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INSIDE

ITER PROGRESS CITED

Recent worldwide conferences have heard of the encouraging progress in ITER Activities.

World Energy Conference

Speaking before the World Energy Conference in Canada, in September 1989, on the subject "World Progress toward Fusion Energy," ITER Council Chairman John Clarke assessed the Activities as "the largest fusion collaboration ever attempted." He reported that "the Definition Phase of the ITER activity has been concluded successfully, and the Design Phase will be finished in 1990." Dr. Clarke outlined the major objectives of the Activities, described the direction and management aspects and provided figures illustrating the scale of the Parties' contributions including both the core design team work and supporting R&D.

General Conference of the IAEA

The Director General of the International Atomic Energy Agency, Dr. Hans Blix, in his statement to the 33rd General Conference of the IAEA, 25-29 September 1989, cited ITER as a project "of major importance to the international fusion community." Also, he informed the Conference that "preliminary discussions will now start about a possible second phase."

ITER CONFINEMENT WORKSHOP

by V.S. Mukhovatov, Physics Project Unit

Confinement scaling expressions based on experimental data are often used for predicting the energy confinement time in tokamaks. The ITER Confinement Workshop was held in Garching from 31 July through 11 August 1989 with the main objective to reassess the existing scaling expressions for the main confinement regimes (L- and H-mode) and to estimate their uncertainties in predicting the energy confinement time for ITER. Twelve experts from four Parties and seven permanent members of the ITER design team took part in the workshop.

New ITER scalings for L-mode have been proposed

The global L-mode confinement data base established by S. Kaye, Princeton Plasma Physics Laboratory, in 1984 was recently extended with data from JT-60, TFTR, JET, DIII-D and other experimental teams as a part of the ITER joint activity and it is now called the "ITER data base." Based on the analysis of this ITER data base, two new L-mode scaling expressions for the energy confinement time, an ITER power-law scaling and an ITER offset-linear scaling, were proposed at the workshop. The reasons for the existence of a large number of scaling expressions, which describe the available tokamak data equally well but give divergent projections for ITER, was clarified. A combination of geometrical parameters was identified to vary little in the data base (i.e., $f_s=0.3Ra^{-0.75}k^{0.5}$ that proved to be close to unity for all existing tokamaks where R and a are the major and minor radii of the plasma column and k is the cross-section elongation). It was shown that most of the commonly known scaling expressions can be reproduced by multiplying the proposed ITER scalings by a power of this factor [e.g., $(f_s)^p$ with $-0.7<p<0.7$] and by other factors for which the dependence considerably differs for different tokamaks, i.e., n_e, A_i and q (n_e is electron density, A_i is the ion mass and q is the safety factor). To improve the accuracy of the scalings, the data with a strong variation in f_s are necessary. Also, the physical reasons for the differences in the dependences of energy confinement time on n_e, A_i and q in different tokamaks need to be clarified.

Joint H-mode data base is needed

The ITER H-mode confinement data base is not yet established, though urgently needed. The results from an analysis of the JET/ASDEX H-mode data base were presented at the workshop. A JET/DIII-D comparison is in progress. It was noted, however, that a two-tokamak data base is not enough to obtain a scaling with respect to the geometrical variables (i.e., to A=R/a, k and a) that are usually kept constant in any particular device. A four-tokamak comparison (i.e., JET, ASDEX, DIII-D and JFT-2M) was suggested for this purpose that requires creation of the joint H-mode data base.

Application of advanced statistical techniques (regression on principal components, ridge regression, etc.) for tokamak data analysis was discussed. To improve the present global analysis, explicit account of the fast particle energy content, the power-deposition and radiation profiles, and effects of the edge localized modes should be considered. For simulation of the ITER operation scenario, a simple transport model was suggested that uses the ITER power-law scaling as a basis.

BLANKET AND SHIELD DESIGN ACTIVITIES

by C. Baker, Head of Blanket Design Unit*

The ITER Blanket Design Unit, supported by each Party's home team, has been involved in shielding analysis (including benchmark problems) and blanket design and concept selection. The blanket design work is concerned with the "driver" blanket which is intended to supply the tritium for ITER for the technology phase.

Shielding Analysis

Shielding benchmark calculations

The results of the shielding benchmark problems defined in the February/March 1989 work session show that, for the most simplified problem of a homogeneous mixture of steel and water, the calculated nuclear responses (heating rates, radiation damage, etc.) in the toroidal field (TF) magnet were consistent to within 10-15 %. This is considered reasonable given the different data libraries and codes used. There were larger discrepancies in the other, somewhat more complicated problems, which require further evaluation. A limited number of additional cases (based on some of the same problems but with more detailed input specifications) have been suggested for further homework activities.

** C. Baker, A. Antipenkov, W. Daenner, T. Kuroda and H. Takatsu*

Design of the inboard shield region

Several optimization studies of the composition of the inboard shield region for both the physics and technology phases have been carried out. Based on these studies a borated-steel/water shield is recommended for use with "thin" (10-20 cm thick) vacuum vessels and a steel/water shield with 3-5 cm of a Pb/B_4C layer for "thick" (30-40 cm) vacuum vessels. With these shields, the total nuclear heating in the TF coils in the inboard region will be 10-12 kW in the physics phase [with a 75 cm first wall/blanket/shield/vacuum vessel (FW/B/S/VV) radial build originally proposed for the physics phase] and 5 kW in the technology phase (85 cm radial build). ITER has now decided to incorporate an 84 cm FW/B/S/VV radial build in both the physics and technology phases. The maximum insulator dose in the technology phase is estimated to be 2-3 x 10^9 rads. (These numbers include appropriate design safety factors.)

Total nuclear heating in the TF coil is 10 - 20 kW

The shielding issues in the region behind the divertor received special attention. With 60 cm of shield/vacuum vessel, the estimated total nuclear heating in the TF coils behind the divertor is 5-8 kW with similar shield materials as in the inboard region. Thus, the total nuclear heating in the TF coil in both the inboard and divertor region is 10-13 kW in the technology phase and 15-20 kW in the physics phase.

The recommended outboard FW/B/S/VV thickness to achieve 2.5 mrem/hr one day after shutdown (for hands-on maintenance) is 150-160 cm in the technology phase. This is provided by the current design. It will be necessary to provide sufficient shielding around all penetrations (e.g. neutral beam lines, vacuum pumping ports, etc.).

Shielding analyses of various penetrations have been done including diagnostic ports, neutral beam injection (NBI) ducts, the divertor/vacuum pumping duct, and assembly gaps between blanket/shield segments. For 1 cm-wide assembly gaps, safety factors of 2 on integrated responses and 3 on local responses are adequate to account for streaming effects. The minimum shield for the NBI duct (at the place of closest approach to the backside of the coil) is 35 cm. The winding pack in the region of the divertor/vacuum pumping duct is well shielded by 45 cm of shield and vacuum vessel. Initial considerations suggest this can be reduced to 35-40 cm. The fast flux peaking factor (defined as the fast neutron flux at the end of a channel compared to the neutron flux penetrating at 150 cm-thick shield) for diagnostic channels ranges from 10^4 for a 4-cm diameter straight channel to $5x10^6$ for a 20-cm diameter channel.

Present ITER configuration meets shielding design criteria

In general it appears that the present ITER configuration will meet all major shielding design criteria. More detailed analysis will be performed to confirm this.

Blanket Concepts

Three basic blanket concepts have been considered

The three basic driver blanket types (solid breeder (SB), aqueous salt (AS) and lithium lead eutectic (LiPb)) have been considered. Options for the solid-breeder (Japan, US, EC) blankets include layered, breeder-in-tubes and breeder-out-of-tubes with the breeder in the form of sintered ceramics or pebbles. The aqueous-salt options (US, EC) included layered and tube designs. The lithium-lead option (USSR) is based on tube geometry. All concepts use low temperature water as the coolant and 316 stainless steel as the structural material.

Estimates of the net tritium breeding ratio (accounting for the divertor region, horizontal ports, side walls/gaps of sectors, plasma stabilization loops, first wall tiles, etc.) have been made. Estimates range from 0.70 for somewhat conservative first wall assumptions (2-cm thick carbon tiles and 2 cm-thick steel) and no tritium from test modules, to up to 0.95 for optimistic first wall assumptions (no tiles and 1-cm steel thickness) and some tritium from test modules. Even with optimistic assumptions, it does not appear that ITER can be self-sufficient in tritium. However, it is expected that the difference can be made up by external supplies.

The SB concepts use Be for neutron multiplication. Tritium is recovered in-situ by flowing low pressure helium through the solid ceramic material. In the AS concept, tritium is bred in the water coolant by adding a lithium salt (e.g. LiOH) to the water. In the LiPb concept, the eutectic material is mainly solid during plasma burns and is melted during off-periods so that the eutectic can be removed from the blanket to recover the tritium.

Comparison of different types of blankets

A detailed discussion was held by experts at a 1989 summer work session to identify the advantages and concerns of each of the blanket options. Based on these discussions, the "best" blanket in several key areas was identified as follows:

Key area	"Best" blanket
+ Radiation tolerance of breeder/ multiplier	LiPb
+ Flexibility of changing from non-breeding to breeding	AS
+ Design tolerance	AS, LiPb
+ Safety	SB
+ Long-term waste	all are about the same
+ Tritium breeding	all are about the same
+ Demonstrated tritium recovery	AS
+ Breeder R&D data base and on-going programme	SB
+ Reactor relevance	1. SB 2. LiPb

Solid-breeder concept has been selected as the "first option"

Based on these considerations, the solid-breeder concept has been selected by the Blanket Design Unit as the "first option" to be incorporated in the reference ITER conceptual design. It is recommended that design and R&D activities continue on the other two options, because all the options have one or more key technical issues that need to be resolved so that a final choice can be made.

Solid-Breeder Blanket Concept

We describe here further details of the proposed solid-breeder design candidates for the "first option" blanket for ITER.

Multilayer configuration

A solid-breeder water-cooled blanket has been designed by the US for ITER based on a multilayer configuration for the inboard and outboard sections. The blanket uses lithium oxide for tritium breeding. The material forms are sintered products for both materials with about 0.8 density factor. The lithium-6 enrichment is 95 % to maximize the tritium breeding ratio. The use of high lithium-6 enrichment reduces the solid breeder volume required in the blanket and the total tritium inventory in the solid breeder material.

The blanket uses one and two thin solid-breeder layers in the inboard and the outboard sections, respectively. Each breeder layer has thin clad steel and is inserted inside a beryllium zone. The water coolant removes the heat from the beryllium zone surfaces which are parallel to the first wall. The blanket has a poloidal manifold for each segment to supply the water coolant for each blanket module in the segment. The water flows in the radial direction in the side walls, in the toroidal direction to cool the blanket module, and in the radial direction to exit from the other side walls. The blanket modules are made by hot vacuum forming and diffusion bonding a double wall structure with integral cooling channels.

Pebble-type blanket

The Japanese team has pursued a pebble-type blanket in which the Li_2O breeder in the form of small spheres (1 mm diameter) is filled. Two types of blankets have been designed for ITER, i.e. MIXture type and SANDwich type.

In the MIX-type blanket, the Be neutron multiplier is filled also in the form of small spheres (1 mm diameter) which is homogeneously mixed with Li_2O pebbles. The optimum mixing ratio of Li_2O and Be in this design is 25 % and 75 % respectively. An enrichment of Li of 30 % is adopted to enhance the breeding performance, however Li_2O without lithium-6 enrichment may have a high enough tritium breeding ratio depending on allowable space for the blanket installation and also by future design efforts.

The breeder temperature is maintained within the range of 450°C - 600°C (nominal) by an arrangement of coolant tubes and a He thermal gap around the coolant tubes. Since the temperature limits for Li_2O is 400°C - 1000°C, power variations of -10 % to +70 % from the nominal condition are permissible for this design. Permissible power variations are design-dependent and can be adjusted by selecting the nominal temperature range.

A SAND-type blanket has been developed as a concept in which the atmospheres of Li_2O and Be are separated from each other. It is one of the approaches which can be applied if the compatibility of Li_2O and Be is a large concern.

The tritium generated in the solid-breeder blankets is continuously recovered by a He purge stream of 0,1 MPa. The estimated tritium inventory in Li_2O in these blankets is from 10 to 190 g with 1% H_2 addition in the purge stream, excluding the Be inventory. More experimental data on tritium release/retention from irradiated Be is required.

Some other options are being considered

The EC hometeam is currently in the middle of an evaluation and selection phase during which three options are being considered: a PIN concept, a SLAB concept, and a PEBBLE BED concept. One of them will enter the predesign phase and will be the focus for the related R&D programme.

All three concepts employ water cooling in the poloidal direction, which is in accordance with the EC first wall concept. A further common feature is the method of breeder temperature control via a gas gap, either with pure helium or with a helium/neon mixture.

IAEA ACTIVITIES IN SUPPORT TO ESTABLISHING THE DATA BASE FOR POWER AND PARTICLE EXHAUST FOR ITER

by R.K. Janev, Division of Physical and Chemical Sciences, IAEA

Establishment of a credible data base is a prerequisite for resolution of the power and particle exhaust problem

The problem of power and particle exhaust has been identified as one of the critical issues for the conceptual design of ITER. This problem combines several inter-related physics and technology areas including the edge-plasma physics, plasma-material interactions, impurity control, plasma-facing materials, etc. The establishment of a credible data base for the underlying physics and candidate materials is an obvious prerequisite for validation of the currently suggested concepts and for exploring new ones to resolve the power and particle exhaust problem. The recently accepted ITER design-related physics R&D tasks fully reflect this approach (e.g. F. Engelmann, ITER Newsletter, Vol. 2, No. 7, p. 3).

Following the general IAEA policy for support to ITER Conceptual Design Activities, the IAEA Atomic and Molecular Data Unit is currently conducting several activities oriented on establishing a number of physics data bases required in the characterization of some of the aspects of the power and particle exhaust problem for ITER. These activities are conducted on an international scale and include participants outside the four ITER Parties.

Three broad programmes are included in these activities:

1) Establishment of a data base for the gas-phase atomic and molecular processes (both collisional and radiative) taking place in the plasma edge.
2) Establishment of a data base for the plasma-material interaction processes, and
3) characterization of the thermal response of plasma-facing components under off-normal conditions.

Contents of the IAEA programmes are defined through interaction with the ITER team

The first two programmes are aimed at providing a data base for a self-consistent description of the edge plasma behaviour and plasma-wall coupling. The third programme is devoted to characterization of candidate materials exposed to high heat fluxes (disruptions, effects of run-away electrons, etc.). The contents of all these programmes have been defined through a continuous interaction with the ITER Physics Group and the pertinent segments of the atomic, solid state and plasma physics communities.

The plasma edge atomic molecular data programme is the most advanced one, and integrates a considerable part of atomic physics community. A co-ordinated research project involving twelve laboratories, the International Atomic Data Centre Network (with participation of ten national atomic data centres), and several individual research projects are currently involved in the creation of the required plasma edge atomic data base. This data base should provide a sound basis for accurate modelling of the impurity and neutral particle transport in the plasma edge, for calculations of plasma edge radiative cooling, helium transport and exhaust, for plasma edge diagnostics, and for exploring the possibilities of enhancement of radiative capabilities of the scrape-off layer and of an intense dissociative divertor plasma cooling.

The plasma-material interaction data activity is focussed on the backscattering, erosion/redeposition and other material release processes. A systematic characterization of these processes for both low- and high-Z materials, including clean and realistic plasma environment conditions will be the subject of a broad co-ordinated research programme to be initiated by the end of this year. The establishment of the data bases for particle reflection and physical sputtering has already been started.

The activity related to the data base on material response to high heat fluxes and material behaviour during off-normal events is in its initial stage. An IAEA experts meeting is envisaged for the next year with the task to categorize and rank different candidate plasma-facing materials with respect to their behaviour under off-normal conditions, and for defining an IAEA co-ordinated research programme in this area.

High responsiveness of the international scientific community

It should be emphasized that in all actions undertaken so far by the IAEA in the development of the above-mentioned activities, the responsiveness of the international scientific community was very high. This fact demonstrates that the ITER design is accepted as a truly international scientific endeavour and that the IAEA efforts to integrate a significant part of the work potentials outside the four Parties in support of the ITER design database reflect this generic scientific interest.

ITER EVENTS - 1989

ISTAC Meeting	Los Angeles	16 - 18 Nov
ITER Council Meeting	Vienna	30 Nov - 1 Dec

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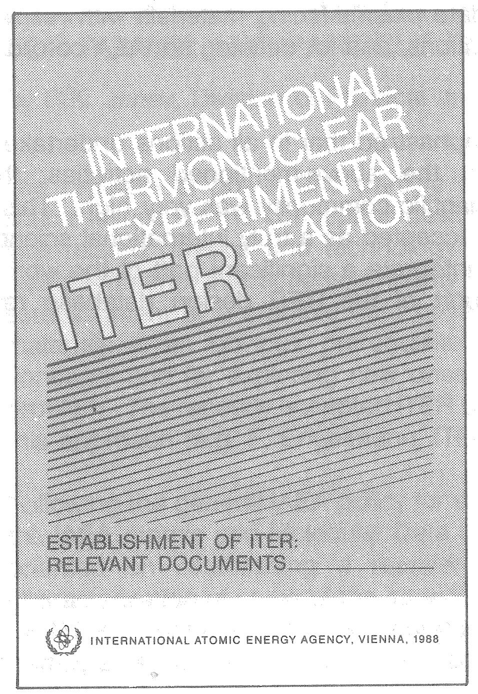

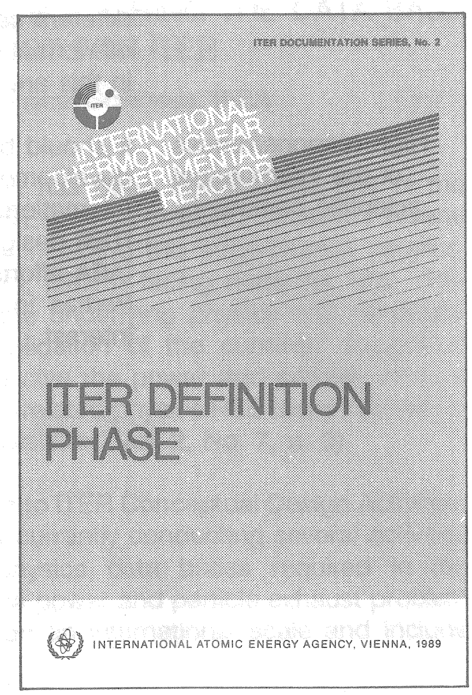

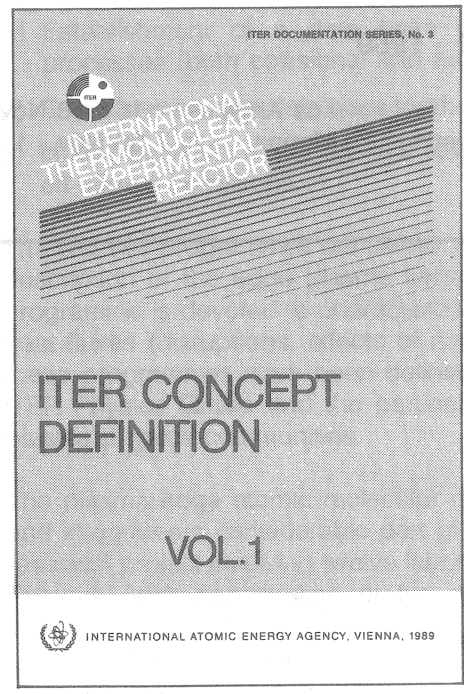

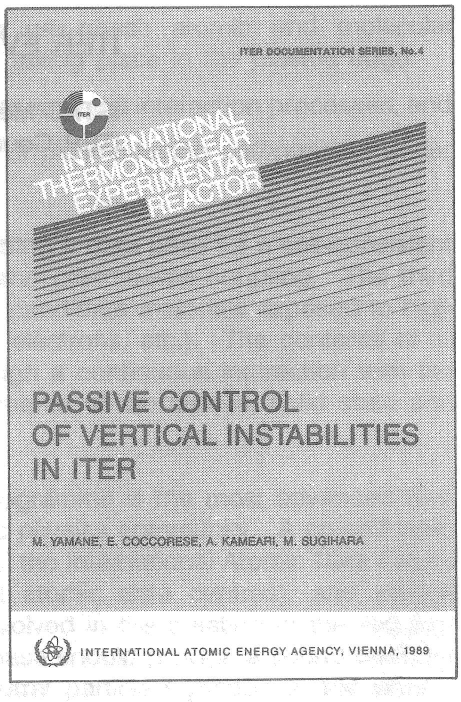

ITER-related information is published by the IAEA

The information generated by the ITER Conceptual Design Activities is published by the IAEA in a Special ITER Documentation Series. The first booklet of this series, issued in 1988, contains documents relevant to establishing the ITER Conceptual Design Activities, including the ITER "Terms of Reference."

The overall results of the Definition Phase were published under the title "ITER Definition Phase" and were distributed to all recipients of the ITER Newsletter. Detail information on the ITER Concept Definition resulting from the Definition Phase, is contained in two volumes (Vol. 1, 71 pages - summary, and Vol. 2, 562 pages).

Also, publication of selected subject reports written by the members of the ITER design team has recently been initiated.