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REPORT OF THE ITER SPECIAL WORKING GROUP ON TASK #2

by Dr. H. Kishimoto and Prof. K. Pinkau, SWG Co-Chairs

A Special Working Group (SWG) was established by the ITER Council (in accordance with Article 10 of the ITER EDA Agreement) to carry out the two following tasks:

1. **The SWG would propose technical guidelines for possible changes to the current detailed technical objectives and overall technical margins, with a view to establishing option(s) of minimum cost still satisfying the overall programmatic objective of the ITER EDA Agreement.**
2. **Pursuant to Art. 2(e), the SWG would also provide information on broader concepts as basis for its rationale for proposed guidelines, and articulate likely impacts on the development path toward fusion energy.**

In accordance with its charters, the SWG reported on Task #1 to the Council in June 1998 and the Council accepted that report at its First Extraordinary Meeting.

On Task #2 the SWG completed its Report to the Council on 30 January 1999. The conclusions contained in this Report are reproduced in full in the box overleaf. The rationale for these conclusions is summarized on the basis of the SWG Report as follows:

ITER:

The ITER Engineering Design Activities (EDA), conducted by the four Parties from 1992-98, led to the designing of a tokamak-based integrated step in fusion R&D towards a demonstration power plant (DEMO). The DEMO would demonstrate quasi-stationary generation of a significant amount of net electrical power. The overall programmatic objective of ITER, which has guided the EDA, is “to demonstrate the scientific and technological feasibility of fusion energy for peaceful purposes.” ITER, as described in the ITER Final Design Report (FDR), would accomplish this objective by demonstrating controlled ignition and extended burn of deuterium-tritium plasmas, with steady-state as an ultimate goal, by demonstrating technologies essential to a power plant in an integrated system, and by performing integrated testing of the high heat flux and nuclear components required to utilize fusion energy for practical purposes.

Concept Development:

The common challenge for all the magnetic fusion concepts is the simultaneous optimization of stability, confinement and power and particle exhaust, in order to increase the atttractiveness of the power plant concept (for example, reducing the magnetic field, plasma current, and peak exhaust heat flux densities, providing easier maintenance, reducing demands on auxiliaries, achieving steady-state operation and mitigating or eliminating disruptions). These efforts include continued optimization of the tokamak performance, as well as the development of other lines such as the helical systems (stellarators, heliotrons, etc.), the reverse-field pinch, compact toroids (spheromak and field-reversed configurations), and the spherical torus/tokamak.

Not only is there promise, at varying degrees, in these different conceptual approaches, but, in addition, they contribute to the improvement and understanding of magnetic fusion systems in general. Furthermore, the requisite technologies are largely common to the different approaches. Such studies are an important part of fusion research and are being developed in parallel with the ITER EDA program and the long-term technology programs of the ITER Parties.

SWG CONCLUSIONS ON TASK #2

The successful development of fusion energy requires meeting the basic challenges of scientific and technological feasibility as well as environmental attractiveness and economic viability. The long-term goal is a convincing demonstration of the resolution of these issues in a demonstration power plant (DEMO). A critical prior step is to integrate high energy gain plasmas at or near steady-state conditions with power-plant prototypical technologies, and demonstrate safe operation of a fusion power system.

The international program is technically ready to proceed with the construction of an experimental facility which in an integrated manner addresses scientific and technological issues before DEMO. Many of these issues can be addressed only in near-power-plant conditions. ITER will provide the conditions required for these critical tests, and it has focused the attention of the world fusion community on key scientific and technical issues. Through this collaboration, the cost and benefits of this important step can be shared by the ITER Parties.

Because of concerns of cost, but coupled with advances in physics and technology made during the ITER EDA, there is now both increased incentive and opportunity to seek an attractive lower-cost design by modifying the detailed technical objectives. A device, in which it is expected to achieve energy gain of at least 10 and explore steady-state operation, at a direct capital cost of approximately 50% of ITER as described in the Final Design Report, would still satisfy the ITER overall programmatic objective, which is "to demonstrate the scientific and technological feasibility of fusion energy for peaceful purposes."

Successful operation of ITER would provide experience of broad, generic value to fusion energy development:

- **the study and control of burning, steady-state plasmas;**
- **the study of the interaction of such plasmas with material walls, together with the removal of thermal energy and helium ash;**
- **the development and performance testing of blankets capable of generating tritium and high-grade heat compatible with efficient electricity generation;**
- **the demonstration of required supporting technologies, and**
- **the demonstration of the safety potential of fusion.**

The SWG has examined a strategy in which the integration of ITER's long-pulse/burning-plasma scientific and technological objectives, which are essential before moving to DEMO, would be deferred until after experimentation has been completed on a new generation of separate specialized facilities addressing selected critical issues. The SWG concludes that this would delay by 10 years or more the key fusion demonstration and integration step, and would increase the total cost of fusion development substantially. It is the unanimous opinion of the SWG that the world program is scientifically and technically ready to take the important ITER step

Technology:

Technology and materials R&D play an essential and multi-faceted role in the accomplishment of the following main challenges in the process of development of a commercial fusion power plant based on the principles of magnetic confinement:

- demonstration of scientific feasibility by developing a stationary burning core having high fusion power gain,
- demonstration of technological feasibility by developing required components and integrating them with the core,
- demonstration of the safety and environmental attractiveness of fusion, and
- demonstration of the economical viability of fusion.

Progress in fusion science has depended on the development of the enabling hardware and methods to create, sustain and control high-temperature plasmas (e.g., magnets, heating and fuelling systems, vacuum technology, etc.). A particular challenge continues to be the development of plasma-facing components to withstand high heat and particle fluxes, as well as off-normal events such as disruptions. In the longer term, technology R&D aims to develop materials and components that will achieve the desired levels of performance (e.g., high temperatures and wall loadings), lifetimes, availability (sufficient component reliability and acceptable change-out times), and safety and environmental attractiveness, with emphasis on in-vessel systems. Such technology R&D, as well as physics R&D, often also has important near-term industrial spin-off applications to fields outside of fusion.

Particularly important is the development of radiation-resistant and low-activation materials, a key part of which is producing data on neutron irradiation effects. In the near-term the program depends primarily on fission reactor irradiation. Reduced-activation advanced materials could be incorporated into the ITER blanket/shield components in later phases. However, specialized facilities for testing materials subjected to 14 MeV neutrons will be required.

ITER approach:

In line with the detailed technical objectives set for the EDA, under the overall programmatic objectives, a fully-documented detailed design of ITER was produced on schedule, with its associated safety analyses and description of manufacturing processes and their associated costs. The Final Design Report (FDR) of the EDA, "ITER Final Design Report, Cost Review and Safety Analysis" was reviewed by the ITER Technical Advisory Committee and by the ITER Parties individually, with a strong input from industry, and approved by the ITER Council. The FDR capital cost estimate was within the range foreseen at the beginning of the EDA by the ITER Council. It was concluded that the ITER machine would fulfil the overall programmatic and the detailed technical objectives. This conclusion is supported by the results of the technology R&D activities conducted during the EDA. Most of these activities have been completed, with some validation tests still to be carried out. Furthermore, the physics program in experiments, theory, and modelling, conducted by the ITER Parties in parallel with the design work, has increased the understanding of the constraints set by plasma physics – confinement, pressure and density limits, helium removal, disruption effects, and divertor operation.

ITER represents a demonstration of fusion technologies under power-plant-relevant conditions: superconducting magnets, additional heating, fuel handling and vacuum pumping systems, plasma-facing components (divertor and first wall) designed to be capable of handling heat and particle fluxes in the power-plant range with steady-state heat removal.

There is an important physics dimension to the integrated capability of ITER, namely the opportunity to optimize, 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 explore the compatibility of improved tokamak modes and profile control in regimes of strong self-heating and steady-state operation – a test for tokamak power plants.

Despite the general achievements presented in the FDR, it appears prudent in the present socio-economic situation to be in a position to offer lower-cost options to enable an effective start of possible future ITER construction. Recognizing this situation, the ITER Council assigned to the SWG the task to "propose technical guidelines for possible changes to the current detailed technical objectives and overall technical margins, with a view to establishing option(s) of minimum cost while still satisfying the overall programmatic objective of the ITER EDA Agreement." Guidelines for a modified-objectives (reduced-cost) ITER were given in the SWG report on this Task of 19 May 1998.

These guidelines would shift the focus of detailed technical objectives from achieving ignition to achieving high fusion-energy gain, without precluding the possibility of achieving ignition. Because progress has been made during the EDA in design, technology development and physics, it appears possible to design a reduced-cost option (further referred to as ITER-II) with attractive performance characteristics. From the preliminary results of the analysis made by the JCT and the Parties, it appears that a machine at approximately half of the direct capital cost of the ITER-FDR machine (further referred to as ITER-I) could satisfy ITER's overall programmatic objective, although with modified fusion objectives, while ensuring that the engineering margins remain such that safety and performance of the device are not impaired.

The table overleaf shows a comparison of some ITER-I reference parameters with corresponding ranges of representative parameters of ITER-II options, which have been developed by the ITER JCT taking advantage of the engineering and technology developed during the EDA.

While additional theoretical and experimental work is required in some areas, projections of ITER's plasma performance show that sustained burn ($Q = 10 \Rightarrow \infty$), and adequate plasma power and helium exhaust can be obtained with operation in a plasma subject to edge-localized modes (ELMs) and internal sawtooth activity. Experimental results from tokamaks and modelling codes confirm the ITER divertor concept of detached or partially detached operation with controlled additions of recycled impurities. Furthermore, due to its better plasma-shaping capabilities compared to ITER-I, ITER-II still does not preclude the possibility of reaching ignition. For a power plant, $Q \geq 20$ is sufficient.

All ITER designs have sufficient flexibility provided by the poloidal field coils and heating and current drive systems to exploit plasma operational scenarios necessary to obtain steady-state operation at $Q = 5$.

Parameter	ITER-I	ITER-II
Major radius (m)	8.14	6.0 - 6.5
Plasma current (MA)	21	13 - 17
Q (=$P_{fusion}/P_{heating}$) (reference plasma)	$Q \Rightarrow \infty$	$Q \geq 10$
Q (=$P_{fusion}/P_{heating}$) (steady-state)	≥ 5	≥ 5
Neutron wall flux (MW/m^2)	1.0	≥ 0.5
Neutron fluence ($MW.a/m^2$)	1.0	≥ 0.3
Fusion power (MW)	1,500	500 - 700
Inductive flat top (s)	1,000	300 - 500

Broader considerations within the mainline Tokamak Program:
The SWG was asked by the ITER Council to "…..also provide information on broader concepts as basis for its rationale for proposed guidelines, and articulate likely impacts on the development path towards fusion energy." Keeping in mind the fusion development described above, the SWG has restricted its attention to the next major steps in the mainline tokamak program.

The overall programmatic objective of ITER is "to demonstrate the scientific and technological feasibility of fusion energy for peaceful purposes", in order to proceed to a subsequent demonstration power plant, DEMO. As discussed earlier, such an accomplishment requires the integration of many scientific and technological features under power-plant-like conditions, and an integration step such as ITER would inevitably be required prior to DEMO.

There are many physics and physics-technology issues to be addressed in advance of DEMO. Examples of such issues include:

- steady-state burning plasma with current-driven profile control and a high bootstrap current fraction,
- a high-performance core with an effective divertor, including high heat-flux steady-state components,
- superconducting magnets with a high power DT burning plasma,
- remote maintenance in a full-scale fusion power system,
- testing of tritium-producing blankets and of structural materials, and
- tritium processing.

Two strategies may be considered to accomplish these objectives. One is the ITER strategy. The other strategy is to delay the integration step and embark, in the near term, on separate specialized facilities addressing selected critical issues. These facilities would be of the same range of size and cost as the present largest experimental devices ($1B to $2B). Within this latter strategy, two classes of such facilities have been considered:

- short-pulse, copper magnet, DT-burning plasma experiments, and
- long-pulse, superconducting magnet, DD steady-state experiments.

For either strategy, a 14 MeV neutron source for materials development is likely needed in parallel.
A burning plasma experiment would provide information on confinement and stability of high-performance DT plasmas. A superconducting DD experiment would provide information on steady-state operation of a diverted tokamak plasma. All three facilities, including the 14 MeV neutron source, would provide valuable experience with fusion materials and technologies.

However, while the first two of these facilities would be designed to address some of the same important plasma science issues as are to be addressed in ITER and could make important contributions, they would do so in conditions falling far short of those in ITER in several important respects:

- they operate at either much reduced fusion power conditions or much reduced pulse length,

- by addressing issues in separate facilities, they fail to address key issues of physics-physics and of physics-technology integration,
- by focusing on plasma science objectives, they do not address the full range of fusion technology objectives of ITER, a prime example being ITER's capability of testing operational blanket modules.

Clearly, the important class of physics performance issues associated with burning plasmas in full non-inductive steady-state operation could not be addressed. The full non-linear interplay between alpha-particle heating, confinement barriers and pressure and current profile control, and their compatibility with a divertor, can only be addressed in an integrated step.

A key question effectively asked of the SWG is whether the combination of specialized facilities under consideration could replace ITER. Given the arguments presented above, the answer is that they could not.

Furthermore, if such facilities were constructed in the place of ITER, and the construction of the integration step were to await results from these facilities, the integration step would presumably be improved, but would be delayed very substantially, perhaps ten years or more, and the total cost of the program would be much increased. The impact in cost and schedule of unnecessarily delaying this integrated demonstration could be devastating to the international effort to develop fusion power, and to the ability of fusion to contribute to the world energy economy in a timely fashion.

It is an important conclusion that fusion development is now scientifically and technically ready to take a step such as ITER, i.e., to enter the regime of fusion energy demonstration.

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STATUS OF THE ITER EDA
by Dr. R. Aymar, ITER Director

This article summarises progress made in the ITER Engineering Design Activities in the period between the ITER Meeting in Yokohama (20-21 October 1998) and February 1999.

The three main focusses of technical activity have been as follows:

- in Design work, the JCT and Home Teams have aimed at establishing design options which meet the revised guidelines for cost and performance targets (EIC ROD 3.1.a). Results of the work to date were discussed at a point design meeting in Garching, end of January (see article on this Meeting, Newsletter Vol. 8, No. 1, January 1999). The report "Study of Options for the Reduced Technical Objectives/Reduced Cost ITER" was presented to TAC before submission to ITER Council;
- in R&D, the JCT and Home Teams have worked to bring the remaining projects to completion - in particular, to complete the model coils and prepare for their testing. Highlights from this work are briefly summarised below;
- the ITER Physics Basis document has undergone further review and revision and is now in the final stages of refereeing and response in preparation for publication in a special edition of “Nuclear Fusion” expected around the middle of 1999. The preparation of this important document has been a major task undertaken within the voluntary framework of ITER Physics. The authors and editors of the various sections of this document are to be commended, and their supporting institutions thanked, for the quality and commitment of effort provided.

The project has worked against a backdrop dominated by the decision by US Congress in the 1999 budget to suspend further funding on ITER and consequential withdrawal of US members from the JCT, close-out of the San Diego JWS, and run-down or re-orientation of other aspects of US participation in the ITER process.

The withdrawal of US members from the Joint Central Team was completed over the period November/December 1998, with the exception of one team member in San Diego and one in Garching, whose assignments were extended for a few months to permit critical contributions to design studies and to complete major R&D. In addition to the withdrawal of US team members, there has been a fall in the number of the Japanese Team members on site because the normal process of rotation and replacement has been interrupted pending completion of an arrangement concerning continuation of ITER activities. Similarly, Russian Team members who left the project in July 98 have not yet been replaced.

In line with the conclusions of the three Parties’ discussions noted in Yokohama, in November, the Director proposed to the three Parties a redistribution of their JCT members between the Garching and Naka Joint Work Sites, and requested the Parties to take the necessary actions to effect the transfers as soon as practicable. The Parties, with the co-operation of JCT members concerned, have made considerable progress towards meeting the goal of a prompt and coherent re-settlement of the staff from San Diego taking effect at around 1 March 1999.

The status of the Team on site as at mid February by Joint Work Site and by Party is summarised in the Table below:

Status of JCT Staff at Mid February 1999

Garching	Naka	San Diego		EU	JA	RF	US		Total
36	32	28		46	27	21	2		96

The cumulative Professional Person Year (PPY) effort on site between 21 July 1998 and 21 February 1999 is estimated at about 64 PPY.

The departure of JCT Members has led to different consequences among the Divisions and Groups. However all Divisions (Safety, Engineering, Integration, Physics Unit) in San Diego have seen their membership decrease so much that a modified structure of JCT has to be proposed and a much larger work contribution needs to be offered by the Parties to keep the previous missions.

Final data is now available on the distribution of tasks committed during the original duration of the EDA. Work has been completed and final reports submitted for 641 out of 681 technology R&D task agreements. It is also possible to finalize the total credits of more than 240 work orders issued under the Comprehensive Task Agreements for Design. These total amounts for R&D in IUA and PPY for Design are well below the planned respective numbers at the beginning of EDA. The Table below summarizes the pattern of assignment to Parties.

Summary of ITER Task allocations to July and post July 1998

	Until July 1998		After July 1998
Party	**IUA**	**PPY**	**IUA**
EU	180,251	202.06	37,000
JA	181,995	176.98	32,905
RF	80,936	140.45	15,600
US	112,767	174.78	–
Totals	555,949	694.27	85,505

Detailed statements of work for the new R&D task sharing during the EDA extension period have been developed with Home Teams in line with the proposals presented to and supported at MAC 14 for a total of 85,505 IUA distributed to Parties as shown below. The first tranche of work packages, covering R&D needed to continue existing R&D activities or to develop generic key technologies, is ready for formal release as soon as Home Teams will be able to accept it.

Arrangements have been made to redistribute selected computing equipment and software used for the JCT work in San Diego to Naka and Garching. Items purchased using the ITER Joint Fund in San Diego will be transferred to the respective Joint Fund Agents at Naka and Garching who will take over title and management responsibilities in accordance with the Joint Fund Financial Rules. In addition, some items of Host Equipment in San Diego are being sent to Garching under the terms of a loan agreement.

Project documentation is being redistributed to Naka and Garching as appropriate together with selected reference documents and other material from the project library in San Diego.

Work has continued on the seven large R&D projects. As illustrated in the highlights below, some are now complete as originally planned; others have witnessed significant progress towards their goals:

L-1 CS Model Coil Project. Manufacturing of the Inner and Outer Modules have been completed, in the US and Japan, respectively, and Acceptance Tests were successfully performed on the both modules.

The Outer Module was delivered to the test site at JAERI in November, 1998 (see Newsletter, Vol.7, No. 11, November 1998); the Inner Module, together with the support structure, has been shipped from San Diego and is expected to arrive at JAERI in mid April. After the installation work at JAERI, cooldown of the CSMC assembly is expected to start in late September or early October, 1999.

L-2 TF Model Coil Project. In the fabrication of the TFMC, two superconducting double pancakes have now been completed by Ansaldo and delivered to Alstom, where they will be assembled into the winding pack and inserted in the case. The conductor for the remaining three double pancakes has been heat treated, insulated and welded into the radial plate; impregnation of one of them is underway and insulation wrapping is nearing completion for the other two.

In the fabrication of the full size case sections, the square tube (representing the inner curved section of the TF coil case) has been completed. The final section weight is 37t. The tube will be cut into 2 U sections and rough machined before shipping for final machining and butt/closure welding trials. Suitable welding processes have already been qualified.

A trial 1 t cast section for the outer case section has been produced. The cryogenic properties of this special purpose alloy have been found to be acceptable for use in the less stressed regions of the TF coil case, such as the OIS/outer leg region. Repair and welding to forged sections of the casting has been demonstrated. The release of the casting of the full size section is now under discussion, with finalisation of the geometry.

L-3 Vacuum Vessel Sector Project. The L-3 project was successfully completed in August of 1998, and all objectives were achieved. Both half sectors were completed and shipped to the Tokai establishment of JAERI on schedule. It then was proved that it is possible to weld together two sector halves made with different segmentations and by different techniques, and achieve tolerances within those required for ITER. This supports the view that several manufacturers world-wide would be able to fabricate the vessel successfully, either alone or in co-operation with other manufacturers. The objectives for the port extension model have also been successfully achieved. All activities were completed in Russia and the port extension has been shipped to Japan in July 98 for integration with the full scale sector model.

L-4 Blanket Module Project. The main materials for the blanket system have been selected by the EU, JA, RF HTs efforts, and characterised in unirradiated and irradiated conditions and their properties improved Remarkable improvements have been obtained by all HTs in the development of the main joining techniques — Be/Cu-alloy, Cu/SS and SS/SS — to be used for the manufacturing of each component. Results of the R&D on ITER water chemistry show that corrosion/erosion of stainless steel and copper alloys is negligible for ITER coolant conditions, provided that water is purified, its conductivity is kept low and a reducing medium is added. Neutronics experiments have validated the transport codes and cross-section libraries that are used for neutronics design calculations of ITER. Results of calculations are used to define design margins. The basic manufacturing feasibility and the good performance of the primary wall, baffle and limiter modules has been assessed by manufacturing and testing several small and medium scale mock-ups in a first stage and prototypical components in a final stage Achieved tolerances in the manufactured prototypes and the results of the thermo-mechanical tests fully meet the requirements. The feasibility of the blanket module integration, of the attachment system and of the hydraulic and electrical connections is being assessed by manufacturing and testing mock-ups of the key parts and by preliminary assembly testing to verify the module installation and the crucial operations.

L-5 Divertor Cassette. The US Home Team completed a 4 t cassette body segment using cast 316LN. All the closure plates were welded using the penetration enhancement compound, and subsequently the component passed 100% ultrasonic inspection, as well as pressure and helium leak tests.

A US mock-up using 3 mm diameter tungsten rods hot pressed into the Cu substrate, showed no damage after testing at 30 MW/m2. and in Japan heating tests on full-scale monoblock divertor mock-ups, using 2-D CfC and CuMn braze. withstood a heat load of 20 MW/m2, 10s for 1,000 cycles without failure.

The EU Vertical Target medium scale, CfC and tungsten armoured prototype, consisting of two poloidal slices assembled to steel back plates, has passed He leak testing. The two units are now being assembled and the manifolds are being welded. The thermal fatigue test in the Le Creusot facility is planned early 1999.

A need identified late in the original duration of the EDA is to develop a technique for joining the CuCrZr tube to the Cu Active Metal Cast (AMC) layer of the CfC monoblock using a HIP process performed at ~ 500°C. This allows the joining to be carried out at the same temperature as the precipitation hardening cycle of the CuCrZr and avoids the over-ageing that occurs with higher temperature joining cycles. Furthermore, the residual manufacturing stresses blamed for infant joint failures are strongly reduced. On this basis, the EU have built CfC monoblock mock-up. The CuCrZr tube shows mechanical and thermal properties comparable with those of a typical solution annealed and aged CuCrZr alloy. The joint between the tube and the AMC appears better than the brazed joints of previous components. A high heat flux testing of this mock-up is planned early in 1999 in JUDITH facility.

L-6 Blanket Remote Handling Project. This project was completed successfully. The full-scale remote handling equipment for blanket maintenance was fabricated by Toshiba and assembled into the Blanket Test Platform at Tokai JAERI for demonstration of ITER blanket replacement. Through performance tests of the equipment, feasibility for handling 4 t blanket modules at any point on the first wall was fully verified.

In addition, the integrated performance test demonstrated the critical operations of rail-deployment/storage, module replacement and transfer, so that the vehicle manipulator system satisfies the maintenance scenario for the ITER blanket. On the basis of these results, the fundamental technology for blanket maintenance has been well established.

L-7 Divertor Remote Handling Project. This project was completed successfully. The validity of the concepts for divertor cassette handling and refurbishment by remote means was demonstrated on two test platforms at Brasimone (EU).

Editor's Note: *Developments in the ITER Physics framework will be the subject of a separate article.*

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