

THE MATERIALS SELECTION IN ITER AND THE FIRST MATERIALS WORKSHOP

by Drs. R. Matera, V. Barabash, G. Kalinin and S. Tanaka, ITER Garching Joint Work Site

The selection of materials and joining technologies to be used in ITER is a trade-off between multiple and often conflicting requirements derived from the unique features of the fusion environment. Materials selection must encompass a total engineering approach, by considering not only physical and mechanical properties, but also the components' manufacturing, their maintainability and reliability, and, finally, how they can be recycled or disposed of at the end of machine operation.

The structural materials for the in-vessel components will operate under the simultaneous influence of different life-limiting factors, such as neutron irradiation, hydrogen atmosphere, dynamic stresses, thermal loads, cyclic mode of operation, and water cooling environment. Even though no safety functions are attributed to the in-vessel components, to achieve good performance and adequate availability of the whole machine they have to remain highly reliable throughout the design lifetime. Ease of fabrication, good weldability, resistance to corrosion, good strength and fatigue resistance, adequate ductility and fracture toughness after neutron irradiation are essential requirements.

The plasma facing materials have to cope with severe operating conditions (cyclic heat loads, high flux of energetic particles, neutron irradiation, etc.). Their selection is mainly based on the expected lifetime under sputtering erosion, thermal ablation, disruption damage and neutron irradiation. For low-Z materials (Be and carbon based materials) tritium co-deposition and inventory play a major role in the selection. For high-Z materials such as tungsten, plasma contamination and loss of melt layer during disruptions become key considerations.

As far as technically feasible, the material choice has been oriented toward industrially available materials and well established manufacturing techniques. This is very much the case for the structural materials of the basic machine (cryostat, magnet case, vacuum vessel), for which a factor is the availability of industrial suppliers with experience in forming and joining technology. The structural integrity of these components throughout the entire design lifetime is important for machine availability and safety.

Taking into account specific requirements for each of the ITER components, the following materials have been proposed for the design:

- Austenitic stainless steel 316 type is the main structural material for most ITER components, in particular: vacuum vessel, back plate, blanket, components of cryostat, etc.

The austenitic stainless steel 316L(N)-IG (ITER Grade), based on the AISI 316L specification but with a narrower composition range of the main alloying elements and a controlled addition of nitrogen, has been selected as the structural material for the in-vessel components. Austenitic stainless steels are the most suitable, as these materials are qualified in many national design codes, have adequate properties and a large experience base for cryogenic applications, and they are antiferromagnetic. Moreover, they have good weldability, forging and casting properties. They have satisfactory resistance to stress corrosion cracking, high levels of fracture toughness and adequate strength. There is an extensive database in the unirradiated condition, an adequate database for material irradiated under ITER conditions, and a large industrial experience in nuclear applications. The characterization of solid and powder Hot Isostatic Pressed (HIPed) steel confirms the HIPing can be successfully applied to the manufacture of the shielding blanket module.

- Incoloy 908 is specifically developed as a Nb₃Sn Cable-In-Conduit-Conductor jacket material.

It can be characterized as a low coefficient of thermal expansion superalloy that contains 4 % by weight of chromium. Its chemistry was derived to maintain a balance between a low coefficient of expansion to match that of Nb₃Sn strand (to minimize J_C and T_C degradation due to differential contraction after the reaction heat treatment), while maintaining its structural properties over a range of Nb₃Sn reaction heat treatments.

- Ti based alloy (Ti-6Al-4V) is the main option for the "flexible" cartridge to connect the blanket module to the back plate.

High strength titanium alloys have advantages compared to other candidate materials in components where impact loading is expected and elastic flexibility is required. The comparison of the candidate materials shows that Ti-6Al-4V alloy has relatively high strength (≈900 MPa yield strength) and high flexibility due to low Young's modulus (approximately a factor of two less than Inconel 718 or SS 316). Titanium alloys are widely used in chemical and aerospace industries. Among the commercial Ti alloys, Ti-6Al-4V was chosen as reference grade. Since it is widely used in different countries, there is an industrial experience in many Hi-Tech fields and the data base of unirradiated material is relatively complete. Available data for irradiated material show that it can be used at doses relevant to the current design.

- A preliminary assessment shows that the high strength Ni alloy, Inconel 718, can be recommended for the attachment of the blanket module to the back plate.

Inconel 718 is the current reference material due to its high strength and good ductility. The material has satisfactory fracture toughness and fatigue lifetime. Inconel 718 is widely used in the nuclear industry. The material is produced commercially in the form of bars, rods, plates, strips, etc. The critical point is stress relaxation under irradiation that will be studied in more detail during planned R&D. As a back-up the high-strength Grade 660, also known as A-286, precipitation hardened stainless steel could be considered.

- Inconel 625 is the main option for the flexible branch pipe connection of the blanket module to the back plate.

The material exhibits good corrosion resistance, high electrical resistivity, weldability with 316L(N)-IG steel and a satisfactory radiation resistance. High electrical resistivity is one of the main requirements allowing a decrease of the electromagnetic loads to the modules. This material is commercially available.

- Two copper alloys have been selected for the heat sink of the Plasma Facing Components (PFCs), one age-hardenable CuCrZr alloy and one dispersion strengthened (DS) alloy, CuA125.

Mechanical properties of both alloys are sufficient for the components to sustain thermal and mechanical loads and achieve the design lifetime. After neutron irradiation, the optimized alloys exhibit better ductility than the nearest commercial produces. In the relevant temperature range, CuCrZr-IG, if correctly heat treated, has better tensile properties and a higher fracture toughness than CuA125-IG. As a consequence of component manufacturing (high temperature HIPing, armour brazing), however, CuCrZr-IG may lose strength and thermal conductivity by overageing. In that case, CuA125-IG would be better as heat sink material. Uniform corrosion rates are relatively low in copper alloys under controlled water chemistry, and there is no evidence of erosion-corrosion effects.

- Beryllium, CFC and Tungsten are the main options for the armour plates of the high-heat flux components.

The plasma facing components have to cope with severe operating conditions and a high erosion rate limiting their lifetime. Beryllium has been chosen as the armour material for ≈80% of the total surface exposed to the plasma (primary wall, upper baffle and as primary option for the port limiters), on the basis of low main plasma pollution, absence of chemical sputtering, oxygen gettering capability, and possibility of armour repair. S-65C beryllium was selected as the reference grade (DSHG-200 is the back-up), because of its resistance to thermal fatigue, availability, and previous experience in JET. Its high sputtering rate makes beryllium less suitable in areas where CX sputtering is the dominant erosion mechanism (lower baffle, upper vertical target), here tungsten provides the best erosion lifetime. In areas hit by high thermal fluxes during normal operation and large energy deposition during plasma instabilities (lower vertical target, dump plate), Carbon Fibre reinforced Carbon (CFC) is selected because it can resist very high heat fluxes and does not melt. However, its use has to be restricted to these regions, because of the problems of chemical erosion and tritium retention, especially in the co-deposited layers. The development of Si-doped CFC could somewhat mitigate the problem of chemical erosion.

- The manufacturing technologies of the in-vessel components involve difference types of joints in the SS structures, between SS and Cu alloys, and between copper alloys and plasma facing materials.

Narrow Gap Tungsten Inert Gas (NG TIG) and Electron Beam (EB) welding are the technologies recommended for the structural welds, HIPing is proposed for the SS/Cu joints. New methods to join Be, W and CFC to the heat sink have been developed. The characterization under thermal fatigue conditions of these joints has demonstrated that for each armour material more than one joining method is available which fulfils, on small-scale mock-ups, the thermal fatigue requirement of PFCs, at least for unirradiated materials. First results on the thermal fatigue behaviour or irradiated mock-ups are encouraging. Coatings could in some instances represent a valid alternative to joining a solid tile onto the copper heat sink. Beryllium plasma spray is a suitable technology for the initial fabrication of the primary wall. The same technology has been successfully applied to tungsten coatings. (Tungsten coatings of the same thickness and the same heat flux limit were produced by Chemical Vapour Deposition

onto copper and Cu/W composite). To repair the uniform damage created by interactions with the plasma on the surface of the PFCs, plasma spray is recommended for metallic armours. For more severe damage, an alternative repair method based on a specially developed Al-Ge rheocast brazing alloy is being developed with encouraging results.

- In the area of plasma wall interactions, good progress has been achieved both at the laboratory level and in the existing tokamaks to better quantify (1) the erosion mechanisms of the PFCs due to sputtering/redeposition and to disruption, (2) the implications of tritium retention and removal on the design, operation and safety of ITER, and (3) the problem of dust formation and collection.

The rationale for the selection of materials and joining technologies of the ITER components is given in the Materials Assessment Report (MAR). The MAR justifies the materials selection on the basis of the available information from the open literature, the ITER R&D results, existing industrial experience, and other available sources. It deals with materials as a part of a specific component, defining the functional requirements, taking into account the manufacturing processes and their impact on the material properties. The effects of in-service conditions on the component structural integrity are included.

The MAR is part of the ITER Final Design Report documentation, as well as the other documents where materials are dealt with from different points of view: the Design Description Documents, the Material Properties Handbook (MPH), and Appendix A of the Interim Structural Design Criteria (ISDC).

The MPH deals with materials as a commodity, independently of their final use. It is a collection of design-relevant data on physical and mechanical properties of a large variety of materials of interest to Fusion Technology. It includes data available in the open literature as well as data from the R&D programme.

Appendix A of the ISDC defines the allowables of the structural materials for the various failure mechanisms considered in the construction code.

The first draft of the MAR for in-vessel components was completed at the end of October 1997 and sent to the HTs for comments and for the preparation of the First Materials Workshop.

During the Workshop, held in Garching on December 1-5, 1997, the second draft was thoroughly discussed and implemented with the additions of the most recent data of the R&D program presented by the HT representatives.

The Workshop was attended by 39 representatives of the four HTs, representing the whole scientific community involved in the ITER Materials R&D program, and by 14 JCT staff members from San Diego and Garching JWS.

During plenary sessions, the HTs' representatives presented the latest results not yet included in the MAR and gave their general comments on the Materials Selection in ITER. In two parallel sessions each chapter of the MAR was thoroughly discussed and revised in order to fully represent a common opinion of HTs and JCT on the final recommendations for the materials and technologies to be used for the in-vessel components.

The outline planning of the working program to be carried out after July 1998 was presented by the HTs. They expressed their views on areas where additional data are considered to be necessary.

LIST OF PARTICIPANTS

EU: P. Chappuis, W. Dänner, G. Le Marois, A. Lind, M. Merola, L. Plöchl, B. Riccardi, M. Rödig, F. Scaffidi-Antonia, B. Singh, S. Tähtinen, F. Tavassoli, E. van Osch, G. Vieider, C. Wu

JA: M. Akiba, T. Hatano, T. Iwadachi, S. Jitsukawa, H. Kawamura, A. Kohyama, K. Nakamura, Y. Nishi, S. Sukuki

RF: V. Belous, S. Fabritsiev, A. Gervash, A. Ivanov, A. Khomutov, I. Mazul, A. Pokrovski, B. Rodchenkov

US: R. Castro, J. Davis, R. Mattas, A. Rowcliffe, M. Ulrickson, R. Watson, S. Zinkle

JCT: V. Barabash, A. Cardella, V. Chuyanov, J. Dietz, G. Federici, Y. Gohar, K. Ioki, G. Janeschitz, G. Kalinin, R. Matera, K. Mohri, R. Parker, S. Tanaka, R. Tivey

US FUSION COMMUNITY DISCUSSION ON FUSION STRATEGIES

by W. A. Marton, USDOE

On April 26 through May 1, 1998, a US Fusion Community Forum for Major Next-Step Experiments was held at Madison, Wisconsin to address a recommendation of a US Fusion Energy Sciences Advisory Committee (FESAC) panel chaired by Hermann Grunder last fall. About 180 scientists and engineers participated, representing virtually all of the US fusion institutions, disciplines and research approaches. The agenda strongly emphasized broad discussion, with more than half of the time reserved for open discussion.

The subject Grunder panel recommendation, which provided the motivation for the forum, consisted of two parts paraphrased as follows, "...it is prudent for the international community to examine ITER options that involve reconsideration of the fundamental trade-offs between cost and mission...directed at examining lower-cost, reduced scope options in the interest of achieving a fusion-energy-producing plasma experiment on the fastest possible schedule" (This part is being addressed by the ITER Special Working Group (SWG) and "in preparation for this international activity, it is essential that the United States initiate a domestic study with broad fusion community involvement to explore the many options" (This part was addressed by the forum).

The forum objectives were the following:

1. Identify a set of candidate "credible" strategy options for advancing fusion energy which have broad US fusion community support,
2. Take a step toward a more effective method of building US fusion community consensus and set the stage for continuing this process, and
3. Provide a sense of the US fusion community views on potential major next steps in fusion energy research as background for the US input into the ITER SWG discussions.

The first day of the forum was dedicated to receiving input from outside the US fusion community. This stimulated the forum participants to think beyond their usual frame of reference and provided a useful starting point for the forum. The following presentations were made:

1. Bob Rosner, University of Chicago, described the experience of the US Astronomy and Astrophysics community in program planning and consensus building.
2. Don Reeder, University of Wisconsin, described the corresponding experience of the US High Energy Physics community including the role of their periodic "Snowmass Meeting."
3. Steve Fetter, University of Maryland, provided perspective on the possibilities and implications of climate change including potential opportunities for introduction of fusion energy into the electric power market after the year 2050.
4. Tom Cochran, Natural Resources Defense Council, provided an environmental perspective on his experience with large nuclear projects such as the Liquid Metal Fast Breeder Reactor and suggested how some of these experiences might relate to ITER.

For the next two days, fusion program strategy options were presented and key fusion program issues were addressed including community ideas for major next step experiment options. The Forum then formed into 6 smaller groups where every participant was invited to present his/her views and to express opinions about strategic options.

On the last day of the Forum, Gerald Navratil, Columbia University, one of the organizers, presented a summary to Martha Krebs, Director of Energy Research in the US Department of Energy, who has responsibility for the US fusion program, including the following conclusions and next steps:

1. Burning plasma physics should be the primary priority in a strategy for a major next step experiment.
2. The size and cost of major development steps give an impression of the size and cost of the final product; therefore, major steps in developing the scientific and technical basis for a practical source of fusion energy need to recognize this.
3. Capability to incorporate innovation should be emphasized if the next step experiment is to be "attractive".
4. Three general strategy options were identified during the Forum as follows:
 - a) Single Integrated Step, in which an integrated step forward is taken now with the Tokamak. This option was discussed in the context of a reduced cost ITER. The sentiment was that the US should

emphasize during the SWG process the incorporation of as many Advanced Tokamak features as possible into ITER as a way of possibly reducing size and cost and increasing attractiveness. The participants felt that if the other ITER partners agree to build such a machine, then the US should participate in the construction.

- b) Multiple Machine Approach, in which smaller, phased, or sequential steps are taken on the way to a demonstration reactor. For this option, the principal elements would be a copper DT ignition device and a superconducting DD steady state device. Examples supported of the former range from Ignitor to the PCAST device; examples supported of the latter are TPX or KSTAR type devices. Since some of these devices could be in the billion or multi-billion dollar cost range, the need for international collaboration was recognized as a necessity.
 - c) Deferral of Major Next Step, in which innovative concept development would be emphasized while further deferring any major next step in fusion development. Those concepts which are able to advance to significant levels of performance while retaining the potential for reactor attractiveness would then qualify for major next step candidacy.
5. A summary of the views of the participants on these strategies follows:
- a) Both the Single Integrated Step and Multiple Machine strategies had substantial support.
 - b) The Deferral strategy had no broad support.
 - c) If the US fusion community were empowered to make the choice now, the multiple machine strategy would be preferred over the single step approach. However, the consensus was that the US should remain in the ITER EDA. Along these lines, a combination of these two approaches was also supported, in which ITER should proceed to a construction decision in 3 years, during which time modest US evaluation of multiple machines should continue. If ITER did not proceed into construction, then the multiple machine approach would be available for further study.
6. In response, Dr. Krebs commented favourably on the restructuring occurring in the US Fusion Energy Sciences program and encouraged a continuing focus on innovation in pursuit of these strategies. In this regard, she indicated that the ITER process has been valuable, that it is important for the ITER Parties to complete the three-year extension with a focus on reduced cost and innovation capability, and to decide whether to construct ITER.
7. The participants are beginning to prepare a report of the Forum, including additional explanation of the rationale behind the three strategies. Working groups were identified for this purpose, and a meeting in San Diego is scheduled for June 18-19, 1998, for the purpose of report co-ordination under the chairmanship of Charles Baker. The current plan is to have the report submitted to DOE by the end of July 1998, for presentation to FESAC later that month.

ITER CENTRAL SOLENOID MODEL COIL HEAT TREATMENT COMPLETE AND ASSEMBLY STARTED

by Dr. R.J. Thome Division Head, and Dr. K. Okuno, Design and CS R&D Group Leader, Superconducting Coils and Structure Division (for the ITER Joint Central and Home Teams)

A major R&D task in the ITER program is to fabricate a Superconducting Model Coil for the Central Solenoid (CSMC) to establish the design and fabrication methods for ITER size coils and to demonstrate conductor performance. This task was extensively covered by the article on the Central Solenoid Model Coil Project*. The heart of the coil is the Cable-in-Conduit conductor which is required to carry 46kA. The finished Model Coil will have a maximum field of 13 T, a bore diameter of 1.6 m, and a stored energy of 640 MJ.

The cable of the conductor has about 1000 strands of Niobium-Tin superconductor. The strands are initially cabled and then inserted into a jacket which provides structural strength for the coil. Strand production and cabling has been done by all four ITER Parties and jacketing of the conductor has been done by the European Union Home Team (EUHT) with Incoloy 908 jacket material supplied by the United States Home Team (USHT). About 6000 m of this conductor has been completed for the CS Model Coil.

* "Central Solenoid Model Coil Project" by R.J. Thome, K. Okuno, B.J. Green (ITER Joint Central Team), R. Jayakumar (US Home Team), H. Tsuji (JA Home Team) for the Project Staff, ITER EDA Newsletter, Vol. 6, No. 3, March 1997.

In order that the strands, which contain the Niobium and Tin, can achieve superconductivity at the operating temperature of about -269°C (4 K), the jacketed conductor has to be reacted at up to 650°C for about 250 hours to form the intermetallic compound, Niobium-Tin. For the ITER program, the cable is contained in a jacket of Incoloy 908 which has nearly the same thermal shrinkage as the superconductor and avoids a degradation of the current density capability of the superconductor which would occur if stainless steel were

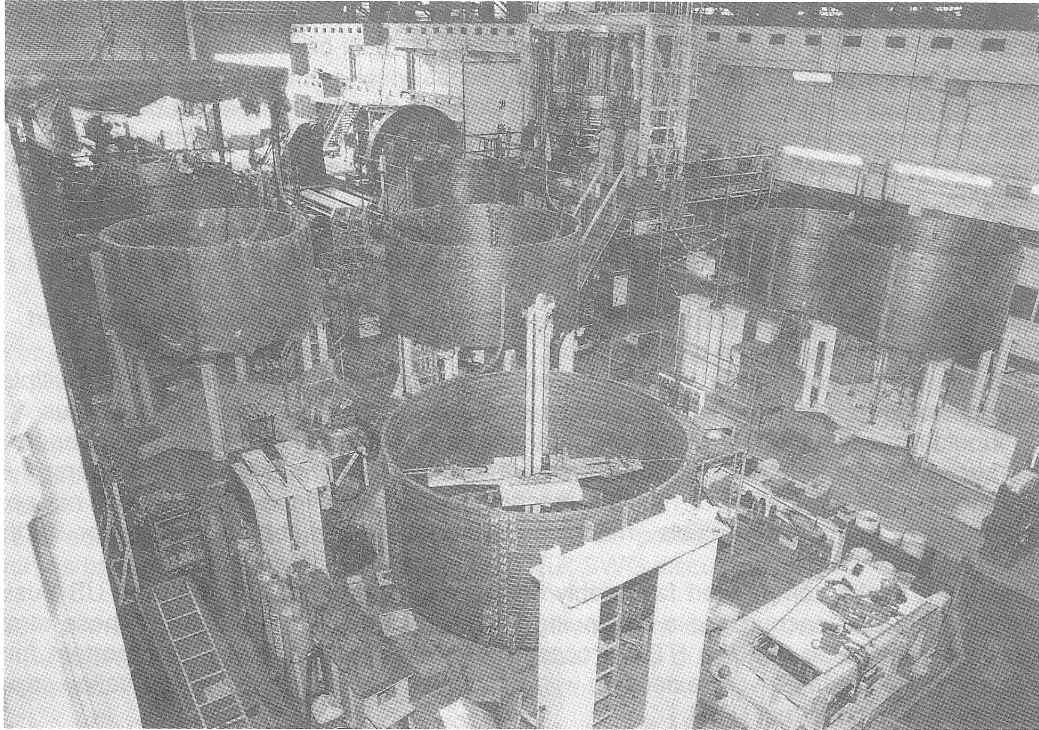


Fig. 1. Several Layers of the CSMC Outer Module in Various Stages of Fabrication at the JAHT Facility

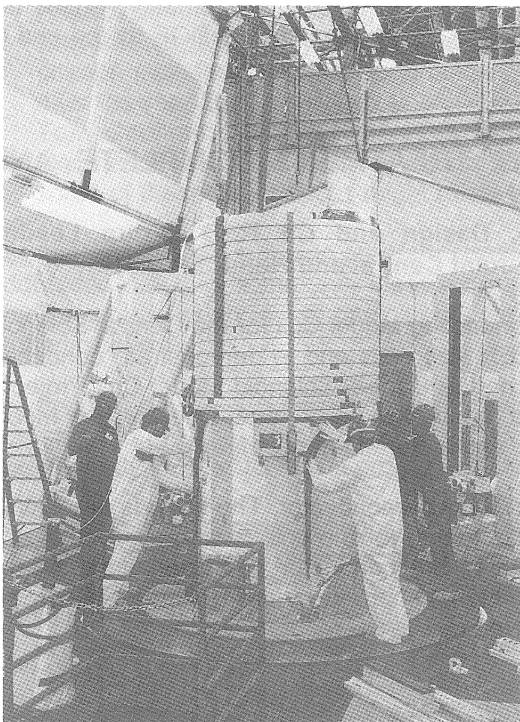


Fig. 2. First Layer of the Inner Module Being Assembled at the USHT Facility

to be used. However, the Incoloy 908 material with its fabrication and manufacturing stresses, is susceptible to so-called Stress Accelerated Grain Boundary Oxidation (SAGBO) at the heat treatment temperature when even minute quantities of oxygen or water vapour are present. As a result, a process consistent with avoiding SAGBO and with large scale coil fabrication requirements was developed as part of the CSMC program. This was an intensive, successful, multi-year effort of basic studies and manufacturing process development between the Japan Home Team (JAHT) and the USHT.

The USHT has the responsibility for manufacturing the 10 layer inner module of the CSMC and the JAHT has the responsibility for the 8 layer outer module. Both teams wind the conductor into the shape required for individual layers before reacting the conductor to form the Niobium-Tin. The USHT used a vacuum furnace to provide an environment for SAGBO prevention for the outside of the jacket, while the inner volume of the jacket (cable space) was

protected by purging with a flow of Argon. The JAHT encloses a wound layer in a retort which is then placed in a furnace at atmospheric pressure. SAGBO is prevented by a purge gas flow with Argon in the retort outside & inside the conductor jacket. Heat treatment of layers of conductor for the CSMC started in 1996 and finished in March, 1998. All 18 layers (about 85 t total) were successfully heat treated without SAGBO. This was a significant milestone for both Home Teams and for ITER.

After heat treatment of each layer, the turn insulation is applied and the layers are assembled into a module. As of mid-April, 98, three layers have been assembled by each Home Team. This represents a successful demonstration of the wind, react, insulate, and transfer process for ITER scale coils.

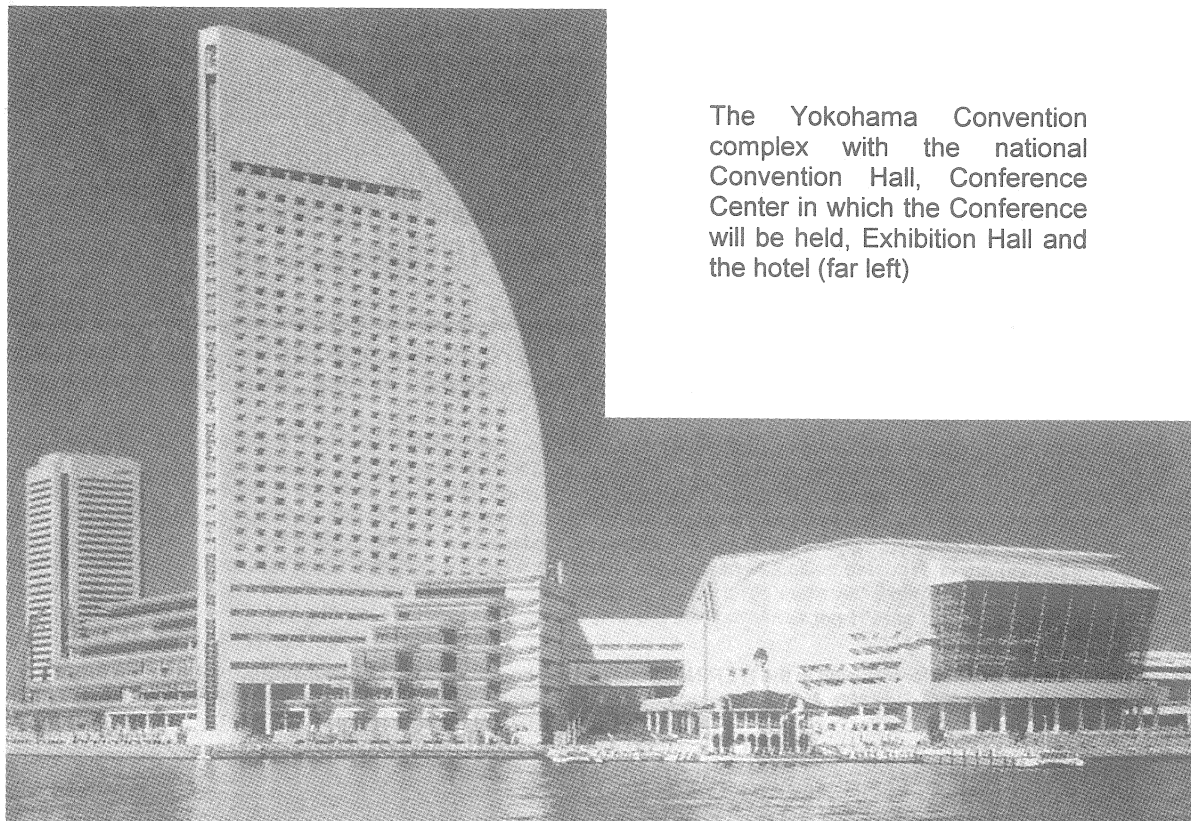
Completion of the CSMC modules is expected in 1998 to be followed by assembly with structural components and testing in a facility at the Japan Atomic Energy Research Institute. The tests will involve the CSMC as well as three insert coils, two being manufactured by the JAHT and one by the Russian Federation Home Team (RFHT).

PROGRAMME OF THE 17TH IAEA FUSION ENERGY CONFERENCE by Drs. T. J. Dolan and U. Schneider, IAEA

The Programme Committee for the 17th IAEA Fusion Energy Conference, to be held in Yokohama 19-24 October 1998, met in Vienna 11-13 May. The Programme Committee consisted of 22 scientists from Europe, Japan, Russia, USA, China, India, Brazil, and the IAEA. Dr. Derek Robinson (UK) served as Programme Committee Chair, and Dr. Teruo Tamano (Japan) as Vice Chair. The Committee considered 370 abstracts that were submitted and selected 107 papers for oral presentation and 240 for poster presentation. The selected papers are covering the following eight categories:

EX Magnetic Confinement Experiments
TH Magnetic Confinement Theory and Modeling
CD Plasma Heating and Current Drive
EDA ITER Engineering Design Activities

IF Inertial Fusion Energy
IC Innovative Concepts
FT Fusion Technology and Reactor Concepts
SA Safety, Environmental, and Economic Aspects of Fusion.



The Yokohama Convention complex with the national Convention Hall, Conference Center in which the Conference will be held, Exhibition Hall and the hotel (far left)

There will again be an Artsimovich Memorial Session. Dr. Robert Aymar, ITER Director, has been invited to present a lecture at the opening of this session, followed by an overview paper on ITER by Dr. Yasuo Shimomura, Deputy to the Director. There will be one oral session on ITER Wednesday afternoon with the following six presentations by the ITER Deputy Directors and other leading members of the Joint Central Team:

Campbell, D.J.	ITER Physics Basis and Physics Roles
Parker, R.	ITER In-Vessel System Design and Performance
Wesley, J.C.	Operation and Control of ITER Plasmas
Huguet, M.	The Integrated Design of the ITER Magnets and their Auxiliary Systems
Haange, R.	Remote Handling Maintenance of ITER
Chuyanov, V.A.	ITER Plant Layout and Site Services

There will also be one poster session on ITER with about 41 posters.

The IAEA will notify authors in early June about the status of their papers. It is planned to print a limited number of copies of the conference proceedings from unedited camera-ready copy as an IAEA TECDOC and to provide a CD-ROM disk of the proceedings using electronic versions of the papers. The format requirements will be specified in the letters to authors.

Since hotel space near the conference center is limited, it is advisable to make hotel reservations well before the 1 August deadline. Otherwise hotel accommodations could be difficult to obtain. The local information about the Conference will be updated on the Conference Home Page in the Internet (<http://www.convention.co.jp/iaea/>).

MR. HIROSHI SHIBATA
In memoriam

Mr. Hiroshi Shibata, Director of Fusion Energy, Atomic Energy Bureau, Science and Technology Agency, Government of Japan, passed away the morning of June 5, 1998, at the age of 37. He had been in this position since October 1997. In spite of the relatively brief period of time, he had shown an outstanding capacity of leadership in the Japanese Fusion Program and contributed much to the ITER EDA at this very important period. We all appreciated his commitment to ITER and his open and constructive attitude. We are praying for the repose of his soul.

Items to be considered for inclusion in the ITER Newsletter should be submitted to B. Kuvshinnikov, ITER Office, IAEA, Wagramerstrasse 5, P.O. Box 100, A-1400 Vienna, Austria, or Facsimile: +43 1 237762, or e-mail: c.basaldella@iaea.org (phone +43 1 206026392).

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