SPECIAL TOPIC

INTERNATIONAL TOKAMAK REACTOR – PHASE TWO A, PART I

Executive Summary of the IAEA Workshop, 1981–1983

INTOR Group*

ABSTRACT. The results of the studies of critical issues affecting the feasibility, objectives and cost of the INTOR design for a tokamak engineering test reactor that have been carried out in the INTOR Workshop over the period 1981–83 are summarized. An improved INTOR design concept is described. The objectives, schedule and test programme of INTOR are discussed, and specific R and D programmes are suggested to resolve design uncertainties.

CONTENTS. 1. Introduction; 2. Role of INTOR in the fusion programme; 3. INTOR objectives; 4. Critical issues: 4.1. Plasma performance; 4.2. Impurity control and first wall; 4.3. Testing; 4.4. Tritium and blanket; 4.5. Mechanical configuration; 4.6. Magnetics and torus electromagnetics; 4.7. Cost-risk benefit; 5. Design description; 6. Machine operation and test programme; 7. Schedule; 8. Research and development; 9. Conclusion; 10. Recommendation for future work References.

1. INTRODUCTION

The International Tokamak Reactor (INTOR) Workshop is a collaborative effort among Euratom, Japan, the USA, and the USSR. It is conducted under the auspices of the International Atomic Energy Agency (IAEA), in terms of reference defined by the International Fusion Research Council (IFRC), an advisory body to the Director General of the IAEA which supervises the INTOR Workshop. The broad objectives of the INTOR activity, as set forth by the IFRC, are to draw upon capability that exists world-wide:

(1) to identify the objectives and characteristics of the next major experiment (beyond the present generation of large tokamaks) in the world tokamak programme;
(2) to assess the technical data base that will exist to support the construction of such a device for operation in the 1990s;
(3) to define such an experiment through the development of a conceptual design;
(4) to study critical technical issues that affect the feasibility or cost of the INTOR concept;
(5) to define R and D that is required to support the INTOR concept;
(6) to carry out a detailed design of the experiment; and, finally,
(7) to construct and operate the device on an international basis.

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

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

* *Euratom*: G. Grieger, G. Casini, F. Engelmann, F. Farfaletti-Casali, M. Harrison, A. Knobloch, D. Leger, P. Reynolds, P. Schiller

Japan: S. Mori, N. Fujisawa, T. Hiraoka, K. Miyamoto, S. Nishio, Y. Sawada, K. Tomabechi

USA: W.M. Stacey, Jr., M.A. Abdou, J.M. Rawls, J.A. Schmidt, T.E. Shannon, R.J. Thome

USSR: B.B. Kadomtsev, G.F. Churakov, B.N. Kolbasov, V.I. Pistunovich, D.V. Serebrennikov, G.E. Shatalov

participants in their home institutions to perform the tasks that had been defined at Workshop sessions. This home-country effort involved more than 100 of the leading magnetic fusion scientists and engineers (about 15–20 man-years of effort) from each of the four partners. The participants met in Vienna four times, for a total time of 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 TFTR, JET, JT-60, T-15) in the world-wide tokamak programme and to assess the technical feasibility of constructing this experiment to operate in about 1990. Detailed assessments of the plasma physics and technology bases for such an INTOR experiment were developed, and physical characteristics were identified which were consistent with this technical basis and with the general objectives of the INTOR device as they evolved in this process.

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

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

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

The conceptual design contributions to the Phase-One INTOR Workshop have been published [6–9]. These contributions formed the basis for the INTOR conceptual design, which is documented in Ref. [10].

The INTOR Workshop was then extended into Phase Two A in July 1981. Emphasis in this phase has been upon the resolution of certain critical technical issues which were identified during Phase One and which affect the feasibility, cost and engineering complexity of the INTOR design concept. This work has been carried forward by teams of experts working in their home institutions under the direction of the INTOR participants, who met in Vienna six times (for a total of about 12 weeks) over the first two years of Phase Two A to define and review the work and to take decisions. The new information developed in the critical issues studies has led to an improvement in the INTOR concept. Several new critical technical issues have been identified. The work to date in the Phase-Two-A INTOR Workshop has been reported in the national contributions [11–14] to the Workshop and is documented in the report [15] of the Phase-Two-A, Part 1 Workshop. This work is summarized in this paper.

The International Fusion Research Council has recommended that the INTOR Workshop be extended to June 1985. The objectives of the Workshop during this period are to investigate certain critical technical issues that are essential to the feasibility and further improvement of the INTOR concept, to define R and D requirements in support of the INTOR concept, to keep under review the results of the world-wide R and D programme, and to improve the INTOR concept as a result of the new information obtained.

The role and objectives of the INTOR experiment are discussed in Sections 2 and 3. Then the results of the Phase-Two-A work are summarized in Section 4. The revised INTOR design concept is described in Section 5. The test programme and schedule are outlined in Sections 6 and 7. Specific R and D requirements are listed in Section 8. Finally, conclusions and recommendations are given in Sections 9 and 10.

2. ROLE OF INTOR IN THE FUSION PROGRAMME

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

(a) Production of several hundred megawatts of electricity and achievement of net electrical power production;
(b) Production of tritium in the blanket, with a net breeding ratio greater than unity;
(c) Demonstration of the development and integration of full-scale components which can be extrapolated to a commercial reactor;
(d) Demonstration of component and system reliability, availability and lifetime at a level that would be acceptable for a commercial reactor;
(e) Demonstration of safe and environmentally acceptable fusion reactor operation that would satisfy the requirements for a commercial reactor;
(f) Demonstration of commercial feasibility (although the DEMO would not need to be itself economically competitive).

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

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

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

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

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

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

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

TABLE I. INTOR TECHNICAL OBJECTIVES

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

(g) Test the safety and environmental acceptability of a fusion reactor.

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

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

3. INTOR OBJECTIVES

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

The programmatic objectives for INTOR are:

(a) INTOR should be the maximum reasonable step beyond the present generation of large tokamaks (TFTR, JET, JT-60, T-15) in the world fusion programme;
(b) INTOR should demonstrate the plasma performance required for the tokamak DEMOs;
(c) INTOR should test the development and integration into a reactor system of those technologies required for the DEMOs;
(d) INTOR should serve as a test facility for blanket, tritium production, materials and plasma engineering technology development;
(e) INTOR should test fusion reactor component reliability;
(f) INTOR should test the maintainability of a fusion reactor;
(g) INTOR should test the factors affecting the reliability, safety and environmental acceptability of a fusion reactor.

The technical objectives of INTOR have been developed to support the achievement of the programmatic objectives, while being consistent with the anticipated technical basis [5] for the design and construction of such an experiment to initially operate in the 1990s. These technical objectives are given in Table I.

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

TABLE II. STAGED OPERATION SCHEDULE

Stage	Number of years	Emphasis	Availability	Annual 14-MeV neutron fluence ($MW \cdot a \cdot m^{-2}$)[b]	Annual tritium consumption (kg)
IA	1	Hydrogen plasma operation Engineering check-out	10%	–	–
IB	2	D-T plasma operation	15%	0.16	3.6
II	4	Engineering testing	25%	0.31	6.9
III	5	Upgraded engineering testing[a]	50%	0.62	13.8

[a] The minimum objective is to achieve about 3 $MW \cdot a \cdot m^{-2}$ within $\leqslant$10 years after the end of Stage II. The design should allow for higher fluences but with a lower probability of success. This could be achieved in several ways; the case given here is for 5 $MW \cdot m^{-2}$ within eight years and is only representative.

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

4. CRITICAL ISSUES

The INTOR Workshop studied during 1981 to 1983 several technical issues that affect the feasibility, practicality and cost of the INTOR concept. This work is summarized in this section and described in detail in the report [15] of the Workshop. Conclusions and recommendations based upon this work are given in Sections 9 and 10.

4.1. Plasma performance

An assessment of radiofrequency bulk heating and current drive and an update of the data base assessment for confinement and beta limits were performed. Several other tasks were also undertaken, including a re-examination of ripple limits, burn control, neutral-beam energy, etc.

4.1.1. Bulk heating

The recommendation [5], made in 1979, to take neutral beams as the reference bulk heating method for the INTOR conceptual design in Phase One was straightforward. Neutral-beam injection (NBI) possessed the overwhelming advantages of exclusivity of data at high power (>1 MW) levels, a good agreement between theory and experimental findings, and a sequence of planned applications that would drive the technology toward INTOR needs. The engineering and technological advantages of RF heating were recognized, but the inadequacy of the data base for the whole spectrum of RF options precluded favouring any of these compared to neutral beams.

Neutral-beam injection heating experiments have progressed considerably since 1979, and the confidence that can be placed in NBI heating is stronger today than it was in 1979.

With respect to RF heating, the situation is markedly different now from what it was in 1979. More than 3 MW of power in the ion cyclotron range of frequencies (ICRF) has been deposited in PLT via second-harmonic heating, the mode favoured for INTOR, using a loop antenna launching structure. The power was given largely, perhaps exclusively, to ions, and the deposition profile was quite centrally peaked. These factors contributed to the high ion heating efficiency (3–4) $eV \cdot kW^{-1} \times 10^{19}\ m^{-3}$, a value that compares favourably with the best NBI heating on PLT. A considerable amount of data has also been gathered in the mode conversion regime, particularly in the extensive set of experiments conducted on TFR. Good heating efficiency has been observed in this case as well, up to the peak power level of 2.2 MW. Perhaps the most troubling single aspect of the largely favourable ICRF data base is the indication, in some cases, of increasing impurity contamination at high power levels. Although this observation is not yet fully understood, there are good reasons to expect that it will be less of a problem in the reactor regime.

Considerable progress was also made in the theoretical modelling of ICRF heating, permitting meaningful tests of the level of understanding of the underlying physical phenomena. Good agreement has been obtained in all cases in which a detailed comparison between theory and experiments could be made. The initial application of these modelling tools to reactor design has resulted in the prediction that heating to ignition with ICRF can be accomplished with less power and under a wider range of plasma conditions than will be the case for neutral-beam injection.

An intense R and D programme will be necessary to develop ICRF heating to the level required for INTOR. This includes both physics experience at increasing power level and duration of the heating pulse and the development of reactor-relevant launching structures. For JET, an ICRF heating system transferring more than 15 MW to the plasma during a heating pulse of 5 to 10 s is already under preparation. An important point is to demonstrate that tokamak plasmas can be heated to high beta with ICRF without any deleterious effects. In addition, since the feasibility of loop antennae in the INTOR environment has not been established, a wider range of launching structures must be investigated. This also requires ICRF waveguide experiments beyond the presently planned low-power coupling experiments.

Nevertheless, there are already now strong technological and engineering design arguments in favour of ICRF heating for INTOR. With the exception of the launching structure, ICRF hardware is already commercially available. The principal engineering advantage of ICRF is the ability to locate the bulk of the equipment in an area remote from the reactor core, thus adding to reliability, simplifying maintenance, and reducing the size of the reactor hall. In addition, the transmission system is compatible with bends to reduce neutron streaming and thus minimize shielding requirements. Other advantages of ICRF relative to NBI include higher efficiency, increased component life, and reduced complexity of required support equipment. An important additional benefit of ICRF in comparison to NBI is the expectation of a more centrally peaked power deposition profile, particularly when the plasma density is above 10^{20} m^{-3}.

Heating experiments in the lower hybrid and electron cyclotron frequency ranges are also making rapid progress. They show heating efficiencies comparable to those of NBI and ICRF, and the trend of the results obtained is consistent with theoretical predictions. However, further advances are needed in both these frequency ranges before either could be considered as a viable candidate for the reference heating system for INTOR. In contrast, electron cyclotron start-up assist, for which a low power is required, appears an attractive possibility, if the R and D programme is successful.

Conceptual designs of launching structures for INTOR were made for ion cyclotron, lower hybrid, and electron cyclotron waves and have shown that such launchers can be made consistent with INTOR requirements, provided that a number of critical components are developed to the required specifications.

The choice between radio-frequency waves and neutral-beam injection for bulk heating in INTOR, at present, rests on a trade-off between the engineering and technological advantages of RF heating, in particular ICRF, and the greater confidence in the physics base for NBI. The recent advances in ICRF physics persuade us that the balance has shifted in favour of ICRF.

Recommendation: ICRF should be the primary option for bulk heating. It must be emphasized, however, that the promise of ICRF can only be realized if the requisite R and D programme is set in motion.

Recommendation: Neutral-beam heating should be the backup option for bulk heating. In the long range, NBI heating would have to be based on negative-ion beams, an area in which recent advances improve the prospects for a system with high efficiency and significantly reduced neutron streaming. The backup to ICRF in the case of a near-term commitment to a tokamak reactor is positive ion beams, but it must be recognized that maintaining this as a viable option entails a significant development effort.

4.1.2. Current drive

Non-inductive current drive experiments are now performed with a variety of methods (neutral-beam injection, lower hybrid waves, electron cyclotron waves). The broadest data base is available for current drive by lower hybrid waves. In fact, experiments on many devices (ALCATOR-C, JIPP-T II, JFT-2, PLT, T-7, VERSATOR II, WEGA, WT-2) have succeeded in demonstrating current drive by lower hybrid waves. For example, in PLT, plasma currents of 200 kA have been sustained for several seconds without Ohmic drive at efficiencies of the order of 2 A · W^{-1}. The same system has even proved capable of raising the plasma current without benefit of an induced electric field. Until very recently, these experiments have been successful only at densities below 10^{19} m^{-3}, raising doubts about the possibility of employing this approach to achieve steady-state operation in a reactor. These fears have been allayed in experiments on Alcator C employing a considerably higher-frequency lower hybrid source, where current drive has been observed at densities approaching 10^{20} m^{-3}.

However, even with optimistic estimates about the extrapolation of these data to INTOR burn conditions, it is estimated that the current drive efficiency would still be no greater than 0.1 $A \cdot W^{-1}$, which limits Q to of order five.

On the other hand, lower hybrid current drive may prove useful in a scenario in which the OH-transformer is periodically re-charged by the RF system at low plasma density but at full plasma current. By conserving the OH flux swing to provide merely the resistive compensation at high densities, such an approach might allow an increase in the burn time and reduce the number of cycles by typically an order of magnitude, provided that current diffusion or impurity accumulation do not impose more stringent limits. In this case, fatigue problems that have their origin in electromagnetic cycling would be greatly reduced. Another possibility could be to use lower hybrid current drive to ramp up the current in a low-density plasma and then use the OH transformer to drive long pulses in a full-density plasma. The present experiments provide a good basis for such uses of current drive. Efficiencies are expected to be of the order of 0.3 $A \cdot W^{-1}$, but the time necessary for start-up would be of the order of tens of seconds.

Recommendation: While the phenomenon of lower hybrid current drive is now reproducible in a variety of devices, the present estimate of the recirculating power requirement in reactor applications renders it premature to contemplate using this mechanism for reformulating the INTOR design around a steady-state concept. It is, however, recommended that a thorough physics and engineering analysis be made of the quasi-steady-state mode of operation, including the variant in which a non-inductive current drive technique is employed between burn cycles to recharge the Ohmic transformer at low density, but full current.

4.1.3. Beta limits and energy confinement

The considerable body of new information on confinement in auxiliary-heated tokamak plasmas has reduced significantly the extrapolation to INTOR conditions in a number of critical plasma parameters. Most significant among these developments is accessing regimes where the theoretical models used to estimate the INTOR β limit are being tested against experiment. Values of β as high as 4.5% and values of poloidal β approaching 2/3 of the aspect ratio have been obtained, although under different conditions. Although the trends found in these experiments are different from those embodied in the INTOR (ALCATOR) scaling used to date, the outlook remains favourable for INTOR, provided the observed improvement of confinement with plasma current persists into the multi-megampere regime and a favourable size scaling is verified. Near-term experiments are targeted to resolve these questions.

To summarize the situation, although the new results obtained on plasma confinement and plasma behaviour at larger β have considerably changed the picture in these fields, no convincing argument has emerged for modification of the specification of INTOR parameters. While a quantification of the confinement potential of INTOR is subject to considerable uncertainty, extrapolation to INTOR conditions of some of the scalings of confinement that were recently proposed yields confinement potentials similar to that predicted by the original INTOR (ALCATOR) scaling upon which the present design parameters are based.

Conclusion: The INTOR design value of $\langle\beta\rangle = 5.6\%$ seems reasonable on the basis of available experimental evidence. The prospects for achieving the burn conditions specified for INTOR remain favourable in the light of the recent experimental results. Consideration of a lower q operating point is attractive, but premature. Therefore, no change in the parameters related to $\langle\beta\rangle$ or energy confinement is suggested at this time.

4.1.4. Other issues

Improved estimates of the effect of the magnetic field ripple on the confinement properties of the INTOR plasma have shown that an edge ripple of $\pm 1.2\%$ at the outboard side is acceptable from the point of view of overall energy transport. In contrast, beam ion losses, towards the end of neutral-beam heating to ignition, may be excessive for such a ripple value; however, for ion cyclotron heating, this situation is expected to be more favourable. Ripple-induced losses of fusion α-particle have also been estimated. Although the results have not yet converged, there are indications that an edge ripple of

±1.2% is acceptable. Therefore, this value is suggested as the upper bound of the design envelope.

The data base relevant to potential burn control mechanisms remains inadequate to permit confident formulation of a preferred scheme. More definitive information on plasma behaviour near β limits would be especially welcome in this regard. Calculations have been made on the magnetic field ripple required to reduce confinement significantly. The ripple at the outboard edge must be increased to at least ±3%, which is a demanding requirement for the toroidal field coil design.

Equilibrium control appears feasible. However, the integration of the passive and active control loops into the blanket and shield system is a demanding task.

Recent experiments have provided important new information about the properties of disruptions, mainly in Ohmically heated discharges. Extrapolating these results to INTOR conditions involves considerable uncertainty, so it is still not possible to give a reliable characterization of disruptions in this device. More experimental information, in particular on discharges closer to thermonuclear conditions, is also needed for progress to be made in defining a disruption control system. The disruption-induced density limit now appears to be less of a concern for INTOR.

For start-up assist and profile control, electron cyclotron heating, having the most solid data base, is recommended. Specific additional R and D work is necessary to demonstrate the viability of the scheme.

A re-examination of the neutral-beam heating to ignition scenarios has resulted in recommending a reduction of the beam energy specification from 175 to 150 keV. The lower energy is acceptable from the physics point of view and is preferable from engineering and technological standpoints.

4.2. Impurity control and first-wall

The primary emphasis of this work was an integrated study of the edge region physics, plasma-wall interaction, materials, engineering and magnetics considerations associated with the poloidal divertor and pumped limiter. The development of limiter or divertor collector plate design concepts, with an acceptable lifetime against erosion, was a major part of the work. A comparative evaluation of the poloidal divertor and the pumped limiter was performed. Other possible impurity control methods were also evaluated.

4.2.1. Poloidal divertor and pumped limiter

The primary purposes of either the poloidal divertor or the pumped limiter are to remove the exhaust of helium from the plasma and to maintain the helium concentration below 5%, to remove heat and to provide for an acceptably small level of wall- or limiter/collector plate-eroded impurities in the plasma. Calculations indicate that both concepts may provide acceptable helium exhaust, and while the pumping requirements may be greater for the pumped limiter, this is not necessarily a major factor. However, the experimental data base for the pumped limiter is restricted to relatively small, low-power experiments, and it is a key issue to determine the scaling with size and power. Both concepts appear capable of handling the heat loads; however, the potential for preventing impurities sputtered from the divertor collector plate, which is far removed from the plasma, is certainly better than for preventing limiter-sputtered impurities from penetrating the adjacent plasma. Moreover, the sensitivity of the pumped limiter performance to plasma positional control may be a problem. There is evidence from ASDEX, PDX and D-III that discharges with poloidal divertors are cleaner. Thus, judging from this relatively limited data base, divertors appear to have a significant advantage over limiters with respect to impurity control.

The choice of materials for the surface of a limiter or divertor collector plate both affects and depends upon the temperature of the plasma at the sheath in front of the surface, because of the energy dependence of the sputtering yield. The sheath temperature is near the plasma edge temperature when $T \gtrsim 100$ eV. However, if there is strong recycling near the collector plate or limiter, the edge temperature will be about 100 eV, and the temperature at the limiter or divertor collector plate will be lower (about 30 eV), because of the smaller electron thermal conductivity at low temperature. Calculations indicate that strong recycling is more easily achieved in a divertor than with a pumped limiter, and strong recycling has been demonstrated in a number of divertor experiments.

Conclusion: Detailed sputtering erosion/redeposition calculations, including self-sputtering but not arcing, resulted in the following conclusions: 1) medium- and high-Z surface materials result in acceptable designs if $T \lesssim 50$ eV; 2) low-Z materials result in acceptable design solutions if $T \gtrsim 700$ eV; 3) it may be possible to find acceptable design solutions with low-Z surfaces in the range $50 \lesssim T \lesssim 200$ eV because the net erosion is predicted to be small, even though the primary erosion and redeposition rates are large; and 4) the erosion rates may be too large to admit acceptable solutions with any materials in the range $200 \lesssim T \lesssim 700$ eV.

Plasma transport calculations have been performed to evaluate the probable values of plasma temperature. With refuelling by gas puffing and recycling the edge density is high and, without high edge radiation, it is estimated that the most probable edge temperature will be about 100 eV far from the plate. The possibility of operating in the more desirable high and low edge temperatures regions was investigated.

The production of high edge temperatures by using pellet injection and high-speed pumping was considered. This reduced the recycling, thereby lowering the edge density and increasing the edge temperature. For a realistic upper limit to the pumping fraction of about 5%, temperatures in the 700–800 eV range were obtained when impurity radiation was neglected. Increased pumping (i.e. up to 20%) produced temperatures in the 1 keV range, but this pumping level is not realistic. It is felt that although temperatures above 700 eV can be obtained in modelling studies, it would be imprudent to base the INTOR design on obtaining these temperatures. This conclusion is, in part, based on the high probability that impurity radiation will reduce the temperature by an unacceptable amount and impurity production will be enhanced by unipolar arcing.

The potential for achieving edge temperatures below ~50 eV in the presence of high edge radiation was studied. An iron or tungsten limiter sputters sufficient material to cause very substantial radiative power losses, but radial transport codes predict for iron and lower-Z materials that a stable radiative edge will be established; and in these conditions parallel transport models of the boundary predict that self-sputtering of tungsten is not a dominant process when more than about 80 MW is radiated. However, similar models, applied to ISX-B discharges, produced similar edge peaking in the impurity concentration, a condition not observed experimentally.

The possibility of reducing further the plasma temperature in the poloidal divertor was investigated. Plasma and neutral transport in the INTOR poloidal divertor was modelled, taking into account the finite parallel electron thermal conductivity and the high recycling of neutrals at the divertor plate. Substantial increases in plasma density near the plate and significant cooling in this region were predicted. The edge density tends to be higher for divertors with high recycling than for pumped limiters. Electron temperatures as low as 25 eV at the plasma sheath were predicted for a high-recycling divertor. Taking into account uncertainties in the edge conditions, this is probably a lower limit on the temperature at the sheath. However, the sheath temperature at the divertor plate should certainly be significantly less than the sheath temperature at the limiter, for comparable plasma edge conditions. If the plasma edge temperature is sufficiently low that $T < 50$ eV at the divertor collector plate, then a high-Z surface can probably be used and the erosion would be quite small.

The divertor configuration that was analysed was similar to the INTOR Phase-One design, with flat collector plates inclined at 30° and 15° in order to reduce the peak heat flux to 2 to 3 $MW \cdot m^{-2}$. The limiter configuration considered was double-edged and shaped to achieve the same maximum heat flux. This limiter has two leading edges with peak heat fluxes of 1 to 2 $MW \cdot m^{-2}$.

For $T < 50$ eV, tantalum or tungsten would be the preferred surface material. Material and lifetime assessment show that beryllium is the preferred surface material for the limiter and divertor collector plate for $T > 50$ eV. Graphite has many desirable properties, but the combination of high erosion rate by chemical/thermal sputtering and the rapid deterioration of thermal properties under irradiation makes the lifetime of graphite tiles short. Boron suffers from fabrication problems, poor thermophysical properties and possibly chemical sputtering. The poor thermophysical properties and self-sputtering of silicon carbide and titanium carbide do not permit adequate lifetime. The thermal conductivity of SiC and BeO is degraded rapidly under irradiation to low values regardless of the initial unirradiated values.

Similar erosion/redeposition characteristics were found for the top surface of the limiter and the divertor collector plate. Designs were developed with Be surface materials that had a lifetime against sputtering erosion of 2.3 years at 50% availability. Because of the concentrated particle fluxes, the erosion of the leading edges of the limiter was unacceptably large. If the plasma edge temperature on the outermost closed magnetic surface is less than 200 eV, the sheath temperature at the leading edge of the limiter is less than 50 eV, and a high-Z material has been considered for use on the leading edge, in order to achieve an acceptable design solution from the erosion standpoint. Further study of this design solution for the limiter leading edge is required.

The high-conductance attachment concept (e.g. brazing) was found to be the most appropriate method for bonding low-Z materials to the heat sink. The maximum allowable thickness of plasma-side tiles is generally limited by the temperature of the tile and the stress and fatigue of the heat sink. The maximum thickness of the tile is 2.5, 0.4, 2.4 and 1 cm for Be, C, BeO and SiC, respectively. The thickness of graphite is limited by low-temperature operation to avoid chemical sputtering. Stresses and fatigue of the heat sink, and consequently the allowable tile thickness, are strongly dependent on the width of the tile.

Electromagnetic analysis of the limiter during plasma disruptions shows that the forces and torques are manageable and arcing can be avoided if the limiter is divided into a number of sectors and the first wall is conducting with a time constant of 50–100 ms.

Studies have been performed to assess the impact of the choice of a poloidal divertor or pumped limiter on the overall mechanical and magnetic configuration and maintenance. The divertor and limiter modules would be configured similarly and their maintainability would be comparable. The poloidal field coil (PFC) configuration is less demanding for the pumped limiter than for the divertor, and the associated power supplies are thus less expensive. If the pumped limiter design is configured within the same toroidal field coil (TFC) system that will accommodate the divertor, the pumped limiter design is a few per cent cheaper than the poloidal divertor design and the maintainability of both designs is comparable. If advantage is taken of the ability to reduce the TFC height with the pumped limiter, an additional few per cent cost reduction is possible.

Recommendation: The single-null poloidal divertor should be the reference impurity control option. The pumped limiter should be retained as a design option. R and D should be pursued for both options to resolve uncertainties. The mechanical configuration and maintenance schemes should accommodate both options.

Recommendation: A design basis plasma temperature at the limiter of 100 eV should be used for the limiter design. The top surface of the limiter would be covered by beryllium tiles. These tiles would be attached by a high thermal conductance bond to a heat sink. Copper alloys are preferred for the heat sink, but vanadium, molybdenum and zirconium alloys remain as potential options. The leading edge of the limiter may not be coated since a high-Z heat sink material results in small net erosions.

Recommendation: A reference plasma temperature of 30 eV at the divertor collector plate should be used for the divertor design. At this low temperature, the surface of the collector plate should be tungsten, tantalum or molybdenum, which would lead to an erosion-resistant, long-lived collector plate under these conditions. However, it is realized that the consequences of operating a high-Z surfaced collector plate at $T > 50$ eV are adverse. Therefore, an alternative design based on a beryllium surface and $T > 50$ eV has also been developed.

4.2.2. First-wall design

The Phase-One INTOR design for the first wall was relatively simple – H_2O-cooled stainless steel in a panel-type design. This design continues to meet the needs for INTOR under the disruption analyses summarized in the next section. On the other hand, the carbon tiles specified for the inboard wall in the back-up Phase-One INTOR design are highly questionable in view of recently published chemical sputtering data. Thus:

Recommendation: The first wall in INTOR should be stainless steel, H_2O-cooled and of the panel-type construction.

4.2.3. Erosion due to disruptions

The model for the surface response to the disruption power flux was refined and the erosion of first wall and divertor collector plate or limiter due to disruptions was evaluated. The reference INTOR disruption scenario, in which the 290 MJ of thermal energy goes to the first wall (45%) with a peak energy deposition of 175 $J \cdot cm^{-2}$ and to the collector plate or limiter (55%) with a peak energy deposition of 270 $J \cdot cm^{-2}$ in 20 ms, was used for the first two cases shown in Table III.

Conclusion: Acceptable divertor collector plate and limiter lifetimes are obtained with the reference disruption scenario, even if the entire melt layer erodes.

If the energy is deposited in 5 ms, instead of 20 ms, and the entire melt layer erodes, the lifetimes of the first wall and limiter or divertor collector plate become marginally acceptable, as shown by the third case. If

TABLE III. LIFETIME ESTIMATE FOR RECOMMENDED DESIGN
(years based on 50% availability)

	Limiter or divertor	First wall
No melt layer erosion	3.8	>30
All melt layer erodes[a] (reference: 20 ms)	2.3	>15
All melt layer erodes[a] (pessimistic: 5 ms)	0.7	2.8
All melt layer erodes[b] (extreme: 5 ms, all energy goes to limiter/divertor)	0.6	>30

[a] Peak energy: limiter/divertor = 270 $J\cdot cm^{-2}$, first wall = 175 $J\cdot cm^{-2}$.
[b] Peak energy: limiter/divertor = 535 $J\cdot cm^{-2}$, first wall = 35 $J\cdot cm^{-2}$.
[a,b] Minor disruptions: limiter/divertor = 170 $J\cdot cm^{-2}$, first wall = 0.

all the energy goes to the divertor plate or limiter and the energy is deposited in 5 ms, annual limiter or divertor module replacement would be required.

4.2.4. Other impurity control methods

Local divertor studies indicate that the special coils which characterize hybrid divertors will produce radial field errors which result in destruction of some of the flux surfaces. It was concluded that the region of flux surface destruction was sufficiently large to make hybrid divertors unacceptable for use on INTOR. Some progress was made in developing conventional bundle divertor concepts that could satisfy the INTOR engineering design criteria.

The use of NBI-driven impurity flow reversal was assessed. Calculations based on a model that had been calibrated to match the PLT flow-reversal experiment, indicated that about 20 MW of the co-injected beam might be adequate to inhibit limiter-sputtered impurities from penetrating to the centre of the plasma, thus possibly assisting in establishing a cool, radiating edge.

4.3. Testing

The operational requirements upon INTOR for engineering testing were evaluated. Three different aspects of testing were considered: 1) fluence requirements for structural materials radiation damage; 2) long-term operational requirements for establishing component reliability; and 3) short-term operational requirements for blanket testing. The results of this section will subsequently be combined with the results of the cost-risk-benefit section to arrive at a recommendation on fluence objective.

4.3.1. Fluence requirements for structural materials radiation damage tests

Testing of structural materials in INTOR will provide a better data base for constructing the DEMO. The evaluation on the necessary neutron fluence for such testing indicates that a fluence of 2 $MW\cdot a\cdot m^{-2}$ will probably be the minimum, because tests with a lower fluence than this value will provide little useful information on any of the important structural material properties. Thus, there are doubts whether the value of 2 $MW\cdot a\cdot m^{-2}$ is sufficient and some of the experts strongly feel that a higher fluence should be aimed at in INTOR testing.

4.3.2. Long-term operation component reliability

The benefit of long-term component operation was quantified in terms of the number of hours of operation that would be required to assure an 80% confidence level in predicting the mean-time-between-failure (MTBF) for that component in the DEMO. Anticipated MTBF were established for INTOR and DEMO components, then the test time required in INTOR was determined from reliability analysis, taking into account the number of such components present. The major benefits that would result from long-term component operation in INTOR are: 1) definition of failure modes; 2) determination of failure rate and distribution; 3) determination of failure recovery time; and 4) identification of design improvements.

Conclusion: There is a substantial incentive to achieve at least 2–3 $MW\cdot a\cdot m^{-2}$ fluence for structural materials damage and component reliability testing.

4.3.3. Blanket testing requirements

The requirements upon INTOR operation were assessed for several different types of blanket tests – neutronics, tritium recovery, materials compatibility, heat recovery and breeder lifetime. In general, the neutronics tests do not impose significant requirements upon operation. The minimum continuous operating times were estimated for solid breeder tritium recovery tests and for heat recovery tests to be about 65 h (about 950 continuous cycles) and about 1250 s (about 5 continuous cycles), respectively. Solid breeder microstructural and thermophysical property changes with radiation are estimated to saturate at about 0.2 $MW\cdot a\cdot m^{-2}$ neutron fluence. Since INTOR will have a lower volumetric nuclear heating rate than DEMO, it will be necessary to simulate the DEMO thermal-mechanical conditions in specially designed test modules in order to obtain relevant information on tritium release, heat transfer and materials compatibility.

4.4. Tritium and blanket

Five areas were addressed: 1) tritium permeation through the first wall, limiter and divertor; 2) tritium containment as it relates to tritium contamination of the reactor environment and its impact on personnel access; 3) key issues related to the design and performance of the tritium-producing blanket; 4) tritium inventory of the processing systems; and (5) tritium safety considerations.

4.4.1. Tritium permeation

Investigation of tritium permeation and inventory in the first wall, limiter and divertor indicates large uncertainties in a number of areas. For all plasma-side materials, characterization of the surface conditions in the actual reactor environment and of the effects of neutron damage trapping results in a large uncertainty in both tritium permeation and inventory. For some materials tritium diffusivities and solubilities are highly uncertain.

The best estimate that can be provided, at present, for the steady-state tritium permeation rate to the coolants for the first wall, limiter and divertor is in the range of 10^2 to 10^4 $\mathrm{Ci \cdot d^{-1}}$. The recommended tritium concentration in the coolant water is 0.1 $\mathrm{Ci \cdot L^{-1}}$.

Several methods for separating tritium from water are available. The capital and operating costs are strongly dependent on the process flow rate, which is proportional to the permeation rate and varies inversely with the allowable tritium concentration in the coolant loop. For a permeation rate of 10^3 $\mathrm{Ci \cdot d^{-1}}$ and a coolant concentration of 0.1 $\mathrm{Ci \cdot L^{-1}}$ the volume of coolant water processed would be $10\,000$ $\mathrm{L \cdot d^{-1}}$. The corresponding capital cost is approximately \$50 M, with an operating cost of approximately \$2 M per year.

Conclusion: Tritium permeation is not a feasibility issue for INTOR. However, the economic penalty can be quite large if the tritium permeation rate is $>10^4$ $\mathrm{Ci \cdot d^{-1}}$. A substantial improvement in the data base for tritium permeation parameters is needed.

A clear goal for the first wall/limiter/divertor designs and for R and D programmes is to ensure that the tritium permeation rate is $<10^4$ $\mathrm{Ci \cdot d^{-1}}$. The time to reach steady-state levels for the tritium inventory and permeation rates can be long depending on neutron damage trapping. This can be an important consideration for components with short life such as the limiter and divertor. The estimated end of life tritium inventory in the first wall, limiter and divertor is in the range of 0.1 to 1.0 kg. Future effort should address the concerns associated with a significant buildup of tritium inventory in the in-vessel components.

4.4.2. Tritium contamination in reactor room

Key aspects related to tritium contamination of the environment of the reactor building and the associated impact on the maintenance personnel access were examined. The potential tritium contamination of the reactor room was evaluated under normal, maintenance and accident conditions.

Under normal operation, the tritium source term is estimated to be <30 $\mathrm{Ci \cdot d^{-1}}$. The dominant leakage comes from the water coolant system. During maintenance, the tritium source term can be as high as 10^3 $\mathrm{Ci \cdot d^{-1}}$, with most of the tritium released during removal of the different torus components. An accident could result in a release of up to 10^5 Ci.

Personnel access for maintenance is the dominant factor in sizing the required atmosphere detritiation system. The capital cost of the atmospheric detritiation system increases rapidly with increasing leak rate into the reactor room. The best strategy is a combination of a detritiation system and bubble suit. The tritium concentration level should be maintained at 5 to 500 $\mu\mathrm{Ci \cdot m^{-3}}$ by the detritiation system. Bubble suits with an independent air supply are required if the worker spends extended periods at levels of >5 $\mu\mathrm{Ci \cdot m^{-3}}$. Bubble suits provide a safety factor of at least 100 against tritium, which means that a worker in a bubble suit can work in a 500 $\mu\mathrm{Ci \cdot m^{-3}}$ environment. Even in an accident, tritium concentration must be maintained below 500 $\mu\mathrm{Ci \cdot m^{-3}}$ or access is not possible.

Tritium release from the reactor building to the environment is another factor that favours lower tritium concentration in the reactor room. If the tritium release to the reactor room is kept below 10 $\mathrm{Ci \cdot d^{-1}}$, the room can be ventilated directly to the environment, while keeping the room concentration below 5 $\mu\mathrm{Ci \cdot m^{-3}}$.

Two additional key points related to the maintenance strategy emerged from this study:

a) It appears that some degree of tritium protection will be required for maintenance personnel to enter the reactor room. The worker efficiency can be reduced by a factor as large as two. Earlier studies for INTOR which established the benefits of personnel access did not account for such a penalty for worker efficiency. Therefore, re-evaluation of the maintenance strategy is necessary.

b) The utilization of robotic units for maintenance operations, particularly those requiring less than a day, appears sufficiently meritorious to deserve a serious study. More than half of the maintenance and repair operations are expected to be completed in less than a day. Therefore, the 24 h 'wait period' for the gamma radiation dose to decay significantly reduces the availability. Furthermore, a significant number of maintenance operations will involve higher tritium exposure than the average case considered above; for example, repair of coolant lines.

4.4.3. Tritium-breeding blanket

A detailed study of solid breeder materials was performed. New data on properties have been obtained, especially for Li_2O. The recommended temperature range for tritium recovery is 400 to 650°C. Radiation effects on tritium inventory may be significant in terms of increasing it to an unacceptably high level of several kilograms. These effects are believed to be strongly dependent on temperature and to be suppressed at temperatures less than 700° to 750°C for Li_2O. Estimated blanket tritium inventory is in the range of 0.3 to 1.0 kg. At present, the available data favour Li_2O among the solid breeding materials.

The sensitivity of the blanket design to power variation was studied. The margin for power variation within the allowable temperature range is of the order ±50%.

For liquid breeder material, attention has been focused on the eutectic $Li_{17}Pb_{83}$. New experimental data on the basic properties of this material have been made available and assessed. They concern in particular physico-chemical data, compatibility with steels, and chemical reactivity with air and water. The blanket conceptual designs have been reviewed, and problems related to accident conditions and reactor layout (preheating systems to keep the breeder always liquid) have been investigated. The loss of coolant accident appears the most severe one. Current experimental data and design studies confirm the interest in blankets based on liquid $Li_{17}Pb_{83}$, but new measurements are needed, in particular to provide data on tritium recovery schemes, corrosion and embrittlement effects of structures, and breeder-coolant interaction effects.

4.4.4. Tritium inventory

A study of the tritium system was undertaken to determine more accurately the contributions to the tritium inventory and the possible sources of tritium release in an accident. The principal new result, relative to Phase I, is that the blanket tritium processing system could have a 500 to 1000 g inventory. Thus, the total tritium inventory, including that in the blanket, is estimated to be 4 to 6 kg. The largest credible tritium release is estimated to be less than 10 g.

4.4.5. Safety considerations

Safety aspects and environmental impact of INTOR were analysed. The most logical means of determining the design goals for tritium concentration and tritium release limitations would be an optimized economic analysis that considers all appropriate factors. Since design details of relevant tritium systems components are not available, accident analysis of those systems should be considered as very preliminary.

At normal operation the main sources of tritium releases are the primary coolant, torus and fuellers. The analysis has shown that it is feasible to maintain tritium concentration in the coolant at a level of $0.1\ Ci \cdot L^{-1}$ and to have tritium releases from the primary coolant into the reactor room several Ci/d. Preliminary analysis has also shown that tritium releases from other major sources (torus, fuellers, blanket) can be kept at the level of 10^{-2} to $20\ Ci \cdot d^{-1}$. The potential tritium releases at accident situations are determined by tritium inventories being in vulnerable form in reactor systems and components.

Recommendation: The design goal for tritium release to the environment under routine operation should not exceed $20\ Ci \cdot d^{-1}$.

To satisfy this design goal and to be able to avoid the necessity of a routine reactor room atmosphere detritiation system, a design goal for tritium release to the reactor room of $10\ Ci \cdot d^{-1}$ is recommended. In the case of tritium releases to the environment on the level of $20\ Ci \cdot d^{-1}$ the permissible site boundary dose of less than 5 mrem·a^{-1} (i.e. 100 times less than according to ICRP recommendations) can be achieved with the application of a stack of moderate height of 1 to about 40 m.

The analysis of tritium inventories and of different accident scenarios has shown that the highest accidental tritium releases are possible at ruptures of some blanket components and at cryopump accidents. However, the limitation of accidental tritium releases on the level of 10 g is feasible.

Recommendation: The design goal for accidental tritium release limitation should be kept on the level of 10 g.

The analysis of environmental impact has shown that the risk due to tritium releases from INTOR, both routine and accidental, if the above-mentioned goals are met, will be negligibly small.

4.5. Mechanical configuration

4.5.1. Toroidal field (TF) coil size

The objective of the Mechanical Configuration critical issue study was to produce the new design concept with a significant reduction in the size of the tokamak device while maintaining the plasma size and performance of the Phase-One INTOR design [10]. As a result of the new concepts developed in this phase, we have produced a new design configuration with a reduction in TF coil size of approximately 15% from a bore of 7.7 × 10.7 m to 6.6 × 9.3 m. The magnetic ripple for this design has increased to only approximately 0.9%. Due to the strong influence of TF coil size on the other tokamak systems such as poloidal field (PF) coils, power supplies and machine structure, the overall cost of the device may be reduced by about 12%.

Recommendation: A reduced size (≈15%) TFC system should be adopted for INTOR.

4.5.2. Torus segmentation

Torus segmentation is the most important consideration for maintenance and access to the plasma chamber. An evaluation was made of several torus segmentation and disassembly concepts that would be consistent with the reduced size TFC system. Torus segmentation concepts that were considered include: 1) number of torus sectors equal to number of TF coils, with (straight-line) radial-horizontal sector removal (similar to Phase-One INTOR concept); 2) number of sectors twice the number of TF coils, with non-radial, translational, horizontal sector removal. A 12-sector option was selected for the INTOR Phase-One conceptual design based on the fact that it provided the maximum access surface for penetrations to the plasma chamber and was the simplest design approach for assembly and removal of the blanket and first-wall sectors.

For the reduced-size TF coil option, the 12-sector design appears to be at the limit for which this concept applies. For this reason, it seems prudent to also develop a 24-sector concept in the event that additional flexibility is required, e.g. concerning the cross-sectional area of the TF coils.

Recommendation: A 24-sector concept should be developed for INTOR, but the 12-sector segmentation concept should be retained, too.

4.5.3. Universal design concept

The INTOR Workshop participants have also developed a 'universal' design concept. The 'universal' design concept incorporates several major changes from the INTOR Phase-One concept related to the vacuum topology, the torus, and the structural design. The configuration also provides sufficient flexibility to accommodate the uncertainty involved in the choice of bulk heating and impurity control methods. The universal design can accommodate neutral-beam injection or ICRF for heating and a poloidal divertor or pumped limiter for impurity control.

An extensive evaluation of several concepts for separate and combined vacuum boundaries for the torus and superconducting magnet vacuum systems was performed. The most favourable vacuum topology was determined to be one in which the torus and superconducting magnetic structure are separated by a common dual vacuum boundary. Several possibilities for torus closure were studied, including a separate vacuum door and integration of the vacuum closure with the torus shield. The latter option is recommended, in order to improve access and maintenance.

An equal and a multiple torus sector configuration have been identified as potential candidate designs for future reactor configuration studies. The torus chamber in both configurations has been modified to accommodate the pumped limiter and the poloidal divertor impurity control system. In the divertor option, the collector plates are tilted in the toroidal direction to prevent a leading-edge condition that would result from the triangular gap formed by adjacent divertor modules.

In the event that the design was specialized to the choice of the pumped limiter, the plasma off-set from the TF horizontal centreline could be slightly reduced. The height of the torus and of the TF coils could then be reduced by approximately 0.5 m, with beneficial effect on the PF coil configuration.

There are many PF coil locations which can provide the magnetic flux to shape a divertor plasma. This is equally so for a pumped limiter plasma. Since a divertor-shaped plasma requires the greatest flexibility in forming a fixed null position for low and high beta, a PF configuration that can accommodate a divertor plasma can also provide a limiter-shaped plasma by only changing the coil currents, not their position. In fact, such a PF coil configuration was used for the 'universal' design configuration.

Recommendation: Wherever appropriate, a 'universal' design concept should be adopted for future critical issues studies in the INTOR workshop. The design concept can then be specialized as design option decisions are made in the future.

4.6. Magnetics and torus electromagnetics

The critical issues studied during Phase-Two-A focused on questions of magnet feasibility, required R and D, electromagnetic effects of disruptions, and requirements for active and passive stabilization.

4.6.1. Magnetic systems

4.6.1.1. Conductor/coolant options for TF coils

Three candidate conductor/coolant options have been evaluated for INTOR: 1) He I, forced-flow, Nb_3Sn; 2) He I, bath-cooled, Nb_3Sn/NbTi; and 3) He II, bath-cooled, NbTiTa. Specific configurations for each of these options were considered viable for INTOR requirements, that is, none exhibits inherent characteristics which would render the option unacceptable from a performance or reliability standpoint. The data base for design and operational experience with existing systems differ somewhat among options, but no large-scale demonstration at the 11–12 T level has been accomplished for any of the cases. Estimates have shown that the required radial build for the winding and TFC case and centripetal force support structure of 1.18 m estimated for INTOR Phase One may be somewhat low. Stress analyses to date have concluded that careful selection of the coil and support structure can lead to operational stress/strain levels within the limits imposed by Nb_3Sn as the superconductor material and by cyclic fatigue of welded stainless-steel structure. Specific areas (e.g. insulated breaks in the cold structure and inclusion of high-strength insulating material for out-of-plane support) require further investigation. Limiter operation leads to somewhat lower stresses/strains.

A study shows that conductor cost over the 8 to 12 T range has a steep dependence on field, but one without discontinuous jumps as conductors or cooling approaches are varied. Niobium titanium and niobium tin show nearly identical cost at 8 T for 4.2 K operation. At 10 T, NbTi conductor at 1.8 K is approximately 30% less costly than Nb_3Sn, and at 12 T, NbTiTa is 30% cheaper than Nb_3Sn.

4.6.1.2. Fault studies

Preliminary studies of fault chains and selected fault scenarios have been performed. It appears that faults can be designed against (e.g. short-circuited single TF coil) or shown to be highly unlikely (e.g. rupture of TF coil). Since about 200 days is estimated for the replacement of one TF coil, a more thorough understanding of fault conditions is needed.

4.6.1.3. Toroidal field coil R and D requirements

The LCP will provide a large-scale demonstration of NbTi and Nb_3Sn at 8 T in a tokamak configuration. In addition, Tore Supra will test NbTi at 9 T and 1.8 K and T-15 will test Nb_3Sn at 8 T. National 12-tesla programmes will test conductors at 12 T, but not provide large-coil experience. The lack of large-coil fabrication and operation for any field above 8 T represents a development and demonstration gap since the field for INTOR is significantly above this level.

Conclusion: The INTOR TF magnet design appears feasible, provided the required R and D is performed. The type of tasks which are needed and the time required to perform them are independent of the INTOR field level, assuming that it is in the 10 to 12 T range. Fault studies should be continued to provide a basis for design base fault definition.

4.6.1.4. PF coil distribution studies

Systematic studies of PF coil distributions have been completed, involving several hundred runs of an MHD equilibrium code. The studies show a strong effect with overall TF coil size. The reduced size TF coils adopted in Phase Two A allow PF system cost reductions of 25% over the baseline scale. A 'universal' PF coil configuration which accommodates both the divertor and pumped limiter option over the full β-range is feasible. There is, as yet, no concept for segmented, multi-turn superconducting coils which appears to be reliable and suitable for remote maintenance of internal coils.

4.6.1.5. PF coil design

Design studies have examined both pool-boiling concepts and internally cooled conductors. In general, both concepts are shown capable of satisfying the coil requirements. For example, studies have confirmed that an 8 T central solenoid is feasible for the six-second start-up ramp, and that it can utilize either conductor approach. Other studies have shown that it is feasible, in principle, to build the large ring coils. There are, however, possible requirements for winding several high-current superconducting magnets on site and in parallel. All studies were based on NbTi as the superconducting material, since it seems possible to keep the field level $\leqslant$8 T. Restriction of AC losses is a major constraint on PF coil design. The new, relaxed plasma start-up scenario has made possible an all superconducting PF magnet solution. Power supply studies indicate that more than one standard PF conductor and a subdivision of power converters into low-and high-voltage units are desirable.

4.6.1.6. PF system fault studies

Studies have examined fault currents induced in various PF coils under the assumption of a terminal short across a given coil. Certain well-coupled PF coils will be subjected to abnormal currents under these conditions, thus requiring active protection schemes. Further analysis will be required at a later stage in the design before design base faults can be defined. Up to 500 days may be required to replace a large ring coil.

Conclusions: The INTOR PF magnet design appears feasible. The required R and D is not yet well established. The coils can be all superconducting, use NbTi and be located external to the TF coils. Fault studies should be continued to establish a design base fault definition.

4.6.2. Torus electromagnetics

4.6.2.1. Disruption effect

The rapid flux change associated with a plasma disruption induces eddy currents in the torus structure which interact with the toroidal and poloidal fields to produce forces and torques. Estimates indicate that these loads are significant and require integration into the sector structural design, but that the load level is manageable. Design measures (e.g. subdivision, insulating breaks) are available to reduce the loads, if necessary.

Estimates of the electromagnetic loads and voltages on the limiter following a disruption indicate a strong dependence on the time constant of the torus and on the electrical continuity of the limiter in the toroidal direction. Results indicate that limiter segmentation is essential to reduce loads to tolerable levels and that many segments may be required to reduce the voltage between segments to a low level. Further study to determine the allowable voltage level for design purposes is necessary. All conclusions are based on a disruption time of 20 ms.

Transient heating of the TF coils and induced currents in the PF coils (assuming low-impedance supplies) appear tolerable. The disruption-induced voltage across gaps in the toroidally segmented torus structure requires further study since its estimated magnitude of 20–40 V lies in the range that may be critical for breakdown to occur. Determination of the threshold level for arcing is needed.

4.6.2.2. Passive stabilization

Passive stabilization against vertical (and radial) displacements by means of saddle coils, complete shells and partial shells has been examined and their relative effectiveness assessed. Coils or shells must be no further out than the blanket outer boundary if they are to meet the following criterion: a given displacement of the plasma must generate induced currents which in turn give rise to a restoring field equal to the field change encountered by the plasma in being displaced. INTOR conditions require approximately 0.1 $T \cdot m^{-1}$ of displacement. Coils or shells placed beyond the blanket outer boundary, for example at the shield outer boundary, will not meet this criterion. A residual growth rate of about 20 s^{-1}, which was assumed to be achievable with 24 passive loops, is needed to restrict the active feedback power to a tolerable level.

Complete shells are approximately twice as effective as optimally located single coils in providing restoring forces. Partial shells which are toroidally continuous need occupy only about 180° poloidally to be nearly as effective as complete shells. Segmented blanket modules which are not toroidally continuous have time constants approximately 1/2 to 1/3 that of continuous shells. The time constant associated with the naturally occurring structures appear somewhat low, and, therefore, higher conductive plates or passive coils should be seriously considered and integrated into the design at a future date. More extensive studies with improved plasma models are necessary to determine if plate-like or shell-like passive elements are more effective stabilizers since their induced fields may be expected to be more uniform in the plasma region. Design integration of the passive loops poses potential problems in limiting the influence of return conductors and in obtaining adequate tritium breeding.

Conclusion: Passive vertical stabilization elements that achieve residual growth rates of 20 s^{-1} must be located no further away from the plasma than the outer boundary of the blanket. Their time constant should be 100 to 200 ms. Design integration of these elements is a critical issue.

4.6.2.3. Active stabilization

The requirements for active stabilization depend on the characteristics of the passive stabilization system. The longer the time constant of the passive system, the slower can be the response of the active system, provided that the passive system meets the criterion for restoration of plasma position. Only passive systems sufficiently close to the plasma meet this criterion, whereas all toroidally continuous shells inside the coil locations, no matter where they are located, impede penetration of its active control fields. On the basis of a residual growth rate of 20 s^{-1} with active control windings outside of the shield, the active control power to cope with vertical excursions in the cm range is a few to a few tens of MVA. This power is estimated to be about a factor of two lower with the limiter than with the divertor. If the control coils are located external to the TF coils, approximately one order of magnitude more power is required.

Recommendation: Active control coils at the position of the outer shield should be given serious consideration. These coils could be saddle-shaped with one coil per major torus sector. The integration of these coils into the design is necessary.

4.6.2.4. Start-up

Delay of start-up voltage, field disturbance by induced currents and vertical field distortion must be compensated. Either an adjustment of the PF coil current scenario or the addition of normal coils inside the TF coils are possible solutions to the voltage delay problem. An adjustment of about +30% in a large ring coil or −20% in the central solenoid, at 200 ms, would be needed.

Start-up coil energy requirements depend on the location of the start-up coils and on the characteristics of any toroidally continuous partial shells which have been included for passive stabilization or for structural reasons. Such start-up coils should be utilized only during times short compared with the plasma build-up. If a toroidal shell with a time constant of 150 ms is present, and if the voltage must be held for 500 ms after reaching 35 V (i.e. after the 150 ms delay) the required energy becomes 170 MJ for the inner location and 580 MJ for the outer location. These reasonable energies will not become prohibitive, provided voltages need not be sustained for long periods and that loop voltages be kept low. Further study on the interplay of system time constants, to arrive at a design optimized with respect to start-up, passive stabilization, active stabilization and control, is essential to refine techniques and models currently in use.

4.7. Cost-risk-benefit

4.7.1. Cost reductions

A number of possible changes in design option, relative to the Phase-One INTOR design [10], were considered in order to achieve an overall reduction in operating or capital cost. The TF coils can be reduced in bore by about 15% before the physics limitation on field ripple is approached. Mechanical configuration studies confirmed that the torus maintenance was still feasible with this reduction in TFC size. Thus, a reduction of about 12% in capital cost (including the reduced PFC system costs) is possible.

Simplification of the PFC system leads to a capital cost reduction of about 4% when the poloidal divertor is replaced by a pumped limiter. If the TFC is further reduced in size, as is mechanically feasible with the

TABLE IV. COST-RISK-BENEFIT SUMMARY

Fluence ($MW \cdot a \cdot m^{-2}$)	0.2	2.0	6.6	10.0
Risk-benefit figure-of-merit (FM)	0.41	0.71	0.79	0.78
Relative costs				
Capital	0.89	0.98	1.00	1.09
Operating	0.34[a]	0.52[b]	1.00[b]	0.94[c]
Total	0.58	0.74	1.00	1.04
Benefit/cost ratio				
FM/capital	0.53	0.72	0.79	0.72
FM/total	0.89	0.96	0.79	0.75

[a] TBR = 0.0.
[b] TBR = 0.6.
[c] TBR = 1.0.

limiter, an additional cost reduction of about 4% is possible.

Cost reductions can be achieved by reducing the heating power margin (75 → 50 MW) and by changing from neutral-beam injection to ICRF. This is judged to be technically feasible, implying a reduction of about 8% in capital cost.

Placing a tritium breeding blanket on the inboard, as well as top and outboard, of the torus to achieve a tritium breeding ratio of about one would reduce the operating cost by about 14%, while increasing the capital cost by about 3%. The engineering consequences have not been examined.

Conclusion: Capital cost reductions of about 20%, relative to the Phase-One INTOR design, are feasible.

4.7.2. *Cost-risk-benefit assessment of performance objectives*

Neutron fluence is a convenient characterization of the INTOR performance objective. Many radiation-damage-related testing capabilities are directly related to fluence. In addition, the accumulation of long-term component operation reliability data can be correlated to the fluence, for a fixed neutron wall load.

A cost-risk-benefit comparison of alternatives with different fluence objectives was performed. A risk-benefit figure-of-merit was defined by taking into account the importance of the information from INTOR to the design basis for the DEMO and the probability that a given INTOR alternative could provide the information required of INTOR. This probability comprised two factors: 1) the design objective (e.g. fluence goal) of the alternative; and 2) the risk associated with achieving the design objective. The figure-of-merit was normalized so that the ideal INTOR alternative, which provided all the information necessary to supplement that from the base programme and complementary facilities to complete the DEMO design basis, would have a value of unity. This figure-of-merit and the costs are shown for four different alternatives, ranging in fluence capability from 0.2 to 10.0 $MW \cdot a \cdot m^{-2}$, in Table IV. There is relatively little difference in the base capital costs, but a rather large difference in operating costs, which results from the different operational lifetimes and tritium costs. The first case only produced about 40% of the information (in an importance-weighted sense) that is required of INTOR, while the other cases produce about 70 to 80% of the required information. The costs of producing the missing information elsewhere, or alternatively, the risk of designing the DEMO without it, has not been factored into these numbers. The additional cost that might be associated with achieving the higher availability that would be necessary with the higher fluence cases has also not been factored into the capital costs.

On the basis of the cost-risk-benefit analysis, the low-fluence alternative is rejected because it produces about 40% of the information (importance-weighted) required from INTOR, and the high-fluence alternative is rejected because of the high risk associated with the design. There is no clear preference between the two intermediate fluence alternatives.

The evaluation of testing requirements in Section 4.3 indicated a strong incentive to achieve about (2–3) $MW \cdot a \cdot m^{-2}$ for structural materials properties (tensile, microstructural change) data and for component reliability data. While there was incentive to achieve higher fluences, this incentive was not so compelling.

Recommendation: INTOR should be designed to achieve a neutron fluence $\phi \approx 3$ $MW \cdot a \cdot m^{-2}$ with a high probability of success. The design should allow for operation to achieve higher fluences, but with a lower probability of success.

5. DESIGN DESCRIPTION

Based on a realistic assessment [5] of the anticipated status of plasma physics research and technology development a few years hence, the conceptual design has been further developed for an INTOR device which could fulfil the objectives worked out during Phase One [10] and which are listed in a previous section.

TABLE V. INTOR DESIGN SPECIFICATIONS

GEOMETRY	
Chamber major radius, R	5.2 m
Chamber volume	320 m^3
Chamber surface area	380 m^2
PLASMA	
Plasma radius, a	1.2 m
Plasma elongation, K	1.6
Plasma aspect ratio, A	4.4
Burn average beta, $\langle\beta\rangle$	5.6%
Poloidal beta, β_p	2.6
Average ion temperature, $\langle T_i\rangle$	10 keV
Average ion density, $\langle n_i\rangle$	1.4×10^{20} m^{-3}
Energy confinement time, τ_E	1.4 s
Plasma current, I_p	6.4 MA
Field on chamber axis, B_T	5.5 T
Safety factor (separatrix), q_I	2.1
Peak thermonuclear power, P_{th}	620 MW(th)
Neutron wall load, P_n	1.3 $MW \cdot m^{-2}$
Toroidal field ripple at outboard edge	0.9%
OPERATION	
Burn time, Stage I/Stages II and III	100/200 s
Duty cycle, Stage I/Stages II and III	70/80%
Number of pulses (lifetime)	7×10^5
Maximum availability goal	50%
HEATING: ICRF	
Number of launchers (active/spare)	3/1
Power at start-up	50 MW
Frequency	85 MHz
Pulse-length capability	CW
START-UP ASSIST: ECRH	
Power	10 MW
FUELLING	
Method	pellet injection and gas puffing
IMPURITY CONTROL	
Method	single-null poloidal divertor
Collector	low plasma temperature at plate: W medium plasma temperature at plate: Be
Power to divertor	80 MW
FIRST WALL	
Power to first wall (excluding neutrons)	44 MW
Outboard: material	D_2O-cooled SS 316
thickness	12 mm
Inboard: material	H_2O-cooled SS 316
thickness	14 mm
Lifetime	15 a (full)
BREEDING BLANKET	
Material	D_2O or H_2O, SS 316 Li_2O
Breeder temperature	400–650°C
Thickness	0.5 m
Location	outboard and top
Breeding ratio	>0.6
Tritium extraction	continuous He purge
TRITIUM FUEL SYSTEM	
Tritium flow rate	64 $g \cdot h^{-1}$
Annual tritium consumption at 25% availability	7 $kg \cdot a^{-1}$
Isotopic enrichment	cryogenic distillation
TRITIUM INVENTORY	
Breeding blanket	0.5–1.0 kg
Storage	2.3 kg
First wall/divertor	0.1–1.0 kg
Tritium handling systems	1.4 kg
TORUS VACUUM SYSTEM	
Initial base pressure	10^{-7} torr
Pre-shot base pressure	3×10^{-5} torr
Pumps	compound cryopumps
Pumping	through divertor chamber
TOROIDAL FIELD COILS	
Number	12
Bore	6.6 m × 9.3 m
Conductor	Nb_3Sn and/or NbTi
Stabilizer	Cu
Maximum field	<12 T
POLOIDAL FIELD COILS	
Total flux	110 V·s
Breakdown voltage	35 V for 0.3 s
Location	external to TF coils
Conductor	NbTi
Maximum allowable field	8 T
POWER SUPPLIES	
Stationary loads	200 MW
Pulsed energy storage	14 GJ

MECHANICAL CONFIGURATION

Twelve or twenty-four blanket sectors assembled with straight-line or translational horizontal motion through windows between TF coils
Semi-permanent inboard, upper and lower shield forming primary vacuum boundary on inner surface
Final closure of primary vacuum boundary on outer boundary of removable torus sectors
Test modules inserted horizontally at mid-plane
All superconducting coils in a common cryostat, except lower ring coil in a separate cryostat
Dedicated sectors: 4 ICRF
1 ECRF
2 fuelling
3 testing
2 I&C

SHIELDING	
Inboard (non-breeding blanket and shield)	1.1 m
Outboard (breeding blanket and shield)	1.5 m

Emphasis was given to developing the conceptual design self-consistently and in sufficient detail in certain important areas so that the critical problems could be identified and the consequences of certain major design decisions could be investigated. This was a necessary prerequisite for studying the critical issues during this phase. The results of these studies have been incorporated into an improved INTOR design concept.

The major features of the improved INTOR design concept are specified in Table V, and a cross-sectional view is shown in Fig.1.

An analysis of the magnetics, MHD equilibrium and stability, energy transport, plasma heating and impurity control has been made to support the plasma physics parameters specified for INTOR. The INTOR plasma, operating with the indicated parameters should achieve an ignited burn with an average thermonuclear power output of 620 MW(th). The plasma current, in excess of 6 MA, should adequately confine alpha particles. The value $\langle\beta\rangle = 5.6\%$ is somewhat greater than the theoretical limit, but is considered to be within the

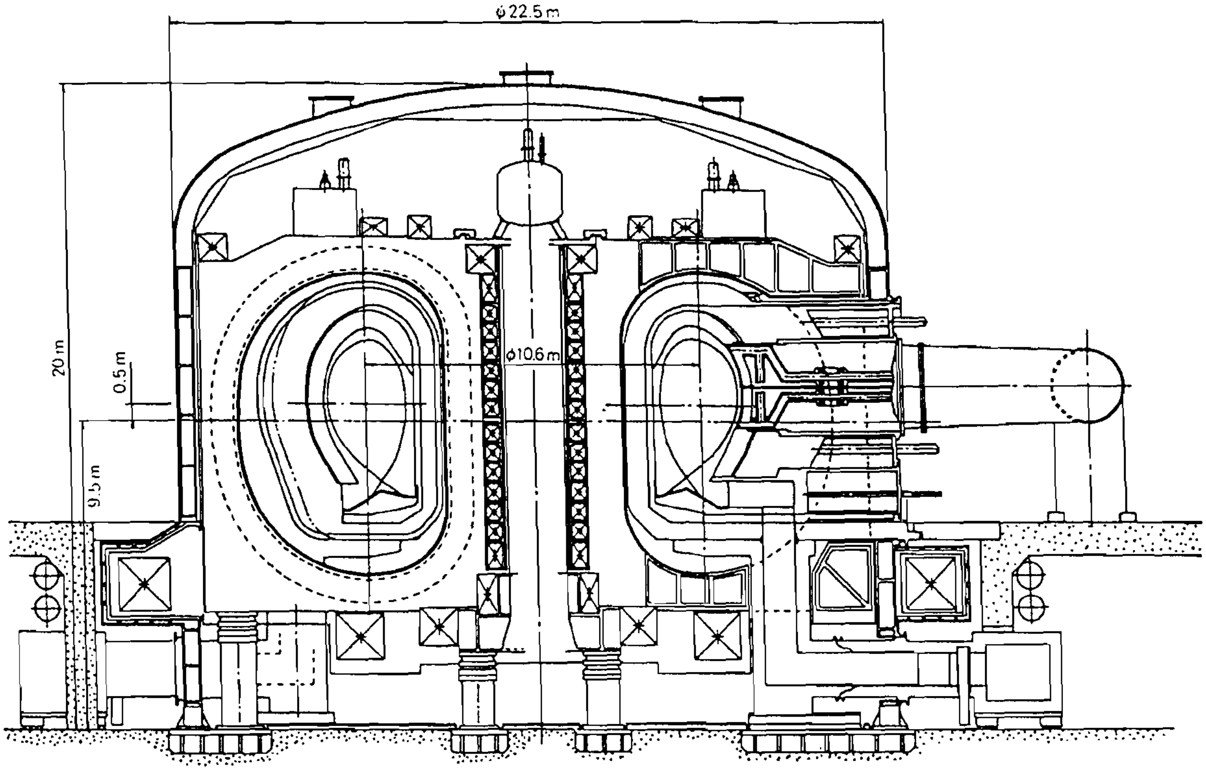

FIG. 1. INTOR universal configuration (artist's concept).

achievable range. Upon comparison with a pumped limiter, preference is given to a poloidal divertor to exhaust helium and to prevent heavy impurities from reaching the plasma in order to achieve the 200 s burn time, which was set at about one fifth of the theoretical magnetic surface diffusion time. Based upon the present best estimate of plasma energy transport losses, the predicted alpha-heating power should exceed that required for ignition. The 50 MW of ion cyclotron heating power allows the plasma to be heated to ignition. Neutral-beam injection (75 MW, 150 keV) is adopted as the back-up option.

A single-null poloidal divertor, with the chamber at the bottom, has been chosen for impurity control, while a pumped limiter is under study as an alternative solution. Analyses indicate that it is possible to magnetically form the divertor channels and to control the separatrix motion to within several centimetres with coils external to the toroidal field coils. A relatively short channel length is adequate because of the high-density mode of divertor operation. Two plasma scenarios have been considered; each with high edge density ($\geqslant 5 \times 10^{19}$ m^{-3}). The reference case is characterized by a low plasma temperature ($\approx$30 eV) at the divertor plate and a high ion flux. The alternative case considers a reduced ion flux with a medium plasma temperature ($\approx$100 eV). For the reference case a tungsten plate bonded to a copper heat sink has been chosen as the best solution. For the medium-temperature case, the use of tungsten is not possible since its self-sputtering coefficient becomes larger than one. Low-Z materials such as beryllium have, therefore, been considered in the form of protective tiles bonded to the heat sink. The tungsten tiles and copper heat sink would need to be replaced every few years. The analysis in support of the single-null divertor 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 engineering design of the divertor collector plate, and the engineering design of a maintainable divertor. As an alternative solution for the impurity control, a double-edged, shaped limiter at the bottom of the plasma chamber has been considered. The temperature regime for the limiter is uncertain in the region 30 to 100 eV and therefore the surface has to be protected with a material (low-Z) with a self-sputtering coefficient which remains smaller than one for all temperatures. A beryllium protection tile bonded to a copper heat sink is the proposed solution.

The mechanical configuration design was driven from the outset by the requirement to provide sufficient access in order to facilitate maintenance and assembly/disassembly. During Phase Two A, a significant reduction in the size of the toroidal and poloidal field coils was achieved while maintaining the plasma size and the performance of Phase One. A semi-permanent inboard, upper and lower shield forms the primary vacuum boundary. The removable torus sectors fit within this semi-permanent shield. Two options are possible, one with 12 and one with 24 removable sectors, which assures the feasibility of reducing the TF coil size. These torus sectors are partially (outboard and upper) the tritium-producing blankets and partially (inboard and lower) the heat-removal shields. The final closure of the vacuum boundary on the outboard is at the outer boundary of the removable sectors. Once the vacuum boundary is cut, each torus sector can be withdrawn horizontally with straight-line or translational motions through a 'window' between adjacent toroidal field coils. The divertor channel is broken up into twelve modules which are removable with straight-line horizontal motion between the toroidal field coils. The single-null divertor was chosen over the double-null divertor in order to achieve this more simply maintainable mechanical configuration.

Semi-permanent, superconducting toroidal and poloidal field coils will be enclosed in a common, semi-permanent cryostat, thus completely separating the cold and warm structures. Only the lower outer PF coil may have a separate cryostat for ease of maintenance. All poloidal field coils will be external to the toroidal field coils and superconducting. Both forced-flow and pool-boiling conductor designs have been developed for the toroidal and poloidal field coils, and in addition a superfluid pool-boiling conductor design using NbTi has been developed for the toroidal field coils. Each of these conductor concepts is under active development, and a final decision can await results from the development programmes.

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

For the purpose of studying the critical issues a 'universal PF coil arrangement' was defined. By proper coil currents this arrangement can produce both single null divertor and pumped limiter configurations. This coil arrangement is by no means optimized for minimum power consumption but has the merit of high configurational flexibility. Once the final configuration is selected, the poloidal field circuit has to be optimized.

A water-cooled, stainless-steel first wall with a panel-type construction is specified. This first wall is expected to last the full lifetime of the device, provided that the melt layer which is predicted to form on the inboard section during a plasma disruption is stable. A tritium-producing blanket will be installed from the outset of operation in order to reduce the operational cost. A solid breeder (Li_2O) blanket that covers the outboard and upper surfaces of the plasma chamber can produce more than 60% of the tritium consumed in INTOR.

An availability of at least 30% during Stage III is required in order for INTOR to carry out its testing mission over a reasonable period of time, but the design should allow higher availabilities (up to 50%) but with decreasing probability of success. Extrapolation of present reliability data leads to availability estimates of about 30–40%, depending upon the degree of redundancy.

It should be noted that the critical issues studies during the past two years concentrated on a limited number of systems, with the consequence that not all systems are specified to the same level of consistency and detail. The present work suffices to define the concept, and to compare some alternatives; achieving consistency on a detailed level is a task for a later phase.

6. MACHINE OPERATION AND TEST PROGRAMME

A preliminary operation and test plan has been developed to provide insight into the design and operational requirements that must be imposed on INTOR. This plan has been developed using judgement as to where INTOR fits within an international fusion development plan, as discussed in Section 2,

TABLE VI. TYPES OF TESTING IN INTOR

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

and taking into account the complementary roles that will be played by other plasma physics experiments and technology testing facilities. A summary of the types of testing included in the test plan is given in Table VI.

Some tests will require nearly continuous operation for some period of time. In particular, some of the tritium recovery tests will require a 70% duty cycle and continuous operation for one week to one month in order to reach equilibrium conditions. On the other hand, thermal-hydraulics testing will require only a 50% duty cycle and continuous operation for approximately one hour. Achievement of the required fluence for reliability testing of blanket modules and other components will require machine availability of about 30%.

The projected test schedule is shown in Fig.2. The figure assumes that INTOR is designed to achieve 3 $MW \cdot a \cdot m^{-2}$ with high probability of success. Achieving higher fluences (6.6 $MW \cdot a \cdot m^{-2}$) is probable and is planned for. The test schedule will have to remain flexible to accommodate any changes in the device operating lifetime, which will be decided during the early phase of operation. As indicated, plasma physics testing will dominate Stage I operation. Stage II testing will consist primarily of plasma engineering and blanket engineering tests and other tests where frequent change-out is required. A minimum time of one month between scheduled reactor shut-downs has been established to permit test change-out without unduly affecting reactor availability. Stage III testing will be devoted to longer-duration tests which do not require frequent reactor shut-down.

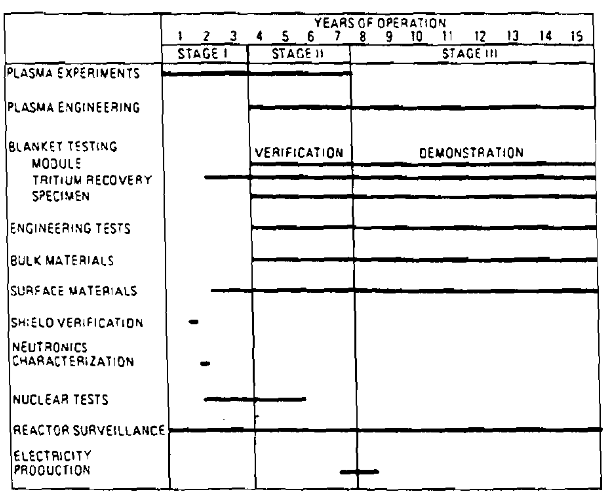

FIG.2. INTOR test schedule.

7. SCHEDULE

The implications of three different INTOR schedules, corresponding to construction initiation in 1986, 1989 and 1992, have been evaluated. The Phase-One INTOR construction schedule was adopted as the baseline schedule. Time spans for each schedule activity were assumed to be unchanged for each start date. The schedules corresponding to the three different start dates are shown in Fig.3.

A review of the INTOR project schedule and its associated logic revealed three basic areas to be considered in conjunction with the impacts imposed by the stipulated start dates. The first of these was the time available for start-up of the INTOR project. The second area concerned the compatibility of the

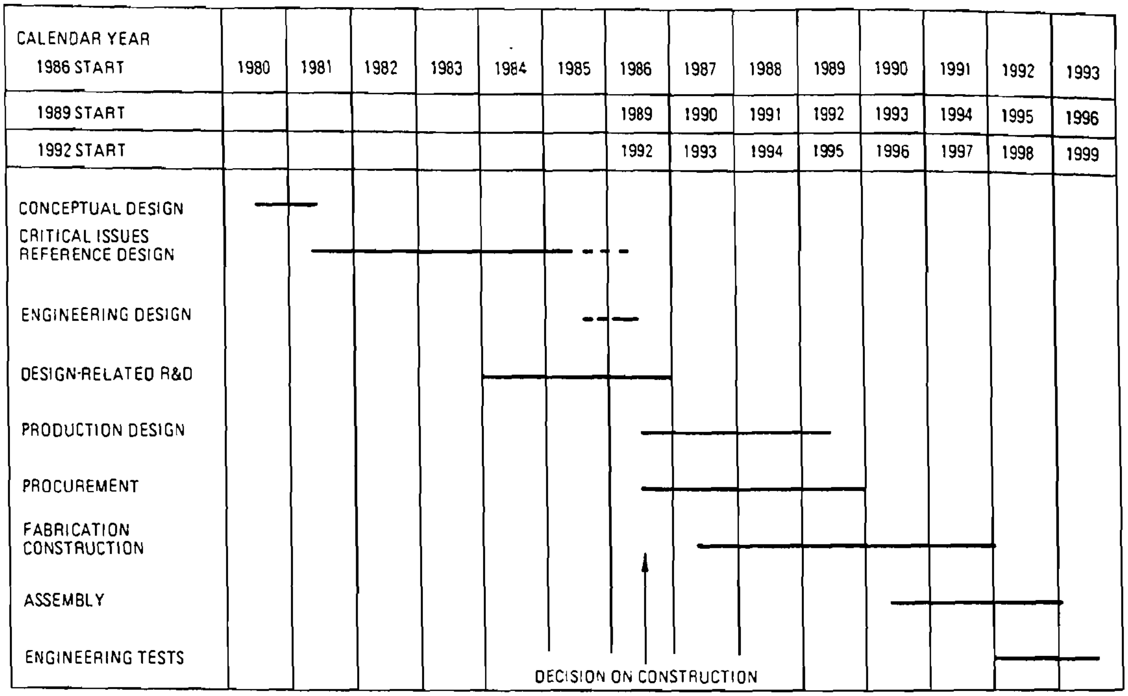

FIG.3. INTOR design and construction schedule.

INTOR alternative schedules with the complementary R and D programmes and facilities. The third area considered the impact on the 'DEMO' machine's schedule arising from the prescribed INTOR start dates.

With respect to initiation of the INTOR programme, and irrespective of judgements on the available data base, a 1986 start date would allow only two years from 1984 to accomplish all the tasks required to lift constraints on start of production. The more significant of these tasks include: establishing the organization for the construction project; establishing the necessary funding arrangements; selecting the construction site; developing the configuration design to a level consistent with the issuance of requests for proposal by major contractors; selecting major contractors; and completing plans for the implementation of production design. Two years are considered to be insufficient for accomplishment of the foregoing tasks; but a time span of five years would seem to be sufficient.

With respect to the compatibility between the supporting R and D programmes and the alternative start dates it is obvious that the later the start date, the greater the amount of test data that can be obtained before making the decision on actual start of construction (which constrains the start of production design and the start of procurement). Of particular importance are the time scales for those results which are expected from facilities having essential lead-times for their construction and operation; the JET – TFTR – JT-60 – T 15 class of experiments is an example. The first two of these devices started to collect data in 1983. Certainly, several years of operation are required for the completion of the data base to be provided by these machines. Also from this point of view one may conclude that the 1986 starting date is inappropriate.

Finally, the influence on the construction date of DEMO can be assumed to be linear.

8. RESEARCH AND DEVELOPMENT

During the course of the work on critical issues in the INTOR activity in 1981–83 several limiting uncertainties were identified which inhibit further development of the design concept. Specific R and D programmes which could resolve these uncertainties have been identified, as summarized in Table VII. These specific R and D programme recommendations supplement the broad R and D programme needs that

TABLE VII. SPECIFIC R AND D PROGRAMME RECOMMENDATIONS

		Priority[a]
Physics		
P.1	Plasma behaviour near beta limits	1
P.2	Confinement scaling in auxiliary heated tokamaks	1
P.3	Plasma equilibrium control	1
P.4	Plasma profile control	1
P.5	Reactor prototypical ICRF heating	1
P.6	ICRF code development	2
P.7	RF start-up assist	2
P.8	High power LH and EC heating	(2)
P.9	Quasi-steady-state mode of operation	(1)
P.10	Characterization of high- and low-temperature edge regimes	1
P.11	Edge particle and energy fluxes	1
P.12	Divertor channel behaviour	1
P.13	Impurity transport	1
P.14	Limiter pumping characteristics	(1)
P.15	Molecular and low-temperature charge-exchange data	2
Nuclear		
N.1	Self-sputtering yield of main candidate materials	1
N.2	Sputtering by tritium	2
N.3	Properties of re-deposited metals	1
N.4	Irradiation effects on non-replaceable high-flux materials (60 dpa)	1
N.5	Irradiation effects on replaceable high-flux materials (30 dpa)	1
N.6	Tritium permeation and inventory, including irradiation effects	1
N.7	Eutectics development	(1)
N.8	14 MeV neutronics integral experiments	2
Engineering		
E.1	High-power ICRF system demonstration	1
E.2	Improved structural concepts for first wall/divertor/limiter	1
E.3	First wall outgassing procedure	2
E.4	Tritium pellet injector	1
E.5	Superconductors for fields above 10 T	1
E.6	Low-loss, high-current 8 T superconductors	1
E.7	TF coil mechanical and electrical properties	1
E.8	Safety circuits to cope with short-circuiting coils	2
E.9	Low-loss poloidal field coil concept	1
E.10	Intermediate-scale PF coil demonstration	1
E.11	Computational tools for transient electromagnetics	1
E.12	Torus maintenance methods and procedures	1
E.13	Adequate torus resistance	1
E.14	Voltage withstand criteria for components within the torus	2
E.15	Pump development	2
E.16	PF coil power supply system optimization	2

[a] Priority
1 – Required for the INTOR reference design: highest priority.
2 – Required for the INTOR reference design: secondary priority.
Numbers shown in parentheses refer to tasks which are not required for the INTOR reference design but which are of importance for other design options.

have been defined in the Zero Phase [5] and the specific R and D programme recommendations made in INTOR Phase One [10].

9. CONCLUSION

The critical issues studies that have been carried out during 1981–83 have clarified our understanding of certain major technical issues which affect the feasibility, cost and engineering design tractability of a next-generation tokamak reactor and have advanced our knowledge of how to design such a device. Some of these intensive studies have been carried to a point where further significant progress must await additional experimental information. Specific R and D recommendations have been formulated to this end. In other areas, such as impurity control, a continuation of the intensive study is warranted. In addition, several new areas in which intensive study would lead to an improvement in the concept were identified.

The results from the critical issues studies have led to improvements in several aspects of the INTOR design concept. The improved design concept described in Section 5 provides a sufficient basis for the immediate implementation of an engineering design of a next-generation tokamak reactor. This concept will naturally become defined in greater detail and probably will evolve in some particulars as the new information from future studies and the R and D programmes becomes available.

10. RECOMMENDATION FOR FUTURE WORK

Four technical issues have been identified to which an intensive, multidisciplinary effort should be devoted: 1) the integrated physics and engineering study of the impurity control system should continue; 2) an integrated physics, engineering, nuclear and magnetic study of the implications of electromagnetics requirements upon torus design, tritium breeding potential and poloidal field system design should be performed; 3) an integrated physics, nuclear and engineering study of the technological requirements and design implications of RF heating, current drive and start-up assist should be performed; and 4) a comprehensive study of tritium containment, radiation shielding and other factors which affect personnel access for maintenance should be performed and contrasted with the technological requirements for almost entirely remote maintenance. In addition, 5) a study of the technical feasibility of partitioning the detailed design and component production tasks among participants so that all could share equally the benefit of technology development is recommended.

A continuing and expanded R and D activity of the Workshop is recommended, also. The design basis assessment of Zero Phase should be upgraded and new R and D requirements should be defined based upon the results of the critical issues studies. The impact on the INTOR concept of new results in R and D should be assessed on a continuing basis. The implications for the INTOR concept of possible shortfalls in the required R and D should be evaluated.

The design concept should be evolved and better defined on the basis of results from these studies and information from ongoing R and D programmes.

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NEW BOOKS

Controlled Fusion and Plasma Physics – Parts I & II: Contributed Papers

> Proceedings of the 11th European Conference, Aachen, 5 – 9 September 1983, published by European Physical Society, Plasma Physics Division, September 1983, in the series Europhysics Conference Abstracts, paperback, Parts I and II: pp 548, ref. Volume 7D, no prices indicated.

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