THE FED-INTOR ACTIVITY

FUSION REACTORS

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A set of critical technical issues affecting near-term tokamak reactors has been addressed by a broad cross section of the U.S. fusion community. The results of this work have guided the evolution of an improved concept for a tokamak experimental reactor. An overview of a two-volume report is presented.

I. INTRODUCTION

The fusion engineering device (FED)-International Tokamak Reactor (INTOR) activity has involved a broad cross section of the U.S. fusion community during 1981-1982 in addressing critical technical issues that affect the feasibility of the FED and INTOR concepts for a first-generation tokamak experimental reactor, which could lead to significant improvements in engineering design or reductions in cost. This activity evolved in mid-1981 from the coalescence of two related activities—the FED design activity and the U.S. INTOR activity.

The design of the FED (Ref. 1) was carried out during 1980-1981 at the Fusion Engineering Design Center under the guidance of the FED Technical Management Board, consisting of senior people in the fusion community. The FED was designed to begin operation about 1990. It was intended to focus development of the systems, nuclear and plasma-production technologies inherent in fusion reactors, and to serve as a demonstration of the integration of those technologies into a workable system.

The INTOR Workshop is a collaborative effort among Euratom, Japan, the United States, and the Soviet Union. It is conducted under the auspices of the International Atomic Energy Agency (IAEA), in terms of reference defined by the International Fusion Research Council, an advisory body to the Director General of the IAEA. An INTOR design[2] was developed during 1980-1981. INTOR also was designed to begin operation about 1990 and had objectives similar to FED plus an ambitious engineering testing mission.

The work that was performed in the FED-INTOR activity during 1981-1982 is summarized in Secs. II through IX and described elsewhere[3] in detail. The reader is referred to the full report for detailed analysis and references to primary sources. The conclusions derived from this work provided the basis for evolving the FED and INTOR designs toward an improved design concept for a next-generation tokamak experimental reactor (i.e., engineering test reactor (ETR) in the United States and INTOR internationally). The objectives of FED-INTOR and this improved concept are described in Secs. X and XI. Conclusions and recommendations are given in the final two sections.

II. PLASMA PERFORMANCE

The primary emphasis of the plasma performance work has been in two areas:

1. review of the physics basis for the design of first-generation tokamak reactors
2. assessment of the prospects for using radiofrequency (rf) for bulk heating.

II.A. Physics Basis

II.A.1. Confinement and Beta

Experiments have moved much closer to ascertaining the validity of the design basis for plasma beta. The highest value obtained to date is $\langle\beta\rangle = 4.6\%$ for plasmas of elongation ~1.4 to 1 in Doublet-III (D-III); $\langle\beta\rangle = 3.2\%$ has also been achieved in near-circular Poloidal Divertor Experiment (PDX) plasmas. These values of beta are near the stability thresholds predicted on the basis of ideal magnetohydrodynamics (MHD). The highest beta values in D-III were obtained by injecting ~3.4 MW of neutral beam power into a low-q, 6-kG plasma. The increase in beam power now being implemented should permit more definitive tests of the theoretical beta limits.

The INTOR scaling law, $\tau_E \propto na^2$, can no longer be regarded as a reliable basis for making reactor design decisions. The linear scaling with density that characterizes the ohmic-heated (OH) data set is not observed in auxiliary-heated discharges. However, confinement is found to improve strongly with current in all beam-heated data sets, and, moreover, there is evidence of additional positive correlation with plasma elongation. Although τ is generally lower in beam-heated cases than in the corresponding OH cases in present day experiments, the fact that the extrapolation to the reactor regime involves a larger increment in plasma current than in density suggests that the new scaling for auxiliary-heated plasmas may not turn out to be a degradation at all. On the contrary, it may actually increase the margin for ignition. The key lies in size scaling, investigation of which then becomes elevated to very high priority in near-term experiments.

Assuming that $\tau_E \propto I_p$, two criteria are sufficient for FED/INTOR to reach ignition:

1. The scaling of τ_E with a (or with R) is at least linear.
2. $\langle\beta\rangle \geqslant 4\%$ is achievable.

A stronger dependence of τ_E on I_p [as noted in Impurity Study Experiment-B (ISX-B)] or a positive correlation of τ_E with κ (as noted in D-III) would result in even more favorable prospects for reaching ignition. The observation of $\langle\beta\rangle > 4\%$ in D-III offers great encouragement with respect to the second criterion. However, drawing conclusions about the $\langle\beta\rangle$ achievable in FED/INTOR requires some extrapolation in both geometry and operating point, so considerable further study is mandated.

Despite the fact that the emerging transport scaling laws for auxiliary-heated plasmas are quite different than those upon which the next generation of large tokamaks and FED/INTOR were designed, the predicted performance parameters still seem to be justified.

Conclusion: The prospects for achieving the ignition and burn conditions specified for FED/INTOR are enhanced by the recent experimental results, and no change in the parameters related to $\langle\beta\rangle$ or energy confinement is required, although a lower q operating point should be considered.

II.A.2. Noninductive Current Drive

Lower hybrid experiments on Princeton Large Torus (PLT), Alcator C, and Versator II have succeeded in demonstrating current drive. In PLT, plasma currents of 200 kA have been sustained for several seconds without ohmic drive at efficiencies on the order of 2 A/W. The same system has even proven capable of raising the plasma current without benefit of an induced electric field. Until very recently, these experiments have been successful only at densities below 10^{13} cm^{-3}, raising doubts about the possibility of employing this approach to achieve steady-state operation in a reactor. In experiments employing a considerably higher frequency lower hybrid source, Alcator C has recently reported extending the density threshold to 6×10^{13} cm^{-3}. The parametric dependences of current-drive efficiency deduced from these experiments are in agreement with theory, but the observed magnitude is less than the theoretical predictions. Even if steady-state current drive could be achieved at reactor-level densities, however, the theoretical efficiency would be ~0.1 A/W, which limits Q to order 5 for FED/INTOR.

On the other hand, lower hybrid current drive may prove very valuable in a scenario in which the OH transformer is periodically recharged by the rf system at low-plasma density but at full-plasma current. By conserving the OH flux swing to provide merely the resistive compensation at high densities, such an approach would increase the burn time and reduce the number of cycles by typically an order of magnitude. Furthermore, fatigue problems that have their origin in electromagnetic cycling would be greatly reduced. Another possibility is to use lower hybrid current drive to ramp up the current in a low-density plasma, then use the OH transformer to drive long pulses in a full-density plasma. The present experiments provide some basis for such uses of current drive. Efficiencies are estimated at ~0.3 A/W.

Conclusion: The experimental basis for noninductive current drive has expanded greatly, but the prospects are still not sufficiently promising to justify adoption for FED/INTOR at this time. Because of its great potential benefit, an intensive study of the physics and technological aspects of the quasi-steady-state mode of noninductive current drive is definitely appropriate.

II.B. Ion Cyclotron Range of Frequency (ICRF) or Neutral Beam Injection (NBI) for Bulk Heating

The original (1979) decision[4] on a primary bulk heating technique for the INTOR reference design was an easy one. Neutral beam injection possessed the overwhelming advantages of exclusivity of data at high-power (>1-MW) levels, a thoroughly convincing comparison between theory and experiment, and a sequence of planned applications that would drive the technology toward FED/INTOR needs. The engineering and technological advantages of rf were recognized, but the inadequacy of the data base for the whole spectrum of rf options precluded favoring any of these options compared to neutral beams.

The NBI heating experiments have progressed considerably since 1979. Higher power (up to 7 MW) injection experiments have heated plasmas to record beta levels, and experiments will be performed in the near future at the 10-MW level to extrapolate NBI heating toward the FED/INTOR conditions. Thus, the confidence that can be placed in NBI heating is stronger today than it was in 1979. On the other hand, the acceptability of NBI heating based on D^+ sources for future tokamak reactors has been seriously questioned because of the engineering complications.

With respect to rf heating, the situation is markedly different now than it was in 1979. More than 3 MW of power in the ICRFs have been deposited in PLT via second harmonic heating, the mode favored for FED/INTOR, using a loop antenna launching structure. The power was given largely, perhaps exclusively, to ions, and the deposition profile was quite centrally peaked. These factors contributed to the high-ion heating efficiency, 3 to 4 $eV \cdot kW^{-1}/10^{13}$ cm^{-3}, a value that compares favorably with the best NBI heating efficiency on PLT.

Developments on the theoretical and modeling front have paralleled the experimental advances, permitting meaningful tests of the level of understanding of the underlying physical phenomena. While these tests are far from complete, good agreement has been obtained in those instances in which detailed comparisons between theory and experiment have been made. The initial application of these modeling tools to reactor design has resulted in optimism that heating to ignition with ICRF can be accomplished with less power and under a wider range of plasma conditions than will be the case for NBI.

The research and development (R&D) program that will be required to elevate the confidence in the physics basis for ICRF heating for FED/INTOR to the same level as NBI heating has not yet been laid out. The ICRF heating experiments must be extended to the 10-MW level. Since the feasibility of loop antenna in the FED/INTOR environment has not been established, a wider range of launching structures must be investigated. This certainly requires ICRF waveguide experiments beyond the presently planned low-power coupling experiments. It must be demonstrated that tokamak plasmas can be heated to high beta with ICRF without deleterious effects.

On the other hand, there are compelling technological and engineering design arguments in favor of ICRF for FED/INTOR. With the exception of the launching structure, ICRF hardware is already commercially available. The principal engineering advantage of ICRF is the ability to locate the bulk of the equipment in an area remote from the reactor core, thus adding to reliability, simplifying maintenance, and reducing the size of the reactor hall. In addition, the transmission system is compatible with bends to reduce neutron streaming and thus minimize shielding requirements. Other advantages of ICRF relative to NBI include higher efficiency, increased component life, and reduced complexity of required support equipment.

The decision between ICRF and NBI rests on a trade-off between the engineering and technological advantages of ICRF and the greater confidence in the physics basis for NBI. The recent advances in ICRF physics persuade us that the balance has shifted in favor of ICRF.

Recommendation: The ICRF should be adopted as the prime heating option for long-range tokamak applications. The principal backup in the long range should be negative ion neutral beams, an area in which recent advances improve the prospects for a system with high efficiency and significantly reduced neutron streaming. The backup to ICRF in the case of a near-term commitment to a reactor tokamak is positive ion beams; it must be recognized that maintaining this as a viable option entails a significant development effort.

This recommendation is made with the realization that an immediate commitment to FED/INTOR is an unlikely prospect so that there should be ample time to resolve the outstanding issues identified above. It must be emphasized, however, that the promise of ICRF can only be realized if the requisite R&D program is carried out.

III. IMPURITY CONTROL AND FIRST WALL

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

III.A. Poloidal Divertor or Pumped Limiter

The primary purposes of either the poloidal divertor or the pumped limiter are to exhaust a small amount of helium (<3%) from the plasma, to remove heat, and to provide for an acceptably small level of wall- or limiter/collector plate-eroded impurities in the plasma. Calculations indicate that both concepts can provide acceptable helium exhaust, and while the pumping requirements may be greater for the pumped limiter, this is not a major factor. Both concepts appear capable of handling the heat loads; however, the potential for preventing impurities sputtered from the divertor collector plate, which is far removed from the plasma, is certainly better than for preventing limiter-sputtered impurities from penetrating the adjacent plasma. There is also evidence from ASDEX, PDX, and D-III that poloidal divertors do function to provide cleaner discharges. Thus, there is a consensus among the physicists involved in this activity (see Acknowledgments) that poloidal divertors have a significant advantage over limiters with respect to impurity control, even though there is a smaller data base for the former.

The choice of materials for the surface of a limiter or divertor collector plate depends on the temperature, T_s, of the plasma at the edge of the sheath in front of the surface because of the energy dependence of the sputtering yield. For the limiter, this temperature is just the plasma-edge temperature, T_{ed}, whereas for the divertor collector plate it may be somewhat less, $T_s < T_{ed}$, because of cooling in the divertor channel.

Conclusion: Detailed sputtering erosion/redeposition calculations, including self-sputtering, resulted in the following conclusions:

1. Medium- and high-Z surface materials result in acceptable designs if $T_s \lesssim 50$ eV.
2. Low-Z materials result in acceptable design solutions if $T_s \gtrsim 700$ eV.
3. It may be possible to find acceptable design solutions with low-Z surfaces in the range $50 \lesssim T_s \lesssim 200$ eV if the net erosion is small, as predicted, even though the primary erosion and redeposition rates are large.
4. The erosion rates may be too large to admit acceptable solutions with any materials in the range $200 \lesssim T_s \lesssim 700$ eV.

Plasma transport calculations have been performed to evaluate the probable values of T_{ed} and T_s. With edge refueling and without high-edge radiation, it is estimated that the most probable edge temperature is in the range $\sim 100 < T_{ed} \leqslant \sim 400$ eV. Special methods for achieving the more desirable high- and low-edge temperature regions were investigated.

Transport codes were used to study the production of high-edge temperatures by using pellet injection. For these studies, the fraction of the recycled plasma pumped by the limiter system was reinjected into the discharge in the form of 1-mm-diam pellets. This reduced the recycling and increased the edge temperature. For a realistic upper limit to the pumping fraction of ~5%, temperatures in the 700- to 800-eV range were obtained when impurity radiation was neglected. Increasing pumping (i.e., 20%) produced temperatures in the 1-keV range, but this pumping level is not realistic. The consensus of the physicists involved in this activity is that although temperatures above 700 eV can be obtained in modeling studies, it would be imprudent to base the FED or INTOR design on obtaining these temperatures. This conclusion is, in part, based on the high probability that impurity radiation will reduce the temperature by an unacceptable amount.

The potential realization of edge temperatures below 50 eV in the presence of high-edge radiation was studied. These conditions were obtained in transport code studies where the impurity transport was assumed to be governed by neoclassical diffusion superimposed on an empirical transport at the hydrogen rate. Under these conditions, the impurities (e.g., iron) peaked at the edge, and a low-temperature edge was obtained with acceptable central radiation. However, this same model, applied to ISX discharges, produced the same edge peaking in impurity concentration, a condition not observed experimentally. The consensus of the physicists involved in this activity is that there is significant risk involved in basing the FED/INTOR collector plate or limiter design on obtaining edge temperatures below 50 eV because of uncertainties in the impurity behavior near the edge.

The possibility of achieving $T_s < T_{ed}$ in the poloidal divertor was investigated. The INTOR poloidal field (PF) divertor was modeled taking into account the finite parallel electron thermal conduction. The high recycling of neutrals at the divertor plate was found to produce substantial increases in plasma density near the plate and significant cooling in this region. An additional reduction in overall edge temperature results from an increase in total edge density with divertor operation. Conditions were obtained with electron temperatures as low as 25 eV at the plasma sheath. Taking into account uncertainties in the edge conditions, this is probably a lower limit on the temperature at the sheath. However, the sheath temperature at the divertor plate should certainly be significantly less than the sheath temperature at the limiter, for comparable plasma-edge conditions. If the plasma-edge temperature is sufficiently low that $T_s < 50$ eV at the divertor collector plate, then a high-Z surface can be used and the erosion would be quite small.

Recommendation: The above considerations lead to the recommendation of 120 eV as the design basis plasma-edge temperature and limiter sheath temperature, and 30 eV as the design basis divertor collector plate sheath temperature.

The divertor configuration that was analyzed was similar to the INTOR Phase I design, with flat collector plates inclined at 30 and 15 deg (see Fig. 1) to reduce the peak heat flux to 2.4 MW/m^2. The limiter configuration (see Fig. 2) considered was double-edged and shaped to achieve the same maximum heat flux. This limiter has two leading edges with peak heat fluxes of 1 MW/m^2.

Material and lifetime assessment show that beryllium is the preferred surface material for the limiter and divertor collector plate for $T_s \geqslant 50$ eV. For $T_s < 50$ eV, tantulum or tungsten would be the preferred surface material. Carbon can be ruled out based on extremely high erosion by chemical/thermal sputtering. Boron suffers from fabrication problems, poor thermophysical properties, and possibly chemical sputtering. The poor thermophysical properties and self-sputtering of silicon carbide (SiC) and titanium carbide do not permit adequate lifetime. The thermal conductivity of SiC and beryllium oxide (BeO) is degraded rapidly under irradiation to low values regardless of the initial unirradiated values.

Similar erosion/redeposition characteristics were found for the top surface of the limiter and the divertor collector plate. Designs were developed with beryllium surface materials that had a lifetime against sputtering erosion of 3.8 yr at 50% availability. Because of the concentrated particle fluxes, the erosion of the leading edges of the limiter was unacceptably large. For $T_{ed} < 200$ eV, the sheath temperature at the leading edge is $T_s < 50$ eV, and a high-Z material can be used on the leading edge to achieve an acceptable design solution from the erosion standpoint. A similar technique would be used for the divertor collector plate if $T_s < 50$ eV locally or globally.

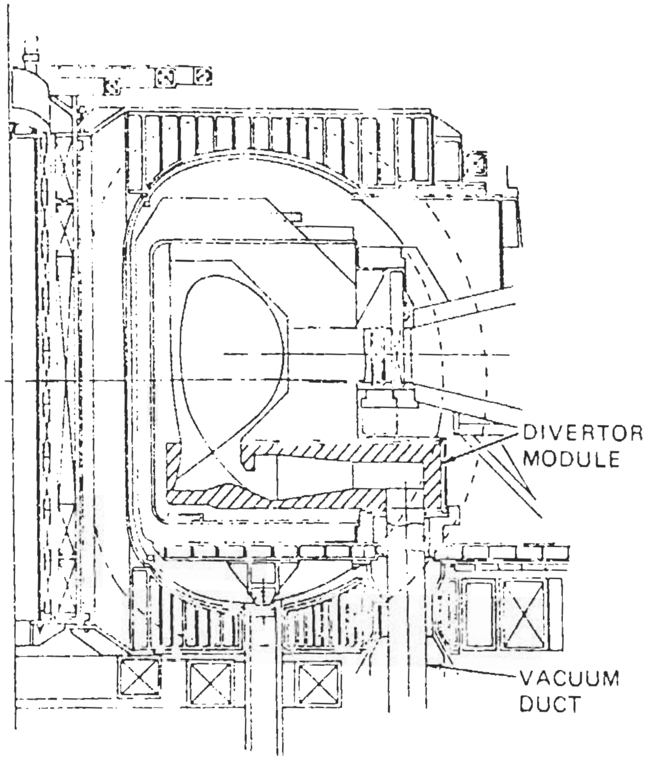

Fig. 1. FED/INTOR poloidal divertor configuration.

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

Electromagnetic analysis of the limiter during plasma disruptions shows that the forces and torques are manageable (320 KN) and arcing can be avoided if the limiter is divided into 72 sectors and the first wall is conducting.

Recommendations: For the design basis plasma-edge temperature of 120 eV, the top surface of the limiter would be covered by beryllium tiles. These tiles would be attached by a high-thermal conductance bond to a heat sink made of V–15 Cr–5 Ti, Cu–2 Be

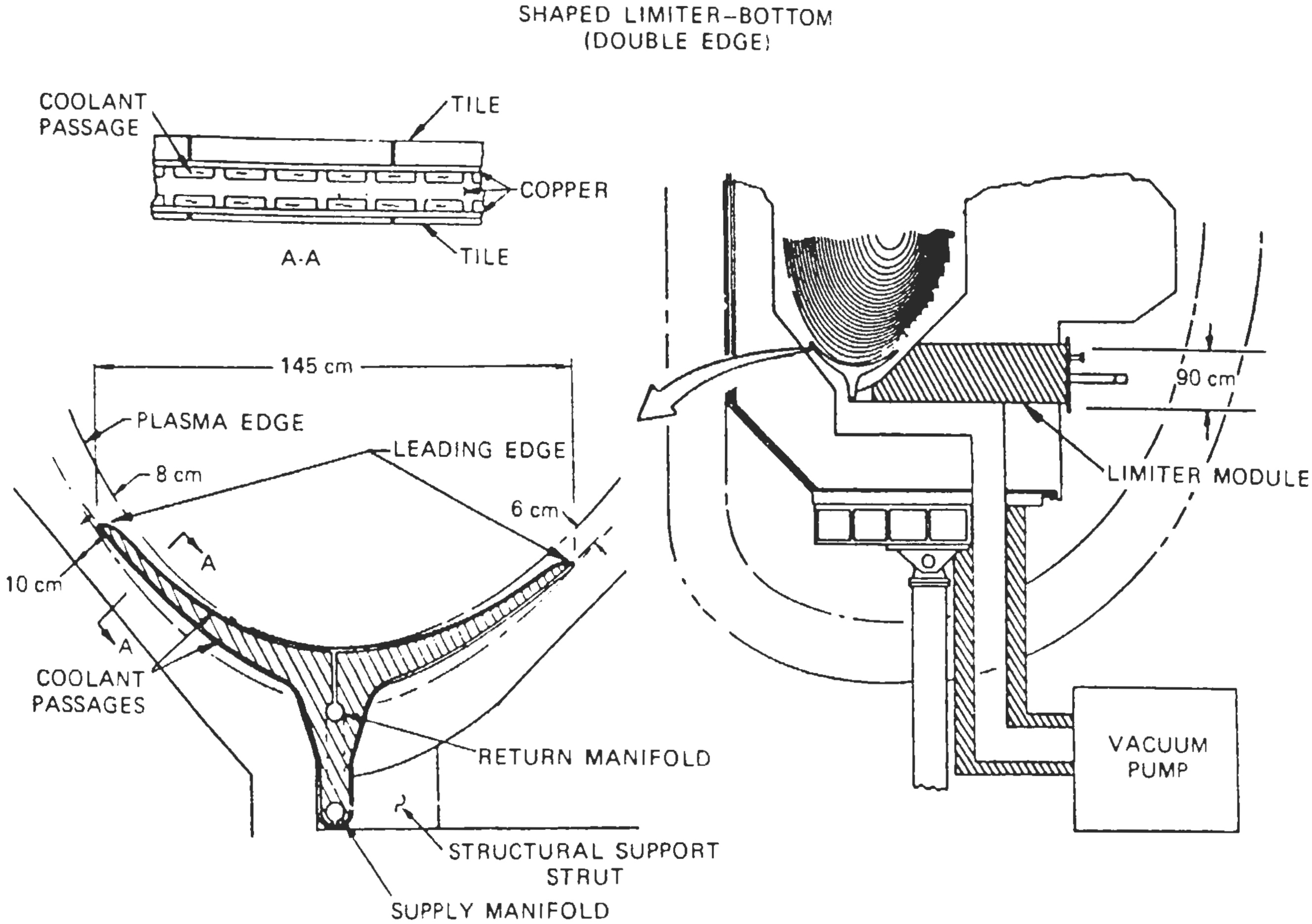

Fig. 2. FED/INTOR pumped-limiter configuration.

or tantalum alloy. The leading edge of the limiter may not be coated since a high-Z heat sink material results in extremely small net erosion.

Recommendations: For the reference plasma temperature of 30 eV at the divertor collector plate, the surface of the collector plate should be tantalum or tungsten, which would lead to an erosion resistant, long-lived collector plate under these conditions. However, the consequences of operating a high-Z surfaced collector plate at $T_s \geqslant 50$ eV are so adverse that it is prudent to base the reference design on a beryllium surface until such time that it can be established with greater confidence that $T_s < 50$ eV is achievable.

Studies have been performed to assess the impact of the choice of a poloidal divertor or pumped limiter on the overall mechanical and magnetic configuration and maintenance. The divertor and limiter modules would be configured similarly (see Figs. 1 and 2) and their maintainability would be comparable.

The poloidal field coil (PFC) configuration is simpler for the pumped limiter than for the divertor, and the associated power supplies are less expensive. If the pumped-limiter design is configured within the same toroidal field coil (TFC) system that will accommodate the divertor, the pumped-limiter design is $\lesssim 5\%$ cheaper than the poloidal divertor design and the maintainability of both designs is comparable. If advantage is taken of the ability to reduce the TFC height with the pumped limiter, an additional $\lesssim 5\%$ cost reduction is possible.

A possibly oversimplified summary of the above follows:

1. Physics favors the poloidal divertor because of the greater potential for impurity control and for achieving very low-sheath temperatures, which could significantly reduce the erosion of the plate if $T_s < 50$ eV.

2. There is very little difference upon which to base a preference from the nuclear and materials engineering point of view, although the leading edge problem with the limiter may slightly favor the divertor.

3. The mechanical configuration and maintenance are comparable for both; the PFC design and power supplies are less demanding for the pumped limiter; and taking full advantage of size reduction, the pumped limiter could result in a ~5 to 10% cost reduction.

Given the level of uncertainty, particularly in the impurity control and erosion/redeposition, and the fact that a common mechanical configuration can accommodate both concepts at the conceptual design level, the following recommendation is appropriate

Recommendation: The single-null poloidal divertor and the pumped limiter should both be carried along at the same level in the FED and INTOR designs. The mechanical configuration and maintenance schemes should accommodate both options.

III.B. First-Wall Design

The Phase I INTOR design for the first wall was relatively simple—H_2O-cooled stainless steel in a panel-type design. This design continues to meet the needs for FED and INTOR under the disruption analyses summarized in Sec. III.C. On the other hand, the carbon tiles specified for the inboard wall in the backup Phase I INTOR design and in the FED design are highly questionable in view of recently published chemical sputtering data.

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

III.C. Erosion Due to Disruptions

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

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

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

TABLE I

Lifetime Estimate for Recommended Design

(Years based on 50% availability)

	Limiter or Divertor	First Wall
No melt layer erosion	3.8	>30
All melt layer erodes[a] (Reference: 20 ms)	1.8	>15
All melt layer erodes[a] (Pessimistic: 5 ms)	0.7	2.8
All melt layer erodes[b] (Extreme: 5 ms, all energy goes to limiter/divertor)	0.6	>30

[a]Peak energy: limiter/divertor = 270 J/cm^2; first wall = 175 J/cm^2.

[b]Peak energy: limiter/divertor = 535 J/cm^2; first wall = 35 J/cm^2.

[a,b]Minor disruptions: limiter/divertor = 170 J/cm^2; first wall = 0.

annual limiter or divertor module replacement would be required.

III.D. Other Impurity Control Methods

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

The use of NBI-driven impurity flow reversal was assessed. Calculations, based on a model that had been calibrated to match the PLT flow-reversal experiment, indicated that ~20 MW of coinjected beam might be adequate to inhibit limiter-sputtered impurities from penetrating to the center of the plasma.

IV. TESTING

The operational requirements on FED/INTOR for engineering testing were evaluated. Three different aspects of testing were considered:

1. fluence requirements for structural materials radiation damage
2. long-term operational requirements for establishing component reliability
3. short-term operational requirements for blanket testing.

The results of this section will subsequently be combined with the results of the cost-risk-benefit section to arrive at a recommendation on fluence objective.

IV.A. Fluence Requirements for Structural Materials Radiation Damage Tests

The impact of materials irradiation testing in FED/INTOR on reducing the uncertainty in the structural materials data base for the demonstration plant (DEMO) design was evaluated for stainless steel and for an advanced alloy, typified by a vanadium alloy. The type of information obtainable is shown in Table II. The value of information from FED/INTOR in reducing uncertainty depends on the availability of a high-fluence neutron source such as Fusion Material Irradiation Test. The impact of different fluence levels in FED/INTOR on the risk associated with the performance of the structural materials in DEMO is summarized in Table III.

IV.B. Long-Term Operation Component Reliability

The benefit of long-term component operation was quantified in terms of the number of hours of operation that would be required to assure an 80% confidence level in predicting the mean-time-between-failure (MTBF) for that component in the DEMO. Anticipated MTBFs were established for FED/INTOR and DEMO components; then the test time required

TABLE II

Structural Material Fluence Effects

Maximum Testing Fluence	Benefits of Testing
Stainless steel	
0 to 1 $MW \cdot yr/m^2$	Little useful information
1 to 2 $MW \cdot yr/m^2$	Preliminary validation (limited) of models
2 to 3 $MW \cdot yr/m^2$	Confirmation of low-fluence effects predicted from other (e.g., fission) testing, particularly for tensile properties
3 to 6 $MW \cdot yr/m^2$	Model verification from observations of microstructure preceding long-term behavior change (possibly observation of onset of swelling)
Above 6 $MW \cdot yr/m^2$	Confirmation of performance (particularly swelling) near end of life
Vanadium alloys	
1 to 2 $MW \cdot yr/m^2$	Tensile properties still changing
2 to 3 $MW \cdot yr/m^2$ [a]	?
3 to 6 $MW \cdot yr/m^2$	?

[a]For components such as limiter/divertor, the lifetime is short because of erosion; 2 to 3 $MW \cdot yr/m^2$ is near end of life in this case.

TABLE III

Effect of FED/INTOR Fluence on Structural Materials Risk* in DEMO[+]

	With Fusion Materials Irradiation Test (FMIT)	Without FMIT
Stainless steel	3 $MW \cdot yr/m^2$ low risk 0 low to medium risk	3 $MW \cdot yr/m^2$ low to medium risk
Vanadium alloys	?? (3 to 5 $MW \cdot yr/m^2$) ?? Medium risk	3 to 6 $MW \cdot yr/m^2$ High to medium risk

*Here medium risk is defined as having a good chance that the material will exceed 50% of design life, but a significant chance that 100% of design life may not be met.
[+]Assuming Fission Reactor Test Program.

in FED/INTOR was determined from reliability analysis, taking into account the number of such components present. The results are summarized in Table IV.

The major benefits that would result from long-term component operation in FED/INTOR are

1. definition of failure modes
2. determination of failure rate and distribution
3. determination of failure recovery time
4. identification of design improvements.

TABLE IV

Test Time Required in FED/INTOR to Attain 80% Confidence in Predicting MTBF for DEMO

Component	Test Time	
	Hour	MW·yr/m²[a]
Pellet injector	1 750	0.3
ICRF system	5 880	0.9
First-wall system	7 000	1.0
Cryogenics system	7 000	1.0
EFC system	9 800	1.5
Diagnostics system	17 500	2.5
TFC system	21 000	3.1
OH coil system	21 000	3.1
Spool structure system	22 400	3.3
Pumped limiter	28 000	4.1
Vacuum pumping system	28 000	4.1
ICRF launcher	35 000	5.2
Torus sector modules	40 600	6.0
ICRF generator	46 000	6.8

[a]$\mathrm{MW \cdot yr/m^2} = \dfrac{1.3 \times \mathrm{hour}}{8760}$.

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

IV.C. Blanket Testing Requirements

The requirements on FED/INTOR operation were assessed for several different types of blanket tests—neutronics, tritium recovery, materials compatibility, heat recovery, and breeder lifetime. In general, the neutronics tests do not impose significant requirements on operation. The minimum continuous operating times were estimated for solid breeder tritium recovery tests and for heat recovery tests. Solid breeder microstructural and thermophysical property changes with radiation are estimated to saturate at ~0.2 $\mathrm{MW \cdot yr/m^2}$ neutron fluence. Since FED/INTOR will have a lower volumetric nuclear heating rate than DEMO, it will be necessary to simulate the DEMO thermal-mechanical conditions in specially designed test modules to obtain relevant information on tritium release, heat transfer, and materials compatibility. These results are summarized in Table V.

V. TRITIUM

Three areas are addressed:

1. tritium permeation through first wall, limiter, and divertor
2. tritium containment as it relates to tritium contamination of the reactor environment and its impact on personnel access
3. key issues related to the design and performance of the tritium-producing blanket.

V.A. Tritium Permeation

Analysis of tritium permeation into the H_2O coolant of the first wall and limiter or divertor plate

TABLE V

Operating Requirements for Blanket Tests

	Tritium[a] Recovery	Heat Recovery
Minimum continuous operating time		
$P_n = 0.4\ \mathrm{MW/m^2}$, $\Delta t_{burn} = 100$ s	~120 h	~3600 s
$P_n = 1.3\ \mathrm{MW/m^2}$, $\Delta t_{burn} = 200$ s	~65 h	~1250 s
Minimum number continuous operating cycles		
$P_n = 0.4\ \mathrm{MW/m^2}$, $\Delta t_{burn} = 100$ s	~2700	~25
$P_n = 1.3\ \mathrm{MW/m^2}$, $\Delta t_{burn} = 200$ s	~950	~5
Minimum fluence	~0.2 $\mathrm{MW \cdot yr/m^2}$	---

[a]Solid breeder.

reveals major uncertainties in the data base, particularly parameters related to surface conditions and neutron damage trapping. For some important candidates for plasma side materials on the limiter and divertor plate, the data base is poor for tritium diffusivities and solubilities. The time to reach steady-state levels for the tritium inventory and permeation rates is found to be a key factor for components with short life such as the limiter and divertor plate. For the general type of designs and materials considered for FED/INTOR, the steady-state tritium permeation rate is estimated to be in the range 10^2 to 10^4 Ci/day. The steady-state tritium inventory in the in-vessel components is in the range of 0.1 to 1 kg. Cost/benefit analysis based on as-low-as-reasonably-achievable guidelines shows that the concentration of tritium in the water should be kept in the range of 2 to 10 Ci/ℓ or lower, depending on the tritium permeation rate. A number of methods exist for separating tritium from water. Based on considerations of available technology and cost, the recommended technology is combined electrolysis catalytic exchange/cryogenic distillation. The capital cost depends strongly on the permeation rate to the coolant and the allowable tritium concentration in the water. For the FED/INTOR conditions, the projected capital cost is in the range of \$5 to 15 million.

Conclusion: Tritium permeation is not a feasibility issue for FED/INTOR. However, the economic penalty can be quite large if the tritium permeation rate is $>10^4$ Ci/day. A substantial improvement in the data base for tritium permeation parameters is needed.

V.B. Tritium Contamination in Reactor Hall

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

Under normal operation, the tritium source term is estimated to be <30 Ci/day. Most of the tritium comes from the water coolant system. During maintenance, the tritium source term can be as high as 10^3 Ci/day with most of the tritium released during removal of the different torus components. An accident could result in a release of up to 10^5 Ci, but most accidents would result in a much lower release.

The maintenance case is the dominant factor in sizing the required atmosphere detritiation system. With an average source term of 100 Ci/day, the total capital cost plus a 10-yr operating cost is \$360, 60, and 18 million to maintain tritium concentration levels in the reactor atmosphere of 5, 50, and 500 μCi/m^3, respectively. The respective worker efficiencies for these concentration levels are 0.6, 0.5, and 0.5. Bubble suits with an independent air supply are required if the worker spends extended periods at levels of ≥ 50 μCi/m^3. Since a tritium level of 5 μCi/m^3 does not permit a significant increase in worker efficiency while it entails a substantial cost penalty, maintaining tritium levels of <50 μCi/m^3 in the reactor hall does not appear to be justified.

Tritium release from the reactor building to the environment is another factor that favors lower tritium concentration in the reactor hall. At a tritium level of 50 μCi/m^3, the environmental release is ~50 Ci/day. This is comparable to routine releases from operating Canada deuterium uranium reactors in Canada (~70 to 100 Ci/day).

For a worker in a bubble suit, the radiation dose received in the reactor hall due to the presence of tritium at ~50 μCi/m^3 is only 20% of the radiation dose that will be received from the 2.5 mrem/h gamma radiation. Therefore, maintaining tritium levels of ~50 μCi/m^3 does not result in substantial increase in the dose burden.

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

1. It appears that some degree of tritium protection will be required for maintenance personnel to enter the reactor building. The worker efficiency can be reduced by a factor as large as 2. Earlier studies for FED and INTOR, which established the benefits of personnel access, did not account for such a penalty for worker efficiency. Therefore, reevaluation of the maintenance strategy is necessary.

2. The utilization of robotic units for maintenance operations, particularly those requiring <1 day, appears sufficiently meritorious to deserve serious study. More than half the maintenance repair operations are expected to be completed in <1 day. Therefore, the 24-h "wait period" for the gamma radiation dose to decay to ~2.5 mrem/h significantly reduces the availability. Furthermore, a significant number of maintenance operations will involve higher tritium exposure than the average case considered above, for example, repair of coolant lines.

V.C. Tritium-Producing Blanket

Experimental data that have become available since the INTOR Phase I design indicate that tritium recovery *in situ* from Li_2O may be more promising than was previously thought. Thus, because of its higher breeding potential and comparable other properties, Li_2O is the preferable solid breeding material. However, the effects of radiation on tritium recovery and the ability to control the thermal conductance across a gap between the solid breeder and the coolant tube remain as feasibility issues for solid breeders. Thus, a liquid breeding blanket alternative should be maintained.

Both Li_2O and lithium-lead (Li-Pb) were analyzed for the breeder blanket. The operating temperature range of 410 to 660°C for Li_2O has been affirmed. For Li-Pb, selection of a coolant is a key issue. The compatibility of Li-Pb with the structural material is a key feasibility issue.

Methods to accommodate variations in the device power level were examined for the solid breeder blankets. The maximum variation in power that can be accommodated is limited to ~30%.

VI. MECHANICAL CONFIGURATION

VI.A. TFC Size

The objective of the 1981-1982 mechanical configuration critical issue study was to produce a new design concept with a significant reduction in the size of the tokamak device while maintaining the plasma size and performance of the Phase I INTOR and the FED designs. An increase in the allowable magnetic field ripple from 0.7 to 1.2% was originally proposed.

As a result of the new concepts developed in this phase, we have produced a new design configuration with a reduction in TFC size of ~15% from a bore of 7.7 × 10.7 m to 6.6 × 9.3 m. The magnetic ripple for this design has increased to only ~0.9%. Due to the strong influence of TFC size on such other tokamak systems as PFCs, power supplies, and machine structure, the overall cost of the device may be reduced by over 10%.

Recommendation: A reduced size (~15%) TFC system be adopted for FED/INTOR.

VI.B. Torus Segmentation

Torus segmentation is the most important consideration for maintenance and access to the plasma chamber. An evaluation was made of several torus segmentation and disassembly concepts that would be consistent with the reduced-size TFC system. Torus segmentation concepts that were considered include:

1. number of torus sectors equal to number of TFCs, with straightline radial-horizontal sector removal (Phase I INTOR and FED)

2. number of sectors equal to a multiple of the number of TFCs, with straightline nonradial horizontal sector removal or a combination of straightline radial-horizontal and toroidal rotation sector removal.

The 12-sector (first) option was selected for the INTOR Phase I conceptual design based on the fact that it provided the maximum access surface for penetrations to the plasma chamber and was the simplest design approach for assembly and removal of the blanket and first-wall sectors. A similar logic led to the 10-sector FED design.

For the reduced-size TFC option, it first appeared that a multiple sector design was necessary. However, with modifications in the vacuum boundary and shielding concepts, it appears that the 12-sector design is still feasible. For the given FED/INTOR size, the present TFC geometry appears to be at the limit for which this concept applies. For this reason, it also seems prudent to develop a 24-sector concept in the event that ripple limits are substantially increased or the TFC cross section needs to be increased for magnetic or structural reasons.

Recommendation: The 12-sector segmentation concept should be retained for FED/INTOR, but a 24-sector concept should be developed, too.

VI.C. Universal Design Concept

The INTOR Workshop participants also agreed to a number of common design features to produce a "universal" design concept that can be used as an update to the FED/INTOR design concept as well as a common reference for future activities. The universal design concept incorporates several major changes from the FED and INTOR Phase I concept related to the vacuum topology, the torus, and the structural design. The configuration also provides sufficient flexibility to accommodate the uncertainty involved in the choice of bulk heating and impurity control methods. The universal design can accommodate NBI or ICRF for heating and a poloidal divertor or pumped limiter for impurity control. The features of the universal design concept are summarized in the following sections.

VI.C.1. Combined Vacuum Boundary

An extensive evaluation of several concepts for separate and combined vacuum boundaries for the torus and superconducting magnet vacuum systems was performed. The most favorable vacuum topology was determined to be one in which the torus and superconducting magnetic structure are integrated into a single combined system.

VI.C.2. Torus Closure

Several possibilities for torus closure were studied, including a separate vacuum door and integration of the vacuum closure with the torus shield. The latter option is recommended to improve access and maintenance.

VI.C.3. Torus Segmentation

Two options will be maintained for torus segmentation - 12 or 24 sectors.

VI.C.4. Divertor and Pumped-Limiter Configuration

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

In the case of the pumped limiter, continuous limiter plates have been specified since the flux lines pass beneath the limiter, allowing impurities created by a triangular gap to be carried back to the plasma. There is insufficient width in the multiple torus sector configuration to extract a full unsegmented limiter. There appears to be insufficient side wall thickness to provide water feed line access to the shield post in the case of the equal torus sector configuration; therefore, the split limiter is preferred for the universal divertor/limiter reactor design. Design layouts have been made showing that there is adequate space for an unsegmented limiter when the reactor is designed solely for the limiter.

In the case where the pumped limiter is located on an outboard 45-deg line, a centered plasma about the TF horizontal centerline can result with symmetrically located PFCs. For this condition, it is possible to reduce the TFC bore in height by 40 to 80 cm, compared with the nonsymmetric plasma condition. This would allow the TFC size to be reduced and also diminish the overturning moment acting on the TFCs. However, it is very difficult (if not impossible) to locate the passive control loops in the blanket for plasma stabilization because of interference with the pumped-limiter module. This configuration also reduces the blanket area, thereby affecting the tritium breeding ratio. For these reasons, this pumped-limiter plate position has been rejected.

VI.C.5. PFC Configuration

A very difficult maintenance situation could be brought about if the lower outboard equilibrium field coil (EFC) has to be replaced and it is located in the same vacuum boundary as the TFC system. Conversely, it is significantly easier to replace this coil if it is in a separate vacuum boundary located just outside the main superconducting magnet vacuum system. Although it may not be essential to locate the lower outboard EFC in a separate vacuum boundary if it truly has a high-reliability value, it is prudent to locate this coil in a separate vacuum boundary in order to enhance its maintenance features.

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

VI.C.6. TFC Configuration

The dominant factor in determining the final mechanical configuration of the tokamak reactor design, regarding equal versus multiple torus sector arrangements, is the cross-sectional area requirement of the TFCs. There must be sufficient area to house the conductor and provide the structural support to constrain it against the magnetic loads. Different conductor and structural designs have been presented in the INTOR Workshop; unfortunately, they do not all fall within the same envelope.

Recommendation: A universal design concept should be adopted for future studies.

VII. MAGNETICS AND TORUS ELECTROMAGNETICS

VII.A. Magnetic Systems

VII.A.1. TF Level Versus Cost

Conductor cost over the 8- to 12-T range shows a steep dependence on field, but one without discontinuous jumps as conductors or cooling approaches are varied. Niobium titanium (Nb-Ti) and niobium tin (Nb_3Sn) show nearly identical cost at 8 T for 4.2 K operation. At 10 T, a Nb-Ti conductor at 1.8 K is ~30% less costly than Nb_3Sn, and at 12 T, niobium titanium tantalum is 30% cheaper than Nb_3Sn. The cost of the conductor at 8, 10, and 12 T is in the ratio of 1.0:1.5:2.25, or approximately proportional to B^2. The conductor for a suitably graded 12-T coil costs ~2.5 times as much as the conductor for a similar size 8-T coil.

The winding cost as a function of field is approximately proportional to the number of turns required, which is a function of the peak field and the current the conductor carried at this field. In the 8- to 12-T range, winding cost is approximately proportional to B^2.

The structural cost, reflecting ~10% of the magnet cost, is proportional to $B^{3/2}$ for the coil cases and to B for the intercoil structure.

The cost of the coils at a fixed field is dependent on their size and is approximately proportional to their peripheries. The variation from the baseline FED to the smallest TFCs considered is ~25%. The cost of the structure for any given size, field, and impurity control approach appears to be rather insensitive to the details of the PFC location.

VII.A.2. TFC R&D Requirements

The Large Coil Program will provide a large-scale demonstration of Nb-Ti and Nb_3Sn at 8 T in a tokamak configuration. In addition, Tore Supra will test Nb-Ti at 9 T and 1.8 K and T-15 will test Nb_3Sn at 8 T. The U.S. 12-T program will test conductors at 12 T, but will not provide large coil experience. The lack of large coil fabrication and operation for any field above 8 T in the United States represents a development and demonstration gap that exists since the field for INTOR is significantly above this level.

Conclusion: An assessment of the required R&D indicates that the type of tasks needed and time required to perform them are independent of the INTOR field level, assuming that it is in the 10- to 12-T range.

VII.A.3. Conductor/Coolant Options for TFCs

Three candidate conductor coolant options have been evaluated for FED/INTOR:

1. Helium I, forced-flow, Nb_3Sn
2. Helium I, bath-cooled, Nb_3Sn/Nb-Ti
3. Helium II, bath-cooled, Nb-Ti.

Specific configurations for each of these options were considered viable for FED/INTOR requirements; that is, none exhibit inherent characteristics that would render the option unacceptable from a performance or reliability standpoint. The data base for design and operational experience with existing systems differs somewhat between options, but no large-scale demonstration at the 11- to 12-T level has been accomplished for any of the cases. The primary impact on system design occurs in the required radial build for the winding and TFC case, which may range from 0.82 m for the higher current densities envisioned for forced-flow configurations to as much as 1.03 m for a Helium I bath-cooled configuration with a conservative stability margin and sufficient internal structure so that the coil case carries no in-plane loads. The nominal 0.87-m radial build estimated for INTOR Phase I may, therefore, be somewhat low, but will continue to be utilized for the build for all of the cases until more detailed structural and operational calculations are performed in the future.

VII.A.4. PFC Distribution Studies

A systematic study of PFC distributions has been completed, involving ~300 runs of an MHD equilibrium code. Results show that the cost of the PF system, coils, and power supplies is a major cost item in the overall machine cost, representing ~20% of the total. The studies show a strong effect with overall TFC size. The reduced-size TFCs adopted in Phase II allow PF system cost reductions of 25% over the baseline scale. The studies also show a cost penalty associated with a poloidal divertor over a pumped limiter and a strong impact of system integration considerations. For example, the cost of an optimized coil set immediately outside the TFC boundary must be increased by 30% when the coils are actually located to avoid interferences at structural and subsystem interfaces.

VII.A.5. PFC Design

Design studies have examined both pool boiling concepts and internally cooled conductors. In general, both concepts are shown capable of satisfying the coil requirements. For example, studies have confirmed that an 8-T central solenoid is feasible for the 6-s startup ramp, and that it can utilize either conductor approach.

The studies have examined fault currents induced in various PFCs under the assumption of an internal short in a given coil. Certain well-coupled PFCs will be subjected to abnormal currents under these conditions, thus requiring active protection schemes. Further analysis will be required at a later stage in the design, but no insurmountable problems are foreseen.

VII.B. Torus Electromagnetics

VII.B.1. Passive Stabilization

Passive stabilization against vertical and radial displacements by means of saddle coils, complete shells, and partial shells has been examined and the relative effectiveness assessed. Coils or shells must be no further out than the blanket outer boundary if they are to meet the following criterion: A given displacement of the plasma must generate induced currents, which, in turn, give rise to a restoring field equal to the field change encountered by the plasma in being displaced. FED/INTOR conditions require ~0.1 T/m of displacement. Coils or shells placed beyond the blanket outer boundary, for example at the shield outer boundary, will not meet this criterion.

Conclusion: Passive stabilization elements must be located no further away from the plasma than the outer boundary of the blanket.

Complete shells are approximately two times as effective as optimally located single coils in providing restoring forces. Partial shells that are toroidally continuous need occupy only ~180 deg poloidally to be nearly as effective as complete shells. The time constant of a partial shell of equivalent 4-cm-thick stainless steel will be ~150 ms. Segmented blanket modules that are not toroidally continuous have time constants approximately one-half to one-third that of continuous shells. The time constant associated with the naturally occurring structures appears somewhat low, and therefore higher conductive plates or passive coils should be seriously considered and integrated into the design at a future date. More extensive studies with improved plasma models are necessary to determine if plate- or shell-like passive elements are more effective stabilizers since their induced fields may be expected to be more uniform in the plasma region.

VII.B.2. Active Stabilization

The requirements for active stabilization depend on the characteristics of the passive stabilization system. The longer the time constant of the passive system, the slower the response of the active system, provided that the passive system meets the criterion for full restoration of plasma position. Only passive systems sufficiently close to the plasma meet this criterion, whereas all toroidally continuous shells inside the coil locations, no matter where they are located, impede penetration of its active control fields. A roll-over frequency criterion is proposed in which the capability to supply active control should be on a time scale equal to the decay time constant of the passive system. The peak power required for a set of control coils located at the shield outer boundary will be ~13, 8, and 5 MW, respectively, for assumed passive system time constants of 75, 150, and 300 ms. If the control coils are located external to the TFCs at the position of the Phase I INTOR or FED PFC set, approximately one order of magnitude more power is required.

Recommendation: It is therefore recommended that active control coils at the position of the outer shield be given serious consideration. These coils could be saddle shaped with one coil per major torus segment. The integration of these coils into the design is necessary.

VII.B.3. Startup

Startup coil energy requirements are dependent on the location of the startup coils and on the characteristics of any toroidally continuous partial shells that have been included for passive stabilization or for structural reasons. Such startup or "blip" coils should be utilized only during times that are short compared with the plasma buildup. Voltages required during times comparable with plasma buildup (i.e., 6-s scale) are more appropriately supplied by the OH system. As an example, if 35 loop volts are to be supplied and held for 500 ms, 40 MJ of stored energy would be required if there were no toroidal shells present and if the coils were located at the outer shell boundary. If the same conditions are to be met by coils at the location of the Phase I INTOR PFCs, 140 MJ would be required. If a toroidal shell with a time constant of 150 ms was present, and if the 500 ms must be held after reaching 35 V (i.e., after the 150-ms delay), the energies become 170 MJ for the inner location and 580 MJ for the outer location. These reasonable energies will not become prohibitive, provided that voltages are not sustained for long periods and that loop voltages are kept low. Further study on the interplay of system time constants to arrive at a design optimized with respect to startup, passive stabilization, active stabilization, and control is essential to refine techniques and models currently in use.

VII.B.4. Disruption-Induced Forces and Torques on Sectors

The rapid flux change associated with a plasma disruption induces eddy currents in the torus shells that interact with the TF to produce a net overturning moment about a line through the machine axis and the middle of a sector. An equal and opposite torque about the plasma axis on the end plates of the sector yields a twisting action with no net moment on the sector. Estimates indicate that these loads are significant and require integration into the sector structural design, but that the load level is manageable.

VII.B.5. Disruption-Induced Forces, Torques, and Voltages on the Limiter

Estimates of the electromagnetic loads and voltages on the limiter following a disruption indicate a strong dependence on the time constant of the torus and on the electrical continuity of the limiter in the toroidal direction.

Conclusion: Results indicate that limiter segmentation is essential to reduce loads to tolerable levels and that as many as 72 segments may be required to reduce the voltage between segments to the 10-V level. Further study to determine the allowable voltage level for design purposes is necessary.

VII.B.6. Disruption-Induced Arcing Between First-Wall Segments

The voltage appearing across gaps in a segmented first wall will be a function of the distribution of resistance around the first wall and in the gaps, and of the existence of other toroidally continuous partial shells that can slow the collapse of flux following a disruption. In a typical INTOR 20-ms disruption. ~40 V/gap will occur if the equivalent first wall and gap wall is 1.0 cm of stainless steel, and if a partial toroidal shell exists at the position of the inner spool piece. If the gap wall thickness was 4 cm, the voltage would be reduced to ~20 V. It would appear that gap voltages may be held below the critical breakdown level by partial toroidal shells and appropriate choice of gap wall thickness. Further investigations into the threshold level for arcing would be desirable. Refined models of the plasma and its motion together with wall heating during the disruption process are necessary.

VII.B.7. Eddy Current Modeling Codes

Presently available computer codes are adequate for conceptual design and design trade-off purposes. However, efficient codes for modeling three-dimensional problems will require development for use in later stages of design.

VIII. COST-RISK-BENEFIT

VIII.A. Cost Reductions

A number of possible changes in design option, relative to the Phase I INTOR design, were considered to achieve an overall reduction in operating or capital cost. The results are summarized in Table VI and discussed below.

The TFCs can be reduced in bore by ~15% before the physics limitation of field ripple is approached. Mechanical configuration studies confirmed that the torus maintenance and assembly scheme of Phase I was still feasible with this reduction in TFC size (see Sec. VI.A). Thus, an ~12% reduction in capital cost (including the reduced PFC system costs) is possible.

Simplification of the PFC system leads to a capital cost reduction of ~4% when the poloidal divertor is replaced by a pumped limiter. If the TFC is further reduced in size, as is mechanically feasible with the limiter, an additional ~4% cost reduction is possible. Because of the uncertainty discussed in Sec. III.A, it is not possible to take advantage of this potential cost reduction at this time.

Cost reductions can be achieved by reducing the heating power margin and by changing from NBI to ICRF. This is judged to be technically feasible (Sec. II.B), implying an ~8% reduction in capital cost.

Placing a tritium breeding blanket on the inboard, as well as top and outboard, of the torus to achieve a tritium breeding ratio of ~1 would reduce the operating cost by ~14%, while increasing the capital cost by ~3%. The engineering consequences have not been examined.

Conclusion: Capital cost reductions of ~20%, relative to the Phase I INTOR design, are feasible.

TABLE VI

Cost Increments—Design Options

Design Option	Change (%) Capital	Operating
Reduced TFC (~15%)	−12	−8
Impurity control (PD → PL)	−4	−6
Heating method (NBI → ICRF)	−2	−1
Heating power (94 → 50 MW)	−7	−3
Tritium production (TBR[a] ~0.6 → ~1.0)	+3	−14

[a]TBR = tritium breeding ratio.

VIII.B. Cost-Risk-Benefit Assessment of Performance Objectives

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

A cost-risk-benefit comparison of alternatives with different fluence objectives was performed. A risk-benefit figure-of-merit (FOM) was defined by taking into account the importance of the information from FED/INTOR to the design basis for the DEMO and the probability that a given FED/INTOR alternative could provide the information required of FED/INTOR. This probability comprised two factors:

1. design objective (e.g., fluence goal) of the alternative
2. risk associated with achieving the design objective.

The FOM was normalized so that the ideal FED/INTOR alternative, which provided all the information necessary to supplement that from the base program and complementary facilities to complete the DEMO design basis, would have a value of unity. This FOM and the costs are shown for four different alternatives, ranging in fluence capability from 0.2 to 10.0 MW·yr/m^2, in Table VII. There is relatively little difference in the base capital costs, but a rather large difference in operating costs, which results from the different operational lifetimes and tritium costs. The first case only produces ~40% of the information (in an importance-weighted sense) that is required of FED/INTOR, while the other cases produce ~70 to 80% of the required information. The cost of producing the missing information elsewhere, or alternatively the risk of designing the DEMO without it, has not been factored into these numbers. The additional costs that also might be associated with achieving the higher availability that would be necessary with the higher fluence cases have not been factored into the capital costs.

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

The evaluation of testing requirements in Sec. IV indicated a strong incentive to achieve ~2 to 3 MW·yr/m^2 for structural materials properties (tensile and microstructural change) data and for component reliability data. While there was incentive to achieve higher fluences, this incentive was not so compelling.

Recommendation: FED/INTOR should be designed to achieve $\phi \cong 3$ MW·yr/m^2 with a high probability of success. The design should allow for operation to achieve higher fluences, but with a lower probability of success.

IX. R&D

During the course of the work on critical issues in the FED/INTOR activity in 1981-1982, several limiting uncertainties were identified that inhibited further development of the design concept. Specific R&D programs that could resolve these uncertainties have been identified, as summarized in Table VIII. These specific R&D program recommendations supplement the broad R&D program needs that have been defined in the past and the specific R&D program recommendations made in INTOR Phase I.

X. OBJECTIVES FOR FED/INTOR

The goal of the magnetic fusion development program is to establish the engineering feasibility of magnetic fusion. A working definition of engineering feasibility is an adequate physics, technology, and engineering data and experience base for the design and construction of a DEMO that would operate at high availability, produce several hundred megawatts of electrical power, breed more tritium than it consumed, and demonstrate the commercial feasibility of fusion. The broad prerequisites for the design and construction of such a DEMO are:

1. adequate data base of basic plasma physics and engineering properties
2. demonstration of the plasma operating conditions required for the DEMO
3. adequate materials radiation damage and plasma-wall interaction data base
4. development of fusion reactor prototypical components

TABLE VII

Cost-Risk-Benefit Summary

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

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

TABLE VIII

Specific R&D Program Recommendations

	Subject	Priority[a]
Physics		
P.1	Reactor prototypical ICRF heating	1
P.2	ICRF code development	2
P.3	Noninductive current drive demonstration	(1)
P.4	Size scaling of confinement	1
P.5	rf startup assist	2
P.6	Plasma equilibrium control	1
P.7	Plasma-edge characteristics	1
P.8	Divertor channel plasma behavior	1
P.9	Divertor impurity transport characteristics	1
P.10	Pumped-limiter heat load distribution	1
P.11	Plasma impurity transport	1
P.12	Pumping characteristics	1
P.13	Plasma radiative edge condition	2
P.14	Plasma high-edge temperature condition	(2)
P.15	Charge-exchange flux distribution	2
Nuclear		
N.1	Tritium permeation	1
N.2	Erosion/redeposition	1
N.3	Material response to disruptions	1
N.4	Self-sputtering yields	1
N.5	High-heat flux component development	1
N.6	Component thermomechanical response	2
N.7	*In situ* recoating	2
Engineering		
E.1	High-power ICRF system demonstration	1
E.2	Intermediate-scale TFC fabrication and operation	2
E.3	TFC superconductor property and cost improvement	1
E.4	TFC mechanical and electrical properties of composites and components	1
E.5	PFC conductor mechanical properties, support requirements, and voltage withstand capability	1
E.6	Flaw growth detection techniques	(2)
E.7	Voltage withstand criteria for torus components	1
E.8	Quench detection and discrimination	(2)
E.9	PFC conductor development	2
E.10	Intermediate-scale PFC demonstration	2
E.11	Dewars and structural coil cases	1

[a]Parentheses indicate that an item does not impact present reference design, but is important for other reasons.

5. integration of reactor-prototypical components into a fusion reactor system
6. reliability testing of reactor-prototypical components
7. testing of a maintenance and assembly procedure and associated equipment
8. testing of the capability for tritium production and recovery and of electricity production by fusion
9. testing of design features that ensure safety and environmental acceptability
10. demonstration of a sufficient level of reactor system reliability and availability to justify extrapolation to the high availability required for the DEMO.

The basis for the design and construction of the DEMO will be provided in part by the base plasma physics and technology development program, in part by a next-generation reactor of the FED/INTOR genre, and in part by complementary test facilities. FED/INTOR will be absolutely essential for items 2, 5, and 10, will be the major source for items 6, 7, and 9, and will be a substantial

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s 3, 4, and 8. The base program
ary facilities will provide the domi-
ons to item 1 and will make substan-
s to items 3, 4, and 8.

e plasma physics experimental and
gram must support the design and
f FED/INTOR and will supplement
n providing the physics basis for the
nstruction of DEMOs. In this context
is viewed as the maximum reasonable
beyond the next generation of large
okamak Fusion Test Reactor (TFTR),
ean Torus (JET), JT-60, and T-15]
amak DEMO and is intended to demon-
hievement of most of the plasma condi-
will be required for tokamak DEMOs.
sics objectives of FED/INTOR then are
te the operation of an ignited or high-Q
ritium (D-T) plasma, to achieve long
reproducible burns with optimized plasma
, and to investigate noninductive current
ievement of these objectives requires satis-
purity control, power and particle balance
nd profile control for parameter optimiza-
losely related objective is the achievement
uty-cycle (≥70%) operation. FED/INTOR
be used to perform certain plasma physics
nts not directly related to learning how to
FED/INTOR, but such experiments should
ed out in other plasma physics devices if

extensive plasma engineering technology and
ent development and testing program will
uired in the development of fusion power
s to the demonstration reactor stage. This
n will both support FED/INTOR in providing
sis for its design and construction, and supple-
FED/INTOR in providing the basis for the
and construction of the DEMOs. The reli-
testing of plasma engineering components
magnets and heating systems) can be carried
o some extent in test facilities, but it may be
cost effective to do such testing in FED/INTOR
proper simulation of the fusion reactor envi-
nent is important.

The screening of candidate structural, breeder,
other materials will be carried out in the base
gram. Blanket, first-wall, limiter, and divertor
nponents will be developed and tested in comple-
ntary facilities, and high-fluence structural radia-
n damage data will be accumulated in fission
actors (for nickel-containing alloys) and in accelera-
r neutron sources. FED/INTOR will then serve
proof-test these components in a fusion reactor
nvironment and to accumulate sufficient radiation
amage data to benchmark that from other sources.

A DEMO must have availability substantially in
excess of 50% for prolonged periods (i.e., years) to demonstrate that fusion is ready to be carried into the commercial stage. A substantial extrapolation toward this level of availability from the few percent that will characterize the next generation of large tokamaks is an important goal for FED/INTOR.

These considerations lead to the following objectives for FED/INTOR:

1. FED/INTOR should be the maximum reasonable physics step beyond the next generation of large tokamaks (TFTR, JET, JT-60, and T-15).

2. FED/INTOR should demonstrate the reactor-relevant mode of plasma operation required for DEMO [ignited or high-Q plasma, controlled burn of >100 s, reactor-level heat and particle fluxes ($P_n \simeq 1\ \mathrm{MW/m^2}$), optimized plasma parameters, and duty cycle ≥70%].

3. FED/INTOR should integrate reactor-prototypical components into a fusion reactor system.

4. FED/INTOR should provide a test of component reliability and system availability with an availability goal of 25 to 50%.

5. FED/INTOR should provide a test facility for plasma engineering (e.g., noninductive current drive), blanket (e.g., tritium recovery), and materials technology development.

6. FED/INTOR should provide a test of fusion reactor maintainability.

7. FED/INTOR should provide a test of high-grade heat and electricity [~5- to 10-MW(electric)] production capability and simultaneously of tritium production and recovery.

8. FED/INTOR should provide a test of those design features that affect the safety and environmental acceptability of fusion.

These objectives will be achieved in stages, as depicted in Table IX. Stage I will be devoted to engineering component checkout and to learning to operate with an optimized D-T plasma. Objectives 1, 2, and 3 will be achieved during this stage. Stage II will be devoted to engineering testing, with emphasis on flexibility in exchange of test modules. Stage III will emphasize high-availability fluence accumulation and reliability testing for materials testing. Objectives 4 through 8 will be achieved during Stages II and III.

XI. FED/INTOR DESIGN CONCEPT

The principal parameters that characterize the FED/INTOR design concept are given in Table X and an elevation view illustrates the more pronounced features in Fig. 3.

TABLE IX

Staged Operation Schedule

Stage	Number of Years	Emphasis	Availability (%)	Annual 14-MeV Neutron Fluence (MW·yr/m^2)[a]	Annual Tritium Consumption (kg)
IA	1	Hydrogen plasma operation engineering check-out	10%	---	---
IB	2	D-T plasma operation	15%	0.16	3.6
II	4	Engineering testing	25%	0.31	6.9
III	b	Upgraded engineering testing	50%	0.62	13.8

[a]At the outboard location of the test modules.
[b]FED/INTOR is designed to achieve 3 MW·yr/m^2 with high probability and to achieve higher fluence levels with reduced probability. A nominal value of 6.6 MW·yr/m^2 is used for a reference design guideline. This fluence will be accumulated during Stage III at a rate depending on the achievable availability. The case given corresponds to the maximum rate and would accumulate 6.6 MW·yr/m^2 after 8 yr of Stage III operation.

The plasma parameters are supported by detailed analyses in conjunction with experimental data. The FED/INTOR plasma, carrying a current in excess of 6 MA, should achieve ignition based on an extrapolation of recent experimental data for auxiliary-heated plasmas, provided that energy confinement scales at least linearly with plasma radius and $\langle\beta\rangle >$ 4% is achieved. This last condition has already been achieved, which also supports the choice of $\langle\beta\rangle = 5.6\%$ for FED/INTOR operation and the resulting values

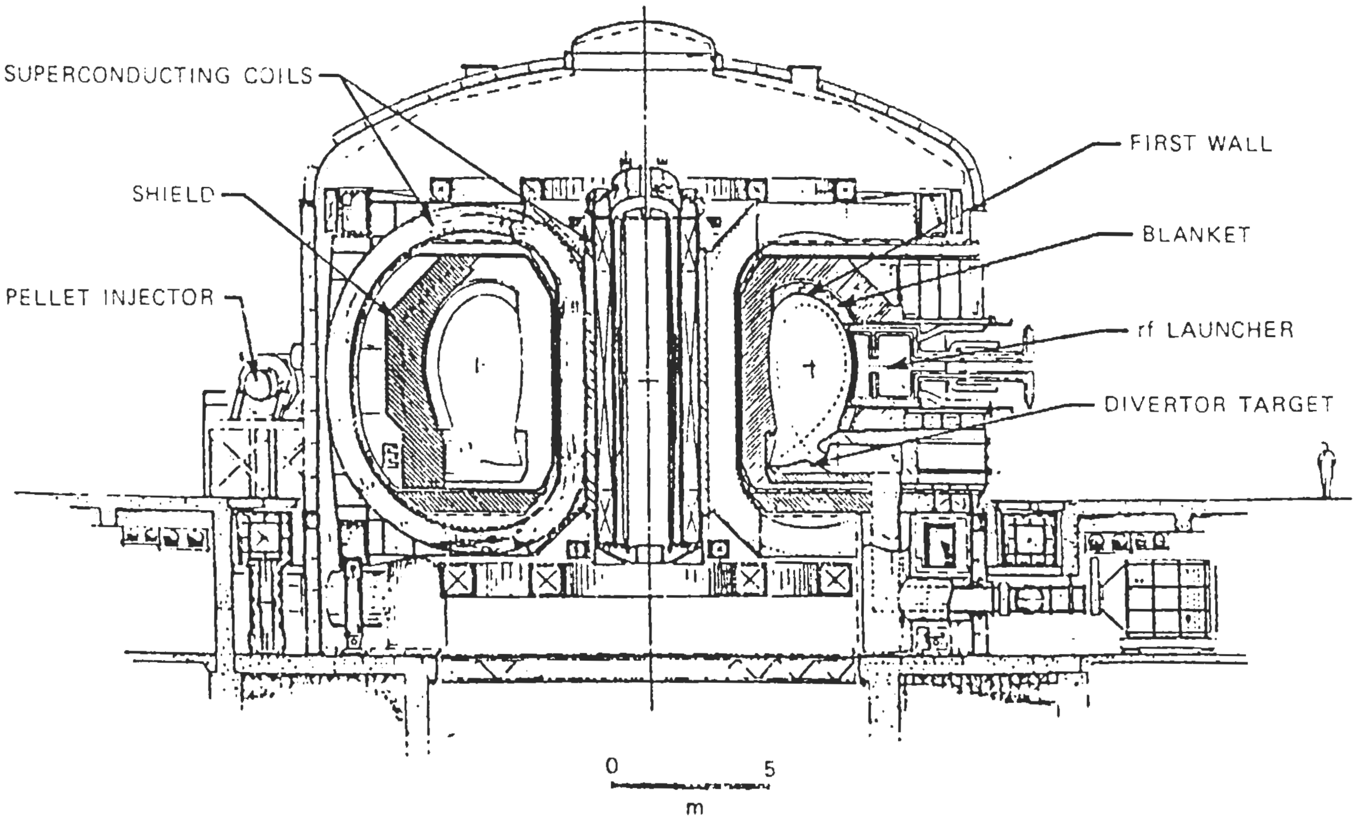

Fig. 3. FED/INTOR elevation view.

TABLE X

FED/INTOR Design Specifications

Geometry	
Chamber major radius, R (m)	5.2
Chamber volume (m^3)	320
Chamber surface area (m^2)	380
Plasma	
Plasma radius, a (m)	1.2
Plasma elongation, κ	1.6
Plasma aspect ratio, A	4.4
Burn average beta, $\langle\beta\rangle$ (%)	5.6
Poloidal beta, β_I	2.6
Average ion temperature, $\langle T_i\rangle$ (keV)	10
Average ion density, $\langle n_i\rangle$ (m^{-3})	1.4×10^{20}
Energy confinement time, τ_E (s)	1.4
Plasma current, I_p (MA)	6.4
Field on chamber axis, B_T (T)	5.5
Safety factor (separatrix), q_I	2.1
Thermonuclear power, P_{th} (MW)	620
Neutron wall load, P_n (MW/m^2)	1.3
Operation	
Burn time, Stage I/Stages II and III (s)	100/200
Duty cycle, Stage I/Stages II and III (%)	70/80
Number of pulses (nominal lifetime)	7×10^5
Maximum availability goal (%)	50
Heating—ICRF	
Mode	Second harmonic D
Number of launchers (active/spare)	3/1
Power at startup (MW)	50
Frequency (MHz)	85
Pulse-length capability	*cw*
Fueling	
Method	Pellet injection and gas puffing
Impurity control	
Method	Single-null poloidal divertor or pumped limiter
Collector or limiter	Beryllium bonded on either vanadium or copper alloy
Power	
To divertor (MW)	80
To limiter (MW)	84
First wall	
Power to first wall (excluding neutrons) (MW)	44
Outboard: Material	H_2O-cooled Type 316 stainless steel
Thickness (mm)	11.7
Inboard: Material	H_2O-cooled Type 316 stainless steel
Thickness (mm)	13.5
Lifetime: (yr)	15 (full)
Breeding blanket	
Material	H_2O, Type 316 stainless steel Li_2O, lead, carbon
Breeder temperature (°C)	400 to 600
Thickness (m)	0.5
Location	Outboard and top
Breeding ratio	0.65
Tritium extraction	Continuous helium purge

(Continued)

TABLE X (Continued)

Tritium fuel system	
Tritium flow rate (g/h)	64
Annual tritium consumption at 25% availability (kg/yr)	7
Isotopic enrichment	Cryogenic distillation
Tritium inventory	
First wall and limiter/divertor (kg)	0.1 to 1.0
Breeding blanket (kg)	0.5 to 1.0
Storage (kg)	2.3
Plasma reprocessing system (kg)	0.2
Pumps, fueling, and other systems (kg)	0.4
Torus vacuum system	
Initial base pressure (Torr)	10^{-7}
Preshot base pressure (Torr)	3×10^{-5}
Pumps	Compound cryopumps
Pumping	Through divertor or limiter chamber
TFCs	
Number	12
Bore (m)	6.6×9.3
Conductor	Nb_3Sn,Nb-Ti
Stabilizer	Copper
Maximum field (T)	~11
PFCs	
Total flux (V·s)	110
Location	External to TFCs
Conductor	Nb-Ti
Maximum allowable field (T)	8
Breakdown coils	
Breakdown voltage (V)	35
Location	External to TFCs
Conductor	Copper
Power supplies	
Stationary loads (MW)	200
Pulsed-energy storage (GJ)	14
Shielding	
Inboard (nonbreeding blanket and shield) (m)	0.80
Outboard (breeding blanket and shield) (m)	1.65

of fusion power and neutron wall load. The 50 MW of ICRF power provides some margin over the minimum that is predicted to be required to heat the plasma to ignition. The ICRF heating system is illustrated in Fig. 4.

The shield and the TFC structure are designed for a nominal lifetime of 7×10^5 pulses, or 6.6 $MW \cdot yr/m^2$ first-wall neutron fluence, which provides a factor of ~2 margin for achieving the minimum lifetime fluence of 3 $MW \cdot yr/m^2$.

The 200-s burn time requires an impurity control system, which is provided by either a single-null poloidal divertor or a pumped limiter located at the bottom of the plasma chamber. Both methods of impurity control appear to have adequate capability for removing helium and the other impurities from the plasma, and both methods can be incorporated in the same location and volume of the reactor. Therefore, the overall reactor configuration is essentially independent of the concept choice for impurity control. The divertor is believed to have advantages in impurity control and helium pumping, but it adds magnetic complexity to the reactor. At this time, uncertainties in scrape-off conditions, erosion by physical sputtering, disruptions, and arcing preclude making a selection of the most acceptable impurity control system, and the two concepts are considered to have equal potential. The design for both the divertor collector plates and the limiter blades consists of a duplex structure with beryllium bonded to either a copper or vanadium heat sink. In the case of the limiter, tantalum is used as the

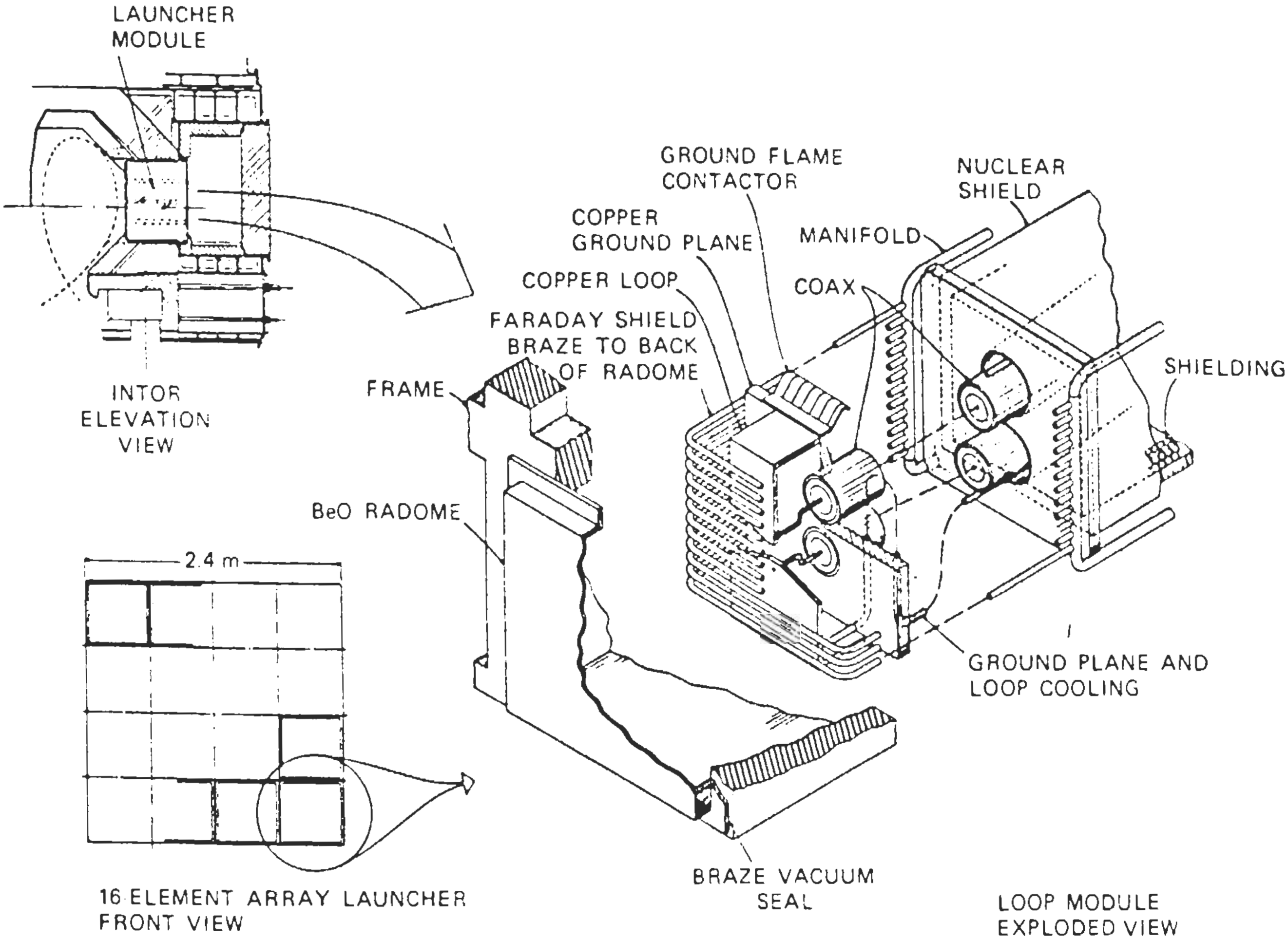

Fig. 4. Ion cyclotron heating system.

surface material at the leading edge to reduce erosion at that location. The estimated lifetime of these high-heat and particle flux components is ~2 yr for Stage III operation.

The mechanical configuration design was driven from the outset by the requirement to provide adequate access in order to facilitate maintenance and assembly/disassembly. A combined vacuum boundary between the superconducting magnet system and the torus plasma chamber has been incorporated into the configuration. The outside portion of the combined vacuum boundary provides a common vacuum region, which contains all the superconducting TFCs and PFCs, except the lower outboard PFC, which is contained in a separate vacuum boundary to allow vertical access for removal of this coil. The combined vacuum boundary allows the coils to be reduced by ~15% relative to the Phase I INTOR and FED designs. Resistive coils external to the TFCs will be used to provide the voltage pulse for plasma breakdown. Other resistive coils will be used for active vertical and radial position control for the plasma. These coils may be segmented and internal to the TFCs for ease of assembly and maintenance. They will respond faster than the main EFCs, but will have limited power requirements because rapid plasma excursions will be stabilized by passive elements.

Both forced-flow and pool-boiling conductor designs have been developed for the TFCs and PFCs, and, in addition, a superfluid pool-boiling conductor design using Nb-Ti has been developed for the TFCs. Each of these conductor concepts is under active development, and a final decision can await results from the development programs.

The rather demanding structural requirements for the TFCs are met by a combination of design strategies. Coil wedging, intercoil support structure, and a bucking cylinder will be used to handle in-plane and centering forces. Gussets, intercoil support structure, a ring girder, the bucking cylinder, and shear ties will be used to handle out-of-plane forces and the overturning moment. A built-up laminated structure will be used.

Twelve torus sectors fit within the internal region of the combined vacuum boundary. These torus sectors are partially (outboard and upper) tritium-producing blanket and partially (inboard and lower) heat removal shield. The final closure of the vacuum boundary on the plasma side occurs at the interface between the outboard surface shield module and the vacuum structure, which enclose the TFC outer leg. Each torus sector can be withdrawn horizontally with straightline motion through a window between adjacent TFCs. The impurity control chamber has been designed to accommodate either poloidal divertor or pumped-limiter modules. Twelve divertor or 24 limiter modules can be removed through the impurity control channel independent of the torus and peripheral components. The torus segmentation concept is shown in Fig. 5.

Extensive analysis supports the design of the first wall, blanket, and shield. A water-cooled, stainless-steel first wall with a panel-type construction is specified. This first wall is expected to last the full lifetime of the device. A tritium-producing blanket will be installed from the outset of operation to reduce the operational cost. A solid breeder (Li_2O) blanket that covers the outboard and upper surfaces of the plasma chamber can produce more than 60% of the tritium consumed in FED/INTOR. Adequate shielding is available for component protection and to allow access 24 h after shutdown.

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

XII. CONCLUSIONS

The critical issues studies that have been carried out during 1981-1982 have clarified our understanding of certain major technical issues that affect the feasibility, cost, and engineering design tractability of a next-generation tokamak reactor and have advanced our knowledge of how to design such a device. Some of these intensive studies have been carried to a point where further significant progress must await additional experimental information. Specific R&D recommendations have been formulated to this end. In other areas, such as impurity control, a continuation of the intensive study is warranted.

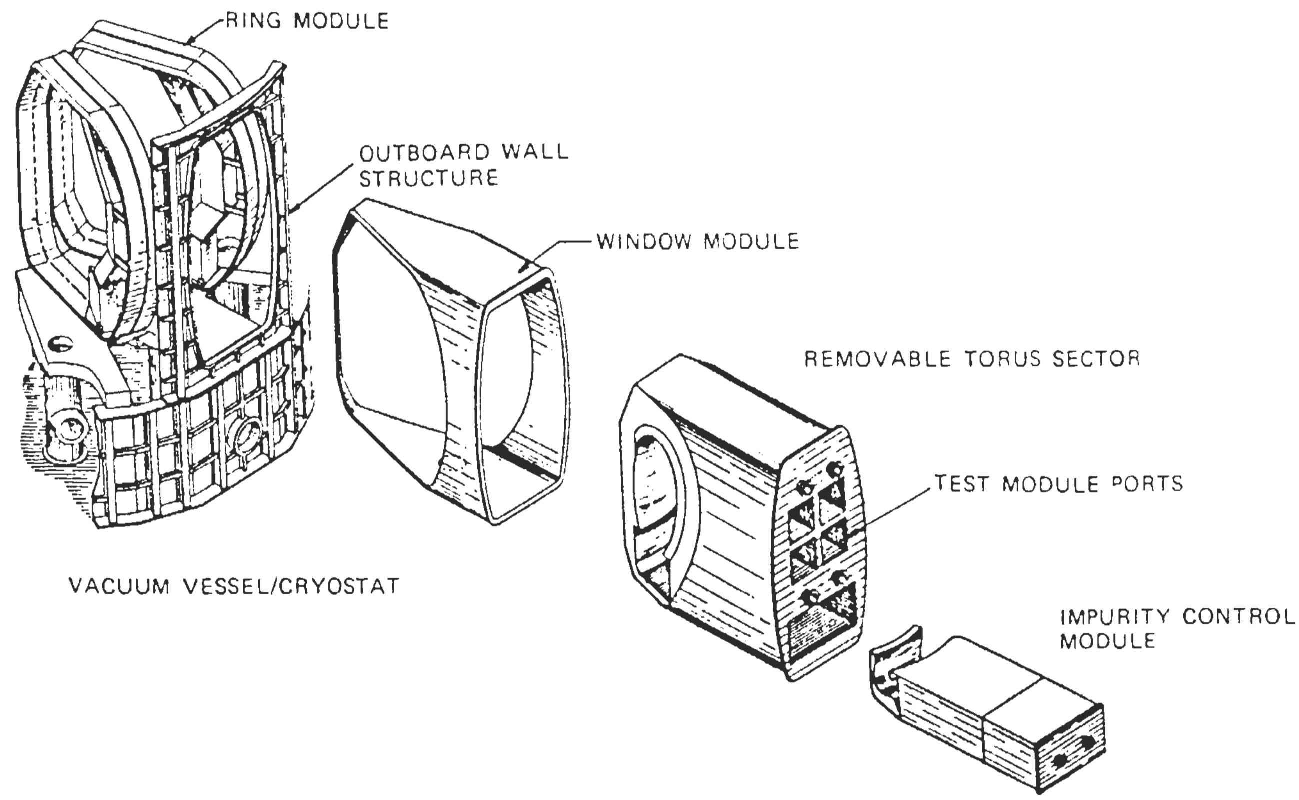

Fig. 5. FED/INTOR torus segmentation concept.

In addition, several new areas requiring intensive study were identified.

The results from the critical issues studies have led to improvements in several aspects of the FED and INTOR design concepts. The improved design concept provides a sufficient basis for the immediate implementation of an engineering design of a next-generation tokamak reactor (i.e., ETR in the United States and INTOR internationally). This concept will naturally become defined in greater detail and perhaps evolve in some particulars as new information from future studies and the R&D programs becomes available.

XIII. RECOMMENDATION FOR FUTURE WORK

Four technical issues have been identified to which an intensive multidisciplinary effort should be devoted:

1. The integrated physics and engineering study of the impurity control system should continue.
2. An integrated physics, engineering, nuclear, and magnetic study of the implications of electromagnetics requirements on torus design, tritium breeding potential, and PF system design should be performed.
3. An integrated physics, nuclear, and engineering study of the technological requirements and design implications of rf heating, current drive, and startup assist should be performed.
4. A comprehensive study of tritium containment, radiation shielding, and other factors that affect personnel access for maintenance should be performed and contrasted with the technological requirements for almost entirely remote maintenance.

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

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