OVERVIEW

INTOR

U.S. CONCEPTUAL DESIGN CONTRIBUTION TO THE INTOR PHASE 1 WORKSHOP

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A conceptual design of a first-generation tokamak engineering test reactor is presented. The objectives, testing mission, and role in the fusion program of such a device are discussed. The physics, nuclear, and engineering design are described, and the projected operational schedule is outlined. An estimate of the cost and design/construction schedule is presented.

I. INTRODUCTION

The International Tokamak Reactor (INTOR) Workshop is a unique collaborative effort among the U.S., the USSR, Euratom, and Japan to define the characteristics and objectives of, assess the technical feasibility of, and develop a design for the appropriate next major experiment in the worldwide tokamak program after the generation of large tokamak experiments presently under construction. The INTOR Workshop is conducted through the auspices of the International Atomic Energy Agency (IAEA).

The Zero Phase of the INTOR Workshop was conducted during 1979. Each of the four participating "countries" 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 critically discuss the contributions of the four parties, and to prepare the report of the Workshop. The bulk 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 the Workshop sessions. This home country effort involved more than 100 of the leading magnetic fusion scientists and engineers (~15 to 20 man-years of effort) in each of the four participating countries. The participants from each country met in Vienna four times, for a total time of ten 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 the tokamak fusion test reactor (TFTR), the Joint European Torus (JET), T-15, JT-60] in the worldwide tokamak program and to assess the technical feasibility of constructing this experiment to operate by about 1990. Detailed assessments of the plasma physics and technology bases for such an INTOR experiment were developed on a worldwide basis, and physical characteristics were identified that were consistent with this technical basis and with the general objective of INTOR being the maximum reasonable step beyond the next generation of large tokamaks.

Each country submitted detailed reports to the Zero Phase Workshop, which were subsequently published.[1-4] The distilled results of these national contributions were combined into a joint report.[5] This report of the Zero Phase Workshop, which represents a technical consensus of the worldwide magnetic fusion community, concludes that the operation by the early 1990s of an ignited, deuterium-tritium

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(D-T)-burning tokamak experiment that could serve as an engineering test facility is technically feasible, provided that the supporting research and development (R&D) activity is expanded immediately. 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 1, the Definition Phase, in early 1980. The objective of the Phase 1 Workshop is to develop a conceptual design of the INTOR experiment. The first six months of the Phase 1 Workshop were devoted to the resolution of critical technical issues that remained unresolved at the end of the Zero Phase. The U.S. contributions to this part of the Workshop have been published.[6]

The INTOR conceptual design has been carried out by design teams working in the home countries (20 to 40 man-years of effort in each country). 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 country) of these design teams have periodically met together at workshop sessions in Vienna (for a total of about 13 weeks during Phase 1) to define the tasks of the home design teams, to review the ongoing design work, and to take decisions on the evolving design. The decisions taken at each workshop session were then incorporated into each national design activity, so that the four national design contributions have progressively converged toward a single design, at an increasingly greater degree of detail, during the course of the 1-yr (July 1980 to June 1981) conceptual design activity.

This paper documents the U.S. contribution to the Phase 1 INTOR Workshop conceptual design activity. A joint conceptual design report will be prepared and published by the IAEA.

The INTOR Workshop has played a major role in identifying and focusing the attention of the world fusion community on the major problems that must be addressed before the next major experiment in the tokamak program can be undertaken. The Workshop has also made a major contribution in developing a consensus on the most likely solutions to these problems. An important new task that was assigned the Workshop at the meeting of the International Fusion Research Council (IFRC) in January 1981 is the definition of specific R&D efforts, including identification of institutions in the participating countries where these efforts could be carried out, that could resolve the major uncertainties that have been identified in the Phase 1 INTOR conceptual design. This IFRC initiative in agreeing for the major fusion countries to work together to resolve these problems may lead to a major international collaboration in the development of magnetic fusion power.

The IFRC of the IAEA, which supervises the INTOR Workshop, has recommended that the Workshop be extended into Phase 2A, Design, for the period July 1981 through June 1982.

II. ROLE OF INTOR IN FUSION PROGRAM

INTOR is viewed as the major experiment in the tokamak program 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:

1. production of several hundred megawatts of electricity and achievement of net electrical power production
2. production of tritium in the blanket, with a net breeding ratio greater than unity
3. demonstration of the development and integration of full-scale components that are extrapolatable to a commercial reactor
4. demonstration of component and system reliability, availability, and lifetime at a level that would be acceptable for a commercial reactor
5. demonstration of safe and environmentally acceptable fusion reactor operation that would satisfy the requirements for a commercial reactor
6. demonstration of commercial feasibility, although the DEMO would not itself be economically competitive.

The role of INTOR in the fusion program can be defined by identifying the physics and technology prerequisites for the design and construction of the DEMOs and then identifying which of these prerequisites can best be satisfied by INTOR and which can best be satisfied by complementary physics experiments and technology test facilities. The broad, general prerequisites for the design and construction of DEMOs are as follows:

1. development of an adequate plasma physics and engineering data base for prediction of the performance of the DEMOs
2. demonstration of the plasma physics performance of the DEMOs
3. development of fusion reactor components
4. investigation of component integration into an overall fusion reactor system
5. investigation of fusion reactor maintainability
6. investigation of component and overall reactor system reliability, at least to some significant fraction of the availability and design lifetime of the DEMOs

7. investigation of electricity and tritium production by fusion

8. investigation of the safety and environmental characteristics of a fusion reactor.

An extensive plasma physics experimental and theoretical program 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 toward a tokamak DEMO and is intended to demonstrate the achievement of essentially all 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 burns with optimized plasma parameters. Achievement of these objectives requires satisfactory impurity control, power balance control, and profile control for parameter optimization. A closely related objective is the achievement of high (⩾70%) duty 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 program will be required in the development of fusion power reactors to the demonstration reactor stage. This program 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, component design concepts, etc., will be carried out in test facilities and that components will be developed and tested under conditions that at least partially simulate a fusion reactor environment in test facilities, prior to the final design and construction of INTOR. INTOR will then serve principally to

1. investigate the compatibility of components within an integrated reactor system

2. investigate the remote maintainability of a fusion reactor system

3. test components and materials in a fusion reactor environment

4. investigate the reliability of components under sustained operation[a] in a fusion reactor environment

5. irradiate materials samples to moderate fluences in a fusion neutron spectrum

6. investigate the production of electricity and tritium in a fusion reactor

7. investigate 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 are readily extrapolatable 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, there is a high degree of commonality among the technologies required for the principal magnetic confinement concepts.

III. INTOR OBJECTIVES

The objectives of INTOR follow from the foregoing considerations of its role in the fusion program and from an assessment of the technical basis that 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 as follows:

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

2. INTOR should demonstrate the plasma performance required for the tokamak DEMOs.

3. INTOR should investigate the development and integration into a reactor system of those technologies required for the DEMOs.

4. INTOR should serve as a test facility for blanket, tritium production, materials, and plasma engineering technology development.

5. INTOR should investigate fusion reactor component reliability.

6. INTOR should investigate the maintainability of a fusion reactor.

[a]Under sustained operation to some significant fraction of the component design lifetime against the limiting phenomenon (e.g., neutron damage, number of pulses) for that component.

TABLE I

INTOR Technical Objectives

A. Reactor-Relevant Mode of Operation

1. Ignited D-T plasma
2. Controlled >100-s burn pulse
3. Reactor-level particle and heat fluxes ($P_n \gtrsim 1$ MW/m^2)
4. Optimized plasma performance
5. Duty cycle $\gtrsim$70%
6. Availability 25 to 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 fueling
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 to 10 MW(electric)
6. Fluence ~5 MW·yr/m^2 during Stage III for component reliability and materials irradiation testing

7. INTOR should investigate the factors affecting the reliability, safety, and environmental acceptability of a fusion reactor.

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

IV. DESIGN DESCRIPTION

IV.A. Overview

A conceptual design has been developed for a device that could fulfill the objectives listed in the previous section. A credible engineering design has been developed that is based on a realistic anticipation of the status of plasma physics research and technology development a few years from now. 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 understood.

The major features for the U.S. INTOR conceptual design are specified in Table III.

TABLE II

Staged Operation Schedule

Stage	Number of Years	Emphasis	Availability (%)	Annual 14-MeV Neutron Fluence (MW·yr/m^2)[a]	Annual Tritium Consumption (kg)
IA	1	Hydrogen plasma operation, engineering checkout	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[b] testing	50	0.62	13.8

[a]At the outboard location of the test modules.
[b]The objective is to achieve ~5 MW·yr/m^2 within ≤10 years after the end of Stage II. This could be achieved in several ways; the case given here is only representative.

TABLE III

INTOR Design Specification

Geometric	
Chamber major radius, R (m)	5.2
Chamber volume (m^3)	320
Chamber surface area (m^2)	380
Plasma	
Plasma radius, a (m)	1.2
Plasma elongation, κ	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, $\langle T_i\rangle$ (keV)	10
Average ion density, $\langle n_i\rangle$ (10^{20} m^{-3})	1.4
Energy confinement time, τ_E (s)	1.4
Plasma current, I (MA)	6.4
Field on axis, B_t (T)	5.5
Safety factor (separatrix), q_I	2.1
Thermonuclear power, P_{th} (MW)	620
Neutron wall load, P_n (MW/m^2)	1.3
Operation	
Burn time (s), Stage I/Stages II and III	100/200
Duty cycle (%), Stage I/Stages II and III	70/80
Number pulses lifetime (10^5)	7
Maximum availability goal (%)	50
Heating–Neutral Beam	
Number of injectors (active/redundant)	4/1
Beam power (MW)	75
Beam energy (keV)	175
Pulse length capability (s)	10
Fueling	
Method	Pellet injectors and gas puffing
Burn Control	
Method	Variable field ripple
Impurity Control	
Method	Single-null poloidal divertor
Collector	Tungsten attached to H_2O-cooled stainless steel heat sink
Power to divertor (MW)	80
First Wall	
Material	H_2O-cooled Type 316 stainless steel
Power to first wall (excluding neutrons) (MW)	44
Thickness	
Outboard (mm)	13.4
Inboard (mm)	15.6
Lifetime (years)	15 (full)
Breeding Blanket	
Material	H_2O, Type 316 stainless steel Li_2SiO_3, Pb, C
Breeder temperature (°C)	400 to 600
Thickness (m)	0.5
Location	Outboard and top
Breeding ratio	0.65
Tritium extraction	Continuous helium purge
Tritium inventory (kg)	1
Tritium Fuel System	
Tritium flow rate (g/h)	63
Tritium inventory (kg)	2.4
Annual tritium consumption at 25% availability (kg/yr)	7
Isotopic enrichment	Cryogenic distillation
Torus Vacuum System	
Initial base pressure (Torr)	10^{-7}
Pre-shot base pressure (Torr)	3×10^{-5}
Pumps	Compound cryosorption
Pumping	Through divertor chamber
TF Coils	
Number	12
Bore (m)	7.75 X 10.7
Conductor	Nb_3Sn, NbTi
Stabilizer	Copper
Maximum field (T)	10.8
PF Coils	
Total flux (V·s)	110
Location	External to TF coil
Conductor	NbTi
Maximum field (T)	7.5
Breakdown Coils	
Breakdown voltage (V)	100
Location	Internal TF coil
Conductor	Copper

(Continued)

TABLE III (Continued)

Electrical Energy Storage	
Number MGF units	6
Total energy storage capacity per generator (GJ)	4
Peak load capacity per generator (MVA)	750
Mechanical Configuration	
Twelve blanket sectors assembled with straightline horizontal motion through windows between TF coils.	
Semipermanent inboard, upper, and lower shield forming primary vacuum boundary on inner surface.	
Final closure of primary vacuum boundary on outer boundary of blanket sectors.	
Mechanical Configuration (continued)	
Test modules inserted horizontally at midplane.	
All superconducting coils in a common cryostat.	
Dedicated sectors:	5 NBI 2 Fueling 3 Testing 2 I&C
Shielding	
Inboard (blanket + shield) (m)	0.85
Outboard (blanket + shield) (m)	1.55
Neutral beam drift tube (m)	1.0
NBI box	0.5 to 0.75

A self-consistent analysis of the magnetics, magnetohydrodynamics (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(thermal). 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 needed to exhaust helium and to prevent heavy impurities from reaching the plasma to achieve the 200-s burn time, which was set at about one-fifth the theoretical magnetic surface diffusion time. Based on the present best estimate of energy confinement and transport losses, the predicted alpha-heating power exceeds that required for ignition by a factor of ~2, and the 75 MW of neutral beam heating power allows the plasma to be heated to ignition for transport losses up to about twice as large as presently estimated.

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 a few centimetres with coils external to the toroidal field (TF) coils. A relatively short channel length (~30 cm) is adequate because of the high-density mode of divertor operation. A divertor collector plate with tungsten tiles mechanically attached to a H_2O-cooled stainless steel heat sink has been designed so that the tiles can be replaced every several years, as necessary. The heat sink should last the design lifetime of INTOR. The analysis in support of the single-null divertor was certainly the most extensive that has been performed for any impurity control scheme; it included self-consistent treatments of the magnetics for separatrix control and 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 semipermanent inboard, upper and lower shield forms the primary vacuum boundary. Twelve blanket sectors fit within this semipermanent shield. These blanket sectors are partially (outboard and upper) tritium-producing blanket and partially (inboard) 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 blanket sector can be withdrawn horizontally with straightline motion through a "window" between adjacent TF coils. The divertor channel is broken up into 24 modules that are removable with straightline horizontal motion between the TF coils. The single-null divertor was chosen over the double-null divertor to achieve a relatively simply maintainable mechanical configuration.

Semipermanent, superconducting TF and poloidal field (PF) coils will be enclosed in a common, semipermanent cryostat, thus completely separating the cold and warm structures. All PF coils will be superconducting and external to the TF coils except for a set of resistive coils internal to the TF coils that provide the voltage pulse for plasma breakdown.

The rather demanding structural requirements for the TF coils are met by a combination of design strategems. Coil wedging, intercoil support structure, and a bucking cylinder are used to handle in-plane and centering forces. Gussets, intercoil support structure, a ring girder, the bucking cylinder, and shear ties will be used to handle out-of-plane forces and 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. An H_2O-cooled, stainless steel first wall with a panel-type construction is specified. This first wall is predicted to last the full lifetime of the device, provided that the melt layer that 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 to reduce the operational cost. A solid breeder (Li_2SiO_3) blanket that covers the outboard 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 the most extensive three-dimensional fusion reactor shielding calculation that has been performed to date.

The availability goal for INTOR is 50% during the last stage of operation. Reliability analyses based on 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 programs. Extrapolation of present reliability data leads to availability estimates of ~30 to 40%, depending on the degree of redundancy.

In retrospect, a relatively few factors can be identified that had a major influence on the design that was developed for a device that could satisfy the INTOR objectives. 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 TF coils than otherwise would be necessary, imposed certain constraints on the TF coil structural support system, and led to the choice of a single-null (rather than a double-null) poloidal divertor (which imposed additional requirements on the PF coil system). The requirement of maintainability led to a choice of all-external PF coil system, which produced large overturning moments on the TF coils and thus led to additional requirements on the structural support system. The requirement for personnel access for maintenance at the outer boundary of the reactor led to considerably more outboard shielding that would be necessary for component protection. The decision to specify high TF strength to hold down the size led, in the presence of external poloidal coils, to additional requirements on the structural support system. The objective of reaching an appreciable fraction of the design lifetime neutron fluence for first wall and blanket components in a DEMO had a significant influence on the specification of neutron wall load and availability goal, on the inboard shield thickness, and on the design of the first wall and magnet structural system against fatigue and crack growth limits.

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

IV.B. Physics Basis

The overall INTOR size and magnetic field parameters have been established as part of the INTOR transport modeling program. These studies, based on the transport coefficients specified in Phase Zero [$\chi_e = 5 \times 10^{17}/n_e$ (cgs); $\chi_i = 3 \times$ Neoclassical], have established the INTOR ignition margin and overall INTOR performance. These coefficients are considered to be conservative since recent tokamak experiments indicate that a more realistic electron energy transport coefficient would be 30 to 40% lower. In addition, high power heating experiments suggest an additional reduction in transport with increasing temperature. Figure 1 shows a contour plot of the power required for steady-state INTOR operation as a function of average plasma density and temperature. The path to ignition and the final operating point are shown in the figure. Ignition is obtained for a beta between 3 and 4%. The specific wall loading is reached at a beta of ~5.6%. At this

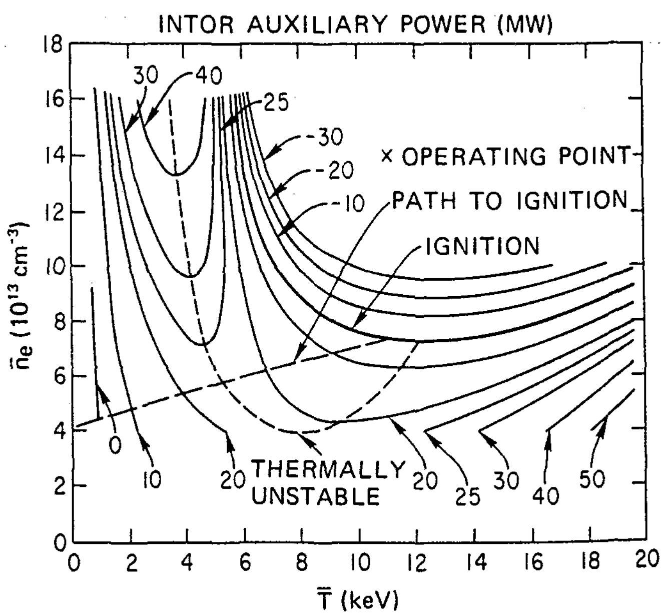


Fig. 1. The heating power required to maintain the INTOR discharge at steady state as a function of average temperature and density is shown. Also shown is the approximate path to ignition and final operating point.

value of beta, the predicted alpha-heating power exceeds that required for ignition by a factor of ~2, taking into account the accumulation of helium in the plasma. Present experiments indicate that these beta values will be realized if vertical elongation produces the theoretically predicted improvement in the critical beta. To obtain these beta values, the INTOR plasma is vertically elongated by 1.6 with a triangularity of ~0.3. The path to ignition is chosen to minimize the required heating power. The specified neutral injection heating power level of 75 MW is sufficient to cover the thermal power losses and changes in heat capacity, with at least a 30% contingency for unexpected losses. The specified neutral beam energy of 175 keV is sufficient to heat the plasma, and, in fact, beam energies as low as 150 keV appear to be adequate. The reduced energy requirement is a consequence of the high-beta outward shift in the magnetic axis, which improves the neutral beam penetration. Recent analysis of the effect of TF ripple on ion energy transport indicates that the present ripple requirement (i.e., the TF ripple must be <0.75% at the plasma edge) is overly conservative and that increased ripples would be acceptable. The INTOR TF coils produce only 0.3% ripple at the plasma edge, and therefore the design is not ripple limited. The operating point is well above the ignition beta; therefore, steady-state operation for this condition will require burn control to provide thermal equilibrium and a stable burn. Variable TF ripple has been chosen as the burn control mechanism. The $N = 6$ ripple produced by decreasing the current in alternate TF coils has good penetration to the magnetic axis. About 1.5% of $N = 6$ ripple should be adequate to provide thermal equilibrium at the operating point. This ripple, combined with the passive stabilization resulting from radial expansion, will produce a stable operation condition.

The total burn cycle of 245 s will include startup, burn, shutdown, and dwell phases. The ohmic heating (OH) coil produces 100 V per turn for 0.1 s to initiate the discharge. The vacuum vessel, with a time constant of 50 ms, reduces the voltage at the plasma. Studies indicate that 50 V per turn may be adequate to initiate the discharge, although only under very clean conditions. The addition of 10 MW of radiofrequency (rf) heating during current initiation could be used to reduce the voltage requirement if this proves necessary. During startup, the plasma current will be increased to 5.4 MA before neutral injection heating is applied. The heating will produce ignition in ~4 s with an associated increase in current of ~1 MA. The divertor will be operational at the beginning of the heating phase. The 200-s burn phase is about ten times as long as the alpha ash buildup time, thereby requiring a helium removal capability. However, the pulse length is less than one-fifth the plasma magnetic skin time, thereby reducing the problems associated with equilibrium evolution. At the end of the burn phase, subignition will be produced by the burn control system, and the plasma current will be reduced to zero in ~5 s. The rate of discharge termination will probably be limited by the necessity to reduce the plasma density to avoid disruptions. Twenty seconds has been allowed to reset the electrical and vacuum conditions for the next pulse.

The first wall loading is a combination of particle and radiation loads that are relatively continuous during the discharge, along with disruption-induced thermal transients. The total alpha heating power of 124 MW must be dissipated on the first wall and inside the divertor chambers. The INTOR divertor chambers are designed to handle 80 MW of this power, while 44 MW falls on the first wall in the form of charge-exchange and radiation. Based on the PDX and ASDEX experiments, we have concluded that the divertor load should divide about equally between the inside and outside divertor chambers. During disruptions, 220 MJ is assumed to impinge on the top, bottom, or inside wall with a frequency of 5×10^{-3} during initial operation, and 10^{-3} during the later phases of the INTOR program.

Recent INTOR physics studies have focused on the magnetic design, with particular emphasis on the divertor geometry. These studies have resulted in a self-consistent plasma equilibrium, PF coil, and mechanical configuration. Figure 2 shows the high-beta magnetic configuration. The PF coils are all external to the TF coils and carry a total current of ~100 MAT. The stored energy in the PF system is ~17 GJ. A maximum power of 4.5 GW is required during the heating phase. The magnetic configuration features a single-null divertor with an open geometry and pumping only on the outside divertor channel. Studies indicate that the single-null divertor will require ~20 cm more scrape-off width at the small major radius side of the discharge than the double-null divertor. However, the single-null configuration was chosen to minimize overall space requirements. The fraction of the diverted plasma that is pumped is limited to that required to exhaust the helium ash (~5%). The remaining neutrals refuel the plasma via the divertor throat. The tritium burnup for this mode of operation should be ~4%. Since the ionization probability for the neutrals inside the divertor channel is very high, the large neutral fueling and plasma flows are confined to this channel. The plasma flows along magnetic field lines carrying mostly particles ionized inside the divertor to the divertor plate at about the sound speed, while plasma enters the divertor throat at a speed that is much smaller than the sound velocity. This low throat velocity results in low plasma density gradients inside the main chamber and therefore high edge

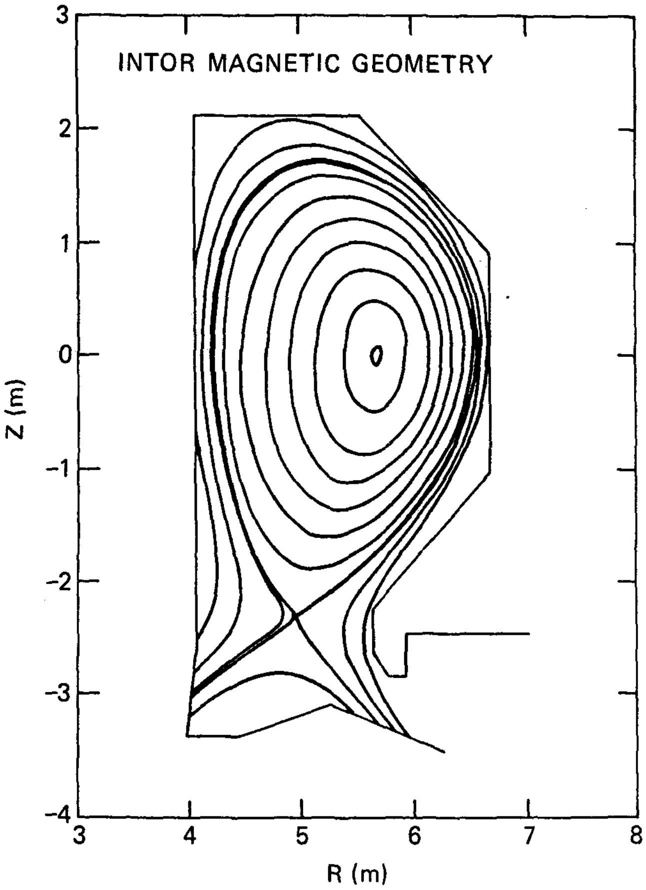


Fig. 2. INTOR poloidal flux surfaces for single-null divertor.

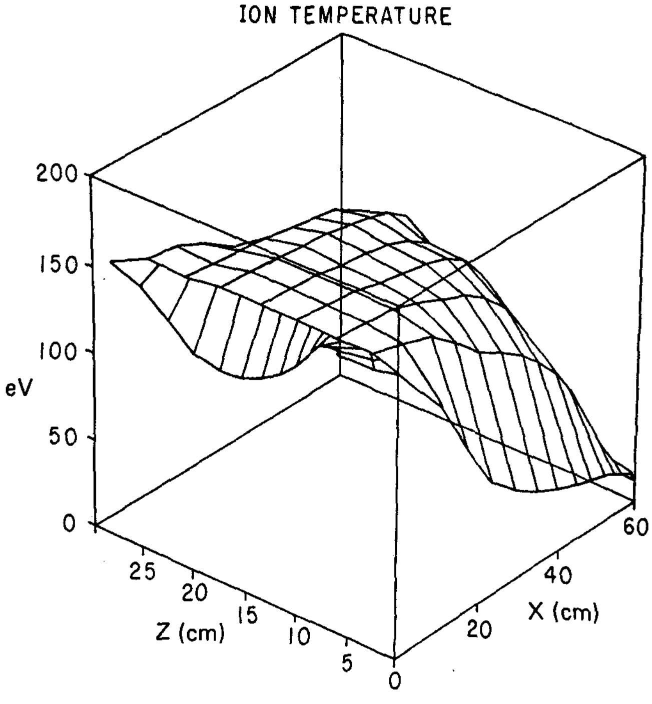


Fig. 3. The ion temperature profile inside the divertor channel is shown. Here, Z measures the channel width and X the channel length, with the divertor plate at $X = 60$ cm.

density. This high edge density lowers the edge temperature and associated material erosion.

Divertor modeling studies have focused on the performance of the high-density PF divertor. These studies have shown that the neutrals are confined to the divertor channel, thereby limiting the charge-exchange erosion of the first wall. Figure 3 shows the ion temperature distribution inside the INTOR divertor. The strong cooling produced by the neutrals reduces the ion temperature from 160 eV at the divertor throat to 22 eV outside the potential sheath at the divertor plate. The impurities eroded from the divertor plate have a very low probability of entering the main discharge as neutrals. In addition, when the impurities are ionized, they are swept toward the divertor plate by the electric field and the plasma flow. The reduction in plasma contamination will facilitate the use of tungsten divertor plates with an increased operational life. The confinement of the charge-exchange erosion to the divertor chamber and the greatly reduced impurity contamination are the primary reasons for choosing the divertor option.

The INTOR physics specifications form a conservative basis for the engineering design. Physics R&D during the initial phases of the engineering design should increase our data base and reduce some of the design constraints.

IV.C. 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 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 influence 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. Refer to Figs. 4 and 5 for elevation and plan views of the tokamak.

IV.C.1. TF Coil Design—Access Requirements

The most significant configuration design feature is the access provided for torus maintenance. The 12 TF coils have been sized with sufficient outside

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 lifetime of the device. Failure of these systems would require a major shutdown of the facility (six months to one year) for repair or replacement. Superconducting TF and PF coils, the inboard portion of the torus shield, and several major support structures have been identified as systems of this type and are designated as semipermanent installations.
3. Sufficient radiation shielding will be provided in the torus and around penetrations to limit activation of components exposed to the reactor cell. "Hands-on" maintenance will be considered for normal operations when the torus internals are not removed, and 2.5 mrem/h will be specified as the maximum dose rate anywhere in the cell after 24 h of shutdown.
4. All systems will be designed for fully remote maintenance to cover cases of emergency.

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.

IV.C.2. All-External PF Coil System

To simplify the INTOR magnetic configuration, essentially all the PF coils have been placed outside the bore of the TF coils. The PF coils can therefore all be superconducting since mechanical joints are not required for assembly. Only a small resistive coil system is placed inside the TF coil bore to provide the initial plasma breakdown voltage. All PF coils have been located above and below the TF opening where the torus sectors are removed.

IV.C.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 within the common cryostat. With this feature, access to the torus is maintained without penetration of the cryogenic vacuum boundary.

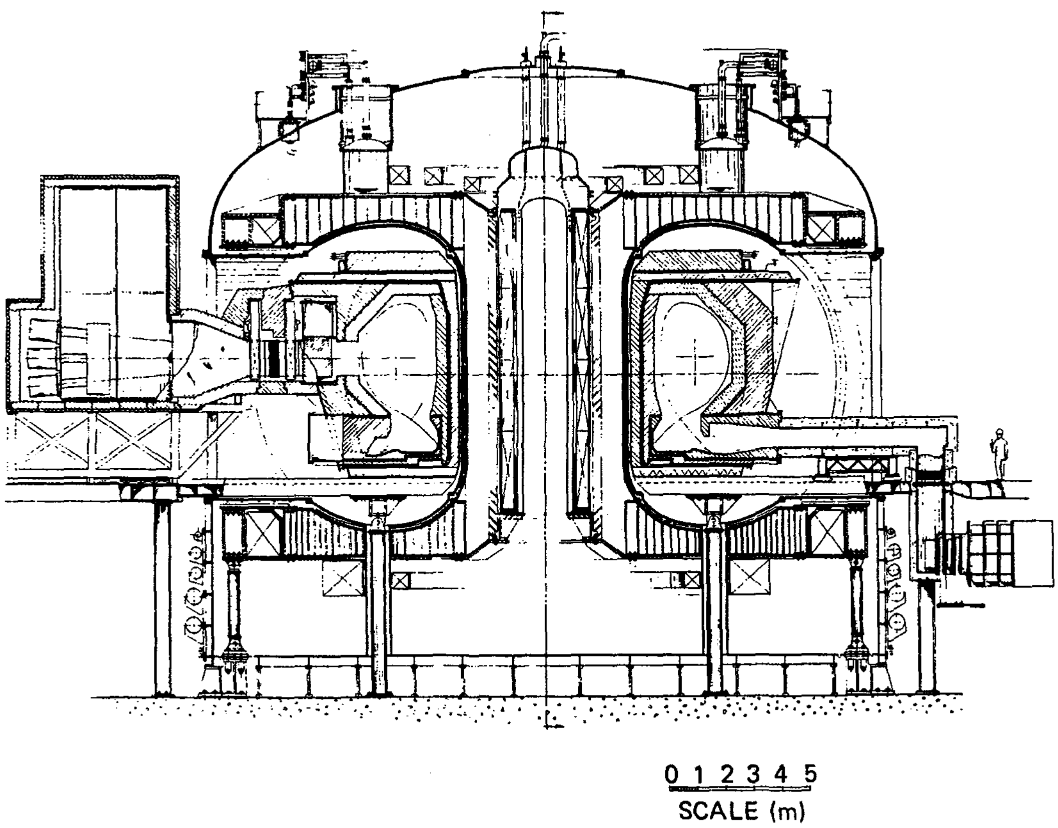


Fig. 4. INTOR elevation view.

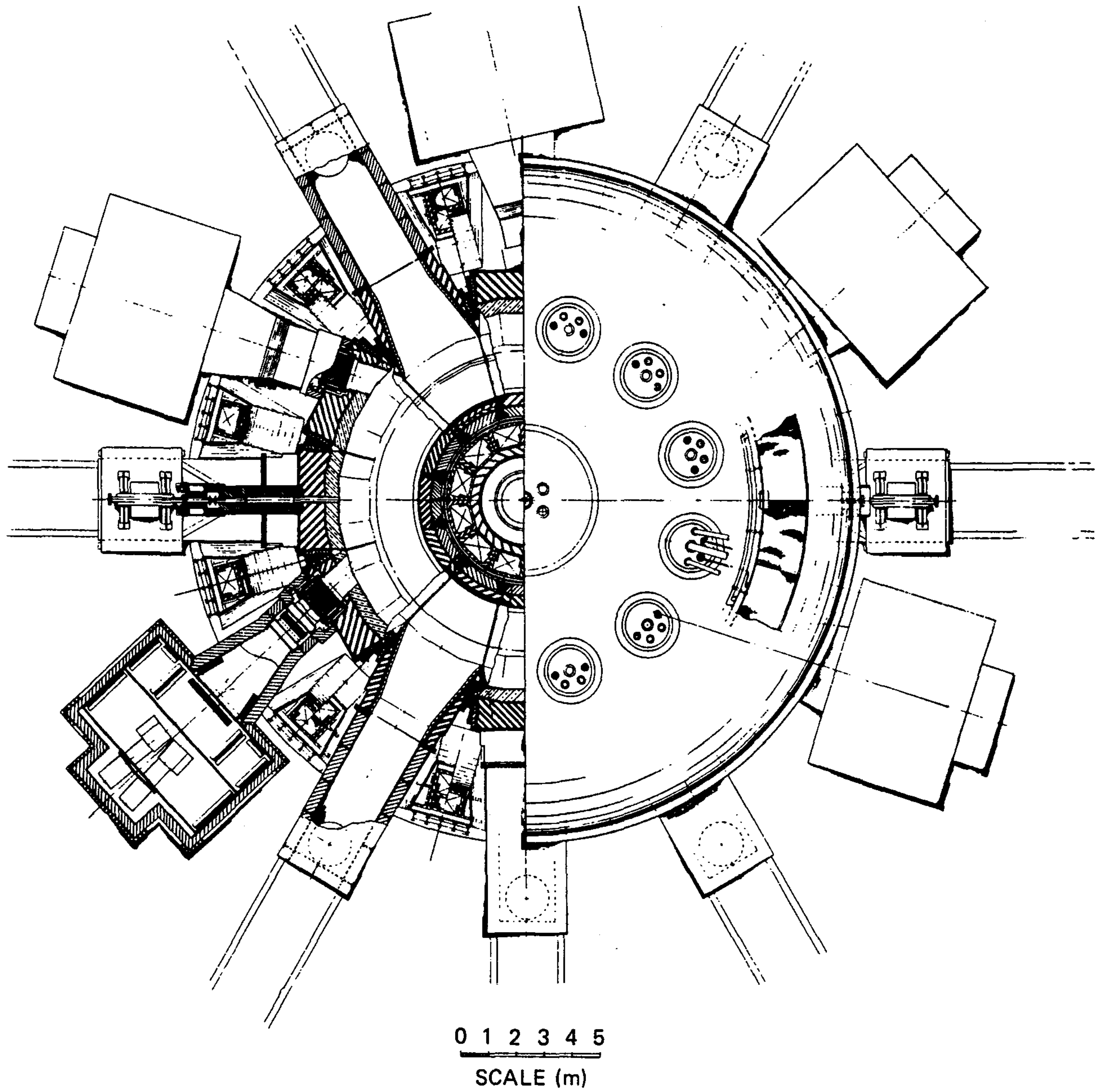


Fig. 5. INTOR plan view.

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.

IV.C.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 semipermanent shield and removable sectors (Fig. 6). The components exposed to the most severe damage from particle and heat loads (first wall and blanket regions) have been combined into a sector that 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 semipermanent shield modules, forms the primary vacuum boundary and is not removed for normal planned maintenance procedures.

IV.C.5. Single-Null Poloidal Divertor

The divertor collector is the most severely damaged torus component and hence must include provisions for frequent repair. Modular divertor sectors have been designed that can be removed in 24 pieces similar to the main torus sectors. A single-null

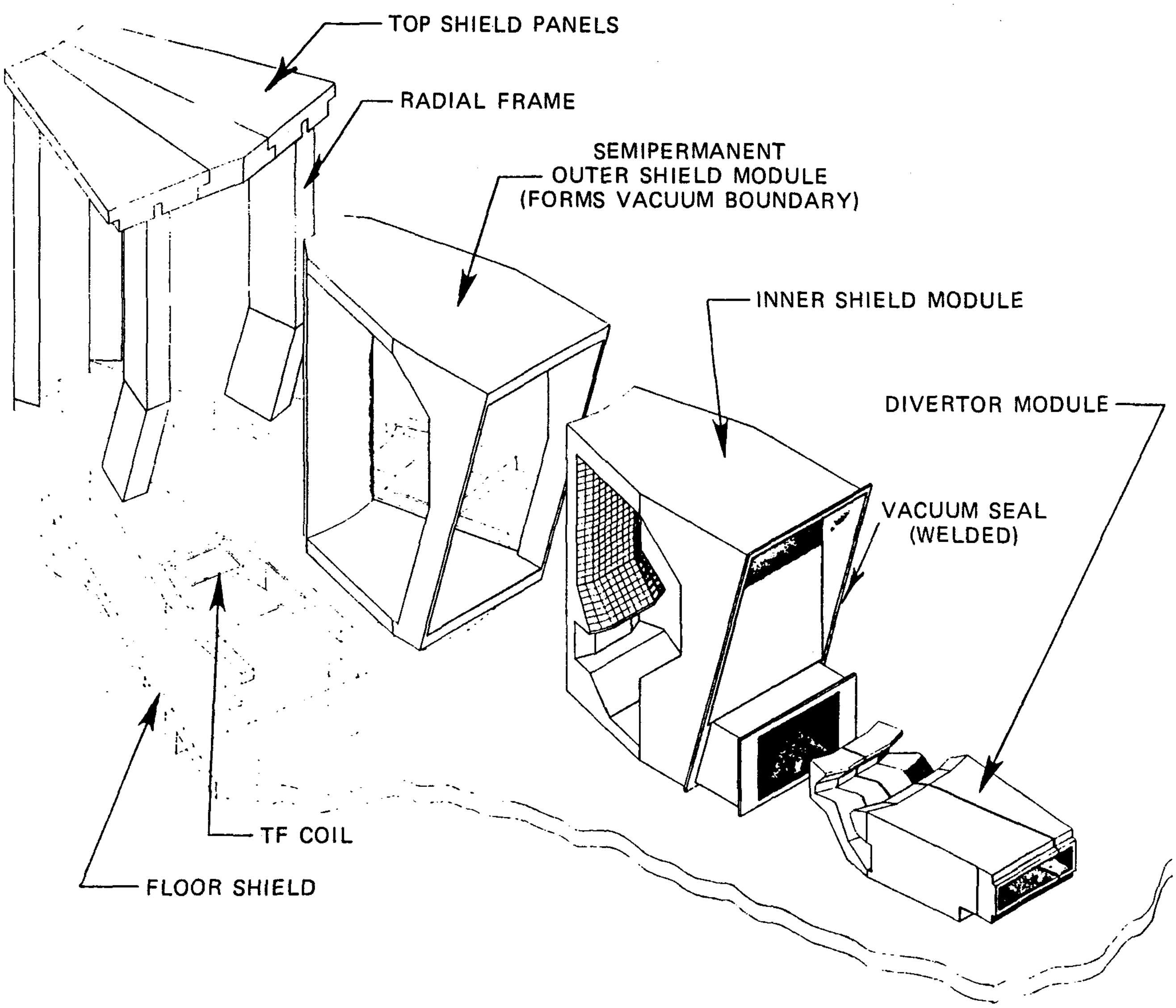


Fig. 6. Torus sector assemblies.

divertor was selected to maintain sufficient access for removal of the modules and for vacuum pumping ducts. A double-null divertor was found to severely limit access to the torus with the addition of divertor collectors and pumping at 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.

All torus vacuum pumping is provided by the divertor pump duct. Only seven pump ducts are provided to avoid interference with access to the neutral beam lines at the other five locations.

IV.C.6. *Structural Support System*

A major design effort has gone into the structural configuration. The all-external PF coils impose very large pulsed fields 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 that maintains the access space for torus sector removal (Figs. 7 and 8).

The structural design was verified by a three-dimensional finite element model using the NASTRAN computer code. The model included a representation of all 12 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 support has been placed entirely at the outside of the machine to provide access to the bottom of the machine (see Fig. 4).

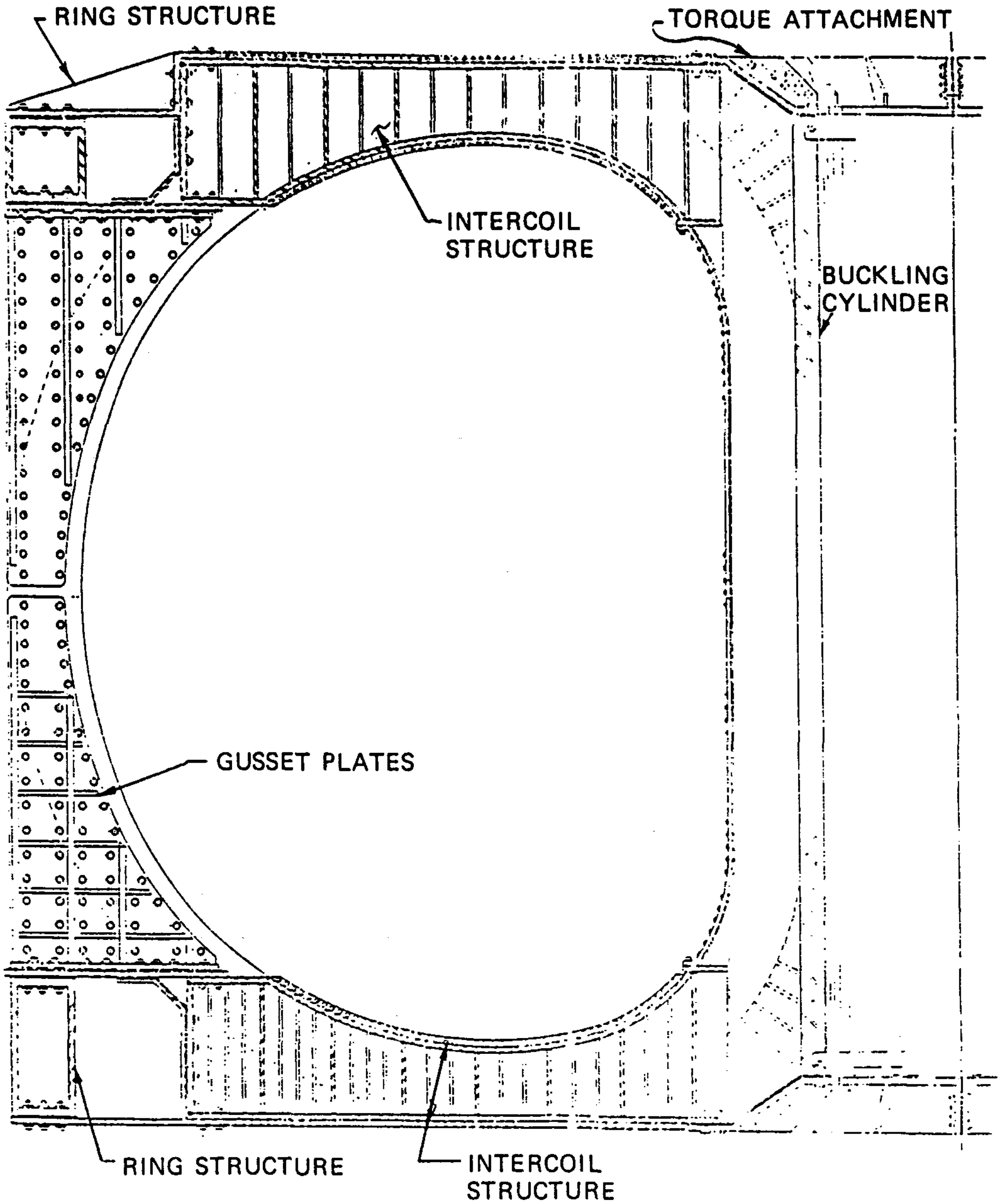


Fig. 7. TF coil support structure (elevation view).

IV.C.7. Tokamak Radial Build

The major dimensions of the INTOR design are summarized in Fig. 9. The radial build dimensions show space allocation for all components as well as the required gaps for assembly tolerance.

IV.C.8. Dedicated Torus Sectors

The facility layout is based on the concept of dedicated bays—the region between adjacent TF coils. 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 dedicated to the pellet injectors for fueling. The final two bays are tentatively dedicated to instrumentation, diagnostics, and control. This approach provides a straightforward interface between the tokamak device and the facility. The dedicated sector allocations are shown in Fig. 10.

IV.D. Magnet and Electrical Systems

IV.D.1. Magnet System

The magnet system consists of the superconducting TF, OH, and electromagnetic field (EF) coils, and a small resistive coil set inside the TF coil

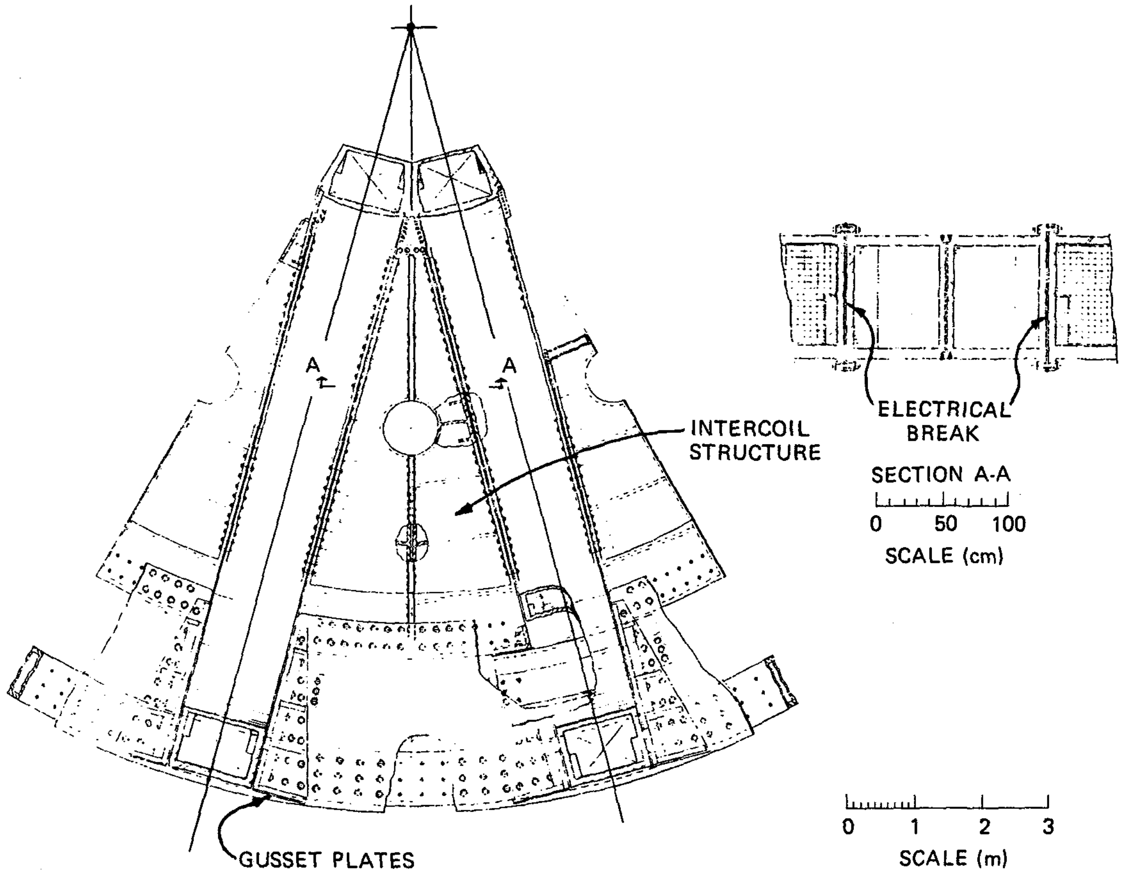


Fig. 8. TF coil support structure.

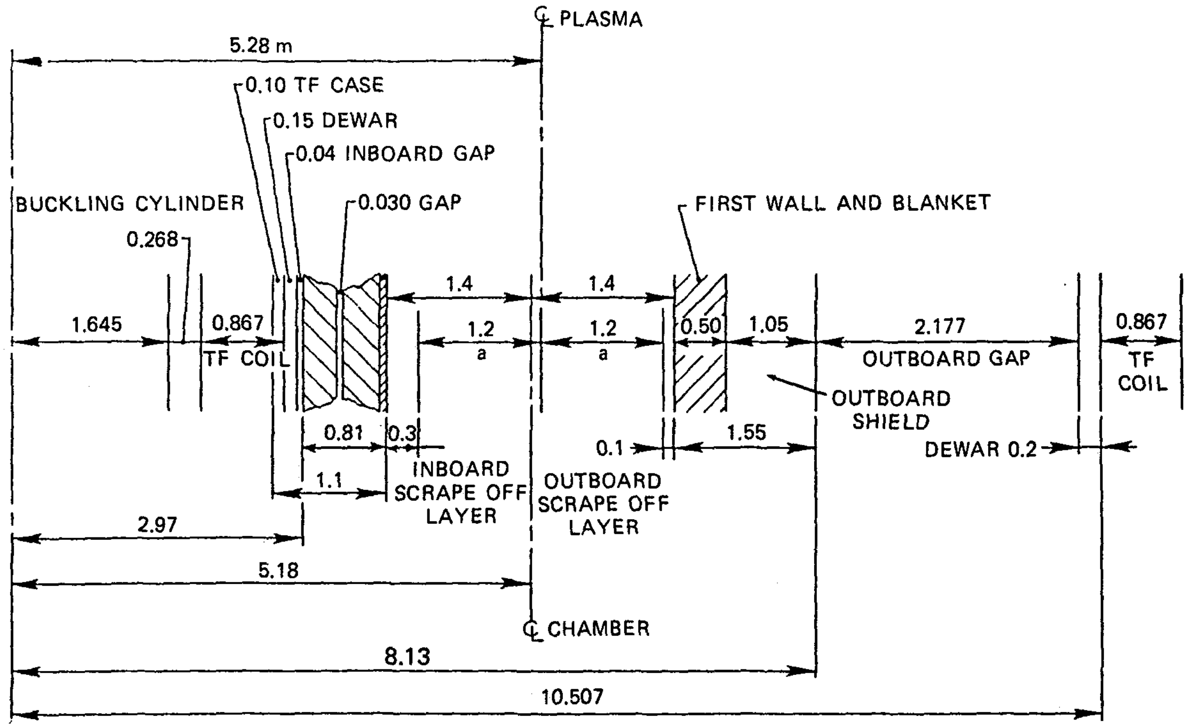
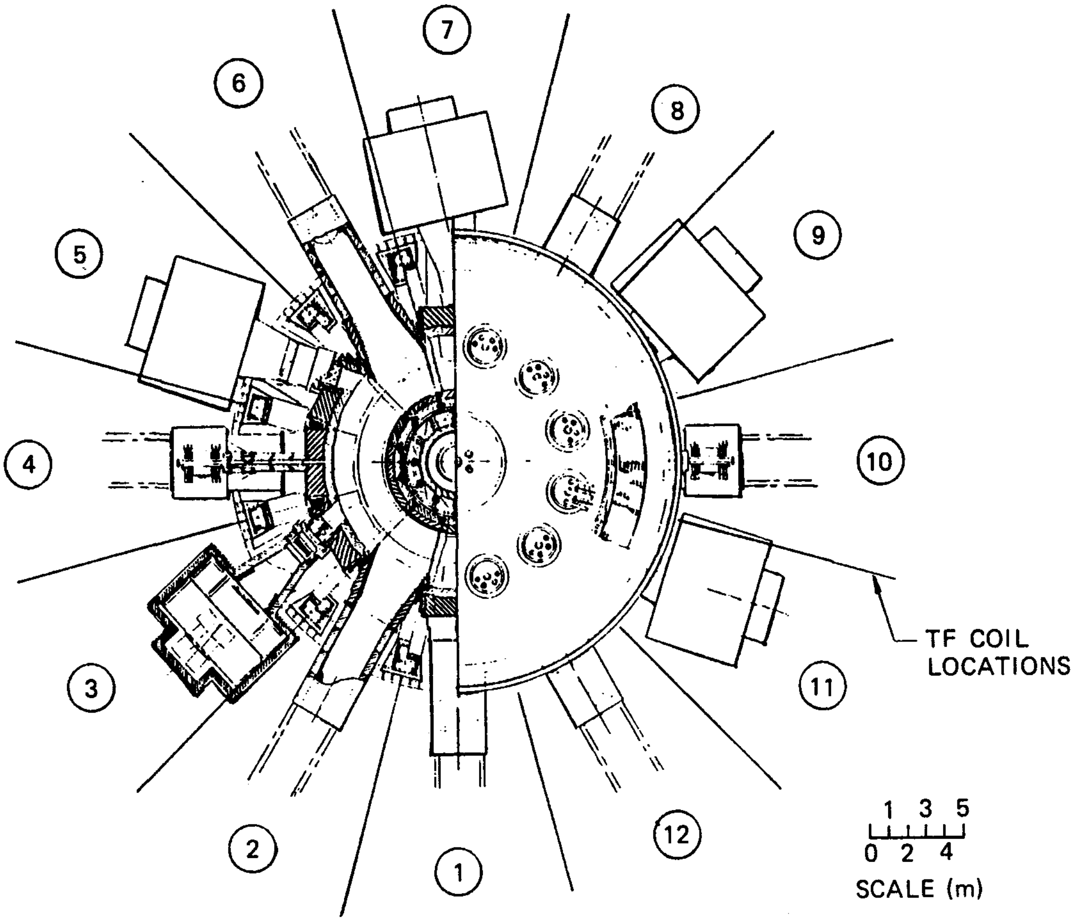


Fig. 9. INTOR device radial build.



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

Fig. 10. Dedicated torus sectors.

bore for plasma initiation. Thermal insulation and refrigeration and cooling systems are also included. The overall magnet system configuration and coil locations are shown in Fig. 11. A brief description of the magnet system follows.

IV.D.1.a. TF Coils. The INTOR TF coil configuration has been developed with sufficient flexibility to incorporate all three of the major cooling concepts presently under development worldwide. The basic TF systems requirement are listed in Table V. The three conductor concepts, all of which are compatible with the overall TF coil configuration and method of support, are characterized in Table VI.

Electromagnetic fields and forces have been computed using the BARK computer code. Eddy currentless calculations were performed using the PURED computer code. The winding pack layouts have been evaluated and designed to provide adequate helium bubble clearance for the pool-boiling concepts. The winding pack substructure was evaluated to assure adequate load paths to the case and supporting structure.

IV.D.1.b. PF Coils. The PF coils utilize a flat cabled conductor with NbTi superconductor. The coils are designed to be pool-boiling cooled by liquid helium at atmospheric pressure. The conductor is wound under sufficient tension in the solenoid so that the winding itself provides structural support. The ring coils utilize a case as support against magnetic forces.

The salient PF parameters are summarized in

TABLE V

TF Coil System Features

Number of coils	12
Field on axis (T)	5.5
Maximum field (T)	10.8
Coil bore size (helium vessel) (m)	7.7 X 10.7
Structural load paths	
a. In-plane running load	Case, winding steel
b. Centering 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. Centering force (MN)	454
b. Overturning moment (MN·m)	342
c. In-plane running load (MN/m)	65.6
d. Out-of-plane running load (MN/m)	52.3

[a]ISS denotes intercoil support structure.

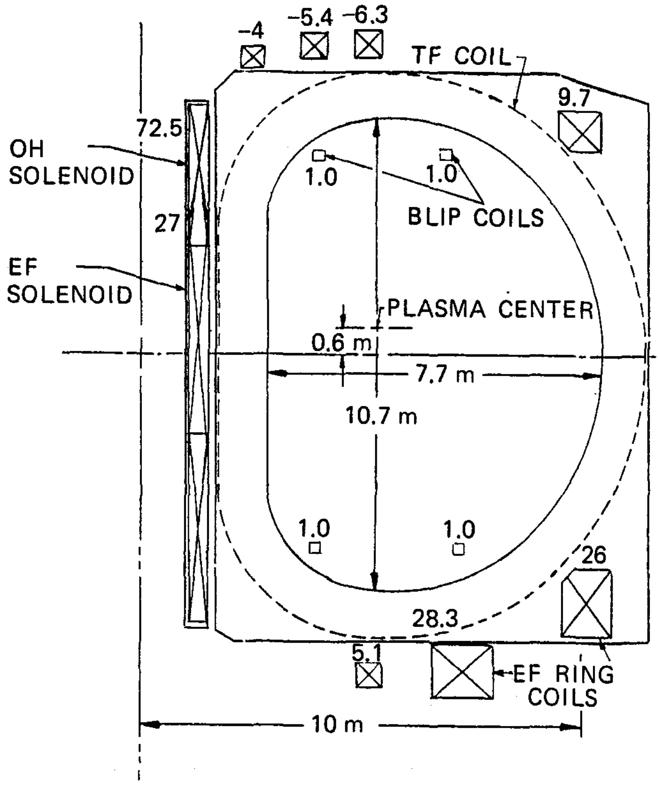


Fig. 11. INTOR magnet configuration (numbers above coils are approximate MAT).

Table VII. The currents in the various coils are shown in Fig. 11 (in MAT).

Electromagnetic fields and forces have been calculated using the BARK computer code. Stability has been verified using the QUENCH computer code for a range of heat loads under maximum credible failure modes. The winding pack layouts have been evaluated and designed to provide adequate helium bubble clearance for the pool-boiling concepts. The winding pack substructure was evaluated to assure adequate load paths to the case and supporting structure.

IV.D.2. Electrical Systems

The INTOR electrical systems consist of power handling and conversion systems for the TF and PF coils and the electrical energy storage system for the pulsed PF and neutral beam systems. These three systems are described below.

IV.D.2.a. TF Coil Power Conversion and Protection. The power conversion system consists of 6 low-voltage power supplies, 2 current controlled voltage regulators, 36 large dump resistors, 12 ground resistors, 12 circuit interrupters, 12 fused transient voltage suppressors, and 12 coil monitoring instrumentation units. The fuses are monitored for continuity and alarm when needing replacement. Six identical groups of components are connected in a series ring circuit to assure that the same current

TABLE VI

TF 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, cowound with steel channel		Spiraled into grooves in plates
Support	Pancakes enclosed in helium case to form coil		Plates bolted together to form coil
Cooling	Pool boiling liquid helium at 4.2 K	Superfluid helium at 1.8 K	Forced flow supercritical helium

TABLE VII

INTOR PF Design Parameters

	Solenoid	Ring Coil No. 20[a]
Geometric		
Mean radius (m)	1.3	9.95
Height (m)	13.2	---
Cross section (m)	---	1.58 (axial) × 2.0 (radial)
Number of pancakes	97	12
Number of turns	1542	516
Magnetic		
Current (kA)	50	50
Peak field (T)	7.5	7.52
Ampere turns (MAT)	72.5	26

[a]The largest ring coil.

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 ~5 h and sustain them during normal operation. The power supplies can also be controlled to invert dc power to ac and 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 12 pulse 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 interconnecting bus work will be sized to provide the slow exponential discharge rate 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. This arrangement limits the transient voltage across the coils and the interrupter that opens first. It also provides better equalization of the current among the coils if a dump resistor module fails. The dump resistors are composed of stainless steel grids of tubes connected in parallel and immersed in water. Nucleate boiling provides heat transfer during a rapid discharge. The grounding resistor between the dump resistors connected across each coil limits the voltage to ground during a rapid discharge. They also have enough resistance to prevent a single ground fault from creating large unequal currents during a rapid discharge.

IV.D.2.b. 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 startup and shutdown 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 buses 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 to create the desired magnetic fields. Make-before-break switches provide the coil backup protection if the inverters fail during operation. Coils are thereby isolated and connected to discharge resistors. Normal discharge rate is provided when the power converters are in operation. Slow discharge backup is not provided for a shorted turn and inverter failures because these events are unlikely to occur simultaneously.

The OH 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 during the burn phase since they are not needed during that time.

IV.D.2.c. Electrical Energy Storage. Four motor generator flywheel (MGF) units each having an energy storage of 6.25 GJ are needed. Four large 25 000-hp wound rotor induction motors are needed to restore the inertia energy after the startup phase of each cycle. Each generator has a peak load capacity of 1200 MVA for the 12-s peak power needed for startup. The energy storage system requirements and design data are given in Table VIII.

IV.E. Heating and Fueling Systems

IV.E.1. Neutral Beam Injection (NBI) System

A detailed trade study was conducted early in this design phase that led to five active beam lines for the U.S. INTOR design. Due to configuration and facility space issues, however, the decision was subsequently made to provide only four active beam lines. The configuration of the beam line is about the same for both cases. There is more cryopanel area per beam line, and the sources are physically larger for the four-beam-line system. The overall cost of the four-beam-line system is lower, but the larger source size presents a higher development risk. A redundant beam line is provided to meet the NBI availability

TABLE VIII

Energy Storage System Requirements and Design Data

Requirements	
Energy–PF Coils (GJ)	22.5
Energy–NBI (GJ)	2.5
Energy–total (GJ)	25.0
Peak load (MVA)	4 400
Design Data	
W. R. induction drive motor	
Number	4
Shaft power (HP)	25 000
Synchronous MGF units	
Number of units	4
Unit peak load capacity (MVA)	1 200
Unit stored energy (GJ)	6.25

requirement. The key features of the five-NBI design are given in Table IX. (Characteristics for the four-beam-line system will be determined in the future.) The system components are shown graphically in Fig. 12.

IV.E.2. Fuel System

The fuel system consists of a gas puffing system, used primarily during the startup, and a pellet injection system for controlling plasma density during the burn phase.

IV.E.2.a. Fuel Gas Puffing System. Both continuous and pulsed flow control of the selected gas are provided in the fuel gas flow control system proposed for INTOR. The fuel gas is selected by the manually controlled remote valves. 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. Either or both of two gas puffing control systems can be selected. If used separately, one flow control valve supplies all 12 gas puffing ports. The manifold divider valve is then open. If both flow control valves are used, the divider valve is closed so that each control valve supplies six of the alternate ports. All or part of the 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. The specie composition of each tank will be determined by monitoring the tritium beta emission.

IV.E.3. Fuel Pellet Injector System

There are two pellet fuel injectors located on opposite sides of the torus. Each injector can inject deuterium and tritium pellets concurrently. A mixed-specie 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. Remote 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 preferrably 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.

Both pneumatic and centrifugal pellet injectors are being developed and could be used for injecting pellets from two feed lines concurrently. The centrifugal injector is rotating many times faster than the feed rate; therefore, two or more species can be injected with the same slinger. 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.

IV.F. 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 radiation heat fluxes from the plasma and radiative 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 startup
4. a beam shine-through region on the inboard wall that receives shine-through of the neutral beams at the beginning of neutral injection

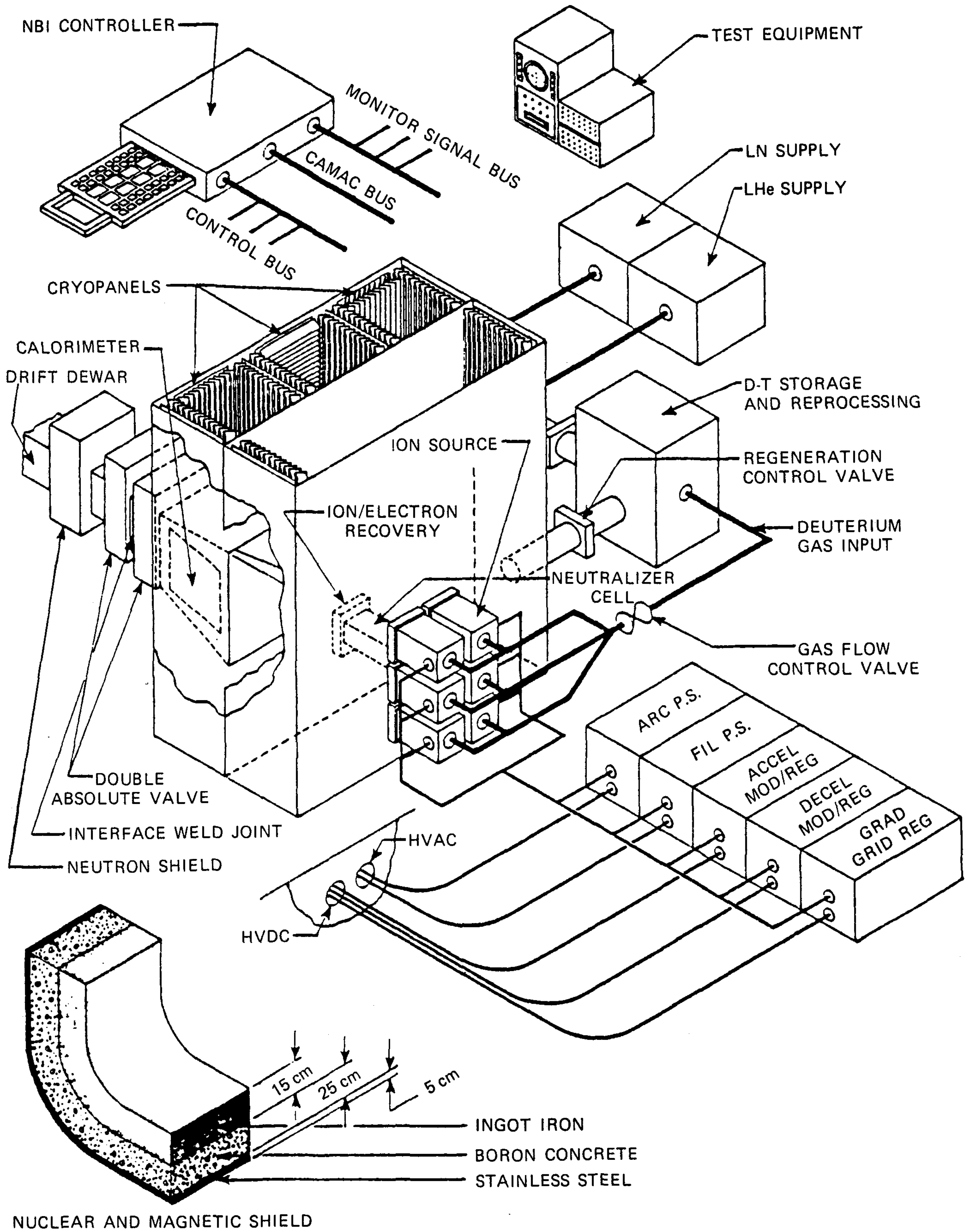


Fig. 12. NBI subsystem components.

TABLE IX

NBI Characteristics

Source Design	
a. Number (to meet 75 MW)	29
b. Size, height X width (cm^2)	14 X 32
c. Current (A)	60
d. Per injector	6
e. Configuration	2 stacks of 3
f. Divergence (deg)	<1.5
g. Energy (keV)	175
h. Pulse length (s)	10
Interface Design	
a. Number of NBIs: 1. operation	5
2. redundant	1
b. Drift duct: 1. height (m)	1.2
2. width (m)	1.0
3. length (cm)	330
4. shutter/valves	1 nuclear/2 vacuum
c. Power density average (kW/cm^{-2})	1.25
d. Injection: 1. angle (deg)	16 & 22
2. power per NBI (MW)	15.9
3. beam cross over	Drift duct center
e. Efficiency (overall NBI): 1. full energy	0.21
2. all energies	0.35
Cryopumps	
a. Gas type	Hydrogen and isotopes
b. Area: 1. drift region (m^2)	39
2. gas cell region (m^2)	7
c. Configuration	Differential LHe shielded by LN chevrons
Ion Deposition	
a. Type	Direct recovery on full energy
b. Capability	≥0.6 efficiency
Species Mix	
a. Ion fraction	0.8, 0.12, 0.08
b. Power fraction	0.62, 0.28, 0.10

5. a region on the outboard wall that receives enhanced particle fluxes caused by ripple effects during the late stages of neutral injection.

Figure 13 is a poloidal view of the reactor showing the location of the various first wall regions. Table X 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. 13). 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. The thin corrugated coolant channels in the panel-type construction selected for the first wall tend to minimize bending stresses and provide longer lifetime 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 0.020 atom/particle at 200 eV and 0.0072 atom/particle at 100 eV for the particle composition given in Table X.

TABLE X

INTOR First Wall Operating Parameters

First Wall	
Total plasma chamber area	380 m^2
Average neutron wall loading	1.3
Radiative power to first wall	40 MW
Charge exchange power	4 MW
Charge exchange current (47% deuterium, 47% tritium, 5% helium, 0.5% carbon, 0.5% oxygen)	1.3×10^{23} s^{-1}
Charge exchange flux	3.3×10^{20} $m^{-2} \cdot s^{-1}$
Charge exchange energy	200 eV
Cycle time (Stage I/Stages II and III)	145/245 s
Burn time (Stage I/Stages II and III)	100/200 s
Total average neutron fluence	6.8×10^{26} n/m^2
Total 14-MeV neutron fluence	6.5 MW·yr/m^2
Total number shots	7.1×10^5
Total disruption energy	220 MJ
Disruption time	20 ms
Total number disruptions	1080
Outboard Region	
Area	266 m^2
Surface heat flux from plasma	11.6 W/cm^2
Surface heat flux from divertor	3.4 W/cm^2
Total surface heat flux	15 W/cm^2
Average nuclear heating	15 W/cm^3
Limiter Region (Outboard wall at R = 6 upper and lower)	
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/m^2
Peaking factor	1.5
Typical particle energy	100 eV
Duration	4 s
Period	t = 0 to 4 s
Ripple Region (Outboard wall at R = 6 upper and lower)	
Area	26 m^2
Heat flux (ripple = ±0.5%)	0.4 MW/m^2
Peaking factor	2
Particle energy (deuterium)	120 keV
Duration	2 s
Period	t = 8 to 10 s
Inboard Region	
Area	114 m^2
Surface heat flux	11.6 W/cm^2
Average nuclear heating	10 W/cm^3
Peak disruption energy density	289 J/cm^2
Beam Shine-Through Region (Inboard wall)	
Total power (15% of injected)	11 MW
Particle energy	175 keV
Duration	2 s
Period	t = 4 to 6 s
Area	11 m^2
Heat flux	1 MW/m^2

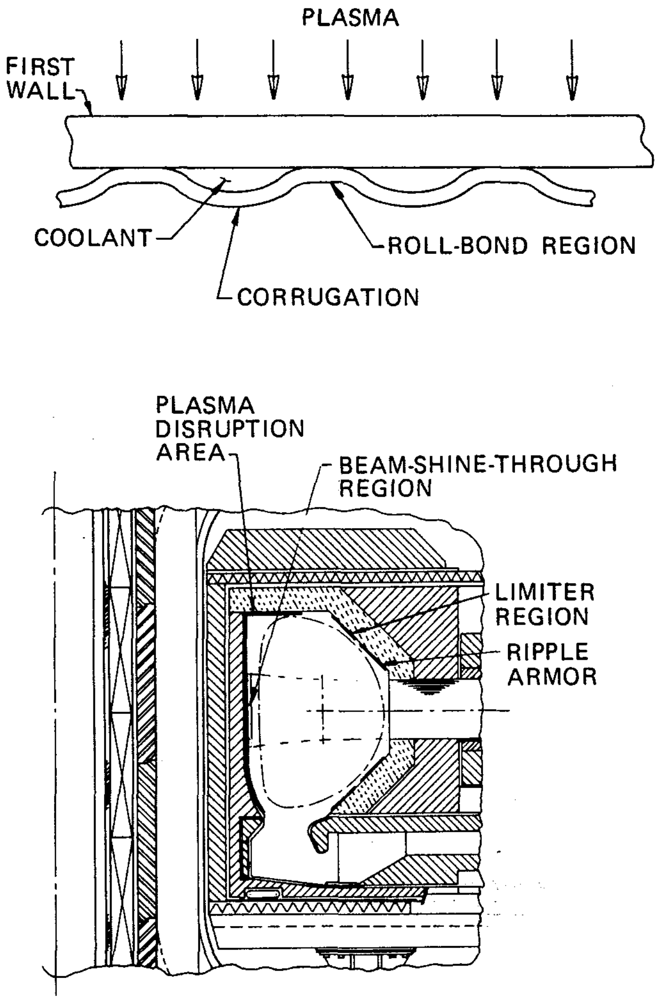


Fig. 13. First wall cross section (top) and first wall configuration (bottom).

The calculated vaporization erosion caused by a plasma disruption is 8×10^{-4} mm per disruption for a 289 J/cm^2 energy density deposited in 20 ms. An uncertainty factor of 2 is used to obtain the design erosion allowance. It is assumed that the melt layer formed during a disruption does not erode.

Table XI is a summary of the lifetime analysis of the first wall system. For the wall thickness requirements necessary to allow for the predicted erosion rates, all regions meet the design temperature, stress, and fatigue criteria for full life operation under the reference conditions. The major uncertainty in this design concept relates to the stability of the melt layer formed during a disruption. A grooved inboard wall concept would accommodate erosion of up to 10% of the melt layer (~0.14 mm/disruption). Further R&D are required to confirm the stability of the melt region during a disruption.

IV.G. Divertor Collector Plate and Channel

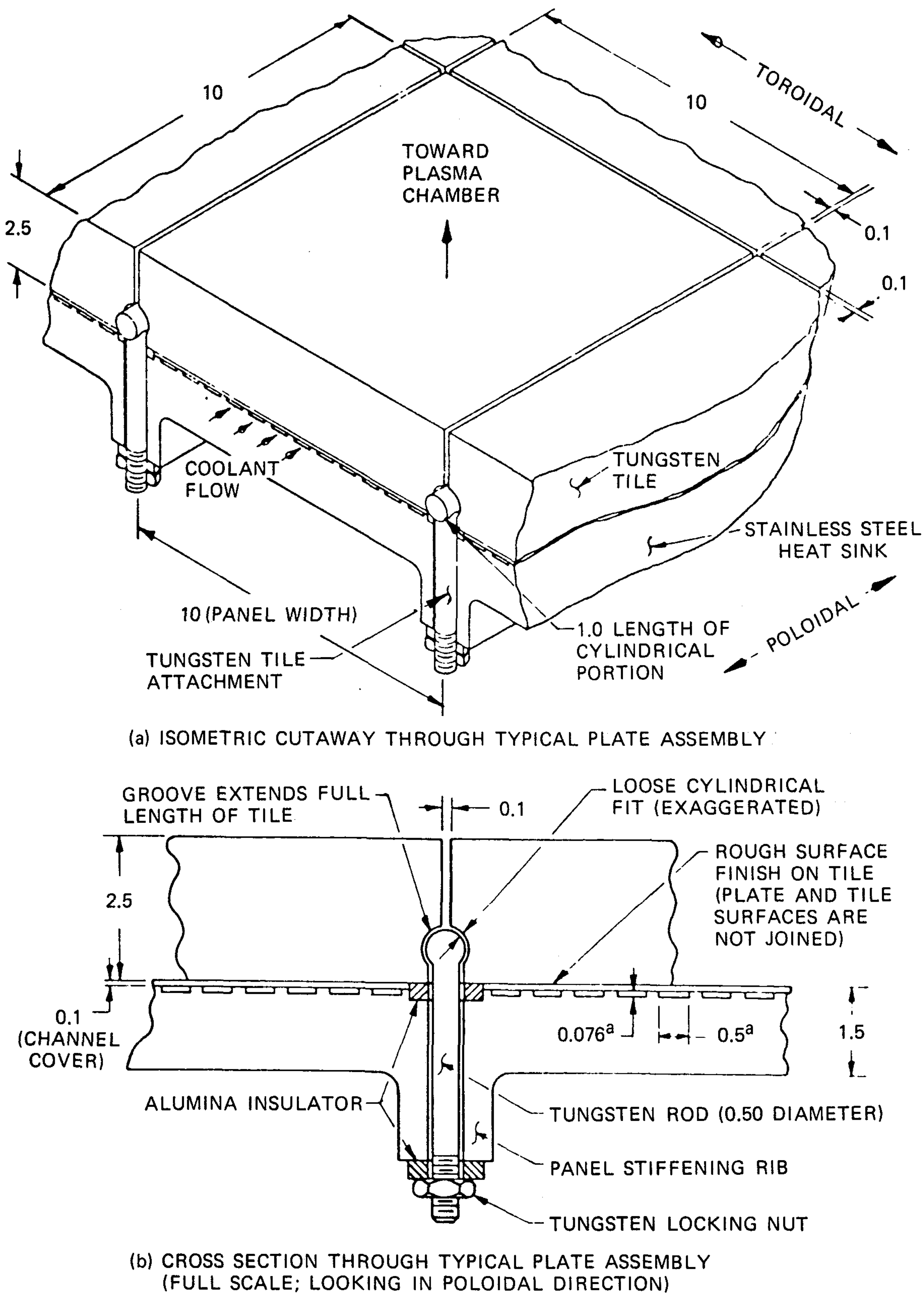
The divertor collector plate design is illustrated in Fig. 14, and the operating and design parameters are listed in Tables XII and XIII. The basic plate assembly consists of tungsten tiles in the shape of rectangular plates that are mechanically attached to an actively cooled stainless steel heat sink. Water is used as the coolant in the heat sink. The collector plates are designed with a poor thermal conductance between the tile and the heat sink, which allows the tungsten tiles to increase in temperature to ~2000°C. At this high temperature, 40 to 50% of the incident heat is radiated back to the divertor and plasma chambers, thereby reducing the thermal gradient in the tile and the heat flux incident on the heat sink. The amount of heat that is radiated back into the

TABLE XI

Summary of First Wall Lifetime Analysis

Region	Total Thickness (mm)	Maximum Erosion (mm)	Maximum[a] Temperature (°C)	Maximum[b] Stress (MPa)	Fatigue Life (cycles)	
					No Erosion	With Erosion[c]
Outboard region	13.4	10.2[d]	310	460	4×10^5	$>10^7$
Ripple region	13.4	10.2[d]	340	460	4×10^5	$>10^7$
Limiter region	14.8	11.6[d]	330	540	2×10^5	$>10^7$
Inboard region	15.6	12.2[e]	280	420	8×10^5	$>10^7$
Beam shine-through region	15.6	12.2[e]	330	420	8×10^5	$>10^7$

[a]Maximum specified temperature = 360°C.
[b]Maximum allowable stress = 650 MPa plasma side, 765 MPa coolant side.
[c]Assumes erosion rate one-half of predicted rate, for conservative design.
[d]Physical sputtering.
[e]Physical sputtering plus vaporization.



[a]CHANNEL DIMENSIONS SHOWN ARE FOR PEAK HEAT FLUX REGION ONLY. ALL DIMENSIONS IN CENTIMETRES. ALL DIMENSIONS ARE TYPICAL.

Fig. 14. Reference divertor collector plate design.

TABLE XII

Divertor Operation Conditions

Design concept	Single null poloidal divertor
Total energy to divertor	80 MW
Ion energy to divertor plates	35 MW
Electron energy to divertor plates	35 MW
Charge-exchange energy to throat and walls	5 MW
Radiation energy to throat and walls	5 MW
Energy to channels	
Outboard	40 MW
Inboard	40 MW
Peak energy flux to channels at null (normal to separatrix)	
Outboard	8 MW/m^2
Inboard	4 MW/m^2
Total ion flux to divertor	$5.5 \times 10^{23}/s$
Average energy of ions	400 eV
Peak ion flux to channels at null (normal to separatrix)	
Outboard	$6 \times 10^{22}/m^2 \cdot s$
Inboard	$3 \times 10^{22}/m^2 \cdot s$
Total neutral flux to divertor throat and walls	$1.6 \times 10^{23}/s$
Average energy of charge-exchange neutrals	200 eV
Uniform neutral particle flux	$7 \times 10^{20}/m^2 \cdot s$
Peaking factor of deposition load	2

plasma chamber deposits an additional ~3.4 W/cm^2 on the outboard first wall. This additional heat load is predicted to not adversely impact the lifetime of the first wall.

The mechanical attachments allow the tile to freely expand and rotate as the temperature changes during the burn cycle. This design significantly reduces the thermal stresses in the tile and allows the tile thickness to be increased to increase the sputtering lifetime. A two-dimensional thermal-hydraulic and stress analysis has been performed on the tungsten tile, and the results indicate the temperatures and stresses in a 2.5-cm-thick tile remain in an acceptable range during the burn cycle. The maximum stresses occur during the ramps up or down in power. Unfortunately, the absence of high-temperature fatigue data prevents the tile fatigue lifetimes to be accurately estimated.

The principal concern for the low conductance design (and possibly for any design employing refractory metals) is the potentially significant chemical sputtering by oxygen. A simple model of tungsten oxidation predicts an oxidation loss rate approximately three-fourths of the physical sputtering loss rate. The complex nature of the environment in front of the collector plate introduces a high uncertainty on this prediction, however. Additional theoretical and experimental work on chemical sputtering are required to resolve this problem.

The heat sink is constructed out of Type 316 austenitic stainless steel. Compared with copper, stainless steel has the advantages of being a standard structural material and of having a known resistance to radiation damage. It also has poor thermophysical properties that lead to large thermal stresses. Thermal stress calculations based on the divertor operating conditions indicate that it is possible to design an austenitic stainless steel heat sink that meets the American Society of Mechanical Engineers (ASME) guidelines for stress and fatigue lifetimes. Based on available radiation effects data, a heat sink of Type 316 stainless steel is predicted to last the reactor lifetime. In addition, a stainless steel heat sink will experience a much lower magnetically induced torque during a disruption, because of its relatively higher electrical resistivity.

IV.H. Tritium-Producing Blanket

A partial tritium breeding blanket will be installed on INTOR to reduce the cost of externally supplied tritium. Liquid (lead-lithium eutectic) and solid breeders were considered. The technology for tritium extraction from a liquid breeder is relatively well understood, but the blanket design is rather complicated, relative to a nonbreeding blanket. The blanket design for a solid breeder is much less complicated, but the uncertainty about radiation effects on tritium release is a major concern. Research programs are currently under way that should resolve this uncertainty within the next year or so. The solid breeder was selected for INTOR based on the lower risk of the engineering design and the fact that the present uncertainty about tritium release will be resolved in the near future.

The breeding blanket in INTOR uses the top and outboard portion of the 12 removable blanket/shield sectors. To permit a blanket design that is easily adapted to the varying width of the top region and to the changes in neutron wall loading with distance from the midplane, two key features were adopted: (a) a modular approach, by which the blanket is divided poloidally into a number of discrete segments, and (b) coolant flow through the module across the full sector width, in the toroidal direction.

An isometric breakout view of a typical breeding blanket module is shown in Fig. 15. The key blanket design features and materials selections are summarized in Table XIV. Design and operating parameters for the blanket are given in Table XV.

TABLE XIII

Divertor Collector Plate Design Parameters

Design concept	Tungsten tiles mechanically attached to a water-cooled stainless steel heat sink
Angular position of plates with respect to separatrix	
Outboard	20 deg
Inboard	45 deg
Peak energy flux to tiles	3 MW/m^2
Peak ion flux to tiles	$2.2 \times 10^{22}/m^2 \cdot s$
Tile	
Material	Fine grained recrystallized tungsten
Dimensions	10 X 10 X 2.5 cm
Effective sputtering coefficient	2.2×10^{-3}
Tungsten loss rate (peak ion flux)	7.7×10^{-10} m/s
Lifetime (peak ion flux–50% duty factor)	1.5 years
Maximum temperature (top surface–end of burn)	2030°C
Minimum temperature (back surface–end of dwell)	1150°C
Heat Sink	
Material	Type 316 austenitic stainless steel
Coolant temperature (in/out)	50/100°C
Dimensions	
Top plate thickness	1.0 mm
Back plate thickness	14.0 mm
Conductance between tile and sink	568 $W/m^2 \cdot K$
Peak heat flux to sink (end of burn)	1.75 MW/m^2
Lifetime	$>10^6$ cycles

TABLE XIV

Summary of Key Features and Materials Section for Reference Design Blanket

Zone	Material	Features
First wall	Stainless steel H_2O coolant	Integrated with blanket
Neutron multiplier	Lead	Faces cooled by first wall and second wall coolant panels Helium gas for thermal conductance
Tritium breeding	Graphite moderator	Helium gas for thermal conductance Separate cooling tubes
	Lithium silicate breeder, Li_2SiO_3	Formed in cylinders around coolant tubes
	H_2O coolant	Contained in single-walled coolant tubes Tubes extend in toroidal direction

The blanket design features a solid lithium compound breeder, Li_2SiO_3, with lithium enriched to 30% of ^6Li. The solid breeder is fabricated at 70% of theoretical density. Adequate tritium breeding is achieved by using lead as a neutron multiplier. Graphite neutron moderator is used in the breeding zone to minimize solid breeder inventory. Tritium is removed from the breeder by a gaseous helium purge stream. Breeder minimum and maximum temperatures during operation are 400 and 600°C. This temperature range and the 70% density facilitate tritium diffusion from the breeder. The low lithium inventory helps reduce the part of tritium inventory related to solubility. Pressurized light water (H_2O) is used to cool all parts

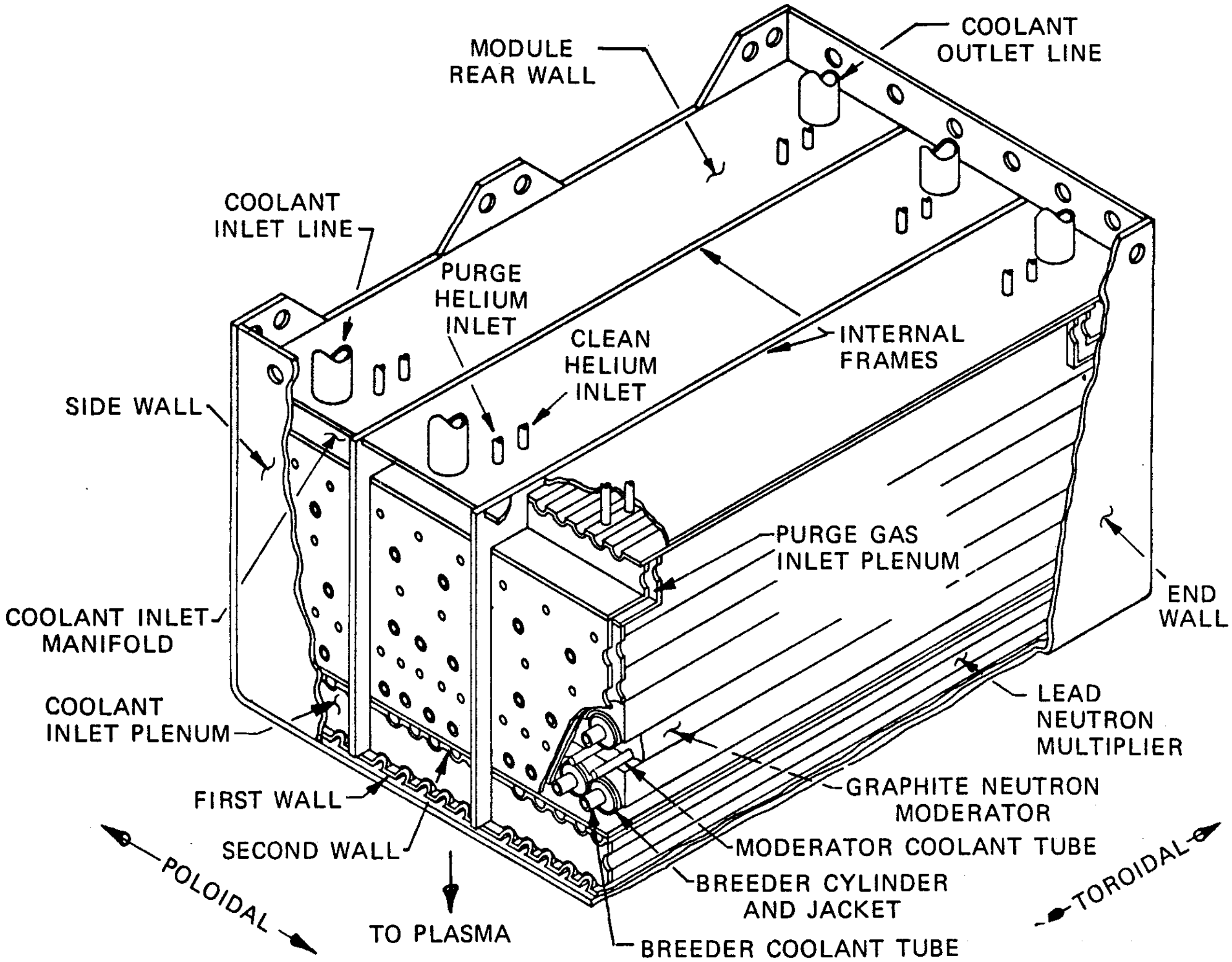


Fig. 15. Reference tritium-producing blanket design.

of the blanket module. Type 316 stainless steel is used for all structural and pressure-carrying components.

The module design integrates the structure and cooling system of the INTOR first wall, blanket neutron multiplier, and blanket breeding zone. Total thickness of the first wall/blanket module is 50 cm. Module length poloidally is nominally 1 m, but this dimension can easily be tailored to fit the modules into the blanket sector geometry.

The first wall panel serves as the plasma-side containment for the neutron multiplier, and its cooling system removes part of the lead neutron multiplier's volumetric heat. The containment at the back face of the multiplier is also an actively cooled corrugated panel (second wall), which removes the remainder of the multiplier's volumetric heat. The first and second walls are joined by intercostals that extend through the 5-cm-thick multiplier, thus combining the two panels structurally into a relatively deep two-cap beam. The panels, the module side walls, and end walls form a pressure boundary around the multiplier. Low-pressure helium in this zone provides good thermal conductance at the multiplier/coolant panel interfaces. Maximum multiplier temperature is predicted to be 290°C, well below the 327°C melting point of lead.

The lithium silicate behind the neutron multiplier is formed in cylinders around single-wall stainless steel coolant tubes. There are three separate rows, or banks, of breeder cylinder/coolant tube assemblies that are separated radially within the breeding zone. The graphite neutron moderator is located between banks and between the third bank and the back wall of the module. Separate, small-diameter coolant tubes cool the moderator.

A thin metal jacket surrounds each breeder cylinder. This jacket provides a pressure boundary between the helium purge gas that flows through the breeder cylinder and the "clean" helium that fills the

TABLE XV

Summary of Reference Design Parameters for Tritium-Producing Blanket

Neutron Multiplier	
Material	Lead
Maximum temperature, °C	290
Melting point, °C	327
Thickness, m	0.05
Theoretical density, g/cm^3	11.34
Effective density, %	100
Second Wall	
Form	Corrugated panel
Structural materials	Type 316 austenitic stainless steel
Maximum structural temperature, °C	<150
Total structural thickness, mm	2.5
Coolant	H_2O
Coolant outlet temperature, °C	100
Coolant inlet temperature, °C	50
Coolant nominal pressure, MPa	0.7
Region thickness, mm	6.0
Breeding Region	
Structural material	Type 316 austenitic stainless steel
Maximum structural temperature, °C	<150
Breeder material	Li_2SiO_3 (lithium enriched to 30% of ^6Li)
Theoretical density, g/cm^3	2.53
Effective density, %	70
Grain size, 10^{-6} m	≤1
Breeder maximum/minimum temperature, °C	600/400
Neutron moderator material	Graphite
Effective density, g/cm^3	1.9
Moderator zone gas	Helium (0.10 MPa)
Region thickness, m	0.43
Coolant	Pressurized water, H_2O
Coolant outlet temperature, °C	100
Coolant inlet temperature, °C	50
Coolant nominal pressure, MPa	0.7
Tritium processing fluid	Helium (0.10 MPa)

remainder of the breeding zone and the multiplier zone. The clean helium provides good thermal conductance between the moderator and the jackets. Jacket temperature during the reactor operation is relatively low to maintain a low-permeability barrier against tritium migration from the purge gas into the graphite.

Coolant inlet and outlet temperatures are 50 and 100°C. The coolant is pressurized to 0.7 MPa (100 psi). The moderator coolant tubes and breeder coolant tubes all connect to coolant inlet and outlet plenums located at the module sides (in the poloidal plane). These plenums also connect to the first and second walls. At the rear of the blanket, the plenums are widened, to serve as a manifold region. These manifolds are connected to large-diameter coolant lines that extend through the bulk shield behind the module.

Helium purge gas inlet and outlet plenums are located between the coolant plenums and the breeding zone. The purge gas flows inside the jackets through several narrow gaps that extend radially through the breeder cylinder. A low partial pressure of oxygen in the 1-atm pressure purge helium reacts with the free tritium at the surfaces of the breeder particles to form T_2O and HTO, which enter the purge gas stream. The purge gas plenums are connected to small-diameter lines that pass through the bulk shield.

The tritium inventory in the blanket should be ~1 kg, based on present knowledge of tritium release data. Radiation effects on tritium release could possibly result in a significantly larger inventory, but these effects are uncertain at present.

The end and rear walls of the module are stiffened sheet metal construction, adequate to contain the 1-atm pressure of the "clean" helium in the breeding zone. They are welded to the side, first, and second walls to form separate pressure-tight multiplier and breeding zones. Actively cooled internal frames within the breeding zone connect to the second wall and breeding zone rear wall. These frames are in line with the intercostals in the multiplier. All the walls, coolant panels, intercostals, and internal frames in the module assembly thus form an efficient, integral structure to react gravity, seismic, pressure, and electromagnetic loads that act on the module components.

IV.I. Radiation Shield System

The shield system was designed to perform several functions and satisfy the design specifications. The main function during reactor operation is to provide sufficient neutron and photon attenuation to (a) protect the different reactor components from radiation damage and nuclear heating, (b) reduce the induced activation level in reactor components, and (c) protect the workers and the public from radiation exposure. The system is designed to permit personnel access into the reactor building, with all shield in place, within 24 h after shutdown.

The TF coils are designed to last the lifetime of the reactor (>6 MW·yr/m^2 integrated first wall neutron wall loading) without change in performance. Therefore, the shield system has to protect the different components of the TF coils from radiation damage. The inboard blanket and shield was carefully optimized to minimize the radiation damage in the inner portion of the TF coils within the allowable radial thickness. The inboard reference blanket and shield specifications are given in Table XVI. The radiation response parameters in the inboard portion of the TF coils are given in Table XVII.

In addition to radiation protection of reactor components, the outboard shield was designed to permit personnel access after shutdown. The optimization process calls for 105-cm shield thickness behind a 50-cm tritium breeding blanket to achieve 2.5 mrem/h biological dose rate in the reactor building outside the bulk shield within 24 h after shutdown. Table XVIII gives the details of the outboard

TABLE XVI

Inboard Blanket and Shield Design

Zone Composition	Zone Thickness (cm)	Zone Composition and Percentage by Volume
First wall including armor	2	78% Type 316 stainless steel, 22% H_2O
Blanket and shield (A)	16.5	90% Type 316 stainless steel, 10% H_2O
Jacket	1.5	100% Type 316 stainless steel
Vacuum gap	3	Void
Jacket	1.5	100% Fe-1422
Blanket and shield (B)	45	90% Fe-1422, 10% H_2O
Blanket and shield (C)	10	72% B_4C (0.9 DF), 18% Fe-1422
Jacket	1.5	100% Fe-1422

TABLE XVII

Radiation Response Parameters in the Inboard TF Coils Normalized to 1.3 MW/m^2 Neutron Wall Loading and 6 MW·yr/m^2 Integral Neutron Wall Load

Maximum neutron fluence in the superconductor (n/cm^2)	3.88×10^{17}
Maximum induced resistivity in the copper stabilizer (Ω-cm)	3.1×10^{-8}
Maximum atomic displacement in the copper stabilizer (dpa)	2.54×10^{-4}
Maximum nuclear heating in the superconductor (W/cm^3)	9.16×10^{-5}
Nuclear heating in the superconductor and the TF case (W/cm)	7.21
Nuclear heating in the superconductor, TF case, and Dewar (W/cm)	16.72
Maximum radiation dose in the thermal insulator (rads)	2.5×10^{9}
Maximum radiation dose in the electrical insulator (rads)	6.8×10^{8}

TABLE XVIII

Outboard Shield Design

Zone Description	Zone Thickness (cm)	Zone Composition and Percentage by Volume
Shield jacket	1.5	100% Fe-1422
Fe-1422 shield	73	99% Fe-1422, 1% H_2O
B_4C shield	15	65% H_2O, 10% Fe-1422, 25% B_4C (0.9 DF)
Lead shield	4	100% Lead

shield, where Fe-1422 alloy is used to reduce the induced radioactivity.

Since the penetration shield significantly affects reactor cost, personnel access requirements and reactor operation, an elaborate three-dimensional radiation streaming calculation for the complete reactor system, including the NBIs 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 modeled in the calculations. Calculations were performed for neutron and photon transport during the reactor operation and decay gamma transport after shutdown. The results show 3-kW nuclear heating in the vacuum pumps of the neutral beam system (six beams each have 1- × 1.2-m beam duct) with the shutter open. The calculations show the biological dose rate in the reactor building outside the bulk shield to be 2.5 mrem/h within 24 h after shutdown with the following penetration shield thicknesses: (a) 100 cm for neutral beam drift tubes, (b) 75 cm for the surfaces of the beam injector box facing the drift tubes, and (c) 50 cm for the rest of the neutral beam system and the divertor ducts.

IV.J. Tritium System

The INTOR design incorporates a complete deuterium/tritium/lithium fuel cycle. The primary goal in the design of the tritium containment and handling systems is to produce a completely integrated tritium fuel cycle.

Major objectives adopted for the design are as follows:

1. A minimum tritium inventory is to be maintained in the entire plant.
2. The tritium impact on the environment is to be minimized.
3. Worker exposure is to be reduced to levels as low as practical.
4. Tritiated waste generation is to be minimized.
5. The tritium systems are to be operated in areas free of gamma or neutron radiation, when possible.
6. The tritium systems are designed to have maximum reliability and availability.

The tritium handling systems perform the following functions:

1. reprocess tritium for fueling
2. process tritium produced in the blanket
3. process the NBI exhaust deuterium
4. process the gaseous wastes
5. detritiate the coolant
6. recover tritium from the atmosphere of the different buildings.

To fulfill the different requirements of the tritium system, the following principles are followed:

1. All parts of the system in contact with tritium must be metallic or ceramic.
2. Triple, or at least double, containment is used.
3. Component redundancy is a necessity.
4. With the goal of minimizing any accidental tritium release and, at the same time, gaining flexibility, the tritium plant might be subdivided along functional lines.
5. A tritium storage with a capacity of 30 days of full operation is foreseen. This storage must be placed far from the tritium system in a safe vault and subdivided into small units.
6. An emergency cleanup system will be available for recovering tritium from room air in the event of an accidental release from secondary containment vessels.

A process flow diagram has been developed describing the process systems used to accomplish the functions described above. The removal of impurities from tritium fuel is based primarily on adsorption on molecular sieve beds at 75 K. The recovery of tritium from impurities is based on catalytic oxidation of impurities, followed by freezeout of the resulting tritiated water, and electrolysis of water. Isotopic enrichment of the mixed tritium/deuterium streams is based on cryogenic fractional distillation in a multicolumn (4- to 5-column) system. The total tritium inventory in the tritium handling systems is estimated to be 220 g, exclusive of the inventory in storage, vacuum pumps, and the breeding blanket. The total inventory is ~3.5 kg, including 1 kg in the breeding blanket.

Tritium permeation through containment vessels is found to present no major problems, except possibly at the first wall where implantation of energetic tritium ions from the plasma may enhance the normal permeation by a huge factor. Experimental information is needed before this matter can be resolved.

IV.K. Diagnostics, Instrumentation, Data Acquisition, and Control

Instruments suitable as candidates for plasma control and diagnostics for INTOR have been reviewed. The listing is similar to the one provided in the Phase Zero report.[5] It is expected that this list will remain basically unchanged until experimentation with machines such as TFTR, T15, JET, and JT-60 begin to add to the list, and help select from the list. Some of the listed instruments will not, at present, function well in the D-T environment (e.g., soft x-ray measurements), but the importance of the measurement is such that concepts and development work should be carried out to make the instruments available. There are few proven instruments for the D-T phase of operation. It is unlikely that additional proven instruments for this phase will become available before TFTR operates in the D-D and D-T modes.

Areas of diagnostic need and phases of operation were reviewed, and it is easily concluded that even if one forceably reduces the instrument list to essential instruments, INTOR will still be the most highly instrumented tokamak yet conceived. Signal levels at the first wall were considered. The first wall is exposed to some 10 MW of synchrotron radiation, 3×10^{15} photon/cm$^2 \cdot$s^{-1} at 3 to 30 keV due to bremsstrahlung radiation and 6×10^{13} n/cm$^2 \cdot$s^{-1} as well as other plasma outputs in the D-T phase.

The impact of the INTOR mechanical configuration on the implementation of the set of INTOR diagnostics was investigated. A great deal of work remains before an integrated instrument package can be presented. It was found, for example, that the poloidal divertor has a serious effect on the ability to obtain local electron density measurements in the plasma, and that the combination of shield configuration and poloidal divertor makes magnetic measurements difficult. Neutron measurement, on the other hand, benefits from the shield, since the shield aides in the collimation of neutrons, and the poloidal divertor has minimal negative impact on obtaining local neutron production rates.

Instrumentation for the nuclear, magnet, heating, etc., systems can be adapted from fission reactor, high-energy physics, and other technologies. However, an integrated system will require substantial development.

IV.L. Facility Layout

The INTOR facilities' description includes the tokamak reactor building, hot cell facilities, and an overall site arrangement.

IV.L.1. Tokamak Building

The tokamak building (Figs. 16 and 17) has been sized to house the tokamak reactor and associated equipment and to contain the tritium and activated materials under postulated accident conditions. The building is a cylindrical reinforced concrete structure 33 m in radius, 39 m high, and lined with a steel shell. The top is enclosed with a dome, providing an additional rise of 13 m. A 2-m wall thickness is required for neutron shielding.

IV.L.2. Hot Cell Area

The hot cell facilities provide the maintenance operations support for the tokamak and supporting radioactive facilities within the tokamak building. The maintenance facility is designed for repair of major equipment modules that are replaced by spares to minimize reactor downtime.

The hot cell facility required for the anticipated maintenance operations is 100 m long $\times$ 55 m wide $\times$ 25 m high. The wall thicknesses of the individual cells vary between 0.8 and 2.0 m. The overhead crane capacity is between 300 and 500 ton.

IV.L.3. Site Layout

The general site arrangement is shown in Fig. 18. The tokamak building is centrally located to minimize pipe and cable runs from major support buildings and facilities.

The tritium processing building contains areas for isotope separation, fuel cleanup, waste treatment, tritium storage, and emergency cleanup systems for the tokamak building atmosphere.

The cryogenic building contains refrigeration units and storage capability for the liquid helium and nitrogen.

The heat exchanger building contains the equipment to transfer the heat from the primary coolant systems to the cooling tower circulating system. Intermediate loops are used for the first wall and shield coolant to reduce possible tritium release.

The essential power building contains the two standby diesel generators and the standby noninterruptible power supplies to help assure that radioactive containing systems continue to function safely during emergencies.

The buildings in the electrical power conversion area contain the equipment required to transform the line power to supply load requirements and to provide fault protection. The MGF building contains the

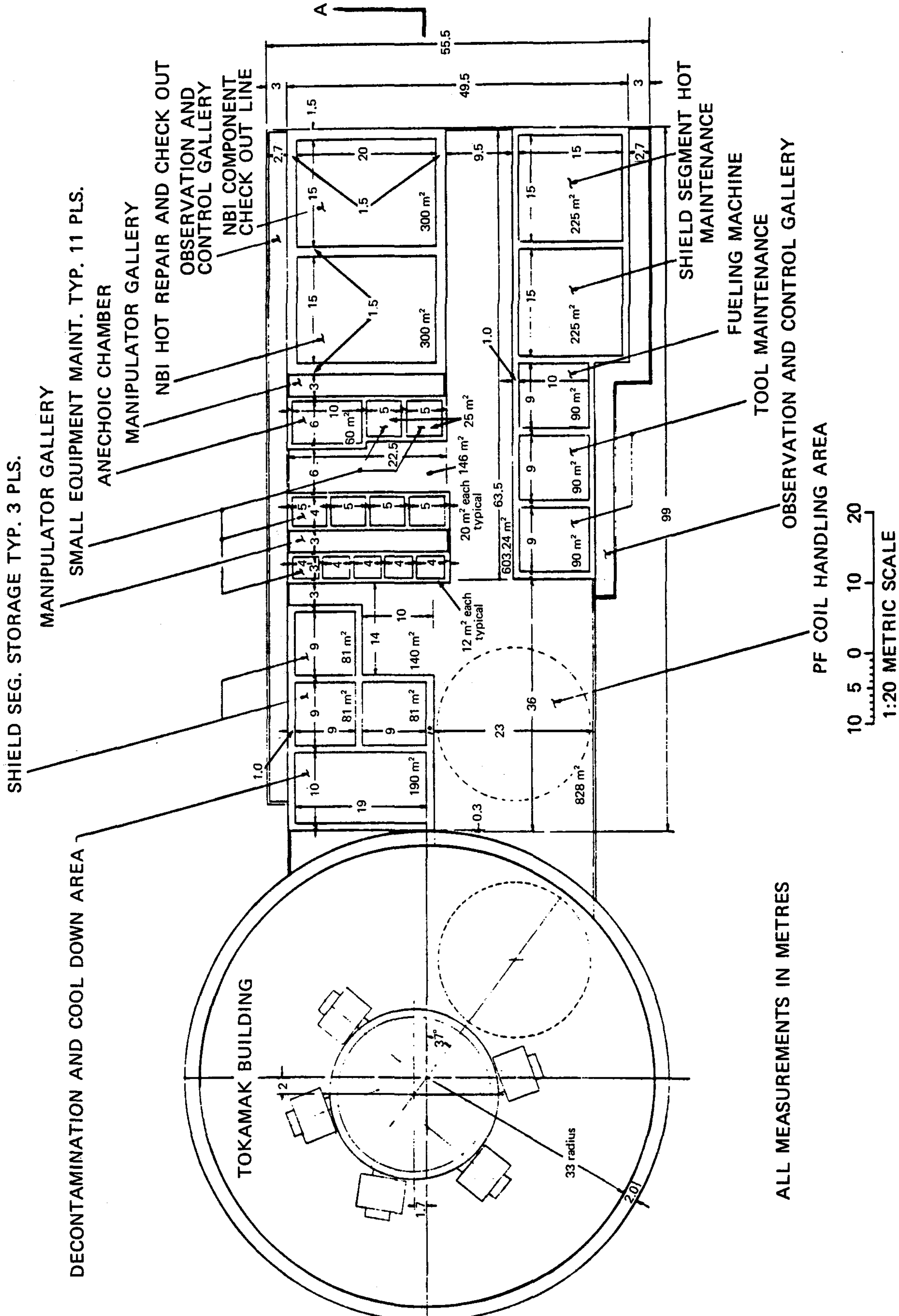


Fig. 16. Tokamak building and hot cell facilities (plan view).

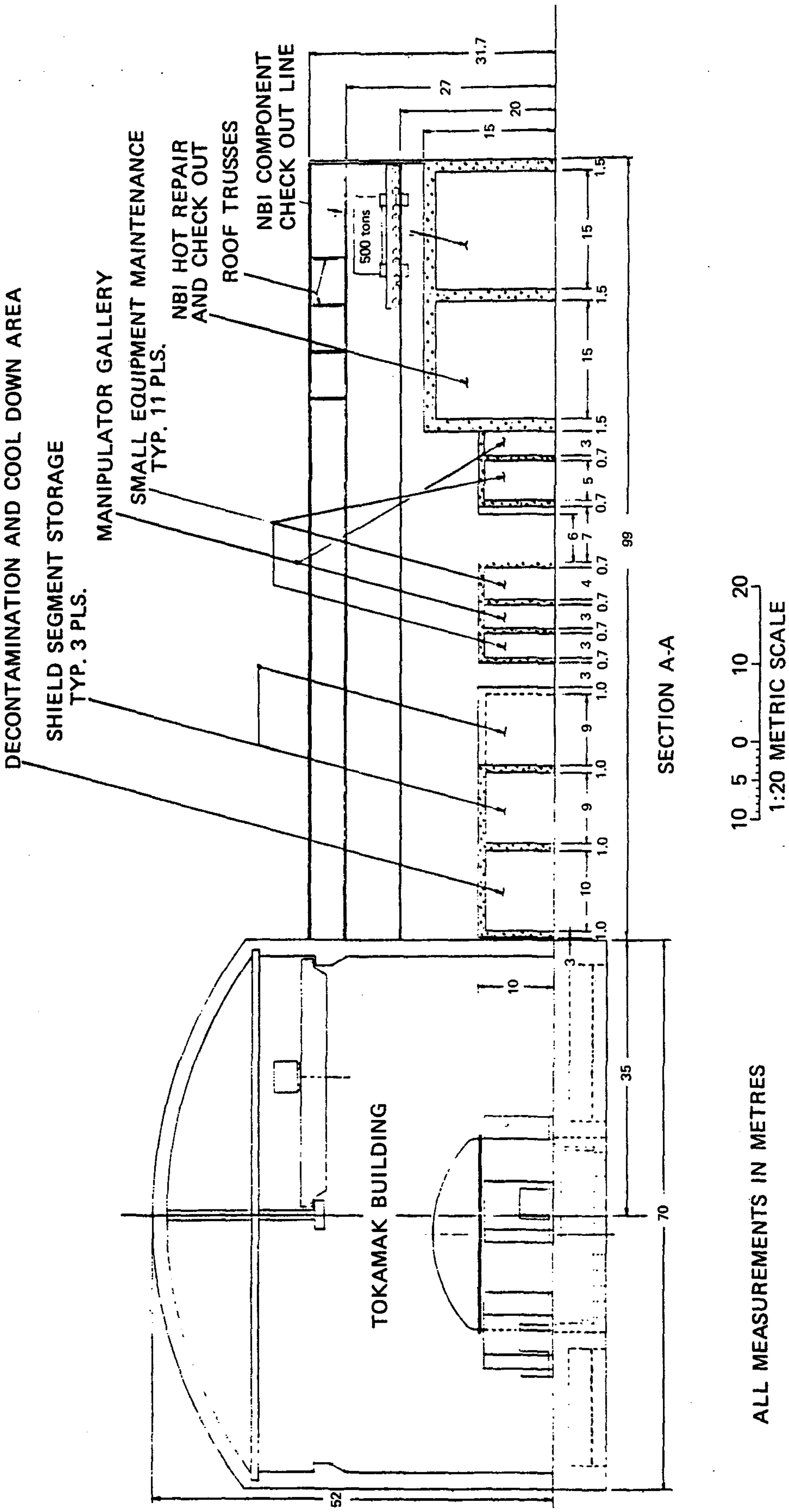


Fig. 17. Tokamak building and hot cell facilities (elevation view).

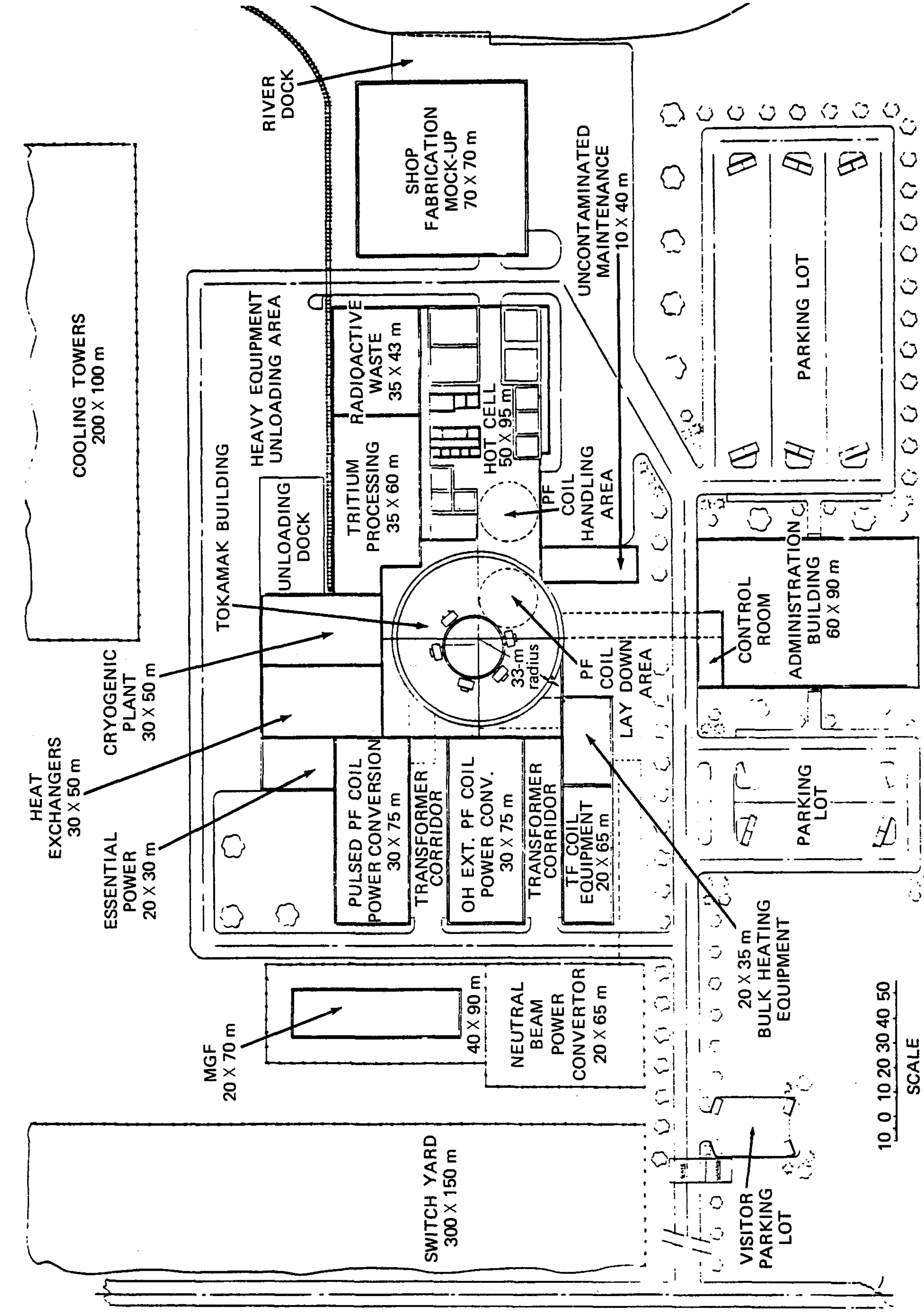


Fig. 18. Site arrangement.

motor generator flywheels used for pulsed energy storage.

The radioactive waste building contains equipment for receiving, storing, and processing for shipment the activated and contaminated solid, liquid, and gaseous wastes.

The uncontaminated maintenance building provides inspection, routine maintenance, and calibration space for uncontaminated or decontaminated equipment removed from the tokamak building or hot cell.

The shop building contains facilities for receiving, assembly, and checkout of plant components, and mockups for training of personnel and verification of procedures and equipment for remote maintenance.

The administration building contains office space for professional and administrative personnel, meeting rooms, cafeteria, and visitor reception areas. The control room from which reactor operations are directed is also located in this building.

V. MACHINE OPERATION AND TEST PROGRAM

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 initial plan has been developed using judgments as to where INTOR fits within an international fusion development plan, as discussed in Sec. II. A summary of the types of testing included in the test plan is given in Table XIX.

V.A. Plasma Physics Tests

Plasma physics tests will emphasize those studies that cannot be performed on other experiments and for which INTOR represents a unique testing resource, e.g., control of equilibrium evolution on the resistive time scale to achieve very long burns. In addition, there will be a variety of tests performed as part of the process of achieving optimized plasma performance of INTOR, e.g., profile control.

V.B. Plasma Engineering Tests

Plasma engineering tests refer to the testing of plasma heating and confinement hardware that interacts directly with the plasma. Testing rf launchers in a thermonuclear environment is one example.

V.C. Blanket Engineering Tests

Blanket engineering tests will emphasize confirmation of results predicted from *ex-machina* tests. Tests will include 1-m^2 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 blanket module tests are planned.

Three different blanket module concepts were reviewed to examine generic problems associated with possible testing on INTOR. These included a water-cooled solid breeder, a helium-cooled solid breeder,

TABLE XIX

Types of Testing in INTOR

<u>Plasma Physics</u>
Vacuum vessel conditioning
OH startup
Neutral beam startup
Long pulse ignition experiments
Performance optimization
<u>Plasma Engineering</u>
Impurity control and exhaust technology
rf heating technology
Burn control technology
Continuous burn methods technology
<u>Blanket Engineering</u>
Prototype module
Tritium recovery
Critical element
<u>Bulk Materials</u>
Irradiation effects on properties of candidate structural materials, insulators, high heat flux materials, breeders, and neutron multipliers
<u>Surface Effects</u>
Retention/reemission characteristics
Plasma impurity release
Surface erosion/redeposition
Surface microstructural changes
Mechanical and physical property changes
<u>INTOR Reactor Material and Component Surveillance</u>
Engineering performance of systems
Failure modes and rates
Maintenance experience
<u>Nuclear Tests</u>
Tritium breeding ratio
Nuclear reaction rates
Volumetric nuclear heating
Neutron and gamma-ray fluxes and spectra
<u>Electricity Generation</u>
Early power generation—end of Stage II
Prototype DEMO blanket—end of Stage III
Breeding and power generation

TABLE XX

Blanket Testing–Irradiation Effects

Long-Term Tests–to end of Stage III (Demonstration)	
Materials Response	
Swelling	Flaw growth
Creep	Compatibility
Fatigue	Hydrogen embrittlement
Mechanical Behavior	
Distortion	
Life/failure modes	
Tritium leakage	
Neutronics	
Activation	
Lithium burnup	
Short-Term Tests–one month to one year (Verification)	
Thermal Hydraulic	
Operating temperatures	
Off normal coolant flow	
Design options	
Neutronics	
Activation	
Breeding ratio/profiles	
Tritium buildup	

and a helium-cooled liquid lithium breeder. In all cases, a module size of 1 m^2 or smaller was found to adequately simulate the effects. The specific blanket concepts were selected only for the purpose of determining the types of tests to be performed. Specific blanket designs for inclusion in INTOR must be developed later in the INTOR program.

General testing specified for the three concepts is summarized in Table XX. Demonstration tests are expected to be used to correlate testing results and analytic predictions for combined materials and synergistic effects and 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 2 or 3 of the design lifetime for the DEMO is considered to be a requirement on INTOR. Should failures occur, failure analysis would provide information on failure modes and guide design variation tests for design improvement efforts.

V.D. Bulk Materials Property Tests

A bulk materials test program has been defined to provide information for (a) a primary and backup structural materials, (b) high heat flux materials, (c) insulators, (d) breeders, and (e) 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 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- × 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. Only one *in situ* cyclic fatigue specimen could be fit in a single capsule. A total of 300 capsules are required to contain all 30 000 specimens. The design permits as many as 153 capsules 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 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 2 or less 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·yr/m^2 test fluence. Tests were started at the beginning of Stage II INTOR operations, and the last specimen was removed at the end of Stage III INTOR operations.

V.E. Surface Tests

A surface test program has been defined to provide data on the surface effects of divertor target, armor, limiter, and first wall materials. The plan requires the use of ~5000 materials specimens. Most specimens are 1 × 1 cm, but some larger samples will be required. The 1- × 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. Specimen locations, cleaning method, temperature, material, and fluence levels are varied in the test program.

V.F. Nuclear Tests

A special test channel is not required for neutron streaming or shielding tests. Radioactivity testing will be performed using specimens from the bulk materials test program.

A need has been defined for neutron characterization testing to provide the distribution of the neutron source and determine the radiation field in strategic locations. This and other key neutronics tests require an accuracy of within ~5% to maximize the usefulness of these tests. Passive measurement techniques are not sufficiently accurate. The tests must be performed during early D-T shots before background irradiation builds up and must be run at approximately five orders of magnitude lower power than normal operation to use direct measurement techniques. Two months of calibration tests are anticipated prior to the start of Stage II testing before background radiation levels become too high.

Nuclear tests on INTOR will provide neutronic measurements on a reactor at operating conditions. These data will validate calculational methods and basic data.

V.G. 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 for a DEMO reactor. Achievement of some significant fraction of the component design lifetime in the DEMO imposes availability and lifetime requirements on INTOR.

V.H. Electric Power Generation

Electric power generation at the end of Stage II should be accomplished by using an INTOR breeding blanket sector that has been designed to accommodate high temperature and pressure. This provides a demonstration of electricity production by fusion under reactor-relevant conditions. Simultaneous tritium and electrical power generation in a prototypical DEMO blanket sector could be performed near the end of Stage III.

V.I. Operational Requirements

Some tests will require nearly continuous operation for some period of time. In particular, some of the tritium recovery tests will require $\geqslant$70% duty cycle and continuous operation for one week to one month to reach equilibrium conditions. On the other hand, thermal-hydraulics testing will only require >50% duty cycle and continuous operation for ~1 h.

V.J. Test Schedule

The projected test schedule is shown in Fig. 19. 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 changeout is required. A minimum time of one month between scheduled reactor shutdowns has been established to permit test changeout without unduly impacting reactor availability. Stage III testing will be devoted to longer duration tests that do not require frequent reactor shutdown.

V.K. Installation

A total of three reactor one-twelfth sectors and one divertor slot have been allocated to testing, and these are shown in Fig. 20. 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 changeout.

V.L. Post-Irradiation Examination (PIE) Facility

A PIE facility has been defined for evaluating specimens from bulk materials, 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, mockup of experiments, radiochemistry analysis, materials property testing, and metallography testing.

VI. SAFETY AND ENVIRONMENTAL IMPACT

Although INTOR is in the conceptual design stage, safety considerations are being incorporated into the design to ensure that the safety advantages inherent in fusion are fully realized. The priorities on safety are the safety of the general public, the plant personnel, and the plant itself—in that order.

The safety and environmental considerations have had a significant influence on the design to date, and must be considered more fully in future design efforts. Perhaps most important is the use of a solid breeding material, lithium silicate, in the blanket, eliminating most concerns about lithium fires being a mechanism for radioactive release. Also, the tritium systems have been designed to minimize leakage of tritium to the containment during routine operations, and to minimize the possibilities for accidental release of tritium. Tritium cleanup systems are provided to mitigate the consequences of tritium release to the containment. The containment systems are being designed to ensure that routine and accidental exposures to the public are both within current guidelines and as low as they practically can be. The shield design was conducted such that radiation exposures to personnel will be well below current limits even after many years of INTOR operation. The design requirements for radiation protection are consistent with the guidelines of the ICRP with an appropriate factor of conservatism for design and release uncertainties. Finally, the use of high pressure-temperature water coolant systems was avoided, thereby reducing the likelihood and severity of pipe break accidents.

In the safety analysis, the principal concerns identified were the tritium in the fuel supply and the radioactive inventories in the structures. Approximately 2.5 kg of tritium will be in the fuel system

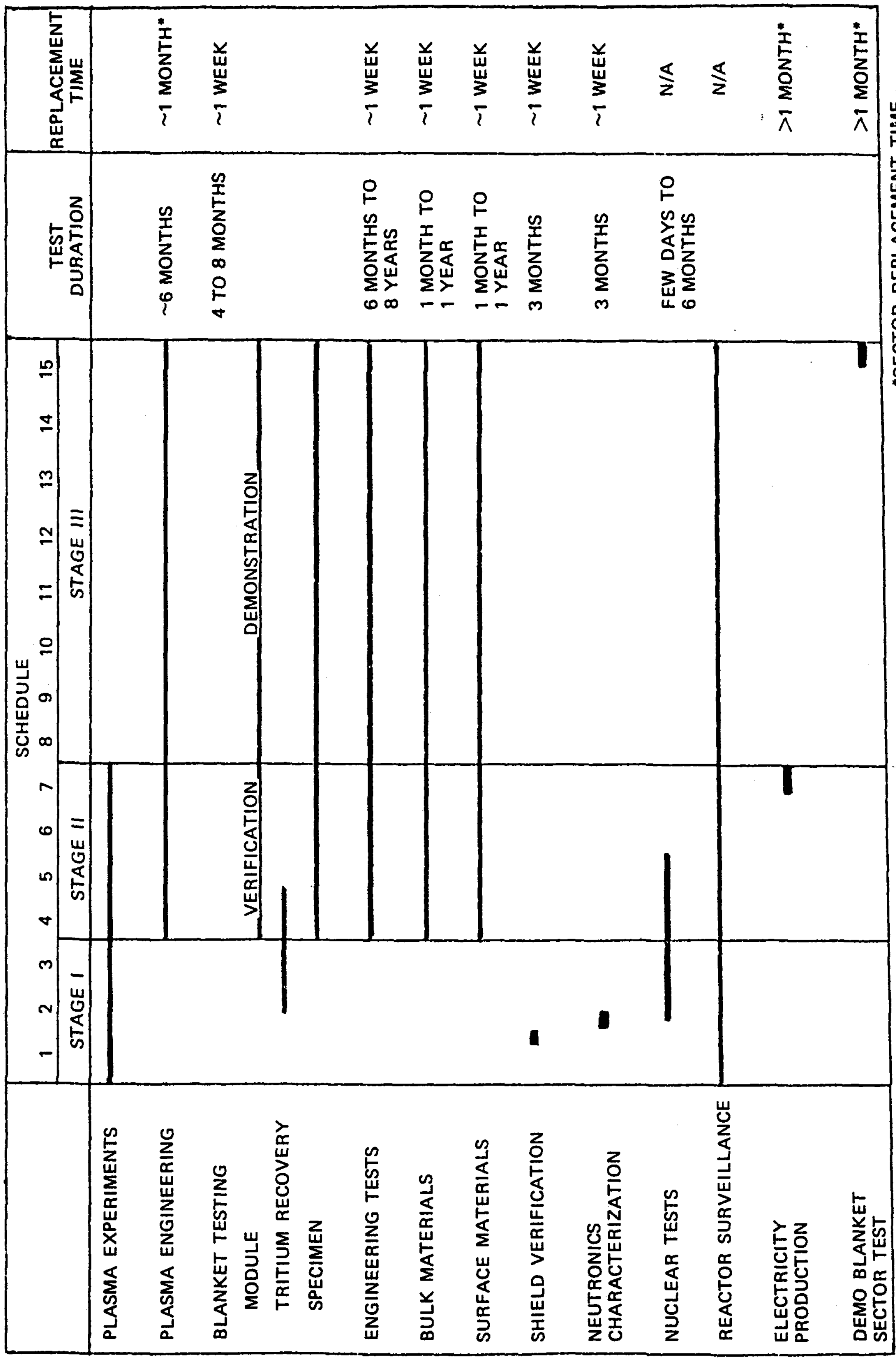


Fig. 19. INTOR test schedule.

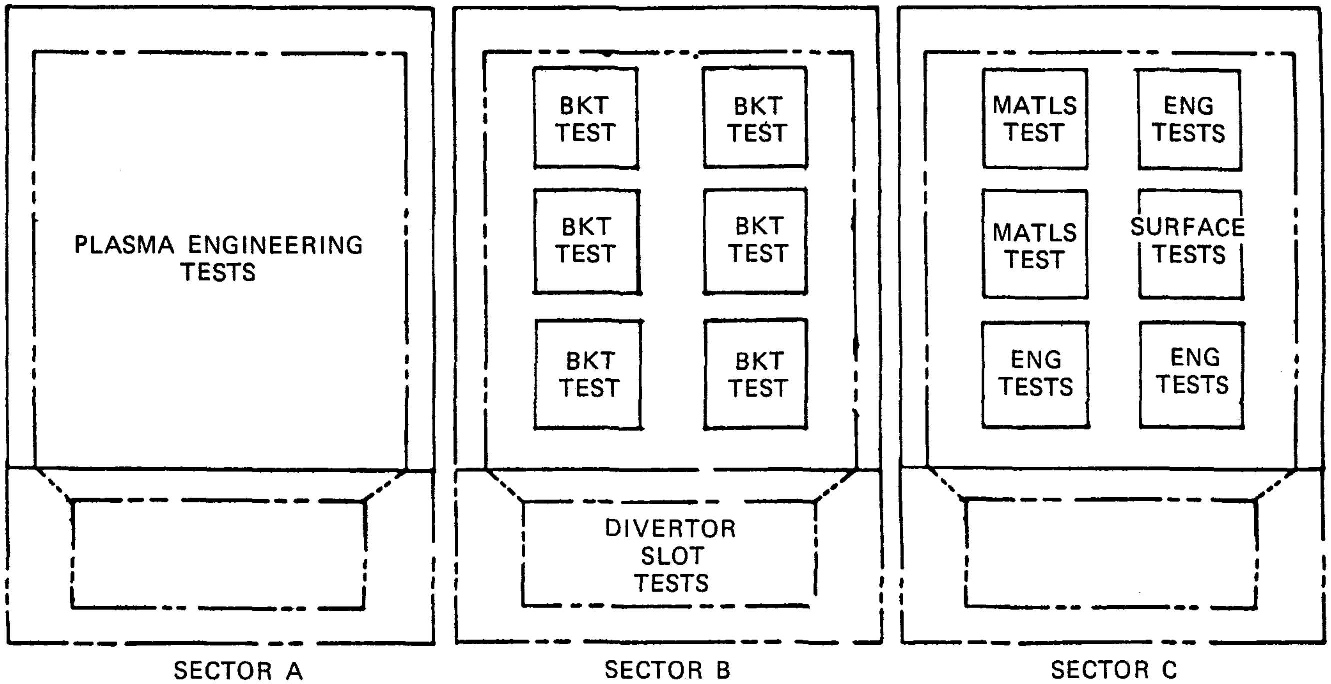


Fig. 20. INTOR test station locations.

and ~1 kg will be in the blanket. The activation product inventories are such that (for example) the first wall alone could contain up to 760 MCi after the first wall design life of 6.6 MW·yr/m^2 is reached. Given these large radioactive inventories, the energy sources in the machine were estimated and evaluated as to their potential for causing radioactive releases. The largest potential energy source was that in the graphite first wall armor in the backup design. Accidental combustion of this graphite in air could result in a release of up to 400 GJ of energy that must be removed by the first wall and blanket cooling system. Next largest were the energy in the TF coil magnetic fields (40 GJ) and the energy (10 GJ) potentially produced if a lithium test module is present and if the lithium were to burn in air. These energy sources were the only substantial sources; no likely way was identified where they could cause a significant release to the environment. The only ways hypothesized for major releases to the environment were seismic events beyond the design basis, a catastrophic failure of the magnet/tokamak system, or human error in design and/or operation. Adequate technology is believed to exist to make the risk associated with releases due to these causes as low as desired when balanced against economic factors. The analysis indicated that the expected normal operational tritium releases of 20 Ci/day would lead to exposures of 10 mrem/yr or less to individuals outside the site boundary. No accidents that would lead to large releases of tritium to the environment were identified. However, as an upper bound case, it was shown that an assumed instantaneous release of the entire vulnerable inventory of tritium (160 g) under worst-case weather conditions would not cause excessive radiation exposure to the public if the site boundary is on the order of a kilometre distant from the release point.

It is important to note that no runaway-type accidents that would affect the public or the plant personnel have been identified by this study.

An environmental impact analysis has also been carried out. This analysis concluded that INTOR will have an acceptable environmental impact when sited with only moderate restrictions on local and regional populations.

VII. COST, SCHEDULE, MANPOWER

VII.A. Cost

The projected direct capital cost for INTOR is \$1.422 billion, as shown in Table XXI. This table presents the costs of each major system of INTOR on the basis of 1981 dollars. Costs were estimated by applying unit cost values or algorithms, i.e., \$/kg, etc., to the total number of units for each system, i.e., kilogram, kilowatts, etc. The total capital cost for INTOR (direct plus indirect) is estimated to be \$2.957 billion. This total includes an allowance of 45% of the direct capital cost for engineering, 15% of the direct capital cost for installation, and 30% of the direct plus indirect cost for contingency.

TABLE XXI

INTOR Cost Estimate

Reactor System	
Torus	$ 62 million
Magnet systems	580
Divertor	32
Neutral beam systems	154
Fueling	5
Support Systems	
Electrical	$277 million
Tritium and fuel handling	21
Cooling systems	48
Instrumentation and control	50
Maintenance equipment	25
Facilities	
Reactor building	$118 million
Other	50
Total Direct	$1.422 billion

VII.B. Schedule

A schedule reflecting design, procurement, fabrication, construction, installation, and preoperational testing for INTOR is presented in Fig. 21. Twelve years are required for the above phases.

VII.C. Manpower

An estimated manpower loading for INTOR is shown in Fig. 22. A peak manpower level (engineering plus construction) of ~2000 persons is required.

VIII. RESEARCH AND DEVELOPMENT

The IFRC has directed the INTOR project to develop specific R&D project recommendations based on the most critical uncertainties that have been uncovered in the INTOR design activity. These projects are to be carried out by the member organizations as part of their overall fusion effort.

The overall scope of the INTOR R&D activity is as follows:

1. Identify the specific R&D projects needed to resolve the most critical uncertainties that have been identified in the INTOR design effort.
2. Identify the research facilities capable of carrying out the required R&D.
3. Develop priorities among the proposed projects.
4. Develop a schedule based on the INTOR schedule.
5. Develop overall program cost information.
6. Develop a set of recommendations to the IFRC based on the above.

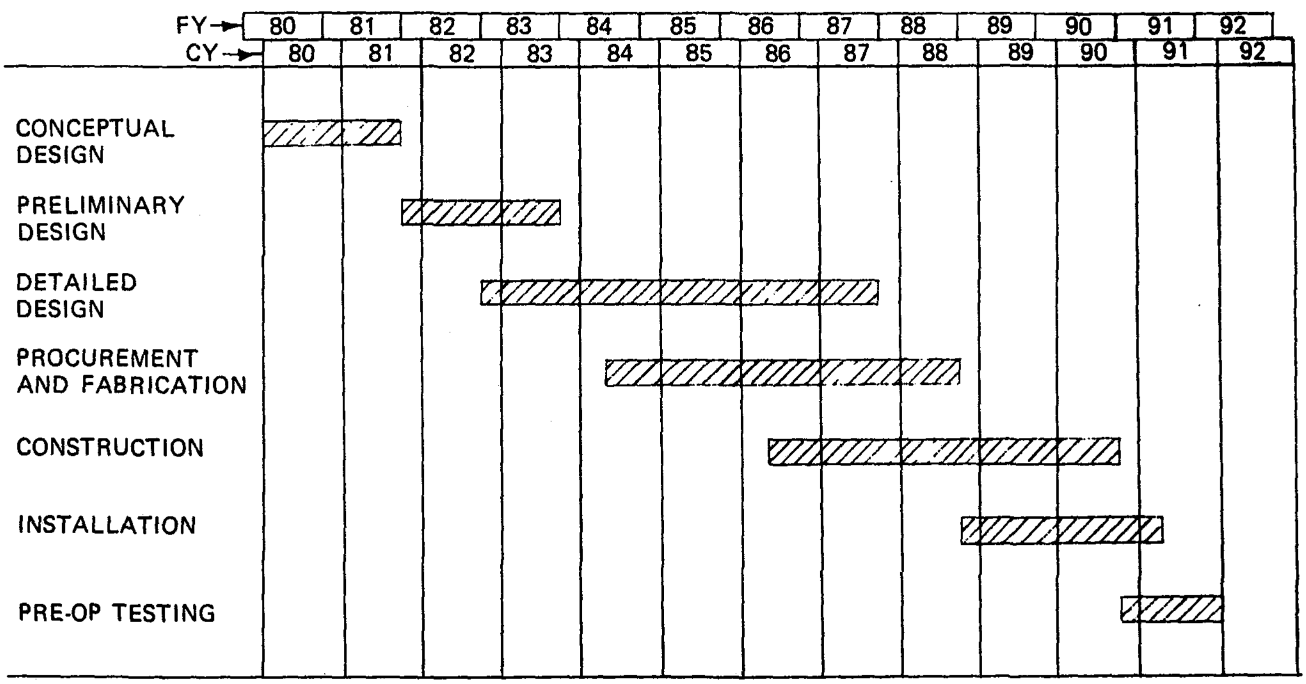


Fig. 21. U.S. recommendation for INTOR program schedule.

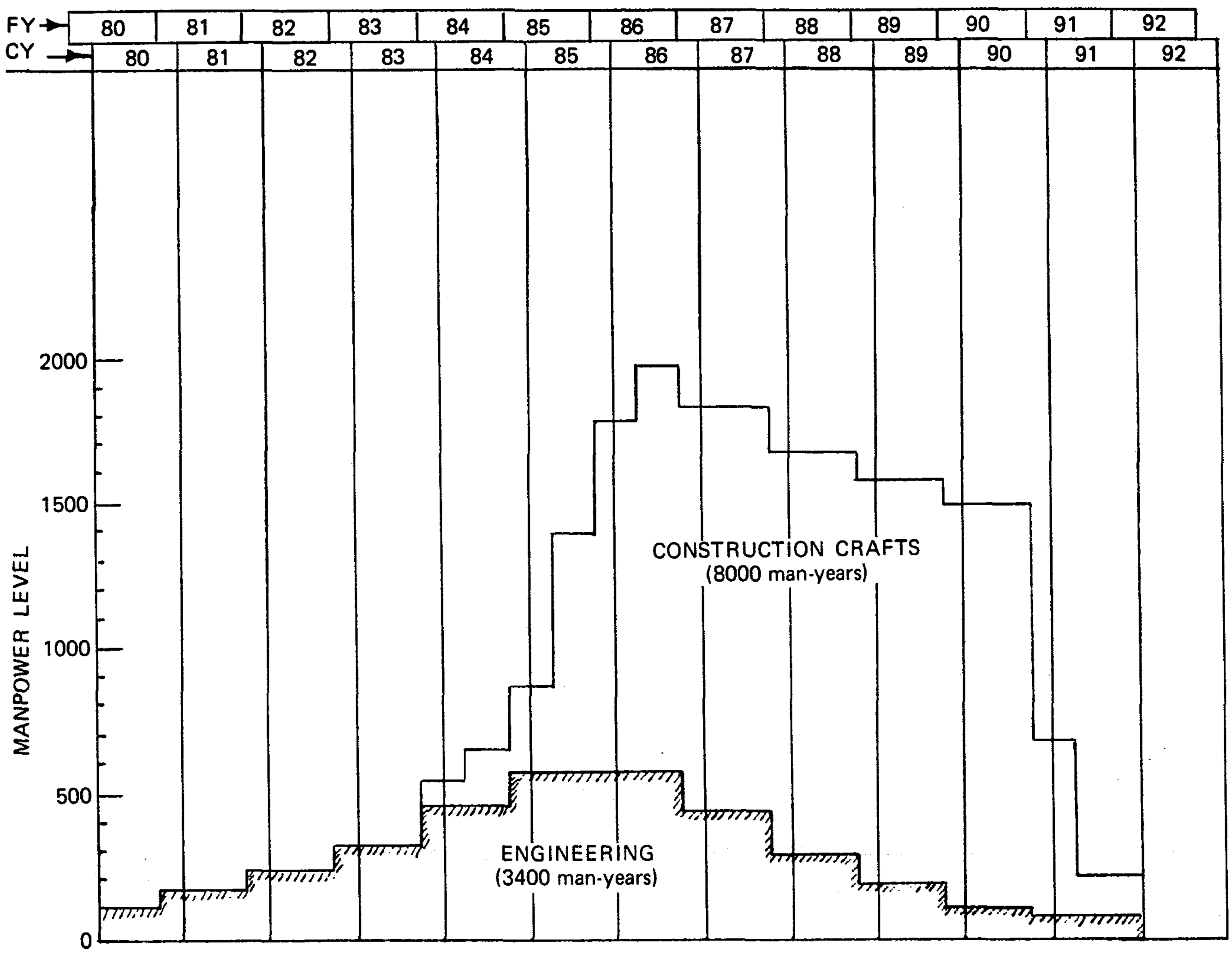


Fig. 22. U.S. recommendation for INTOR manpower spread.

To expedite this program, recommendations will be developed for the August meeting of the IFRC. A preliminary identification of R&D projects has been developed by the U.S. INTOR team in the areas of physics, nuclear, and engineering. The projects are listed in Table XXII. It is emphasized that these are specific projects; the broader R&D needs in support of INTOR were identified previously.[5]

IX. CONCLUSIONS AND RECOMMENDATIONS

A workable conceptual design has been developed for a tokamak that could, with a reasonable degree of confidence, meet the performance objectives specified for INTOR. There is some margin in the design to allow for present plasma physics and technology uncertainties, and R&D programs are either in place or have been identified that 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 such as INTOR have changed considerably over 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 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 analyzing the technical issues within the specific design context, while simultaneously trying to design around the problems, taking the present conceptual design as a base. Such a procedure would involve a reexamination of early design and technical

TABLE XXII

Specific R&D Projects

Physics 1. Beta limit scaling with triangularity 2. High-density divertor operation 3. Divertor channel model tests 4. High-Z impurity backflow from divertor 5. Ripple-induced hot ion transport model tests 6. Ripple-induced thermal transport model tests 7. Disruption characterization 8. Startup voltage experiments 9. Vertical feedback control theory and experiments 10. Pellet fueling studies 11. Current profile control experiments Nuclear 1. Stainless steel behavior during plasma disruptions 2. Graphite tile attachment 3. Chemical sputtering of graphite	Nuclear (continued) 4. *In situ* recoating 5. First wall grooves 6. Solid breeder development 7. Solid breeder/coolant tube interfaces with predictable thermal conductance 8. Compatibility of neutron multiplier and solid breeder materials 9. Tritium permeation 10. Chemical sputtering of refractory metals (tungsten) 11. High-temperature fatigue of tungsten 12. Radiation damage in copper and copper alloys Engineering 1. Fatigue and fracture mechanics data for 4 K coil materials 2. 175-keV neutral beam systems

objectives decisions that have led to difficulties as the engineering design has been carried out on a more detailed level. A second iteration on the INTOR design, within this context and incorporating such new information as may become available from the R&D programs, would be of great value in developing the basis for ultimately designing and constructing the next major experiment in the tokamak program.

The INTOR Workshop has served a useful function in identifying and focusing worldwide attention on specific uncertainties that have a major impact on the design of a tokamak such as INTOR and in identifying specific R&D projects that could be undertaken to resolve these uncertainties. Expansion of the Workshop activities to include active coordination of a limited number of worldwide R&D efforts aimed at resolution of these specific uncertainties would be a significant step toward a meaningful international collaboration in the development of magnetic fusion power.

APPENDIX

U.S. INTOR Organization 1980-1981: Conceptual Design

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Organization Index

ANL	Argonne National Laboratory
B&R	Burns & Roe, Inc.
BNL	Brookhaven National Laboratory
Bechtel	Bechtel National, Inc.
Culham	Culham Laboratory
DOE	U.S. Department of Energy
Ebasco	Ebasco Services, Inc.
EG&G	EG&G Idaho, Inc.
FEDC	Fusion Engineering Design Center, Oak Ridge National Laboratory
GA	General Atomic Company
GAC	Grumman Aerospace Corporation

Organization Index (Continued)

Ga. Tech	Georgia Institute of Technology
GE	General Electric Company
HEDL	Hanford Engineering Development Laboratory
LANL	Los Alamos National Laboratory
MDAC	McDonnell Douglas Astronautics Company
MIT	Massachusetts Institute of Technology
Mound	Mound Laboratory
ORNL	Oak Ridge National Laboratory
PPPL	Princeton Plasma Physics Laboratory
SNLA	Sandia National Laboratory, Albuquerque
SNLL	Sandia National Laboratory, Livermore
W	Westinghouse Electric Corporation
U. Wis.	University of Wisconsin-Madison

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This paper is based on the work of more than 100 people in the U.S. fusion community (see Appendix).

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