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MASTER

INTOR - A FIRST-GENERATION TOKAMAK EXPERIMENTAL REACTOR

FUSION RESEARCH REPORT
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ABSTRACT

An intensive, year-long, international evaluation of the next major tokamak beyond the generation of large experiments currently under construction was carried out during 1979. This evaluation consisted of the definition of objectives, an assessment of the physics and technology base and R and D needs and the identification of a set of parameters that physically characterize the machine.

1. INTRODUCTION

The International Tokamak Reactor Workshop was held under the auspices of the IAEA during 1979 to develop an international consensus on the objectives, characteristics, technical and scientific basis, and feasibility of constructing the next major experiment in the tokamak program beyond the next generation of large tokamaks (JET, JT-60, TFTR, T-15). Specifically, the Workshop:

(1) assessed the plasma physics and technological bases that now exist or could be expected to exist in the early 1980s;

(2) identified R and D requirements that will not be met by existing programs;

(3) defined objectives;

(4) identified a plausible set of physical characteristics that are consistent with the objectives and with the projected data bases; and

(5) made a recommendation on the feasibility of constructing such a device to initially operate about 1990.

Parties to the INTOR Workshop were the European Community, Japan, the Soviet Union and the United States. Each party was represented by four participants, who met together four times (for a total of 10 weeks) to define the effort, to review and discuss the contributions of the four parties, and to prepare the report [1] of the Workshop. These participants guided the work of more than 100 senior scientists and engineers in each participating country working in their home institutions to develop material for the Workshop.

The INTOR Workshop concluded that [1] "A substantial physics and technology data base for INTOR exists today, and this data base will be expanded over the next few years by currently planned programs. However, certain crucial information will not be developed by currently planned programs. Much of this missing information could be developed on the INTOR time scale by the expansion and/or acceleration of existing R and D programs and by the establishment of new R and D programs. On this basis, it is concluded that it is scientifically and technologically feasible to undertake the construction of INTOR to initially operate about 1990, provided that the supporting R and D effort is expanded immediately to provide an adequate data base within the next few years in a number of important areas. Furthermore, it is concluded that the construction of an INTOR-like device to operate in the early 1990s is the appropriate next major step in the development of fusion power."

The second phase of the INTOR activity, an 18-month definition phase, began in January, 1980. The objective of the definition phase is to produce a preconceptual design of INTOR. The initial focus of this definition phase is upon the resolution of certain major design issues which were identified during the INTOR Workshop in 1979.

This paper summarizes the objectives and characteristics of INTOR, the assessment of the data base and R and D requirements, and the major design issues to be resolved.

2. INTOR OBJECTIVES AND CHARACTERISTICS

2.1. Objectives

If successful, the next generation of tokamak experiments (JT-60, TFTR, JET and T-15) will provide fairly conclusive information on how to achieve ignition plasma parameters. The establishment and sustainment of a burning D-T plasma seems, therefore, in practical reach. In order to meet the requirements of fusion reactor operation, certain technologies used in previous experiments for heating and maintaining the plasma must either be significantly advanced or replaced by new technologies. In addition, fusion power demonstration requires the development and demonstration of new technology for tritium and electricity production by fusion, the development and testing of materials, the development of reliable components, the demonstration of net electricity production and net tritium breeding and the demonstration of reliable and economic operation of a fusion reactor.

This agenda sets an urgent need for a machine that demonstrates the plasma physics required for fusion power and that provides for the development, testing and, to some extent, demonstration of the required technologies. INTOR is intended to fulfill these tasks. In particular, INTOR must develop the technologies needed for a fusion power demonstration reactor (DEMO), which would then demonstrate net electrical power, tritium breeding and economic feasibility. INTOR should be of the tokamak type since the data base is largest and the stage of development is most advanced for this confinement concept and since the gross characteristics of the technologies to be developed for magnetic confinement fusion are relatively independent of the choice of confinement concept. These considerations led to the following programmatic objectives for INTOR:

(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 physics requirements for DEMO.

(3) INTOR should demonstrate those technologies required for DEMO that must be incorporated as an intrinsic part of the reactor (e.g. superconducting magnets, remote handling technology).

(4) INTOR should be a test facility for blanket, tritium production, plasma engineering, materials and other technology developments required for DEMO.

(5) INTOR should serve as a test and demonstration facility for magnetic fusion technology in general.

(6) INTOR should demonstrate the reliable operation of a fusion reactor.

The technical objectives of INTOR then follow from the programmatic objectives and the physics and technological data base. These objectives are given in Table 1.

2.2. Characteristics

The concept for a machine that is capable of meeting the technical objectives of INTOR and that is supported by the data base assessment has been developed by the Workshop. The concept recommended as input for the definition phase is characterized by a relatively large ($\sim$280 m^3) D-shaped plasma heated to ignition by neutral beam injection power of about 75 MW for 5-6 s up to a mean temperature of about 10 keV. Controlled burn should then be achieved for more than 100 s at a plasma-to-magnetic pressure ratio $\langle\beta\rangle = 5\%$, which is marginally stable theoretically. A divertor is needed to exhaust helium and control impurity influx in order to achieve this long of a burn. The plasma current will exceed 6 MA, which is quite sufficient to confine fast alpha-particles.

Table 1
INTOR Technical Objectives

I. Reactor-Relevant Mode of Operation
  a) Ignition of a D-T plasma.
  b) Controlled $\gtrsim$ 100 sec burn pulse.
  c) Reactor-level particle and heat fluxes ($P_n \gtrsim 1\ \mathrm{MW/m^2}$).
  d) Duty Cycle $\gtrsim$ 70%.

II. Reactor-Relevant Technologies
  a) Superconducting toroidal and poloidal coil technology.
  b) Plasma composition control (e.g. divertor) technology.
  c) Plasma power balance control technology.
  d) Plasma heating and fueling technology.
  e) Tritium fuel cycle technology (excluding breeding and extraction).
  f) Remote maintenance technology.
  g) Vacuum technology.
  h) Fusion power cycle technology.

III. Engineering Test Facility
  a) Tritium breeding blanket and extraction technology testing.
  b) Advanced structural and breeding materials, coolants, etc.
  c) Blanket technology for simultaneous electricity production and tritium breeding.
  d) Materials testing.
  e) Advanced plasma engineering technology testing.

IV. Demonstration
  a) Electricity production by fusion.
  b) Tritium production by fusion.
  c) Safe and reliable operation of a fusion reactor.
  d) Availability $\sim$25 - 50%.

Both the toroidal and external poloidal field coils will be superconducting. INTOR will require about 200 MW_e power. The neutron power load on the first-wall will be about 1.3 MW/m^2 (with an option at 1.6 MW/m^2), and the total fusion power will be about 620 MW_t. A stainless-steel, water-cooled shield operating with relatively low (100-200°C) maximum structural temperature will be employed on the basic machine. The major suggested parameters for INTOR are given in Table 2.

Since a principal function of INTOR is to demonstrate the feasibility of reactor components and to serve as a test facility for technology development, simplicity and reliability of the basic machine are of the utmost importance, even if this is in conflict with certain aspects of reactor-relevance. The technical feasibility of INTOR is then intimately related not only to the supporting technology development programs but also to the manner in which each technology is incorporated into INTOR.

Certain reactor-relevant technologies must be incorporated as intrinsic - not easily replaceable - components (e.g. superconducting magnets, remote handling system) in order to be tested in INTOR. These technologies must be developed to a high level prior to the construction of INTOR, since failure of such intrinsic components would cause INTOR either to fail or to be significantly delayed in achieving any of its objectives. For such intrinsic technologies, INTOR will demonstrate an already-developed technology in a fusion reactor environment.

With certain other reactor-relevant technologies (e.g. electricity and tritium producing blanket), there exists the option of incorporation as intrinsic components or as relatively easily replaceable test components. Inclusion of a new technology as an intrinsic component at

Table 2
INTOR Suggested Parameters

Major radius, R (m)	5.2
Plasma radius, a (m)	1.3
Chamber radius, r_w (m)	1.4
Elongation, κ	1.6
Burn time (s)	≥ 100
Duty cycle (%)	≥ 70
Burn average $\langle\beta\rangle$ (%)	$5(6^a)$
Av. DT density $\langle n_i \rangle$ (m^{-3})	$1.4(1.5^a) \times 10^{20}$
Av. ion temperature $\langle T_i \rangle$ (keV)	10
Plasma current, I (MA)	6.4
DT thermal power, P_{th} (MW)	$620(750^a)$
Neutron wall load, P_n (MW/m^2)	$1.3(1.6^a)$
Lifetime (no. of burn pulses)	$5 \times 10^5 - 10^6$
NB heating power, P_B (MW)	75
NB energy, E_B (keV)	175
Fueling	pellet, gas puffing
Impurity control	divertor
Toroidal field coils	NbTi and/or Nb_3Sn
Toroidal field at centerline, B (T)	5.5
Poloidal field coils	NbTi
Tritium inventory (kg)	3
Maximum availability (%)	25 - 50
Tritium consumption during Stage II (kg/yr)	6

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a) For 1.6 MW/m^2 option.

the outset places a greater demand upon the development program supporting that technology and entails an increased risk that failure of that component will cause INTOR to fail or to be significantly delayed in achieving any of its objectives. On the other hand, inclusion of a new technology as a test component that could be inserted after INTOR had operated for some time and that could be relatively easily replaced places less of a demand upon the supporting development program and entails considerably less risk that component failure will significantly delay INTOR in achieving its other objectives. A further advantage of inclusion of a new technology in test components is that INTOR can serve as a test bed for the development of that technology, with successively improved test components being tested in INTOR.

The philosophy of INTOR is to minimize the risk of failure and the demands upon the supporting technology development programs by including new technologies in test components to the maximum extent that is consistent with achieving the programmatic and technical objectives. This philosophy leads to the concept of staged operation, with the possibility of incorporating some of the technologies that have been tested in test components during an early stage into intrinsic components during the last stage in order to provide a full-scale engineering demonstration.

The objectives of INTOR could best be achieved in a sequential order by staged operation. During Stage I, which might encompass the first three years of INTOR operation, the first two sets of technical objectives listed in Table 1 would be achieved, i.e., a reactor-relevant mode of operation would be established and some intrinsic reactor-relevant technologies would be demonstrated. The production of 5-10MW_e in nonbreeding blanket modules could be achieved towards the end of this first stage.

Stage II, which would encompass the following four years, would then be fully devoted to engineering testing, with an availability goal of ∿25%. This would include the testing of blanket modules, of tritium production and extraction, of plasma engineering technologies (e.g. rf launchers), and of materials properties. An important criterion for INTOR during this stage is to provide sufficient flexibility to accomodate a variety of experimental programs at the same time. This would allow the simultaneous testing of various blanket concepts, with different structural and breeding materials, coolants and design concepts, together with the testing of plasma engineering technologies. The reliability of many components would be tested during this stage, and improved components could be installed in some instances. High temperature coolants from tritium-producing blanket modules could also be used to demonstrate simultaneous electricity production and tritium production.

Stage III of INTOR operation would be determined by the choice of alternatives for the utilization of the machine and would be of 5 years or more duration. In option III-A, stage III would be a straightforward continuation of stage II with accelerated engineering testing in test modules. There would also be a strong emphasis on achieving the highest possible availability (∿50%) and on operating at a high neutron wall load (1.6 MW/m^2) to accumulate a significant neutron fluence for materials irradiation testing and to test component reliability. This option would allow for the uninterrupted continuation of engineering testing in INTOR, but would be demanding of component reliability and redundancy.

Option III-B would be predicated upon the assumption that the manifold engineering constraints make it necessary to partially rebuild the machine to test a full blanket sector before a concept can be applied to DEMO. In the extreme, option III-B could be expanded to allow a full blanket replacement (probably excluding the inboard part), which would provide for the demonstration of substantial tritium (kg level) and electricity (100 MW_e level) production in a reactor-relevant way. The selection between the options A and B for stage III can be made at a later time, provided the design of INTOR offers a high enough modularization and flexibility for keeping the choices open.

3. PHYSICS

3.1. Physics Status Assessment

For the parameter range of present-day tokamak devices, there is a well-documented experimental data base on energy confinement. The overall energy confinement time τ_E is found to obey an empirical scaling law of the form

$$\tau_E(s) = 5 \times 10^{-19}\ \bar{n}_e(cm^{-3})a(cm)^2 \quad ,$$ with the dominant

losses occurring through anomalous (i.e., non-classical) cross-field electron thermal transport. Ion thermal transport is found to be within a factor of about three of classical predictions. Uncertanties exist in the extrapolation of these scaling laws to the high-temperature regime to be reached in INTOR. In particular, there are sound theoretical reasons for expecting some degradation in confinement at high temperature; however, the highest-temperature experiments to date (notably in the PLT tokamak) have given some indications of a slight improvement in confinement with rising temperature. Additional experiments in present tokamak facilities with high-power neutral-beam heating, as well as the next generation of tokamaks, will extend the data base on confinement further toward the INTOR regime in the early-to-mid 1980s.

As part of the data-base assessment, each of the participating countries carried out studies of the requirements for reaching ignition in INTOR, on the basis of one-dimensional transport codes which model in some detail the particle and energy balance in a tokamak. Adopting a model for electron thermal transport based on the empirical scaling law for energy confinement discussed above, and assuming ion thermal transport three times larger than classical, it was found that ignition could be achieved in a pure DT plasma of minor radius in the midplane $a = 1.2$m, vertical elongation factor $\kappa = 1.6$, and field strength $B = 5.0$T on axis. (The final suggested parameters, namely $a = 1.3$m and $B = 5.5$T, represent about 60% better $n\tau_E$ capability at fixed β.) Some typical results of these calculations are shown in Fig. 1, for the case where the plasma is heated to ignition by 60MW of neutral beam power at 150-200 keV energy injected nearly perpendicular to the main magnetic field. Shown in the figure are the value of $< \beta > \equiv 8\pi <p>/B^2$ at ignition, and the heating pulse t_{ig} required, for a range of relevant plasma densities $<n_e>$. (The final suggested parameters of the neutral beam system, namely 75 MW at 175 keV with a 10 sec pulse length capability, represent some increment above the minimum theoretical requirements.)

Figure 1 shows that ignition occurs at $<\beta> \simeq 4\%$, a value that includes contributions to the pressure from beam ions and alpha particles. The plasma temperature and $<\beta>$ - value rise during the burn phase, due to the increase in reaction rate with increasing temperature, until some mechanism of burn control becomes effective. Over a fairly broad range of relevant temperatures, the neutron wall loading that can be achieved in the burn phase is proportional to $\beta^2 B^4 a$. For INTOR, assuming parabolic density and temperature profiles, and taking B = 5.5T, a burn-phase value $<\beta> = 5\%$ (including a contribution of 1.5% from fast alpha particles and thermalized helium) provides the minimum specified neutron wall loading of 1.3 MW/m^2. Theoretical limitations on $<\beta>$, arising from pressure-driven MHD "ballooning" and "kink" instabilities, permit stable equilibria at least up to $<\beta>> = 4\%$, with $<\beta> \simeq 4.5\%$ probably attainable with fully optimized profiles. Thus, in its burn phase, it seems likely that INTOR will operate slightly into the theoretically-unstable regime, a regime that has already apparently been attained in the TSX-B and T-11 tokamaks. High-power injection experiments will explore the $<\beta>$ - limits in D-shaped plasma cross-sections (i.e., configurations similar to those planned for INTOR) in the early 1980s.

Adequate control of the shape of the plasma equilibrium can be provided by a poloidal field design in which most, if not all, poloidal field coils are exterior to the toroidal field coils, (a configuration that has considerable advantages from the point of view of assembly, disassembly, and maintenance). Such a poloidal field design can also provide feedback control of the instability in which the plasma column as a whole tends to be displaced vertically, provided the growth time of such a mode is lengthened to at least 20msec by eddy currents induced in the blanket/shield structure, as seems likely to be the case.

Neutral-beam injection was selected as the primary heating option for INTOR, although the injectors required represent a significant step beyond those that are being developed for the TFTR-generation of devices, as indicated in Table 3. In particular, to achieve pulse lengths of 10 secs, it will probably be necessary to develop direct recovery systems (i.e., recovering the energy of the unneutralized beam ions), both to achieve an acceptable overall power efficiency, and to minimize power handling requirements on the ion beam dump.

RF heating will be retained as an alternative to neutral beam heating in INTOR, because of its potential for technological advantages that will become even more important in a commercial reactor. At the minimum, provision should be made for the testing in the reactor environment that will be present in INTOR of whatever RF technique is most successful in forthcoming megawatt-level RF experiments on present tokamaks. Provision of supplemental auxiliary RF heating at the 5-10 MW level (especially electron cyclotron heating at about 140 GHz) might be necessary in the start up phase.

Table 3
Neutral Beams for INTOR Compared with the
4-Injector TFTR System

	TFTR	INTOR
Beam Energy (keV)	120	175
Neutral Beam Power (MW)	33	75
Pulse Length Capability (secs)	1.5	10*
Monatomic Ion Fraction	0.85	≈ 0.9
Full Energy Fraction of NB Power	0.76	~ 0.8
Direct Recovery Efficiency (%)	0	~ 60
Overall Power Efficiency (%)	36	~ 30
Power Density at Port (kW/cm^2)	3.0	~ 2.5

*Plus a capability to restart 10 MW during the burn.

The start-up phase of INTOR is illustrated (very schematically) in Fig. 2. Computer calculations, calibrated against present-day experimental results, show that purely-ohmic start-up in INTOR can be accomplished at low density (and very low impurity level) with a peak one-turn voltage around the plasma of only 100 volts, but such a voltage is about the limit of what can be provided by a superconducting induction coil of dimension constrained by the overall dimensions of INTOR, and consistent with known limits on $\dot{B}$ at the coil. Thus, there is considerable incentive for lowering the start-up voltage by auxiliary electron cyclotron heating applied for the first approximately 100m sec as indicated in Fig. 2, provided this does not inhibit current penetration into the plasma by exacerbating the skin effect.

Control of the plasma density, density profile, and species mix, to ensure a roughly 50 : 50 DT plasma will be important in INTOR. The optimum path to ignition (the "variable density start-up" case shown in Fig. 1) involves neutral beam heating of a target plasma with density $< n_e > \sim 7 \times 10^{13}$ cm^{-3} (beginning at t ~ 5 secs in Fig. 2), and then increasing the density to $< n_e > \sim 1.4 \times 10^{14}$ cm^{-3} by the end of the heating pulse when ignition occurs (t ~ 10 secs in Fig. 2). In fact, the injected beam particles will themselves provide essentially all of this density increase, although these particles will be entirely deuterium. Thus, the target plasma must be almost entirely tritium. On the basis of an empirical law characterizing present-day experiments, the maximum density achievable in ohmically heated plasmas with INTOR parameters is expected to be $< n_e > \sim 3 \times 10^{13}$ cm^{-3}, substantially short of the density $< n_e > \sim 7 \times 10^{13}$ cm^{-3} required at the beginning of the neutral beam pulse to prevent excessive "shine-through" of 150-200 keV beams. Thus, it may be necessary to apply some further auxiliary heating (shown from t ~ 3 secs to t ~ 5 secs in Fig. 2), such as an additional lower-energy injector or an RF system (electron cyclotron heating would be especially attractive), during a phase in which the tritium density is increased substantially, before the main heating pulse begins.

The requirement to confine energetic ions (thermal ions, beam ions, and alpha-particles) imposes severe requirements on the exact degree of axisymmetry of the toroidal confining field, i.e., on the "field ripple" that necessarily arises due to the discrete toroidal field coils. Moreover, our understanding of the physical loss processes associated with field ripple is still evolving. With respect to confinement of thermal ions, it seems that the INTOR requirements (specifically, that the ripple contributions to cross-field ion thermal conductivity should not exceed about three times the classical thermal conductivity), can be met by a 12-coil toroidal field system with coils of 8m horizontal bore and 10m vertical bore. Coils of about this size are needed, in any case, for accessibility reasons. The effect of the ripple on beam ion confinement is less clear, but preliminary calculations indicate that ripple losses could exceed 15% of the injected beam energy, unless the injector is inclined at an angle of at least 15° to the perpendicular.

If the field ripple could be adjusted upward (to about 1.2%, peak-to-average, at the plasma edge), it would provide a significant mechanism for energy loss, and one that increases sharply with increasing temperature. Accordingly, variable field ripple could provide an attractive technique for "burn control", i.e., for preventing "runaway" of the plasma temperature due to the increase in reaction rate with increasing temperature. Alternatively, one could operate in a non-ignited (but very high Q) mode, with the burn controlled by control of the input power.

Present-day tokamaks are subject to plasma "disruptions", in which the plasma energy is dumped rather suddenly upon the vacuum vessel, and the plasma current is transferred either to the vacuum vessel or to nearby poloidal field coils. Disruptions occur most frequently when the discharge parameters (density and q-value) are pushed toward their limits. However, even in "disruption-free" modes of operation, there is a finite probability of a disruption occurring, typically of order 10^{-2} but varying from facility to facility. In recent years, there has evolved a rather good physical understanding of the processes that lead to a disruption. Specifically, the plasma current profile is believed to evolve to one that is unstable to several resistive MHD modes, which interact with each other in a way that quickly destroys confinement. In INTOR, plasma disruptions will produce very severe thermal loads upon the first-wall/ limiter/divertor-plate, as well as severe electro-mechanical effects. Active magnetic feedback of the precursor MHD modes is theoretically possible, but the needed feedback coils, close to the plasma, would present very severe practical difficulties in an INTOR-like reactor environment. However, by appropriate observations of the magnetic perturbations associated with the precursor modes, it may be possible in a very large device such as INTOR to anticipate a disruption in sufficient time to take effective counter-measures against the consequences of the disruption, e.g., by inserting some sacrificial limiter to protect the main limiter-and/or-first-wall coating. Certainly, one of the main objectives of the initial stage of INTOR operation will be to learn how to operate in a mode that minimizes the occurrence of disruptions, in the hope that the disruption probability can be reduced to of order 10^{-3} in the later engineering-test stages.

Impurities represent the most serious threat to the achievement of ignition and a long burn pulse in INTOR, since the surface heat load on the first-wall and/or limiter will be much higher than in present-day tokamaks. Even a very slight contamination of the plasma by partially-stripped high-Z impurity ions leads to substantial radiation losses, and contamination by low-Z impurity ions at levels typical of many present-day tokamaks leads to severe depletion of the reacting DT fuel, at fixed β. Moreover, if the helium produced by the DT reactions is all retained in the plasma, the burn in INTOR will be quenched after about 30 secs, unless the $< \beta >$ - value can increase significantly above 6%. Although several non-divertor schemes for impurity control were proposed (and should be investigated further) the INTOR Workshop was unanimous in recommending that some type of magnetic divertor be incorporated into INTOR.

It is important to understand the functions that a divertor is intended to perform in a tokamak reactor such as INTOR. First, as we have seen, the divertor must exhaust helium at the same rate that it is created by fusion reactions. However, for the expected helium particle confinement time of about 2 secs, only about 10% of the helium that recycles at the edge of the plasma must be pumped away, to maintain an acceptable level of helium contamination. Second, the divertor should remove a large fraction of the power which would otherwise be deposited on a limiter, thus protecting the plasma from impurities generated at the limiter. Divertors can, in principle, also provide for exhaust of DT ions. However, to achieve an acceptable burn-up of the tritium fuel in a reactor tokamak such as INTOR, it is desirable for the DT neutrals to recycle back from the divertor to the main plasma chamber. In this case, it will not be possible to prevent bonbardment of the first wall by energetic DT neutrals produced by charge exchange. Accordingly, it seems likely that INTOR must be equipped with a low-Z coated first-wall, to minimize the effects of charge-exchange sputtering.

The selection from among the various alternative divertor concepts for INTOR will have a profound effect on other aspects of the design of the device, and is discussed in section 3.3.

3.2. Physics R and D Needs

The INTOR Workshop identified three major research and development areas in which the present world-wide program is inadequate, and a substantial increase in effort will be required to meet the needs of INTOR, namely:

- Impurity control; divertor physics and technology
- Disruption detection and control
- Neutral beam heating technology

The Workshop identified a number of other areas in which the needed research and development effort can probably be carried out as part of presently-planned programs on existing or authorized facilities, namely:

- Energy and particle confinement, especially ripple effects and helium transport
- Limits on β, and effect on transport
- Plasma shape and profile control, especially current profile control
- RF physics and technology, needing an accelarated program if near-term experiments are successful
- Fuelling
- Start-up, shutdown, and burn control

3.3. Physics Major Design Issues

The INTOR Workshop examined various alternative divertor concepts, and a number of divertor designs were carried out specifically for INTOR.

The bundle divertor, in which the main confining field is diverted in a single local region of the torus, is attractive from the point of view of assembly and maintenance. The main difficulty, in an INTOR-sized device, is that the requirements for achieving acceptably low field perturbation (ripple) on axis are necessarily in conflict with the requirement to provide adequate space for shielding the copper divertor coil(s). In bundle divertor designs proposed for INTOR, the best that could be achieved was about 40 cm for shielding and about 1% field perturbation on axis. Moreover, a copper-coil bundle divertor would consume about 100 MW power. The Workshop concluded that a satisfactory bundle divertor design for INTOR had not been developed.

Conventional poloidal divertors maintain the axisymmetry of the magnetic configuration, and employ an interior poloidal field coil to divert the field lines into a divertor chamber that passes all around the major circumference of the torus. However, the interior divertor coil was considered unworkable in the INTOR reactor environment. On the other hand, a poloidal divertor design with all poloidal field coils outside the toroidal field coils, as shown in Fig. 3, seems to be a viable concept for INTOR. Such a concept provides only "weak" divertor action, in that the exhaust channels are short and the divertor plates are quite close to the main plasma. Nonetheless, calculations indicate that a divertor of this type should be capable of performing the minimum functions required in INTOR, namely to exhaust helium and to protect the plasma from limiter-generated impurities. Indeed, the plasma scrape-off, even in a relatively short exhaust channel, should be effective in preventing helium and impurity atoms from returning (recycling) from the divertor to the main plasma chamber, at the same time permitting DT atoms (more energetic due to charge exchange in the scrape-off) to recycle relatively freely. Thus, a poloidal divertor of this type seems to offer a workable solution to the potentially most severe impurity problems in INTOR, and the Workshop agreed that near-term design activity should be based on this concept.

Two additional critical design issues were identified by the Workshop, burn control and disruption suppression. An adequate mechanism for maintaining a more-or-less constant power burn against the effects of thermal runaway and, possibly, limited impurity accumulation must be identified. At present, variable toroidal field ripple seems most promising. While disruptions can probably not be altogether avoided, it should be possible to limit the frequency to 10^{-3} and to mitigate the consequences.

4. ENGINEERING

4.1. Engineering Status Assessment

Any engineering embodiment of the objectives and physics considerations outlined above requires appreciable design sophistication and advanced technology. Novel, sometimes severe, design criteria will require innovative approaches to the design of individual systems.

The close coupling of all reactor systems implies subtle and complex design management problems. Nonetheless, engineering solutions which support the objectives of INTOR exist.

4.1.1. General Engineering Design

Each engineering specialist perceives INTOR in the context of his own specialty. To the systems integrator, charged with the responsibility for optimizing the overall design, perhaps the most striking feature is the sheer diversity of interactive effects and systems which will be simultaneously operative in the reactor. Structural loads result from an unusually large, coupled array of electromagnetic and thermal effects. The number of advanced technologies which must be combined is large; e.g. superconducting magnet, high vacuum, cryogenic, tritium handling, and nuclear energy conversion systems. High levels of neutron flux and fluence place unusual burdens on materials technology, while reactor availability requirements call for shrewd application of remote maintenance capabilities. Thus, the analytical structure which serves as the basis for trade-offs affecting absolute performance level, reliability, safety and cost is complex.

Nonetheless, certain key observations can be made, first, a physical characterization exists which self-consistently supports the spacial and functional requirements specified in section 2. Figure 3 represents one of several possible approximations to the design solution. The space allocations are based on design and construction experience for the various systems shown and are judged practicable. For some reviewers, the components might appear a bit "squeezed" toward the central vertical axis of the machine, thus implying higher peak mechanical stresses, higher refrigeration losses, and more difficult remote maintenance procedures than would be necessary in a larger design. However, enhanced plasma performance and lower cost for most systems must drive the layout toward this compaction of inboard systems.

A second observation is that remote maintenance requirements profoundly affect the reactor's layout and the specific design of nearly all components. This situation follows from the ambitious reactor availability target and simply from the prudent requirement that one be able to repair the device after initiation of D-T operations. As part of the assessment activity, a systems interface matrix was generated. The elements of the matrix provide a rough quantitative indication of the affect the design of a given system has on the design of other systems. Remote maintenance systems stood out as the most interactive of all, thus reinforcing the belief that remote maintenance considerations must be an integral part of the design from the onset of **preconceptual** activity.

A third observation, closely coupled to the previous discussion on reactor availability and remote maintenance, concerns the relative placement of toroidal and poloidal magnet systems. Placement of all poloidal coils outside of the bore of the toroidal coil is a strongly preferred option even at the costs of substantially increased poloidal coil system stresses and power requirements. Internal poloidal coils cause severe topological problems which would inhibit experimental operations with test modules and complicate remote maintenance procedures. Particularly unacceptable are the placement of poloidal divertor coils in or near the vacuum vessel as per scale-up of present experimental devices. The radiation environment is considered too hostile for these room temperature pulsed coils; the reliability of the divertor system would be poor and the size and topological complexity incompatible with fast remote maintenance. Fortunately, new simplified concepts for divertors have been put forth in the past year as discussed in section 3. The designs take advantage of the finding that the long pulse operation of the reactor does not require an "efficient" impurity divertor; the key seems to be provision of a less efficient system which serves principally to act as a helium ash pump. These simplified concepts, some of which do not require placement of coils within the toroidal coil, are being further developed.

One positive finding of the INTOR systems integration assessment is worth mentioning even though it does not relate to any single technical issue. There already exists in the world the design management techniques and experience to successfully conduct a well-integrated design activity. This base exists largely outside of fusion research but can be adapted to the needs of INTOR.

4.1.2. Magnetics and Power Systems

It is the use of large magnetic systems which gives fusion power both a unique conceptual elegance and many practical problems. Despite the size and complexity of the magnetic systems, the outlook is bright. Research and development activities throughout the world are properly addressing the relevant technological issues and, with a modest scale-up of effort, will be capable of providing the required data base for INTOR. The discussion of magnetic systems naturally divides itself into the three categories of toroidal, poloidal, and divertor systems.

In order to meet the field and pulse length requirements for INTOR, it is clear that the toroidal coil must be superconducting. Although consensus does not exist on the optimum choice of superconductor, stabilizer, and cooling system, there is general agreement that development programs already underway are exploring all likely options. A substantial data base will allow an intelligent examination of the trade-offs in the choice of superconductor (NbTi versus NbSn), stabilizer (Cu versus Al) and cooling (pool boiling versus forced). Experts are very confident that a reliable toroidal system can be constructed. The system can be designed to withstand the effects associated with startup and plasma disruptions. The inboard radiation shield thickness specified for INTOR is deemed adequate to protect the coil against insulation damage and loss of stabilizer conductivity. It is thought that superconducting toroidal coils can be constructed with sufficient reliability that no failure is expected during the life-time of INTOR. The overall design of the reactor need not reflect a rapid removal and replacement scheme for the toroidal system. Nevertheless, prudence would dictate adoption of a realistic means for replacement on a several-month timescale. Toroidal field coils with demountable joints are judged impractical for INTOR because of stress problems and dewar complications, not because of contact resistance problems.

The poloidal coil system consists of the main induction coil and the field-shaping and control coils. The induction coil consists of a main solenoid located in the central bore of the machine and a few distributed turns to control the flux return from the main solenoid. A superconducting coil system is preferred over a normal coil because INTOR can be made more compact and lower in cost. Coils already in operation are capable of 10 tesla per second rates at up to 7.5 tesla fields. These parameters are consistent with those demanded by INTOR, but experiments with larger coils having an order of magnitude more energy are required. The electromechanical effects which result from the 100 V/turn required for plasma startup are severe. Use of rf or other techniques to reduce the initial one-turn voltage requirement would appreciably simplify the magnetics design.

Field-shaping and control coils are required to attain the equilibrium and stability conditions required by INTOR's elongated plasma. Engineering considerations cited above give substantial impetus to the adoption of superconducting coils located outside the toroidal coil. However, a full stability analysis, including effects of all intermediate structures, has not been completed. If it should prove necessary to place a few coils within the toroidal coils, then the situation is less attractive, but tolerable. Normal conducting coils would have to be used because of heat load and radiation damage effects.

Divertors, which are presently specified for INTOR, impose severe restraints on the magnet system. Designs performed prior to 1979 seem to be incompatible with acceptable magnet design practice. Fortunately, the new concepts mentioned previously suggest that designs could be developed for which the magnet technology basis would be adequate.

Little need be said about the power supply assessment for INTOR. Analysis of peak power and total energy requirements shows that the present generation of equipment, already under construction for machines like TFTR, is entirely adequate. The only exception might result from any increase of neutral beam energy above the presently planned 175 keV, or from the adoption of rf as a principal mode of plasma heating. The total power requirement for the facility is about 200 MW and represents no severe siting problem.

4.1.3. Reactor Availability, Reliability, and Remote Maintenance

A major objective of INTOR is the attainment of high availability (25% - 50%). The assessment has shown that operation of INTOR with high availability implies severe reliability requirements and profoundly affects the approach to remote maintenance.

In order to assess the reliability requirements for the components of INTOR, a rather detailed analysis was conducted. Over 55 components were studied for an assumed configuration of the machine. Dominant failure modes were formulated for each of these components, and the time to complete a remote repair was estimated. A reasonable schedule was assumed for machine operating time and scheduled maintenance. The availability target then implied a net time available for unscheduled repairs and thus established a set of possible reliability factors for the individual components. Realistic reliability factors were also solicited for the various components from appropriate experts and compared to the deduced set of required values. The conclusion was that the availability goals for INTOR can be met, but only with stringent development, testing and quality control.

The costs and the benefits of fully remote operation versus the costs and benefits of semi-remote or "hands-on" maintenance was performed. More specifically, the costs of incremental shielding around the machine which would allow personnel access to the reactor hall were estimated as a percentage of the capital investment. This percentage was found to be less than the percentage increase in availability for the machine because of the greater speed of repair associated with semi-remote and "hands-on" maintenance. Thus, the addition of the shielding is viewed as a cost-effective measure. Although all systems must be remotely maintainable, the lesson is that such a requirement should not rule out some "hands-on" maintenance.

The technology for the remote maintenance of INTOR largely exists and can be adopted on an appropriate time-scale. However, the techniques for designing INTOR for remote maintainability are at an early stage of development.

The vacuum technology for pumping neutral beam lines and for pumping the plasma chamber between burn pulses should be adequate for INTOR needs. However, the development of high speed pumps to remove helium during the burn is required. Moreover, the neutron and gamma heat fluxes on the cryopanels could lead to large refrigeration costs if conventional cryocondensation pumps operating at 4.2 K are used.

4.1.4. Operations

The successful operation of INTOR will place substantial demands on our abilities to diagnose and control very complex experimental operations. INTOR diagnostics will encompass a uniquely large and diverse package of physics and engineering diagnostics. Thousands of channels of data will have to be handled expeditiously and reliably.

Preliminary assessment of the diagnostics, control and data acquisition system for INTOR has confirmed that most of the required expertise is already being developed. A great deal of experience in the use of plasma diagnostics on tokamaks exists and is being expanded as the new generation of tokamaks comes on line. Diagnostics adequate for feedback control and those needed for interpretation of data (and thus eventual optimization of performance) have been identified. A major problem may arise because of the severe environment in which diagnostics will be required to operate. Several plasma diagnostics components, if utilized in the same manner as at present, would rapidly fail in the high neutron and gamma fluxes encountered during D-T operation. In the case of the engineering diagnostics, most of the fundamental techniques have been developed in other fields (e.g. fission, high-voltage).

In the case of the computer systems and the associated interface equipment, no need exists for INTOR development specifically. Because of rapid progress in the industry, the technology should be adequate for any foreseen requirements.

4.2. Engineering R and D Needs

The INTOR Workshop identified two major research and development areas in which the present world-wide program is inadequate, and a substantial increase in effort will be required to meet the needs of INTOR, namely

- Superconducting pulsed coil technology
- Vacuum pumping of helium.

The Workshop identified a number of other areas in which the needed research and development effort can probably be carried out as part of presently-planned programs on existing or authorized facilities, namely

- Superconducting toroidal coil technology
- Energy storage and transfer
- Remote maintenance and assembly
- Penetration shielding for personnel access and component protection
- Diagnostics, data acquisition and control.

These R and D needs are specified in detail in the report [1] of the Workshop.

4.3. Engineering Major Design Issues

The INTOR Workshop identified six major engineering design issues: location of the poloidal field coils, location of the vacuum boundary, location and configuration of the blanket test modules, the remote maintenance approach, the design availability objective, and the establishment of design standards. The issues associated with the poloidal field coil location were discussed previously. The location of the primary vacuum boundary at the first-wall, in or behind the primary shield but within the toroidal field coil cage, or at the building wall were all suggested, and the second option is generally favored.

Provision of a realistic environment for testing tritium extraction requires that the blanket test modules have a surface area of several square meters facing the plasma chamber, and the requirement to produce $\sim$10 MW_e implies that about 30 m^2 of plasma chamber surface area be covered by test modules. In principle, these modules can be run in vertically from the top, horizontally along the midplane, or on about a 30° diagonal above and below the midplane.

The development of a philosophy on component replacement and of a fundamental remote maintenance approach is basic to the design of INTOR. Certain components must be maintainable on a routine basis, while others can or will have to be designed for very high reliability so that rather long times can be allowed for replacement or repair. The identification of which components fall into each category is required prior to the development of an engineering design. A number of remote maintenance approaches (in situ repair, component removal, segment removal, etc.)

must be examined in developing a fundamental approach that is consistent with the availability requirements of INTOR.

The machine availability that realistically can be specified as an objective for INTOR depends upon the component reliability criteria, which affect component development programs, and upon the redundancy level in the design, as well as upon the remote maintenance approach.

Design standards, or guidelines, are needed in order to proceed with a design. In some instances appropriate standards may be borrowed from other fields. But in many instances (e.g. superconducting magnets, tritium release rates) it will be necessary to formulate new design guidelines.

5. NUCLEAR

5.1. Nuclear Status Assessment

5.1.1. First Wall, Blanket and Shield

The assessment by the INTOR Workshop revealed that because of the large number of tokamak reactor studies done over the past 10 years (more than 20), a considerable variety of first-wall, blanket and shield design solutions appear to be workable under INTOR operating conditions. The possible exceptions to this statement are limiters, divertor collection plates, and passive liners to withstand disruptions.

It was concluded that design procedures, design codes and testing methods currently exist to perform adequate neutronics, thermal-hydraulic, and structural analysis of the vacuum vessel, test modules, blankets, shields, and first walls. Furthermore, there was reasonable confidence that water cooled first walls could be designed to withstand the 10^6 heat pulses to the first wall if they did not exceed 30 to 60 watts/cm^2 on a steady state basis. Concern was expressed that there is both inadequate information about the magnitude of the heat dumped to the first wall due to a plasma disruption and the ability of an unprotected first wall to handle such an event. The use of radiatively cooled armor plate was discussed but no specific solution was agreed upon during this phase of INTOR assessment.

There appears to be a growing, and potentially adequate, data base for predicting failure modes of unirradiated first walls. However, in practically every material except stainless steel there is an inadequate data base on irradiated materials properties to put into the first wall lifetime codes. It is anticipated that an adequate data base for Ni containing materials will be present by 1985, but the prospect for obtaining such data on other materials before construction of INTOR is slight.

There seems to be a major uncertainty with regard to the implications of high voltages induced in the blanket or first wall during start up and/or during disruptions. The exact placement of electrical insulators and their operating lifetime in the radiation environment is a major unknown, and further work is required to clarify this situation.

5.1.2. Tritium

Eight major areas of concern were identified with respect to the use of tritium in INTOR: fuel purification and recycle, tritium breeding in test modules, storage of tritium, minimizing the active inventory of tritium, external sources of tritium for INTOR, containment of tritium, transportation of tritium, and cleanup of tritium releases in the plant. It was generally concluded that with respect to these topics, either data and facilities are presently available, or that they would be available by the mid-1980s to proceed with the design and construction of INTOR.

The level of knowledge and the capability required to design the fuel purification system is expected to be reasonably in hand by the mid-1980s. The main body of information is expected to come from experimental facilities such as the Tritium Systems Test Assembly (TSTA) in the USA, which is already under construction, and from the planned Tritium Engineering Test Laboratory (TETL) in Japan and Vacuum/Tritium Stand (VTS) in the USSR.

The large scale production of tritium from lithium containing materials has been demonstrated on a batch-wise basis in a number of countries, but the ability to continuously extract tritium from a hot breeding blanket module while simultaneously maintaining a low tritium inventory remains to be demonstrated. Extraction methods and behavior of breeding components under irradiation could be developed in fission reactor facilities by the mid-1980s.

The inventory of tritium required in INTOR is 2 to 3 kg. Failure proof storage systems must be identified. The development of such systems seems to be straight-forward.

The "active" tritium inventory (i.e., in the fuel fabrication, injection, exhaust, and reprocessing system) is considerably more vulnerable to accidental release and therefore will have to be reduced to as low a value as possible. Estimates of this inventory range from a few hundred grams to a few kg's. The wide spread in estimates is due to the uncertainty of fractional burnup per pass, recycle time of cryopumps, and amount of tritium adsorbed throughout the INTOR chamber and exhaust systems. Efforts to quantify these parameters will be a prime motivation of research in the next few years.

In the event that INTOR does not provide its own tritium, then roughly 65-95 kg of tritium, depending upon the option chosen for stage III, will have to be supplied from fission reactors over a 12 year period. While this amount of fuel is by no means trivial, it could be obtained entirely from dedicated tritium production facilities in the USA or partially from the heavy water coolant of CANDU reactors. The cost of this fuel could range from as low as 4 million dollars per kg to as high as 8 million dollars per kg. The international regulations governing such large transfers of tritium need to be closely examined.

The mechanisms of hydrogen diffusion through metals are reasonably well understood. However, it remains to be demonstrated that sufficiently reliable tritium permeation barriers can be produced, particularly for heat transfer surfaces and those subjected to intense neutron bombardment. Such data can come from TSTA, TETL, and VTS, while in situ experiments in fission reactors should shed light on the problems anticipated for irradiated coatings.

5.1.3. Materials

In general, it was concluded that the data base either exists now, or that it will exist by the mid-1980s, to be able to make a reasonable judgement on the mechanical performance of materials in INTOR. Furthermore, there is reasonable optimism that the structural materials can be used for the lifetime of the device.

Of the four main classes of potential structural materials considered for INTOR (austenitic steels, ferritic steels, titanium alloys and aluminum alloys), the austenitic steels are the favored choice. Ferritic steels have better thermal stress properties and more resistance to irradiation induced swelling, but the post-weld heat treatments and the shift in the ductile-to-brittle temperature makes operations below the 300-400°C range questionalbe. Titanium alloys present some advantage from the thermal stress and fatigue lifetime standpoint, but the general lack of data on irradiated materials and on formation of hydrides below 300°C makes them unattractive for use in INTOR at this time. Aluminum alloys have the potential advantage of lower thermal stresses and higher permissible first wall thickness, lower long-lived radioactivity, and lower tritium permeability. On the other hand, a lack of irradiation data above 100°C, decreased ductility due to post-irradiation heating (baking) above 100°C, a lack of irradiation data on materials with appropriate helium content, and doubts about commerical reactor-relevance for electricity production make the Al alloys unattractive for the primary structural alloy in the shield. It may be possible to consider using Al alloys in the first wall operating at about 100°C.

There is only one class of materials, austenitic steels, that has been irradiated with approximately the proper combination of displacement damage and helium production. By 1985 it is possible that a few small specimens of non-nickle containing alloys will be irradiated in FMIT under "typical" high energy neutrons conditions. However, the number of specimens and the range of temperatures, stress conditions and alloy variations will be too small to allow construction to begin in 1985 if any other alloy except stainless steel were to be used.

The data base on solid breeder materials (e.g. Li_2O, $LiAlO_2$, etc.) is small but building at a steady rate. Tritium release data looks favorable under low neutron irradiation conditions, but tests at much higher fluences and at high temperatures are necessary in order for solid breeders to be ready for incorporation in test modules.

Damage limits to the Cu and Al stabilizers of the superconducting magnets have been established. The critical damage limit of 5×10^{-5} dpa for copper and 10^{-5} dpa for Al have been accepted for INTOR. These limits are much lower than those set for the superconductor Nb_3Sn (0.002 dpa) or NbTi ($\sim$0.06 dpa). The limit to the electrical insulators in the magnets was set at the 10^9 to 10^{10} rad level.

The use of carbon in the plasma chamber appears to be marginally acceptable. Recent work by Japanese scientists has revealed that chemical sputtering can be reduced by a passivation mechanism which is not understood at the present time. Further work in this area may remove this barrier to the extensive use of carbon as limiter, beam dumps, armor plate, or even coatings of the first wall.

The assessment revealed that the experimental data in physical sputtering is now sufficient to predict wall erosion rates once the particle spectra and angle of incidence are specified. Wall erosion of unprotected INTOR first walls is not serious if the wall flux is below 10^{15} particles cm^{-2} s^{-1} and if the particle energy is less than 100 eV. However, the anticipated particle fluxes may be somewhat higher.

A great deal of progress has been made in the area of low-z coatings for first walls, limiters and particle collection plates. However, as of now there is no clearly superior candidate that has been demonstrated to be compatible with both the plasma and cyclic operation of the power cycle. TiB_2 and TiC hold great promise, but more tests are necessary for a final decision in INTOR.

5.2. Nuclear R and D Needs

The INTOR Workshop identified two major research and development areas in which the present world-wide program is inadequate, and a substantial increase in effort will be required to meet the needs of INTOR, namely

- First-wall, limiter and divertor plates
- Insulators and coatings.

The Workshop identified three other large areas in which the needed research and development effort can probably be carried out as part of presently-planned programs on existing or authorized facilities, namely

- Stainless steel properties
- Integrated tritium fuel handling system
- Tritium cleanup, safety and monitoring systems.

These R and D needs are specified in detail in the report [1] of the Workshop.

5.3. Nuclear Major Design Issue

The major near term design issue has to do with the first wall. There is no question that water cooled stainless steel tubes, on the order of 2 mm thick can handle the steady state heat flux from INTOR ($\sim$30 watts cm^{-2}). However, there is relatively little agreement on the uniformity of the heat flux around the torus. Moreover, the magnitude of heat fluxes associated with plasma disruptions is so large as to be of overriding concern in the design of the first wall.

Another concern stems from the high charge exchange neutral flux that could come from the edge of the plasma. For example, charge exchange neutral fluxes of 10^{16} to 10^{17} cm^{-2} s^{-1} at 100 eV have been postulated. Such fluxes would cause erosion rates of 0.5 to 1 mm/year, completely unacceptable for many reasons. A logical way to avoid this problem is to use sacrificial armor plate which is in front of the first wall but cooled radiatively. Such protection may also be required to mitigate against plasma disruptions.

The issue to be decided in the next year of the study is; can INTOR operate with a bare first wall (if so, is it made from Al or steel) or is a solid armor plate or other protective mechanism required around the entire torus? A corollary issue is the establishment of a reasonable upper limit for the neutron wall load for which INTOR is to be designed.

6. SAFETY AND ENVIRONMENT

This particular topic had to be approached from a rather generic sense because no detailed design was available. The main purpose of this effort was to identify areas of disagreement or lack of regulations between the various countries.

The major issues that were investigated for the assessment were tritium fuel, activation products, use of lithium in INTOR, effects of magnetic fields and magnet failures, potential toxic materials, and hazards associated with high vacuum, large cryogenic systems, and high voltage.

Since it was decided not to use large amounts of lithium containing materials in INTOR this was not perceived to be a major problem. There were no major toxic materials identified in INTOR as it is currently conceived, so that area was not pursued in great detail. Some observations on the remaining topics are given below.

Regulations pertaining to the control of (1) radioactivity, (2) toxic materials, and (3) industrial procedures associated with: (a) high vacuum technology, (b) large cryogenic systems, and (c) high voltage systems currently exist in the member nations, and these are felt to be adequate for starting the design of INTOR. However, there are few, if any, regulations that are specific to fusion devices, and a concentrated effort to establish these guidelines for INTOR must start as soon as possible. It is likely to take 5-10 years to formulate such regulations in compliance with public review and legislative requirements.

The most critical example of where specific attention must be paid to fusion devices is in the area of tritium containment. The analytical models relating release of tritium to exposure to humans currently exist. In addition, the effects of tritium on the population for a given exposure are also fairly well understood at this time. What is required is for an authoritative, international group to put the two together to establish release limits specifically for D-T fusion devices. These quidelines are necessary before INTOR progresses into the detailed design phase.

There are also two regulations currently in force in some countries which could place undue constraints on INTOR. The first is the 20 g limit on the amount of tritium in approved shipping containers. To supply initial inventories of several kilograms could require an unduly large number of shipments. There is no insurmountable problem here, only the time delays associated with the design of larger containers and regulatory licensing of such units. The second problem is with the current accountability practices. It is currently required (at least in the USA) that the entire tritium inventory of a facility be established to the nearest 0.01 g twice a year. Considering that inventories of about 3 kg could exist in INTOR and that 5-10 kg/year might be burned up, such a strict accounting may be impossible. The regulations should be examined to see if such a limit can be raised for INTOR or if such a limit is even appropriate for such a test device.

At the present time, the only material in INTOR which might be subjected to safeguards requirements is tritium. It is currently designated as a sensitive, rather than a strategic, material because of its high value (about 4000 to 8000 $/g). It is anticipated that this status will not change for INTOR.

Specific regulations for the continued exposure of individuals to high magnetic fields are also needed. At the present, the practice in the field is to use ad hoc values from the accelerator field, which may or may not stand up to the close scrutiny of safety review panels.

Guidelines currently exist for the use of non-ionizing microwave radiation, but few, if any, regulations exist for rf heating sources that might be used in INTOR. Such regulations should be established in the next decade.

Considerable investigation has already been devoted to identification of accident sequences for tokamak reactors. Based on conceptual designs of commercial plants, event trees have been constructed to identify accident scenarios as an aid in improving the safety aspects of the next generation of conceptual designs and for quantitative assessment of accident consequences. These event trees are not usable at this time to determine defendable values of risk (probability of an accident times the consequence) because of a lack of specific reactor design details and lack of an adequate data base for component failure rates. Such a data base should be developed; however, it will not likely be sufficiently developed in time for quantitative application to INTOR. The state of development of mechanistic computer models for fusion reactor accidents is preliminary. Further development is needed if defendable safety analyses are to be done for INTOR other than "worst case" analyses. The codes will also need to be verified by comparison with experimental data from well-characterized and instrumented safety-related experiments, which are just starting to be performed.

Acknowledgement

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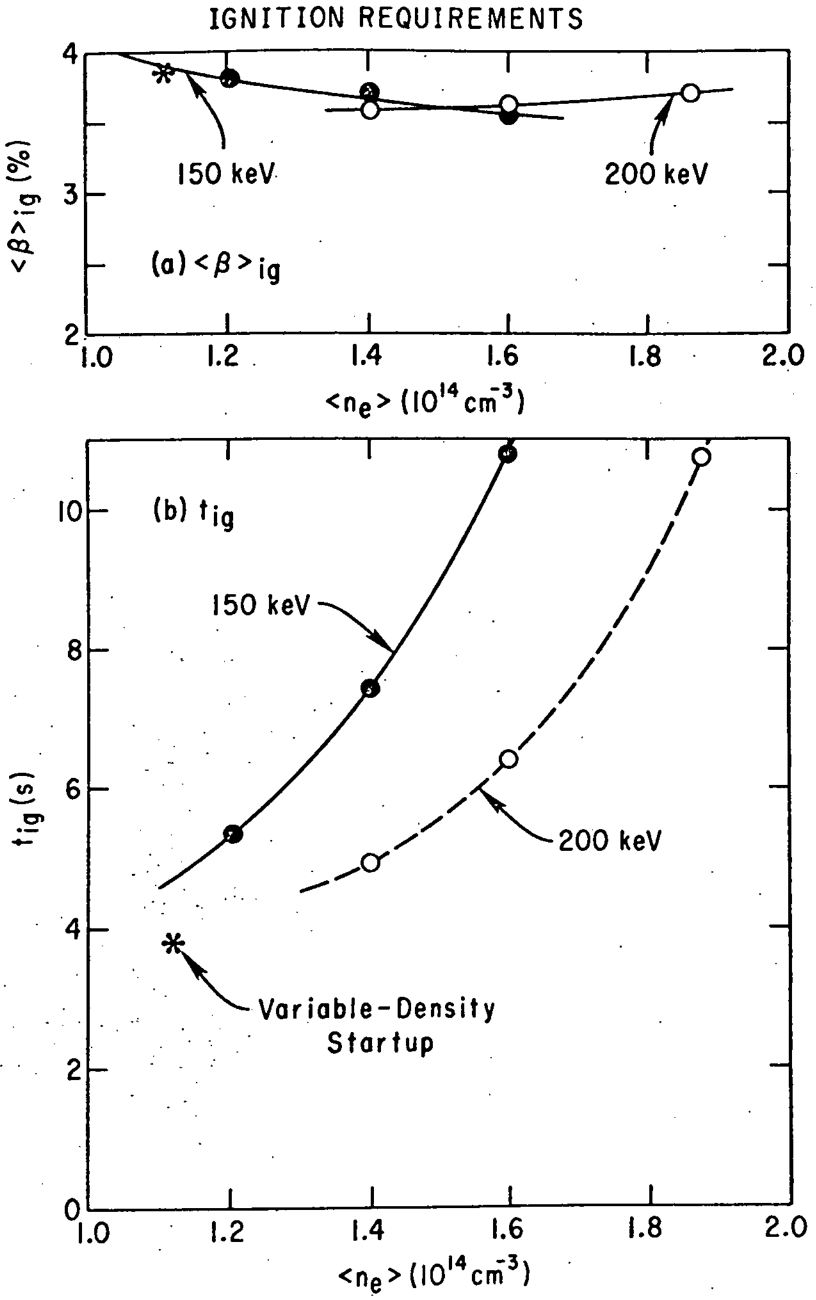
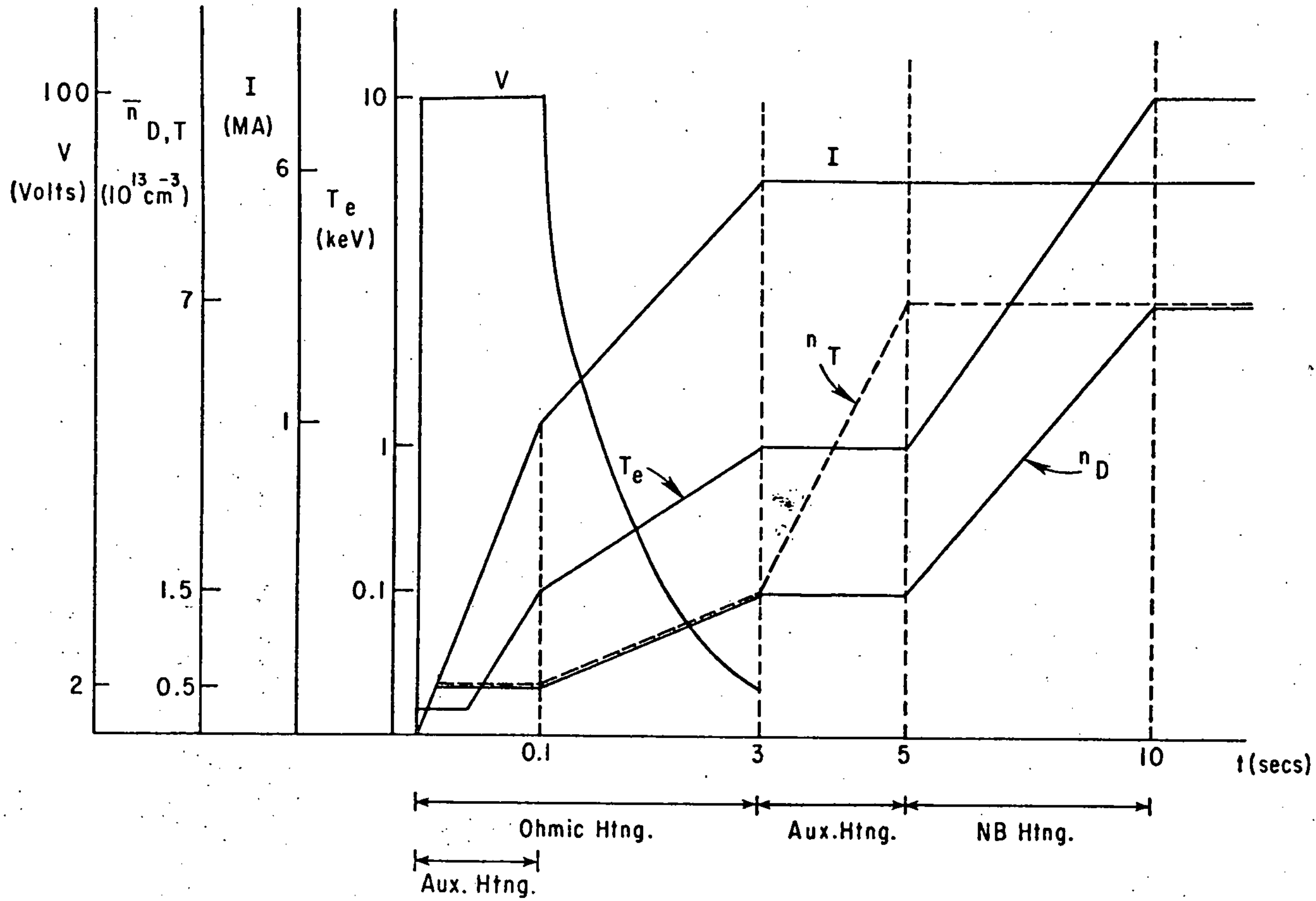
References

1. "INTOR: The Report of the International Tokamak Reactor Workshop," IAEA Report, Vienna (1980); also to be published in Nucl. Fusion (1980).

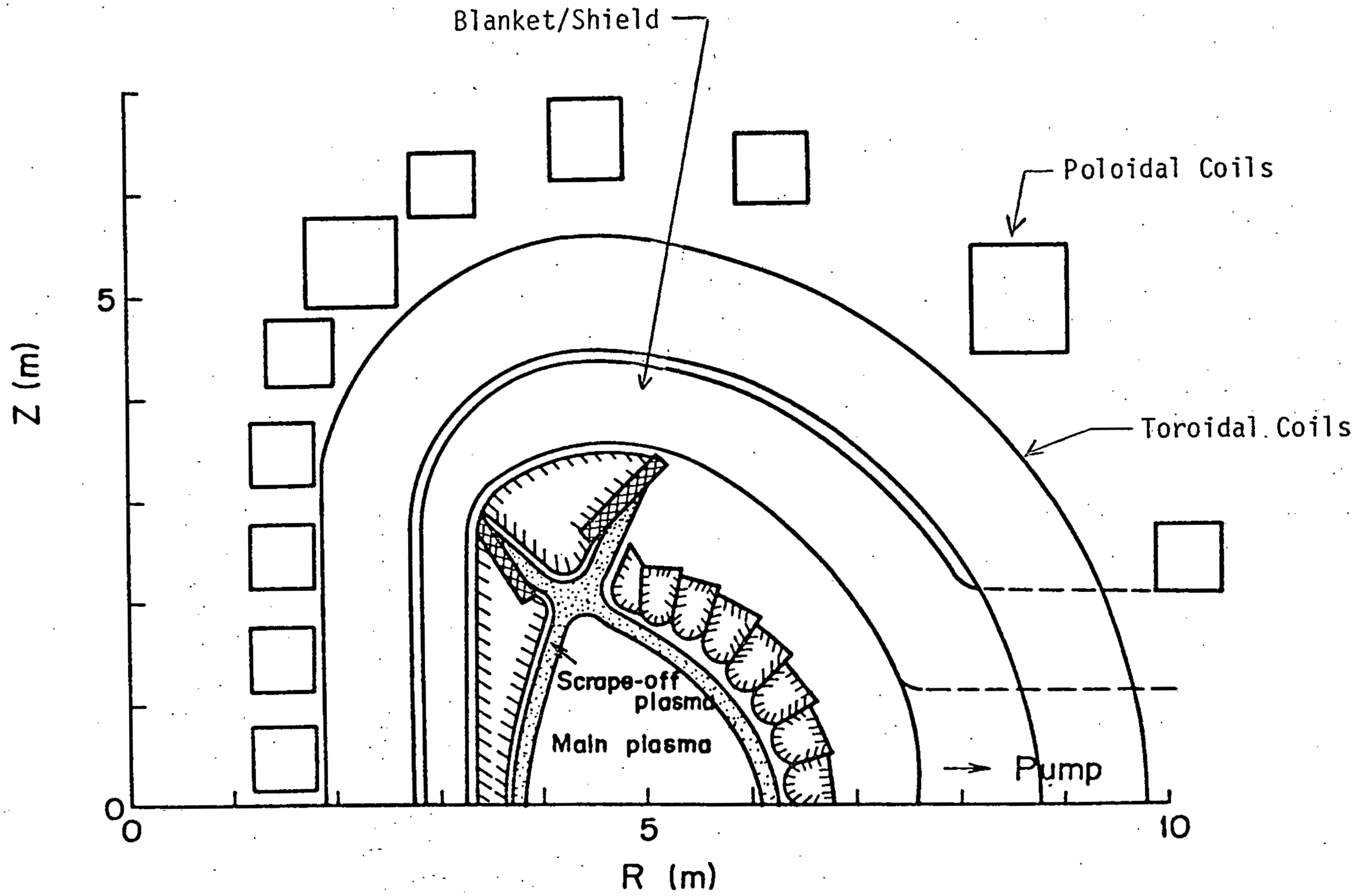
Figure Captions

Figure 1	The required $\langle\beta\rangle_{ig}$ and heating pulse t_{ig} to achieve ignition in INTOR, for a range of plasma densities, assuming 60 MW of neutral-beam power at 150-200 keV, including an optimized 150 keV "variable density" start-up.
Figure 2	Schematic illustration of the start-up scenario for INTOR.
Figure 3	An exterior-coil poloidal divertor proposed for INTOR.


IGNITION REQUIREMENTS
(a) $<\beta>_{ig}$
$<\beta>_{ig}$ (%)
2
3
4
150 keV
200 keV
1.0
1.2
1.4
1.6
1.8
2.0
$<n_e>$ (10^{14} cm^{-3})
(b) t_{ig}
t_{ig}(s)
0
2
4
6
8
10
150 keV
200 keV
Variable-Density Startup
1.0
1.2
1.4
1.6
1.8
2.0
$<n_e>$ (10^{14} cm^{-3})



START-UP PHASE OF INTOR


Blanket/Shield
Poloidal Coils
Toroidal Coils
Scrape-off plasma
Main plasma
Pump
Z (m)
5
0
R (m)
0
5
10