Alternatives for Addressing the ITER Mission

I ITER Background

At the time of the start of the ITER partnership in the late 1980's, each of the four ITER Partners were considering major next steps in fusion research which would follow the then-current generation of large tokamaks: JET, JT-60, T-15 and TFTR. In particular, the European NET and the Japanese FER conceptual design studies were particularly well advanced and were seen as providing the data and experience bases necessary to design a demonstration reactor (DEMO). With all Parties recognizing that some form of major next step was required, it was both natural and desirable that they joined together in the ITER partnership, to share cost and to accomplish more than one Party could alone.

As the ITER design proceeded, two policy considerations prevailed. The first was that ITER must provide the burning-plasma physics, the steady-state operation and the fusion technology data bases which would be required for DEMO. This meant that, in addition to requiring substantial operation with full deuterium-tritium fuel, ITER must provide operational experience with the superconducting magnets, tritium-breeding blankets, thermal power handling, and other technologies essential for DEMO.

The second consideration was that ITER must not fail to reach its operational objectives. The tokamak consequently needed to be sized to achieve those objectives under the operating conditions that had been best documented in the world tokamak program.

The application of these two policies resulted in the current EDA design. Nonetheless, midway in the EDA it was noted that ITER as designed would also be able to operate in one or more of the "advanced tokamak" operating modes which, although less completely documented, have been observed in most of the world's tokamaks (albeit only transiently) and which promise more attractive cost-of-electricity for a tokamak power plant.

The EDA design has been reviewed by all of the Parties and has been judged to meet the requirements set for it. However, the projected capital cost for construction, estimated at $7.15 billion (1997$), or ~$10 billion with inclusion of management, R&D during construction and contingency (as practiced in the US), appears beyond the willingness of the Parties to finance in the prevailing international circumstances. As a consequence, there has been initiated a study, SWG Task 1, to determine whether and how ITER's cost could be reduced significantly (with a target of 50%) while still meeting its overall programmatic objective, namely "to demonstrate the scientific and technological feasibility of fusion energy for peaceful purposes".

Several approaches to achieve this goal are possible. One approach is that leading to the current ITER design.

Another approach retains a single D-T machine having the requisite technologies, e.g., superconducting magnets. It relaxes the targeted operating conditions under the “standard ITER” operating assumptions but retains the capability still to achieve many of the full ITER detailed technical objectives under the advanced tokamak operating conditions which offer so much better performance at the reactor level.

The SWG under Task 2 has also prepared information on broader concepts in support of certain goals of Task 1. An example identified here moves away from a single facility, adopting a "modular strategy" and seeks to acquire the requisite DEMO data from a family of concepts, spread in time and locale. Clearly, this approach would lack integration, but, then, it would offer increased flexibility. Other approaches will be identified also.

Use of advanced materials for the structure and first wall of a reactor will be required for fusion to reach its full environmental potential. The current ITER design, or any foreseeable variation which might ensue, cannot provide a sufficient level of neutron fluence to carry out lifetime tests of the materials and components which will be required for DEMO. Therefore, in parallel with the common view of the required next step, there will be required, in addition, a dedicated high-fluence, point neutron source for the development and qualification of materials and, likely, a volume neutron source for component testing. Such facilities provide other opportunities for international cooperation.

It is often asserted that ITER should constitute a "single step to DEMO." This descriptor, while conceptually attractive, does not characterize the underlying strategy fully. A point neutron source, and likely a volume neutron source (or component test facility), are elements of even the full-ITER strategy. Furthermore, if an alternate concept such as the stellarator would prove superior to the tokamak, an engineering test reactor based on that alternate would certainly be needed prior to a DEMO. Nonetheless it is clear that the "single step to DEMO" philosophy brings the strong advantage that it maximizes the degree of demonstrated integrated testing achieved in the next step. The chief disadvantage of this strategy is that the cost and risk of the single step are both high, making it difficult to finance the device, even at reduced cost, and difficult to assure its success.

The following sections provide background on the present ITER design and the reduced-scale ITER and begins the discussion on broader concepts. The initial focus of the broader concepts is on a modular approach.

II The present ITER EDA Design

II.1 Burning Plasma Physics

The ITER EDA design provides for nominal 21 MA plasma operation at B=5.7 T with a single-null divertor, a major radius of 8.14 m, a minor radius of 2.80 m, an elongation of k_{95} = 1.6 and a triangularity of d_{95}=0.24. The fusion power is 1500 MW with a burn duration of $\geq$1000s.

Projections of ITER's plasma performance show that sustained ignition and adequate plasma power and helium exhaust can be obtained with operation in an ELM_y, saw-tooth mode in the range of 1000 to 1500 MW . There is a ~10% margin in overall plasma energy balance for the most probable performance, although Q-values as low as four are within the uncertainly range of the projections. Experimental results from tokamaks and modeling codes confirm the ITER divertor concept of detached or partially-detached operation with controlled additions of recycled impurities. Under degraded confinement (30% compared to nominal value) and with a density limited to the Greenwald value, ITER would produce 1300 MW of fusion power in a driven mode (with up to 100 MW of additional heating, depending on the confinement) with a current of 24 MA (q ~ 2.6) and 850 MW at 21 MA. For such a driven mode the required b_N is below projected limits and the edge power is above the nominal L to H-mode power threshold.

II.2 Steady-state Advanced Tokamak Physics

The ITER EDA device and balance of plant are designed to provide for steady-state operation. The ITER design has sufficient flexibility to exploit advanced plasma operational scenarios necessary to obtain steady-state operation at the 1000 MW level. With presently-achieved current drive efficiencies, the 100 MW of auxiliary power provided would drive up to about 5 MA. Thus it is necessary to obtain high-bootstrap current, high-beta operation (b_N ~ 3 - 4) with a total plasma current of 12 to 15 MA. Producing ~1000 MW will require enhanced energy confinement. One candidate that has been investigated is reversed- or negative-central shear configuration. Such modes were not judged during the present EDA to be sufficiently demonstrated to use as a design basis for baseline operation. However, because these modes are now under continuing active study in the world's tokamak programs, the EDA design was determined to be flexible enough with respect to plasma shaping to enable exploration and exploitation of the advanced operating space they promise (although in operating in these advanced modes, the current design cannot take full advantage of the chamber cross section). The full potential implications of advanced modes (e.g., upgradability for wall-mode feedback control) were not investigated.

II.3 Plasma Technologies

ITER represents a complete demonstration of all relevant plasma technologies: superconducting magnets for both TF and PF systems can operate up to 13T; there is provision for up to 50 MW each of plasma current/current drive systems from some combinations of negative-ion based neutral beams, electron and ion cyclotron systems and lower hybrid systems (although issues of position control may make this problematic); a complete fuel handling system is provided, including fuel injection (gas puffing and pellet injection), vacuum pumping and fuel processing including tritium handling; and plasma facing components (divertor and first wall) are designed to be capable of handling reactor prototypical heat and particle fluxes with steady-state heat removal. The design for all these components reflects the full range of operational modes including inductive-driven, long-pulse ignition; driven modes with current drive for steady-state operation, and off-normal conditions including disruptions.

II.4 Fusion-nuclear Technologies and Testing

The ITER design provides for all the components necessary for handling up to 1500 MW of fusion power on a steady-state basis, including a first wall/shield/blanket (tritium-breeding capability in the Enhanced Performance Phase only) and heat transport/rejection systems. A complete fuel (tritium) handling and processing system is also provided. Provision is made for full remote maintenance for all in-vessel components as well as replacement of the vacuum vessel and magnets, if ever required. The vacuum vessel serves as the principal, but not only, radiation confinement barrier and is designed to nuclearcode standards. A thorough safety analysis has been carried out to insure that public evacuation will not be required under any credible, even very low probability, accident scenarios.

The choice of the basic coolant (low-temperature water) and structural materials (316 LN SS) of the in-vessel components was based on the need for a well-supported design data base for a next-step device. Such a choice of coolant and structures material is not suitable for power reactors, thus provision is made for large test modules which can incorporate advanced material/coolant tritium breeding design concepts. The ITER design provides for 4 to 5 test modules with a total first wall area of $22 m^2$. The neutron flux is 1.2 MW/m^2 with a fluence goal of about 1 $MW\text{-}a/m^2$.

These values are considered adequate to perform tests on the feasibility of basic performance, integration issues of testing in the actual fusion environment, and some data on radiation effects on performance. These values are not adequate to develop a complete data base on failure modes and reliability nor on the high neutron fluence effects. The former task would likely fall to a volume neutron source and then DEMO; the latter would be the task of the point and/or volume neutron sources, discussed earlier.

II.5 Integration

The ITER design represents the first, detailed effort to confront one of the major feasibility issues for fusion energy systems: the ability to provide sufficient access, clearances and mechanical design features that satisfy at the same time remote maintenance needs, plasma performance requirements, and a robust structural system to handle off-normal events like disruptions. This mandates an integrated approach to magnet system interactions, the number and location of TF coils, PF coil location, size and location of ports, mechanical interfaces for all the in-vessel components and modularization of the divertor and first wall/shield components. A systems approach must be used for remote maintenance, transport systems, repair facilities and general facility layout.

There is an important physics dimension to the integrated capability of the ITER EDA design. For example, ITER offers the opportunity, indeed must achieve it to meet its mission, to study simultaneously plasma-core and edge conditions sufficient to achieve good energy confinement which are compatible with divertor conditions to accommodate the particle and heat fluxes. ITER also provides the opportunity to demonstrate the compatibility of advanced tokamak modes and profile control with plasmas dominated by internal heat production -- a necessary test for tokamak power reactors. The important first steps in this physics integration have been taken by JET and TFTR. ITER would extend their results into the regime of strong self-heating and steady-state operation.

III A Reduced-scale ITER

The target for cost reduction for ITER is 50% of the full EDA cost. Seeking such reductions presents both an increased technical risk, as measured by the possibility of a truly disappointing burning-plasma operation (Q-value) arising from less-than-assumed energy confinement, and an opportunity, as measured by an increased theoretical capability to exploit the advanced operating modes observed in current tokamaks. Several options are under study. Judgment of this approach must, therefore, be less design specific and must focus on the opportunities and trade-offs it presents.

A reduced-scale ITER would be designed to target only sub-ignited conditions under baseline ITER physics-design rules. For example, studies suggest that a device with R_o = 6.0-6.5 m, I_p ~ 13-14 MA. could achieve Q ~10. These designs, if operated at full bore in advanced modes, could then could also ignite with, e.g., with HH ~1.1-1.25 times higher than the baseline case, provided a full complement of auxiliary heating, current drive, and fueling systems were installed. (Although a 50% shortfall in confinement would correspondingly result in Q values much lower, perhaps 2-3.) Further reductions in radius and current would require advanced operation (HH >~ 1.2 and β_N >~ 3.5) even to achieve minimal Q (say >5) conditions.

One of the design requirements for any reduced-scale ITER would be that most of the full ITER detailed technical objectives could be restored under advanced tokamak operation. To the extend that this capability to accommodate performance beyond the baseline entails external systems which could be fitted later, it might not impact the construction costs. However, to the extent that this capability requires a more robust core tokamak capability, e.g., increased cooling in the wall, it could impact the cost.

The major access to advanced modes is through better control of the plasma current and pressure profiles using sophisticated auxiliary systems. The additional three years of the EDA will permit the ongoing experimental program to provide more data in support of optimizing this technique. However, it is important for the reduced-scale designs to retain the maximum access for auxiliary systems, controls and diagnostics to take advantage of new opportunities.

The projected costs savings of reduced-scale ITER designs generally assume higher elongation of 1.7 to 1.8, and a lower helium (and impurity) level. However, it is clear that achieving elongation values in this range will be very difficult without certain poloidal coils being located internal to the toroidal coils. Because of the impact on the reactor attractiveness brought about by strong shaping, it is important that the ellipticity increase to 1.7 to 1.8 and the triangularity to 0.3 to 0.4. Such higher ellipticity and triangularity values would require significant changes from the present ITER design, such as internal coils.

Studies show that it should still be possible to have a multi-hundred second, inductively-driven, flat-top duration even in a smaller machine, because the incremental volt-seconds required over those needed for the current ramp remains a small percentage of the total. One cost reduction option would reduce the shielding and permit a higher level of irradation of the toroidal coils. This approach would lead, perhaps, to a pulse duration of a few hundred seconds under burn conditions and would limit the total fluence capability of the device to the point where a VNS was mandatory before the step to DEMO. However, some reduction in shield thickness could be achieved without prejudicing the operating time.

III.1 Burning plasma physics

Provided the advanced modes work as well as expected (i.e., as seen for short pulses in present experiments), a reduced-scale ITER could have a similar performance to the reference ITER, and therefore it could achieve all of ITER's burning plasma objectives. While this would entail higher technical risk than the reference ITER, the operating mode would have greater power plant relevance, provided they have been achieved in more strongly shaped configurations.

III.2 Steady-state Advanced Tokamak Physics

Whereas the current ITER would explore advanced tokamak operating modes primarily as part of its second, Extended Performance Phase, a reduced scale ITER would exploit these modes to achieve its burning-plasma and neutron-wall loading objectives much earlier in its operation. Pulse extension leading to steady state operation would consequently be more naturally carried out in these operating modes. The design can also be better optimized for AT operation, e.g., by having a larger natural elongation and by allowing such operation to make better use of the full chamber cross section.

III.3 Plasma Technologies

The plasma technology objectives of the present ITER would also be met by a reduced-scale version, e.g., use of superconducting toroidal and poloidal coils, heating and current-drive systems, fueling techniques, disruption control techniques, steady-state heat removal, etc. All would be required to operate in a manner directly relevant to DEMO.

III.4 Fusion-nuclear Technologies and Testing

Both the present and reduced ITERs would require radiation protection, remote handling, tritium handling, etc., and both would impose the same levels of safety and environmental protection. The neutron flux would be reduced from the 1 MW/m^2 required in the present ITER to ~0.5 MW/m^2 in the baseline operation of a reduced ITER. However, one of the design requirements would be that the latter could be restored under advanced tokamak operation. Narrowly speaking, this would require only higher b_N, but improved confinement would also be required to avoid prohibitive associated thermal loads.

ITER's wall loading and fluence testing represents the fusion-nuclear technology objective at greatest risk in a reduced-scale ITER. Achieving 1 MW-a/m^2 would place high demands on availability. The reduced-scale option, however, would have greatly reduced tritium-supply requirements, and probably would not have to breed its own tritium.

III.5 Integration

The technology integration objectives for a reduced-scale ITER should be met as well as in the present ITER, provided the steady-state objectives are met. The physics integration objectives would be even better satisfied in the reduced-scale option, provided the advanced tokamak operating can be achieved and sustained in steady state under conditions of strong self heating.

IV. Broader Concepts - Modular Strategy

A modular strategy would replace the ITER device with a normal conductor D-T burning plasma experiment and a steady-state D-D advanced tokamak experiment. These devices would then certainly be accompanied or followed by a volume neutron source, or component test facility. If it proved that an alternate concept was superior to the tokamak (for example with respect to disruptions, beta, and/or steady-state operation), the volume neutron source might be constructed in the alternate configuration, and serve to bridge the new concept to DEMO. Even in the case where the tokamak is most successful, it has been argued that it might be cost-effective to construct the volume neutron source in a different plasma configuration.

The cost of the modular strategy is difficult to estimate. The normal conductor D-T burning plasma experiment could, in principle, be as small as the Italian Ignitor design, or as large as the US "PCAST" machine, with the US BPX design somewhere in between. The steady-state D-D advanced tokamak might be as inexpensive as the Korean KSTAR device, or as aggressive as the Japanese JT-60SU design, with the US TPX device in between. The summed cost of the BPX plus TPX combination is about $2.5B, or about 1/4 of the full ITER cost. It is hard to assess the cost of a volume neutron source, but it seems reasonable to extrapolate that the full cost of these machines need not be significantly more than half the cost of the present ITER EDA design. The "modular" nature of this strategy would allow expenditures to be made in smaller units, with less risk. Furthermore the potentially largest investment in the high-duty-factor D-T device would be made only after demonstrated success with the other two devices.

The time profile for the modular strategy is certainly longer than for either ITER or the reduced-cost ITER. ITER's time profile is roughly 3 decades. One decade for construction, one for basic performance phase operation, and one for testing in the Enhanced Performance Phase. The modular strategy might complete construction somewhat more quickly (say eight years) due to its factor of ~ two lower initial cost. One decade of operation might be needed for the two international devices. Perhaps after eight years of operation of these devices construction could start on the volume neutron source, and then this device would require eight years to construct and a decade to operate. The total time profile is 3 x 8 + 10 = 34 years, not so far from the ITER timetable.

It may also be judged desirable, perhaps even necessary, to add another step to the modular strategy before DEMO, i.e., an integrated test facility. In this case the overall cost of the modular strategy could even exceed the cost of the present ITER design and would add a decade or more to the date of DEMO. Alternatively, the test of integration might be accomplished at lower cost by dedicating the first phase of the DEMO operation to this task. In either event, the modular strategy offers more flexibility to solve problems, so perhaps it offers a somewhat greater likelihood of achieving its somewhat reduced objectives

IV.1 Burning Plasma Physics

The modular strategy may have one important advantage over the reduced-cost ITER strategy in this area, since a cost-effective normal conductor B-P device could be designed to have as much ignition margin as the full ITER, or even more. So long as the pulse length were substantially greater than the alpha slowing-down time, issues of alpha particle physics could be adequately addressed in this device. With pulse length much greater than energy confinement time, some issues of burn control could also be examined. However the cost for a pulse length much greater than the helium containment time, such as embodied in the PCAST machine, might be prohibitive. It is also difficult to accommodate helium pumping in a compact device. Thus this aspect of burn control and divertor integration most likely could not be addressed in a short-pulse D-T burning plasma experiment. The issue of alpha particle physics and burn control as a function of j(r) would have to be addressed as in present devices, by using the time evolution of the current to control, in a gross sense, the current profile.

Issues of alpha particle physics and burn control in advanced regimes could also be partially addressed in the steady-state D-D advanced tokamak device. Such a device, designed with adequate shielding for D-D operation, can accommodate short-pulse D-T operation, to test the stability of the a population. Some issues of burn control could be tested by adjusting the heating systems to inject power in proportion to n^2<sv>. Helium pumping and exhaust could be tested in this device, but not with full integration of a divertor with a reactor-scale core.

IV.2 Steady-state Advanced Tokamak Physics

The smaller scale of the SS/AT device compared to even a reduced-cost ITER could arguably lead to greater flexibility for studying advanced tokamak physics. For example the TPX device was designed to support both double and single null plasmas, with variable elongation and triangularity, over a very wide range of beta and li. It was also designed for a series of possible upgrades to the heating and current drive systems, and would have accommodated pressure profile control and feedback stabilization systems. A smaller D-D device such as this, with international financial support, would be able to adopt upgrades and modifications more quickly than a D-T ITER. It would be able to investigate in detail the interaction of an advanced tokamak core plasma with successful divertor operation, but the integration would not be near full reactor scale. Such a device would of course lack the ability to study alpha particle heating in advanced configurations, and could only study alpha particle stability by introducing tritium for a short period. NINB and ICRF could be used, however, to investigate fast-particle dynamics in advanced regimes.

IV.3 Plasma Technologies

The modular strategy addresses technologies, as well as physics, in a modular fashion. All of the same technologies that are addressed in ITER would be addressed in the combination of devices of the modular strategy. It could even be argued that the trade-off between risk and integration is well balanced in the modular strategy. For example, the divertor plates in the D-T experiment could be relatively simple, inertially cooled systems. This would greatly reduce the risk of major problems with remote maintenance in the inhospitable D-T environment. The SS/AT device would certainly need to incorporate actively cooled divertors, but the radiation environment could be such that human intervention would be possible for brief periods, after some cool-down. In a similar vein, maintenance of the superconducting coils of the SS/AT would be much easier than in ITER. The issues of tritium retention and disruption survival could be addressed in these devices, without the fundamental success of the devices depending on prior resolution of either of these two areas of concern.

IV.4 Fusion-nuclear Technologies and Testing

Neither the SS/AT nor the short-pulse D-T burning plasma experiment would offer significant nuclear technology testing. Ultimately, however, the volume neutron source, or component test facility, would, at least in principle, provide more vigorous testing of blanket and nuclear components, and of internal high-heat flux components in a D-T environment, than the full ITER. Again the fully integrated demonstration aspects of ITER nuclear component testing would be lacking, especially if the volume neutron source configuration is not the same as that planned for DEMO, but the probability of success in this strategy would arguably be enhanced by qualifying component concepts in one or the other of the first two devices, before subjecting them to full tests in the volume neutron source.

IV.5 Integration

A key question is: What needs to be integrated with what? A high-performance core plasma needs to be integrated with an effective divertor, including high-heat flux steady-state components. The SS/AT device provides great flexibility to address this problem, but not with a full reactor-scale core plasma. Superconducting magnets need to be integrated with a D-T burning plasma eventually, but this is not an area of great uncertainty. It is more critical to demonstrate successful operation of such magnets in a tokamak magnetic environment. It is important to integrate heating and current drive systems with a steady-state burning plasma. Many, but not all, aspects of this will be addressed in the first two devices, and the volume neutron source should address it fully. Remote maintenance must be integrated with a working fusion plasma system. This issue would be addressed in pieces in lower-risk steps in the first two devices, and then more fully in the volume neutron source.

Perhaps the best test of integration in this strategy is the degree of confidence that success would provide in moving to DEMO. It appears that success with a SS/AT, a normal conductor burning plasma experiment, and a volume neutron source, would give less confidence in the step to DEMO than the full-scale ITER . This could be overcome by adding another step before DEMO.

Another way to compare the "single-step to DEMO" strategy with the modular strategy, is to consider that the modular approach focuses more on research and development, while the single-step strategy focuses more on demonstration. Thus success of the modular strategy is more assured, but the ultimate potential of the single-step strategy is greater.

V. Conclusions

The full ITER, a reduced cost ITER, and a modular strategy, represents credible progress towards the development of fusion power. The initial investment cost for the full ITER has proven to be too high for the international fusion community. If the reduced-cost ITER can reach its target of 50% cost reduction, while maintaining Q=10 capability in ELMy H-mode and advanced-mode operation in full-bore plasmas, then it is a very attractive alternative.

Since this is not assured, however, and since the international funding even for a half-price ITER is not assured, it is essential that the international ITER process examine carefully a modular strategy, in parallel with the investigation of the reduced-cost ITER option.