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INSIDE

1990 WINTER JOINT WORK SESSION OF ITER

by K. Tomabechi, Chairman, ITER Management Committee

Aiming at further integration of ITER design

The Winter Joint Work Session was held at the ITER technical site, Garching, Federal Republic of Germany, from January 22 through March 23, in order to assess critical technical issues of ITER design, aiming at futher developing of a self-consistent integrated design of ITER. The session was very active, with participation of about ten experts from each of the four Parties as usual and, in addition, many short-term specialists who attended specific design-related technical discussions.

During the session, several specialists' meetings were held at Garching to support ITER designers. Topics covered by the meetings were shielding experiment and analysis, transient electromagnetics and plasma control, reference material data base, magnet materials, beta limits and profiles, and physics and modelling of LHW-assisted ramp-up. The specialists participated in these meetings, discussed the relevant technical issues by delineating the problems, reviewing the state of the art, and defining the further work to be carried out.

Basic design concept confirmed

Based on the results of the home work, which were brought to the Session by each of the home teams, as well as on the outcome of the specialists' meetings convened, the design of ITER progressed during the session. The basic design concept of ITER was confirmed as reasonable with no new technical information for altering the concept drastically, although some critical technical issues require further effort to resolve them. Then the ITER Team developed detailed home tasks to be carried out by the beginning of the coming Summer Joint Work Session which starts on July 2, 1990.

Long-term R&D plans in preparation

In parallel to the design activities, effort was made for developing plans for both long-term physics R&D and long-term technology R&D to be required should the ITER Engineering Design Activities (EDA) be initiated by the ITER Parties. Effort was also made to estimate cost and schedule for such activities.

Based on the information provided by each of the home teams for the long-term R&D to be required for the EDA, preliminary plans were developed by the ITER Team, with assistance given by the cognizant experts who participated in the meetings to review and refine draft documents of the required R&D work. The R&D plans thus developed, though still preliminary, were presented at the 5th ISTAC meeting held on March 21 - 23, at Garching.

The plans will be presented also to the ITER Council in Vienna at the end of April. During the Summer Joint Work Session, the ITER Team will further refine the plans and produce final reports on the subject, following further guidance to be given by the ITER Council.

IMC estimates cost and schedule of an Engineering Design Phase

Concerning estimates of cost and schedule for the EDA, effort was made to define the relevant conditions for such activities with a proposition of producing well documented and technically convincing proposals which would allow the Parties to make a decision for construction commitment of ITER.

Preliminary estimates of cost and schedule for the EDA include the approximate number of professional man years to be required for the design work and the period of time to be needed for achieving technical readiness to enter into the Construction Phase, while the time schedule would be essentially dependent on the pace of the relevant R&D work to be conducted. The results of the estimates were presented to the ISTAC on March 21 - 23 and will be presented also to the ITER Council on April 26.

Next joint work session scheduled

At the end of the Winter Session, the ITER Team developed a draft work plan for the coming Summer Joint Work Session of July 2 through November 16, where the Conceptual Design will be completed and the final report will be written. The major milestones set forth in the draft work plan are: (1) virtually complete machine design by the end of August, (2) complete first draft of the long-term physics and technology R&D for the EDA also by the end of August, (3) present the results of the Conceptual Design Activities at the IAEA Washington Conference on Plasma physics and Controlled Nuclear Fusion Research on October 1 - 6, and (4) complete draft of ITER Conceptual Design Final Report by the beginning of November.

Therefore, the Summer Joint Work Session will be a very busy one, aiming at completion of the ITER Conceptual Design Activities as planned.

ISTAC DISCUSSES LONG-TERM ISSUES

by N. Pozniakov, ISTAC Secretary

Long-term ITER-related R&D plans were the focus of the ISTAC discussion at its recent meeting in Garching on 21 - 23 March. The respective charges were given to the ISTAC by the ITER Council (IC), requesting ISTAC advice in preparation for the IC meeting at the end of April, where the Council plans to examine various aspects of possible Engineering Design Activities (EDA) of ITER.

Broad Long-Term Physics R&D proposed

The draft Long-Term Physics R&D Programme (1991-95), worked out by the IMC, addresses seven different areas:

1. power and particle exhaust physics,
2. disruption control and operation limits,
3. enhanced confinement,
4. heating and fuelling physics,
5. long-pulse operation and optimization of discharge start-up and shut-down,
6. physics of a burning plasma, and
7. diagnostics.

The ISTAC found that the Programme is generally responsive to the Committee's recommendations from its meetings of June and November 1989, where the long-term physics R&D issues were discussed preliminarily. In particular, the diagnostics programme is being developed in response to a specific recommendation by ISTAC. Areas 1 - 4 are important to the design and are to be addressed on existing facilities during the EDA. Areas 5 and 6 relate mainly to the optimization of the operational scenario of ITER. Information in these areas would be useful even if it could be obtained only during the construction phase of ITER, i.e., after 1995. Area 7 is to include the development and testing of prototypes and will require a significant level of ITER-specific funding; key aspects of the diagnostics-machine interface, as well as diagnostics to be used for plasma control and safety purposes, must be addressed before the start of construction.

Categorization of tasks recommended

The Long-Term Physics R&D Programme is comprehensive and encompasses a major fraction of the existing world tokamak activities. At the same time, in view of the extremely broad scope of the overall Programme that will have major impact on ITER, the key recommendation of the ISTAC was that the various R&D tasks should be categorized according to the following criteria:

- tasks that must be fulfilled during the EDA in order to validate the engineering design,
- tasks whose fulfillment during the EDA would allow the optimization of the engineering design beyond minimum requirements, and
- tasks that relate primarily to the optimization of the operational phase of ITER and could be fulfilled even after the EDA.

This approach would allow concentration of efforts on timely fulfillment of the most important tasks to validate the design.

In this connection, the ISTAC noted that several areas needing urgent action had already been identified. Some of these areas are not adequately covered in the existing fusion programmes of the four Parties and would require, therefore, more attention to ensure that the relevant tasks are given sufficient priority.

Detail proposals for long-term technology R&D elaborated

With regard to the Long-Term Technology R&D Plan, the ISTAC found that the ITER Team has done an outstanding job of compiling a comprehensive list of technology R&D requirements, of identifying facilities in the home countries of the Parties that can be upgraded to address these requirements, and of identifying new facilities that are required. Preliminary cost estimates of the long-term technology R&D have been made as well. The technology R&D requirements were defined within each of eight major areas: magnets, containment structures, assembly and maintenance, current drive and heating, plasma facing components, blanket, fuel cycle, and diagnostics. They include both the development of new or extended technology and the development and testing of components.

The ISTAC made some recommendations in view of further development of the Long-Term Technology R&D Plan. The Committee proposed that within each technology area, the R&D tasks should be grouped into the following categories:

- tasks required to validate technology of primary design option,
- tasks required to develop components of primary design option, and
- tasks required to validate technology of backup design option.

Focus on critical problems should be ensured

The relative magnitude of the suggested R&D in the different technology areas should be reviewed to ensure that the most critical problems are being adequately addressed. The possibility of consolidation of R&D tasks for the different technology areas should be provided to ensure full coverage of critical R&D, and to avoid duplication in non-critical areas. In those areas, for which the technology of the primary design option is not yet established, the R&D should be planned to validate a backup option. On the other hand, R&D to support more than one backup option can be justified only in exceptional cases.

The ISTAC believes that its recommendations could improve a mission-oriented approach to the Long-Term Technology R&D essential to produce the technology data base, important for making the decision whether to proceed to the ITER construction.

Also, the ISTAC discussed the IMC estimates of cost and schedule of the EDA. The respective comments will be presented to the ITER Council at its forthcoming meeting. The ISTAC members were informed by the IMC on the status of the design activities. And finally, following the tradition, they exchanged news on recent fusion developments which may have impact on the ITER development.

PROPOSALS ON WAYS AND MEANS FOR EDA OUTLINED

by M. Roberts, Chairman, ITER Council's Ways and Means Working Party and H. Donoghue, Secretary, ITER Council's Ways and Means Working Party

The ITER Council's Ways and Means Working Party met in Vienna from 13 to 16 March 1990 for the third of their three scheduled meetings. Their goal for this meeting was to set down the results of their exploration, within the ITER Conceptual Design Activities (CDA) framework, of "those supporting elements needed for the possible conduct of the Engineering Design of ITER." On target until the last few minutes (thanks to the much-appreciated help of the ITER Council Secretariat), a virus in the computer system effectively quarantined the Final Report! Fortunately, it emerged a few days later, unscathed.

The Working Party has outlined what they believe to be a resonable approach to ITER Engineering Design Activities (EDA).

Approach to EDA based on logical sequence of actions needed

Their starting point was the logical sequence of actions, both technical and non-technical, implied by the technical needs of engineering design. This raised questions, notably on the timing of site selection and of decisions on construction. In addition, the Working Party bore in mind the ITER Council's view, expressed in its December 1989 letter to the Parties, that "the appropriate next technical step in ITER should be engineering design" and that "it would be highly desirable to avoid any unnecessary hiatus in the technical work after completion of the Conceptual Design Activities at the end of 1990." As a consequence, the Working Party considered it important to explore, in some detail, the logical sequence of tasks, technical and otherwise, which would be necessary immediately after the completion of the CDA, whether or not these tasks are normally seen as part of engineering design activities.

This logical sequence from the end of the CDA to the start of construction is reflected in all of the Working Party's explorations of "supporting elements." For example, in exploring task-sharing for the EDA, their concern was not only what models might be helpful but also at what point would all the elements needed for the sharing of tasks according to these models become available. Similarly, in exploring organization and management for the EDA, they bore in mind the evolution of tasks to be undertaken.

Members of the IC Ways and Means Working Party

Broad spectrum of the EDA-related issues explored

The Working Party's report contains an exploration of the possible organization and management of the EDA, an approach to task-sharing, matters associated with the EDA work site, financial matters, handling of intellectual property, possible formal arrangements (including the role of the IAEA), and matters related to the construction site and to environment, safety and licensing. It also contains indications of points on which further exploration would be necessary and useful. Discussions with the ITER Management Committee (IMC) were very useful to the Working Party in its work.

Idea of an umbrella Agreement

As regards a formal framework for the EDA, the logical sequence led the Working Party to the idea of an umbrella Agreement covering the whole of the EDA, with protocols governing the tasks and details, in due course. In their view, the first protocol, defining a first set of tasks (outlined in the Report) could be agreed simultaneously with the Agreement, allowing work to start immediately. Their idea is that the Agreement could be in place by end-1990. The Working Party also considers that a Declaration by all four Parties at high level, indicating their intention to proceed to a joint EDA, could provide a help to the whole process.

Editor's Note

Continuing the series on the major fusion research centres of the ITER Parties, the Newsletter introduces JET Joint Undertaking of EURATOM, which is a large-scale international project. The Joint European Torus, JET, is one of the world's largest tokamak devices implementing advanced technology and demonstrating high operation parameters. Both in organization and science and technology, the experience of JET is essential for a Next Step device which may be ITER.

THE JET PROJECT AND ITS IMPACT ON NUCLEAR FUSION RESEARCH

by P.H. Rebut, JET Joint Undertaking, Abingdon, U.K. *)

Necessity of international co-operation

During the early 1970's, it was clear that to achieve plasma conditions near those in a fusion reactor, much larger experiments were required, which were likely to be beyond the resources of any individual country. In 1973, it was decided in Europe that one large tokamak device, the Joint European Torus (JET), should be built as a joint venture, and a design team was set up to prepare a design. Approval to proceed with the project was given at the end of 1977.

On 1st June 1978, the formal organization of the project, the JET Joint Undertaking, was set up near Abingdon, U.K. The Project Team is drawn from EURATOM and the fourteen member nations (the twelve EC countries, together with Switzerland and Sweden). Funding, which is currently at the level of $\sim$100 M ECU per year, is provided 80% by EURATOM, 10% by U.K. as host nation, and the remaining 10% by members. The construction of JET, its power supplies and buildings was completed on schedule and broadly to budget by mid-1983. The experimental research programme started that year.

JET is the largest project in the co-ordinated programme of EURATOM, whose fusion programme is designed to lead ultimately to the construction of an energy-producing reactor. Its strategy is based on the sequential construction of major apparatus such as JET, the Next European Torus (NET), and DEMO (a demonstration reactor), supported by medium-size, specialized tokamaks.

Objective of JET

The objective of JET is to obtain and study a plasma in conditions and dimensions approaching those needed in a thermonuclear reactor. This involves four main areas of work:

1. to study various methods of heating plasmas to the thermonuclear regime,

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*) Dr. P.H. Rebut is a member of the ISTAC

2. to study the scaling of plasma behaviour as parameters approach reactor range,
3. to study the interaction of plasma with the vessel walls and how to continuously fuel and exhaust the plasma, and
4. to study the production of α-particles in the fusion of deuterium and tritium atoms and the consequent heating of plasma by these α-particles.

The JET machine: parameters and performance

JET is a tokamak device, whose overall view is shown in Fig. 1. It is now about midway through its original experimental programme. The technical design specifications of JET have been achieved in all parameters and exceeded in

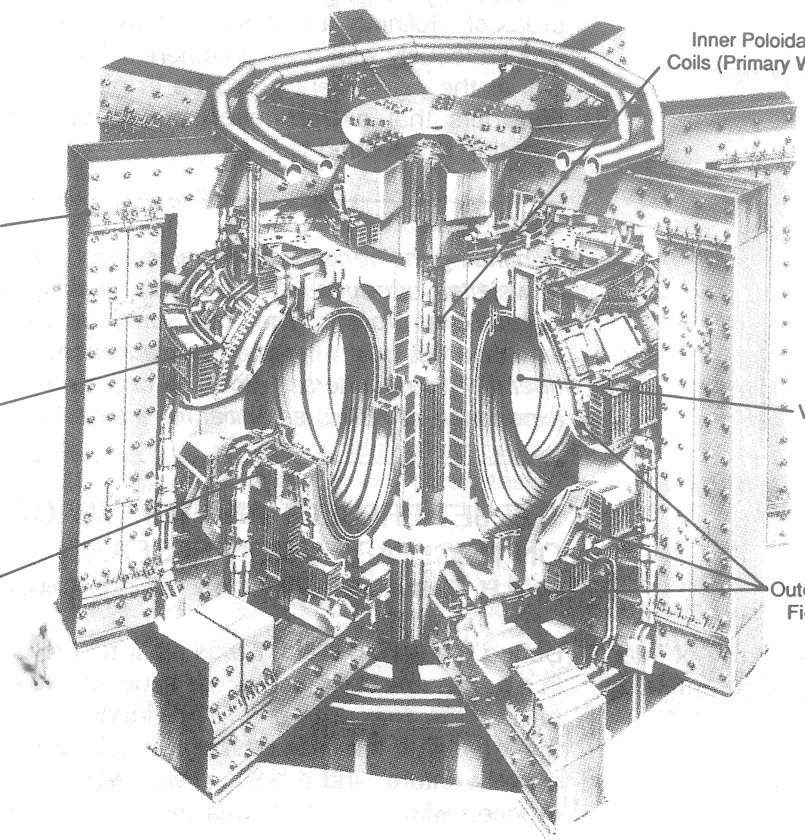


Fig. 1. Overall View of the JET Tokamak

several cases (see Table I). The plasma current of 7 MA and the current duration of up to 30 seconds are world records and are over twice the values achieved in any other fusion experiment. The neutral beam injection heating system has been brought up to full power (~21 MW) and the ICRF heating power has been increased to~18 MW in the plasma. In combination, these heating systems have provided 35 MW power to the plasma.

TABLE I. JET PARAMETERS

Parameter	Design Values	Achieved Values
Plasma Major Radius (R_o)	2.96 m	2.5 - 3.4 m
Plasma Minor Radius (Hor)(a)	1.25 m	0.8 - 1.2 m
Plasma Minor Radius (Vert)(b)	2.1 m	0.8 - 2.1 m
Toroidal Field at R_o	3.45 T	3.45 T
Plasma Current	4.8 MA	7.0 MA
Neutral Beam Power	20 MW	21 MW
ICRF Heating Power	15 MW	18 MW

JET results with carbon wall

Recently, JET has concentrated on studying interaction of the plasma with the wall and the effect of impurities generated. During 1987/88, the plasma vessel was operated with carbon tiles on the inside of the vacuum vessel walls to provide a low-Z material facing the plasma. In this situation, plasma temperatures up to 23 keV were reached and the plasma densities (up to $\sim 2\text{x}10^{20}\text{m}^{-3}$) and energy confinement times (up to 1.5 s) were within the range required in a reactor. These values, achieved in individual experiments, were not all reached simultaneously, however. Operating JET with a magnetic limiter configuration (X-point configuration), two regimes of energy confinement have been observed: a higher confinement regime (H-mode), having energy confinement times about twice the normal values (L-mode). In both regimes, confinement degradation occurs in that the plasma thermal energy does not increase proportionally to the heating power. Therefore, considerably more power is needed to increase the plasma temperature and energy. In these experiments, carbon impurities in the plasma at high power levels became a problem that hinders further enhancement of plasma parameters.

Impact of impurities on JET plasmas

When JET operated with all-carbon walls, impurities created the following problems.

- The production of impurities increased with the input power to the plasma. In ohmic discharges, Z_{eff} varied as I_p/n where I_p is the plasma current and n is the density.
- At high power, the heat load on the tiles was too high, and the plasma evolution presented a catastrophic behaviour, the so-called "carbon catastrophe" or "carbon bloom." Increased plasma dilution, increased power radiation, reduced neutral beam penetration and a threefold fall of the fusion yield resulted from the carbon influx.
- For lower input power with long duration, problems were also encountered. Without fuelling, deuterium was pumped by the carbon and replaced by impurities, resulting in a large dilution of the plasma.
- The maximum density achieved without occurrence of plasma disruptions appeared to be limited by edge radiation.

TABLE II. JET PARAMETERS WITH CARBON AND BERYLLIUM FIRST WALL

	With C	With Be
ICRF Power Input (P_{RF})	18 MW	18 MW
NB Power Input (P_{NB})	21 MW (at 80 kV)	18 MW (at 140 kV)
Total Input Power (P_{tot})	33 MW	33 MW
Plasma Energy (W_p)	11 MJ	13 MJ
Electron Temperature (T_e)	12 keV	13 keV
Ion Temperature (T_i)	23 keV	27 keV
Density (n_D)	$2\text{x}10^{20}\text{m}^{-3}$	$4\text{x}10^{20}\text{m}^{-3}$
Confinement Time (τ_E)	up to 1.5 s	up to 1.8 s
Effective Charge (Z_{eff})	2 - 5	1.2 - 4
Dilution Factor (n_D/n_e)	0.4 - 0.75	0.75 - 0.95
Neutron Rate (ns^{-1})	$1\text{x}10^{16}\text{s}^{-1}$	$4\text{x}10^{16}\text{s}^{-1}$
Fusion Product ($n_D T_i \tau_E$)	$2.5\text{x}10^{20}$ m^{-3}keVs	$8\text{x}10^{20}$ m^{-3}keVs

JET results and achievements with beryllium

In 1989, to further improve JET results the carbon tiles and inside of the vacuum vessel were first covered with evaporated beryllium and, later, the belt limiter and RF antennae tiles were replaced by beryllium tiles. The effect of a beryllium first-wall on the impurity influxes was as follows:

- oxygen impurity was essentially eliminated from the plasma,
- carbon was the main impurity, but its influx was lower by a factor of ~2 than with carbon walls,

- the effective charge, Z_{eff}, was significantly reduced in ohmic plasmas (down to 1.2) and with strong additional heating (down to <2), and
- a severe carbon influx ('carbon bloom') was still a problem for inner wall and X-point plasmas; it remains to be a serious limitation in H-mode studies.

Reduced impurity levels have allowed operation at higher densities and improved the general JET performance. The improved results, set out in Table II, are following.

- The pumping of deuterium with Be was more efficient than with carbon walls. This permitted low density and high temperature (27 keV) operation for times >1s.
- The density limit increased to (nRq/B)~30 (with a record peak density of $4x10^{20}m^{-3}$ with pellet fuelling). This limit is principally a fuelling limit and not a disruption limit as found with carbon limiters.
- H-modes were created with ICRH alone for periods >1s. Their confinement characteristics were similar to those with neutral beam (NB) heating alone.
- β-values up to the Troyon limit were obtained in low field (B_T = 1.4 T) double-null X-point plasmas.
- The neutron yield doubled to $3.7x10^{16}ns^{-1}$ and the equivalent fusion factor Q_{DT} increased to~0.8.
- The fusion parameter ($n_D \tau_E T_i$) increased to $8\text{-}9x10^{20}m^{-3}$skeV for both high (20 keV) and medium temperaturs (9 keV).

α-particle simulation in JET

The behaviour of α-particles has been simulated in JET by studying fast particles such as 1 MeV tritons, and He^3 and H minority ions accelerated to energies of a few MeV by ICRF heating. The fast population (in the MeV range) has up to 50% of the plasma's stored energy and possesses all the characteristics of ignited plasmas (except for anisotropy). Up to 100 kW of fusion power has been generated and Q~1% has been reached. Comparison of the measurements with theoretical predictions suggest that the trapping and slowing down of the fast particles are close to classical expectations. This is particularly encouraging for α-particle heating as the fast particle parameters are all in the range expected for D-T α-particles.

Achieving control of the impurity influx into the plasma is a prerequisite for building a tokamak reactor. For high-Z impurities, radiation losses may prevent attainment of the temperature required for ignition. The presence of low-Z impurities dilutes the concentration of reacting ions and therefore reduces the α-particle power. Under present conditions, the lifetime of the plasma-facing components would be severely limited. The degree of impurity control which might be achieved has a direct consequence on the size of the Next Step device.

By virtue of its size, its plasma performance and its long pulse capability, JET is in a good position to address the problem of impurity control in the basic geometry for the next step tokamak. A new phase is proposed for JET which aims to:

Proposed new phase for JET

demonstrate effective methods of impurity control in operating conditions close to those of the next step tokamak; that is in a stationary plasma of "thermonuclear grade" in an axi-symmetric pumped divertor configuration.

The objectives of the pumped divertor in JET are:

- control of impurities in the plasma,
- decrease of the heat load on the target plates,
- control of the plasma density, and
- demonstrate exhaust capabilities.

The progress made in controlling these impurities should also improve the - particle production and heating during the JET tritium phase. A cross-section of the proposed divertor arrangement is shown in Fig. 2.

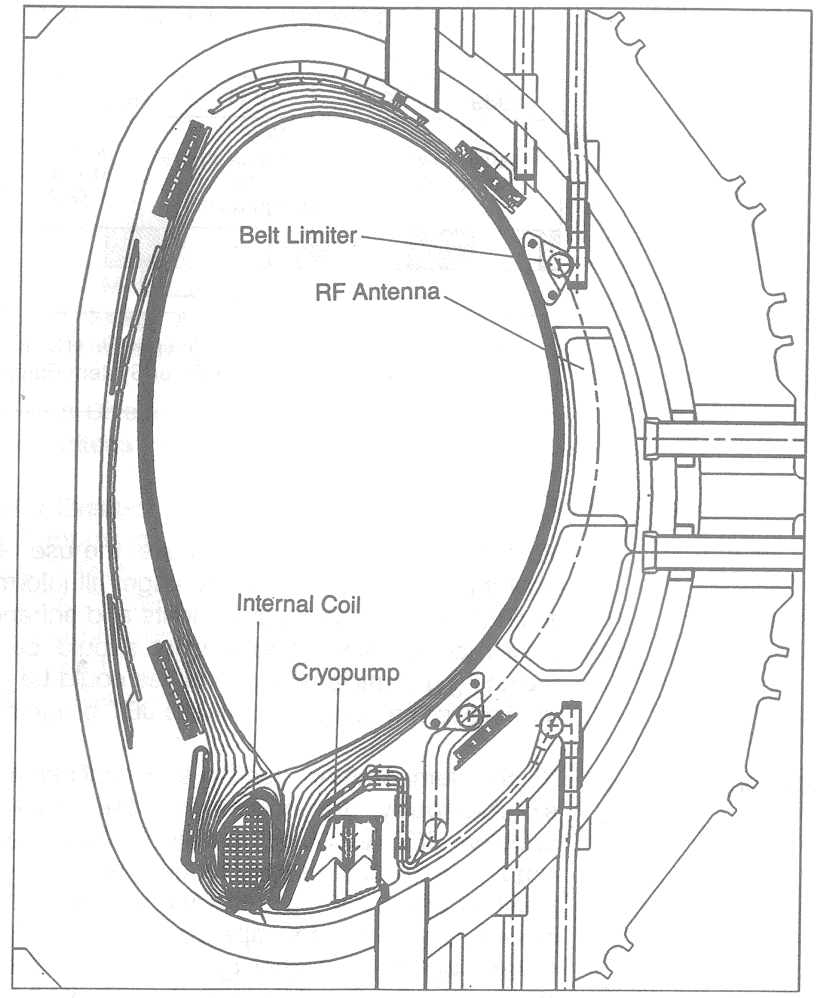


Fig. 2. A Cross-Section of the Proposed JET Axi-Symmetric Pumped Divertor

Methods of controlling impurities to be considered and tested on JET during the new phase are either passive or active. The objective of passive impurity control is to reduce the production of impurities at source by proper choice of plasma-facing components. The present approach on JET is towards low-Z materials such as beryllium or beryllium carbides, but alternative routes may have to be considered and tested. The objectives of active impurity control are: minimize the impurity content in the plasma; reduce the heat load on divertor plates to values which can be sustained continuously; control the plasma density; and exhaust the ashes in a reactor.

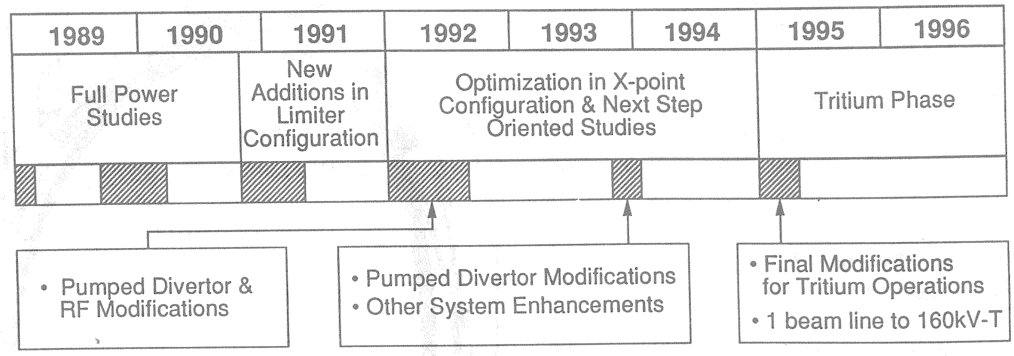
Expected JET performance and tentative schedule for the new phase

In the pumped divertor configuration, the capability of JET should allow:

- 6-MA single-null X-point configuration for up to 10 s with an axisymmetric pumped divertor in the bottom of the vacuum vessel,
- 3-MA double-null X-point operation for up to 20 s at a magnetic field of 3.4 T, and
- an increase by more than a factor of 2 of the expected α-particle power.

A schedule for the JET programme in the new phase is shown in Table III. The earliest date for a pump limiter in JET is 1992. In the light of experimental results, further optimization would be likely $\sim$18 months later.

TABLE III. JET PROGRAMME IN THE NEW PHASE



To provide time for the new phase, the use of tritium in JET would be postponed until the end of 1994. At this stage, all information on particle transport, exhaust and fuelling, first wall requirements and enhanced confinement regimes needed to construct the Next Step device, should be available. Final tests with tritium, including α-particle heating studies could be performed in the following two years, leading to the completion of the JET programme by the end of 1996.

Impact on Next Step devices

Plasma temperature, density and confinement values already achieved, but not simultaneously, are individually close to the requirements of the next step. In addition, JET results on scaling of these parameters have allowed some of the requirements of a reactor to be specified. In particular, the next step tokamak must be about 2.5 times the linear dimensions of JET for a similar magnetic field, have a plasma current capability of 25-30 MA (ignition could be reached at a lower current but with a higher q), and an output of several GW. The plasma must be maintained for very long times, such as 1 hour, rather than the 20-30 seconds bursts presently used in JET. Sufficient knowledge now exists to design such a device, but a number of plasma engineering problems remain to be solved. These relate mainly to the interaction of the plasma with the vessel walls, e.g. control of impurities, fuelling and exhaust. JET has the capability of studying these problems and will be doing so in the second half of its programme.

JET is providing important information for a Next Step device, with specific emphasis in such areas as:

- control of impurity content, with full additional power for sufficiently long times (in quasi-steady state),
- improved understanding of particle and impurity transport, prediction of pumping requirements and simulation of transport of fusion ashes,
- achieving steady-state in enhanced confinement regimes (i.e. H-modes, peaked profiles),
- testing lifetime of divertor plates and assessing first-wall materials to be used,
- fuelling methods, in particular, high-speed pellet injection system, and
- performing technology tests and assess systems for the Next Step (tritium plant and remote handling, heating and current drive, etc.)

Contribution to ITER activities

JET is already making a substantial contribution to the ITER activities, which will be continued, particularly with regard to specific questions posed by ITER, as follows:

- disruption control and operational limits,
- enhanced confinement,
- profile effects,
- long-pulse operation (including non-inductive current drive), and
- simulation of α-particles.

ITER EVENTS CALENDAR - 1990

ITER Council Meeting	Vienna	26 - 27 Apr
Joint Work Session	Garching	2 July - 16 Nov
ITER Council Meeting	Washington	8 - 9 Oct
ISTAC Meeting	Vienna	28 - 30 Nov
ITER Council Meeting	Vienna	13 - 14 Dec

Specialists' Meetings at Garching in support of joint design work:

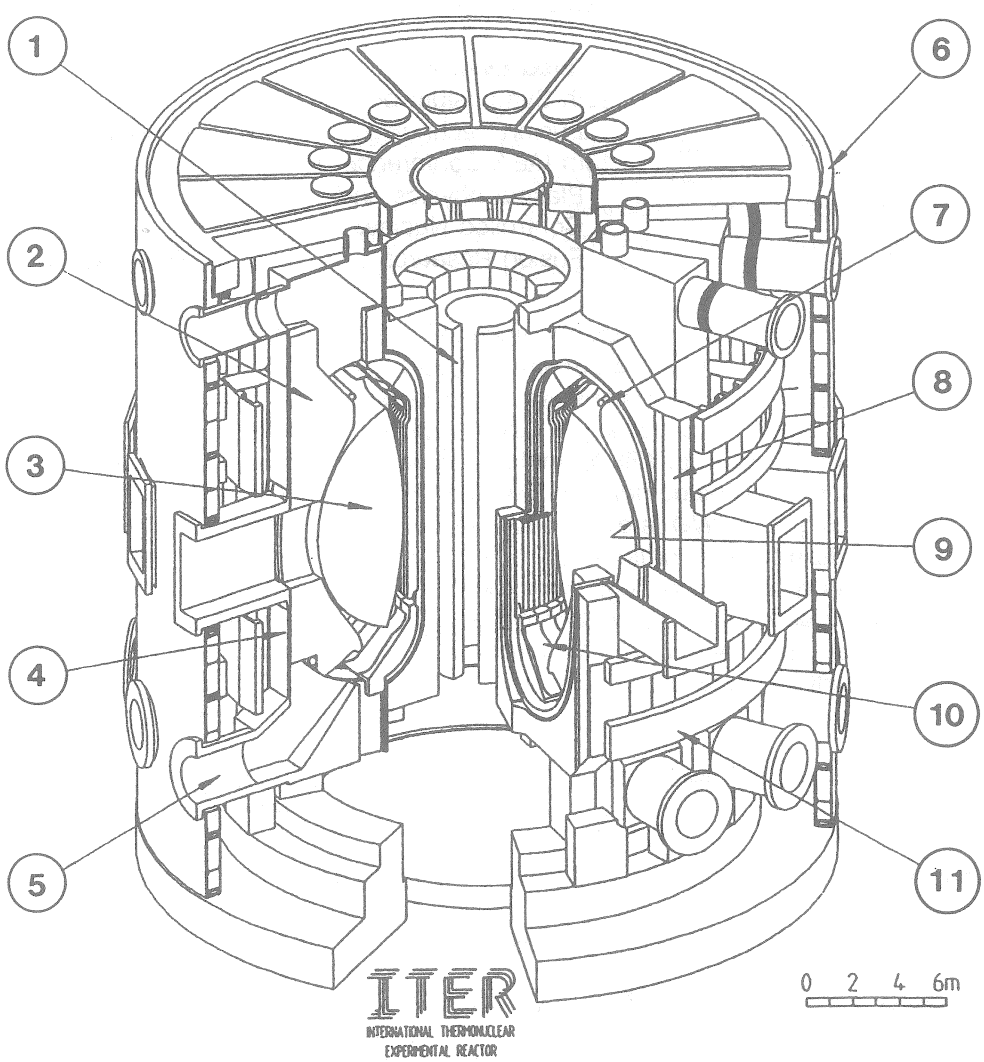
Plasma Operation Control in ITER		23 - 27 July
Advanced Divertor		20 - 24 Aug
Design Criteria		10 - 14 Sep

Related Events:

16th Symposium on Fusion Technology	London	3 - 7 Sep
13th IAEA Conference on Plasma physics and Controlled Nuclear Fusion Research	Washington	1 - 6 Oct

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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