INTOR appears capable of handling the required power load and exhausting the helium, and is expected to prevent the accumulation in the plasma of impurities eroded from the first wall and divertor collector plate, although this latter aspect has not yet been thoroughly analysed.

Most of conclusions drawn in Phase Zero concerning the operation scenario (shown in Fig.2) have been confirmed. New results on current initiation have allowed the maximum loop voltage to be lowered to 70 V. The provision of 5 to 10 MW of RF heating to assist current rise and to minimize volt-seconds needed in this phase is desirable, although it is not specified in the present design. A duration of the Ohmic heating phase of 5 s appears to be reasonable. Heating to ignition by 75 MW of 175 keV neutral beams continues to be the preferred option. RF heating methods (ion cyclotron, lower hybrid, and electron cyclotron heating), which are under rapid development, all appear consistent with the INTOR conceptual design and could be used if their physical and technological viability would be demonstrated. The length of the burn phase was increased to 200 s, which is still short (a fifth) of the global skin time in INTOR. For the control of the burn temperature, it has become less certain whether relying on enhanced ripple losses can provide a solution. On the other hand, evidence for the existence of a soft beta limit, where transport losses increase, has come from recent experiments. It is anticipated that a control scheme combining different mechanisms will be required. Further work is necessary to define a satisfactory solution. The duration of the shut-down phase was increased to 15 s in accordance with new modelling results. Furthermore, a preliminary analysis of the gas release rate from the wall during the dwell time between shots was performed, the conclusion being that 20 s appear to be sufficient for this phase.

In summary, the physics data base for INTOR has been considerably expanded during the last 18 months. The main points are: (1) a better analysis and new experiments have yielded an improvement, by about a factor of two, of plasma confinement in the density range relevant to INTOR; (2) more detailed theoretical and experimental investigations on beta

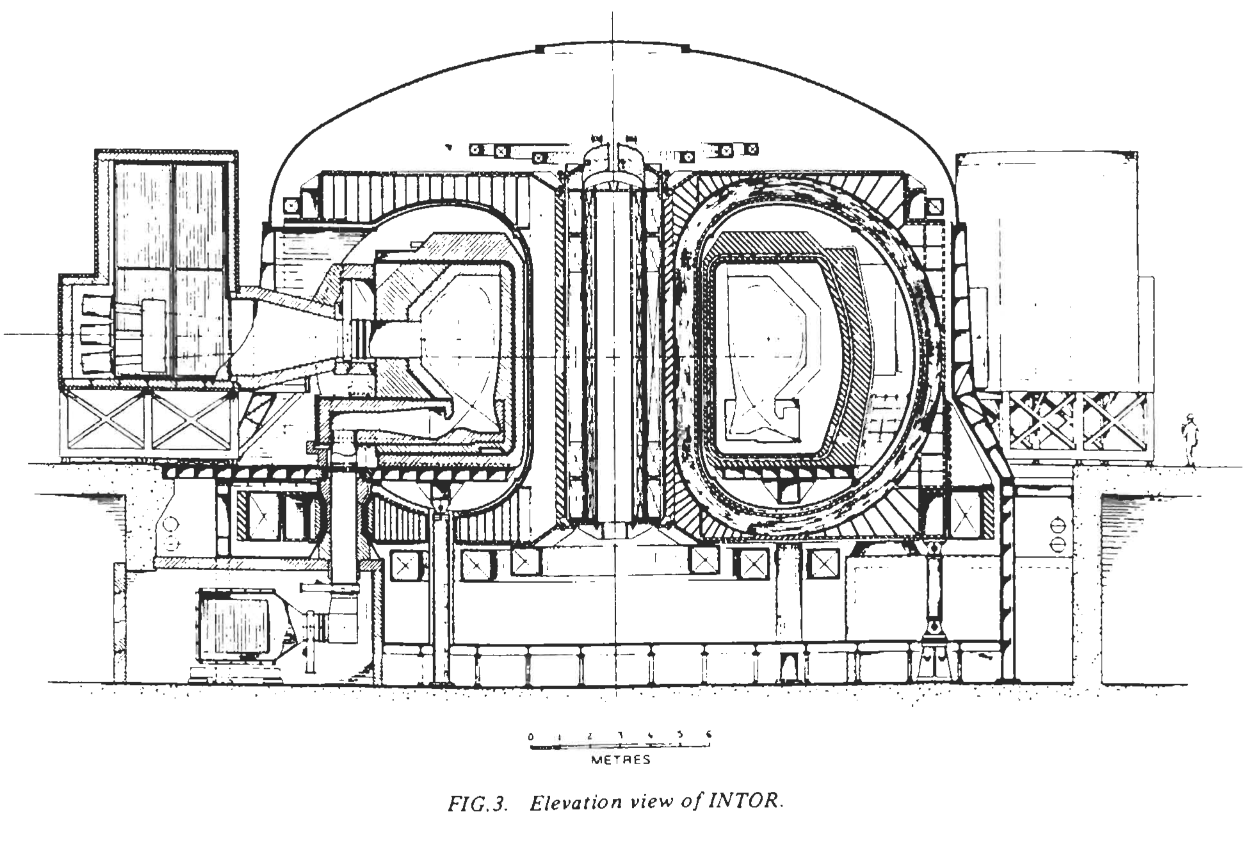


FIG.3. Elevation view of INTOR.

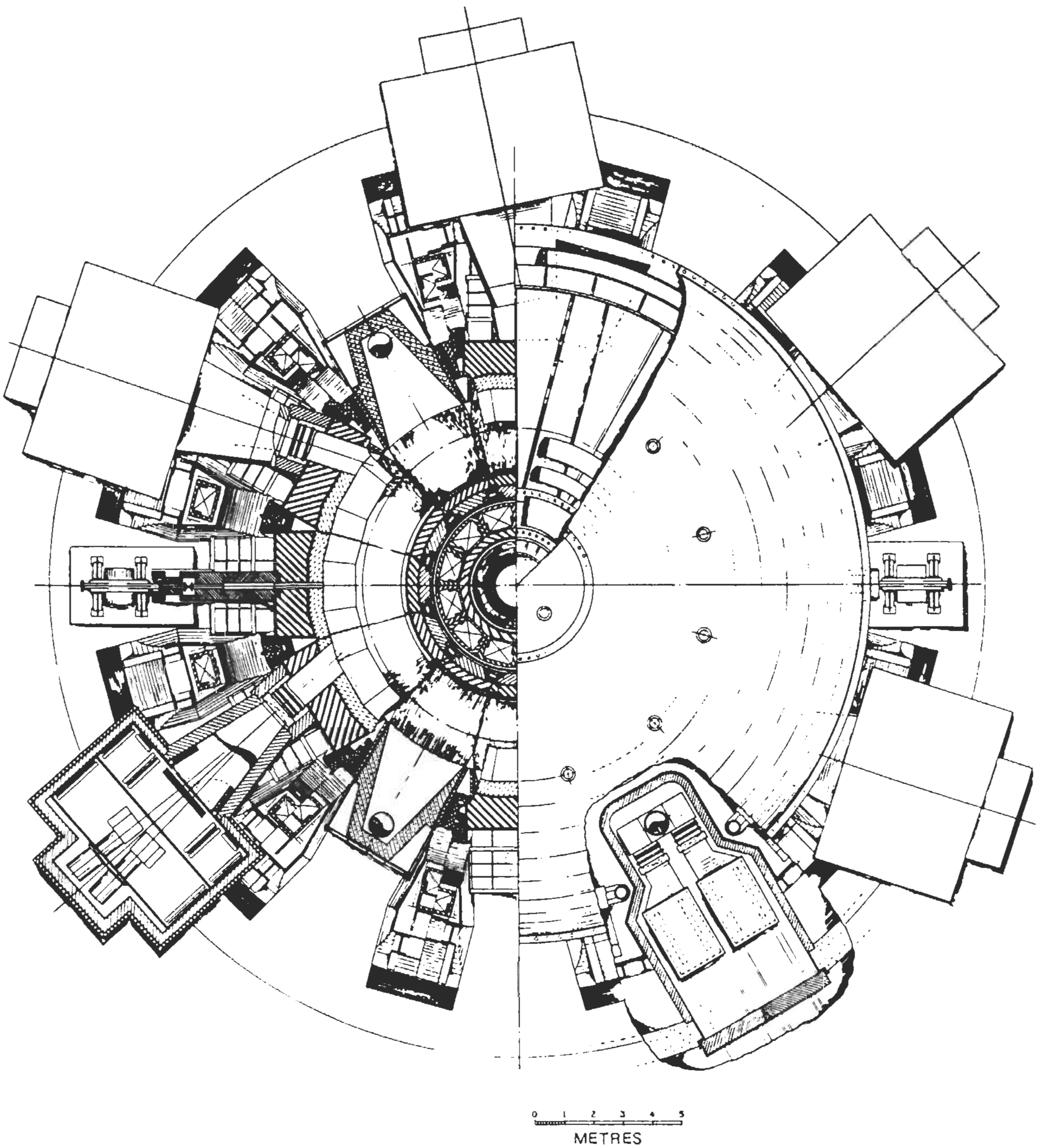


FIG.4. Plan view of INTOR.

limits have confirmed that reaching the beta required for INTOR may be possible; (3) an increase of the density limit in tokamak experiments, both with and without additional heating, supports the INTOR start-up procedure; and (4) a large body of new experimental and modelling results on the characteristics of the plasma edge and poloidal divertor operation has been obtained which has allowed the specification of an impurity control and helium exhaust concept for INTOR.

3.3. Mechanical configuration and maintenance

Activation of components by high-energy fusion neutrons, the presence of tritium and complex electromagnetic features of the tokamak device have the

TABLE IV. INTOR MAINTENANCE PHILOSOPHY

1. The tokamak will be designed from the outset to be maintained and repaired by the use of existing technology for remote maintenance equipment such as manipulators, viewing systems and transfer mechanisms.
2. Certain systems must be designed and developed with very high reliability so that failure will not be expected within the life-time of the device. Failure of these systems would require a major shut-down of the facility (six months to one year) for repair or replacement. Superconducting toroidal magnetic field (TF) and poloidal magnetic field (PF) coils, the inboard portion of the torus shield and several major support structures have been identified as systems of this type and designated as semi-permanent installations.
3. Sufficient radiation shielding will be provided in the torus and around penetrations to limit activation of components exposed to the reactor room. "Hands-on" maintenance will be considered for normal operations when the torus internals are not removed. 2.5 $\mathrm{mrem \cdot h^{-1}}$ will be specified as the maximum dose rate anywhere in the room after 24 h of shut-down.
4. All systems will be designed for fully remote maintenance to cover cases of emergency.

potential for seriously delaying maintenance and repair operations. Maintenance considerations were therefore established at the outset of the INTOR design study as the fundamental consideration in the development of the design configuration.

A basic maintenance philosophy was established for the conceptual design (Table IV) to allow maintenance requirements to affect the design configuration. Implementation of this philosophy has led to a modularized design concept, and designing to achieve the required access has had a significant impact on the design of the tokamak systems.

The main features of the INTOR design configuration. are summarized in the following sections. Elevation and plan views of the tokamak are given in Figs 3 and 4.

3.3.1. Toroidal magnetic field coil design – access requirements

The most significant configuration design feature is the access provided for torus maintenance. The twelve toroidal field (TF) coils have been sized with sufficient outside dimensions so that a complete torus sector, consisting of one twelfth of the total, can be withdrawn by a simple straight motion between the outer legs of the coils. These twelve torus sectors fit within a semi-permanent upper, inner and lower shield frame.

3.3.2. Poloidal magnetic field coil system

To simplify maintenance in INTOR, all of the poloidal field (PF) coils have been placed outside of the bore of the TF coils. The PF coils can, therefore, all be superconducting since mechanical joints are not required for assembly. All of the PF coils have been located above and below the TF opening where the torus sectors are removed. A small solenoidal, cryo-resistive coil is placed within the Ohmic heating solenoid to provide the breakdown voltage pulse.

3.3.3. Cryogenic vacuum topology

Since all of the PF coils external to the TF coil bore are superconducting, it was possible to design a single vacuum cryostat to contain all of the coils. The vessel includes individual enclosures for the outer TF coil legs as part of the common cryostat. With this feature, access to the torus is maintained without penetration of the cryogenic vacuum boundary. Another important feature of this design is that there is a complete separation of the cold and warm components, which eases the structural design requirements for thermal movements of the large structures.

3.3.4. Plasma chamber vacuum topology

The torus system, consisting of a first wall, blanket, shield and divertor collector, has been configured in two major parts: a semi-permanent shield and removable sectors (Fig.5). The components exposed to the most severe damage from particle and heat loads (first wall and blanket regions) have been combined into a sector which can be removed separately from the torus shielding. More importantly, the vacuum seal for this sector is entirely on the outside of the torus. The seal weld is on a rectangular flange easily accessible between the TF coil outer legs. The other portion of the torus, consisting of the structural frames and semi-permanent shield modules, forms the primary vacuum boundary and is not removed for normal planned maintenance procedures.

3.3.5. Single-null poloidal divertor

The divertor collector is the most severely damaged torus component, and its design must include

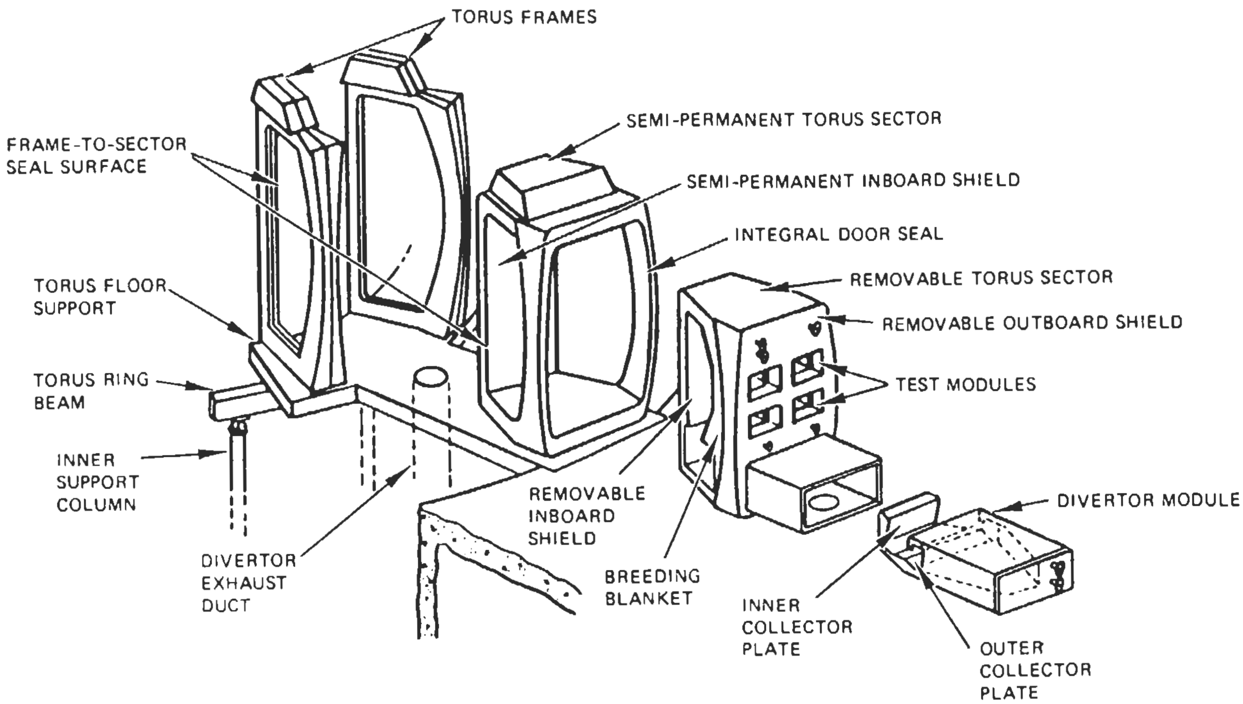


FIG.5. Divertor duct options.

provisions for frequent repair. Modular divertor sectors have been designed which can be removed in a similar way as the main torus sectors. A double-null divertor was found severely to limit access to the top of the torus. For this reason, a single-null concept has been selected in spite of the more difficult design problems associated with the asymmetric PF system and increased particle loadings.

In the single-null divertor system, the divertor chamber is located at the bottom of the toroidal plasma chamber. This results in a shift of the plasma upwards by 0.6 m relative to the TF coil horizontal centre line, in order to centre the assembly of toroidal chamber and divertor region inside the TF coil system and to facilitate its maintainability. The configuration of the divertor region is indicated in Fig.5. It consists of twelve divertor modules, each located inside a removable torus sector. Each divertor module is provided with an exhaust duct, from which it can be disconnected easily without dismantling the pumping system. The collector plates and the other divertor surfaces subject to high erosion are protected by tiles of refractory metal attached to the surface and are water-cooled. Each cassette unit, including its shielding plug, is extended outwards to the outer boundary of the torus system, where it can be disconnected easily from the outside.

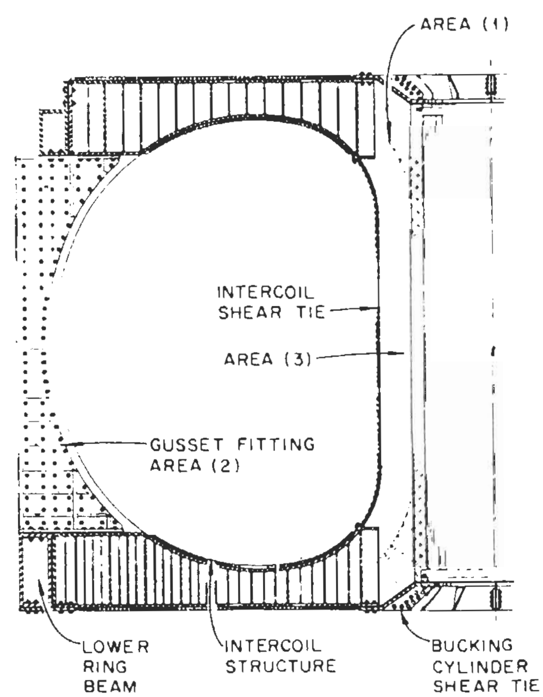


FIG.6. TF coil support structure – elevation view.

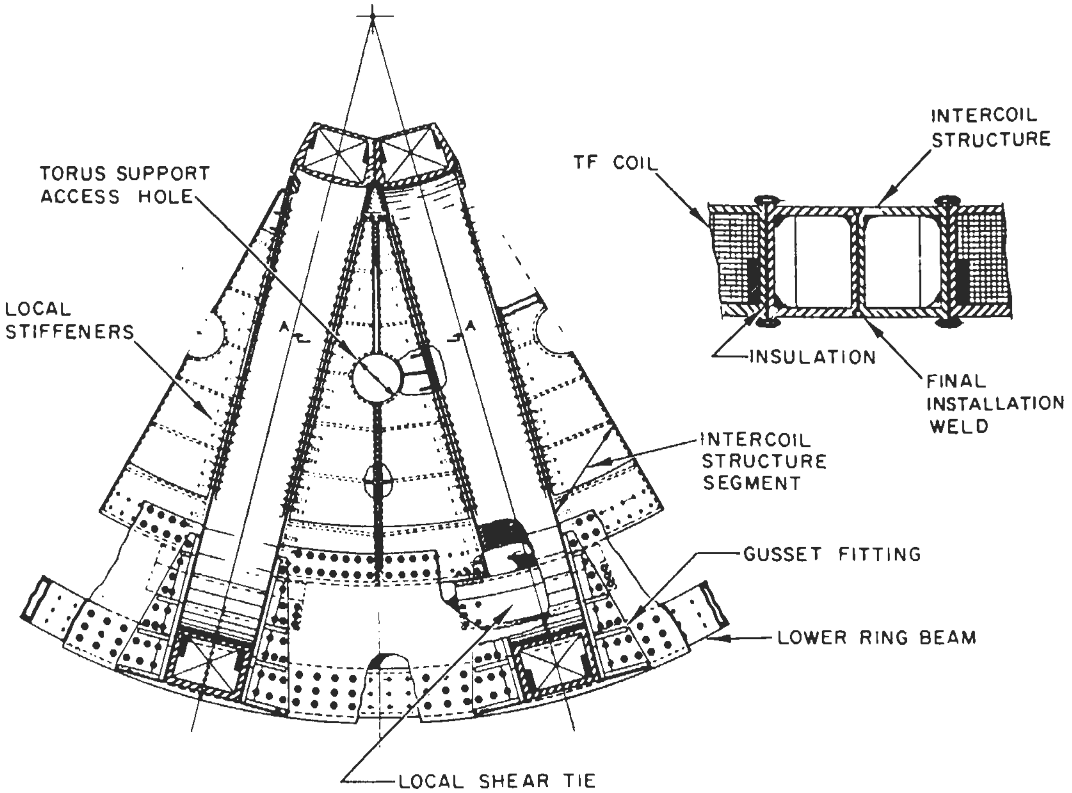


FIG. 7. TF coil support structure – plan view.

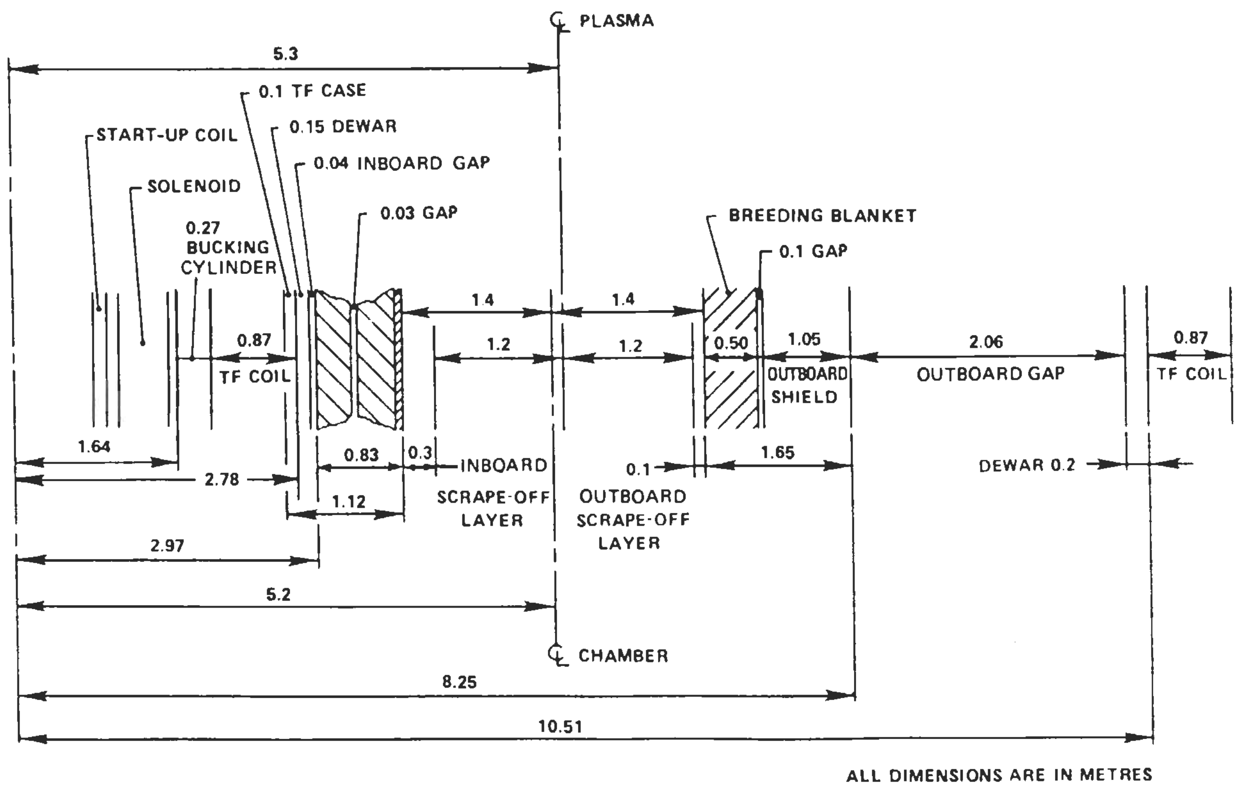
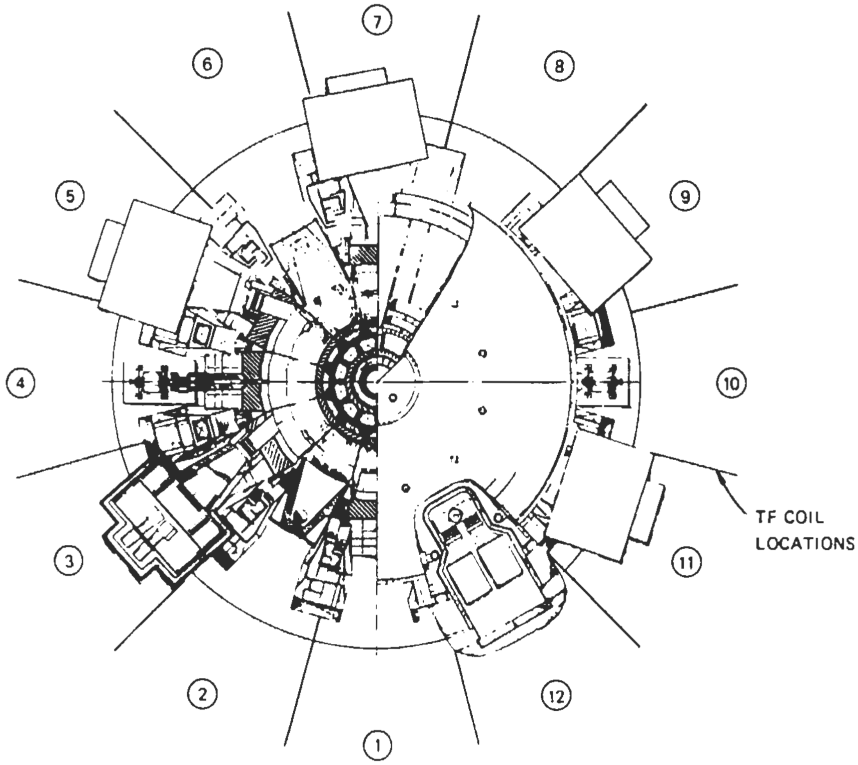


FIG. 8. Radial build of INTOR.



COMPONENT	SECTOR POSITION
NEUTRAL BEAM INJECTORS	3, 5, 7, 9, 11
FUELLING	4, 10
TESTING	1, 2, 12 (INCLUDING RF)
DIAGNOSTICS, INSTRUMENTATION AND CONTROL	6, 8

FIG.9. Dedicated torus sectors.

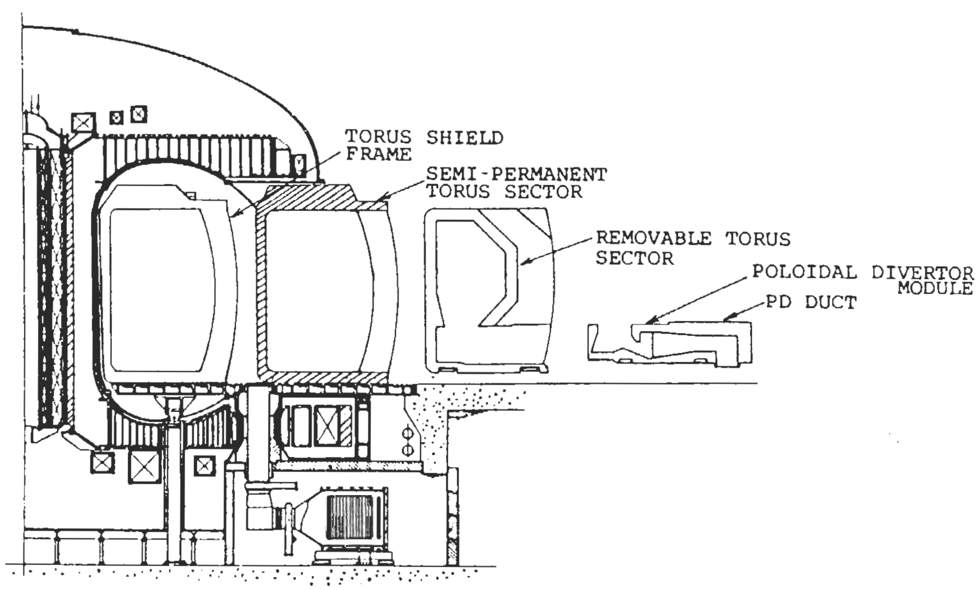


FIG.10. Torus assembly sequence.

3.3.6. Structural support system

A major design effort has gone into the structural configuration. The large TF coils combined with the all-external PF coils impose very large pulsed magnetic forces on the TF system. Thermal considerations require that all the structural support be provided at cryogenic temperatures. A structural configuration consisting of mechanically attached reinforcing members has been designed which maintains the access space for torus sector removal (Figs 6 and 7).

The structural design was verified by a three-dimensional finite-element model code. The model included a representation of all twelve TF coils and their support structure. Local stress analyses were performed to investigate details of the design, including the local plate bending.

Another feature of the structural configuration is the gravity support system. The magnet system support has been placed entirely at the outside of the machine to provide access to the bottom of the machine (see Fig.3)

3.3.7. Tokamak radial build

The major radial dimensions of INTOR are summarized in Fig.8. The radial build dimensions show space allocation for all components as well as the required gaps for assembly tolerance.

3.3.8. Dedicated torus sectors

The facilities layout is based upon the concept of dedicated bays – the region between adjacent toroidal field coils (Fig.9). Three bays are dedicated to testing – two for blanket and materials testing and one for plasma engineering hardware and diagnostics testing. Five bays are dedicated to the four active and one redundant neutral-beam injectors, and two bays are tentatively dedicated to instrumentation, diagnostics and control. This approach provides a straightforward interface between the tokamak device and the facility.

3.3.9. Assembly and maintenance

The INTOR device and facility were designed consistently with the assembly and maintenance philosophy. Figure 10 shows the torus assembly sequence.

Failure rate is another important factor in establishing the maintenance equipment requirements. The assumed frequency of repair of several major tokamak components is given in Table V.

Maintenance equipment requirements were also studied for general-purpose transport equipment, a general purpose manipulator, special-purpose equipment, and viewing systems.

TABLE V. FREQUENCY OF COMPONENT REPAIR

Type of repair	Frequency of repair (per year)	Components
Small scale	2–10	Divertor, NBI ion source
Medium scale	1	First wall, blanket (removable torus sector)
Large scale	0.1	Magnet, shield structure (semi-permanent torus sector)

TABLE VI. FEATURES OF TOROIDAL FIELD COIL SYSTEM

Number of coils	12
Field on plasma chamber axis	5.5 T
Maximum field	11 T
Coil bore size (helium vessel)	7.7 m × 10.7 m
Structural load paths	
(a) In-plane running load	Case, winding steel
(b) Centring load	Bucking cylinder, ISS[a]
(c) Out-of-plane running load	Gussets, ISS
(d) Overturning moment	Ring girder, gussets, bucking cylinder, ISS, shear ties
Maximum magnetic forces	
(a) Centring force	454 MN
(b) Overturning moment	237 MN·m
(c) In-plane running load	66 $MN \cdot m^{-1}$
(d) Out-of-plane running load	30 $MN \cdot m^{-1}$

[a] ISS = intercoil support structure.

TABLE VII. TOROIDAL FIELD CONDUCTOR OPTIONS

	Pool boiling at 4.2 K	Pool-boiling superfluid	Forced-flow alternative
Superconductor	NbTi/Nb_3Sn hybrid	NbTiTa	Nb_3Sn
Conductor	Flat, cabled		Rectangular, cabled, sheathed with steel
Winding	Spiral, pancake, co-wound with steel channel		Spiralled into grooves in plates
Support	Pancakes enclosed in He case to form coil		Plates bolted together to form coil
Cooling	Pool-boiling liquid He at 4.2 K	Superfluid He at 1.8 K	Forced-flow supercritical He

3.4. Magnetic and electrical systems

3.4.1. Toroidal field coil system

The INTOR TF coil configuration has been developed with sufficient flexibility to incorporate any one of the three major conductor cooling concepts currently under development world-wide. The basic TF systems requirements are listed in Table VI. The three conductor concepts, all of which are compatible with the overall TF coil configuration and method of support, are characterized in Table VII.

The three conductor cooling concepts are liquid-helium-bath-cooled at 4.2 K (option A); superfluid-liquid-helium-bath-cooled at 1.8 K (option B); and forced-flow-liquid-helium-cooled (option C). Option A is based on a mature technology of cooling for which there is considerable design, fabrication and operating experience; the disadvantage is that Nb_3Sn, which because of its relatively brittle nature is more difficult to design and fabricate, must be used as a superconducting material. Option B avoids the difficulties of fabrication with Nb_3Sn (at the lower cryogen temperature, NbTi may be used as the superconducting material), but at the expense of introducing a technology (superfluid helium) for which there is little experience. Option C has significant technical advantages over both bath-cooled options – better heat transfer from the conductor and easier to insulate – but this option also relies on technology (forced-flow cooling) for which there is little experience. The ultimate choice among the three options will be made in the future on the basis of results from development programmes.

Electromagnetic fields and forces have been calculated. Eddy current losses have been evaluated and their effect included in the cryogenic requirements. Winding packs have been configured and evaluated to verify adequate paths for transfer of loads to the supporting structure.

3.4.2. Poloidal field coil system

The INTOR PF system has been developed with the requirement that there be no coils interior to the bore of the TF coils. A system meeting this requirement has evolved with three coil sets:

(a) A superconducting solenoid located in the bore of the bucking cylinder, which provides the Ohmic heating function to induce and maintain the plasma current.

(b) A cryoresistive solenoid, co-axial with and located in the bore of the superconducting solenoid, which assists plasma breakdown and plasma current initiation.

(c) A set of eight superconducting ring coils located above and below the TF coils, which provide the equilibrium function.

The PF system is not symmetric about the mid-plane of the TF coil set, a consequence of the use of a single-null divertor. A summary of major design parameters for the PF coil sets is given in Table VIII.

The PF coil system has been analysed to determine the fields and forces. Losses have been calculated and included in the requirements for the cryogenic system. The winding packs have been configured to provide adequate system load paths to the supporting structure.

Two conductor concepts have been developed, both of which are compatible with the functional requirements and the space envelopes. One is a forced-flow-cooled concept, the other is a 4.2-K liquid-helium-bath-cooled concept. The ultimate choice between these options will be made in the future on the basis of results from development programmes.

3.4.3. Alternating-current power system

The AC power system provides the power needed for INTOR operations and has ample reserve for future

TABLE VIII. SUMMARY OF PARAMETERS FOR PF COILS

	Superconducting solenoid	Cryoresistive solenoid	Superconducting ring coils
Number of coils	1	1	8
Size	Mean diameter 2.87 m Length 11.45 m Radial build 0.22 m	Mean diameter 1.99 m Length 11.45 m Radial build 0.59 m	1.5 m × 1.5 m winding pack cross-section (typical) Diameter 24.2 m (largest ring coil)
Current (MA-turns) (end of burn pulse)	38	41	Total for eight coils 60 Largest coil 24

TABLE IX. MAJOR AC LOAD POWER ESTIMATE

Load description	Estimated power (MW)
MGF drive motors	120
All coolant pumps	20
Cryogenic system	41
Vacuum system	5
Auxiliary heating (steady state)	20
Tritium system	5
Burn power supplies	15
Facility power (HVAC, lighting, etc.)	15
TOTAL	241

growth. A heavy-duty utility line provides the power for the major loads estimated in Table IX. An independent light-duty utility line, having lower voltage, provides facility power so that essential power exists if the power to the heavy-duty line fails. Also, if the light-duty line fails, the heavy-duty line can provide the facility power. The AC power systems have to provide: ample steady-state power, make-up power for the motor-generator sets, facility power, limited back-up pulsed load capability, standby back-up power for essential loads, power to non-interruptible loads, and power for co-ordinated fault protection. Diesel generators are proposed to provide power to critical power loads if both power lines fail. They are connected to the two primary busses of the facility power system. Tie breakers connect the heavy-duty substation with the light-duty substation so that maintenance and equipment testing can be performed during a utility power failure.

3.4.4. TF coil power conversion and protection

The power conversion system consists of six low-voltage power supplies, two current-controlled voltage regulators, 36 large dump resistors, 12 ground resistors, 12 circuit interrupters, 12 transient voltage suppressors, and 12 coil monitoring instrumentation units. Six identical groups of components are connected in a series ring circuit to ensure that the same current flows through all coils under normal and most abnormal operating conditions.

Six identical reversible power supplies convert AC power from the regulator to DC power needed to charge the coils in about five hours and sustain them during normal operation. The power supplies can also be controlled to invert DC power to AC and to discharge the coils. A current controller adjusts the voltage regulators so that the AC voltage to the interleaved power supplies provides the desired TF coil current. The low-voltage power supplies are built with bridge rectifier power modules connected in parallel, with enough added capability so that they can provide the 15 kA with one module removed for servicing. The standardized modules reduce the initial cost of the power converters and minimize the inventory of spare parts.

The power supply inverters will be used to provide the slow discharge needed for protection under most abnormal conditions. Energy dump resistors and circuit interrupters are provided for initiating rapid discharge of the TF coils. The dump resistors are connected across the coils and across the circuit interrupters in order to limit the transient voltage across the coils and the interrupter that opens first.

3.4.5. PF coil power conversion and protection

Each PF coil has rectifier-inverter power supplies (converters) and low-voltage burn power supplies interleaved between coil sections.

Each power converter consists of standard modules connected in series and parallel to provide the required currents and voltages. The high-voltage converters provide power to charge and discharge the superconducting PF coils during the start-up and shut-down phases of the cycle. The burn power supplies provide the low sustaining voltage needed during the 200-s burn phase of the cycle. Each power converter requires a pair of high-current medium-voltage busses between the coil sections and the converters.

Standardized instrumentation packages constructed in modular units will be provided to monitor the superconducting coil for faults and to control the current flow in the PF coils. DC circuit interrupters provide the coil back-up protection if the inverters fail during operation. Coils are thereby isolated and connected to discharge resistors. A normal discharge rate is provided when the power converters are in operation. Slow-discharge back-up is not provided for internal coil shorts and inverter failures because these events are unlikely to occur simultaneously.

The Ohmic heating solenoidal coils require power converters having bidirectional current flow. Therefore, both forward and reverse rectifier-inverters are needed. During the burn phase, the high-voltage converters are bypassed to reduce the internal impedance and power losses. The forward rectifier-inverters are turned off and bypassed during the burn phase since they are not needed during that time.

3.4.6. Electrical energy storage system

The AC energy storage system reduces the peak power demand on the utility line. It receives power from the utility line during the long burn phase of the cycle and from the inductive loads of the PF converters during the discharge phase and stores this energy for the pulse loads of the subsequent cycle. The AC energy storage system also provides power distribution and fault protection for the reversible power converters and the auxiliary heating power supplies. Motor Generator Flywheel (MGF) sets are proposed for isolating the major pulsed loads from the utility line and supplying the required energy storage. The pulsed energy storage requirements that must be met by the MGF sets are given in Table X. Five MGF units, each having an energy storage of 4.5 GJ, are needed for the PF coil system and NBI system. Each generator has a peak load capacity of 800 MV·A for the five second peak power needed for start-up.

TABLE X. MGF ENERGY STORAGE REQUIREMENTS

A.	PF coil			
	(a)	Start-up		
		Energy from MGF units	20	GJ
		Peak load on MGF units	3.1	GW
			(3.9	GV·A)
	(b)	Shut-down		
		Energy to MGF units	12	GJ
		Peak input to MGF units	2.05	GW
			(2.56	GV·A)
B.	Neutral beam-lines			
	Start-up			
	Energy from MGF units		2.5	GJ
	Peak load on MGF units		250	MV·A
C.	Total requirements:			
	(a)	Start-up	22.5	GJ
			(3.9	GV·A)
	(b)	Shut-down	12	GJ
			(2.56	GV·A)

3.4.7. Cryostat

The cryostat for the magnet system has been designed with a single vacuum vessel encircling all the cold structure. Each coil will have its own helium vessel. A helium vapour cold shield, conforming to the inside of the vacuum vessel, has been incorporated to reduce radiation heat transfer to surfaces at liquid helium temperature. Concepts for joints with high electrical resistance have been proposed to reduce induced toroidal currents in the cryostat vacuum vessel, if necessary.

TABLE XI. CRYOGENIC SYSTEM – TOTAL LOAD

	Helium		Nitrogen
	Refrigeration (kW)	Liquefaction (l/h)	Refrigeration (kW)
Magnetic system	42.5	1800	1100
Neutral beam + RF	10.5	200	–
TOTAL	53	2000	1100

TABLE XII. NBI MAJOR CHARACTERISTICS

Source design		
(a)	Number per injector	6
(b)	Current	68 A
(c)	Energy	175 keV
(d)	Pulse length	10 s
Interface design		
(a)	Number of injectors	
	Operation	4
	Redundant	1
(b)	Drift duct	
	Height	1.2 m
	Width	1.0 m
	Length	330 cm
(c)	Power density average	1.56 $kW \cdot cm^{-2}$
(d)	Injection	
	Angle	16 and 22°
	Power per NBI	18.7 MW
(e)	Efficiency (overall NBI)	
	Full energy	0.22
	All energies	0.35
Cryopumps		
(a)	Area	
	Drift region	64 m^2
	Gas cell region	30 m^2
(b)	Configuration	Differential Liquid He shielded by LN chevrons
Ion deposition		
(a)	Type	Direct recovery on full energy
(b)	Capability	≥ 0.6 efficiency
Species mix		
(a)	Ion fraction (H^+, H_2^+, H_3^+)	0.8, 0.12, 0.08
(b)	Power fraction	0.62, 0.28, 0.10

3.4.8. Cryogenic system

The cryogenic system for INTOR has been based upon the forced-flow liquid-helium coil-cooling option for both the TF and the PF coils. The cryogenic system consists of two cooling circuits: a liquid-helium system, providing a temperature of 4–4.5 K, to maintain operating temperatures in the superconducting coils; and a liquid-nitrogen system, providing a temperature of 80-95 K, to cool the cryoresistive solenoid and the cold shield between the 4 and 300 K (ambient) surfaces. The liquid-helium system also provides cooling for the neutral-beam injection equipment.

The cold mass being cooled has the following major parameters:

Weight of structure cooled to liquid-helium temperature: 9000 tf
Weight of structure cooled to liquid-nitrogen temperature: 150 tf
Radiation heat transfer surface area: 3000 m^2
Cold-mass supports; length: 1 m; cross-section area: 2 m^2.

The helium system must have both a liquefaction and a refrigeration capability. The liquefaction is needed because power leads will be helium vapour cooled. Both systems are of forced-flow type.

A summary of the loads on the overall cryogenic system is given in Table XI. The total requirements are 53 kW of refrigeration at 4.2 K and 2000 $litre \cdot h^{-1}$ of liquefaction.

3.5. Heating and fuelling system

3.5.1. Neutral-beam injection system

An injected power of 75 MW is provided by four injectors. One additional injector provides redundancy. The injection angles of the six individual sources in an injector range between 16° and 22°. Each injector interfaces with the tokamak through a 1.0-m-wide by 1.2-m-high penetration fitted with a single absolute valve. One metre of shielding around this drift duct makes the total interface area 9.6 m^2 per injector. The injector's main assembly is housed inside a nuclear shield igloo which can be transported separately or with the igloo. The main NBI box measures 6 m wide by 8 m high by 4 m deep. Each injector weighs about 100 tonnes. As planned, the igloo is made to allow easy dismantling for removal of the NB injector to the hot cell or maintenance in situ. The weight of the igloo is ~350 tonnes. Power supplies and ancillary

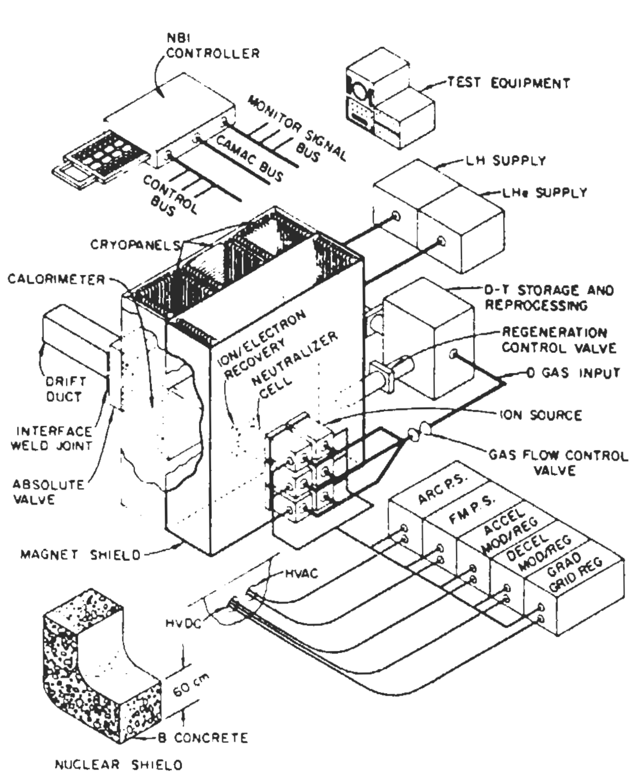


FIG.11. Neutral beam injection system.

equipment (such as cryogenic refrigeration, instrumentation and control processors, cooling, etc.) are located outside the reactor hall. A summary of the major characteristics of the NBI system is given in Table XII.

Figure 11 delineates the beam line components and their configuration. Note that the ion sources are arranged into two stacks of three. A common focal point for all source beams is located at the centre of the drift duct. The drift duct shape is optimized to this source configuration and beam cross-over to minimize the wall loading.

A gas efficiency of 50% is assumed for the ion source. A little over 80 torr·litre·s^{-1} is pumped in cold gas out of the six ion sources. With a chamber differential pumping approach, it takes 30 m^2 of cryopump area at 10^{-4} torr on the gas cell side of the baffle and 64 m^2 in the drift region at 4×10^{-6} torr to handle the 80 torr·litre·s^{-1} of cold gas. These are cryocondensation-type pumps protected by liquid nitrogen chevrons.

The drift duct has an absolute valve which is only open during beam operation. This reduces the possibility of tritium or ^{3}He backstreaming. A 15-cm thick ingot iron magnetic shield encases the gas cell chamber. This reduces the tokamak stray fields to <8 G in the vicinity of the ion source and gas cell.

Nuclear hardness is addressed through use of shielding and proper choice of material. The insulators in the source, direct-recovery and bending magnets, are either shielded or constructed of non-organic material.

3.5.2. Fuelling system

The fuelling system consists of a gas puffing system and a pellet injection system.

Both continuous and pulsed flow control of the selected gas are provided in the fuel gas flow control system proposed for INTOR. The selected gas can be predominantly deuterium, predominantly tritium, or a mixed gas species from the fuel recovery process system. Downstream pressure controllers regulate the pressure of the selected gas to the control valves. All or part of the twelve available gas puffing ports can be selected for operation. Both normal and automatic control modes are proposed. This system offers redundancy to improve reliability, and considerable flexibility in selecting and controlling fuel gas flow to the plasma chamber. The fuel gas tanks that contain tritium are shared with the fuel recovery system. They also service the pellet injector system described below.

There are two pellet fuel injectors located on opposite sides of the torus. Each injector can inject deuterium and tritium pellets concurrently. A mixed-species fuel may be substituted for the deuterium or tritium. Downstream pressure regulators control the flow of gas to the nitrogen-cooled chiller tanks located externally to the pellet injector. Remotely controlled selector valves determine the fuel gas admitted to the liquefiers. The liquefiers are located inside the insulated fuel injectors. The liquid fuel to the pellet former flows preferably by gravity feed rather than forced flow. The pellets are then admitted to the dual-feed pellet injector at a rate controlled by a process computer. The programmable computer uses electron density and neutron measurements to calculate the pellet injection rate of both channels. Either fuel injector can provide for satisfactory operation, although the plasma distribution in the toroidal direction may be more uniform with both injectors operating.

The design pellet injection velocity is 2×10^3 m·s^{-1}. Both pneumatic and centrifugal pellet injectors are being developed and could be used for injecting pellets

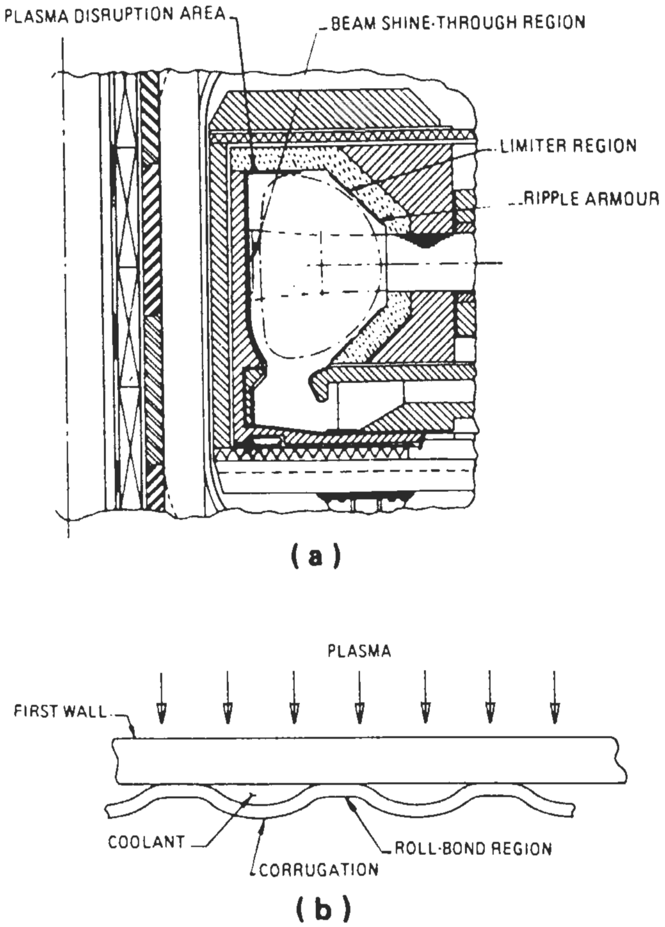


FIG.12. First-wall configuration.
(a) Poloidal view of reactor;
(b) Cross-section of first wall.

from two feed-lines concurrently. The centrifugal injector is rotating many times faster than required for the feed-rate; therefore, two or more species can be injected with the same device. A pneumatic injector can also be developed with two or more launch tubes in the same barrel to accomplish the two-channel injection. At this time, it is desirable to design the shielded drift tube and isolation valve to accommodate either type of injector and to defer a choice until further development is carried out.

3.6. First-wall system

A conceptual design of a first-wall system that will survive the total reactor life has been developed for INTOR. The first-wall system consists of: (1) an outboard region that serves as the major fraction of the plasma chamber surface and receives particle and

TABLE XIII. INTOR FIRST-WALL OPERATING PARAMETERS

FIRST WALL	
Total plasma chamber area	380 m^2
Average neutron wall loading	1.3 $MW \cdot m^{-2}$
Radiative power to first wall	40 MW
Charge-exchange power	4 MW
Charge-exchange current (50% D, 50% T)	1.3×10^{23} s^{-1}
Charge-exchange flux	3.3×10^{20} $m^{-2} \cdot s^{-1}$
Charge-exchange energy	200 eV
Total disruption energy	220 MJ
Disruption time	20 ms
Total average neutron flux	6.8×10^{26} $n \cdot m^{-2}$
Integrated 14-MeV neutron wall loading	6.5 $MW \cdot a \cdot m^{-2}$
Total number of shots	7.1×10^5
Total number of disruptions	1080
OUTBOARD WALL	
Area	266 m^2
Surface heat flux from plasma	11.6 $W \cdot cm^{-2}$
Surface heat flux from divertor	2.0 $W \cdot cm^{-2}$
Total surface heat flux	13.6 $W \cdot cm^{-2}$
Average nuclear heating	13 $W \cdot cm^{-3}$
LIMITER (Outboard wall at R ≈ 6 m – top and bottom)	
Width	1 m
Area (each)	38 m^2
Total ion flux	3×10^{23} s^{-1}
Total heat flux	10 MW
Total ion heat flux	5 MW
Heat flux density	0.3 $MW \cdot m^{-2}$
Peaking factor	1.5
Typical particle energy	100 eV
Duration	4 s
Period	t = 0–4 s
RIPPLE ARMOUR (Outboard wall at R ≈ 6 m – top and bottom) (does not coincide with the limiter)	
Area	26 m^2
Heat flux (ripple = ± 0.5%)	0.4 $MW \cdot m^{-2}$
Peaking factor	2
Particle energy (D)	120 keV
Duration	2 s
Period	t = 8–10 s
INBOARD WALL	
Area	114 m^2
Surface heat flux	11.6 $W \cdot cm^{-2}$
Average nuclear heating	13 $W \cdot cm^{-3}$
Peak disruption energy density	289 $J \cdot cm^{-2}$

TABLE XIII. (cont.)

BEAM SHINE-THROUGH REGION	
(inboard wall)	
Total power (5% of injected)	4 MW
Particle energy	175 keV
Duration	2 s
Period	t = 4–6 s
Area	4 m^2
Heat flux	1 $MW \cdot m^{-2}$

radiation heat fluxes from the plasma and radiatiative heating from the divertor, (2) an inboard region that receives radiative and particle fluxes during the plasma burn and the major fraction of the plasma energy during a disruption, (3) a limiter region on the outboard wall that serves to form the plasma edge during the early part of start-up, (4) a beam shine-through region on the inboard wall that receives shine-through of the neutral beams at the beginning of neutral injection, and (5) a region on the outboard wall that receives enhanced particle fluxes caused by ripple effects during the late stages of neutral injection. Figure 12 is a poloidal view of the reactor showing the location of the various first-wall regions. Table XIII summarizes the operating parameters for the first-wall system.

The reference concept for all first-wall regions is a water-cooled stainless-steel panel (see Fig.12). The wall thickness of the special regions, e.g. the limiter and inboard regions, is increased to allow for enhanced erosion, caused by the preferential heat or particle fluxes. The 20% cold-worked type-316 stainless steel, which is selected as the structural material, provides adequate radiation damage resistance for full life and an allowable design stress intensity sufficient to meet code specifications for the reference conditions. Stress corrosion cracking of the stainless steel in the low-temperature, low-pressure water should be avoided by control of the water chemistry. The thin corrugated coolant channels in the panel-type construction selected for the first wall tend to minimize bending stresses and provide longer life-time than tubes. The outboard wall is integral with the blanket and serves as the containment for the neutron multiplier.

The erosion rates and thickness requirements for the various regions of the first-wall panel have been evaluated. The physical sputtering erosion rates are based on effective sputtering yields of 0.017 atoms per particle at 200 eV, and 0.006 atoms per particle at 100 eV, for the 50% D – 50% T charge-exchange flux incident on stainless steel. The calculated vaporization erosion caused by a plasma disruption is 8×10^{-4} mm per disruption for a 289 $J \cdot cm^{-2}$ energy density deposited in 20 ms. An uncertainty factor of two is used to obtain the design erosion allowance of 1.8 mm on the inboard wall during the 15-year reactor life. A thin melt layer (~140 μm) is predicted to form on portions of the inboard wall during a plasma disruption. It is assumed that this thin melt layer does not erode during the short time (~10 ms) that it is molten.

Table XIV is a summary of the life-time analysis of the first-wall system. Regarding the wall thickness requirements necessary to allow for the predicted erosion rate: all regions meet the design temperature, stress and fatigue criteria for full-life operation under the reference conditions. The major uncertainty in this

TABLE XIV. SUMMARY OF LIFE-TIME ANALYSIS

Region	Total thickness (mm)	Maximum erosion (mm)	Maximum temperature[a] (°C)	Maximum stress[b] (MPa)	Fatigue life (cycles) No erosion	Fatigue life (cycles) With erosion[c]
Outboard wall	11.7	8.7[d]	260	360	3×10^6	10^7
Ripple region	11.7	8.7[d]	297	400	1×10^6	10^7
Limiter region	12.8	9.8[d]	280	410	8×10^5	10^7
Inboard wall	13.5	10.5[e]	275	408	9×10^5	10^7
Beam shine-through region	13.5	10.5[e]	332	495	2×10^5	10^7

[a] Maximum specified temperature = 350°C.
[b] Maximum allowable stress = 650 MPa plasma side, 765 MPa coolant side (cold-worked material).
[c] Assumes erosion rate, one-half of predicted rate for conservative design.
[d] Physical sputtering.
[e] Physical sputtering plus vaporization.

design concept relates to the stability of the melt layer predicted to form during a disruption. A grooved inboard wall concept would accommodate erosion of up to 10% of the melt layer (~0.14 mm/disruption). Further research and development are required to confirm the stability of the melt region during a disruption.

Aluminium was considered as an alternative first-wall structural material because of its superior thermal conduction properties, hence lower thermal stresses. Stainless steel was, however, favoured because of the substantially greater data base, its pre-eminent role in world-wide fusion development programmes, and its greater ability to withstand disruptions.

The back-up first-wall concept consists of a radiatively cooled graphite liner on the inboard wall, with a water-cooled stainless-steel panel for the outboard wall. The graphite tiles (30 cm square by ~4 cm thick) are installed on rails and require removal of the blanket shield sector for replacement. These tiles, which operate at ~1500 K, produce ~40 MW of additional nuclear heating load that must be radiated to the outboard wall and the wall behind the tiles. The outboard wall is similar to the reference stainless-steel panel design.

3.7. Divertor collector plates

The impurity control system in INTOR is a single-null poloidal divertor located at the bottom of the plasma chamber, as shown in Fig.3. A summary of the operating conditions is given in Table XV. The total power to the divertor is 80 MW, which is equally divided between inner and outer channels. A total of 70 MW of that power impinges directly on the divertor collector plates, resulting in high surface heat and particle fluxes, in addition to the neutron flux. These severe operating conditions mean that the divertor collector plate is the most severely damaged torus component, and hence it is designed for frequent replacement. Modular divertor sectors have been designed which can be removed in twelve pieces – one piece per torus sector. A single-null divertor was selected to maintain configuration simplicity and to facilitate maintenance. All torus vacuum pumping is provided through the divertor pump duct.

Most of the divertor design effort has focused on the divertor collector plates because of the severe operating conditions. The collector plates will potentially be subjected to large temperature gradients and thermal stresses, large physical sputtering rates, and radiation damage in the form of swelling, embrittlement, and

TABLE XV. DIVERTOR OPERATING CONDITIONS

Design concept	Single-null poloidal divertor
Total power to divertor	80 MW
Ion power to divertor plates	35 MW
Electron power to divertor plates	35 MW
Charge-exchange power to throat and walls	5 MW
Radiation power to throat and walls	5 MW
Power to channels – outboard	40 MW
– inboard	40 MW
Peak power flux to channels at null	
(normal to separatrix) – outboard	$8\ MW\cdot m^{-2}$
– inboard	$4\ MW\cdot m^{-2}$
Total ion flux to divertor	$5.5 \times 10^{23}\ s^{-1}$
Average energy of ions	400 eV
Peak ion flux to channels at null	
(normal to separatrix) – outboard	$6 \times 10^{22}\ m^{-2}\cdot s^{-1}$
– inboard	$3 \times 10^{22}\ m^{-2}\cdot s^{-1}$
Total neutral flux to divertor throat and walls	$1.6 \times 10^{23}\ s^{-1}$
Average energy of charge-exchange neutrals	200 eV
Uniform neutral particle flux	$7 \times 10^{22}\ m^{-2}\cdot s^{-1}$
Peaking factor of deposition load	2

creep of the plate mateirals. To reduce the impinging fluxes, the inner plate is inclined at an angle of 30° and the outer plate is inclined at an angle of 14.5° with respect to the separatrix. These inclinations reduce the peak surface heat flux to $2\ MW\cdot m^{-2}$ and the peak particle flux to $1.5 \times 10^{22}\ m^{-2}\cdot s^{-1}$.

Two design concepts have been examined in detail during Phase 1. Both designs consist of protection plates (tungsten or molybdenum) which are attached to heat sinks composed of a standard structural alloy (stainless steel or copper). The protection plate is eroded during particle bombardment and requires replacement. Tungsten and molybdenum have the lowest sputtering rates of all refractory materials. The major difference between the design concepts is in the method of attachment for the protection plate. The first design concept employs a braze, resulting in a high thermal conductance between the plate and heat sink. The second concept employs mechanical attachments, resulting in low thermal conductance between the plate and heat sink.

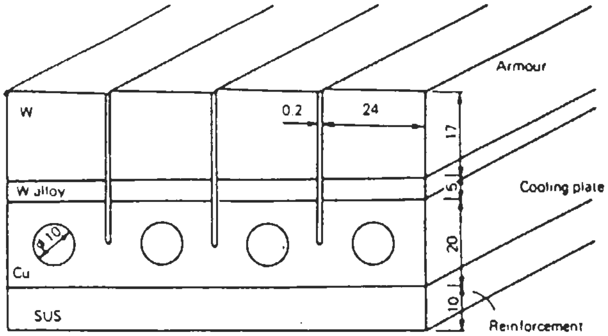


FIG.13. Brazed divertor plate concept.

In the brazed concept illustrated in Fig.13, the plate temperatures remain low, but the plate is constrained from thermal expansion by the braze joint. The problems of high-temperature operation are alleviated, but the restraint of the braze bond results in high stresses that could lead to cracking of the plate or the copper alloy heat sink, or failure of the braze joint. The braze bonds are not characterized well enough at this time to ensure the integrity of the bond during operation in INTOR. The copper alloy heat sink may incur radiation damage which could limit its life-time to much less than one year of operation.

In the mechanically attached concept, shown in Fig.14, the mechanical attachments allow the plates to expand freely and to rotate as the temperature changes during the burn cycle. This design results in low thermal stresses and allows the protective plate thickness to be increased in order to increase the sputtering life-time. During the burn cycle, the plate temperatures increase from 1600°C to 2300–2500°C, at which point 40–50% of the incident heat is radiated back to the divertor and plasma chambers. This reduces the thermal gradient in the protection plate and the heat flux to the stainless-steel heat sink, which is designed for the life of the reactor. The major concerns of the design involve the high temperature of operation which leads to re-crystallization and embrittlement. At elevated temperatures, chemical sputtering by oxygen impurities may be large, approaching as much as 75% of the physical sputtering rate. Also, the temperature of operation depends upon the emissivity of the protection plate.

Both concepts appear feasible, but contain considerable uncertainties. Further investigation is required before a final design can be selected.

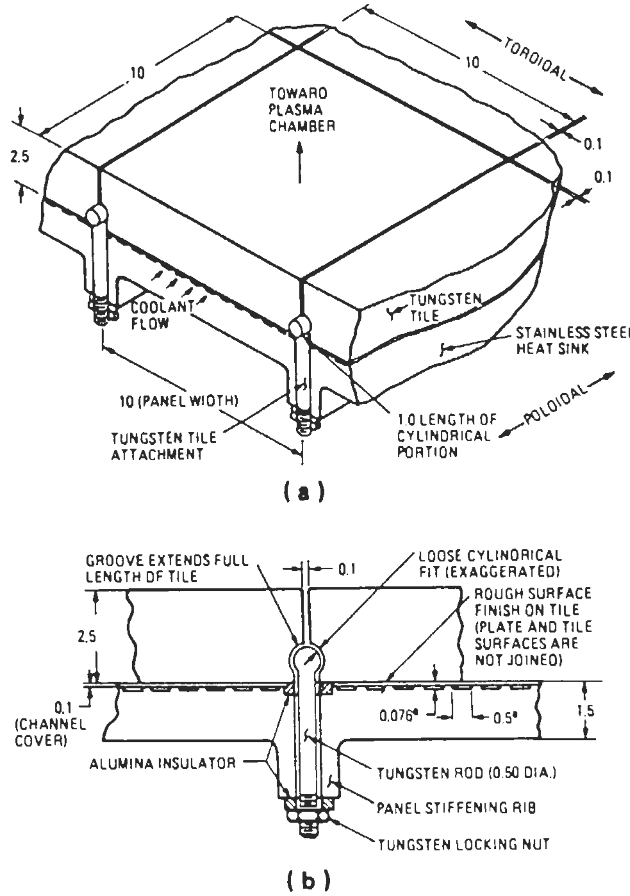


FIG.14. Mechanically attached divertor plate concept.
(a) Isometric cut-away through typical plate assembly;
(b) Cross-section through typical plate assembly (full scale; looking in poloidal direction).

(All dimensions in cm; all dimensions are typical.)
[a]Channel dimensions shown are for peak heat flux region only.

3.8. Tritium-producing blanket

Incorporation of a tritium-breeding blanket in the INTOR design is based on both economic and tritium availability considerations. The cost of tritium is estimated to be on the order of a billion dollars for the reactor life-time and the availability from existing sources is questionable. From the engineering point of view it is prudent that the tritium-breeding blanket be limited to the outboard and upper regions of INTOR. A minimum breeding ratio of 0.6 was recommended as the criterion for the INTOR blanket. Since economics and tritium availability are the primary reasons for a tritium-producing blanket in INTOR, it is not essential that the blanket design, materials and

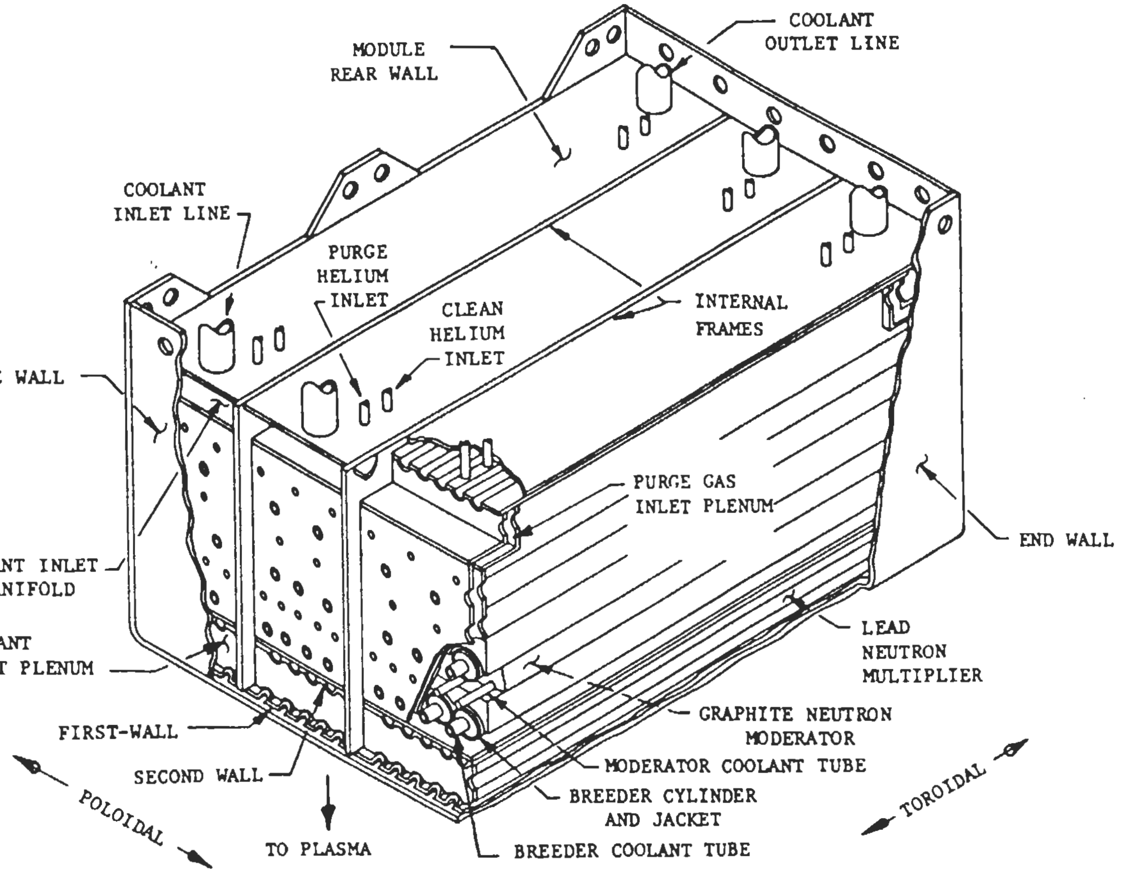


FIG.15. Reference blanket design 1.

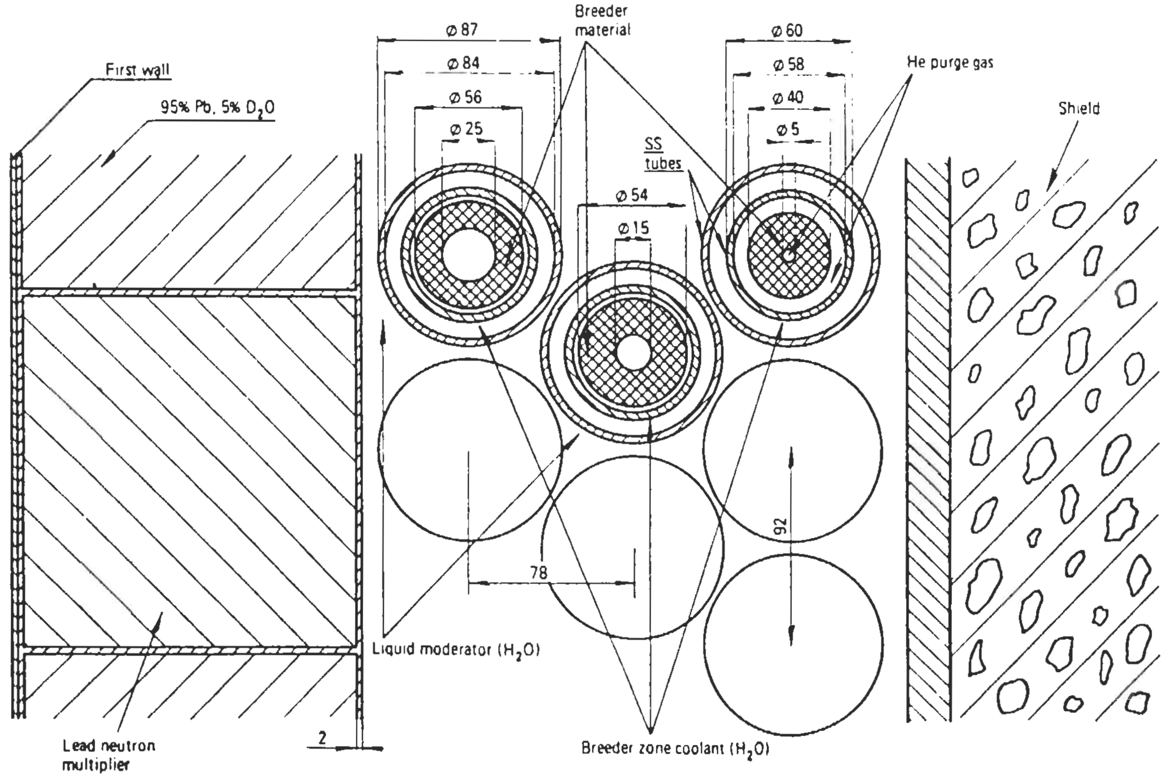


FIG.16. Reference blanket design 2.

operating parameters be reactor-relevant at the expense of increasing the risk in the engineering design.

Solid and liquid breeder materials were evaluated for the INTOR blanket design. The solid breeding materials offer the advantages of engineering design simplicity and relatively low stored chemical energy. The major question regarding the use of solid breeding materials relates to tritium recovery. The greatest uncertainty involves the possible effects of radiation on the tritium release mechanisms. Of the different solid breeding materials, the ceramic compounds are believed to offer the most potential for acceptable tritium recovery.

The reference blanket concepts are based on Li_2SiO_3, with Li_4SiO_4 as a back-up material. (Alternative blanket concepts with Li_2O and $Li_{17}Pb_{83}$ as the breeding materials were also developed.) A lead neutron multiplier is incorporated into the blanket to achieve the desired breeding ratio, and the first wall is structurally integral with the blanket. Low-temperature water coolant is used for all blanket regions. Two reference blanket designs were chosen. The primary difference between them relates to the choice of moderator and the location of the breeding material with respect to the coolant. The first design, referred to as Design 1, used graphite as a solid moderator, with the breeding material located outside the coolant tube (Fig.15). In the second design (Fig.16), referred to as Design 2, water (H_2O) is used as a liquid moderator and the breeding material is concentrically located inside the coolant tube. In both designs a lead multiplier is placed between the first wall and the breeding region. Low-temperature H_2O is used for the coolant in the breeding region, whereas D_2O is used as the coolant for the first wall and neutron multiplier. The main parameters for the two designs are summarized in Table XVI.

The Design 1 concept incorporates two key features which permit a blanket design that is easily adapted to the varying width of the top region and to changes in neutron wall loading with distance from the mid-plane: (1) a modular approach that divides the blanket poloidally into a number of discrete segments, and (2) coolant flow in the toroidal direction across the sector width. The Li_2SiO_3 is fabricated at 70% theoretical density with lithium enriched to 30% ^{6}Li. The lithium silicate is formed in cylinders around the 1 cm ID stainless-steel coolant tubes. As shown in Fig.15, there are three separate rows, or banks, of breeder/coolant-tube assemblies. The solid graphite moderator is located between these banks and in front of the rear module wall. Separate, small-diameter coolant tubes are also located in the moderator region. A thin metal tube (jacket) surrounds each breeder cylinder to prevent reaction of the graphite with the breeder material. Tritium is removed from the ceramic breeder by a low-pressure (~0.01 MPa) helium purge stream. The breeder temperature is maintained between 400 and 600°C to facilitate tritium release. A 5-cm-thick lead neutron multiplier is located between the breeding region and the first wall. A water-cooled panel, which separates the lead from the breeder materials, provides cooling for adjacent regions. The lead is cooled on the front side by the water-cooled first wall which also serves as part of the blanket containment.

The Design 2 blanket concept can also be adapted to the varying width of the top region and to changes in neutron wall loading with distance from the mid-plane by adopting a modular approach and with a

TABLE XVI. REFERENCE DESIGNS PARAMETERS FOR TRITIUM PRODUCING BLANKET

	Design 1		Design 2
Neutron multiplier			
Material		Pb	
Thickness		5 cm	
Maximum temperature		290°C	
Melting point		327°C	
Coolant	H_2O		D_2O
Breeder region			
Breeder material		Li_2SiO_3	
Breeder max/min temperature		600/400°C	
Effective density		0.7	
Breeder element diameter	4–6 cm		6 cm
Enrichment of ^{6}Li		30%	
Tritium processing gas		He	
Structural material		316 SS	
Maximum structural temperature		150°C	
Coolant		H_2O	
Coolant inlet/outlet temperature		50/100°C	
Coolant pressure		0.7 MPa	
Neutron moderator	C		H_2O
Breeder region thickness	43 cm		26 cm
Effective blanket coverage		0.6	
Net tritium breeding ratio		0.65	

coolant flow in the toroidal direction across the section width. This concept has as its main advantages: (1) simplified fabrication of the solid breeder material, and (2) minimal breeder material inventory and blanket thickness by using a most effective moderator. This concept incorporates three rows of breeding elements behind the neutron multiplier region. Each element consists of a hollow cylindrical Li_2SiO_3 breeder encased in a stainless-steel tube which is cooled externally by water coolant contained in a second, outer concentric tube. Both the inside and outside diameters of the ceramic breeder are adjusted in each of the rows to accommodate radial and poloidal variations in the nuclear heating. Tritium recovery is accomplished similarly to Design 1 by flowing low-pressure (~0.01 MPa) helium through the central void and annular gap surrounding the solid breeder. As before, the low-density (70% of theoretical density) solid breeder and the controlled temperature (400–600°C) facilitate recovery. The three rows of elements are surrounded by low-temperature H_2O water moderator that is separate from the coolant. The water moderator and breeder elements are contained in a stainless-steel vessel located directly behind the 5-cm-thick lead neutron multiplier region, which is similar to that in Design 1.

A net tritium breeding ratio of 0.65 is attainable in both designs. The total blanket tritium inventory was estimated at as much as 200 g, based on data for unirradiated material. Theoretical estimates have shown that radiation effects could increase this inventory to ~1 kg. Thus, the range of 0.5–1.0 kg is specified for the INTOR breeding blanket tritium inventory.

3.9. Radiation shield system

The INTOR shield system was designed to:
(1) protect reactor components from excessive radiation damage and nuclear heating; (2) reduce the induced activation level in reactor components; and
(3) protect the workers and the public from radiation exposure. The system is designed to permit personnel access into the reactor buildings with all shielding in place within 24 h after shut-down. The shield system consists of the torus bulk shield, the penetration shields and the reactor building walls that serve as biological shielding.

Because of space limitations on the inner side of the torus, the inboard shield was carefully optimized. The total thickness in the mid-plane, from the plasma side of the first wall to the inner surface of the conductor region of the toroidal field coil, is 1.1 m. The reference configuration of the inboard shield includes a 40-cm-thick region of 90% stainless steel plus 10% water and a 34-cm-thick region of 60% borated steel plus 40% water. The actual shield thickness, including the first-wall and a 3-cm gap is 0.8 m. The stainless-steel first wall and the magnet case and dewars provide additional radiation attenuation. Table XVII summarizes the radiation response in the inboard region of the TF coils.

In addition to radiation protection of reactor components, the outboard shield was designed to permit personnel access after shut-down. The reference design calls for 105 cm shield thickness behind a 50 cm tritium breeding blanket to reduce the biological dose rate to 2.5 $mrem \cdot h^{-1}$ outside the bulk shield within 24 h after shut-down. The main region following the blanket is 0.7 m thick and contains 90% steel and 10% water. This region is followed by a 0.28-m-thick region of borated steel and water. A 0.04 m lead layer is the outermost region. Steel with low nickel content is used to reduce induced radioactivity.

Since the detailed design of the penetration shield will significantly affect reactor cost, personnel access requirements and reactor operation, an elaborate three-dimensional radiation streaming calculation for the

TABLE XVII. SUMMARY OF RADIATION RESPONSE PARAMETERS IN THE INBOARD REGION OF THE TF COILS
(based on 1.3 $MW \cdot m^{-2}$ neutron wall load and 6 $MW \cdot a \cdot m^{-2}$ integral neutron wall load)

Maximum neutron fluence in the superconductor	4×10^{17} $n \cdot cm^{-2}$
Maximum induced resistivity in copper stabilizer	3×10^{-8} $\Omega \cdot cm^{-1}$
Maximum atomic displacement in copper stabilizer, dpa[a]	2.5×10^{-4}
Maximum dose to thermal insulator	2.5×10^{9} rad
Maximum dose to electrical insulator	7×10^{8} rad
Maximum nuclear heating rate in superconductor	9×10^{-5} $W \cdot cm^{-3}$
Total nuclear heating in twelve TF coils	7 kW
Total nuclear heating in magnet vacuum vessel	9 kW

[a] = displacements per atom.

complete reactor system, including the neutral-beam injectors and divertor ducts, was carried out. The general-purpose Monte-Carlo code MCNP was used for the calculations, with a continuous energy representation for the nuclear cross-sections from ENDF/B-IV. The energy and spatial distributions of the D-T plasma source neutrons were modelled in the calculations. Calculations were performed for neutron and photon transport during reactor operation and decay gamma transport after shut-down. The results show 3-kW nuclear heating in the vacuum pumps of the neutral-beam system with the shutter open. The calculations show the biological dose rate in the reactor building outside the bulk shield to be 2.5 $mrem \cdot h^{-1}$ within 24 h after shut-down, with the following penetration shield thicknesses: (1) 1 m for neutral beam drift tubes; (b) 0.75 m for the surfaces of the beam injector box facing the drift tubes; and (c) 0.5 m for the rest of the neutral-beam system and the divertor ducts.

3.10. Tritium and vacuum systems

3.10.1. Tritium systems

The INTOR design incorporates a complete deuterium-tritium-lithium fuel cycle. The tritium systems perform the following functions: to reprocess tritium for fuelling; to process tritium produced in the blanket; to process tritiated wastes and tritium contained in the coolant; and to recover and control the amount of tritium in all buildings. All these functions must be performed, keeping a minimum tritium inventory, minimizing the tritium impact on the environment and tritium waste generation, and reducing worker exposure to levels as low as practical. The tritium systems are to be operated in areas free of gamma or neutron irradiation and must be designed to have maximum reliability and availability, including redundancy of components.

The tritium flow rate in the tritium reprocessing system for one continuous day of operation is 1035 $g \cdot d^{-1}$ during Stage I, and 1597 $g \cdot d^{-1}$ during Stages II and III, including both the bred tritium and the external supply. The tritium inventory in the plasma reprocessing system is on the order of 200 g, to which must be added 390 g contained in vacuum pumps, fuellers, pellet fabrication units and blanket tritium recovery systems. Taking into account a storage of 2300 g (30 days of full operation) and the in-blanket inventory of 0.5–1.0 kg, the total INTOR tritium inventory is in the range of 3.4–3.9 kg.

The exhaust plasma is processed cryogenically. All gaseous impurities are trapped at cryogenic temperature, except He, which is separated by using a falling film condenser just before the cryogenic distillation unit in which the isotopic separation occurs. Four columns and two equilibrators are required in this unit. The separated isotopes are sent back into fuellers or pellet fabrication units.

INTOR is equiped with a breeding blanket in which Li_2SiO_3 is used as a breeding material. Tritium is removed from this blanket, mainly as T_2O, by means of a helium purge stream. The first step of the bred tritium-processing system consists of a conversion of all the gaseous fraction that is not in water form into water by passing through a catalytic reactor. Then the water is trapped at cryogenic temperature, separated from the helium stream and electrolysed. The recovered tritium is then directed to the fuel clean-up system.

One important concern is the permeation of tritium through the first wall and divertor plates. The implantation of a high concentration of tritium atoms in the inner surface of the first wall, combined with a relatively slow recombination of these atoms at the first surface, could significantly enhance the amount of tritium that permeates through the first wall into the coolant. Further work is needed in this area.

In order to limit the total tritium release to less than 10–20 $Ci \cdot d^{-1}$, the coolant must be processed to maintain a sufficiently low tritium concentration. The tritium containment systems must be designed so that all tritium releases (normal or accidental) remain within the confines of the plant. The atmospheric tritium level in all areas must be maintained at $<5\ \mu Ci \cdot m^{-3}$. Triple containment is used where reasonable and practical; double containment is used throughout. Decontamination is facilitated by dedicating an appropriately sized atmospheric tritium recovery unit to each tritium containing area. Interfacing the heating, ventilation and air conditioning system with the air tritium recovery system prevents release of tritium to the outside environment. Following these rules, the design goal of a total tritium release of about 10 $g \cdot a^{-1}$ could be reached.

3.10.2. Vacuum systems

3.10.2.1. The torus vacuum system

The initial base pressure in the toroidal vacuum chamber should be 10^{-7} torr. Baking and discharge cleaning will be used. The assembled chamber is baked up to 200–300°C. The maximum baking temperature is limited by the lead blanket zone. The surfaces facing the plasma are cleaned of oxygen and carbon by a hydrogen discharge, the chamber walls being hot. The effective pumping speed of the chamber required to provide the initial base pressure is 5×10^4 litre·s^{-1}.

During the dwell time the pressure in the vacuum chamber should reach 3×10^{-5} torr within 20 s. An effective pumping speed of 1.5×10^5 s^{-1} is adequate.

In a stationary plasma burn mode it is necessary to keep the helium concentration in the plasma below 5%. To accomplish this, about 1% of the helium atoms escaping from the plasma should be removed from the chamber. An assessment shows that the effective helium pumping speed should be about 2×10^5 litre·s^{-1}. Thus, the pumping requirement during the burn time is decisive in determining the pumping speed requirement.

The pumping system of the toroidal chamber consists of twelve pumping units symmetrically placed relative to the reactor axis. Each pumping unit is connected with the divertor chamber by a vertical exhaust duct of 1.0 to 1.2 m diameter and 9 m length. To provide the effective helium pumping speed of 2×10^5 s^{-1}, the pumping speed of each of the 12 pumps for helium should be 5×10^4 s^{-1}.

Compound cryopumps are to be used to pump helium and D-T mixture. Two compound cryopumps are installed in parallel in each of the twelve pumping units. This arrangement allows regeneration of twelve pumps while twelve pumps are in service.

Apart from the compound cryopumps, turbomolecular and oil-free mechanical pumps are employed in the units for pumping during vacuum chamber initial preparation to obtain 10^{-7} torr background pressure and for use in cryopump regeneration.

3.10.2.2. Neutral-beam injector vacuum system

The injector vacuum system includes the vacuum chamber itself, which contains a gas neutralizer and the main high-vacuum pumps, and other high-vacuum and fore-vacuum pumps, valves and gauges. The positive-ion sources are connected to the vacuum chamber.

The injector vacuum system should provide: initial base pressure of 10^{-7} torr; vacuum conditions in ion sources, neutralizers and injection track which provide effective ion neutralization, minimum beam losses due to re-ionization not exceeding a few per cent, and minimum diffusive cold deuterium flow from the injector through the drift tube into the discharge chamber; minimum tritium content flowing from the discharge chamber into the injector; capability to replace ion sources without breaking the vacuum in the injector; and capability for pressure measurement and vacuum interlocking in the operating pressure range.

Liquid helium-cooled condensation pumps are used as a basic means for high-vacuum pumping. The specific deuterium pumping speed on helium cryopanels is ~8·litre·s^{-1}·cm^{-2}. To reduce cryopump speed, a differential pumping system is used in the injectors, where helium panels are placed in diaphragm-separated chambers. The differential pumping system provides reduction of the cryopanel area up to 45–50 cm^2. Cryopumping speed in each injector is (3–4) $\times 10^6$·litre·s^{-1}.

Other modes of pumping, i.e. turbomolecular and oil-free mechanical pumps, are used to reach the initial base pressure and during regeneration of cryopanels.

Vacuum gates arranged at the injector drift tubes provide the reactor chamber vacuum at injector replacement and cryopanel regeneration. There are also fast shutter valves in the injection lines that are opened only during injection in order to limit tritium flow into the injectors. All ion sources are connected to the injector through gate valves so that the sources can be replaced without warming the cryopanels.

3.11. Diagnostics, instrumentation, data acquisition and control

A preliminary study of the systems needed to instrument, diagnose and control INTOR has been completed. The need has been identified for two essentially different system levels to separate the control function from instrumentation and diagnostics.

Plasma diagnostics must be extended in many respects for operation in the neutron environment of a D-T plasma. The instrumentation that will be required for the plasma in INTOR will be more extensive than in any previous tokamak.

The instrumentation required for the technological systems can be adapted from other fields (radiation detection from nuclear fission reactors, magnet from high-energy physics, etc.). The development of an integrated system will, however, require a substantial effort.

The computer systems technology required for data acquisition and diagnostics is adequate.

3.12. Site criteria and facility layout

3.12.1. Site criteria

The main criteria for the choice of a site are:
(a) Sufficient area of land for buildings and services, and for an exclusion radius for the general public, the area to be in a region of low population density;
(b) Appropriate land conditions from the points of view of seismic activity and ground load bearing;
(c) Availability of electric power supplies, including a line that provides a well-stabilized supply;
(d) Availability of adequate water and cooling supplies for both the construction and operation phases;
(e) Possibility of good arrangements for radioactive waste storage and disposal for very large pieces of equipment and slightly tritiated waste;
(f) Satisfaction of the safety requirements for accidental release of tritium as gas or as oxide;
(g) Availability of man-power resources required for the construction and operation of the reactor and for carrying out experiments, together with the associated housing and amenities;
(h) Availability of transport for unusually large items of equipment during construction and repair, for tritium during operation and for personnel for normal daily travel and for travelling abroad;
(i) Proximity of a well developed industrial base for the manufacturing of components.

3.12.2. Facility layout

A conceptual plant design study for the INTOR plant was undertaken to: define major buildings and their functional requirements; identify the radioactive boundary; and develop a preliminary plant arrangement.

Figure 9 illustrates the allocation of the facilities needed for operation and testing around the torus, namely: neutral-beam injectors (four active plus one redundant); fuelling devices (two); pumping ducts and pumps for the plasma exhaust evacuation (12); test module facilities (3); and instrumentation and control systems (2).

The following criteria have been followed for the layout of the facilities:
(a) Alternate positioning of the neutral-beam injectors;
(b) Fuelling devices on opposite positions as compared with the centre-line of the torus, in order to facilitate fuel injection;
(c) Test facilities near each other, in order to ease the transport of samples and test components to the hot cell area after irradiation;
(d) Combination of the expected tests on radiofrequency heating and electricity production in a single test sector location, because these two experiments are expected to be performed at different stages of the machine operation.

Figure 17 shows a plan view of the reactor building, and Fig.18 shows the vertical view of the reactor building. Figure 19 shows the entire plant layout.

Since detailed design and analysis have not yet been performed for INTOR facilities, the preliminary arrangement represents only one of several possible alternatives. The objectives of detailed design would be to reduce the cost of the facilities while providing satisfactory levels of performance. The major features of the present arrangement are:

(1) A centrally located, cylindrical tokamak building designed to withstand the maximum credible overpressure. The tokamak building would house the tokamak and major supporting equipment while allowing for the necessary maintenance and repair activities. The build ing may have to be designed to withstand high levels of seismic activity in order to meet important safety criteria on radioactive release.

(2) Primary cooling loops are contained in an annulus within the main tokamak building. This arrangement is possible because a simple heat injection system is used without the high temperatures and pressures required by power-generating turbines. The removal of afterheat is not expected to be difficult and can be accomplished by circulating the primary coolant using either normal or emergency power sources. The tritium-processing system is installed in a building which is located adjacent to the tokamak building.

(3) The repair and maintenance building is also a central part of the INTOR complex. This building is connected to the reactor building and the radwaste storage and cooling area by flexible joints so that the effect of earthquakes can be isolated. Radioactive materials are conveyed from the reactor building to the radwaste area through the repair and maintenance building. This configuration makes simple radiation control zones.

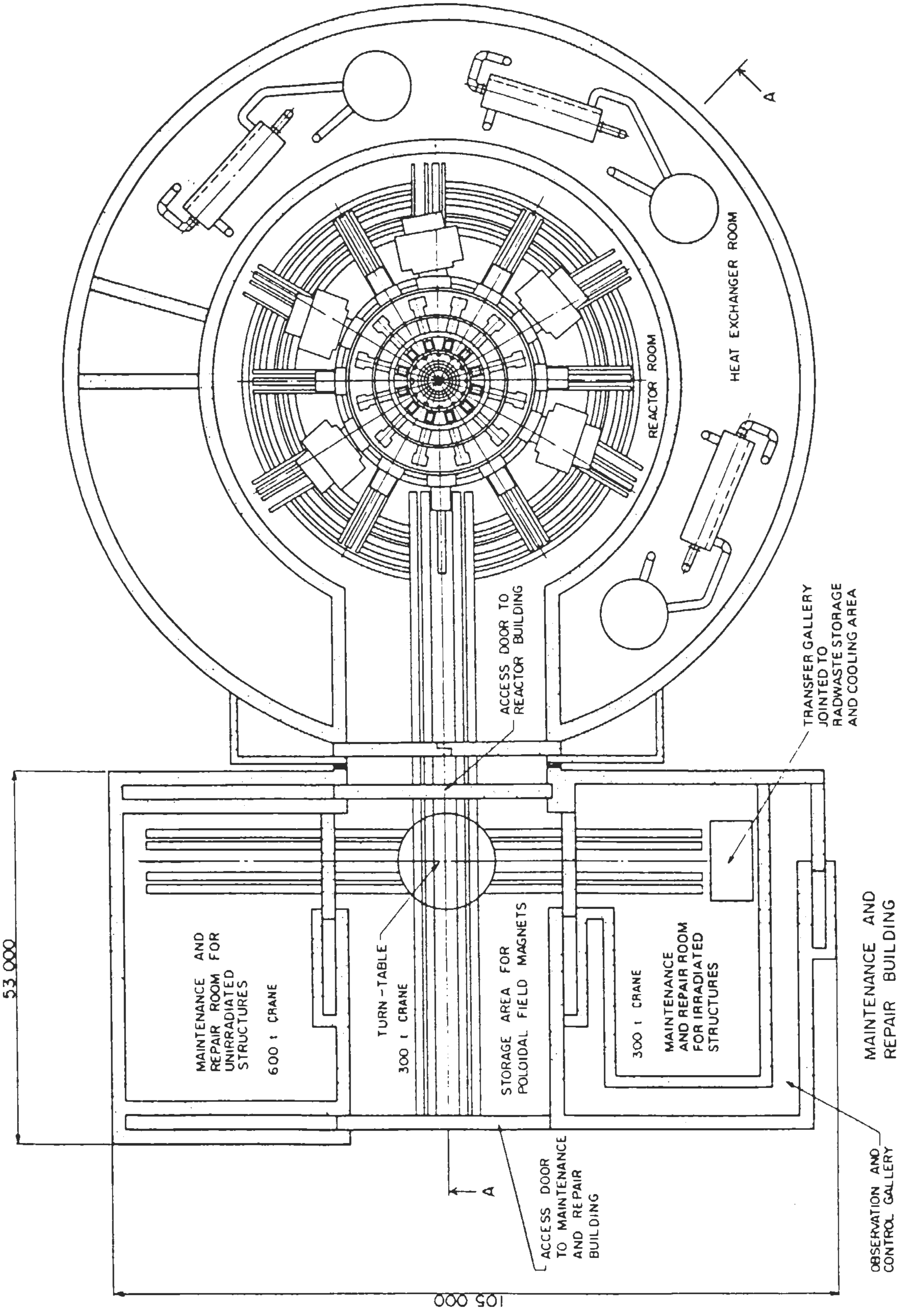


FIG.17. Plan view of the reactor building (ground floor).

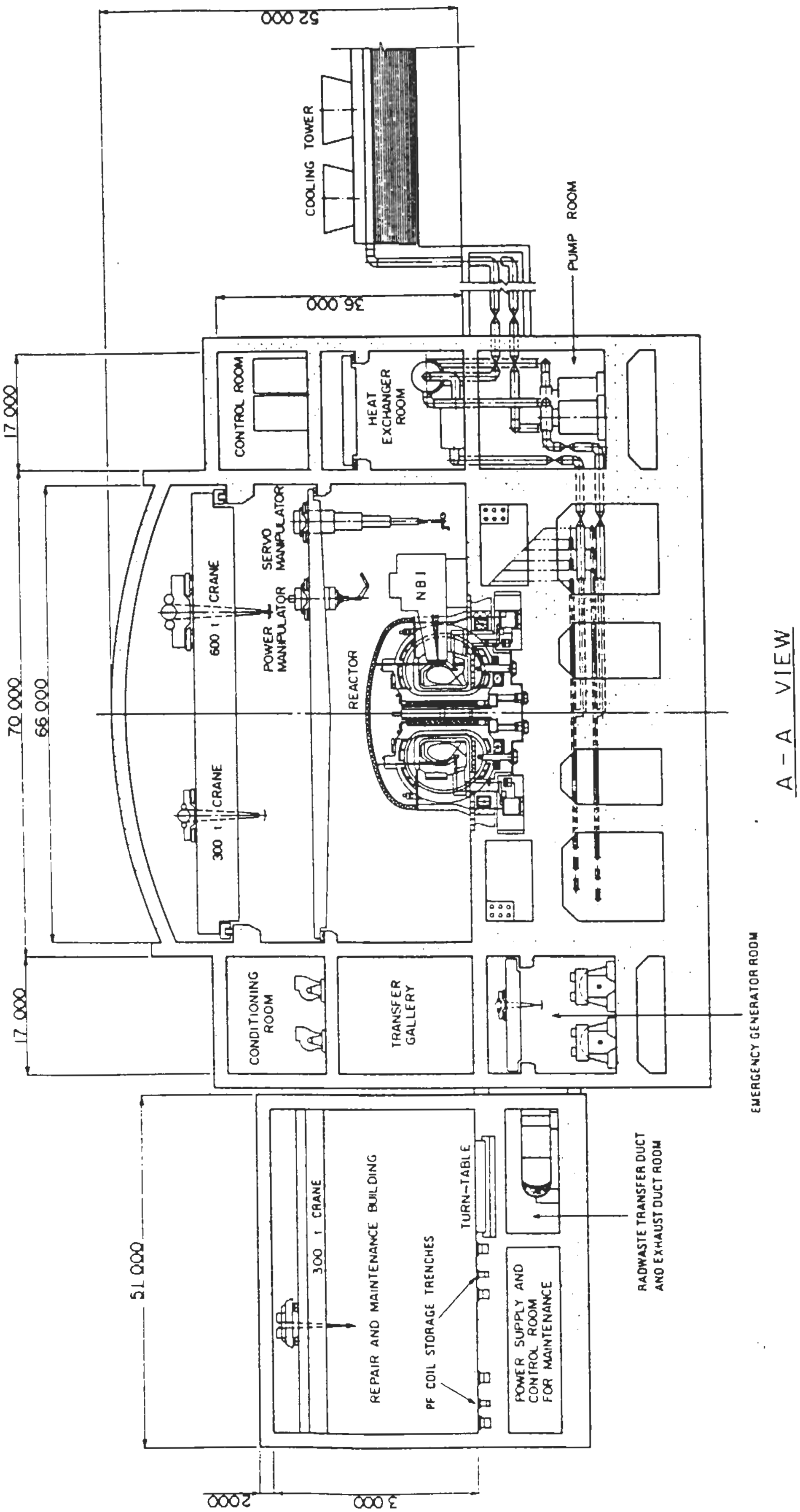


FIG.18. Cross-sectional elevation of the reactor building – section on A-A of Fig.17.

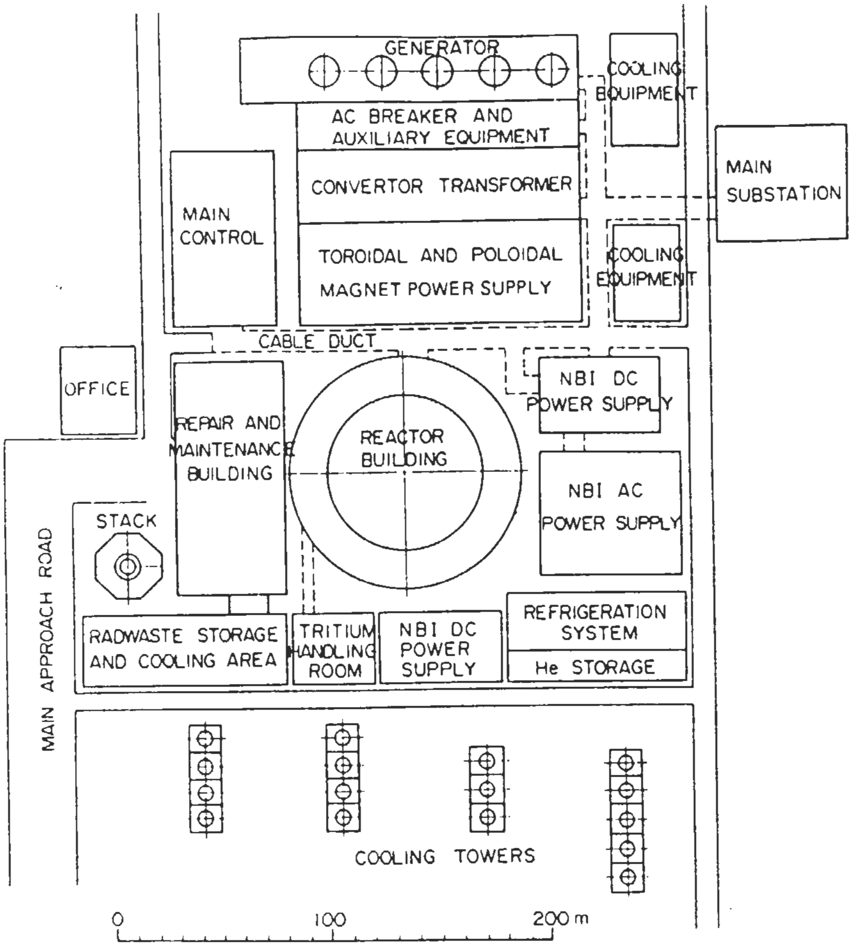


FIG.19. INTOR plant layout.

4. MACHINE OPERATION AND TEST PROGRAMME

A preliminary operation and test plan has been developed to provide insight into the design and operational requirements that must be imposed on INTOR. This plan has been developed using judgement as to where INTOR fits within an international fusion development plan, as discussed in Section 1, and taking into account the complementary roles that will be played by other plasma physics experiments and technology testing facilities. A summary of the types of testing included in the test plan is given in Table XVIII.

Plasma physics tests will emphasize those studies that cannot be performed on 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 refer to the testing of plasma heating and confinement hardware that interacts directly with the plasma. Testing radio-frequency launchers in a thermonuclear environment is an example.

Blanket and engineering tests will emphasize confirmation of results predicated upon ex-machina tests. Tests will include 1-m^2 prototype blanket modules, tritium recovery capsules and tests of critical life-limiting elements of the blanket. The critical element tests are used to provide closer simulation of DEMO reactor conditions and to permit use of accelerated testing. At least four long-term prototype blanket module demonstration tests are planned.

Short-term blanket tests will be used for design verification, and they require test periods ranging from ~ 1 month to 1 year. Demonstration tests are expected to be used to correlate testing results and analytical predic-

TABLE XVIII. TYPES OF TESTING IN INTOR

PLASMA PHYSICS	
	Vacuum vessel conditioning
	Ohmic heating start-up
	Neutral beam start-up
	Long-pulse ignition experiments
	Performance optimization
PLASMA ENGINEERING	
	Impurity control and exhaust technology
	RF heating technology
	Burn control technology
	Continuous burn methods technology
BLANKET AND ENGINEERING TESTS	
	Prototype module
	Tritium recovery
	Critical element
BULK MATERIALS	
	Irradiation effects upon properties of candidate structural materials, insulators, high heat flux materials, breeders and neutron multipliers
SURFACE EFFECTS	
	Retention/re-emission characteristics
	Plasma impurity release
	Surface erosion/re-deposition
	Surface microstructural changes
	Mechanical and physical property changes
INTOR REACTOR MATERIAL AND COMPONENT SURVEILLANCE	
	Engineering performance of systems
	Failure modes and rates
	Maintenance experience
NUCLEAR TESTS	
	Tritium breeding ratio
	Nuclear reaction rates
	Volumetric nuclear heating
	Neutron and gamma-ray fluxes and spectra
ELECTRICITY GENERATION	
	Early power generation – end of Stage II
	Prototype DEMO blanket – end of Stage III

tions for combined materials and synergistic effects and to provide information on performance changes with irradiation. These test modules will be left in the reactor until the end of Stage III. Testing to within a factor of two or three of the design life-time for DEMO is a requirement on INTOR.

Should failure of a blanket test module occur, failure analysis would provide information on failure modes and guide design variation tests for design improvement efforts.

Bulk materials property tests. A bulk material test programme has been defined to provide information for: (1) primary and back-up structural materials; (2) high-heat-flux materials; (3) insulators; (4) breeders; and (5) multipliers. Material properties of interest, the number of materials and variations, along with the number of test temperatures, fluences, duplications and other test conditions, resulted in the identification of 30 000 specimens for test in INTOR. Single-variable tests to characterize the effects of displacement rate, temperature and stress were included in defining requirements. Investigation of specimen volume requirements indicated that all 30 000 specimens could be tested in a single 1-m × 1-m test pocket and resulted in identification of a 5-cm-diam. × 15-cm-long standard capsule to contain a varying number of specimens. This capsule can contain as many as 1440 swelling and phase stability specimens. One in-situ cyclic fatigue specimen fits in a single capsule. A total of 300 capsules are required to contain all 30 000 specimens. As many as 153 capsules can be in the reactor at a given time. At each change-out interval, 60 capsules are removed and replaced. The temperature of each capsule can be controlled to operate at a specified level between 50°C and 700°C. Individual capsules can be removed without having to disconnect services. A second 1-m^2 test pocket will be used for single variable tests or to increase the fluence to the specimens.

The test duration was established by considering the fluence requirements of the DEMO first wall and blanket. Extrapolation by a factor of two to three is considered acceptable. To provide data over a range, it was decided to remove samples at intervals of 4%, 10%, 30%, 50% and 100% of the 6.6 $MW \cdot a \cdot m^{-2}$ design fluence objective. Tests will be started at the beginning of Stage-II INTOR operations and the last specimen will be removed at the end of Stage-III INTOR operations.

Surface tests. A surface test programme has been defined to provide data on the surface effects of divertor target, 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 the allocated 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 and in the divertor chamber. Specimen locations, cleaning method, temperature, material and fluence levels are varied in the test programme.

Nuclear tests. Nuclear tests have been defined to measure tritium breeding ratio, nuclear heating, reaction rates, neutron and gamma spectra and induced activation.

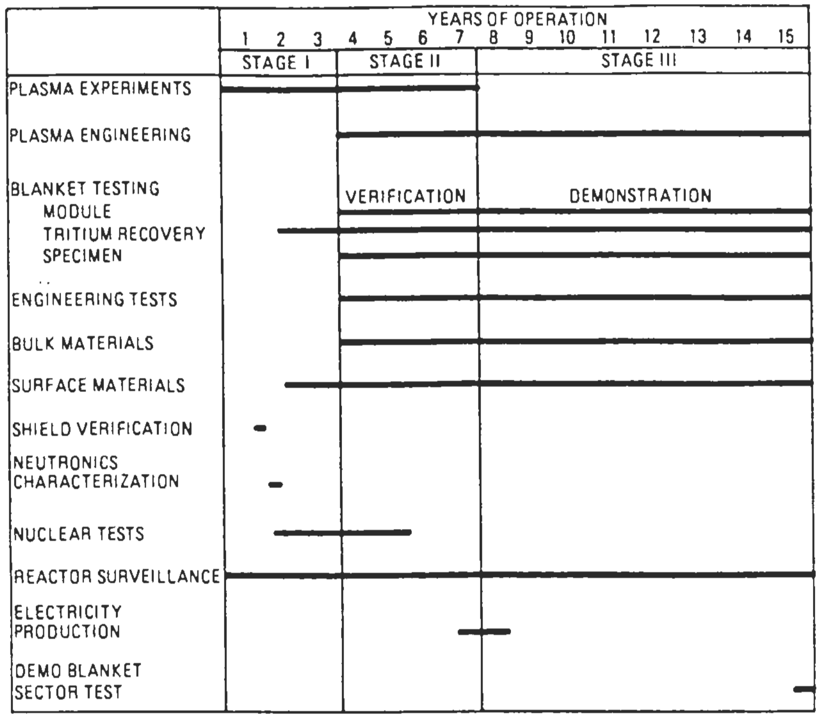


FIG.20. INTOR test schedule.

A need has been defined for neutron characterization testing to provide the distribution of the neutron source and to determine the radiation field in strategic locations. This and other key neutronic tests require an accuracy of within 5% in order to maximize the usefulness of these tests. Therefore, the tests should be performed during early D-T shots before background irradiation builds up and should be run at approximately five orders of magnitude lower power than normal operation in order to use direct measurement techniques.

Material and component surveillance. The basic INTOR reactor components and operation will provide useful information. Monitoring failure rates, failure modes and maintenance times will permit improvements to be achieved for a DEMO reactor. Achievement of some significant fraction of the component design life-time in the DEMO imposes availability and life-time requirements upon INTOR.

Electrical power generation at the end of Stage II and the beginning of Stage III should be accomplished by a special breeding blanket sector that is installed during the initial construction of INTOR, following successful blanket operating and module tests during Stage II. Simultaneous tritium and electrical power generation in a prototypical DEMO blanket sector could be performed at the end of Stage III.

Operational requirements. Some tests will require nearly continuous operation for some period of time. In particular, some of the tritium recovery tests will require a 70% duty cycle and continuous operation for one week to one month in order to reach equilibrium conditions. On the other hand, thermal-hydraulics testing will require only a 50% duty cycle and continuous operation for approximately one hour. Achievement of the required fluence for reliability testing of blanket modules and other components will require machine availability of 40–50%.

Test schedule. The projected test schedule is shown in Fig. 20. As indicated, plasma physics testing will dominate Stage-I operation. Stage-II testing will consist primarily of plasma engineering and blanket engineering tests and other tests where frequent change-out is required. A minimum time of one month between scheduled reactor shut-downs has been established to permit test change-out without unduly impacting reactor availability. Stage-III testing will be devoted to longer duration tests which do not require frequent reactor shut-down.

Installation. A total of three reactor sectors and one divertor slot have been allocated to testing. A standard-size 1-m^2 test pocket is used to provide test flexibility. Test module support devices are located in the reactor building basement to leave clearance around the reactor for machine maintenance and test module change-out.

Post irradiation test facility (PIE). A PIE facility has been defined for evaluating specimens from bulk materials and surface effects materials and for dissecting the blanket test modules. It is assumed that a complete PIE facility will be located at the INTOR site. The facility will provide for handling and disassembly, decontamination, mock-up of experiments, radiochemistry analysis, materials property testing and metallography testing.

5. SAFETY AND ENVIRONMENTAL IMPACT

As part of the INTOR Phase-One design, a preliminary assessment of the principal safety and environmental concerns of the proposed reactor has been prepared. The purpose of this assessment is to identify these concerns so that they can be addressed and solved, where possible, early in the design of the machine. This interaction between safety analysis and design is crucial in achieving the objective of ensuring that INTOR can be constructed and operated without undue risk to operating personnel and the general public. The primary safety concerns for INTOR have been identified as the radioactive inventories (tritium and activation products) and energy sources and mechanisms that could result in release of a portion of these inventories. No safety concern has been identified for INTOR that is not amenable to solution by prudent design, construction and operating procedures.

5.1. Radioactive sources

The INTOR tritium inventory has been estimated to be approximately 3.4 to 3.9 kg. Of this inventory about 2.3 kg is in the storage facility and 0.5–1.0 kg is resident in the blanket. The most vulnerable tritium inventories for release during an accident are those in the plasma chamber, vacuum system, and fuel processing system. The inventory in storage is in a separate vault and is relatively invulnerable to accidental release.

Activation products will be present in the solid structure and equipment surrounding the plasma chamber, the coolant streams, and the reactor building atmosphere. Estimates of the inventory range from approximately 3.2×10^8 Ci to 1.3×10^9 Ci. The most significant radioactive materials are isotopes of iron, manganese, chromium and cobalt that are constituents of stainless steel. A small portion of this inventory (24 to 70 Ci) is circulating in the primary coolant water from corrosion or sputtering. The water coolant will also contain a large quantity of ^{16}N (~ 1 Ci$\cdot$cm^{-3}); because of its short half-life, this isotope is not, however, considered to be a significant problem. Estimates were also made of the concentration of ^{41}Ar that would be present in the reactor hall during operation. Through proper operation of the ventilation system, air activation should not prohibit access to the reactor hall.

5.2. Energy sources and potential accidents

The energy sources present in the INTOR design have been identified and preliminary assessments have been made of potential accident scenarios for which these sources may lead to release of radioactivity. The largest of the energy sources in INTOR is that in the toroidal field magnets, of approximately 40 GJ. A number of accident scenarios have been postulated for which this energy could lead to damage of the magnets or associated equipment. Most of these scenarios lead to the formation of electric arcs in the magnet system; the consequences of such arcs are, however, uncertain. The credibility of such accident scenarios has not been established and experimental data are needed on the behaviour of superconducting magnets under accident conditions.

Analyses have been performed to determine the consequences of a loss-of-coolant accident on the first wall and blanket. These analyses, which were performed for Design 1 (graphite moderator), have shown that 200 to 300 s are required to reach first-wall melting following a total loss of coolant, even if the plasma continues to burn. Thus, there is ample time to take corrective action to prevent damage to the first wall. There is, however, a higher probability of the occurrence of lead multiplier melting; therefore, the system must be designed to prevent or be insensitive to lead melting.

Analysis of hydrogen fires has shown that little potential exists for gross facility damage. If the entire tritium supply, together with an equal amount of deuterium, were released to the containment building, the concentration would be one to two orders of magnitude below that required to sustain ignition in air (4%). The reactor building design should avoid creation of local areas where explosive concentrations of hydrogen isotopes could accumulate.

The INTOR plasma will contain approximately 0.23 GJ of thermal energy. Analyses have been performed to determine the effect of this energy during a major disruption. The analysis has shown that a melt layer of approximately 0.14 mm is formed at the hottest point during a disruption, with an energy deposition density of 289 $J \cdot cm^{-2}$ and a disruption time of 20 ms. Also, a layer of approximately 10^{-3} mm is vapourized. The end-of-life thickness of the first wall is specified to be 4 mm; therefore, it is highly unlikely that a single disruption could cause a first-wall failure.

Analysis of the effects of cryogenic system failure was performed to determine the effect on the reactor building pressure. Analysis of an accident was performed whereby the entire contents of the liquid helium inventory was released to the containment building. For this case, the overpressure of the building was about 0.3 atmosphere. Large containment buildings can be designed to withstand this overpressure; the cryogenic inventory should, however, be compartmented to prevent such large spillages.

5.3. Radioactive release and consequences

Routine releases of tritium were estimated to be in the range of from 10 to 20 $Ci \cdot d^{-1}$. Dose calculations were performed for ground-level releases and various stack heights. The results showed that even the ground-level release did not exceed the design guide value for the general public (5 mrem/year) for distances of 800 m or greater. Also, to illustrate the potential consequences of large releases, calculations were performed for a 10 g tritium release in the oxide form, for a ground-level release and for various stack heights. A 10 g release in the oxide form is considered to be a very conservative assumption for design purposes since it is expected that the installed tritium clean-up system(s) will reduce any release to the building by several orders of magnitude before release to the environment. For the ground-level release, the doses were within the 25 rem accident guide value for distances of 400 m and greater. For releases from an elevated stack, doses were less than 0.5 rem at all distances and stack heights.

No accidents were identified that would lead to large quantities of activation products being released to the environment. Estimates made of routine releases to the environment of activated reactor building gases and fluids showed that the doses from the routine activation product releases were only about 1% or less of the routine tritium releases.

6. SCHEDULE FOR DESIGN, CONSTRUCTION AND SUPPORTING RESEARCH AND DEVELOPMENT

A plausible schedule for the design and construction of INTOR can be derived from previous experience with large fusion machines and with fission reactors. Such a schedule is shown in Fig. 21. This schedule is tight, but realistic. It does not, however, allow for delays which could possibly arise from intergovernmental decision-making requirements. It is also predicated upon the assumption that the required supporting R and D programmes are expanded or initiated immediately.

A preliminary evaluation was made of the extent to which the R and D needs that were identified in Phase Zero [5] were being met by existing and planned programmes, consistent with the INTOR schedule shown in Fig. 21. The programme of any single country (Euratom is considered as one country for this purpose) was found to address about 50% of the R and D items identified in Phase Zero. Combining the programmes of two countries led to coverage of about 70% of the items. Combining the programmes of three or all four of the countries resulted in a coverage of about 80–90% of the items. It was not possible to evaluate how adequately each item was covered. Furthermore, it must be emphasized that the above percentages refer to the number of items identified in the Phase-Zero report, without accounting for the magnitude of the effort associated with each item.

7. RESEARCH AND DEVELOPMENT

The broad R and D programme required to support the design, construction and operation of INTOR was identified during the Phase-Zero Workshop [5]. These broad R and D needs were confirmed during Phase One, and to them should be added the development of the technology to support a solid tritium-breeding blanket in INTOR.

During the Phase-One conceptual design activity a number of critical uncertainties were identified which prevented firm design decisions from being taken. In most instances, it was apparent that these uncertainties could be resolved by specific, short-term R and D. Such R and D has been defined, and a list is given in Table XIX.

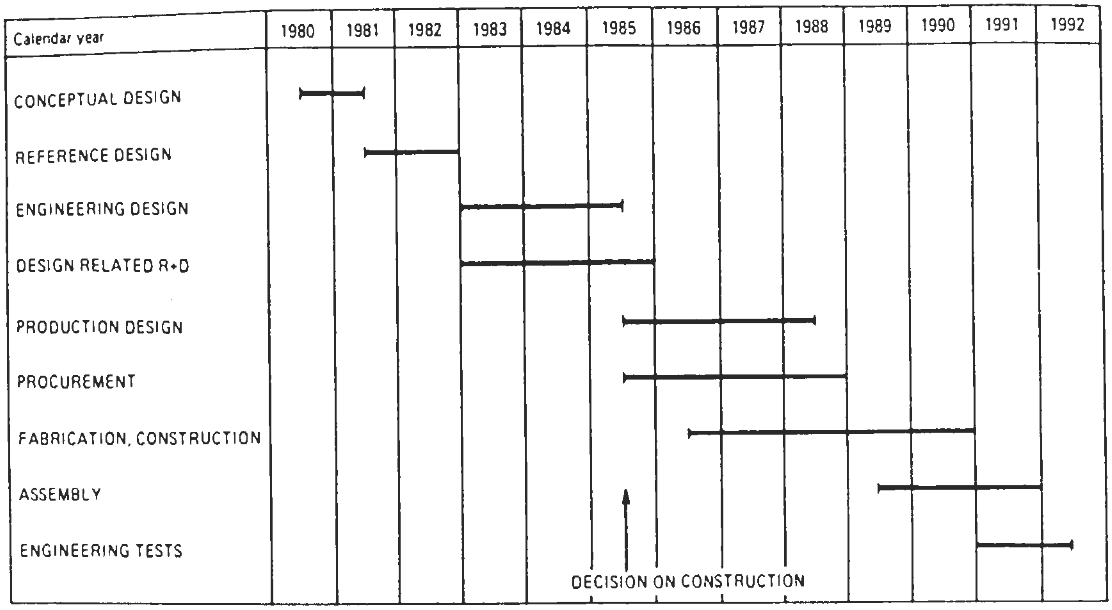


FIG.21. INTOR design and construction schedule.

8. CONCLUSIONS

A conceptual design has been defined which could, with a reasonable degree of confidence, be developed into a workable engineering design of a tokamak that meets the performance objectives for INTOR. There is some margin in the design to allow for present plasma physics and technology uncertainties, and R and D programmes are either in place or have been identified which could resolve most of these uncertainties over the next several years.

One specific objective of this design activity was to develop the design in sufficient detail to discover the real problems that lie beneath the surface. In this respect, the activity was certainly successful, because our perspective and understanding of the real problems in the design of a tokamak reactor like INTOR have changed considerably in the course of the design activity. While we have found design solutions to these problems and have developed a workable overall design, we would be the first to acknowledge that we may not yet have developed the optimal design.

The INTOR Workshop activity has now reached a stage where the major technical issues and the major design problems are inseparable. Further progress requires analysing the technical issues within the specific design context, while simultaneously trying to design around the problems. Such a procedure would involve a re-examination of early design and technical objectives decisions that have led to difficulties as the engineering design has been carried out on a more detailed level. Assecond iteration on the INTOR design, within this context and incorporating such new information as may become available from the R and D programmes, would be of great value in developing the basis for designing and constructing the next major experiment in the tokamak programme.

The INTOR Workshop has served a useful function during Phase One in identifying and focussing world-wide attention upon specific uncertainties which have a major impact upon the design of a tokamak like INTOR, and in identifying specific short-term R and D that could be undertaken to resolve these uncertainties. This identification of specific R and D needs complements the identification of broad R and D needs in Phase Zero [5].

The international workshop mode of operation has proven to have many benefits. Each party to the workshop has provided unique information and insight into the solution of design problems, which has been a major advantage of international co-operation. The workshop mode of operation has provided a critical review and auditing function for design concepts, with the result that these concepts have evolved and progressed in a way that would not have been possible within a single national design activity. Finally, the evolution of a conceptual design represents an international consensus on a plausible definition of the next major experiment in the tokamak programme.

TABLE XIX. SPECIFIC SHORT-TERM RESEARCH AND DEVELOPMENT TO SUPPORT INTOR DESIGN

		Priority [a]
Physics		
P.1.	Beta limit scaling with triangularity and asymmetry	1
P.2.	High-density divertor operation (a) Divertor channel model tests (b) High-Z impurity backflow	1
P.3.	Atomic cross-sections for plasma edge processes	2
P.4.	Ripple-induced hot-ion transport model tests	2
P.5.	Ripple-induced thermal transport model tests	1
P.6.	Disruption characterization	1
P.7.	Low start-up voltage and volt-second experiments	2
P.8.	Vertical feedback control theory and experiments	1
P.9.	Pellet fuelling studies	2
P.10.	RF heating modelling and experiments	3
P.11.	Code development for consistency checks and optimization (including transport, equilibrium, stability, poloidal field configuration, etc.)	2
P.12.	Current profile control	1
Nuclear		
N.1.	Stainless steel surface behaviour under plasma disruptions	1
N.2.	Aluminium surface behaviour under plasma disruptions	3
N.3.	Graphite tile attachment	3
N.4.	Chemical sputtering of graphite	3
N.5.	In-situ recoating	3
N.6.	First-wall grooves	3
N.7.	Solid breeder development	1
N.8.	Liquid breeding material development	3
N.9.	Solid breeder/coolant tube interfaces with predictable thermal conductance	2
N.10.	Tritium permeation	1
N.11.	Chemical sputtering of refractory metals (tungsten and molybdenum)	1
N.12.	High-temperature fatigue of tungsten and molybdenum	1
N.13.	Behaviour of brazing under thermal cycle conditions	1
Engineering		
E.1.	Fatigue and fracture mechanics data for room-temperature and 4 K materials	1

[a] Priority
1 – Required for the reference INTOR design, highest priority.
2 – Required for the reference INTOR design, secondary priority.
3 – Not required for the reference INTOR design, but of importance for other design options.

9. RECOMMENDATION

The INTOR activity is technically ready to proceed into the next phase, on the basis of progress to date. Appropriate activity in the first year, or so, of the next phase should include parallel efforts: (1) to review critical technical issues and objectives in order to optimize the design concepts; and (2) to develop the design in greater thoroughness.

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