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INSIDE

FINAL MEETING OF ITER COUNCIL

by P. N. Haubenreich, ITER Council Secretary

All agree that CDA were successfully concluded

Achievements recognized

In a meeting on 11 December 1990 at IAEA Headquarters in Vienna, the ITER Council concluded its overall supervision of the ITER Conceptual Design Activities.

The Council heard the ITER Management Committee report that the scope of work set forth in the CDA Terms of Reference had been been fully accomplished and that arrangements had been made for IAEA publication of results. The Council then received the ITER Scientific and Technical Advisory Committee's final report, with its favourable evaluation of the CDA results, and a requested report from a group of experts identifying and suggesting allocation of responsibility for critical tasks in technology R&D. The Council then finished drafting the ITER CDA Final Report, to be printed and widely distributed by the IAEA.

In concluding remarks, the Council praised the effective teamwork and remarkable achievements of the many scientists and engineers who had participated in the CDA at Garching and at home. Finally, the Council expressed its appreciation to the IMC, the ISTAC, and the ITER Secretariat for their roles, to the IAEA for its support, and to the host organization for the site and support for the joint work.

ISTAC MEETING

by N. Pozniakov, ISTAC Secretary

The 12-member ITER Scientific and Technical Advisory Committee, meeting on 28-30 November in Vienna, evaluated the completed results of the CDA with respect to established objectives and possible future co-operation on ITER.

ISTAC finds CDA results meet objectives, form basis for further work

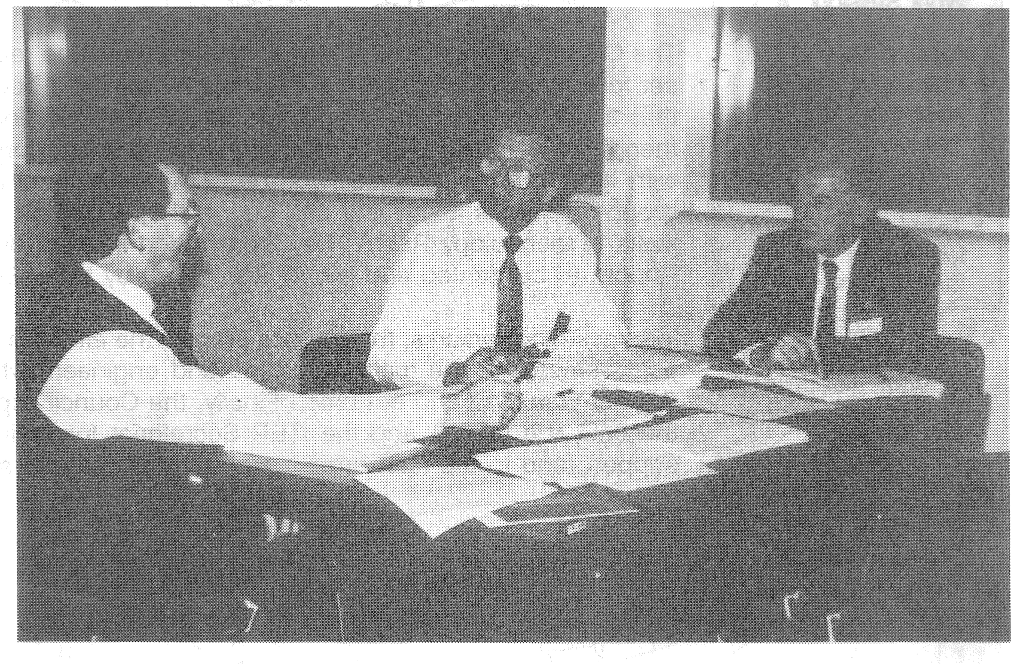
The Committee agreed on general conclusions, including the following. "The conceptual design meets major programmatic and technical objectives formulated in the ITER Terms of Reference. The ITER conceptual design, along with the developed long-term physics and technology R&D programmes, is sufficient to proceed to the engineering design supported by relevant R&D activities." The ISTAC also reached conclusions on specific aspects of the work and formulated suggestions for further co-operation among the Parties.

COMPLETION OF 1990 SUMMER JOINT WORK SESSION

by K. Tomabechi, Chairman, ITER Management Committee

Final session of CDA work at Garching

The final Joint Work Session of the ITER Conceptual Design Activities, which began on July 2, 1990 at the technical site Garching, Germany, was completed on November 16, 1990 with the achievement of all the objectives assigned to this session. About 50 full-time professionals participated in the Joint Work Session.



Scenes during ISTAC reviews. Top, left to right: M. Tanaka, V. Krylov and P. Rebut. Bottom, left to right: D. Post (Leader, Physics Project Unit), P. Rutherford and F. Troyon.

In addition, many short-term participants attended specialists' meetings and workshops. The work at Garching was conducted with the support of the home teams.

By the end of August 1991, critical issue assessments and machine design integrations were virtually completed. Draft reports of the physics and technology R&D needs for the Engineering Design Activities were developed. The results of this work were presented at the ISTAC meeting, held at the IAEA Headquarters in Vienna on September 12-14, 1990.

Results reviewed by ISTAC, told to fusion community

The highlights of the work conducted during the Conceptual Design Activities were presented in a set of technical papers at the 13th International Conference on Plasma Physics and Controlled Thermonuclear Fusion Research held in Washington D.C., USA, on October 1-6, 1990. (See list of these presentations in the November 1990 ITER Newsletter.) The presentations were well received by the audience with active discussions.

Reports will be published by IAEA

In November, the results of the R&D conducted during the Conceptual Design Activities, both in the physics and technology areas, were assessed and compiled into a single document. Furthermore, all technical results of the work conducted in the past two and a half years were documented in a series of technical reports (see Table I). Manuscripts of all these reports were prepared for publication by the IAEA in the IAEA/ITER Documentation Series.

Thus the Joint Work Session was completed successfully on schedule with the outcomes compiled into a series of technical reports. The work was presented at the ISTAC meeting on November 28-30, 1990, and reported to the ITER Council on December 11, 1990.

TABLE 1. ITER TECHNICAL REPORTS AT END OF CDA

IAEA/ ITER/ DS/No.	Title	Pages	Co-ordination
18	ITER Conceptual Design Report	200	IMC
19	Physics and Technology R&D for ITER Conceptual Design	100	IMC
20	Research and Development Needs for ITER Engineering Design	100	IMC
21	ITER Physics	300	Post/Uckan
22	ITER Parametric Analysis and Operational Performance	100	Iida/Perkins
23	ITER Operation and Research Programme	100	Post/Shatalov/ Shimomura
24	ITER Test Programme	200	Shatalov
25	ITER Tokamak Device	100	Salpietro/Shatalov/ Doggett
26	ITER Magnets	200	Miller
27	ITER Poloidal Field Ssytem	200	Shimomura
28	ITER Containment Structures	200	Sadakov
29	ITER Blanket, Shield and Materials Data Base	200	Smith/Shatalov
30	ITER Plasma Facing Components	160	Vieider/Shatalov
31	ITER Fuel Cycle	200	Dinner
32	ITER Current Drive and Heating System	200	Parail
33	ITER Diagnostics	150	Mukhovatov
34	ITER Assembly and Maintenance	200	Honda
35	ITER Plant	100	Iida/Kolbasov
36	ITER Safety	100	Iida/Raeder

PROGRESS IN THE ITER BASIC DEVICE DURING CDA

by E. Salpietro, Leader, ITER Basic Device Project Unit

Since the end of the ITER Definition Phase, two years of intense activity have elapsed. The ITER Basic Device has evolved in finalizing the overall device design philosophy and criteria as well as in improving the data base for the design and defining the R&D needs for the EDA. The integration of the machine was achieved through a continuous iteration of the safety, reliability, nuclear and physics testing requirements. The present ITER concept and parameters were chosen at an early stage of the CDA to allow detailed analysis of the concpet to be carried out. Based on these analyses, it was possible to define the strengths and weaknesses of the original concept to allow some readjustment of parameters at the start of the EDA. Overall it was demonstrated that the proposed concept is satisfactory for achieving the machine objectives. The existing data base supporting the machine design is well established and the necessary R&D programme to develop the industrial technology to the level required to support the ITER construction has been defined. A considerable effort has been devoted to properly document the activity in order to ease the start of possible Engineering Design Activities.

Conceptual design evolved over two years

The ITER concept, whose parameters are reported in Table I, is characterized by:

- Self supporting systems (e.g. Toroidal Field Coils),
- Full remote maintenance with provisions for hands-on maintenance where possible,
- All superconducting coils outside the vacuum vessel,
- Fast active control of the plasma with copper coils inside the vacuum vessel,
- Double vacuum barrier for remote leak detection,
- Containment structures designed to resist all <u>hypothetical</u> accidents, and
- Components designed in such a way that a failure will not propagate to other components.

The in-depth-analysis carried out during the CDA revealed that the following areas needed improvement:

Further improvements identified

- Thicker inboard nuclear shielding to reduce the heat loads in the TF coils,
- Provide more current in the PF system to allow fully inductive physics operation up to I_p = 28 MA,
- Re-consider the possibility to fully replace the in-vessel components (i.e. blankets during the lifetime) to improve the flexibility of the machine.

To incorporate these improvements, it is expected that the machine parameters will increase only by a few percent with no significant cost impact. It is also expected that a critical review of the concept shall be carried out at the beginning of the EDA to confirm the inputs for the design derived from the experimental physics and technology programme.

R&D during CDA improved data base

During the CDA progress was been made in the data base for the design. Major achievment in the remote maintenance, magnets, current drive and heating systems are:

- Qualification of remote handling components (e.g. proximity sensors, motors) to be used in the ITER environment was initiated: part is already complete and the remainder is due in 1991.

- Vehicle and manipulators for divertor plates handling scale of 1:5 were successfully tested.

- Partial boom scale 1:1 for in-vessel maintenance under construction to be tested in 1991.

- Magnet double pancakes with subsize superconductors successfully tested.

- Superconducting wires with acceptable characteristics for ITER already produced and tested.

TABLE I. KEY ITER DESIGN PARAMETERS AND REQUIREMENTS

Major radius (m)	6.0
Minor radius (m)	2.15
Elongation K(95%)	1.98
Safety factor q (95%)	3.0
Plasma current (MA)	22
Field at plasma (T)	4.85
Number TF coils	16
Number PF coils	14 external (s.c.) 2 internal (normal)
PF system flux swing (V-s)	325
Mean radius of central solenoid (m)	1.73
Mean radius to outer PF coils (m)	11.5
Mean height to divertor PF coil (m)	9.0
Maintainability	by remote-handling methods

- Last but one stage ITER superconductor cable successfully tested under ITER conditions (B, T, $\varepsilon_{\parallel}$, $\varepsilon_{\perp}$).

- Test facility for testing of full size superconductor is close to completion.

- Full size superconductor lengths ready to be tested.

- Qualification of the conductor element (wires, steel jacket, chromium plating, brazing) successfully carried out.

- For the heating and current drive systems the RF sources were developed, although the efficiency needs to be improved.

Technology R&D foreseen

The R&D programme for the period 1991-96 foresees an expenditure of about $400 M in the four areas: magnets, remote handling, containment structures, heating and current drive systems. The major items of expenditure are:

- Magnet model coils and testing
- Remote handling prototype tools qualification for in-vessel maintenance
- Fabrication and testing of parts (e.g. full-size insulated sector bellows)
- NBI scalable beam line
- EC scalable line

The results were only achieved through the dedication of the participants to the joint work sessions in Garching, the joint efforts of the home teams and the contributions of the experts invited to assist the ITER Team.

ITER SPECIALISTS' MEETINGS IN GARCHING

submitted by N. Fujisawa, IMC Secretary

1. EXPERTS' WORKING SESSION ON THE PHYSICS OF ALTERNATIVE POWER AND PARTICLE CONTROL CONCEPTS

Status of physics surveyed, EDA R&D suggestions made

The objective was to review the status of the physics associated with alternative concepts for power and particle control in ITER. The major impact will be upon the R&D programme, although those concepts which are vigorously pursued at present might influence the EDA in the relatively near term, i.e., 1 or 2 years.

The results from TORE SUPRA provide increased support for the potential benefits of ergodisation relative to a plasma bounded by poloidally and toroidally asymmetric limiters. However, in order to impact upon the EDA of ITER, the potential advantages of the TORE SUPRA ergodic configuration must be related to the capabilities of the toroidally symmetric poloidal divertor. There appear to have been no recent advances in this direction.

In respect to the present ITER design, field line tracing indicates that, if the internal coils were appropriately connected and energized, they could be used to ergodize a thin layer ($\sim$1.3 mm for a coil current of 10kA) of the outer edge plasma adjacent to the separatrix. The layer extends for $\pm$ 40° in the poloidal direction. Plasma edge modelling of a comparable configuration indicates that an ergodizing current of 20kA would reduce the peak power load on the divertor plates of ITER options A1 and B6 by about 10%.

The capability for using the internal coils of ITER as ergodizing windings either in the $n=1$ mode or in higher n modes should be included in the EDA. The practicability of using this system to avoid toroidal peaking by locked modes should also be investigated.

R&D on wider ergodic edges (such as in the TORE SUPRA configuration) is of long-term relevance to ITER and should be encouraged but greater emphasis is required in respect to identifying the practicability of its application to tokamak reactors.

Biasing of the newly installed ring of the DIII-D divertor has been used to produce E x B drifts which displace the power load peaks across the plates. Suitably directed E x B forces also increased (by at least a factor 5) the neutral gas pressure in the divertor pump duct. In principle, an oscillating bias potential could be used to sweep the divertor plasma channels across the plate at a higher frequency than magnetic sweeping. Biasing would require that, at least, the outer pair or the inner pair of the divertor plates be electrically insulated.

Radial electric fields such as those produced in PISCES simulation experiments and in discharges with ALT-2 in TEXTOR modify transport in the edge plasma and there is now a considerable experimental and theoretical data base related to these issues.

Measurement of the sputtering yields of copper samples in the SPRUT 4 plasma simulation device indicate that erosion would be reduced if the divertor plates could be biased close to the plasma potential.

The relative merits of using oscillating bias of the divertor plates or of magnetic sweeping of the null-point as the method of spreading the power load and reducing plate erosion should be explored during the EDA. In addition, further confirmation in tokamak experiments is required before the oscillating bias concept can be accepted as a physically viable alternative.

A major objective of the proposed JET divertor is to retain impurity ions within the divertor by the friction forces that arise as a consequence of injecting gas upstream of the divertor. Most of the input power to the divertor could then be dissipated by impurity radiation. Simulation of JET conditions indicates that about 10 to 20% of the plasma particle flux upon the plate must be injected as neutrals in order to retain the impurities if the upstream plasma density is less than about 10^{20} m^{-3}. No recirculation is required at higher edge densities. JET therefore envisages that operation with a clean core plasma is likely only at edge densities higher than those envisaged for ITER.

Two-dimensional simulation of the recirculation of Franck-Condon D/T atoms to localized regions of the private flux plasma of the ITER divertor indicates that their penetration into the plasma is inadequate to fully eliminate backflow of D/T ions from the divertor. Nevertheless, the peak power load and peak plasma temperature at the plate are almost halved if the neutral particle recirculation fraction is 10%. This is explained by the localized nature of recirculation assumed in the simulation which gives rise to a local reduction of the gradient in electron temperature along the magnetic field.

The puffing of 60 torr-l/s of N_2 gas into the private flux region of DIII-D substantially increased radiative power losses within the divertor but it also increased Z_{eff}. However puffing D_2 at a rate of 120 torr-l/s reduced the peak power loads on the plates, it increased $<n_e>$ and Z_{eff} was also slightly reduced. The change in power loads might be attributed to a combination of the behaviour noted above together with the rise in $<n_e>$.

To form a gas divertor target it is necessary to cool the divertor plasma to about 1eV so that volume recombination of the plasma electrons and ions can occur. Previous modelling studies of cooling the plama by surrounding it with a charge exchanging gas indicate that energy can only be extracted from a relatively thin surface layer except in conditions where the ionisation rate is much smaller than the charge exchange rate, i.e., the plasma temperature must be below about 3eV. The implications are that both a long divertor channel and a high upstream density are required. However, recent experiments in the PISCES plasma simulator indicate that some form of enhanced cross field diffusion of plasma particles sets in at electron temperatures below about 10 eV. The interpretation is still under discussion but, if the effect can be substantiated and extrapolated to ITER-like conditions, it might be beneficial in spreading divertor plate power loads provided that the plasma could first be cooled to the appropriate temperature.

A three-pronged approach to R&D for gas injection (or recirculation) is recommended: (1) tokamak experiments, (2) modelling studies and (3) plasma simulator experiments. Work on methods for recirculating substantial quantities of D/T gas should also be pursued. The present physics guideline regarding the ratio of core to edge plasma densities in ITER should be reviewed.

The study of plasma-liquid metal interactions in T-3M indicates that the plasma performance with a conventional carbon limiter and with a screen of gallium droplets is comparable, at least after the first 20 ms of the 30 ms discharge. Extrapolation of these results to ITER is uncertain.

Liquid surface divertor plates (and limiters) are relevant to the long-term search for improved erosion and power handling capabilities for ITER. Therefore, R&D in this area is encouraged.

2. SPECIALISTS' MEETING ON ADVANCED DIVERTOR CONCEPTS

The objectives of this meeting were: (1) to review the status of innovative and advanced divertor concepts that could potentially provide improved solutions to the difficult divertor problem for ITER, and (2) to develop recommendations on which, if any, concepts should be pursued in the EDA R&D effort.

Specialists evaluated concepts for divertor thermal design

The proposals were categorized according to the following:

- Advanced divertor materials,
- Liquid metal divertor concepts,
- Helium pumping enhancement,
- Other divertor concepts, and
- Innovative coolant concepts.

Advanced divertor materials considered included boronized carbons, advanced carbon-fibre-composites (CFC's), thermionic electron emitters, and liquid metal target materials. Boronized graphites and CFC's provide advantages over normal graphites and CFC's with respect to physical and chemical sputtering, radiation enhanced sublimation, and tritium retention. More work is recommended on physical properties (particularly thermal conductivity), thermal shock resistance, effects of neutron irradiation, and manufacturing.

Further work was also recommended on advanced CFC's particularly with respect to thermophysical properties, thermal shock resistance and neutron radiation effects.

The potential of thermionic electron emitters for divertor applications in ITER was considered doubtful and further work in this area was not recommended.

Sputtering data on proposed liquid metal targets, e.g. gallium, are very limited. Because of the high interest in liquid metal divertors discussed below, it was recommended that further work on sputtering of selected liquid metal targets be conducted.

Considerable information was presented on the liquid metal divertor concepts. Concepts discussed included flowing films on a support plate, free flowing droplets, and free flowing curtains. The primary advantages of the liquid metal divertors are possibly higher heat load capability (3-8 times solid plates), increased lifetime, and runaway electron protection. Main issues include stability during disruptions, edge effects for film flow, sputtering and tritium inventory uncertainties, and design uncertainties. It was recommended to include liquid metal investigations in the EDA R&D effort, to increase design work by national teams, and to consider incorporation of liquid metal divertors as an alternate option in the EDA phase.

Two helium pumping enhancement schemes were presented: use of permeation to selectively remove D/T from the pumping duct, and use of special materials to selectively pump helium (helium self-pumping concept). Both methods could potentially provide benefits for the impurity control systems in ITER. If both can be used together the potential is even greater. Since neither concept has been fully demonstrated, further work is required to evaluate their feasibility. Because of the significant benefits, further R&D is recommended.

The solid ball curtain concept which utilizes recirculating carbon balls as the divertor target was proposed. It was concluded that the current proposal of falling graphite balls was not compatible with the ITER design. R&D in support of ITER was not recommended; however, further development could be conducted by individual parties.

A mechanically swept divertor concept was proposed as an alternative to magnetic sweeping. Because of several difficulties that were identified this concept was not recommended for ITER, but the efforts of national teams were encouraged.

Floating carbon tiles on a liquid metal film was proposed as a possible option that would permit in-situ replacement of graphite divertor tiles. Several serious concerns with this concept were identified. Consequently, this concept was not recommended for further development unless gap conductivities can be enhanced.

A unique helium cooled divertor concept was proposed. The participants acknowledged the design effort and the ingenious technical innovation; however, it was concluded that this concept was not relevant to the ITER design. Further work in support of ITER was not recommended.

TWIN-LOOP CONCEPT FOR PLASMA VERTICAL STABILIZATION

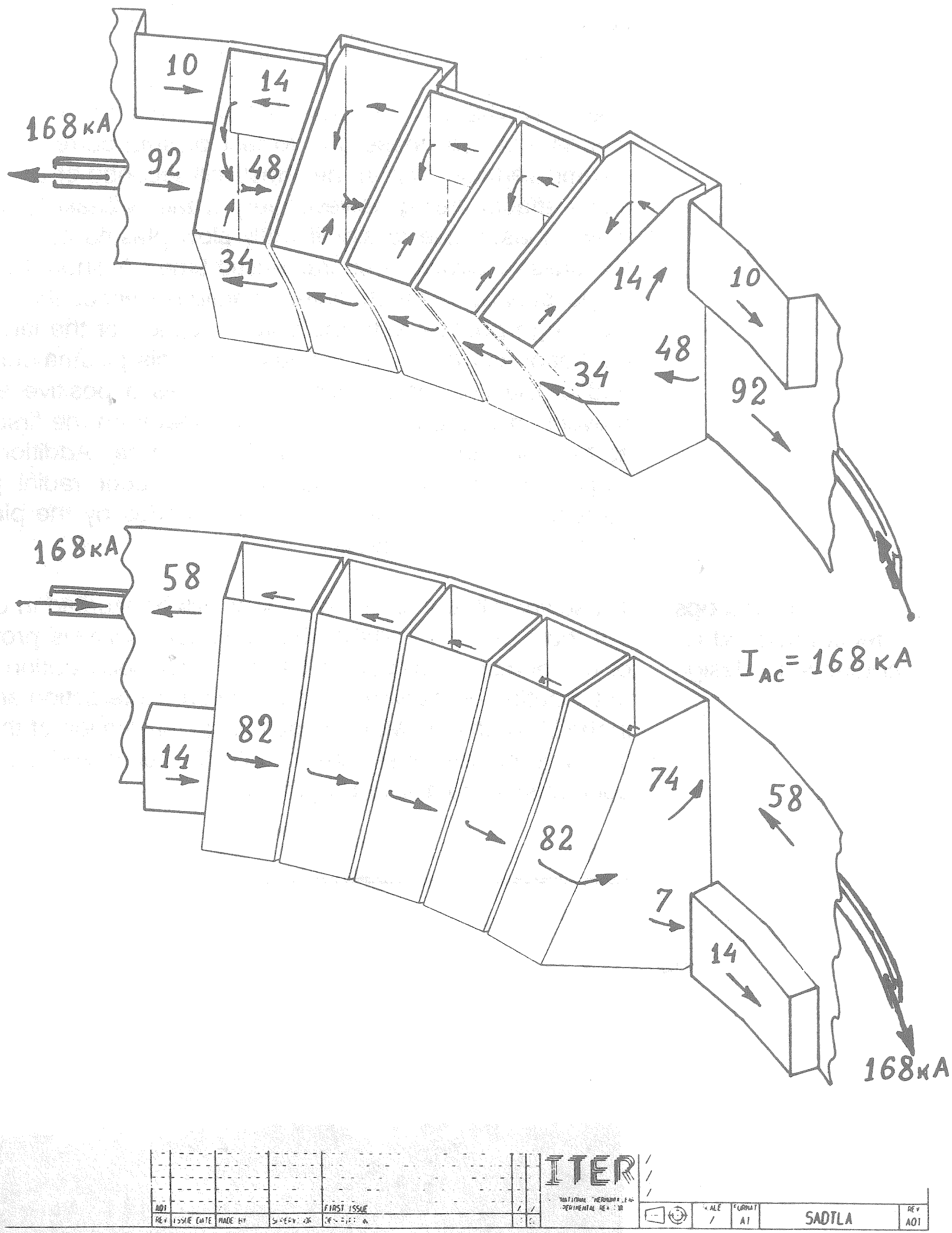
by S. Sadakov, Leader, Containment Structure Design Unit

Editor's Note

The following article is a continuation from the November issue of the Newsletter. In the first part of the article, a new passive stabilization concept, named twin-loops, was basically presented. This second part provides more details on the twin-loops performance.

Continued description of performance

Two interesting aditional effects of twin-loops were predicted qualitatively in September 1989 and shown clearly by the following numerical simulations. The first one is the transformation of the active coil pulse currents through blanket



Currents induced in blanket loop by active coil current pulse

loops from the blanket back wall on its face wall (Fig. 1). In the typical case about 2/3 of the active coil currents are compensated by the opposite currents in the vacuum vessel toroidal loops, and the remaining 1/3 are transformed by the blanket loops on to the face plates, which is a significantly more effective position from the point of view of the plasma stabilization. It was also clearly shown that additional cutting of the copper plates of the vacuum vessel bulk structure near an active coil can improve this effect leading to a "zero" or "negative" electromagnetic shielding of the active coil by the torus conductive structures. Thus, fast active control and relatively longer instability growth time can be reached.

A result of twin-loops up-down parts galvanic independence, toroidal eddy currents in "+/+" or "-/-" modes can be effectively induced in its up/down parts due to plasma current fast variation or its radial displacement. Eddy currents in "+/+" mode result in an additionally increased plasma elongation; in "-/-" mode, in a decreased one (for positive plasma current). It leads to the clear dependence of the stabilizing effect on the plasma current fast variations. Stability margin and instability growth time are improved additionally in the case of fast plasma current increase and, in opposite, these parameters will be reduced in the case of plasma current drop. These effects were demonstrated in a disruption numerical simulation with non-rigid plasma model.

The EDDYC-2 code shows the improvement of the passive stabilization during thermal quench phase due to fast plasma current increase (~10%), and plasma column vertical jump in the latest milliseconds of the current quench phase. The same effects, being however very small, will take place at the start-up and shut-down phases due to significantly slow plasma current changes in comparison to the passive structure characteristic time. It should be pointed out that the twin-loops provide a much better stabilizing effect than the saddle ones during all discharge phases, with the only exception of the latest milliseconds of the major disruption current quench phase. But this plasma column fast vertical jump in the end of the current quench phase gives a positive effect, which consists in the prevention of the high peak thermal loads on the first wall due to plasma contact at the machine midplane in the inboard area. Additional positive effect of the twin-loops on the plasma behaviour is a better radial plasma stabilization due to currents generation in "+/+" or "-/-" modes by the plasma radial displacement in the thermal quench phase.

Twin-loops recommended for engineering design

These fine effects of the twin-loops will be studied in detail during the EDA phase, but the existing results of the numerical analysis provide a good basis in favour of the principal solution to use the twin-loops option in the ITER design. Future optimization of the twin-loops/active coils interaction and geometry should be done in the EDA phase. More detail design integration of the twin-loops/active coils with the vacuum vessel structure (both for sectorized and continuous vacuum vessel concepts) should also be provided.

The ITER Secretariat at the IAEA Headquarters in Vienna, who compiled, typed, edited and disseminated the ITER Newsletter during the two and a half years of the Conceptual Design Activities. From right to left: P.N. Haubenreich, C. Basaldella and N.L. Pozniakov.

THE EDITOR OF THE ITER NEWSLETTER, IN THE NAME OF THE READERSHIP, THANKS THE FOLLOWING AUTHORS FOR THEIR CONTRIBUTIONS AND COLLABORATION DURING THE ITER CONCEPTUAL DESIGN ACTIVITIES

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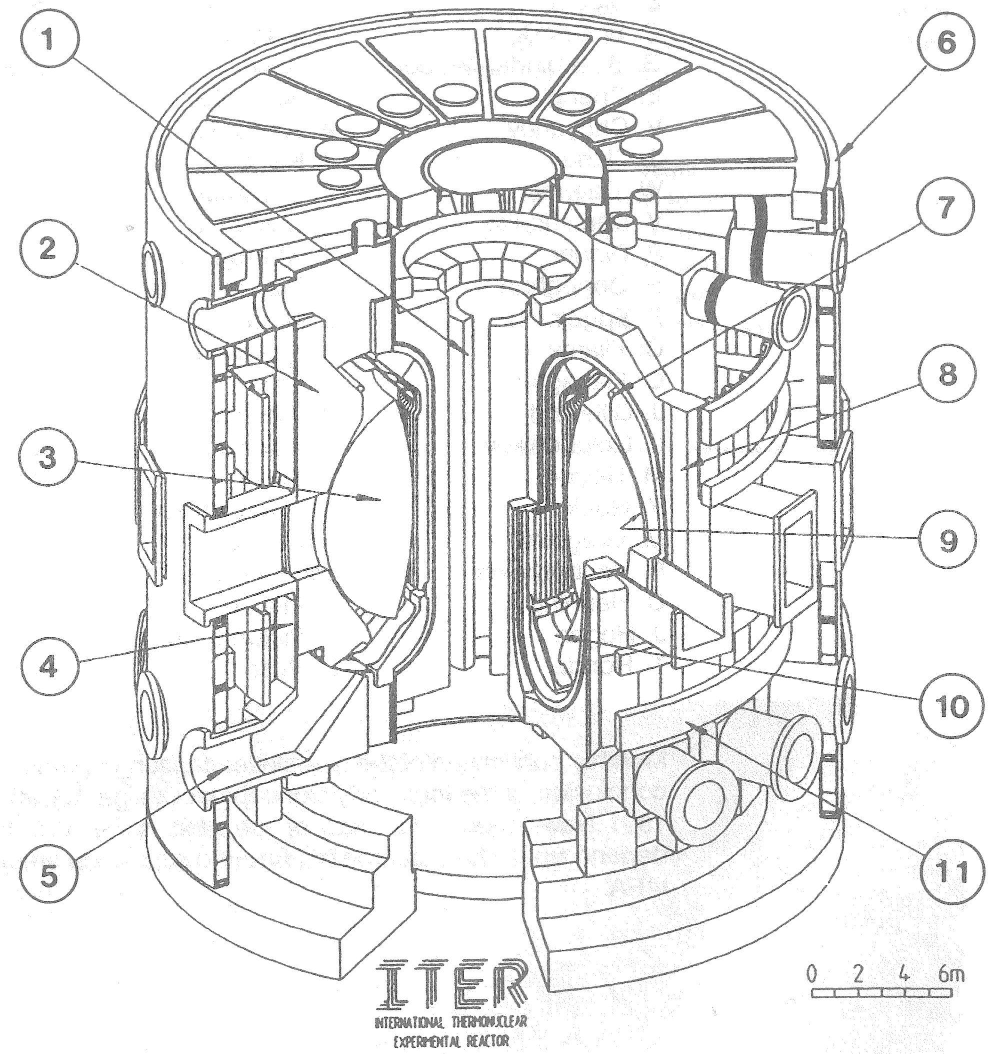
Monthly publication of the Newsletter ceases with this December 1990 issue, at the conclusion of the four-party Conceptual Design Activities. There will be no January 1991 Newsletter. The date of the next issue and the frequency thereafter will depend upon the outcome of current discussions among the CDA Parties and the IAEA.

ITER REFERENCE PARAMETERS

Plasma major radius, R (m)	6.0
Plasma half-width at midplane, a (m)	2.15
Elongation, 95% flux surface	1.98
Toroidal field on axis, B_o (T)	4.85
Nominal maximum plasma current, I_p (MA)	22
Nominal fusion power, P_f (MW)	1000



1- CENTRAL SOLENOID
2- SHIELD/BLANKET
3- PLASMA
4- VACUUM VESSEL-SHIELD
5- PLASMA EXHAUST
6- CRYOSTAT
7- ACTIVE CONTROL COILS
8- TOROIDAL FIELD COILS
9- FIRST WALL
10- DIVERTOR PLATES
11- POLOIDAL FIELD COILS

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