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INSIDE

ITER COUNCIL MEETING

by Paul N. Haubenreich, ITER Council Secretary

The eighth meeting of the ITER Council was held in Washington on 8 October, following the 13th International Conference on Plasma Physics and Controlled Nuclear Fusion Research.

Planning for R&D during engineering design

A major emphasis of this meeting of the Council was on the planning process for research and development that would be required if the Parties proceed with Engineering Design Activities. In the preceding weeks, the plans for physics and technology R&D that had been developed jointly under the direction of the ITER Management Committee were completed. In September both plans were reviewed by the ISTAC (see article in October Newsletter) and the Technology R&D Plan was reviewed also by the *ad hoc* group of experts appointed by the Council at their meeting in July. Both reviews produced positive conclusions but everyone recognized needs for early decisions on task-sharing and effective co-ordination of worldwide efforts to ensure timely, sufficient results.

The Council agreed upon the steps that will be taken by the end of 1990, in accordance with the Terms of Reference, to provide the results of the Conceptual Design Activities to the Parties for use in connection with their consideration of further steps toward the ultimate programmatic objectives of ITER. These steps will include the preparation of a comprehensive set of technical reports for publication by the IAEA.

IAEA CONFERENCE IN WASHINGTON

by John R. Gilleland, U.S. ITER Managing Director

ITER conceptual design presented to world fusion community

The International Conference on Plasma Physics and Controlled Nuclear Fusion Research, held every two years, is the preeminent international conference for announcing new results in fusion research. The 13th Conference, held in Washington this year during the first week of October, provided an opportunity for the ITER team to introduce the nearly completed conceptual design work to the international fusion community. The team presented their design in eight talks and 19 posters, listed in the accompanying table.

U.S. Energy Secretary Watkins: "Fusion research sets the pace among scientific fields for the breadth and vigor of its international partnerships. ... I think it is important that we maintain our momentum. ... You can be assured that the United States will thoroughly explore continued participation in ITER activities beyond the end of the current Conceptual Design Activities."

The oral presentations included four summary talks in a plenary session, covering: the project, by the Chairman of the Management Committee, Ken Tomabechi (Japan); an overview of the design, by Romano Toschi (EC); the physics basis by Douglass Post (US); and the engineering basis by Ettore Salpietro (EC). The other papers that were presented orally covered operational issues, nuclear engineering issues, the testing programme and safety. The poster session provided additional details on the physics and engineering design issues, including energy confinement, magnets, remote handling, etc. A booklet containing preprints of all 27 papers was distributed to the Conference attendees.

The oral sessions were well attended and lively, with many questions from the audience. The ITER team made the point that there has been a lot of progress in tokamak physics in the past three years. For example, documentation and analysis of H-mode confinement data from several machines have helped to clarify, on an empirical basis, several aspects of energy confinement scaling; the resulting "ITER scaling" is consistent with the ITER reference parameters. There was also parallel progress in fusion technology.

Physics progress cited

Judging from the response of the conference participants, ITER made a very favorable impression on the community. Many compliments were made on the book of preprints, the talks and the posters. The participants were generally impressed with the breadth of the project and the scope and quality of the work that had been done. Many people remarked that it was clear that the team had made an honest and forthright presentation of both the problems and the strengths of the design. The general atmosphere was that the time had come for the international fusion community to take ITER seriously as a potential machine that the world could build. For example, one of the first questions from the audience was: "What can be done to speed up the projected schedule for the construction of ITER?"

One question from the audience: "What can be done to speed construction?"

The clear demonstration at the Conference that an international design team composed of Europeans, Japanese, Soviets and Americans could work together as a single group and do a very good job of the conceptual design offered encouragement to the fusion community that it would be possible to build a large experiment like ITER as an international project.

TABLE 1. ITER PRESENTATIONS AT IAEA CONFERENCE

Oral	
ITER: Conceptual Design	K. Tomabechi
ITER: Design Overview	R. Toschi
ITER: Physics Basis	D. Post
ITER: Engineering Basis	E. Salpietro
ITER: Operational Scenario	Y. Shimomura
ITER: Nuclear Engineering	G. Shatalov/C. Baker
ITER: Test Programme	M. Abdou
ITER: Safety Aspects	H. Iida
Posters	
Energy and Particle Confinement in ITER	N.A. Uckan et al.
Operational Limits and Disruptions in ITER	T. Tsunematsu et al.
Power and Particle Control for ITER	S.A. Cohen et al.
ITER Current Drive and Heating Physics	V.Parail/ W.M.Nevins et al.
Plasma Operation Control in ITER	K. Borrass et al.
ITER Superconducting Magnets Systems	R. J. Miller et al.
The ITER Basic Machine Layout	F. Casci et al.
ITER: Remote Maintenance	T. Honda et al.
Transient Electromagnetics in ITER	L. Bottura et al.
Tritium Breeding Blanket	D. Smith et al.
Plasma Facing Components in ITER	G. Vieider et al.
ITER Fuel Cycle Design	P.J. Dinner et al.
ITER Plant Systems and Site Requirements	C.A. Flanagan et al.

TABLE 1. (Cont'd.)

ITER System Studies and Design Space Analysis	L.J. Perkins et al.
The ITER Poloidal Field System	J. Wesley et al.
ITER Operation and Diagnostics	V. Mukhovatov et al.
Physics Research and Development Programme	F. Engelmann et al.
A Preliminary Analysis of the ITER Energy Confinement H-Mode Data Base	J.G. Cordey et al

SAFETY AND ENVIRONMENTAL ANALYSES IN CDA

by H. Iida, Head, System Analysis Project Unit

Inherent safety advantages of fusion incorporated in design

Fusion has inherent advantages from the safety and environmental points of view, such as the absence of any criticality accident and of radioactive products from the fusion reaction itself. In a fusion reactor, only an overpower issue exists comparing to a damaging power excursion, which in principle can happen in poorly designed fission reactors. Although structures of a fusion reactor will be activated by neutrons emitted from fusion reactions, the radioactivity will be significantly less than the total radioactivity produced in a fission reactor of similar power level. Most of fusion radioactivity is in the solid form and is not prone to mobilize, so its release from a fusion plant is not a likely event. Most of fusion radioactivity decays faster than fission reactor radioactivity, thus making the radwaste handling issue easier.

However, such promising fusion safety potentials do not mean automatic achievement of super safe design and operation of fusion machines. In order to benefit from those safety potentials, a proper consideration of safety issues in actual design of the machine is essential. In the ITER conceptual design, careful examination has been done to identify any possibility of an uncontrollable accident and to find how it can be eliminated. Fire of carbon-based material, hydrogen and/or carbon monoxide explosion, and large-scale damage of the torus due to magnet accidents were studied. Analyses showed that on the basis of current assumptions these accidents could be eliminated by the proper design.

Many accident analyses were carried out during the CDA. They are grouped into four areas: (1) in-vessel, (2) ex-vessel, (3) tritium system, and (4) magnet.

Safety analyses of in-vessel occurrences

Coolant pipe rupture accidents are conceivable since the plasma facing components will be operated in a hostile environment. In such accidents, major concerns are hydrogen and carbon monoxide production due to chemical reaction between steam and carbon-based material, and overpressure produced by steam within the vacuum vessel. The amount of hydrogen and carbon monoxide produced depends on the temperature of the first wall and divertor plates. In the ITER design, the temperature of 90% of the surfaces is below 1000°C, while 9% ranges up to 1500°C and 1% up to 1800°C. Analyses show that production of hydrogen and carbon monoxide is not at a hazardous level even if air is introduced into the vacuum vessel. In addition, the ITER design incorporates an inert gas atmosphere around the plasma vacuum boundary. This prevents air ingress into the vacuum vessel so that the combustion of hydrogen and carbon monoxide gas is not possible. The calculated overpressure is well below the design value of the vacuum vessel so that it would work as the primary containment boundary, even in such accidents. In some cases, high temperatures (~250°C) may be required for conditioning of plasma facing components. In such a case, by using proper fluids and by avoiding use of high-temperature water, overpressure can be lowered enough to maintain the confinement function of the vacuum vessel.

The possibility of a carbon-based-material fire was examined. The temperature of a carbon-based material on the first wall is high enough to react with air if the vacuum boundary failed during plasma burn. However, analyses show that this reaction is self-terminating and can not continue long. The estimated amount of reacted carbon is less than 1% of the total residing in the vacuum vessel. In the actual ITER design, the inert gas atmosphere around plasma vacuum boundary would serve as a barrier against air ingress into the vacuum vessel, practically preventing fire.

Safety implications of loss of coolant were evaluated

Effects and consequences of a major loss of coolant accident outside the vacuum vessel were examined. Because of rapid pressure and flow decrease in the cooling system, heat transfer between coolant and plasma facing components will be lost almost immediately. One concern is the melting of a divertor plate, which is subject to very high heat load during plasma operation. Analyses show that melting may be avoided by using proper material, such as a refractory metal, so that plasma burn will be terminated passively due to carbon contamination before the temperature of the divertor plate reaches a risky level.

Safety of tritium system analyzed on basis of experience

A detailed Failure Mode and Effects Analysis (FMEA) was performed for the tritium handling systems. Tritium system safety technology is being established in the existing experimental facilities. Multi-confinement approach and many passive safety features are already implemented in tritium technology. ITER will fully utilize the existing experiences.

Abnormal magnet events will not damage surroundings

In the magnet safety area, the most severe accidental conditions, which pose large, abnormal electromagnetic forces on the toroidal field (TF) coils, were identified. The structural analyses on these accidental condition tend to confirm that the deformation of the coils remain acceptable and will not breach any surrounding components that contain significant radioactivity.

As a conclusion of the design work in the safety area, ITER is expected to have no uncontrollable accident.

Radiological effects of hypothetical events are addressed

Based on the various analyses of each accident, a few hypothetical accidents were selected to assess upper-bound radiological consequences. Doses at the site boundary due to such accidents are estimated to be lower than anticipated regulations, 100 mSv (10 rem), even though calculations employ conservative assumptions. Thus, ITER is expected to satisfy anticipated safety dose limits.

In fission reactors many engineering safety systems are included, such as emergency core cooling system, containment spray system, etc. In the ITER design, the active engineering safety systems are not yet well implemented because of the preliminary stage of the design. Later, the essential safety systems can be incorporated reducing accident consequences appropriately.

We are also studying the extreme case which neglects functions of all active safety equipment and confinement by the reactor building. This constitutes the concept of a passive safety. In this case the calculated dose at site boundary exceeds 100 mSv. Our ultimate target is to reduce it to the level of 100 mSv. This would make fusion a highly safe option. To this end, it is important to continue to make efforts in every stage of design and operation to reduce tritium and activation product inventories as far as possible.

ITER should meet environmental regulations

Safety and environmental analyses during the CDA show that it is feasible to make ITER meet the anticipated regulatory dose limits. Yet, there remains room for improvement - both to increase safety margins and to demonstrate more of fusion's safety and environmental potential. These are challenges for the EDA.

ITER SPECIALISTS' MEETING ON SAWTOOTH BEHAVIOUR AT HIGH BETA

by T. Tsunematsu and J. Hogan, Physics Project Unit

Sawtooth activity: a control problem

The sawtooth oscillation of the electron temperature near the plasma center has been observed on most tokamaks. As much as 10% of the thermal energy can be lost in about 100μs for intensely heated plasmas. During the process, the poloidal magnetic flux may be redistributed near the plasma center. Sawtooth activity, therefore, is considered to cause a serious problem in the burn control of high Q plasmas where the major heating power is supplied by the alpha particles, and it also has a negative impact on plasma position control through possible uncontrolled change in the internal inductance. On the other hand, sawtooth activity could help ITER performance by removing impurities (including He ash) from the plasma center.

Broad database used to characterize sawtooth behavior

To assess the present data base and establish the basis of the projection to ITER, a specialists' meeting on 'Sawtooth Behaviour at High Beta' was held at Garching, August 6 - 8, 1990. There were 25 participants in the meeting: six from EC, one from Japan, one from the Soviet Union, four from the U.S.A. and thirteen from the ITER team. Thirteen presentations were made. In addition to theoretical reports, data from JET, TFTR, T-10, ASDEX, TEXTOR and PBX were presented. Major topics discussed at the meeting were the characterization of sawtooth behaviour in the high-β(β_{pol})regime (in terms of the period, amplitude, crash time, saturation, etc.), measurements of the safety factor on the axis and position of the q=1 surface (q: the safety factor), time dependence of the pressure profile during the sawtooth, and control of sawtooth activity and impurity accumulation.

Measurements and model comparisons

Experimental results on the characteristic behaviour of sawtooth oscillations and comparison with theoretical models were presented. The typical sawtooth oscillation cycle consists of a long period (up to seconds) of gradual change in the plasma equilibrium parameters, a precursor phase (not always observed) which may last from ten to hundreds of milliseconds, when slowly growing modes can be seen, and a sudden crash, on the 10 - 100μs timescale which alters the plasma equilibrium parameters. During the precursor and crash periods a localized region of intense soft X-radiation is observed ('hot spot') along with a colder surrounding region with an m=1 structure ('$T_e$island'). The magnetic field topology is an important determinant of the plasma behaviour during a sawtooth cycle; however, it is not directly observable. Time and space-resolved soft X-ray emission (SXR), electron cyclotron emission (ECE) and time resolved external magnetic diagnostics are usually available. Some time- (or even shot-) averaged polarimeter or beam measurements of the q-profile (on JET, TFTR, TEXTOR, PBX-

M and ASDEX) are available for the plasma state just before or after the sawtooth. Nonlinear, 3D MHD computational predictions of SXR and ECE signals are compared with these diagnostics.

While the experiments report a variety of sawtooth phenomena (full reconnections on TFTR and both full and partial events on JET, TEXTOR and T-10), the nonlinear simulations presented at this meeting predict only Kadomtsev-like, or "full" reconnection. According to this model the poloidal flux is exchanged during the crash within a region larger than the q=1 radius. Experiments also observe values of q_0 which are too low to be consistent with the full reconnection model. The models also predict too short a crash time.

Sawtooth period

The sawtooth repetition period is required to estimate the impurity accumulation (or ash removal efficiency). Based on considerations of 3D modeling and fits to some experiments, a semi-empirical model predicts the sawtooth time is proportional to $R^2T^{3/2}$. When all experiments are considered, however, neither the temperature nor the size scaling are clear, although an increase of the sawtooth period with T is observed. For ITER considerations, we must be careful with the interpretation of this semi-empirical result: experiments report that the sawtooth period may easily be made as long as the experimental pulse length by using co-parallel injection, and with increasing difficulty by using balanced parallel and counter-parallel injection [see page 4]. This indicates that the q- or j-profile can be changed due to heating and the resultant current-drive methods. Thus, the semi-empirical scaling, if confirmed, should be regarded as a scaling for the <u>minimum</u> τ_{ST} in ITER.

Precursor of sawtooth

Reliable identification of sawtooth precursors is required for control. Numerical studies predict the existence of modes which grow slowly for the entire time of the sawtooth period, but for which the experimentally observeable quantities (mode kinetic energy and 'hot spot' displacement) are detectable only milliseconds before the crash. It seems that effects on the transport timescale, which are not included in the numerical models, may modify the detailed behaviour, so that the prediction of crash times depends on detailed knowledge of the resistivity and thermal conductivity profiles. Experiments on TFTR show that there are often no precursors. Thus, no reliable predictions of sawtooth precursor phenomena can be made for ITER at this time.

Suppression of sawtooth activity

Suppression of the sawtooth has been demonstrated experimentally with various methods: LHCD and counter NB injection on ASDEX, pellet injection and ICRF heating on JET, parallel NB on PBX, co-injection on TEXTOR, and EC heating near the q=1 surface on T-10. Among these methods, LHCD and NBCD are available for sawtooth control of ITER in the present reference scenario for current drive and others may be available as alternative methods.

Sawtooth stability is a fundamental problem for the steady-state phase, since the neoclassical Ohm's law predicts that $q_0 << 1$. An estimate of the level of driven current needed to suppress sawteeth in ITER was made. Current is injected in the direction of the main plasma current to broaden the j-profile, especially near the edge, and to use the naturally peaked ohmic current in the center to provide the necessary shear there. A relatively small counter-driven current (~38kA), localized near the axis, is used to raise q_0 to near unity (~0.9).

Does sawtooth affect impurity accumulation or dispersal?

The 3D models do not make specific predictions of sawtooth effects on impurity accumulation or dispersal. Dispersal of central impurities may be a qualitative prediction of the full reconnection model. However, no detailed comparison with experiments was presented for this topic, and there are discrepancies between experiments and the codes regarding the degree of reconnection and its effect on plasma parameters.

Methods of controlling sawtooth behavior are R&D subject

One of the major results in this meeting for ITER is the presentation of possible methods for sawtooth control. Several methods such as NBCD and LHCD are available for ITER by using the present reference scenario for current drive, and possibly even within the already provided power. The other alternative methods by using ICRF heating may also be possible to suppress sawtooth. Although there are uncertainties in the physics picture used to establish the guidelines, the major issues described above are emphasized in the ITER long-range physics R&D.

Editor's Note

Large-scale international co-operation in fusion, such as ITER or NET, benefits from and leads to consolidation of the worldwide activities in this area. There are many indications of interest in and willingness to contribute to the major projects from the side of smaller-scale national activities of various countries. We believe that many of these activities, which are not directly associated with ITER, are nevertheless relevant and interesting for our readers because some critical issues of the design of a next step machine are addressed. As an example, we present fusion activities of the Austrian Research Centre, Seibersodorf.

FUSION ACTIVITIES IN AUSTRIAN RESEARCH CENTER SEIBERSDORF

by I. Šmid, Department of Materials Technology

Electrical energy from nuclear power is not being produced in Austria, since a majority has voted against the opening of the nuclear power plant in Zwentendorf in a referendum carried out in 1978. This circumstance may explain the country's special relation to nuclear power. In spite of this development there is a very positive view on controlled thermonuclear fusion. Austria is not directly associated with or a member of an ITER Party. However, some national research activities as well as joint work in collaboration with foreign institutions are carried out.

Austrian Research Centre Seibersdorf (ARC), located near Vienna, was established in 1956. Research and develoment work is performed in many scientific and technical areas, such as engineering, information processing, life sciences, systems research and environmental technologies. Fusion activities of the ARC evolved during the years from the specific capabilities of several projects. Some of the present activities that may contribute to the development of fusion science and technology databases necessary for the next step devices are presented below.

Plasma facing components tested under NET and ITER relevant conditions

The performance of plasma facing components for divertor application is being tested under high-heat-flux conditions. The divertor design used, which is based upon an advanced carbon material brazed to a refractory heat sink (TZM, Mo.41Re), was developed and manufactured by the Next European Torus (NET) Team, Garching, with an Austrian industry partner. The present work emphasizes the evaluation of the loading experiments which are performed in collaboration with Sandia National Laboratories Albuquerque. The general thermal response, the permissible peak heat fluxes and fatigue properties are being studied. Presently up to 14 MW/m^2 can be applied to the surface in steady-state mode, and full mechanical integrity is still observed after 1000 pulses at moderate loading

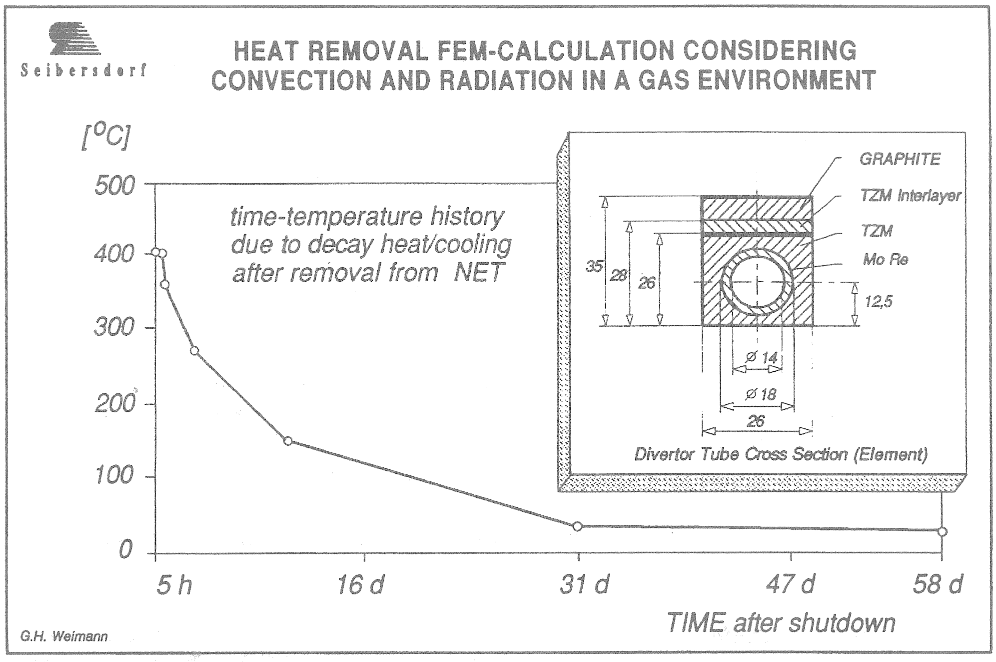


Fig. 1. Predicted temperature of dismounted and vertically stored divertor units consisting of several elements after neutron irradiation during full time operation of NET (technology phase, 830 days total exposure time). Convection cooling of the top, bottom (but not of the lateral) and tube inner surfaces was assumed, the minimum distance to other stored divertor units is 60 cm. In the figure the cross section of one single divertor element out of the whole unit is given.

conditions. The results contribute to the development and optimization of such components, in particular of the brazing technique. The postactivation behaviour and afterheat production under conditions of an ignited plasma are being studied with respect to the exchangeability and storage of worn out and activated divertor components (Fig. 1).

Different first wall structural materials, in particular austenitic and ferritic steels, are being investigated for their chemical compatibility with breeding materials, namely liquid Li and Li17Pb83. To study the performance of the blanket of a D-T-based fusion reactor, static as well as dynamic corrosion experiments are being carried out using a lithium loop. A decrease of the amount or replacement of elements with an unfavourable activation behaviour creates the need to test other than the known structural or reference materials. Consequently, a wide spectrum of materials and material combinations - optimized for fusion application - is being investigated.

Safety issues are being studied

Given the importance of safety issues, numerical methods are being applied to determine the precautions needed for handling and maintenance of the activated materials with respect to environmental considerations. In addition, the requirements for storage and waste management are evaluated as a part of a complex study of the behaviour of radioactive isotopes in the eco-system, including experimental determination of the transfer of isotopes between soil and plants.

Since present tokamaks are operated without high energy helium or alpha particles in the plasma, calculational methods have to be applied to simulate the transport mechanism of those particles. The used predictive plasma simulation codes are equipped with alpha transport routines to calculate the alpha power deposition in the plasma. Self-consistent models for alpha particle transport are not yet available; thus semi-empirical models developed for main ion transport have to be used. Further plasma instabilities triggered by fusion alphas are of essential importance for the operation of next generation tokamaks, which will maintain a plasma under thermonuclear conditions. Finally, the influence of alpha ripple losses on some construction parameters of fusion reactors is being investigated. These topics have been studied by ARC, partly in collaboration with the JET Team in Culham and the Massachusetts Institute of Technology in Cambridge.

TWIN-LOOP CONCEPT FOR THE PLASMA VERTICAL STABILIZATION

by S. Sadakov, Leader, Containment Structure Design Unit

Importance of vertical plasma position control

In a fusion reactor with a highly elongated plasma, vertical plasma position control is one of the key issues. In the present day tokamaks, the vacuum vessel, with conductive elements attached directly to it, serves as a passive stabilizing structure. In ITER-like machines the vacuum vessel is very far from the plasma because of sectorized blanket and other in-vessel components. This requires other design solutions to ensure plasma vertical stability. Typically the saddle loops attached on the outboard blanket segments are used for this purpose (Fig. 1).

Conventional saddle loop approach was originally proposed

The conventional saddle passive loop approach was originally proposed for the plasma vertical stabilization in ITER. Later, the evolution of the ITER parameters to the highly elongated plasma (K=2.2) drastically reduced the stability margin ($m_s = (f_{stab} - f_{dest})/f_{dest}$, where f_{stab} is the stabilizing force due to eddy currents and f_{dest} is the desabilizating force due to static currents). For the most deteriorated set of plasma parameters (minimum β_p and maximum internal inductance l_i), rigid displacement model and only 1cm electrical gaps between the saddle loops, the stability margin has an unacceptably low value: $m_s \leq 0.3$.

Additionally, unresolved design integration problem existed due to geometrical intersection of the saddle loops vertical legs, on one hand, and many horizontal ports (neutral beam injection, electron cyclotron heating, blanket testing and diagnostic) on the other hand.

New concept named twin-loops

To solve these technical problems, a new passive stabilization concept, which has been named twin-loops [1], was proposed and developed by the ITER team. Twin loops can be considered as a next step after "galvanic toroidal loops" analyzed earlier. Numerical study for the verification of the twin-loop concept for ITER parameters and design was carried out by the Poloidal Field Design Unit in the time period from September 1989 to September 1990 [1,2]. Complete numerical simulation results and relevant conclusions are given in Ref. [3].

Advantages of twin-loops

Twin-loops (Fig. 2) consist of highly conducting plates, attached to the outboard blanket segments face, side and back walls above and below horizontal ports area, and of the same conducting plates, attached to the adjacent inner surface of the vacuum vessel bulk structure. There is no special vertical electrical connection between the upper and the lower parts of the twin-loops. This is attractive for allocation of all horizontal ports. Also there is no requirement on galvanic connections between blanket segments and the vacuum vessel (due to inductive coupling being used); thus the problems of different thermal expansion movements of these structures are simplified.

Fast plasma vertical displacement induces simultaneously local circulating currents, flowing around each conducting loop on the blanket segments, and toroidal currents, flowing around the torus through highly conducting plates on the vacuum vessel bulk structures and through all resistive elements. Reverse parts of the

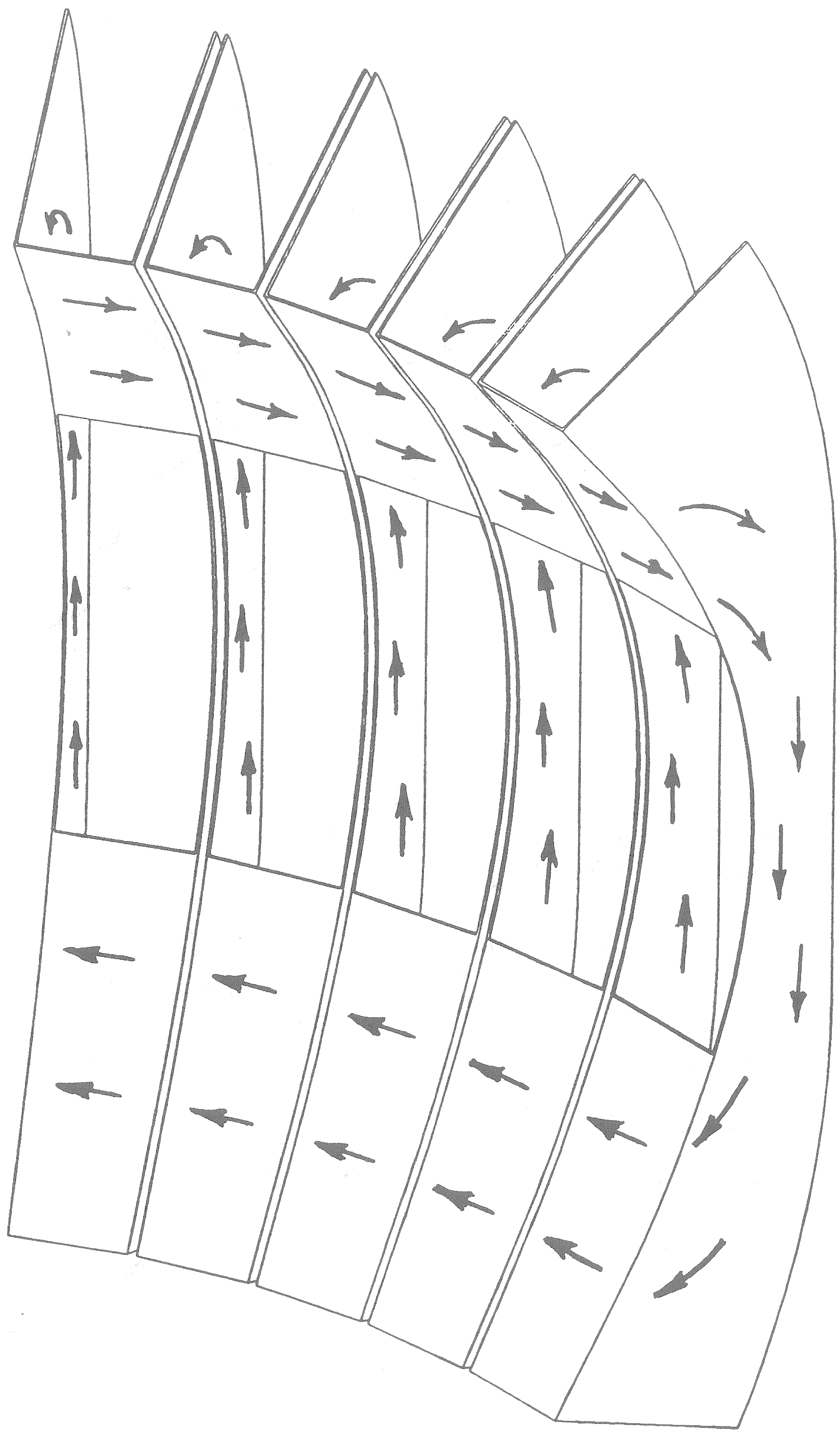


Fig. 1. Conventional saddle loops geometry and induced currents paths on the outboard blanket segments

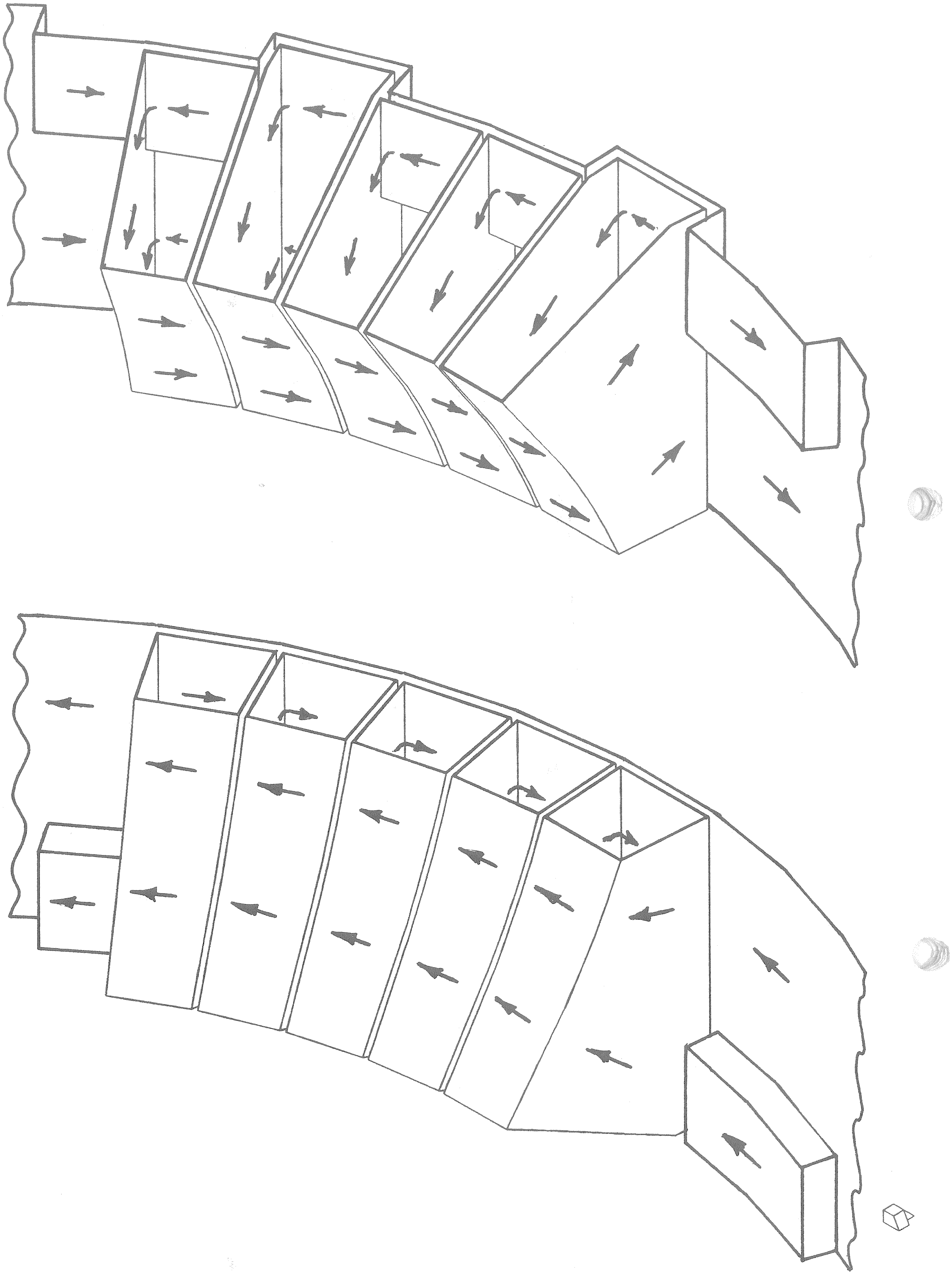


Fig. 2. New twin-loops geometry and induced currents paths on the outboard blanket segments and on the vacuum vessel

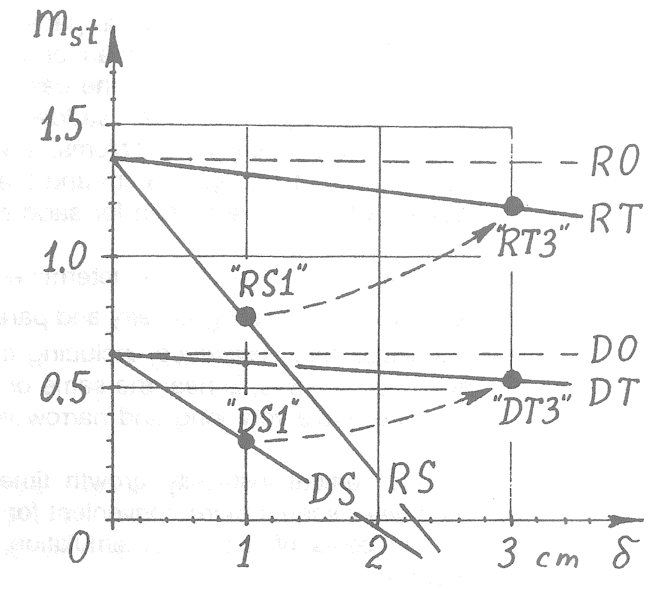


Fig. 3. Summarized values of the stability margin (m_{st}) as a function of all electrical gaps thickness (δ) for near-to-reference plasma parameters and:

- saddle (RS),
- twin (RT), and
- idealized "zero-gaps" (RO) passive loop options,

and also for deteriorated plasma parameters and:

- saddle (DS),
- twin (DT), and
- "zero-gaps" (DO) options.

RS1 and DS1 are operating points for saddle loops, 1-cm gaps. RT3 and DT3 are operating points for twin-loops with 3-cm gaps. (Interpretation of the results from Ref. [1].)

circulating currents on the back walls of the blanket segments and toroidally continuous currents on the adjacent wall of vacuum vessel have similar values and opposite directions, which leads to the compensation of the produced magnetic fields in the plasma area. Consequently, a passive stabilizing effect is provided mainly by the currents flowing on the face walls of blanket segments (the same as for saddle-loops option).

Why was twin-loops approach proposed for ITER?

The twin-loops approach for ITER was proposed for consideration in September 1989 on the basis of the following simple logic.

- If we compare stability margin values of the saddle- and twin-loops options for the same machine/plasma geometry, the same shape of passive loops face plates and the same values of all electrical gaps, then there is only one reason for different m_s values. It is the difference in gap inductance, which depends linearly on gap length and thickness, and inversely proportional on gap width. For the twin-loops configuration gaps the length is about 2-3 times smaller and the width is 3-4 times larger than for saddle-loops. Therefore, for the same gap thickness the inductance in the case of the twin-loop option will be ~10 times less than for the saddle-loops. Finally, taking into account manufacturing/assembly problems, it was decided to increase the thickness of all gaps from 1 cm up to 3 cm and therefore to keep gap inductance for twin-loops ~3 times smaller than for saddle-loops (Fig. 3).

- Current decay time (T_{st}) is determined mainly by the passive structure total resistance. For ITER geometry and parameters ($20\mu\Omega$ torus toroidal resistance), twin-loops total resistance, including the current fraction in the stainless steel resistive elements, is near the same or less than for the saddle-loops, because the latter have very long and narrow vertical legs.

- Finally, vertical instability growth time is determined as $T=m_a x T_{st}$, and this parameter seems more convenient for different passive structures comparison on the basis of numerical simulation, including results for non-rigid plasma models.

Numerical simulation of the vertical instability in ITER for the twin-loops concept was provided by some authors using different numerical codes and various machine conducting structures and plasma models (rigid and non-rigid) [1, 2, 3]. All these calculations confirmed the predicted better stabilizing effect of the twin-loops in comparison with saddle-loops. Summarized results are illustrated in Fig. 3, which compares the stability margin values as a function of all electrical gap values for the saddle-loop, twin-loop and idealized "zero-gap" structures with the same shape of the face conducting plates and machine/plasma geometry.

Using the twin-loops, typical stability margins for the most deteriorated plasma parameters increase from ~0.3 for 1-cm electrical gaps up to ~0.5 for the 3-cm electrical gaps. Typical instability growth time changes correspondingly from T ~20-40ms up to T~30-60ms. A dynamic plasma model shows about 20% shorter growth time than rigid models.

REFERENCES

1. Yu. Gribov, et alia. "Twin loops for vertical control of highly elongated plasma" (to be published in "ITER documentations series", IAEA, Vienna)

2. A. Astapkovich, et alia. "Transient electromagnetics in ITER", 13th International Conference on Plasma Physics and Controlled Nuclear Fusion Research, Washington, D.C., USA, 1-6 Oct. 1990.

3. "ITER Poloidal Field System", Technical report of ITER Poloidal Fields Design Unit (to be published in "ITER documentations series", IAEA, Vienna)

(This article will be continued in the next issue of the Newsletter.)

ITER MAJOR EVENTS - 1990

ISTAC Meeting	Vienna	28 - 30 Nov
ITER Council Meeting	Vienna	11 - 12 Dec

INTERNATIONAL CONFERENCE ON FUSION REACTOR MATERIALS (ICFRM-5)

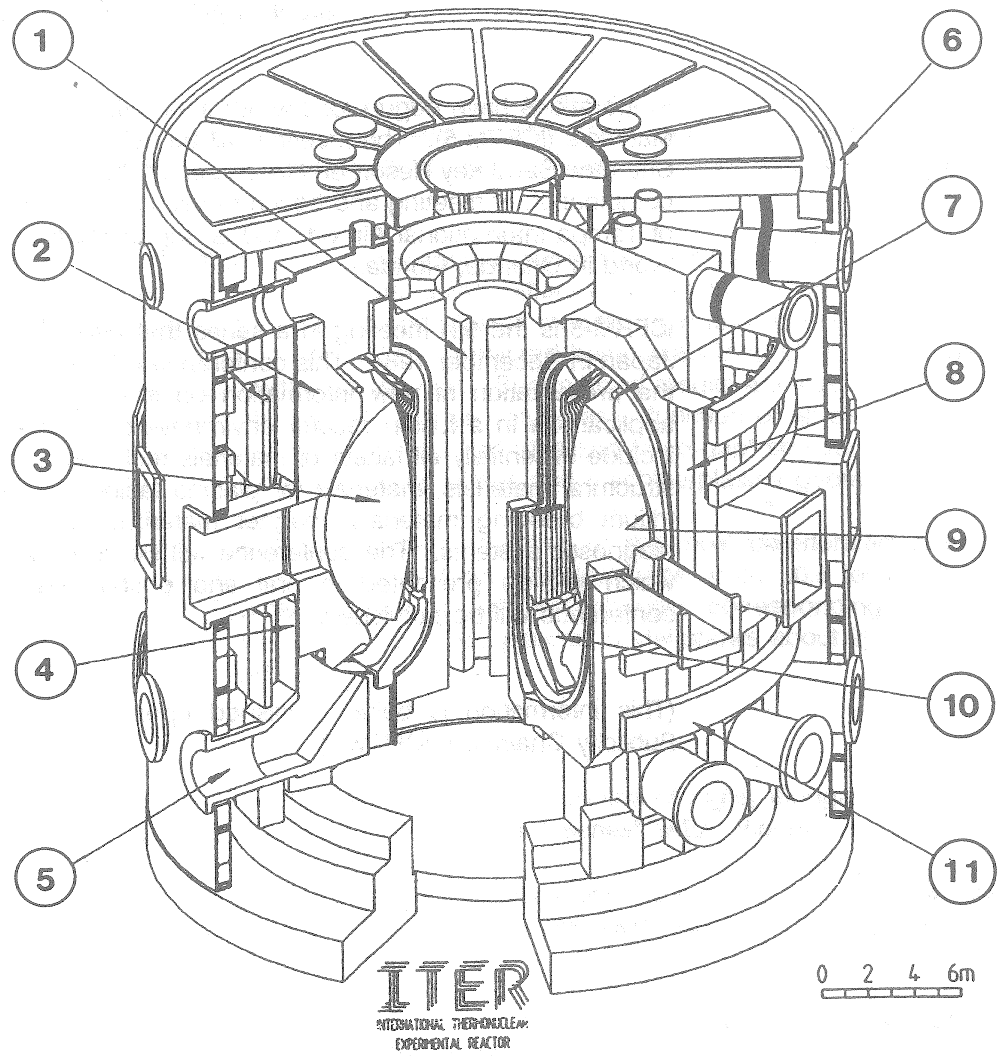
Preparations have begun for the 5th International Conference on Fusion Reactor Materials (ICFRM-5). The meeting will be held in Clearwater, Florida (USA) at the Sheraton Sand Key Resort on November 17-22, 1991. This resort has an excellent combination of meeting and beach/recreation facilities. It is located 21 miles west of Tampa International Airport, and is approximately 2 hours by car from Disney World in Orlando, Florida.

ICFRM-5 is the 5th meeting in a series that began with ICFRM-1, held in Tokyo, Japan in December 1984. This conference will, as in the past, provide a forum for the presentation of new information on the behaviour of materials relevant to applications in a fusion reactor environment. The scope of the conference will include essentially all facets of materials technology for fusion reactors including structural materials, materials for plasma facing and high heat flux components, tritium breeding materials, magnet materials, and ceramics for electrical and diagnostic systems. The conference will include invited and contributed papers which will be presented in oral and poster sessions. Proceedings of the conference will be published.

(This information is being published upon request by Mr. Philip J. Maziasz, Publicity Chairman, ICFRM-5)

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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