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CSMC AND CSIC CHARGING TESTS SUCCESSFULLY COMPLETED

by Dr. H. Tsuji, Head, Superconducting Magnet Laboratory, JAERI Naka

It is a great honour to report the completion of all charging tests of the Central Solenoid (CS) Model Coil and the CS Insert Coil on August 18, 2000. It is emphasized with great satisfaction that during the six-month period during which the tests were carried out, no accident or injury have occurred.

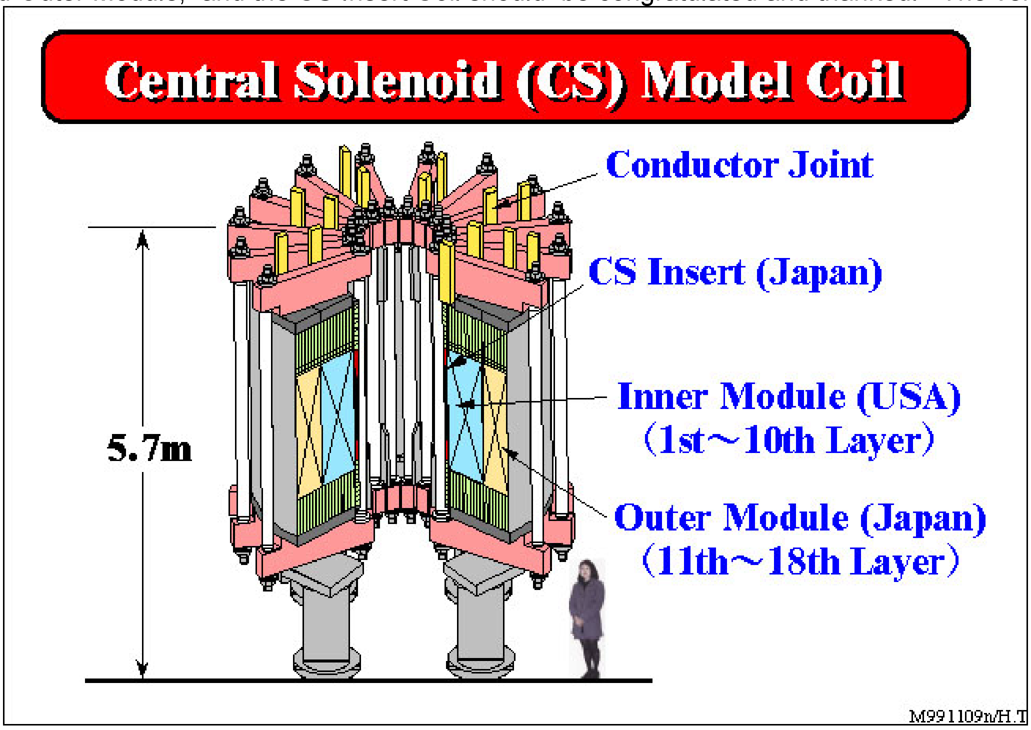
By these tests, all of the following project goals were successfully met:

- DC Charging: Generate 13T and 640 MJ by the current of 46 kA;
- Pulsed Charging: Charge the coil up to 13 T by a ramp rate of +0.4T/s;
- Discharge the coil from 13 T by a ramp rat of -1.2 T/s;
- Cycling Charging: Charge and discharge the CS Insert Coil for 10,000 times.

The warm-up of the coil system with a weight of 180 tons started the next day, on 19 August, in the morning and the coils returned from the 4-K cryogenic world back to the 300-K human world by the end of August.

The next important job for the scientists involved is to evaluate the huge amount of data obtained during the six-month tests and to reflect the results in appropriate reports. In this endeavour, the continued encouragement and participation of the ITER community will be essential.

On this occasion, all colleagues from the ITER Home Teams, the Joint Central Team, institutes and industries over the world, who participated in the development and fabrication of the CS Model Coil's Inner Module and Outer Module, and the CS Insert Coil should be congratulated and thanked. The very insistent



and successful efforts of the CSMC Test Group Members who took part in the experiment at JAERI Naka should be particularly commended. The names of the Cryogenic Operation Group (Kouichi Imahashi, Takeshi Ohuchi, Kiichi Ohtsu, Junichi Okayama, Tsutomu Kawasaki, Masaru Kawabe, Yoshiyuki Takaya, Fujio Tajiri, Hiromi Hanawa and Takeomi Hirohara), who carried out the operation of the Hellics Refrigerator continuously in three shifts per day for six months, deserve particular mentioning.

The brief history of the development, fabrication and testing of the CS Model Coil and CS Insert Coil is given in the box below.

BRIEF HISTORY OF THE DEVELOPMENT, FABRICATION AND TESTING OF MODEL COILS

July 1992	Development and fabrication of coils initiated
November 1998	Fabrication of the CS Model Coil Outer Module was completed. The Module was transferred to Naka Fusion Research Establishment of JAERI
February 1999	Fabrication of the CS Model Coil Inner Module was completed and the Module was shipped from the US to Japan
May 1999	Arrival of the Inner Module at JAERI Naka
May 1999	Fabrication of the CS Insert Coil was completed and the Coil was transferred to JAERI Naka.
1 June 1999	An international ceremony for the completion of the coils' fabrication was held at the Naka Fusion Research Establishment of JAERI
2 June 1999	Start of the installation of the CS Model Coil Inner Module and Outer Module and the CS Insert Coil, together with the coil supporting structures fabricated by the US Team
October 1999	Completion of the installation and start of the initial check-outs
November 1999	Start of the initial cool-down of the coil system the total weight of which being 180 tons
December 1999	Helium leak at 20 K was detected. The cool-down was stopped.
January 2000	The coil system warmed up to 300 K
February 2000	The leak in the cooling plumbing of the support structure was located. A new cooling pipe was installed
13 March 2000	Restart of the initial cool-down
4 April 2000	Superconducting transition of all the modules at 17.5 K achieved
11 April 2000	The charging test of the CS Model Coil started
19 April 2000	DC charging goal of the CS Model Coil (46 kA, 13 T, 640 MJ) achieved
23 May 2000	The first pulse charging goal of the CS Model Coil (charge up of the coil up to 13 T by a ramp rate of +0.4 T/s) achieved in the joint experiment with the JT-60 Power Supply Group
24 May 2000	Second pulse charging goal of the CS Model Coil (Discharge of the Coil from 13 T by a ramp rate of -1.2 T/s) achieved
5 June 2000	DC charging goal of the CS Insert Coil (40 kA, 13 T) under a backup field of the CS Model Coil achieved
3 July 2000	Cyclic test of the CS Insert Coil between 0 kA, 13 T and 40 kA, 13 T under a backup field of the CS Model Coil started
7 July 2000	Initial 1000 cycles done
11 July 2000	Switch to the pulsed charging of the CS Insert Coil and CS Model Coil in the joint experiment with the JT-60 Power Supply Group
20 July 2000	Rapid charging of the CS Insert Coil by a ramp rate of +1.2 T/s up to 13 T successfully performed
21 July 2000	Rapid charging of the CS Model Coil by a ramp rate of +0.6 T/s up to 13 T successfully performed
25 July 2000	Cycling test of the CS Insert Coil resumed
9 August 2000	10,003 cycles successfully made
11 August 2000	No major degradation of the superconducting performance of the CS Insert Coil by the cyclic operation of 10,000 cycles was confirmed
14 August 2000	Start of reversed field charging between the CS Model Coil and the CS Insert Coil and measurements in the superconducting perform-ance under a compressive strain state
18 August 2000	All charging tests of the CS Model Coil and the CS Insert Coil completed

THE ITER DIVERTOR CASSETTE PROJECT MEETING

by Drs. M. Akiba, JAERI, and R. Tivey JCT

The Divertor Cassette Project (ITER Large Project L-5) topical meeting was held on April 5-7, 2000 at the JAERI, Naka site. The meeting focused on the progress made by the three parties under task agreements on the development of carbon-fibre composite (CfC) and tungsten armoured high heat flux plasma-facing components.

The majority of R&D effort has focused on the development of two armour options for the vertical target, one with a carbon/tungsten combination (carbon near the strike point) and the other all tungsten. In line with the recommendations of the Technical Advisory Committee (TAC), within these broad options, design and manufacturing variants are being developed that offer the potential of improving reliability and/or reducing costs. The R&D programmes of the Home Teams in developing the technologies, and in building and testing components, are structured so as to complement one another with a minimum of overlap.

EU HT tests at Le Creusot on a prototypical vertical target have already shown that the reference design (carbon monoblock and tungsten macrobrush) can meet the ITER requirements on a DS Cu heat sink. The mock-up sustained 1000 cycles at 10 MW m^{-2}, before the tungsten macro-brush armour was then tested at 15 MW m^{-2} for 1000 cycles and the carbon armour at 20 MW m^{-2} for 2000 cycles. Finally, the carbon armour was shown in a critical heat flux (CHF) test to survive > 30 MW m^{-2}. The EU is now concentrating its efforts on adapting the reference design to be suitable for use with CuCrZr, by joining the armour using Hot Isostatic Pressing (HIP-ing) at ~ 500°C, the temperature used for hardening the CuCrZr alloy. EU R&D will also study the possibility of using wider monoblocks as a means of reducing cost while still meeting the high heat flux requirements.

The JCT also reported that recent post irradiation examination (PIE) of specimens confirm that ductility (uniform and ultimate elongation) is similar for both the precipitation-hardened alloy, CuCrZr, and dispersion strengthened copper (DS Cu). Therefore, CuCrZr remains the reference alloy, because of its much higher fracture toughness.

Building on its success with the carbon monoblock, based on a 20 mm diameter tube, the JA HT is on schedule to investigate the feasibility of using an annular flow design for the vertical target before the end of the EDA. If successful, this offers the potential of a cheaper and more robust design with simplified coolant manifolds.

The RF HT, on the other hand, is developing a tungsten-armoured vertical target that meets the high heat flux requirements and provides good, though not optimal, CuCrZr mechanical properties by using a fast brazing technique based on ohmic heating of the component. In fact, all three HTs are continuing to build and test small mock-ups in order to develop tungsten armours with 20 MW m^{-2} capability. Using 10 mm cubed tiles an RF mock-up survived more than 1000 cycles at > 26 MW m^{-2} without damage or distortion. At Jülich, an EU mock-up using the reference macrobrush geometry (pins 4.5 x 4.5 x 10 mm) on a CuCrZr heat sink survived 1000 cycles at 18 MW m^{-2}, again without damage, and a monoblock lamellae (0.2 mm thick) on a CuCrZr tube survived more than 600 cycles at 18 MW m^{-2}. Other promising options scheduled for testing are designs using hot pressed tungsten pins into a Cu substrate (JAERI) and flat tungsten tiles on a hypervapotron heat sink (CEA), which should avoid the high temperatures in the pure Cu interlayer blamed for the thermal creep distortion observed in some mock-ups at high heat flux. This hypervapotron mock-up is one of a pair (the other has carbon flat tile armour), which is part of the programme aimed at testing the postulated cascade failure effect, where the loss of a single tile leads to the rapid detachment of the neighbouring downstream tile.

Both the JA and EU HTs are developing non-destructive examination (NDE) techniques for joint testing. JA are using in-bore ultrasonic probes, while the EU have demonstrated success in using both ultrasonic and thermographic (carbon armour only) methods.

On the issue of material selection, recent tests carried out in the JUDITH facility in Jülich show that the tungsten candidate alloy, 1%La_2O_3-W, has no benefit over pure tungsten in terms of post-irradiation fatigue, and during disruption simulation tests in JAERI the same alloy showed a three times higher material loss than that observed for pure tungsten. There was little difference observed in the material loss of unirradiated and irradiated carbon during thermal shock tests (Jülich), with the exception of siliconised carbon (SEPcarbNS11), which showed a four-to sevenfold increase depending on irradiation temperature.

A “hot” liner has been proposed as a means of mitigating the co-deposition of tritium with carbon. The RF reported that using a “hot” liner a reduction in co-deposition can be expected. This is based on the results of two separate experiments carried out at the Institute of Physical Chemistry (IPC), Moscow, using methane gas excited by, in one test, a radio frequency source and, in the second, a magnetron. However, a significant level of hydrocarbon radicals with low sticking coefficients will still pass through the liner and these will have the potential to deposit on the cold surfaces beyond the liner. IPC also report that no significant variation in co-deposition downstream of the liner was observed, for liner temperatures in the range 500-1200 K. The RF is continuing this R&D, and in addition will consider using a cold plate beyond the liner to trap all the tritium-bearing hydrocarbons. In ITER such a plate might be heated off line in order to reclaim the tritium. Finally, the EU are constructing a test rig at IPP Berlin that will generate a more divertor-relevant source gas by firing a plasma beam at a carbon target. First results from this facility are expected before the end of 2000.

In summary, the R&D continues to make significant progress in the development of the cost effective technologies that are needed to make both a carbon/tungsten and an all tungsten target viable options for plasma-facing components of the ITER divertor.

The next meeting is provisionally scheduled for April 2001 in St. Petersburg.

LIST OF PARTICIPANTS

EU: P. Chappuis (CEA Cadarache), M. Merola (EFDA CSU Garching), M. Rödig (FZJ)

JA: M. Akiba (JAERI), K. Ezato (JAERI), K. Sato (JAERI), M. Taniguichi (JAERI), H. Nakamura (JAERI), K. Yokoyama (JAERI), M. Dairaku (JAERI)

RF: R. Ginyatulin (Efremov Institute), I. Mazul (Efremov Institute)

JCT : V. Barabash, C. Ibbott, R. Tivey

BLANKET R&D AND DESIGN TASK MEETING

by Dr. K. Ioki, ITER Garching JWS

A blanket R&D and design task meeting was held at JAERI Oarai Establishment and ITER Naka JWS on 17-19 July 2000. The shielding blanket and breeding blanket designs and R&D were reviewed for two days and for one day, respectively. Progress on R&D and the designs of the shielding and breeding blankets were reported and the completion of blanket R&D tasks by July 2001 were discussed in this meeting. The meeting participants visited testing facilities including an EB (electron beam) heat load test facility in the hot cell laboratory at JAERI Oarai. Participants also observed the full-scale blanket module which had been sectioned by partial cutting, and a full-scale partial mock-up of the separate FW panel recently fabricated by the JAHT at JAERI Naka Establishment. The JAHT tasks were mainly discussed and recent results of the EUHT and RFHT were also reported in this meeting.

Shielding Blanket Design

The JCT presented the latest design of the shielding blanket. The blanket design has been modified consistently with the separate cooling manifold concept. The blanket design in the NB (neutral beam) port region should be fixed in the near future. More detailed design of the shield block with radial cooling channels in a coaxial configuration has been developed.

The FW design has been improved in a few aspects. The lateral Cu-plate has been eliminated in the SS backing plate of the FW panel due to lower nuclear heating in ITER-FEAT. A slight modification of the cooling channel layout is proposed by the JAHT. The JCT and EUHT have proposed a blanket module design with additional deep slits in the shield block to significantly reduce electromagnetic (EM) loads. The EUHT has continued thermal and mechanical analysis of the FW panel attachment system with shear ribs and high strength bolts accessed from the rear side of the module. The JAHT has performed a detailed 2D thermal analysis of a "finger-structure" FW panel, which has confirmed low thermal stress conditions. The JAHT has performed a structural analysis of the FW panel attachment system with a welded central shaft support. A race-track cross-section of the central shaft can reduce its toroidal width. YAG Laser welding will be used for the FW panel replacement in this concept. Dynamic behaviour of the FW panel with the central shaft support will be checked. Reliability of the FW panel attachment system with shear ribs and high strength coppercored bolts will also be assessed. Engineering margins should be evaluated for either concept of the FW panel attachment system.

The RFHT reported two new designs of the limiter module assembly system. One design is a structure with a bolt connection, which is compatible with refurbishment in the hot cell. The RFHT also reported progress on the structural analysis of the blanket attachment system, including the shear key and flexible support.

The JCT and the EUHT have calculated eddy currents and EM loads in the blanket module and the module attachments, such as the electrical strap and the hydraulic connections. The calculated values are within the allowables. The JAHT plans to perform EM analysis on the blanket module by using a 3D solid model during disruptions and VDEs. The EM loads on all blanket modules including combined cases will be calculated in the near future.

Shielding Blanket R&D

JAHT R&D

Qualification of Be and Be/Cu joints and mock-ups testing

The JAHT has focused on low-temperature hot isostatic pressing (HIP) (T< 650°C) to avoid sensitisation of SS and the formation of an intermetallic compound in the Be/Cu joint. Two interlayers have been selected; Al/Ti/Cu, Cu. High mechanical properties are obtained for the two approaches in 4 point bending tests (250 MPa at RT; ~ 200 MPa at 200°C). Using the OHBIS facility in JAERI Oarai, high heat flux tests (5 MW/m^2, 150 cycles) have confirmed their good performance. New mock-ups will be manufactured for thermal fatigue testing (1000 cycles).

Module fabrication development

A prototypical mock-up of the blanket module with separate FW panels is to be fabricated. The shield consists partly of radial cooling channels and partly of poloidal cooling channels. A full-scale FW panel is to be fabricated by HIPing with DSCu heat sink and a partial FW panel with Be armour. Mock-ups of the FW panel with CuCrZr heat sink will also be fabricated and tested as a possible cost reducing approach.

Development of the slot fabrication methods

Electrical discharge machining and end milling have been utilised for the slot fabrication on the FW panel with Be tiles. A water jet has been successfully used for deep slot fabrication in the shield block. Wire cutting will also be applied as an alternative. The most cost effective method will be selected based on the results.

Meeting Participants in the Shielding Blanket Session (Naka JWS)

EUHT R&D

Fabrication and testing of FW mock-ups and shield block prototype

The EUHT reported high heat flux testing at FE200 in Le Creusot on FW mock-ups fabricated with solid-HIPed DSCu and CuCrZr, and powder-HIPed CuCrZr. All the mock-ups have been successfully tested by a large margin compared with the ITER design requirements. A full-scale shield block prototype representative of a double curved module of the 1998 ITER design has been fabricated by powder HIPing. The manufacture of a medium scale mock-up for FW joining development was also reported.

Module fabrication development

Four separate FW panels are being fabricated. Two panels are being fabricated with HIPed and brazed Be/DSCu joints respectively. Two panels are being fabricated by powder and solid HIPing respectively, and with HIPed and brazed Be/CuCrZr joints respectively. The two separate FW panels with DSCu heat sink will be tested at a thermal fatigue test facility in ENEA Brasimone by May 2001. Fabrication development (e.g. drilling trials on stainless steel plates, fabrication of serpentine tubes with small bending radii, fabrication of small scale powder HIPped mock-ups) is progressing to develop more cost effective manufacturing alternatives.

Studs with a high thermal conductivity core for the FW panel attachment

High strength studs to fix the separate FW panel onto the shield block require a high thermal conductivity core to avoid excessive heating and to maintain the preload during plasma operation. Candidate materials for the core are pure copper and sodium (or NaK). Stud specimens will be fabricated for mechanical testing.

RFHT R&D

Fabrication and testing of the port limiter

The RFHT presented the R&D status of the fabrication and testing of small port-limiter mock-ups. Each panel of the mock-up is made by HIPing, and beryllium tiles are bonded to the copper substrate (DS copper and CuCrZr) by brazing. The panels are joined together by EB welding. A heat load test will be carried out on the mock-up at a facility in Efremov Institute. Fabrication of a middle-scale mock-up of the port limiter is underway. Each panel is full size in the radial direction and half size in the poloidal direction. The fabrication technology of the full-scale port limiter will be established by this R&D.

Meeting Participants in the Breeding Blanket Session (JAERI Oarai Establishment)

Fabrication and testing of the flexible mechanical attachment
Fourteen flexible supports of Ti-alloy have been fabricated by the RFHT, and thermal and structural tests have been performed at an elevated temperature. The buckling stability and fatigue strength of the flexible support have been confirmed in the tests.

Blanket Material R&D (JAHT R&D)

Irradiation tests on Cu alloy/SS joints
The test specimens of the joining materials for the limiter/divertor pipes consist of cast materials prepared by the RFHT, and brazed and friction-welded materials prepared by the JAHT. Tests on unirradiated materials have been performed, and tests on irradiated materials are in progress at JMTR in JAERI Oarai. Tensile properties of the brazed materials are poor and the brazing method needs to be improved. Tensile properties of the cast and friction-welded materials are good in unirradiated and irradiated conditions.

The test specimen of the joining materials for the blanket module structure are HIP jointed DS-Cu/SS and SS/SS. Results of unirradiated specimens show that the bonding tensile properties are similar to the DS-Cu base metal, but the impact value is considerably reduced. The task will be completed by March 2001.

Corrosion test of SS, Cu-alloy, Cu/SS joints
The JAHT has started irradiation-assisted stress corrosion cracking (IASCC) tests of HIPed SS and SS/SS joints, and studies of crevice effects on SCC. Crevice tests will also be carried out under ITER water chemistry conditions. The results are expected in April 2001. In the meeting, the importance of the crevice test at welds was emphasised and the relevance of the accelerated tests was discussed.

Irradiation test on Ti alloys for the flexible attachment. The mechanical properties of the Ti alloy used for the blanket flexible support are to be tested by the JAHT. Two types of Ti-6Al-4V materials have been considered: aged and annealed. The first has higher tensile strength, and the second has higher fracture toughness when unirradiated. At present, mechanical, impact strength and fracture toughness tests, have been completed before irradiation. A set of specimens has completed irradiation at JMTR. The post-irradiation tests will be completed by March 2001.

Irradiation test on insulating coating. The mechanical and electrical performance of alumina insulation coating is to be measured for the keys of the blanket and the flexible cartridge. Two types of coatings have been produced, with different adherence layers between the alumina and the steel plate. The specimens will be irradiated and tested under impact loads. The results will be available at the beginning of 2001.

Reweldability test of irradiated stainless steel
Tubular and straight specimens with 3 mm thickness have been prepared and irradiated. They are now being welded in a hot cell with YAG Laser. Mechanical tests after welding will be carried out by the end of this year, while a second irradiation up to 3 He appm will take place by the beginning of 2001. Final tests and reporting will be completed before the end of the EDA extension.

Breeding Blanket Design

The JCT plans to restart the breeding blanket design after August. The issues will be (i) selection of the basic structure such as a BIT (Breeder In Tube) concept, (ii) temperature control of the breeding material and the Be moderator, (iii) design of the He purge gas line. The EUHT has developed a preliminary design of a HCPB (helium-cooled pebble bed) concept for the ITER breeding blanket, but the coolant temperature would seem to be too high. The local TBR (tritium breeding ratio) is 1.14(Li_3SiO_4) and 1.10 (Li_2ZrO_3, Li_2TiO_3). The JCT noted that the breeding blanket is to be water-cooled. The JAHT plans to start a design improvement activity based on the 1998 ITER breeding blanket design. The work will be performed reflecting the low-temperature tritium release R&D result and the ITER design changes such as the separate blanket cooling concept and the lower coolant temperature.

Breeding Blanket R&D

The JAHT reported the status of the tritium recovery test from the breeder at JMTR. Tritium recovery from lithium titanate (Li_2TiO_3, grain size ~5 μm) at lower temperature (200°C~) will be clarified quantitatively. An intermediate report will be prepared at the end of 2000. The JAHT will perform a pebble packing test. Effects of the thermal cycles and vibrations are to be tested. In this test the packing factor distribution is measured by an X-ray-CT device.

The EUHT reported the status of thermomechanical interaction testing between Be and cooling plates at the SMARTS facility in ENEA Brasimone. The thermal conductivity and heat transfer coefficient of a Be pebble bed have been measured in Super Pehtra at FZK in Germany. Data from a binary pebble bed during thermal cycles have been obtained. The electrical resistivity of the Be pebble bed has been measured by ITN in Portugal. The electrical resistivity increases with temperature and decreases with applied mechanical pressure. The BeO layer increases the effective electrical resistivity and electrical insulation will not be required on the wall surface of the pebble bed. Mitigation of water/steam chemical reactivity of Be pebbles is being studied by SCK CEN in Belgium. The effect of possible coatings will be studied.

The RFHT reported the out-of-pile testing for He-cooled DEMO blanket with ceramic breeder and in-pile testing of an ECD (experimental channel device) with pebble-bed breeding zones in the IVV-2M reactor.

Conclusions

The shielding blanket design has progressed in the blanket manifolds, attachment system and blanket module. Further work on the blanket module structure and the FW panel attachment system is required to prepare the Final Design Report by the end of this year. The blanket R&D is proceeding on schedule without any significant delay. The EDA blanket R&D tasks will be completed during the EDA period. However, the schedule of the heat load tests on separable FW panels is very tight, and some tests will be carried out after the end of the EDA.

List of Participants

EUHT:	P. Lorenzetto, L. Petrizzi, M. Heller
JAHT:	Y. Ohara, M. Enoeda, Sa. Sato, K. Furuya, T. Hatano, Y. Kosaku, T. Kuroda, S. Kikuchi, Y. Yanagi, J. Ohmori, Sh. Sato, S. Jitsukawa, H. Kawamura, E. Ishitsuka, K. Tsuchiya, M. Nakamichi, H. Yamada
RFHT:	A. Epinatiev, K. Skladnov
JCT Garching:	V. Chuyanov, K. Ioki, A. Cardella, F. Elio, N. Miki, T. Osaki, M. Yamada

IAEA TECHNICAL COMMITTEE MEETING ON FUSION SAFETY

by Dr. T. Dolan, Head, Physics Section, IAEA

The 7th international Atomic Energy Agency (IAEA) Technical Committee Meeting (TCM) on Fusion Reactor Safety has been held in Cannes, France, 13-16 June 2000,. The objective of this TCM was to exchange information on all aspects of fusion safety from present machines to future power plants.

The Technical Committee Meeting purpose was to learn lessons from current installations and studies important to the future. 35 papers were presented on seven topics.

Most of the papers were either based on ITER or were relevant to ITER. 45 participants from 10 countries have contributed to the discussions. A brief summary of the presentations within these seven topics is given below. This article is a condensation of the meeting summary prepared by Session Chairs G. Marbach, W. Gulden, I. Cook, C. Gordon, S. O'Hira, K. Moshonas, and D. Petti.

SAFETY APPROACH

C. Gordon discussed "Lessons learnt from the ITER safety approach for future fusion facilities", i.e. the need for a fusion specific approach to safety design guidelines.

D. Petti told about "Future directions in US fusion safety and environmental activities" showing the redirection away from a program focussed on an energy mission toward a program that is focused on advancing plasma science, fusion science and fusion technology, as the knowledge base for fusion energy development. US fusion activities have been broadened to include examination of a greater number of materials and coolants, safety support for a wider range of magnetic and inertial concepts and modification of safety analysis tools to analyze these different systems.

B. Kolbasov reviewed the "Fusion safety studies in Russia in 1996 - 2000" which were mainly related to ITER.

M. Costa analyzed the "Safety classification of the ITER fusion plasma shutdown system and resulting safety requirements" by relating them to the INES scale of IAEA.

In his talk on "Selection of DBEs for ITER EDA final design by GEMSAFE methodology" T. Sawada showed how 21 design basis events (DBEs) were derived and classified into three categories considering their occurrence probabilities and expected scales of their consequences.

The status of the "Development of a national licensing framework for ensuring safety of the fusion experiment facility in Korea" was reported by M.-J. Lee, emphasizing the successful completion of the first phase, the development of the conceptual design of KSTAR (Korea Superconducting Tokamak Advanced Research). Actual construction work of the facility is now underway to commission in 2004 as scheduled.

In his presentation "Safety activities JAERI related to ITER" S. O'hira emphasized that the inherent and high level of passive safety features implemented in the design of ITER FEAT can lead to realization of reasonable safety requirements in terms of the ITER safety objectives i.e., plasma is terminated by inherent plasma characteristics without any active measures, decay heat is low enough to cool only by radiation without cooling systems, fusion power and plasma energies can be handled within the design loads without extension to accidents, and that the design requirements of components with very high functional reliability for operation will also cover that for safety.

SAFETY IN NORMAL OPERATION

There were four papers : a general paper on inputting occupational safety considerations to the fusion design process, with special reference to ITER and three papers on occupational dose assessment for conceptual fusion power stations.

Taking into account also the results from other sections of the meeting, it is clear that, for fusion power stations, design for occupational safety must be jointly optimised together with other aspects of safety, environmental impact, capital cost and (especially) availability. This is particularly true in the context of schemes for fast replacement of blankets and divertors to produce high availability.

COMPUTER CODES FOR FUSION SAFETY

The papers show considerable progress in the area of computer codes for safety analysis over previous TCMs. Previously efforts were focused on identifying safety issues to be analysed and developing the tools to do the analysis. Now, as shown in this session, the effort is on refining the models to address specific safety design issues and on validating a complete set of analysis tools.

There were 6 papers presented related to the modelling of fusion specific phenomena or validation/benchmarking of these.

"Validation, verification and benchmarking in support of ITER-FEAT safety analyses" by L. Topilski presented the codes used in ITER-FEAT safety analyses and the status of their validation. In particular the validation of codes against ICE and EVITA experiments and a benchmarking exercise for water in the vacuum vessel were noted. "Modelling of fusion phenomena - fusion code benchmark" by P. Sardain reported on an international series of defined benchmark calculations of 9 codes against each other for ingress of steam/water into a vacuum chamber looking at pressurisation, critical flows and relief into a pressure suppression tank.

Participants in the Meeting

"Three dimensional calculations on dust mobilisation behaviour in fusion reactors at Loss-of-Coolant Accident" by K. Takase reported on a detailed calculation of air ingress into a vacuum chamber, dust mobilisation and transport compared with a simple experimental set up. "Modelling of ice formation and condensation on a cryogenic surface" by T. Marshall gave predications for the EVITA experiment on water/steam ingress onto a cryogenically cooled surface. "Fracture mechanics evaluation of a crack generated in SiC/SiC composite first wall" by R. Kurihara examined crack growth in a possible first wall material for a fusion reactor. The final paper by B. Merrill, "Modifications made to MELCOR code for analysing lithium fires in fusion reactors" reported on enhancements to access the "fusion safety fluid property database" and model lithium reactions.

Overall the interplay between design, safety analysis to identify key phenomena, data and parameters, and the experimental and benchmarking efforts to validate the codes and determine uncertainty was emphasised.

ACCIDENT ANALYSIS

In this session, results of two experimental studies of accidental conditions and two analytical studies of hypothetical events and assessment for ITER test-blanket were presented. Most of the studies were related to the ITER, but the others also included useful information to evaluate accidental condition of the ITER. Tungsten volatilization and hydrogen generation from the brush design have been studied under exposure to steam.

Also results of two-phase flow experiments with an Integrated ICE Test Facility were presented. In the experiments the fluid flow configurations inside the vacuum vessel at the ICE events were observed visually and pressure transients were measured quantitatively. Two-phase flow analyses were carried out with the TRAC code and the experimental results were validated numerically. From this study it was clarified that the ITER pressure suppression system is very effective to reduce the pressure rise during the ICE events and the thermal-hydraulic characteristics in the ITER can be predicted numerically.

The possibility of in-vessel hydrogen deflagration and detonation under a hypothetical accidental condition of the ITER-FEAT was analyzed. The maximum possible adiabatic pressure of deflagrations and momentum of detonation occurring at any pressure during transient air ingress into the vessel were evaluated for mixtures and clouds of stoichiometric mixture of hydrogen and air. The results for the ITER-FEAT limit of 5 kg showed a maximum deflagration pressure below the 2 bar design limit of the vacuum vessel (VV) and a detonation momentum which is within comparable design loads of the VV and its support structure, such as for the dropping of a blanket module from the top of the VV to the bottom.

Temperature transient calculations were reported for a hypothetical beyond-design-basis type accident in the European ITER design concept and the ITER-FEAT design. It has been established that, even when making the most extreme assumptions, the maximum outboard first-wall temperature of around 100 K occurs at about 1 year after loss of coolant. Thus, for all cases considered temperatures remain far below melting point for any of the plant structures.

The analysis performed so far has confirmed the expectation of little safety impact from the test modules, except in a few areas.

TRITIUM SAFETY

There were 4 papers presented in this session. The first two presented experiences at JET. The third discussed the collection and comparison of components, and the fourth compared the results of tritium release experiment in a caisson assembly with calculated predictions. The following conclusions were drawn:

- Tritium inventory control at JET relied on weekly inventory calculations as the on-line measurements of tritium transfers were insufficiently precise to ensure inventory control. Differences between inventory control calculations and measurements were of the order of 10 %.
- Analysis of samples collected in the process streams of the JET exhaust detritiation system proved that the dryers are capable of meeting and exceeding the design detritiation factor (DF) of 1000.
- Facility-specific failure rate data between US TSTA and Japan's TPL showed good agreement. The generic data from mound laboratory and Darlington TRF showed more variability for comparable components. Establishing a basis for failure rates is important for supporting accident analysis as well as for assessment of normal operations.
- Tritium release experiments inside TPL/JAERI's caisson assembly for tritium safety (CATS) provided information to predict tritium behaviour after a tritium leak accident in a ventilated room.

WASTES AND DECOMMISSIONING

A broad spectrum of papers were presented in the area of waste management.

Dr. M. Zucchetti gave an overview of the waste management strategy in Europe based on the SEAFP study. He reviewed recent results on the amounts of material that could be recycled inside the nuclear industry, recycled outside the nuclear industry and cleared as non radioactive material. He also reviewed the analysis of risks from fusion specific repositories and presented the key isotopes that dominate the risk at very long times and discussed recent encouraging results on detritiation of materials.

Mr. Klein discussed decommissioning activities at the BR3 site. It is considered a model decommissioning project in Europe.

Mr. Siller presented results of analysis of human intrusion scenarios for geological disposal of fusion wastes.

Dr. Forrest presented his recent work on SAFEPAQ-II, a relational database used to assemble, manipulate, store and retrieve cross section, decay, and nuclear reaction data needed for neutron activation calculations. The database is quite impressive and is a very efficient, comprehensive and useful tool.

Dr. Taylor reviewed results of calculations he performed on the neutronic aspects of SiC with respect to it safety and environmental performance. He examined the decay heat, gamma dose rate (important in maintenance and recycling), biological hazard potential as a measure of the waste hazard, and clearance index as a measure of the ability to clear ex-vessel components.

Dr. Broden looked at the management of fusion wastes for six potential fusion power plants in the SEAFP study using current rules for plans. In many of the SEAFP designs, about 30% of the material in the designs can be cleared using current IAEA recommended clearance limits and 60% can be recycled within the nuclear industry.

Thus, it was clear at this conference that there have been significant advances in the area of fusion waste management and decommissioning. The world fusion program is moving away from simple characterization of fusion waste to the development of an integrated strategy for management of fusion waste that emphasizes recycling and clearance.

SOCIO ECONOMIC ASPECTS

Dr. Cook explored the issues of the interface between studies of the safety and environmental characteristics and socio-economical aspects of fusion energy.

It was stressed in particular that safety and environmental objectives are really a top level priority for future energy as underlined by the utilities and the fusion energy advantages from this point of view are well documented. Even with a near term technology, cost of electricity from fusion energy becomes attractive in comparison with other energy sources if a carbon tax or any other account of environmental consequences is foreseen.

On the same topic Mr Hamacher presented the first results on external costs of future fusion plants obtained in the framework of the Program Socio Economic Research of Fusion (SERF) which was conducted by European Fusion institutions under the auspices of EURATOM.
The results bused on the options developed in the former SEAFP (Safety and Environmental Assessment of Fusion Power) indicate that external costs of fusion do not exceed those of renewable energy sources.

The number and diversity of participants associated with the quality of presentations are the best demonstration that the issue of safety in fusion is really a topic which is now mature, specific and well documented.

The best demonstration of this point is the definition of a fusion specific approach in the ITER project. This methodology is based on a generic methodology for nuclear installations, as defense in depth, but specific, well documented safety design guidelines have been produced.

In the case of a fusion facility, the main safety function is confinement, and various strategies are suitable especially with passive safety features as implemented in ITER-FEAT.

In conclusion, it should be emphasized that the safety activities on ITER have led to a great improvement of fusion safety assessment. Clarification of the safety approach, detailed safety analysis of accidents and occupational norms are now supported by a major effort of code validations. The lessons from current installations (JET, tritium labs) are also very important. Power Plant Studies demonstrate safety and environment advantages of fusion energy providing provisions are met in the design. Therefore, as in the case of ITER, the way to achieve the safety and environmental advantages of fusion and to facilitate the development of fusion energy is to follow an integration process between safety, materials selection, and design.

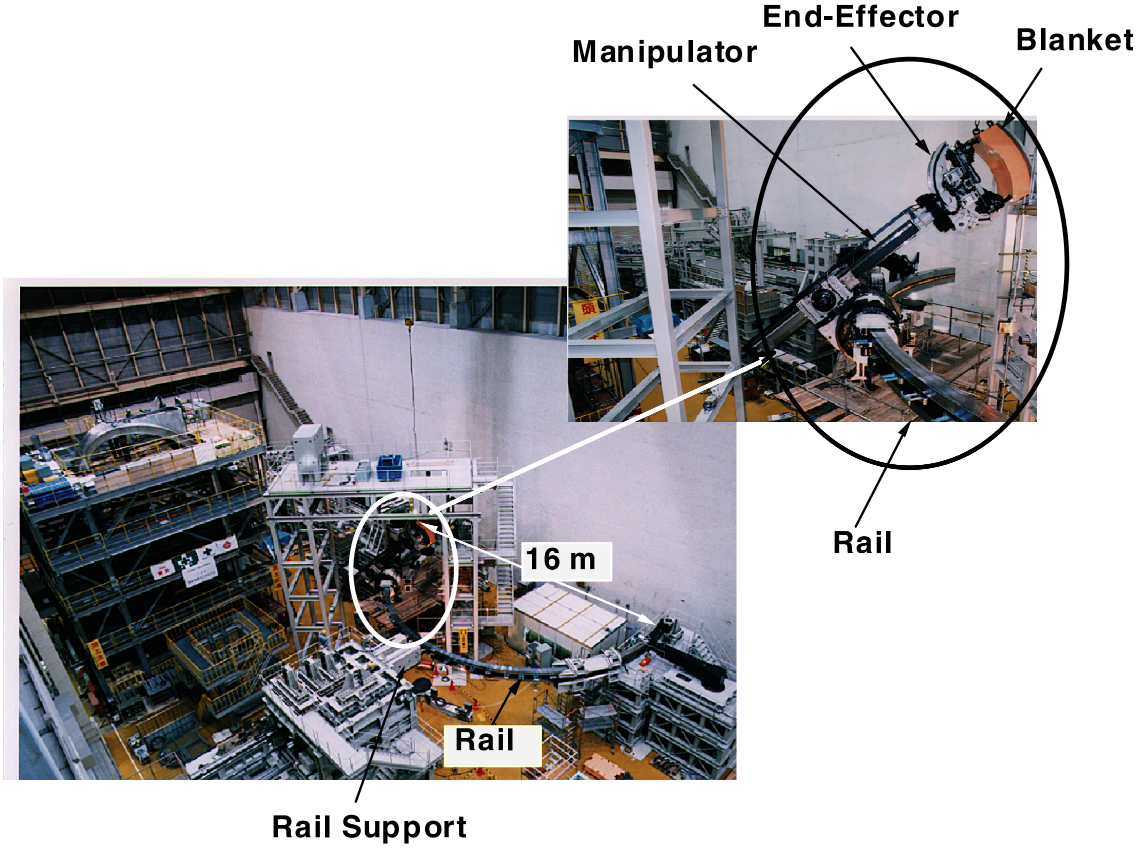
ITER L-6 Large Project "Blanket Remote Handling and Maintenance"

by Dr. K. Koizumi, Head, Reactor Structure Laboratory, Naka Fusion Research Establishment, JAERI

The main objective of the blanket module remote handling project is to develop and demonstrate the ability to remotely maintain blanket modules, including manipulating a 4 t module at a distance of 6 m with an accuracy of $\pm$ 2 mm. A rail-mounted vehicle system has been developed to handle the heavy blanket module within the limited space and with the required precision.

The basic performance tests of the full scale model were successfully completed in 1998 on the Blanket Test Platform (BTP) constructed at Tokai JAERI. The test platform comprises the module handling equipment, auxiliary remote handling tools and a blanket mock-up structure to reproduce the physical environment of a 180^{o} ITER in-vessel region. A suppression control technique to reduce dynamic deflection and vibration of the arm to negligible levels has subsequently been developed and successfully tested.

Blanket Remote Handling Test Platform and Full-Scale Manipulator



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