SPECIAL TOPIC

INTERNATIONAL TOKAMAK REACTOR — PHASE TWO A, PART III

Executive Summary of the IAEA Workshop, mid-1985 to 1987

INTOR GROUP*

ABSTRACT. The INTOR Workshop started in 1979 with the aim to define, design and construct the next major experiment in the world tokamak programme. The Workshop was organized to run in phases. The paper describes the results of Phase Two A, Part III, which ran from mid-1985 to end 1987 and was concentrated on critical issues affecting the feasibility, objectives and cost of the INTOR design for a tokamak engineering test reactor. The paper also summarizes the results of INTOR Workshop studies on DEMO requirements and innovations which hold the promise to lead to improvements of the tokamak reactor concept. Instead of the updating of the early INTOR conceptual design of Phase One, which is necessary and which was planned for Part III of Phase Two A, a summary is presented which contains all principal conclusions from the work on critical issues and innovations as well as their implications relative to the INTOR design concept. Finally, a critical and comparative analysis of the existing INTOR-like designs is given.

CONTENTS.

* *EC*	*Japan*	*USA*	*USSR*
G. Grieger	S. Mori	W.M. Stacey, Jr.	B.B. Kadomtsev
E. Coccorese	N. Fujisawa	D.A. Ehst	A.M. Epinatiev
F. Engelmann	T. Honda	C.A. Flanagan	B.N. Kolbasov
F. Farfaletti-Casali	H. Iida	Y.-K.M. Peng	A.I. Kostenko
M.F.A. Harrison	B. Ikeda	N. Pomphrey	A.S. Kukushkin
A.F. Knobloch	T. Kobayashi	D.E. Post	R.N. Litunovskij
D. Leger	T. Mizoguchi	D.L. Smith	I.V. Mazul
E. Salpietro	T. Okazaki	P.T Spampinato	V.I. Pistunovich
P. Schiller	T. Tsunematsu		G.E. Shatalov
W.R. Spears			V.L. Vdovin
G. Vieider			S. Yakurin
J.-G. Wégrowe			

1. INTRODUCTION

The International Tokamak Reactor (INTOR) Workshop was installed as a collaborative effort among the European Community, Japan, the USA and the USSR, to be conducted under the auspices of the International Atomic Energy Agency (IAEA), in terms of reference as 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 were set forth by the IFRC, to draw upon capability that exists worldwide:

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

The INTOR activity was carried out in phases. At the end of each phase, the participating governments reviewed the progress of the activity and decided 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 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 parnters. 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 worldwide 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 devices as they evolved in this process.

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

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 their home countries (20–40 man-years of effort by each partner). The national conceptual design contributions to the Phase One INTOR Workshop have been published (see Ref. [2]). These contributions formed the basis for the INTOR conceptual design, which is documented in Ref. [2].

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 teams, to review the ongoing design work and to take decisions on the evolving design. The decisions taken at each Workshop session were then incorporated into each 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.

By the same activity also a number of critical technical issues were identified with the potential for considerable improvements in the feasibility, cost and engineering configuration of the INTOR design concept if further work would lead to more advanced solutions than were available at the end of Phase One. With this in mind, the INTOR Workshop was extended into Phase Two A, which was split off from Phase Two, "detailed design of the experiment". In Phase Two A, emphasis was placed on the resolution of the critical technical issues mentioned above. This work turned out to be fruitful and rewarding, but also time consuming. As a consequence, Phase Two A was extended twice so that, finally, there were three parts of Phase Two A: Part I from July 1981 to the end of 1982, Part II covering 1983 to mid-1985, and Part III covering mid-1985 to 1987. Part I was concentrated on plasma performance, impurity control and first wall, testing requirements, tritium and blanket, mechanical configuration, magnetics and electromagnetics, and cost–risk–benefit. The last item is a cost–risk–benefit comparison of alternative INTOR designs with different fluence objectives if INTOR is considered to be the only step which has to bridge the gap between present-day devices and DEMO.

These critical issues studies improved our understanding of major technical issues affecting the feasibility, cost and engineering design tractability of a next-generation tokamak reactor, advanced our knowledge of how to design such a device and led to improvements in several respects of the INTOR design concept. Some of these intensive studies have been carried to a point where further significant progress must await additional experimental information. Specific R&D recommendations have been formulated to this end. In other areas, such as impurity control, a continuation of the intensive study was warranted. In addition, several new areas were identified for which intensive studies held the promise to lead to additional improvements of the INTOR concept. This work was 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 that was developed in the critical issues studies led to certain improvements in the INTOR concept. The work in the Phase Two A, Part I, INTOR Workshop was reported in the national contributions (see Ref. [3]) to the Workshop and in the report of the Phase Two A, Part I, Workshop [3].

Part II of Phase Two A was concentrated on impurity control, plasma heating and current drive, transient electromagnetics, maintainability and technical benefit of partitioning INTOR component design and fabrication. The last item provided a rather detailed discussion of the merits and demerits of various forms of international collaboration in constructing and operating an INTOR-like device. Also a reassessment of the scientific and technical database supporting the INTOR concept was undertaken. As a consequence of these studies, some of the major parameters of the INTOR design concept were modified. This was done in order to illustrate in which way the INTOR design concept might have to be modified to take account of the new results, leaving the self-consistent updating to be done in a later INTOR phase.

This work was carried forward by teams working in their home institutions under the direction of the INTOR Workshop participants, who met five times (for a total of 11 weeks) in Vienna to define and review the work. The work of the Phase Two A, Part II, Workshop is reported in the national contributions (see Ref. [4]) and in the report of the Phase Two A, Part II, Workshop [4].

Part III of Phase Two A started with the intentions: (i) to address the following critical issues: impurity control, beta and confinement, heating and current drive, electromagnetics, configuration and maintenance, and first wall and blanket; (ii) to reassess the DEMO requirements; (iii) to study potential innovations that are not yet supported by developed physics or technology but hold the promise to lead to improvements of the tokamak concept; and (iv) to incorporate the results of all the work done during Parts I to III of Phase Two A in an updating of the INTOR conceptual design.

In the course of the Part III studies, discussions started on an international collaborative activity — the joint design and construction of a next-step facility with aims similar to those of INTOR. At this moment, the work orientation of the INTOR Workshop was rediscussed and it was decided to stick to the work on critical issues, DEMO requirements and innovations, because these items are of immediate relevance also to any near-term tokamak design activities, and, instead of the intended introduction of the results of the work into an updating of the INTOR conceptual design, to prepare a short summary of those results and to use this part of the capacity of the INTOR Workshop for a critical analysis of existing INTOR-like designs.

This work was carried forward in the usual manner, i.e. by teams of experts working in their home institutions under the direction of the INTOR participants who met in Vienna five times (for a total of about ten weeks) over the two years of Part III of Phase Two A, to define and review the work and to take decisions. This work was supported and/or prepared by INTOR-related experts' meetings on impurity control, innovations, DEMO requirements, current drive, confinement, disruptions and comparison of INTOR-like designs. The work done in Part III of the Phase Two A INTOR Workshop is reported in the national contributions [5–8] to the Workshop and in the report on Phase Two A, Part III.

The cumulative INTOR work to date was a major factor in laying the ground-work for proceeding to the design of the next major experiment in the world tokamak programme. Its objectives and general characteristics were defined. A preliminary conceptual design was developed early in the INTOR process and was used to identify critical technical issues and R&D requirements. The critical technical issues have been partly resolved and partly elucidated by studies that were initiated by the Workshop. The methods used in reactor design by the four groups were further developed and compared in order to test their consistency. The national designs of the four groups and the physical and technical constraints upon which they are based were evaluated. Finally, ways in which the INTOR design concept would have to be updated, based upon this work, were identified.[1]

2. SUMMARY

2.1. GENERAL INTRODUCTION

During Part III of Phase Two A of the INTOR Workshop, studies on 'critical issues' were carried out in six areas: impurity control, operational limits and confinement, current drive and heating, electromagnetics, configuration and maintenance, and blanket and first wall. An assessment of innovations proposed by the INTOR-related Specialists' Meeting on this subject [9] for further consideration was combined with the work on critical issues. In addition, the physics, engineering and nuclear databases of the INTOR design concept underwent another re-evaluation. The results of all this work are summarized in Sections 2.2.2 to 2.2.9.

An analysis of INTOR-like national designs and their comparison with INTOR started with an INTOR-related Specialists' Meeting held in Vienna from 23 to 27 March 1987. The conclusions drawn during this meeting, together with results of related discussions within the INTOR Workshop, are summarized in Sections 2.3.1 to 2.3.7.

To gain the time necessary for doing this analysis of INTOR-like designs within the INTOR Workshop activity of the present Phase, it was decided to cancel the original intention to update the INTOR conceptual design by incorporating the results of the three INTOR phases on critical issues as well as the results of the assessment of innovations and of the now available database. Instead, the conclusions of all the INTOR work are summarized in Section 2.4. Also, it is discussed how the INTOR conceptual design would have to be changed as a consequence of this work.

2.2. CRITICAL ISSUES, INNOVATIONS AND DATABASE ASSESSMENT

2.2.1. Introductory remarks

Three of the critical issues of Part II of Phase Two A of the INTOR Workshop were continued in Part III because considerable progress was expected from further work. These issues were impurity control, current drive and heating (with emphasis on current drive) and electromagnetics. New topics tackled were operational limits and confinement, as well as configuration and maintenance, the latter one aiming at a critical comparison of different maintenance approaches. Blanket and first wall, the sixth topic of the critical issues, was taken up again because it was expected that new information would allow the conclusions of earlier phases to be evolved.

In the course of the Part III studies, the INTOR Workshop was also charged with an analysis of proposed innovations to improve the tokamak concept. A collection of proposals and a first analysis were made during an INTOR-related Specialists' Meeting [9]. The proposed innovations which looked promising and of sufficient impact were then taken up by the

[1] Note that in this paper the references are either to INTOR reports on earlier phases [1–4] and on the national contributions to Phase Two A [5–8] or to the reports on INTOR-related meetings on tokamak concept innovations [9], fusion reactor critical issues [10] and plasma disruptions [11]. Particularly the national reports on which INTOR is based contain extensive lists of references to which the reader is referred.

relevant critical issues groups for further analysis. The results of this work on innovations are contained in the sections on critical issues, together with comments on the recent evolution of the database.

2.2.2. Impurity control

During Part III of INTOR Phase Two A, work on impurity control was directed towards: (a) updating the previous assessment of experimental data, (b) apprais-ing the relevance to INTOR of a number of innovative concepts [9], and (c) improving the consistency of plasma edge modelling, with particular emphasis on model validation and improved prediction of divertor performance in INTOR-like reactors. In addition, scoping studies were initiated in order to provide a wider and more flexible approach to engineering design problems.

2.2.2.1. Experimental data

There has been a substantial amount of new data from both poloidal divertor and limiter experiments in tokamaks. Now there is further evidence that a diver-tor with an open geometry, of the type envisaged for INTOR, is capable of producing high recycling condi-tions which are desirable to minimize sputtering of the divertor target. The concentration of impurities within the main plasma is generally lower for a divertor than for a limiter. Nevertheless, the concentration of low-z impurities (notably oxygen) is affected less than that of high-z impurities. There is often substantial emission of radiation within the divertor region and this is indicative of high recycling. It appears that the H-mode can be most readily accessed by operation with a poloidal divertor. In contrast, H-mode operation has been observed only in one limiter experiment. A detrimental aspect of the H-mode is that, in certain conditions, it causes impurities to accumulate on the plasma axis. Also, the density at the plasma edge tends to be lower, and particle and power exhaust is often in bursts, which implies more demanding working condi-tions for the divertor plates. In the case of limiter operation, the temperature of the adjacent plasma is high and this is likely to lead to high rates of sputter-ing and erosion of the limiter surface. Exhaust of neutral gas can be quite efficiently performed by a pumped limiter.

2.2.2.2. Innovative impurity control schemes

Five innovative schemes for impurity control in INTOR have been considered, namely: (i) flow rever-sal of impurities as a consequence of co-injection of the neutral beams, (ii) formation of a stable radia-tive edge at the periphery of the plasma column; (iii) formation of an ergodic edge layer; (iv) burial of helium in the divertor region, and (v) use of liquid divertor plates. The first three schemes are not yet sufficiently well verified to be considered as candidates for the INTOR impurity control system. The last two schemes show promise, and further theoretical and experimental work, together with the appropriate design analyses, should be encouraged.

2.2.2.3. Plasma edge modelling

Improvements were made in the two-dimensional numerical models used for both interpretation and prediction of plasma edge behaviour. Comparison of the simulated edge conditions with conditions observed in experiments has enhanced the confidence in such modelling. In addition to complex two-dimensional modelling, several simpler models were developed which, when calibrated by comparison with the more complex models, can be used for scoping studies.

As a consequence of this work, it can be concluded that a high recycling divertor with a tungsten target is the best available concept for maintaining a clean main plasma and for ensuring low erosion of the target during a fully inductive operational scenario in INTOR. However, there are substantial uncertainties in plasma transport and in confinement requirements; therefore, further research and development together with continuous reassessment of the expected perfor-mance of this impurity control system in INTOR are needed. Maintenance of a clean plasma during an ignited burn with current drive is likely to be difficult because of the increased exhaust power and low plasma density. A stable radiating edge layer would be helpful in this respect and flow reversal would be a beneficial adjunct. Continuous current drive with a sub-ignited and clean plasma presents considerable problems and it is possible that it can only be achieved with very low power amplification factors. It is expected that, during inductive ramp-up, adequately high recycling conditions can be established within the divertor; this is, however, less certain in the case of non-inductive ramp-up. Consideration has been given to the use, in INTOR-like conditions, of low-z target material such as carbon and beryllium. During an inductively driven ignited burn, such targets sputter at rates which are approximately 10^2 times that of tungsten, but, even so, the target lifetimes exceed the likely duration of the physics phase of operation. Such materials are unlikely to be suitable for the more extended technology phase unless the divertor target surface can be readily renewed.

The relative merits of single-null and double-null poloidal divertors have been assessed. Experimental data indicate that H-mode operation can be most easily achieved with a single-null divertor, whereas the conditions needed for attainment of high elongation and ready control of vertical position favour the double-null divertor. Edge plasma modelling predicts that the peak power loading of the divertor target is lower in the case of the double-null divertor and that, on the basis of present understanding, adequate exhaust of the helium ash can be obtained by pumping only the bottom divertor chamber.

The concept of a divertor gas target, in which the plasma temperature is so low that volume recombination occurs and sputtering is negligible, has been reassessed. Unfortunately, the gas target approach requires a scrape-off plasma pressure which appears to be in excess of that envisaged for INTOR.

Recent modelling also shows that, under certain conditions, the plasma temperature at the target of a high recycling divertor may, during the burn phase, oscillate between a lower value of about 5 eV and an upper value of 15–20 eV. This effect could be beneficial for helium removal because it reduces the unfavourable effect of the thermal force acting on helium ions within the divertor layer. Nevertheless, improved modelling of impurity transport indicates that the pumping requirements for exhaust of helium ash may be more demanding than those specified in Phase Two A (Parts I and II), i.e. approximately $2 \times 10^5\ L \cdot s^{-1}$ of helium. However, in a compensatory sense, there are now data which show that a reduction in the helium pumping requirements can be obtained by optimization of the combined effects of pump-duct geometry and inclination of the divertor target towards the entrance of the duct. The innovation scheme of burial of helium in a continuously recoated metal layer within the divertor chamber could be a useful adjunct to the scheme using vacuum pumps.

Recent modelling confirms the previous prediction that divertor action provides efficient screening of the main plasma from impurities present in the edge. However, there remain many uncertainties (cross-field transport of impurity, ion sputtering by superthermal ions, etc.) and continuing experimental and theoretical studies are required.

An overall conclusion from the studies in Part III of Phase Two A is that for INTOR the poloidal divertor will have many advantages over the pumped limiter. Nevertheless, certain aspects of impurity control are at present uncertain and both the conceptional design and the operational scenario of INTOR should be flexible in these particular respects.

2.2.3. Operational limits and confinement

Operational limits to stable tokamak operation, disruptions and the confinement properties of tokamak plasmas are key issues for INTOR. In these areas an updating of the database was undertaken and innovative ideas were analysed, in particular with respect to enhancing the beta limit. A specific effort was dedicated to advancing the ideal MHD stability analysis beyond the limits explored in the past.

2.2.3.1. Beta limit

Experimental results on the operational limit to the plasma beta correspond to values of the Troyon factor g in the range of 3–3.5% ($T \cdot m \cdot MA^{-1}$). Improved analysis of the observations led to a slight decrease in g with respect to earlier findings. The same range of values was found in ideal MHD stability analyses of D-shaped plasmas having an aspect ratio A of approximately 4; this was extended to cover moderately large elongations ($K < 2$) provided the MHD q-value at the plasma edge (q_a) is more than 3. For larger elongation and lower q_a, there is an appreciable degradation of the Troyon factor. For such plasmas, up–down asymmetry causes a decrease of the beta limit, typically by 10–20%, for cases of practical interest.

Equilibria with a safety factor on axis (q_0) of less than 1 were considered, but a clear advantage of operating in this regime could not be identified. At extremely low aspect ratio ($A < 2$), several results indicated an enhancement of g up to about 4.5% ($T \cdot m \cdot MA^{-1}$), provided q_a is above a critical value, which increases with decreasing A.

For indented plasmas, the ideal ballooning stability limit is enhanced but the kink mode is destabilized so that efficient wall stabilization is essential for achieving high beta. It remains uncertain whether this can be provided. The second stability regime of ideal ballooning modes can be reached either in D-shaped plasmas with sufficiently high q_0 or in sufficiently indented plasmas. However, in these cases the kink mode instability is enhanced. Furthermore, a wide range of the plasma has to be nearly shear-free, a situation in which low-n internal modes tend to be destabilized. Also resistive destabilization of high-n modes is a concern.

In conclusion, moderately elongated D-shapes ($K \approx 2$) appear attractive for INTOR and allow the plasma beta to be enhanced, but more unconventional solutions to increase beta are at present too uncertain to rely upon.

2.2.3.2. Density limit

The density limit, if extrapolated according to scalings similar to the common Murakami–Hugill scaling, tends to be a more stringent limitation to the plasma pressure (at temperatures $T < 10$ keV) than the beta limit. However, the physics understanding of this limit is incomplete, and results for discharges with intense additional heating generally show an enhancement of the density limit and indicate deviations from the Murakami–Hugill scaling. In JET, the density limit appears when the radiation losses become equal to the power input — a criterion which, when extrapolated to INTOR, predicts an appreciably higher density limit than that obtained by the Murakami–Hugill scaling. Quantitative predictions, however, sensitively depend on the plasma edge parameters in this case. The limit to the safety factor, at least at modest values of beta and for conventional circular and D-shaped plasmas, is at $q_a \approx 2$ (which for poloidal divertor configurations is to be referred to 95% of the magnetic flux).

2.2.3.3. Disruptions

Operational limits are often due to the appearance of disruptions. The available database on major disruptions was analysed and the disruption specification for INTOR was updated. In view of results from JET and TFTR, very short energy quench times (of the order of 0.1 ms) must be considered to be a possibility in INTOR. The energy deposition profile in a poloidal divertor configuration remains unknown; therefore, a deposition of up to the total plasma kinetic energy on either the divertor plates or the first wall must be considered. The current quench rate is determined by the evolution of the plasma parameters after an energy quench, taking into account the electromagnetic coupling to the surrounding passive conducting structures and the capacity of the active position control device. If efficient position control is provided for, a maximum current decay rate of $3 \times 10^8\ A \cdot s^{-1}$ appears appropriate for INTOR.

2.2.3.4. Confinement

Extrapolation of plasma confinement to INTOR implies large uncertainties. It is considered that reliance on operating INTOR in a regime of improved confinement ('H-mode') is a reasonable working hypothesis, although there are still major uncertainties with respect to the reactor relevance of this regime. These are related, for example, to the possibility of controlled steady-state operation with limited impurity contamination and to the compatibility with RF heating and current drive as well as the compatibility with efficient power and particle exhaust for acceptable working conditions of the divertor plates and the first wall. Also the scaling of energy confinement in the H-mode remains uncertain, in particular with respect to plasma size and plasma temperature (or, equivalently, heating power), but to some extent also with respect to other parameters such as plasma current and density. These issues are key research items in the ongoing tokamak physics programme and are expected to be clarified within a few years.

2.2.4. Current drive and heating

One of the major objectives in Part III of Phase Two A was the evaluation of the feasibility of innovative ideas which would improve the tokamak concept. Current drive and heating were selected as potential candidate areas of improvement and several innovative methods were discussed in the Specialists' Meeting on Tokamak Concept Innovations [9]. On this basis, current drive has been emphasized in Part III, and a Specialists' Meeting on Non-Inductive Current Drive was held to assess the present status of experimental and theoretical developments [10].

The Specialists' Meeting on Non-Inductive Current Drive concluded that there has been considerable progress in this area. This progress includes all functions for which non-inductive current drive can be used, namely steady-state current drive, current ramp-up and transformer recharging, control of the current profile and MHD behaviour, as well as current initiation. Lower hybrid (LH) wave injection is the most developed technique for current drive; neutral beam injection (NBI) also shows promise, although its database is still more limited. Progress is sufficient for a tentative extrapolation to reactor conditions.

Recent high power experiments have made impressive progress in plasma heating by means of different methods, as shown by the following examples:

Neutral beam injection

TFTR: $T_{i0}/T_{e0} \approx 27/6$ keV, $\bar{n} \approx 3 \times 10^{19}\ m^{-3}$, $P_{inj} = 13.5$ MW

Lower hybrid waves

JT-60: $T_{e0} \approx 6$ keV, $\bar{n} = 1.7 \times 10^{19}$ m^{-3}, $P_{inj} = 2.4$ MW

ASDEX: $T_{e0} \leq 4$ keV, $\bar{n} \leq 10^{19}$ m^{-3}, $P_{inj} = 0.75$ MW

PLT: $T_{e0} \geq 5$ keV, $\bar{n} \approx 10^{19}$ m^{-3}, $P_{inj} = 0.6$ MW

Ion cyclotron waves

JET: $T_{e0}/T_{i0} \approx 7/5.5$ keV, $\bar{n} \approx (3\text{–}4) \times 10^{19}$ m^{-3}, $P_{inj} = 5.5$ MW

T-10: $T_{e0} \approx 8\text{–}10$ keV, $\bar{n} = 1.5 \times 10^{19}$ m^{-3} (He), $P_{abs} \approx 2.2$ MW

2.2.4.1. Lower hybrid waves

The efficiency of lower hybrid waves for current drive in its different modes (steady state up to a level of 2 MA, ramp-up, transformer recharge, start-up), for electron heating, for sawtooth stabilization and for profile control has been confirmed. A current drive efficiency of the order of that anticipated for INTOR-like devices has been achieved experimentally, and current drive operation at high density has also been demonstrated for adequately chosen frequency.

Convergence between theory and experiments is good in most domains. The use of lower hybrid waves is the best documented current drive method with a broad range of applications; thus, its position as a major candidate for most of the functions to be fulfilled by external power in next-step devices is reinforced. However, poor penetration in high temperature, dense plasmas is predicted.

2.2.4.2. High frequency fast waves

High frequency fast waves (HFFW) have the potential advantage of better penetration than lower hybrid waves and have a comparable current drive efficiency. The database has grown recently, but it is still too meagre to allow firm conclusions on the potential for reactor grade plasmas to be made.

2.2.4.3. Ion cyclotron waves

The position of ion cyclotron (IC) waves as one of the major candidates for plasma heating in next-step and future devices has been confirmed. Current drive by means of low frequency fast waves in the range $\omega < 5\ \omega_{CD}$ (avoiding ion resonances) has favourable prospects regarding efficiency, penetration and insensitivity to alpha particles. However, the present database on low frequency fast wave current drive is not sufficient to confirm this potential.

2.2.4.4. Neutral beams

The abundant database on beam injection confirms the potential of this method for plasma heating. In recent experiments on neutral beam current drive in TFTR and JET, driven currents of more than 0.5 MA have been achieved. In TFTR, the interpretation of these experiments is consistent with the existence of a significant bootstrap current.

2.2.4.5. Electron cyclotron waves

The usefulness of electron cyclotron (EC) waves for plasma heating has been demonstrated. The main handicap for this application is still the absence of efficient high power generators, although source development is making rapid progress. The utility of EC waves for startup assist and MHD stabilization has been proven as well as their unique potential for well localized absorption. High power coupling through the plasma edge is not a problem. This points to the use of EC waves in INTOR for functions requiring localized intervention (local current drive, including the innermost plasma region, control of magnetic islands).

2.2.4.6. Other schemes

Further methods, such as current drive and heating by ion Bernstein and Alfvén waves, or the use of synergistic effects by a combination of several methods, present interesting prospects. Bernstein wave heating in particular has given first attractive results.

2.2.4.7. Benchmark calculations

The results of dedicated studies on steady state and cyclic current drive and on profile control for INTOR-like devices led to the following additional conclusions. Benchmark calculations for INTOR ($\bar{T}_e = 20$ keV, $\bar{n}_e = 0.7 \times 10^{20}$ m^{-3}) have shown comparable current drive efficiency ($\gamma \approx (0.3\text{–}0.5) \times 10^{20}$ m$^{-2}\cdot$MA$\cdot$MW^{-1}) for a number of drivers: LH waves above 3 GHz; NBI; high frequency (0.2–1 GHz) fast waves; and low frequency (20–70 MHz) fast waves. For fast waves and NBI, the high value of $\bar{T}_e$ was chosen so as to increase γ.

A deuteron energy in the range of 0.4–0.8 MeV is optimum for NBI current drive for INTOR-like devices. Thus, one necessary condition for the use of

NBI is the successful development of the technology on the basis of negative ion acceleration.

Beam injection (or possibly use of IC or EC waves) to drive part of the current in the plasma core, with lower hybrid waves for current drive in the plasma exterior, is an attractive possibility for steady state current drive when the respective techniques are used in the regions where their performance is optimum.

Bootstrap current calculations for INTOR show that, if this neoclassical current contribution occurs, the external current drive power can be reduced by 50–90%, depending on the plasma density profile.

Acording to the modelling studies, current profile control by means of HFFW should be flexible. Neutral beams have also shown the potential of current profile control; this is restricted, however, by shinethrough constraints. Although current profile control by LH waves has been demonstrated experimentally, it cannot be relied upon in the core of reactor grade plasmas; it could be used in low density phases of the discharge. The potential of EC waves for local intervention was confirmed.

For transformer recharging or initial current ramp-up, LH waves are useful as a driver. At low $\bar{n}_e$ and $\bar{T}_e$ ($\approx 4 \times 10^{18}$ m^{-3} and ≈ 2 keV), a centrally peaked current (8 MA) can be maintained in the presence of a reverse electromagnetic force of -0.01 V$\cdot$m^{-1}; a power of $P_{CD} > 50$ MW and $Z_{eff} \approx 10$ are needed for INTOR recharge/ramp-up specifications. A positive feature of this scheme is less cycling of the wall loading and of β_p. However, the considerable shine-through poses difficult engineering problems.

2.2.4.8. Conclusions

In summary, several options for plasma heating can be extrapolated to reactor level on the basis of their present physical and technical achievements. For electron cyclotron waves, recent progress in the development of appropriate generators may soon overcome the main handicap of this method. Ion cyclotron waves, lower hybrid waves and neutral beam injection (at moderate energy level) can already be relied upon.

Among several proposed methods for current drive applications, only the use of lower hybrid waves, with a large database, and neutral beam injection, with a more limited database, offer reliable prospects. For cyclic current drive, the use of lower hybrid waves is the method of choice. For steady state applications, both methods have drawbacks: lower hybrid waves because of a limited penetration into hot and dense plasmas, and NBI because the high beam energy required necessitates the use of negative ion technology, the development of which is still at an early stage. A combination of the two methods appears at present the best option and is recommended for any iteration of the INTOR design.

The recent experimental evidence in favour of the existence of the bootstrap current, in conjunction with non-inductive current drive, invites a new optimism regarding the possibility of a steady state commercial reactor.

2.2.5. Electromagnetics

During Part III of Phase Two A, attention was concentrated on performing and comparing benchmark calculations of inverse equilibrium codes, on the study of operational scenarios, on plasma control issues, on the effects of a plasma disruption, and on innovative proposals to minimize the machine size and cost.

2.2.5.1. Benchmark calculations

The poloidal field (PF) coil configuration and currents are determined by the plasma current, the plasma current density distribution, the plasma shape and the allowable coil locations. The problem to be solved is finding the optimum PF coil locations and currents for the minimum objective function (e.g. superconductor mass, total current or magnetic energy). The result is not affected significantly by the objective function used.

Benchmark calculations were carried out by the four INTOR participants for specified double-null and single-null plasma configurations.

The results of all four participants were in very good agreement for the double-null plasma configuration. For the single-null plasma configuration the results were more diverse. Overall, the ampere-turns in the PF coils ranged from 110 MA to approximately 140 MA (about the same percentage difference for the double-null case). This was due mainly to the high multipolarity of the PF coil system and the effect of mutual 'screening' of closely located coils plus small differences in the plasma shape. The conclusion of this exercise was that all codes give the same results if the same constraints are used.

2.2.5.2. Operational scenario issues

The reference operational scenario for INTOR is the fully inductive scenario.

Alternative scenarios have been investigated with the aim of assessing their advantages and disadvantages. The steady state (SS) scenario has the advantage of lower AC losses in the superconducting coils and of a reduction of the power supply requirements for the PF coils and the number of loading cycles. It also permits of a vacuum vessel with lower toroidal electrical resistance, with consequent improvement of the growth time of the vertical stability but with higher power supply requirements for the active coils.

The SS scenario would permit some reduction of the machine size for the same plasma performance since the transformer requirements are reduced. The size reduction is greater for small plasma elongation than for large plasma elongation and triangularity because of the higher equilibrium current required in the central solenoid.

In a quasi-steady-state (QSS) cyclic scenario the plasma is inductively driven during burn for the available volt-seconds, and the plasma current and temperature are then reduced so as to permit an efficient transformer recharge. The enhancement of the burn time with regard to the purely inductive scenario is determined by the plasma configuration and the duration of the burn in the inductive scenario. For large elongation and triangularity of the plasma and long inductive burn, the improvement achieved by the QSS scenario is limited. Better improvement is achieved for machines with small plasma elongation and triangularity and with short inductive burn. Additional current profile control may be required in all cases.

2.2.5.3. Vertical versus horizontal access

Extensive analyses of the impact of the various possible access schemes on the torus components have been performed. The results can be summarized as follows: (a) For large elongation plasmas, less magnetic field energy is required in a vertical access scheme than in a horizontal access scheme. (b) For smaller elongation plasmas, there is no clear-cut advantage for one of the two access schemes regarding magnetic stored energy.

2.2.5.4. Closed loop plasma control

For the control of the radial position of the elongated plasma, only a slow control system is required since the plasma is stable against displacements [1]. The easiest method is to introduce a PF amplifier to control the current in the outer PF coils. In addition, feedback is needed for the remaining PF coils to control the plasma current and shape (elongation, triangularity, null-point position, etc.) as functions of time during the pulse.

The control of the vertical position of the plasma has to be fast since the plasma is unstable in this direction, with a growth time for INTOR and INTOR-like devices of the order of 5–20 ms. The radial stabilizing field can be produced by a pair of axisymmetric coils in series, but these cannot be placed outside the TF coils because the time varying field would produce very high AC losses.

Components which are further away from the plasma, particularly the vacuum vessel, have a slightly stabilizing effect on the plasma.

Estimates of the control power required for the INTOR stabilizing coils are in the range of 20–50 MW. The growth time of the vertical instability, for the same passive structure, is shorter for plasmas with large elongation than for plasmas with small elongation.

The single-null and double-null plasma configurations differ in the requirements they impose on the control system. For single-null configurations there is a coupling between radial and vertical displacements, but this is not the case for double-null configurations. Also the symmetry of the double-null system diminishes the problems of the plasma position detection system. The overall conclusion is that the active control system for double-null configurations is simpler, more reliable and less demanding in power than that for single-null configurations with the same average plasma elongation.

2.2.5.5. Plasma disruption

The various effects of plasma disruption have been investigated. The transfer of part of the plasma current from the plasma to the vacuum vessel leads to radial forces on the vessel, and the high resistance bellows sections in the vessel cause saddle currents which produce an overturning moment on the rigid sections of the vessel. For the first wall, which is more segmented and lacks toroidal continuity, the effects of plasma disruption are much smaller. Furthermore, in the case of vertical motion before the current quench, net vertical forces occur which may be large.

During a disruption, voltages occur between the gaps in the blanket system and the first wall. In the PF coils, voltages are induced during a disruption, but they are reduced to an acceptable level by the screening effect of the vacuum vessel. If the internal impedance of the power supply is low, the currents in the PF coils change and this may cause a coil quench. For modelling of the plasma disruption and estimates of the consequences, evolutive equilibrium codes (PROTEUS, D STAR, TSC, ECC) can be used. These codes assume that the plasma current and its distribution evolve according to the plasma models calibrated on present experiments. In this case, all external currents flowing in the passive structure and coils are modelled.

2.2.5.6. Innovations

The following innovations were proposed: (i) the use of high current density and high field superconductors for both TF and PF coils to reduce plasma radius and to minimize the machine size; (ii) the use of non-metallic structural materials in the magnet system to improve the electromagnetic characteristics such as penetration time and eddy current losses; (iii) the use of ferromagnetic steel for first wall/blanket materials.

If the current density and the magnetic field are increased in both TF and PF coils, the reduction in overall machine size can be significant. However, in both coil systems the current density is limited by structural requirements rather than by the superconductor material properties. The superconductor includes only a small fraction of the coil area, the rest being occupied by coolant, insulation, copper (for thermal quench protection) and structural reinforcement. Improvements in structural materials are seen as the major avenue to increasing the winding pack current density. All participants have national programmes researching into new jacket and case materials.

Non-metallic materials have limited application in the coil structural system because of their low stiffness. It would be necessary to make the components much thicker; in many critical regions, such as the bucking cylinder, this would increase the machine size.

The use of ferromagnetic materials in the first wall and blanket appears acceptable regarding electromagnetics, although the residual magnetism of such structures might cause problems during maintenance.

2.2.6. Configuration and maintenance

The following six topics were studied with a view to finding possible improvements of the INTOR design: Vertical and horizontal access configurations; application of shape memory alloys; ferromagnetic inserts for ripple reduction; PF coil redundancy; replacement of divertor and first wall to reduce the downtime; and containment of tritium and activated dust. Discussions on the first topic had started already during Part II of Phase Two A. The second, third and fourth topics came from the Specialists' Meeting on Tokamak Concept Innovations and were judged to be worthy of further discussion. The fifth and sixth topics are a continuation of the critical issues partly discussed in Part II of Phase Two A. The conclusions obtained are described in this section.

2.2.6.1. Vertical and horizontal access configurations

Each group developed vertical access configurations for comparison with the INTOR reference design and with other horizontal access configurations. These configurations include purely vertical access (USA), oblique access (EC, Japan) and a combined vertical and horizontal access (USSR). The main differences and impacts were examined with regard to the following features: torus segmentation and access port size, TF coil bore and supporting structure, active coils and passive structures, arrangements for plasma position control, bellows structure, reactor building size and shape, PF coil system, interfaces for heating and current drive devices, and maintenance procedures and equipment.

Some differences were pointed out, but it was difficult to make quantitative evaluations except for the impact on the PF coil system. It was found that for small elongation plasmas there are no clear advantages regarding stored energy and power supply. For large elongation plasmas the PF system requirements can be significantly reduced by locating the PF coils closer to the midplane of the machine.

For the vertical access configuration the available access port size is smaller than that for the horizontal access configuration, and thus a larger number of segments is required for the removable part of the torus structure. The greater segmentation permits a lower weight of the replaceable module, but tritium breeding efficiency and passive stabilization of the plasma are reduced. In the vertical access configuration the upper active control coils are segmented. The number of TF coils for the horizontal access approach should not exceed twelve when just a straight-line maintenance procedure is provided. For the vertical access configuration the number of TF coils can be increased, in which case the TF coil bores can be smaller.

For both configurations, the heating and current drive devices are installed horizontally at the midplane. For removal of first wall/blanket segments of the vertical access configuration, overhead maintenance handling equipment is required. This equipment must be capable of providing for translation, lifting and, in some cases, tilting of components. The discussion of facilities for the vertical and horizontal configurations was based primarily on work done for the Next European Torus (NET) and the Fusion Engineering Reactor (FER). In the FER concept the size of the reactor hall is such that the peripheral reactor equipment as well as equipment for maintenance and initial assembly can be accommodated. A rectangular building is preferred to a circular one because of the shorter span and easier construction. In the NET concept the upper deck of the rectangular reactor hall is a tight intermediate containment where personal access is not permitted. Its size is such that equipment for maintenance and initial assembly can be accommodated. The peripherals are housed in tight isolated cells around the reactor where personal access is permitted. This concept permits a smaller building span.

Generally, both approaches seem to be feasible; vertical access should be suitable for large plasma elongation and horizontal access should be suitable for small elongation.

2.2.6.2. Applications of shape memory alloys (SMA)

The following applications of SMA have been proposed: SMA sleeve pipe connector for coolant lines, mechanical quick connector, vacuum seal structure for the access door and SMA jack system for lifting of the blanket/shield module. These applications are judged feasible and have the advantages of reducing the reactor downtime and making it unnecessary to develop special tools such as an auto vacuum seal welder/cutter, an auto pipe welder/cutter and a bolt runner.

All of these appliances are to be placed behind the radiation shield. Theoretical studies have shown the possibility of using SMA also behind the blanket. Irradiation experiments will have to be performed.

2.2.6.3. Ferromagnetic inserts for ripple reduction

The effectiveness and the engineering feasibility of ferromagnetic inserts for ripple reduction were investigated. It was concluded that ferromagnetic inserts can reduce TF ripple effectively at the plasma edge and across the plasma cross-section, permitting a reduction in TF coil size and number.

Typical calculations show that the use of ferromagnetic inserts in the reference INTOR machine makes it possible to reduce the TF coil size by 0.5 m or to reduce the number of TF coils from twelve to ten. No major difficulties in the integration of ripple inserts into the reactor structure are envisaged. Forces acting on the inserts (10–25 MN) seem to be manageable.

2.2.6.4. PF coil redundancy

Measures to cope with failures in the lower PF coils were studied. The following options are possible: (i) to design PF coils so that they are replaceable without disassembly of the whole machine (untrapped coils); (ii) to provide redundant sections of coils in order to increase the coil availability; and (iii) to provide for later installation of a replacement lower outboard coil in an untrapped position. With the second approach, the time required to eliminate minor fault consequences can be reduced.

2.2.6.5. Replacement of divertor and first wall to reduce the downtime

Reviewing the INTOR design of Phase Two A, Part II, the following improvements were suggested:
(i) The baking time required after maintenance is approximately two weeks, which is the largest contribution to this operating period. Thus, vacuum or inert gas coverage should be maintained during the maintenance period so that no baking is needed.
(ii) Damaged components such as first wall armour tiles should be replaceable in situ without the necessity of removing large modular structures.

For divertor plate cassette replacement, it is recommended to use a transfer cask under vacuum or inert gas. In-situ replacement of armour tiles for the first wall, using a remote handling manipulator placed in an 'ante-chamber', is recommended. This can be realized by providing easily replaceable tile attachments and developing in-vessel inspection systems. However, the components (blanket and shield) should nevertheless be designed to be replaceable.

2.2.6.6. Containment of tritium and activated dust

A crucial point during maintenance, when the torus is opened, is tritium contamination of the reactor hall by outgassing of the reactor inner surface and additional contamination from the activated dust produced by sputtering or disruptions.

Several measures to limit or reduce the risk of contamination of the reactor hall have been proposed:
(i) Cooling of the transfer casks with a shiller system to minimize tritium outgassing; thus, any spread of

contamination inside the reactor hall can be prevented by containing the dust inside the casks. (ii) Use of plastic shroud enclosures for the containment of withdrawn pieces so as to stop the spread of tritium contamination. In this case, only the metallic dust afterheat poses problems. (iii) For small openings, the use of a separate aspiration system for air induction into the torus is considered as a means of preventing the spread of tritium and dust in the reactor hall. All the proposed methods are for general use in tritium technology and are expected to be applicable to later fusion reactors.

A large containment area located in the upper space of the machine, called tight intermediate containment, is proposed for the vertical access concept. Personnel access to this area would be excluded.

A further method, using carbonization to fix the dust by a glow discharge, was examined. However, this method is applicable only under vacuum conditions and there is no database to support its use in the INTOR maintenance procedure. In addition, the amount and grain size of dust particles are not yet known. More studies and sampling in existing devices (JET, JT-60, TFTR) are required.

2.2.7. First wall and blanket

2.2.7.1. Database for new materials

Data for new materials have been reviewed: austenitic and ferritic steels, graphite and carbon/carbon (C/C) composites, ceramic breeder materials, liquid breeder materials, divertor materials and magnet materials.

Austenitic steels. Additional information includes: sensitivity to aqueous stress corrosion, low temperature radiation effects on mechanical properties, effect of radiation on welds and experience in fabrication of wall components. Aqueous stress corrosion cracking of austenitic steels, particularly in the presence of irradiation, has been identified as possibly serious issue for the reference INTOR first wall/blanket structure. Significant loss of tensile ductility (uniform elongation much less than 1%) and fracture toughness have been observed after low temperature (<300°C) irradiation to ~15 dpa at a He/dpa rate of ~15. The fatigue properties of cold-worked material and solution annealed material are similar; radiation has only a modest effect on the fatigue properties of austenitic steel. The tensile and fatigue properties of austenitic steel structures bonded by the HIP process with >95% bonding ratios (in the surface area) are similar to the properties of the base metal. For the neutron fluence aimed at in INTOR, austenitic stainless steel is the only appropriate structural material for the low temperature first wall and blanket.

Ferritic/martensitic steels. The selection of ferritic/martensitic steels for the first wall/blanket structure in the low temperature, cyclic operating conditions of INTOR is not reasonable. The ductile-to-brittle-transition temperature (DBTT) of ferritic steels is increased by more than 200°C during low temperature (<300°C) irradiation. The effect of hydrogen is of particular concern at the low temperatures where the release of the internally generated hydrogen may be inhibited. The hydrogen effect may be even more critical for irradiated material (resulting in a change of the DBTT) and/or for welds.

Graphite and C/C composites. Three aspects of graphite and C/C composites have been evaluated, namely the form of redeposited material, radiation effects and tritium retention. A thin layer of amorphous carbon has been observed on the entire chamber wall of the present machines after operation with graphite. The amorphous material has a large capacity for retaining tritium (0.4 tritium atoms per carbon atom). The predicted radiation lifetime of nuclear grade graphite at 800–1200°C is <1 MW·a·m^{-2}. The effect of a high helium generation rate (He/dpa ~ 300) is unknown, but it may be significant at low temperatures (<1200°C). The C/C composites have the advantage of significant tensile strength and fracture toughness, but they are predicted to be significantly less radiation resistant than nuclear graphites because of their large anisotropy.

Ceramic breeder materials. Significant R&D has been conducted, the emphasis being on the candidate ceramic breeder materials; Li_2O in flowing helium at low moisture generally corresponds to thermochemical data. A significant transfer of LiOH from high temperature zones to lower temperature zones has been observed. The experimental data show that the swelling in $LiAlO_2$ remains low up to 3 at.% ^{6}Li burnup. The other two ceramic breeder materials show a considerable increase in swelling with burnup at irradiation temperatures of 700 and 900°C (4% for Li_2O, 2.5% for Li_4SiO_4 and <0.5% for $LiAlO_2$ at 700°C). In-pile tritium release from Li_2O, $LiAlO_2$ and Li_4SiO_4 has been examined. From Japanese results, the fastest tritium release seems to be from Li_2O, followed by tritium release from Li_4SiO_4 for similar temperature and grain size. The tritium release rates from ceramic breeders are sensitive to grain size and temperature. Especially for $LiAlO_2$, for which the tritium diffusivity is lowest, the required small grain size is of concern for other reasons. At sufficiently high temperatures or with small grain size, most of the tritium should be released from all candidate ceramic breeder materials.

Liquid breeder materials. Eutectic 17Li–83Pb alloy and aqueous lithium/salt solutions have been studied as liquid breeder materials. The estimated operating temperature limit based on corrosion by 17Li–83Pb is <400°C for austenitic steel and 450–500°C for ferritic steel. Tritium permeation from 17Li–83Pb causes concern because of the low solubility of tritium. There is serious concern regarding stress corrosion cracking by lithium salts and regarding tritium recovery.

Divertor materials. Primary candidate materials include tungsten for plasma facing tiles bonded to a copper heat sink. Liquid metal materials and helium burial materials are studied as innovative target materials. The cyclic fatigue data obtained by torsion fatigue tests for the tungsten–copper brazed specimens show that the lifetime of brazed specimens in the low strain range agrees well with that of copper; in the high strain range the lifetime depends on the strength of the interface of the bond material and tungsten. The loss rates of liquid metal films have been estimated for three candidate materials: lithium, tin and gallium. The evaporation losses are much less than the sputtering losses up to 500°C for lithium and up to 900°C for tin. The predicted sputtering rates for liquid metal surfaces do not differ substantially from those for solid state materials. The tritium inventory in a liquid lithium layer is calculated to be of the order of a few grams, and the one predicted for a liquid tin layer is even less. The corrosion data for gallium show that the refractory metals are generally highly resistant, the steels are moderately resistant, and nickel, copper and aluminium have very low corrosion resistance. For helium burial, the candidate materials are vanadium, nickel and iron. The minimum energy for effective helium trapping (~30 at.% trapping fraction) is estimated to be ~30–50 eV.

Magnet materials. The radiation limits for Nb_3Sn are estimated to be 2×10^{18} to 2×10^{19} cm^{-2}. The dose limits for epoxy insulators and polyimide insulators are predicted to be in the range of $(0.5–1) \times 10^9$ rad and 1×10^9 to 1×10^{10} rad, respectively, depending on the shear stress requirements.

2.2.7.2. Disruption analysis

Since disruptions have a major influence on the design and performance of the first wall and divertor, special emphasis has been placed on analyses and experimental observations of these effects. A parametric erosion analysis has been performed for the first wall and divertor materials, for a range of conditions, and the erosion predicted for a revised disruption scenario has been determined. Also, the effect of disruptions on the fatigue life of a steel wall has been evaluated and experimental observations on simulated disruptions have been reported.

The parametric disruption analysis considered disruption times of 0.1–20 ms and deposited energy densities of ~100–1000 $J \cdot cm^{-2}$ on stainless steel, graphite and tungsten. The extent of vaporization, melt layer thickness for the metals and effects of vapour shielding have been determined. On the basis of a tentative disruption scenario, in which the thermal quench is assumed to occur in ~0.1 ms, with most of the energy going to the tungsten divertor plate, the predicted lifetime erosion of the tungsten is ~17 mm and that of the steel wall is ~1.7 mm. For this case the melt layers are assumed not to erode in the short disruption times.

The effects of these disruptions on the fatigue life of a steel wall is a major concern that has not been completely resolved. Analyses conducted by the USA and Japan indicate that surface cracking of the wall will occur as a result of severe disruptions; however, propagation of the crack will not occur and hence the normal fatigue life will not be significantly degraded. Analyses conducted by the EC indicate that stresses created at the surface by the disruptions are sufficient to severely degrade the fatigue life of a steel wall, by approximately one order of magnitude. It is apparent that the limits of the ASME and RCC-MR design codes are exceeded by these conditions.

Experimental observations of simulated disruption effects with electron beams on steel indicate more melting and less vaporization than predicted by the analytical codes. The difference is attributed primarily to momentum transfer from the electron beam, which is not accounted for in the codes. It remains to be determined whether significant momentum transfer is expected in an actual plasma disruption.

2.2.7.3. First wall designs

The first wall design activity for INTOR was concentrated on the evaluation of critical issues associated with a bare steel wall and with a graphite protected wall. The following general conclusions have been made:

- For better reliability, it is preferable to use a non-reactor-relevant concept with low pressure/low temperature water as a coolant;
- Austenitic stainless steel is the only appropriate structural material for the low temperature first wall;
- Differences of opinion exists regarding the preference of cold worked or solution annealed stainless steels; these differences are based primarily on tradeoffs between designs, welding/joining constraints and different allowable design stresses;
- It should be noted that there remains a considerable uncertainty on the disruption scenario; this makes the development of optimized first wall/divertor systems very difficult;
- There are two possible design concepts for the first wall: a bare stainless steel first wall design concept and a protected first wall design concept;
- The choice of the extension of the protection of the first wall steel structure by low-z tiles is based on the different conclusions regarding the effects of disruptions on the fatigue life of a steel wall and on the different interpretations and/or applications of design codes to the allowable lifetime and/or stresses:
  - EC: Initial first wall extensively covered with graphite tiles, later reduction of protection;
  - Japan: Local protection by guard limiters which are designed for rapid replacement;
  - USA: No protection, unless required because of specified plasma operating conditions;
  - USSR: Initially a partial protection, which may be reduced later.
- Sub-limiters for startup and for protection against runaway electrons may be needed; however, no analyses have been performed.

Bare stainless steel first wall concepts. A bare stainless steel first wall is currently the reference first wall for the INTOR design [4]. The primary advantage of the bare steel wall includes design simplicity and a well established database. Key issues identified for further study include: (i) effectiveness of vapour shielding during disruptions, (ii) effects of disruptions on fatigue life, (iii) melt layer stability during disruptions, and (iv) advantages and disadvantages of cold worked versus solution annealed stainless steel, particularly as affected by welding/joining. For bare first wall designs with a fatigue lifetime of 2×10^5 cycles, the allowed peak nominal heat fluxes are assessed to be 0.2–0.4 $MW \cdot m^{-2}$, depending on: the initial minimum first wall thickness of 3–6 mm (especially for the lower value, stable disruption melt layers are required); the evaluation of disruption effects via different fracture mechanics methods; and the coolant channel geometry (coolant direction, double containment, design pressure).

Protected first wall concepts. Two types of first wall protection are proposed: (i) local protection by 'guard' or 'sub' limiters for rapid replacement and (ii) extensive coverage by protection tiles with radiation cooling.

With radiation cooled graphite tiles on austenitic steel first wall structures, high nominal peak heat fluxes of 0.5 $MW \cdot m^{-2}$ at 2×10^5 cycles can be achieved; these are limited by both steel fatigue and graphite temperatures/erosion, considering a thin steel wall thickness, re-radiation and residual tile heating.

On the other hand, there are also several critical issues which will limit the use of graphite tiles as first wall protection for next-generation tokamaks. These critical issues are: (a) irradiation damage is expected to lead to severe anisotropic swelling; a lifetime of less than 1 $MW \cdot a \cdot m^{-2}$ is predicted for nuclear grade graphite and even less for C/C composites; (b) there is general concern regarding the integrity of the tile attachment both for bonded and mechanical solutions; (c) the total erosion due to physical and chemical sputtering has been estimated for graphite (assuming about 1% oxygen) to rise to about 15 mm per year at 1800°C; (d) the form and location of redeposited material is also a key issue; graphite may be redeposited as amorphous carbon with different properties; and (e) the outgassing properties of graphite will present major vacuum problems during startup.

2.2.7.4. Divertor design

The emphasis of the divertor design activity has been on a detailed analysis of the collector plate reference design, a study of alternative solid state collectors, new plasma disruption scenarios, and development of innovations for liquid metal collector and self-pumping divertor applications.

The reference concept for the divertor plate is still the use of an actively cooled plate, consisting of tungsten tiles on the plasma side bonded to a cooled copper or copper-based alloy heat sink. By the use of buffer materials between the tungsten armour and the copper alloy heat sink, it is possible to obtain a structure where ASME criteria are satisfied and the allowable number of cycles exceeds 2×10^4 under a thermal load of $\sim 5\ \mathrm{MW \cdot m^{-2}}$. The key issue here is the choice of the optimal fabrication method of the tungsten–copper bonds.

The use of monolithic molybdenum plates and the application of carbon based materials for the armour are alternatives to the tungsten–copper divertor plates. In comparison with the leading concept, both of these options could be of advantage only for a limited range of conditions.

Analysis of new disruption regimes shows that significant divertor plate thermal erosion will take place. For tungsten, for example, evaporation after 2×10^4 cycles of operation constitutes 3.1 mm during Phase One and 0.6 mm during Phase Two.

Two innovations — helium burial in divertor plates and use of liquid metal divertor collector plates — have been developed. The first innovation permits a reduction of the required pump capacity, tritium processing and fuelling, and an increase of the helium removal efficiency and of the space for nuclear and tritium breeding modules. The second innovation leads to a solution of the problem of erosion and thermal fatigue, and provides divertor plate operation without excessive reduction of the total INTOR lifetime. Both suggested concepts are thought to have sufficient potential to justify further development.

2.2.7.5. *Tritium-breeding blanket*

A wide range of blanket concepts have been studied, with the primary objective to investigate the feasibility of providing tritium self-sufficiency without compromising the overall reliability of the machine and with modest R&D requirements. The evaluation led to the following main conclusions for the concepts considered:

Ceramic breeder with water or helium coolant

- Water cooling at low temperature is preferred, considering the reliability target and the achievable compactness of the design.
- The ceramic breeder (Li_2O, Li_4SiO_4, $LiAlO_2$) is arranged in the form of hot pressed plates or pebble beds surrounding the coolant tubes and is contained in modules with austenitic stainless steel as structural material.
- Beryllium or possibly lead are required in significant quantities (up to 80% of the blanket volume) as neutron multiplier for adequate tritium breeding.
- Tritium extraction is performed via a low pressure helium purge flow, possibly doped with hydrogen for reduced tritium inventory.
- The main critical issues are the integrity of the pebble bed, the consequences of coolant tube leaks and ruptures, compatibility problems with the multiplier and stress corrosion by water.

Lithium–lead eutectic breeder (self-cooled or water cooled)

- Water cooling is preferred, mainly considering the R&D requirements, which would be significantly higher for the self-cooled concept because of MHD effects;
- For water cooling at about 10 MPa, the coolant tubes are submerged in the liquid breeder, which is contained in long poloidal pressure tube modules;
- More detailed engineering studies led to the manufacture of a nearly full size blanket module;
- Beryllium may be needed in limited quantities (20% of the blanket volume) for achieving tritium self-sufficiency on the outboard side only;
- For tritium extraction, the liquid breeder is slowly circulated at a rate of up to ten times per day;
- The ongoing R&D effort led to reasonable confidence in the basic feasibility of the concept for most of the previously identified critical issues, such as corrosion of the structural material by the eutectic and the safety consequences of a coolant tube rupture. More development work is required on the tritium extraction systems and on aqueous stress corrosion of austenitic steel.

Aqueous lithium salt blanket

- First conceptual studies indicated the possibility of a simple blanket solution by using lithium salt (LiOH or $LiNO_3$) dissolved in low temperature coolant water; the initial shielding blanket could later be made the tritium breeding blanket;
- To achieve tritium self-sufficiency, at least 65% of the blanket volume would have to be beryllium;
- Tritium extraction from salt-free coolant water is a well established technology based on CANDU experience. However, it is estimated that the large processing capacity required for maintaining acceptable tritium levels in the coolant will cost up to fifty million dollars for the associated tritium

extraction system of INTOR. In addition, the technology has to be extended to salt-containing water;
- The major technical feasibility issue relates to stress corrosion of the austenitic steel structure, including irradiation effects.

It is concluded that there are several blanket concepts which are likely to achieve tritium self-sufficiency without major impact on the machine reliability and with a modest R&D effort.

2.2.7.6. Tritium system

No changes were made to the reference tritium system of INTOR [2, 3]. The only subsystem that may be subject to changes is the breeding blanket tritium recovery system, this being strongly dependent on the selection of the breeding material.

2.2.7.7. Radiation shield

The criteria and constraints and the design of the shield were reviewed. A wide range of radiation operating limits for TF coil materials was presented by all teams. Equivalent neutron fluences in the range from 1×10^{18} to 2×10^{19} cm^{-2} were considered. The main reason for the differences is the lack of new experimental data with simultaneous simulation of the reactor conditions. Also, there are different opinions as to the use of an epoxy insulator or a polyimide insulator. No changes in radiation limits were recommended.

The inboard radiation shield thickness is 70 cm; a magnet structure of about 10 cm of stainless steel serves as additional shielding. The shield composition corresponds to that suggested in Ref. [4], with the exception of the boron carbide layer. The radiation response in the inboard TF coils is as follows:

Fast neutron fluence	1.4×10^{18} $n \cdot cm^{-2}$
Copper disintegration	8.3×10^{-4} dpa
Nuclear heating rate	2.3×10^{-4} $W \cdot cm^{-3}$
Total nuclear heating of TF coils	8.6 kW

These values could be increased by a local peaking factor of three; they would still be average values of permissible radiation limits for magnetic materials.

No changes for the criteria and the composition of the outboard radiation shield were recommended.

2.2.7.8. Materials and nuclear technology R&D

Critical materials and nuclear technology R&D needs for INTOR were defined. Critical issues discussed in connection with austenitic steel structures include: susceptibility to aqueous stress corrosion, low temperature fracture toughness at appropriate He/dpa ratios and fluences, and effects of irradiation on the mechanical properties of welds/joints. The points discussed for graphite and C/C composites include the effects of radiation with high helium generation (He/dpa ~300) on the swelling and mechanical properties of graphite (C/C) and the properties of redeposited carbon. Critical data requirements for ceramic breeder materials include tritium transport and release characteristics at low (<500°C) temperatures, irradiation induced swelling, mechanical integrity of irradiated material under cyclic loading, and thermo-chemical stability and compatibility. Methods for tritium recovery from LiPb and aqueous/salt solutions as well as the corrosion characteristics of austenitic steels in LiPb and salt solutions must be evaluated in more detail. Data requirements for divertor materials include water corrosion of refractory metals and heat sink materials, and the evaluation of the bonding integrity of tungsten/copper duplex structures under irradiation. Extensive R&D is needed for the concepts of self-pumping helium burial and liquid metal divertors. Additional data for the effects of radiation on superconducting magnet materials are needed.

Technology needs for the first wall include: effects of disruptions on the fatigue properties of steel and melt layer stability, tritium retention in graphite, and effects of disruptions on the mechanical integrity of graphite and C/C composites. Blanket module testing under relevant conditions is required. The melt layer stability during disruptions and the effect of disruptions on the fatigue life of divertor plates must be determined. Liquid metal MHD effects must be evaluated for the new liquid metal divertor concepts. In situ testing of the self-pumping concept must be conducted to demonstrate the proof of principle.

2.2.8. Additional physics issues

In the area of additional physics issues, three topics have been discussed: fast alpha particle confinement, operational scenario and burn temperature control.

2.2.8.1. Fast alpha particle confinement

Alpha particles produced by D–T fusion reactions are expected to affect the behaviour of fusion plasmas in many ways. When the alpha particle production is strong enough to make alpha particle heating dominant, the plasma temperature and pressure profiles may change and may modify the transport and stability properties and thus influence the plasma confinement. Plasma–wall interactions will probably not be influenced in general by the change in plasma profiles, but fusion alpha particle losses to the walls may be an important issue. In 1985 and 1986, efforts have been made to quantify the fusion alpha particle losses. These losses are due to unconfined orbits, ripple trapping and ripple induced diffusion of banana orbits (stochastic diffusion).

Modelling of alpha particle losses confirmed the earlier finding that direct losses of alpha particles for a plasma current of 6 MA are at most a few per cent. The fraction of alpha particles lost through trapping in ripple wells is also small. The main mechanism governing alpha particle losses is ripple induced diffusion of banana orbits. There are, however, considerable quantitative differences in the results of the various analyses of this effect. In Ref. [7], it is predicted that the corresponding power loss is 2–3%; in Ref. [6], a value of 10% is found. In Ref. [8], the lower limit to the power losses is calculated to be 6–8%. The alpha particle flux to the wall is localized in the part between the TF coils and above (or below) the plasma midplane at small poloidal angles, but the predicted deposition profiles are different.

A comparison of the models and numerical procedures explains the difference to some extent. The profiles of the ripple and of the safety factor are important issues. The effects of elongation, finite beta, temperature and density profiles are also relevant.

More work is needed to refine the analysis, to improve the physics models used for determining the alpha particle losses and to validate the results by experiments.

2.2.8.2. Operational scenario

The operational scenario of INTOR [4], relying on purely inductive operation with limited burn pulse length (>100 s), was confirmed as the reference option. In addition, it should be ensured that INTOR is sufficiently flexible for the application of alternative operational scenarios, including non-inductive current initiation and ramp-up, transformer recharge and possibly steady state operation if some problems can be solved.

Alternative scenarios are attractive for future reactor operation: (a) Non-inductive current initiation allows the initial high loop voltage requirements to be reduced. (b) Non-inductive current ramp-up and transformer recharging, for which lower hybrid waves seem adequate, may permit an extension of the inductive burn pulse, depending on the plasma configuration and the poloidal field system. (c) For steady state operation, a non-inductive current drive system with a power of about 100 MW is anticipated. Steady state operation is a highly desirable operational mode for a reactor. Already with rather moderate power (≈ 50 MW), it would be possible to provide partial non-inductive current drive and current profile control during burn; this will probably be needed for satisfactory plasma performance and will permit an extension of the pulse length. If a substantial part of the current were driven by the bootstrap mechanism, considerable savings of the external power for current drive would result.

2.2.8.3. Burn temperature control

The reference assumption for INTOR operation during ignited burn is that a plasma temperature of about 10 keV is maintained. Because of the uncertainties in plasma transport behaviour under thermonuclear conditions, it is at present not clear whether this will be a natural feature of operating at a transport threshold, such as a soft beta limit or a strong confinement degradation with electron temperature. Therefore, it is still important to pursue studies of active burn temperature control methods. Possible principal schemes for this are: variation of the toroidal field ripple, controlled impurity injection, major radius compression/decompression, controlled fuel injection, and operation in a sub-ignited regime driven by additional heating [1–4].

During Part III of Phase Two A, studies have only been performed on one of the schemes for feedback control of the plasma temperature during ignited burn, namely controlled fuel injection using pellets [8]. In these studies, a $1\frac{1}{2}$-D model for describing plasma equilibrium and transport was used. It was shown that, in principle, this method can provide efficient burn temperature control.

2.2.9. Additional engineering issues

In the area of additional engineering issues, two topics have been discussed: compact reactor concepts, as innovations, and engineering scoping studies, performed in anticipation of a design upgrade of INTOR.

2.2.9.1. *Compact reactor concepts*

A special task of the INTOR Workshop was the study of innovations which would significantly improve the prospects of tokamak development and which would lead to an attractive, viable tokamak fusion reactor. Numerous ideas were submitted for consideration; several of these include compact reactor concepts.

Two suggestions for the use of copper coils were made. The first concept is the spherical torus, which is a confinement concept with very small aspect ratio, obtained by retaining only the indispensable components, such as the toroidal field coils inboard of the plasma torus. This concept is characterized by high toroidal beta (>0.2), natural large elongation (>2), large plasma current (>7 MA$\cdot$T$^{-1}\cdot$m^{-1}), strong paramagnetism and strong magnetic helical pitch. This concept has features which can be combined to produce a spherical torus plasma in a unique physics regime that permits compact fusion at low field and modest cost.

The second concept is the elongated tokamak, which calls for extreme shaping of the plasma by elongation (values of >4). Benefits associated with this concept include good confinement, high beta, and high plasma current density at moderate magnetic fields and stresses. The high current density suggests the capability of Ohmic ignition. Maintenance and repair are facilitated by rapidly demountable toroidal field coils. For each of these concepts it is necessary to demonstrate experimentally that the potential physics benefits can be realized; only when this has been proven is it possible to make conclusions regarding the feasibility of the concept.

For the use of superconducting magnet systems, two suggestions were made. The first concept is an all superconducting steady state tokamak based on a minimum major radius and strong plasma shaping. This concept relies on high magnet current densities, high-field plasma shaping coils, minimum neutron shielding and steady state operation assuming current drive. The resulting design can achieve high beta conditions in the first stability regime in a very compact device with modest cost. The concept is dependent upon the development of efficient current drive methods.

The second superconducting magnet concept is the microwave tokamak. This concept aims at an attractive high Q, steady state reactor in which the total plasma current is driven non-inductively by a combination of ECH, wall reflection of synchrotron emission and bootstrap current. Further development is required for the microwave sources needed for this concept.

2.2.9.2. *Engineering scoping studies*

A series of scoping studies was performed, with the purpose of evaluating a number of possibilities that could be considered for the eventual update of the INTOR design. These studies were directed at finding which of many possible changes would seem feasible and practical, and would have a significant impact on the overall design and performance. However, no design update of INTOR was made during Part III of Phase Two A, but the results of these studies remain valuable for considerations of future tokamak designs.

A set of scoping studies was recommended by all groups. This set includes the following elements:

— Reduction in size and/or number of TF coils;
— Single-null versus double-null divertor;
— Non-inductive current drive;
— Quasi-steady-state operation;
— Pumped limiter with ergodic edge;
— Combined use of NBI for heating, current drive and impurity flow reversal;
— Use of a pusher coil for higher beta;
— Higher plasma current;
— Integration of these individual elements.

These scoping studies were performed using various tokamak-system-codes and other similar analysis techniques. These codes typically make it possible to incorporate a large number of physics and engineering variables and constraints. Several high sensitivity areas with significant impact on the design were identified. The major items that could have a significant impact are the following:

— Use of non-inductive current drive for either steady-state or quasi-steady-state operation;
— Use of negative-ion neutral beam injection for the combined use of plasma heating, current drive and impurity flow reversal;
— Adoption of a higher plasma current;
— Use of a more elongated plasma shape;
— Use of ferromagnetic inserts to achieve a reduction in the size or number of TF coils and to maintain the desired plasma edge ripple.

Items examined in this study which do not appear sufficiently attractive or well enough established to warrant further consideration are the use of a pumped limiter with an egodic edge temperature, and the use of a pusher coil to achieve an indented plasma shape and thereby enhanced plasma performance.

Finally, the results of these studies indicate that both the single-null and the double-null divertor configuration have advantages as well as disadvantages and should continue to be explored. For higher values of elongation, double-null plasma configurations appear to be favourable.

The results of these scoping studies provide a quantitative basis for potential changes which can lead to an improvement of the next-generation tokamak design concept.

2.3. ANALYSIS OF INTOR-LIKE DESIGNS

2.3.1. Introductory remarks

In November 1986 — well within Part III of Phase Two A of the INTOR Workshop — the International Fusion Research Council (IFRC) recommended that the INTOR Workshop conduct, during 1987, critical analyses of the existing INTOR-like designs, with the aim of contributing to the basis for future design work for an Engineering Test Reactor (ETR). As a first step, members of the INTOR, FER (Japan), NET (EC), OTR (USSR) and TIBER (USA) design teams met at the IAEA Specialists' Meeting (23–27 March 1987) to document, in a common format, discuss and compare the programmatic and technical objectives, the engineering and physics design constraints (such as stress limits, beta limits), the main features 'driving' the design concept (choices made by the designers, such as incorporation of non-inductive current drive or a horizontal maintenance and assembly scheme), and the design specifications (major parameters, choice of materials, choice of heating method) of the five designs. The five designs are characterized by their gross parameters as listed in Table I.

The programmatic and technical objectives were discussed by the design team leaders present at the meeting. This discussion is given in Section 2.3.2. The design constraints, design 'drivers' and design specifications were compiled and compared by design team members. The results of their work are summarized in Sections 2.3.3 to 2.3.5.

On the basis of this work, the INTOR Workshop undertook two further analyses, (i) an evaluation of the operational flexibility of the various approaches, which

TABLE I. MAJOR PARAMETERS OF INTOR-LIKE NATIONAL DESIGNS

PARAMETER	INTOR[a]	NET	FER	TIBER	OTR
Major radius (m)	5.00	5.18	4.42	3.00	6.30
Minor radius (m)	1.2	1.35	1.25	0.83	1.50
Fusion power (MW)	585	650	406	314	500
Plasma current (MA)	8.0	10.8	8.8	10.0	8.0
Average beta (%)	4.9	5.6	5.3	6.0	3.2
Safety factor, q_I	1.8	2.1	1.8	2.2	2.1
Plasma heating method/power (MW)	ICRH/50	TBD[b]/50	ICRH/50 LH/20	LH/10 NBI/40	ICRH/50
Number of TF coils	12	16	12	16	12
Maximum field of TF coils (T)	11	11.4	12	12	11.7
Volt-seconds	112	181	50	58	210
Neutron wall load peak/average ($MW \cdot m^{-2}$)	1.6/1.3	1.5/1.0	1.5/1.0	1.6/1.0	1.05/0.8
Tritium inventory (kg)	3.1–4.6	2	2	TBD[b]	3.5–5.0
Test first wall area (m^2)	12	40	9	19	

[a] Phase Two A, Part II.

[b] Not yet selected.

allows them to cope with uncertainties still existing in the database, and (ii) a rather extensive systems analysis aimed at arriving at a unique description of the various designs and at extracting the most essential design drivers. This work is summarized in Sections 2.3.6 and 2.3.7.

2.3.2. Objectives

Each of the four national designs aims at providing an essential step forward towards a well balanced point between the present generation of experiments and DEMO. In this general aim, these designs are very similar to the reference design of INTOR. Even on a more detailed level, there is a great deal of similarity in the objectives: the achievement of reactor relevant plasma operating conditions, the incorporation of reactor relevant technologies in the machine components and the provision for engineering testing are broad, common objectives. All of these designs are based on a start of construction in about 1993.

The differences between the various groups regarding strategic considerations, budget availability, etc. led to differences in the quantification of these general objectives. The fluence objective varies from 0.3 $MW \cdot a \cdot m^{-2}$ for FER to 5.0 $MW \cdot a \cdot m^{-2}$ for OTR, with associated variations in materials and component testing capabilities and availability requirements. Ignition is an objective for FER, INTOR and NET, while steady state operation at $Q \geq 5$ is an objective for TIBER, and OTR has a high-Q objective. Tritium self-sufficiency is an objective for OTR, while no tritium will be bred in FER, except in test modules. OTR is the only design with the objective of demonstrating nuclear fuel production.

In addition, there is also a difference in the objective of the design studies themselves, as distinct from the objectives of the devices, which has caused differences in the designs. For example, the TIBER design activity had as an objective the study of the extent to which a compact design could be achieved by making aggressive assumptions about the development and incorporation of new technologies which are still to be developed.

The following subsections give descriptions of the objectives which have led to the different designs, together with the underlying philosophy, as presented by the design team leaders at the Specialists' Meeting in March 1987.

2.3.2.1. FER, Japan

(prepared by K. Tomabechi)

FER is designed to produce primarily an ignited, long burn plasma at minimum construction cost and technical risks; this is to be achieved by avoiding the use of technologies which are not directly relevant to the achievement of the primary objectives and which can be developed by other means. Therefore, the utilization of FER for technology development is limited to the extent possible in a device of reasonable cost and technical risk.

Regarding tritium breeding technology, the risks associated with the incorporation of a breeding blanket to obtain a tritium breeding ratio close to one are considered to be so high that they would significantly affect the operability of the machine, although a number of design concepts for tritium breeding have been proposed. Thus, FER is designed without a tritium breeding blanket, and external supply of the tritium needed for the machine is assumed.

On the other hand, a variety of tests, including tritium production tests or even high temperature heat extraction tests, can be performed by using test modules, at a relatively low neutron fluence of 0.3 $MW \cdot a \cdot m^{-2}$. Only a limited number of tests, such as structural material tests and long endurance tests of components, are excluded. To achieve a fluence of 0.3 $MW \cdot a \cdot m^{-2}$, the total quantity of tritium needed is estimated to be approximately 8 kg, which is within the range of tritium supply from external sources as expected for FER.

Another significant feature of the FER design is the operational scenario, which relies on non-inductive plasma current ramp-up, for which encouraging experimental data have been obtained in recent tokamak experiments. The incorporation of non-inductive current ramp-up into the operational scenario allows the designers to reduce the volt-seconds necessary for the long burn pulse (800 s) to 50 $V \cdot s$. The reduction of the volt-seconds has a positive impact on the FER design and results in compactness of the machine, less heat in the superconducting coils and less power supplies.

2.3.2.2. INTOR–IAEA collaboration

(prepared by the INTOR Steering Committee)

Besides the more general objective of INTOR to be the maximum reasonable step beyond the present generation of large tokamaks (TFTR, JET, JT-60, T-15), its programmatic objectives follow from a deeper consideration of its potential role in the world fusion programme. The target of all fusion programmes is the successful construction and operation of a demonstration fusion reactor (DEMO). A definition of the DEMO objectives and a discussion of the general prerequisites for the design and construction of demonstration reactors[2] led to the working hypothesis that a single step from the present generation of large tokamaks to DEMO, together with other parts of the fusion programme, may be sufficient to develop all the information necessary for the construction of demonstration reactors.

Thus, the primary physics objectives of INTOR are to investigate the operation of an ignited D–T plasma and to achieve long, controlled, reproducible burns with optimized plasma parameters. For the achievement of these objectives, satisfactory impurity control, power and particle balance control and profile control for parameter optimization are required. A closely related objective is the achievement of high duty cycle operation. INTOR may also be used for certain plasma physics experiments that are not directly related to the study of INTOR operation; however, it would be preferable to carry out such experiments in other plasma physics devices.

An extensive programme of technology and component development and testing is required to bring fusion power reactors to the demonstration stage. This programme will support INTOR in providing the basis for its design and construction, and it will support the INTOR programme in providing the basis for the design and construction of demonstration reactors.

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 which, at least partially, simulate a fusion reactor environment.

Within the general fusion programme, magnetic confinement concepts other than the tokamak are also 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 one concept will supplement the tokamak as the frontrunner before the DEMO stage. Thus, it is important that INTOR can be used also for testing of technologies that are needed for the other magnetic fusion concepts. Fortunately, the technologies required for the principal magnetic confinement concepts are, to a high degree, common to all systems.

The programmatic objectives of INTOR follow directly from the ongoing considerations of its role in the general fusion programme and from an assessment of the technical basis that may become available within the next years for its design. The development of the technical objectives has been such that they support the achievement of the programmatic objectives of INTOR; at the same time, they have to be consistent with the technical basis anticipated for the time of INTOR construction. The programmatic objectives will be achieved at different stages of INTOR operation.

An important question is to what extent materials and components testing should be included in the INTOR objectives. It was clear from the beginning that end-of-life testing of DEMO components or materials was far beyond what was possible with the technologies available for the basic INTOR machine. Even one-third of the intended DEMO fluence seemed to be beyond the scope of INTOR. This led to the strategy to study the radiation damage of reactor materials by simulation methods and to use INTOR for a calibration of the simulation methods in a 14 MeV neutron environment. Since it was also expected that, with the available time and budget, only stainless steel could be developed to all manufacturing details required for DEMO components, a fluence goal of 3 $MW\cdot a\cdot m^{-2}$ was considered adequate. At present, also ferritic steels are considered for use in DEMO. In this case the fluence needed for calibration would be around 6 $MW\cdot a\cdot m^{-2}$, which might be too large to aim at with INTOR. The minimum fluence goal related to the development and testing of blanket modules is about 0.8 $MW\cdot a\cdot m^{-2}$. In any case, the original goal of 3 $MW\cdot a\cdot m^{-2}$ is retained, to allow for some very valuable components testing and for extended blanket module testing.

Self-sufficiency in tritium production is not a programmatic objective of INTOR. Module and sector tests should allow reliable extrapolation to be made. A tritium breeding blanket was introduced in the design of the basic INTOR machine upon request of the IFRC, in order to reduce the demands on tritium supply and thus the operating cost. The request for full reactor relevance of this blanket was reconsidered, since it was necessary not to interfere with the reliable operation of the basic INTOR machine. This could be achieved with a tritium breeding rate of about 60%.

[2] The DEMO objectives and prerequisites have been checked by an INTOR consultants meeting in 1986 (see Ref. [10]).

2.3.2.3. NET, European Community
(prepared by R. Toschi)

It is the strategy of the European Fusion Programme to go directly from JET to NET; therefore the need for a significant extrapolation with respect to the operating conditions is anticipated. As a consequence, in NET the following provisions are made:

— Access to a wide range of plasma conditions, to be able to reach ignition whilst also providing engineering test capabilities. This calls for a staged approach with different requirements for the physics and technological stages. In particular, in the physics stage, operation with various plasma sizes, shapes and a current of up to 15 MA must be possible.
— Minimization of the complexity of the device during the early (physics) learning phase; consequently, no tritium breeding blanket, for example, will be introduced in NET during the physics phase.
— Possibility of introducing improved critical components (in particular the components facing the plasma) during operation; consequently, easy replacement of in-vessel components, for example, is an important guideline for the design.

Construction is to start as soon as the physics database is adequate. This is expected to be the case by 1993, and therefore the selected technologies and design solutions must be proven by this date. A thorough assessment of the feasibility of the apparatus and of whether the design solutions adopted satisfy the reliability requirements is scheduled for 1989.

The following design principles are therefore adopted: (a) The basic machine must be semi-permanent, highly reliable and compatible with a variety of operating scenarios, and must have margins of beyond-schedule operation. (b) The shielding blanket to be provided for the physics phase should have the potential of being developed to breed tritium in the technological phase. (c) Because of severe and at present poorly defined operating conditions, the components facing the plasma will be developed by going through iterations of the design during operation. These components must be designed for operating conditions corresponding to a neutron wall loading of about 1 $MW \cdot m^{-2}$. The first set has to have a lifetime corresponding to at least the duration of the physics phase (≈ 0.03 $MW \cdot a \cdot m^{-2}$). The possibility of rapid replacement is a particularly important issue.
(d) The breeding blanket is not required to provide nearly the full tritium supply (rather a breeding rate of 0.4). Blanket testing is to be done in modules and full sectors (at least two sectors simultaneously).
(e) Access to the machine must be compatible with the different heating methods under consideration.
(f) Inductive current drive must allow for a pulse duration of > 100 s; also non-inductive current drive must be possible. (g) The reliability requirements of the basic machine and shielding must allow for operation of up to 30 000 hours.

2.3.2.4. OTR, USSR (prepared by V. Pistunovich)

OTR is intended to provide experience in the development, construction and maintenance of a fusion reactor. For this purpose, OTR will be designed to achieve burning by intensive thermonuclear reactions, either with steady state operation or with a long pulse mode of operation. OTR will have to operate at a fluence level of 3–5 $MW \cdot a \cdot m^{-2}$ to provide data on the reliability of the units and components of a commercial fusion reactor. On the other hand, OTR is intended to serve as a test facility for materials testing, which again requires operation at a fluence of at least 3 $MW \cdot \cdot m^{-2}$. The achievement of these objectives requires high availability of the machine and operation with the first wall load at a level of 1 $MW \cdot m^{-2}$. Tritium self-supply is desirable both for economic reasons and with regard to fuel supply.

The construction of OTR is to be based both on nearly developed and on existing technologies. Its reliability during the operating phase has to be high enough to meet the above objectives. However, it is also necessary to have operational margins for an extrapolation from the existing data and for an evaluation of the impact of changes on the dimensions and the power level of OTR and its components.

Testing of two experimental sectors is planned for the third stage of OTR operation. This testing can provide experimental data for commercial fusion reactors. The two sectors can represent different options of full-scale reactor blankets and can be used to obtain a technological basis for commercial fusion reactors.

OTR should provide experience for environmentally acceptable fusion reactor operation. This objective can be achieved with a highly reliable blanket and tritium system and by using a low temperature and low pressure water coolant in the basic sectors of OTR.

OTR should also demonstrate the possibility of electricity and nuclear fuel production in fusion reactors. This objective can be achieved by including a system for electricity production in the reactor configuration and installing two experimental sectors with nuclear fuel elements.

2.3.2.5. TIBER, USA (prepared by G. Logan)

TIBER is intended to demonstrate the reliable operation of an integrated fusion device with reactor relevant technology. TIBER must be designed to test the reliability, safety and availability of integrated fusion reactor components, at reactor level particle and heat fluxes. For such tests to obtain results supportive for future practical reactors, reliability has to be a key part of the design. This is accomplished through specific approaches, including (a) reduction of cyclic stresses through maximum use of steady state operation, (b) use of non-inductive current drive for current profile control and disruption stabilization, and (c) high fluence operation at *nominal* plasma and coil parameters (beta, current, PF coil field and stress) that are sufficiently below the *maximum* design values, such that plasma disruptions are significantly reduced and the life of PF coils is significantly enhanced. The impact of (a) permits higher stress limits and a more compact magnet set. The impact of (b) improves the reliability of the TIBER tokamak operation; however, it increases the required R&D for the auxiliary systems external to the torus, which systems must provide a controlled current drive as well as heating. The impact of (c) improves the reliability through a reduced tendency for disruptions (lower beta/critical beta, higher q/q-minimum), but at the expense of a higher cost relative to a minimum machine designed for the maximum design limits.

TIBER should complete the physics database for long-pulse burn. To best complete the physics basis needed for practical and reliable future reactors capable of long and extended burns, TIBER is designed to operate in steady state with non-inductive current drive. The key physics motivation is to enable current profile control for times long compared to the plasma current diffusion time-scale. The result should be fewer disruptions, improved availability and higher fluence capability through steady state operation.

TIBER should demonstrate the performance of nuclear components at reactor relevant conditions. It must be designed to test key nuclear components. These components are the first wall and blankets, shielding (bulk and penetration shields), tritium processing systems, high-heat-flux components (lifetime and energy extraction) and associated assembly/maintenance equipment. The purpose of using blanket test modules is (a) to demonstrate that the fusion fuel cycle can be made self-sufficient for future devices, on the basis of the data obtained, and (b) to demonstrate that fusion energy can be extracted and used safely (but omitting the steam turbine generator). Operation of the whole TIBER facility should demonstrate that reactor equipment and personnel can operate with the required radiation protection. The major impact of the nuclear testing mission would be expected in facilities external to the torus itself, designed specifically to carry out the test programme.

TIBER should make use of selected innovative engineering design features to achieve a design with reduced cost. The TIBER design objective has a strong emphasis on reducing cost, through the incorporation of new technologies; these can be developed with an aggressive R&D programme, in time for TIBER (operation beginning about the year 2000). The key features for a compact, low cost TIBER torus are high current density, polyimide insulated superconductors, TF centring loads transmitted through reinforced central PF coils and a centre post, a common external torus vacuum boundary, and high density tungsten inboard shielding. The R&D programme for these compact torus features must lead to preliminary qualification by 1991 and final qualification for an assumed construction start in 1993. The R&D programme for the auxiliary system must qualify lower hybrid current drive for ramp-up and inductive assist by 1991 and for additional steady state current drive systems by the beginning of TIBER operation (2000).

2.3.3. Physics constraints

Physics constraints are posed by those parameters and limits used in a design that are derived primarily from physical laws of nature and where the designer has limited choice. The physics assumptions and constraints for each of the four national ETR designs and INTOR are globally quite similar (Table II). The differences in the designs are mainly due to the choice of and the emphasis on different features (see Section 2.3.5) and the use of different engineering constraints (see Section 2.3.4). On the whole, the national designs tended to adopt more conservative physics assumptions than the INTOR design, especially with regard to beta and q_{edge}.

All of the designs rely upon H-mode confinement and have incorporated an open poloidal divertor for this. In addition, they rely upon positive scaling of confinement and beta with the plasma current, the result being that the specified currents are in the 10 MA range. In fact, all of the designs use a Troyon

TABLE II. PHYSICS CONSTRAINTS

PARAMETER	INTOR	NET	FER	TIBER	OTR
I_p (MA)	8	10.8	8.74	10	8
K (at 95% of magnetic flux)	1.6	2.05/1.7	1.7	2.4	1.5
$\tau_{E,\ required}$ (s)	1.4	1.9	1.7	0.44	1.7
$\tau_{E,\ ASDEX\text{-}H}/\tau_{E,\ required}$	2.9	3.0	2.3	6.8	3
n ($10^{20}\cdot m^{-3}$)	1.6	1.7	1.14	1.06	1.7
Murakami[a] (10^{19} $T^{-1}\cdot m^{-2}$)	19	23	15	8	25
Beta required (%)	4.9	5.6	5.3	6	3.2
Troyon coefficient (%)	4	3.5	3.5	2.8	3.5
Impurity control, divertor	SN	DN/SN	SN	DN	SN
Pulse length (s)	150	350	800	55	600
Heating	ICRF	TBD	ICRF (LH for ramp-up)	LH + (LH for NBI ramp-up	ICRH

[a] Estimated using line average density.
SN: single-null divertor.
DN: double-null divertor.

type of scaling for the beta limit, although the choice of the Troyon coefficient is somewhat different in each design. The plasma is elongated so as to achieve a high current. The elongations vary from 1.5 to 2.4. All of the designs have an adequate margin for ASDEX H-scaling, but for none of the designs is ignition possible with most L-mode scalings.

All of the designs rely upon densities for ignited operation that are at the high end of the present tokamak database. When utilizing current drive with subignited operation, it is possible to make more conservative assumptions with respect to the density limit. The Murakami parameters range from 15 to 25 for ignited operation and are approximately 8 for Q = 5 operation.

The edge safety factor varies among the designs, but the cylindrical safety factors are very similar for all of them.

All of the designs rely upon operation with an open, high recycling divertor to provide power and particle exhaust. Advanced fuelling techniques, including high velocity pellets and other schemes, have been considered for all designs.

A toroidal field ripple in the range of 0.75–1.2% is anticipated to be adequate for fast alpha particle confinement, although major uncertainties remain.

The physics assumptions for current drive and heating are quite similar among the designs. For those that do not rely upon current drive, ICRH is generally the heating method adopted as far as a choice has been made. Where current ramp-up and transformer recharge is used, lower hybrid wave heating is the method of choice. Those designs which incorporate steady state current drive rely upon 400–500 keV neutral beams for central current drive and upon lower hybrid waves for current drive at the edge. The penetration of lower hybrid waves is felt to be inadequate for high temperature operation.

2.3.4. Engineering design constraints

In addition to natural design constraints, the choices made in systems or aspects of a design represent design limits, which have to be realized by other systems or aspects of the design. For example, the decision to use a double-null, highly elongated plasma poses limits on the mechanical configuration of the machine. The major engineering design constraints are given in Table III.

TABLE III. MAJOR ENGINEERING DESIGN CONSTRAINTS

ITEM	INTOR	NET	FER	TIBER	OTR
Field ripple at the edge (%)	1.2	1.5	0.75	0.8	1.0
Impurity control	SN	DN/SN	SN	DN	SN
Plasma elongation (at 95% of magnetic flux)	1.6	2.05/1.7	1.7	2.4	1.5
Maximum radiation to TF coil insulator (rad)	10^9	5×10^8	3×10^9	10^{11}	10^9
Allowable TF coil stress	ASME	600 MPa	600 MPa	600 MPa	600 MPa
Maximum first wall heat flux ($MW \cdot m^{-2}$)	0.4	0.4	0.4	0.3	0.4
Allowable first wall stress	ASME	RCC-MR	ASME	ASME	200 MPa
Directed peak heat flux ($MW \cdot m^{-2}$)	5	5	2	3	5

SN: single-null divertor.
DN: double-null divertor.

In the engineering category, the design constraints were arranged in three groups: mechanical and configuration; electromagnetics, heating and current drive technology; and nuclear.

2.3.4.1. Mechanical and configuration constraints

In the mechanical and configuration group, the major engineering design constraints and the range of possibilities for the five designs are as follows:

Magnet configuration: Placement of all coils in a common cryostat or placement of magnets in a self-contained cryostat.

Method of reacting magnet loads: Use of a bucking cylinder, wedging of the inner legs of the TF coils, or reacting the TF coils directly from the central solenoid.

Vacuum boundary: Use of a common boundary for the plasma and magnets or use of a separate vacuum containment for each system.

Number of replaceable modules: The number varies from 12 to 48, depending upon the overall device configuration.

Component replaceability: Most designs assume many of the components to be designed to last for the lifetime of the device, with no plan for replaceability; other designs make no such assumption.

Tritium breeding: This varies from no tritium breeding (other than in test modules) to full tritium self-sufficiency.

Maintenance approach: Two major approaches are considered: horizontal removal and vertical removal of torus components.

Plasma configurations: The plasma configurations vary from modestly elongated (1.5) single-null divertor plasmas to highly elongated (2.4) double-null divertor plasmas.

Radial dimensions: The five designs vary dramatically in the overall plasma major radius, and naturally also in the thickness of the components and the space allocations for the major radius. For example, the region for plasma scrape-off varies from 9 to 30 cm, the total inboard blanket/shield thickness varies from 48 to 105 cm, the total width of assembly gaps and spaces varies from 2 to 20 cm, and the thickness of the TF coil inner leg varies from 49 to 110 cm.

2.3.4.2. Electromagnetics, heating and current drive technology

In the area of electromagnetics, heating and current drive technology, the major differences are in the electromagnetics. All of the designs use similar heating and current drive technologies.

For the TF coil system, there are a number of different engineering design constraints, which are primarily related to the environment as perceived necessary for the desired performance of the superconductor. These constraints include the total and peak nuclear heating levels. The total nuclear heating level varies from about 8 to 72 kW of nuclear heat deposition; the related peak nuclear heating levels vary from about 0.3 to 5 $kW \cdot m^{-3}$. Radiation protection requirements for the superconductor and the associated insulator also vary significantly; the radiation dose varies from about 2×10^8 rad to 10^{10} rad. Other significant variables are the conductor current values (16–35 kA), the average winding pack current density ($\sim$10–22 $MA \cdot m^{-2}$), the magnetic energy (4–45 GJ) and the maximum quench voltage to ground ($\sim$7–20 kV).

For the PF coil system, the dominant differences are related to the total volt-seconds to be provided ($\sim$50–210 V·s). In addition, there are differences in the allowable maximum rate of change of the magnetic field ($\sim$0.5–3 $T \cdot s^{-1}$), differences in the OH current ramp time ($\sim$13–30 s), differences in the breakdown voltage being used (10–35 V) and differences in the total magnetic stored energy ($\sim$4–11 GJ).

2.3.4.3. Nuclear systems

In the area of nuclear systems, there are significant differences in the engineering design constraints for a number of aspects of the first wall and blanket, the divertor and the shielding.

For the first wall and blanket systems, the differences are related to the target lifetime fluence values (which range from 0.3 to 3 $MW \cdot a \cdot m^{-2}$); the allowable stresses in the structural material, which are also connected with the number of lifetime cycles of operation; the tritium breeding requirement; the first wall protection assumptions; and the assumptions related to the disruption scenario.

For the divertor, the differences are related to the incorporation of different concepts for the physics and technology phases and to the differences in the disruption scenario.

For shielding, the differences are related largely to the need to protect the magnets; therefore, these differences are connected with the allowable fluence to the superconductor, the allowable dose to the insulators and the nuclear heating limits. Another constraint relates to the desirability of minimizing the overall thickness of the inboard shield region to minimize the size and cost of the design.

2.3.5. Design driving features

The five INTOR-like designs differ in a number of significant features which tend to 'drive' the characteristics of each design. Each of these design driving features represents an aspect of the design where the designer has a choice from among a number of options. These choices are made in accordance with the overall mission and the supporting programmatic and technical objectives established for each design.

The selection of each of the design driving features by the design teams is also influenced by a number of important considerations related to each national programme. These include the perceived timing for the necessary development and construction of each device, the perceived understanding of the scientific and technological database as it stands at present and as it might develop over the time period until the start of construction, and finally the maturity of the technology required to support each design and its stated mission.

Table IV presents a comparison of the major design driving features for the five INTOR-like designs. Many of these features are related to the scientific aims and the present knowledge. These include the need to achieve ignition or not, the nature of the operating scenario (inductive or non-inductive current drive, or some hybrid combination), the pulse length, the degree of plasma shaping (elongation), the type of impurity control (single- or double-null divertor), the nature of startup and the plasma heating method. The remaining major driving features result from operational and technological considerations, such as whether to breed tritium or not (and how much), the target fluence and the nature of the desired nuclear testing, and finally the approach to maintenance of the internal torus components (horizontal or vertical access).

2.3.6. Operational flexibility

Descriptions of NET, TIBER, FER and OTR presented at the Specialists' Meeting on Engineering Reactor National Design Concepts have been supplemented by information on the flexibility of each design. This information, together with that for INTOR, has been reviewed and the capabilities of these concepts to operate under various combinations of alternative plasma conditions, alternative plasma geometries and alternative operational scenarios have been identified.

In view of the present uncertainties in both the physics and engineering databases, the potential for flexibility of a device can be compared under three main headings: (i) the ability of a device to minimize the risk connected with attaining the mission goal in view of present uncertainties; (ii) the ability of a

TABLE IV. MAJOR DESIGN DRIVING FEATURES

FEATURE	INTOR	NET	FER	TIBER	OTR
Operating mode	Ignited	Ignited	Q > 20–30	Q > 5	Q > 5
Pulse length (s)	150	>200	800	d.c.	600
Current drive	inductive	inductive	hybrid	non-inductive	inductive
Fluence ($MW\cdot a\cdot m^{-2}$)	3.0	0.8	0.3	3.0	5.0
Tritium breeding rate	0.6	0.3	0.0	1.0	1.05
Plasma heating method	ICRH	TBD[a]	TBD[a]	NBI + LH	ICRH
Impurity control	SN	DN/SN	SN	DN	SN
Access for maintenance	horizontal	vertical	horizontal	horizontal	horizontal
Weight of largest replaceable component (t)	300	60	250	32	300
Availability/period	25%/10 years	8%/11 years (25%/1 year)	7%/6 years	30%/12 years	50%/9 years

[a] Not yet selected.

device to expand the mission goal by enhancement of such parameters as plasma current, plasma pressure, burn time, neutron wall loading and end-of-life fluence; and (iii) the ability of a device to enable a wider choice of operational scenarios, particularly continuous current drive.

Consideration of the attributes of the various designs shows that all of them have a substantial potential for flexibility of operation.

2.3.7. Systems analysis

A systems analysis of the five INTOR-like designs was performed to evaluate and determine the impact of changes in a given design. Such a quantitative analysis can provide valuable insights regarding how different choices affect a given design. The results of this analytic comparative study of the five INTOR-like designs should allow the fusion community to take such design impacts into account in the development of the next generation of tokamaks.

Systems analysis methodology has progressed significantly during the last few years. The capability to represent a tokamak point design and the complexity of the main subsystems has been developed. Several of the computer systems codes that have been developed now incorporate numerical optimization methods for a given figure of merit which enable simultaneous changes of many selected variables subject to specified constraints. Many of the codes involve iterative routines to obtain a unique solution for the specified input. Nomograph routines have been evolved to permit rapid parametric evaluation of design options for given assumptions. Finally, simplified systems of equations have been developed on the basis of present understanding of tokamak physics and engineering dependences. These systems codes make it possible to quickly determine the impact of design variations. Good computational tools have thus been developed with which next-generation tokamak designs can be studied in reasonable detail.

A measure of the validity and usefulness of a given systems analysis method is its ability to reproduce the major features and the performance characteristics of a variety of designs. One test of this ability is to use the systems analysis method to obtain as accurate a reproduction as possible of the mechanical features, the performance, the physics and the engineering parameters of an existing design. The various national systems methodologies were applied to the task of replicating each of the five INTOR-like designs.

The results of these calculations for the four national designs and for INTOR indicate the ability of the systems analysis methods to accurately represent the general characteristics as well as many specific features and parameters of the designs. This demonstration gives confidence in the ability of parametric studies to realistically indicate the impact of making changes in a design.

Sensitivity calculations were performed by all participants to determine what changes are introduced in the respective design by making selected changes in the input. In some calculations, only one aspect of the design was changed (a mechanical feature or dimension of a component, a physics assumption or parameter, or an engineering assumption or parameter). The impact on the design resulting from this single change was then determined.

From the results of the individual sensitivity calculations those items which have a strong impact on the overall design can be determined, as well as items which have considerably less impact. By a systematic assessment of item changes and their impact on the design, the items with the strongest influence can be identified. This information can then be used in the detailed design process as a guidance for the designers.

From the results of these evaluations the following was found:

— *Items with the strongest sensitivity to changes:*
  ignition margin or Q*
  safety factor (q)
  elongation
  shield attentuation*
  Z_{eff} and reactivity*
  neutron wall load
  beta scaling coefficient (Troyon factor)
  toroidal field coil stress.
— *Items with the weakest sensitivity to changes:*
  fluence
  burn time
  presence/absence of bucking cylinder*
  presence/absence of inner blanket*
  shield thickness
  scrape-off layer (inboard)
  edge ripple*
  plasma inductance*
  plasma profiles
  volt seconds
  radiation dose to insulator*.

This was the general ordering observed, but the various sensitivities found in the four studies indicate appreciable scatter in the results.

The reasons for this are (i) differences in the models (level of detail of the power balance, confinement time scaling law); (ii) different figures of merit (cost, major radius); (iii) different design points around which the sensitivity was calculated; and (iv) different constraints (ignition margin, wall load, fusion power). The latter point may be the most important one, since each team has evolved different sets of design constraints, including items that were judged to be very important. Each design team has also derived certain outputs which other design teams assume to be fixed inputs (e.g. the three constraints mentioned above).

In addition, the national designs have evolved to different levels, but in all instances the parameter space for the design already has many constraints. These constraints narrow considerably the space available for valid designs when one parameter is altered. Even for the same device with constraints on different items, the parameter space over which the sensitivity analysis is conducted will be different, thereby contributing to the different results.

Nevertheless, given these differences, it is greatly encouraging that there is general agreement on the parameters with strong sensitivity and on the parameters with weak sensitivity.

There is an important distinction between sensitivity and overall design impact. To draw practical conclusions from the sensitivity studies, it is necessary to consider the probable range of uncertainty of a parameter when determining the importance to the design of changes in that parameter. Parameters with strong sensitivity may have a strong impact on the design even if the range of variation is small. However, parameters with weak sensitivity may have a strong impact on the design if the range of variation is large. This practical consideration should be made in the design process. This issue was given preliminary consideration by each delegation.

Calculations in which a group of items was changed were also performed; for example, it was of interest to determine the impact of substituting at one time all physics related assumptions made in one design into another design. The impact of changing the engineering assumptions or the general features of a given design is also of interest.

The differences in the individual assumptions of the INTOR-like devices can be grouped into categories, such as physics, engineering or features. The physics category typically includes terms such as beta and beta coefficient, safety factor, plasma temperature and density, edge ripple and plasma profile factors. The engineering category typically includes the dimensions of components (OH and TF coils, buckling cylinder, shield), stress levels, radiation levels and gaps. The

* Studied by only one delegation.

features category typically includes plasma configuration (elongation and triangularity), maintenance approach, fluence level, tritium breeding, single-null or double-null divertor and operating scenario.

Each participant performed a limited number of analyses of these global effects. These studies were performed in a manner similar to the individual sensitivity studies described above. For example, calculational transitions were made from INTOR to FER (Japan), INTOR to TIBER (USA), INTOR to OTR (USSR), NET TO TIBER (USA) and NET to FER (EC and Japan).

The results of these studies indicate that the delegations were successful in demonstrating the ability to transfer global groups of changes (all physics, engineering, features) from one design to another. It has been shown that the systems analysis codes can be used to differentiate between the national designs by making global effect substitutions.

These substitutions provide a valuable insight into the differences between the various INTOR-like designs. However, the impact on the design depends strongly on which items are included in the categories physics, engineering and features. Because of differences in the specification of these global effect categories, general conclusions about which category has the greatest effect are difficult to draw. The results depend both on the definition of the categories and on the design points that are compared. The national studies indicate that significant effects have to be expected for each category.

Cost and major radius are valuable figures of merit for measuring sensitivity, although they occasionally lead to disparate results. For a given design, the impact of changes must be interpreted with caution since these changes may imply impacts on less measurable design aspects such as the risk associated with the design, new technology development or new physics, maturity of technology and time for construction.

At present, there is great interest in how the estimated capital cost is affected by various aspects of tokamak design. It is recognized that for the national designs national procedures are used to account for engineering, fabrication, transportation, installation and project management costs. These approaches are all different, not only in units of currency but also in the treatment of the various elements in the costing process. Recognizing these differences, a cost comparison was performed for each of the five INTOR-like designs to determine the relative ranking of capital costs as predicted by each national group.

The relative capital costs were normalized to the INTOR cost estimate. The results indicated that the costs for INTOR and NET are approximately the same, the cost of FER is about 10% lower than that of INTOR, the cost of TIBER is about 35% lower than that of INTOR, and the cost of OTR is about 25% higher than that of INTOR. In general, the spread of the calculated relative costs for a given design was 5–15%.

These relative cost comparisons are able to reflect the incremental differences between the various designs. This ability derives from common assumptions for the design of the various components and major systems. However, these cost comparisons must be interpreted with care, since the various designs are based on different assumptions regarding the timing of construction, the amount of the necessary supporting development and research, and the aggressive or conservative approach regarding the maturity of the technology in the design. Factoring these considerations into the design can alter the cost comparisons dramatically.

Overall, the comparative systems analysis has demonstrated that valuable insights can be derived. An important aspect is that such studies can be performed rapidly and with little cost, and that the results can provide guidance in the evolution of the designs. Especially in the early stages of design, such an analysis is of great interest, since it enables a rapid evaluation of a number of options. The database generated can give quantitative support for initial choices among different options. This allows the design teams to engage in more detailed design work on the basis of a reasonably established initial baseline design.

Valuable insights have been obtained by this international collaborative system analysis. Therefore, it would be desirable to continue these studies in order to resolve differences in the definition of terms, to better understand the detailed modelling of specific systems, and to apply these models to a common design.

2.4. IMPLICATIONS FOR THE INTOR DESIGN CONCEPT

The principal conclusions from the work of Phase Two A, Part III, are summarized in the preceding sections. The implications of these conclusions relative to the INTOR design concept are discussed in Chapter XII of the report on Phase Two A, Part III. A summary is given in the following subsections.

2.4.1. Impurity control

Modelling studies and experimental data still support the choice of the poloidal divertor for impurity control and the choice of a high-z (tungsten) divertor collector plate surface. Thus, the major aspects of the recommended impurity control system are the same as in the reference INTOR design concept. A number of modifications to the INTOR design concept may be necessary, however. A low-z limiter for startup may be required. If the present uncertainty regarding the severity of disruptions remains, then it may be prudent to install a protective armour on the first wall, at least during the physics phase. The value of Z_{eff} may have to be increased from 1.5 to 2.0, in which case allowance would have to be made for a corresponding increase in the power radiated to the first wall.

2.4.2. Operational limits and confinement

A variety of H-mode energy confinement scaling laws have been proposed over the last few years. On the basis of these laws, the INTOR design concept is considered to have adequate confinement capability to achieve ignition, if there is no substantial degradation with heating power.

The INTOR design concept somewhat exceeds both the Murakami–Hugill limit and the Greenwald density limit, but it should be noted that these limits are exceeded by as much as a factor of two in experiments with intense auxiliary heating. Thus, the density in INTOR is very probably below the actual density limit.

Analytical and experimental results indicate that the Troyon beta limit g-factor must be reduced from the value of 4 used in the INTOR design concept to 3.0–3.5 and that the safety factor q_I must be increased from 1.8 to at least 2. Since

$$\beta\ (\%) = \frac{g I_p\ (\mathrm{MA})}{a\ (\mathrm{m})\ B\ (\mathrm{T})}$$

$$q_I = \frac{5\left[\frac{1}{2}\left(1+\left(\frac{b}{a}\right)^2\right)\right]^{1/2} B\ (\mathrm{T})\ a^2\ (\mathrm{m})}{R\ (\mathrm{m})\ I_p\ (\mathrm{MA})}$$

a combination of increasing the plasma current, the magnetic field and the plasma elongation (b/a) and/or reducing the major radius in the INTOR design concept is probably necessary to achieve the performance objective (e.g. neutron wall load).

2.4.3. Current drive and heating

There is now a substantial experimental and theoretical database on non-inductive current drive, for example by lower hybrid waves or neutral beams, or a combination of both, so that it can be considered as an option to achieve the basic performance objectives of INTOR. However, the predicted efficiency is low, and the required power may be of the order of 100 MW if the plasma parameters are optimized for current drive. Thus, while inductive current drive is retained as the reference option in the INTOR design concept, it is suggested to use non-inductive current drive in a new INTOR-like design concept, provided that such a design can be shown to be feasible and to have substantial advantage over an inductively driven design.

New experimental data support the previous choice of ICRH as the reference heating scheme in INTOR. However, if neutral beams and lower hybrid waves were chosen for current drive in a new INTOR-like design concept, it would be appropriate to use them also for heating (and, in the case of neutral beams, possibly for added impurity control by flow reversal).

2.4.4. Electromagnetics

It has been established that the active control coils should be located inside the toroidal field coils and outside the shield. Also, it has been confirmed that the first wall/blanket structure is adequate for passive stabilization.

Modelling studies indicate that the INTOR poloidal field coil system could be designed more optimally. In particular, the coils should be placed closer to the midplane. Leaving a large midplane window for horizontal access imposes a moderate penalty in terms of stored energy for little elongated to moderately elongated plasmas, but a large penalty for highly elongated plasmas.

2.4.5. Configuration and maintenance

The reference INTOR maintenance concept is horizontal removal of large torus segments, which requires that a rather large 'window' for access be left at the midplane, with the consequence that no poloidal field coils can be located near the midplane. Analysis of this maintenance scheme and comparison with a vertical or oblique removal concept led to the conclusion that the simpler maintenance procedures associated with horizontal maintenance outweigh the penalty in poloidal field coil optimization for small to moderate plasma elongation, but that the vertical or oblique maintenance scheme is preferable for moderate to large plasma elongation, for which the penalty in poloidal field coil optimization becomes too large. Thus, if the plasma elongation has to be increased to more than two, as may be necessary to satisfy the plasma operating limits (see Section 2.4.2), then a change from the horizontal maintenance concept to the vertical or oblique concept may be required. Also a combination of the two concepts might have its merits.

For the reference INTOR maintenance concept, it is recommended to use a transfer cask for containing tritium and dust, in order to meet the requirement of personnel access to the reactor hall. Because of recent developments, an in situ maintenance scheme is recommended for components facing the plasma (e.g. protective tiles on the first wall).

The use of iron inserts to reduce the field ripple would enable a reduction of about 50 cm in the toroidal field coil bore or a reduction in the number of toroidal field coils from twelve to ten, without significantly complicating the configuration. Thus, the use of iron inserts is recommended.

2.4.6. First wall and blanket

Analyses of the divertor collector plate, the first wall and the breeding blanket confirm the choices that were made in the INTOR design concept. The reference divertor plate concept of tungsten tiles bonded to a water cooled copper heat sink is predicted to have a lifetime of 2×10^4 cycles, limited during normal burn by fatigue and erosion. This implies that the divertor plate must be replaced ten times during the lifetime of INTOR. It is still recommended to use a bare, water cooled austenitic stainless steel first wall, unless new information indicates that the frequency of disruptions would be much greater than is assumed in the present disruption scenario.

The reference breeding blanket concept, with ceramic breeding material, an austenitic stainless steel structure and water cooling, is still recommended. It is possible to use water at relatively low pressure, which is recommended for better reliability.

In the studies it was found that a beryllium multiplier together with certain stratagems can be used to achieve a tritium breeding ratio greater than unity and hence to make INTOR self-sufficient in tritium production without increasing the inboard dimension or the level of risk. Accordingly, it is recommended to equip INTOR with a (non-reactor-relevant) tritium producing blanket adequate to provide tritium self-sufficiency.

2.4.7. Design sensitivity

Systems analyses indicate that the size and cost of an INTOR-like design is very sensitive to the ignition margin, Z_{eff}, the plasma elongation, the safety factor, the value of the g-factor in the Troyon beta limit, the neutron wall load, the shield attenuation and the allowable stress in the toroidal field coils. Thus, the size and cost of INTOR could be reduced by future developments that would lead to improved energy confinement, improved impurity control, stability at larger plasma elongation, a lower safety factor, larger values of the Troyon g-factor, a higher limit of radiation damage on the magnet insulators, and magnet structural materials operating at higher stress levels.

3. CONCLUSIONS

The cumulative INTOR work to date has been a major factor in laying the groundwork for proceeding to the design of the next major experiment in the world tokamak programme. Its objectives and general characteristics have been identified. A preliminary conceptual design has been developed early in the INTOR process and used to identify critical technical issues and R&D requirements. The problems in connection with the critical technical issues have been partly resolved and partly a better understanding of the problems was reached. The modelling methods used in reactor design by the four groups have been further developed and compared to test their consistency. The national designs of the four groups and the physical and technical constraints upon which they are based have been evaluated. Finally, ways in which the INTOR design concept should be updated on the basis of this work have been identified.

REFERENCES

[1] INTOR GROUP, International Tokamak Reactor, Zero Phase (Rep. Int. Workshop Vienna, 1979), IAEA, Vienna (1980); see also Summary in Nucl. Fusion **20** (1980) 349.

[2] INTOR GROUP, International Tokamak Reactor, Phase One (Rep. Int. Workshop Vienna, 1980 and 1981), IAEA, Vienna (1982); see also Summary in Nucl. Fusion **22** (1982) 135.

[3] INTOR GROUP, International Tokamak Reactor, Phase Two A, Part I (Rep. Int. Workshop Vienna, 1981–1983), IAEA, Vienna (1983); see also Summary in Nucl. Fusion **23** (1983) 1513.

[4] INTOR GROUP, International Tokamak Reactor, Phase Two A, Part II (Rep. Int. Workshop Vienna, 1984 and 1985), IAEA, Vienna (1986); see also Summary in Nucl. Fusion **25** (1985) 1791.

[5] European Community Contributions to the INTOR Phase Two A, Part III Workshop, Rep. EUR-FUBRU/XII-139/88/EDV1, Commission of the European Communities, Brussels (1988).

[6] Japanese Contributions to the INTOR Phase Two A, Part III Workshop, Rep. Japan Atomic Energy Research Institute, Tokai-mura (1988).

[7] USA Contributions to the INTOR Phase Two A, Part III Workshop, Rep. INTOR/88-1, Georgia Institute of Technology, Atlanta, GA (1988).

[8] USSR Contributions to the INTOR Phase Two A, Part III Workshop, Rep. Kurchatov Institute, Moscow (1988).

[9] Tokamak Concept Innovations (Rep. Specialists' Mtg Vienna, 1986), IAEA-TECDOC-373, IAEA, Vienna (1986).

[10] Fusion Reactor Critical Issues, Reports of three Specialists' Meetings on Impurity Control Modelling (Vienna, Sep. 1985), Non-inductive Current Drive (Garching, Sep. 1986) and Confinement in Tokamaks with Intense Heating (Kyoto, Nov. 1986) and Report of a Consultants' Meeting on DEMO Requirements (Yalta, June 1986), IAEA-TECDOC-441, IAEA, Vienna (1987).

[11] INTOR Related IAEA Specialists' Meeting on Plasma Disruptions, Vienna, 27–28 July 1987.