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)

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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 copper-