copper alloys. An interesting piece of research was in the development of a lithium-copper alloy that combined dispersion strengthening with greater sputtering resistance than pure copper.

MATERIAL ENGINEERING AND DESIGN

Thirteen papers were presented on the material requirements for elements ranging from testing of blanket components to material property needs for commercial reactors. These papers tended to concentrate on the engineering aspects of solving design problems associated with fusion reactors. The NET team described an approach to protecting the stainless steel first wall of their reactor with graphite tiles bonded to a molybdenum structure. Complementing this presentation was a paper describing the material development needs required to design plasma interactive components and the use of existing facilities to conduct the testing. The need for data was underscored in a presentation on the design issues encountered in designing the compact ignition tokamak. This study found that the dominant uncertainties in the strength properties of copper alloys and insulating composites were controlling the size of the toroidal field coils, which in turn was impacting the size of the experiment. In a change of pace, the implications of safety and environmental challenges on materials selection was also presented. This study identified the safety criteria that fusion needs to address in the design of reactors, along with possible solutions. The results of this study indicate that the single most significant factor in determining the safety and environmental characteristics of a deuterium-tritium reactor is the choice of materials near the plasma and first wall. Rounding out this session was a study that examined the criteria that commercial fusion reactors would probably be judged against before being introduced into the power grid and how materials development can be used to help improve this selection.

FUTURE SYMPOSIA

The next conference in this series will be held on October 4–8, 1987, in Karlsruhe, Federal Republic of Germany.

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SUMMARY OF THE FUSION REACTOR TECHNOLOGY CONTRIBUTIONS TO THE FOURTH EUROPEAN NUCLEAR CONFERENCE—ENC-86, GENEVA, SWITZERLAND, JUNE 1–6, 1986

INTRODUCTION

For the first time, Fusion Reactor Technology was presented under a separate heading in the European Nuclear Conference Series, reflecting the importance this technology has gained in the nuclear world. This was reflected as well in the Foratom exhibition where many participants exhibited their achievements in the field of fusion technology.

Approximately 2050 participants from 30 countries attended ENC-86. Contributions to the fusion reactor technology sessions consisted of 11 oral papers and 20 posters, which represent roughly 7% of the total contributions to the conference.

The present report reviews the most significant aspects of the fusion-related papers. Since all but one of the papers are European, this report reflects the European fusion technology approach. The papers were mainly of a general, informative character; hence, the present review is directed more toward the nonfusion specialist.

EXPERIMENTAL SYSTEMS

The oral session opened with a paper on the construction experience with the Joint European Torus (JET) in operation in the Culham Laboratory, United Kingdom, since 1983. Although this story has already been told, it is still fascinating. It reflects the strong management role taken by the JET team, the enthusiasm of all those involved, and the intimate team spirit apparent throughout JET and the contractors, which led to the assembly of the machine in a remarkably short time. The collaboration of 17 research organizations from 12 different countries and from contractors and suppliers from all over Europe resulted in completion of the machine close to budget and to a program established 5 years earlier.

In the invited contribution, P. H. Rebut highlighted the achievements of JET. At present, with moderate radiofrequency (rf) and neutral beam injection heating, JET is still a factor of 30 to 50 away from the Lawson criterion conditions. By the end of the operation, it is expected to reduce this factor to 3 to 5.

With the French Tore Supra, presented by the project leader Aymar, Europe intends to explore the field of the superconducting tokamak, with parameters close to those of the Tokamak Fusion Test Reactor and T15. Tore Supra is a medium high field machine (4.5 T) presently under construction in Cadarache. It is planned to be in operation by the end of 1987. The aim of Tore Supra is not to achieve ignition but to test a superconducting tokamak in conditions close to the operating conditions of a thermonuclear reactor and to contribute to the studies of initial heating of a plasma by transferring high power. The cooling of the superconductor is done by superfluid helium at 1.75 K.

SYSTEM STUDIES

Four system studies were presented at the conference:

1. near-future Next European Torus (NET) by R. Toschi (NET project leader)
2. reduced scale fusion reactor approach by Bathke [Los Alamos National Laboratory (LANL)]
3. stellarator reactor approach by Wobig (Max Planck Institute, Garching)
4. DEMO approach (Culham Laboratory team).

The NET is presently in the predesign phase. According to planning, the start of construction is foreseen in 1994, with operation in 2000. In the European fusion program, NET is the single step between JET and DEMO. NET aims, first of all, at full demonstration of reactor-relevant plasma performance. All technologies essential for a fusion reactor will be included.

The main performance objectives are:

1. safe ignition margin
2. long-pulse duration (~100 times longer then JET)
3. extensive testing program of in-vessel components.

The LANL view on the future of the magnetic confined fusion reactor was presented under the title, "Prospects for Improved Fusion Reactors." As a result of their study, they conclude that the size and cost of the fusion power core and of the reactor plant equipment, leading to an increased mass power density, has to be reached in order to make the fusion reactor competitive. This approach of the reduced fusion power core results in a higher power density and an increased wall loading. Figures of 20 MW/m^2 were cited. This will put enormous constraints on the first wall, requiring special high-heat flux materials. Although the physics data base is better developed for a tokamak than for the poloidal field (PF)-dominated systems, they claim that, based on recent advances, the latter offers better prospectives.

The apparent success of the stellarator/heliotron experiments during recent years has given new confidence in their development to fusion reactors. The main advantages of the stellarator line are:

1. inherent capability of steady-state operation, eliminating the cyclic thermal loading of first wall and blanket
2. absence of transient magnetic fields except for the startup phase
3. single coil system
4. disruption-free operation.

Three dimensionality introduces several new technical difficulties, however, as compared with axisymmetric devices, and issues such as the maximum achievable beta value, the transport processes, and the required plasma dimensions still remain critical with respect to the economic feasibility.

The Culham Laboratory DEMO system study is a further extrapolation of the International Tokamak Reactor and/or NET with an increase in size, wall loading, and a fusion power aiming at an actual electrical output of ~1100 MW. Similar to NET, a ceramic and a liquid breeder blanket is proposed. The liquid breeder is a Pb-30 Li alloy whose melting point is roughly 100°C higher than of the eutectic Pb-17 Li, putting a supplementary constraint on the compatibility with the structural material. To meet the high-heat load in the divertor, water-cooled thin tubes, fabricated from a W-Re alloy, with a thin layer of flowing molten tin, are proposed. Neutronic calculations have shown that a global tritium breeding ratio (TBR) in excess of unity can be attained.

COMPONENTS DEVELOPMENT

The development of large components is needed for future fusion devices. To increase efficiency and reduce cost, international collaboration between the European fusion research laboratories and industries is being established.

An example of the European approach of collaboration was presented by the Commissariat à l'Energie Atomique (CEA), France, and Kernforschungszentrum Karlsruhe (KfK), Federal Republic of Germany, for the development of vacuum components and for the common conceptual design and supporting research and development for the nuclear parts of NET; and JET and Sulzer in the field of neutral beam heating techniques.

On behalf of Euratom, KfK is participating in the International Energy Agency Large Coil Task. It has established a tight collaboration with industry for the development and fabrication of its coil. Prior to shipment, their coil was tested in the KfK test facility TOSKA. KfK is now engaged in the development of (a) a PF coil (8-m diameter) for Tore Supra based on a new conductor technology, and (b) the high-field conductor for the NET toroidal field (TF) coils (maximum field = 11 T) based on an A15 superconductor.

In the near future the TESPE experiment, a laboratory-scale arrangement of TF coils with a bore of ~1 m, will be used for magnet system safety experiments.

BLANKET AND FIRST WALL

About one-third of the fusion reactor technology papers dealt with blanket and first-wall aspects. A survey of the problems of the first-wall materials in fusion reactors and a comparison with the problems with canning materials in fission reactors were given by P. Schiller.

For the investigation of first-wall effects, two probe manipulators have been developed. The Swiss Federal Institute for Reactor Research (EIR) has delivered a robotic device equipped with sensors, fully remotely controlled, to the Tokamak Experiment for Technology Oriented Research (TEXTOR). It has been in service since 1984 for inserting and extracting first-wall probes. Culham Laboratory, in collaboration with the General Electric Company, is constructing a "JET surface probe fast transfer system," which allows probes to be inserted between limiters and the first wall and at the vertical parts of the torus of JET. The system is near completion.

The 14-MeV fusion neutrons may cause important changes of the mechanical properties of first-wall and blanket structural materials. No fusion neutron irradiation test facility currently exists that can give end-of-life (EOL) data. Therefore irradiation facilities that can simulate the damage caused by fusion neutrons are proposed.

EIR and the Swiss Institute for Nuclear Research (SIN) are installing a high-energy (600-MeV) high-current density (4 $\mu A/mm^2$) proton irradiation test facility, PIREX II, for simulating the 14-MeV neutron effects. According to Victoria (SIN), EOL damage states may be obtained within reasonable experimental time with this facility. An *in situ* testing system with the possibility of stress cycling is being developed. The actual thickness of the specimens will be limited by the allowed thermal gradient in the specimens.

A liquid and a solid breeder blanket are foreseen in the NET design. For the liquid the eutectic Pb-17 Li has been selected. Results from compatibility studies between Pb-17 Li and the candidate austenitic American Iron and Steel Institute (AISI) Type 316L stainless steel and martensitic WN 1.4914 structural materials from the CEA-Saclay and the Studiecentrum voor Kernenergie Centre d'Étude de l'Energie Nucléaire (SCK/CEN)-Mol have confirmed the lower corrosion rate of martensitic steel in the temperature range of

400 to 450°C. The corrosion rates measured in the LELI loop in Mol on the martensitic steel are higher than those reported earlier by other laboratories. A dissection of the complete LELI loop has shown large differences in the corrosion aspects in various parts of the loop. This gives another dimension to compatibility and needs further attention.

A large effort of the breeding materials program of the European Communities is devoted to ceramic breeders and ceramic breeder blanket concepts. Most helium-cooled solid breeder blankets exhibit a poor breeding capability. The CEA has developed attractive helium-cooled blankets based on breeder pin assemblies. They consist of hexagonal pin bundles, composed of alternate stacks of beryllium and γ-$LiAlO_2$ hollow pellets, externally cladded. The breeder composition is 20% γ-$LiAlO_2$ (60% enriched in ^6Li) and 80% beryllium that acts simultaneously as moderator and as neutron multiplier.

A design proposal for NET gives a TBR > 0.8 and for DEMO ~1.5. Attention has been given to the beryllium-breeder compatibility and to the beryllium-tritium radiation-induced aspects.

In another contribution, Gervaise (CEA) gives a detailed analysis of the neutronic problems in various fusion reactor blankets mainly with respect to

1. tritium production
2. cooling (H_2O, helium)
3. neutron multiplier (beryllium, lead)
4. breeder material (lithium, Pb-17 Li, Li_2O, $LiAlO_2$).

He points out the causes of inaccuracy of the obtained breeding ratios.

KfK has further developed their pebble-bed blanket design, which offers, according to M. Dalle Donne, good prospects for breeding. Intimate mixing of Li_4SiO_4 and beryllium pebbles gives a breeding ratio of 1.47. This can still be increased to 1.53 by addition of $ZrH_{1.7}$ to the back of the blanket. They have now changed the design of their pebble-bed canisters to breeder out-of-tube due to the possible compatibility problems between Li_4SiO_4 and beryllium.

The KfK breeder materials program was presented by Kummerer. It has been shifted from lithium-metasilicate to lithium-orthosilicate, mainly because the latter offers higher breeding prospects.

The CEA ceramic breeder materials program is oriented toward the fabrication and characterization of $LiAlO_2$. The program is very well structured. Much attention is paid to the influence of grain size on tritium release and on mechanical and physical properties. In- and out-of-pile results, obtained on a 0.38-μm grain material, are encouraging. Compatibility studies have shown sulfidation of AISI Type 316L stainless steel. As a result, the sulfur content in the $LiAlO_2$ pellets has been reduced to a level lower than 200 ppm.

MAINTENANCE

During the deuterium-deuterium phase, the vacuum vessel of JET will become slightly activated. Therefore adequate containment and personnel protection equipment will be needed for in-vessel maintenance in the near future. A torus access cabin has been developed for manual access operation inside the vacuum vessel. This will facilitate operation if beryllium is used as a first-wall protection.

During the deuterium-tritium phase, planned in 1991, the vessel will become too active to allow man access in the vessel. For this phase the torus access cabin will be equipped with an articulated boom. This boom has already been tested successfully. It is primarily intended to carry a servomanipulator into the vessel. It also allows transport of large components of limiter sections and rf antennas. A range of special tools for remote cutting and welding of components has been developed. The design study of the servomanipulator work is done by CEA-UGRA.

SAFETY AND ENVIRONMENT

The containment of tritium is one of the most important safety aspects of a fusion reactor. Djerassi has calculated the tritium concentration in each containment for normal, accidental, and maintenance conditions by means of a computer code TRITO. His main conclusion is that tritium has to be kept as close as possible to the source. Analysis of the blanket risk in case of a loss of coolant has shown that the environmental health consequences are quite low. The damages to the plant itself, however, may be very severe.

A complete fuel cleanup system based on existing technology of hydrogen isotope diffusion through a Pd-Ag membrane was presented by CEA.

First tests, performed by CEA, of a system for purification of very low tritium contaminated gas by catalytic oxidation at *ambient temperature* followed by liquid nitrogen cold trapping are very promising.

Absorptive removal of tritiated water by zeolites was studied by KfK. A major concern is that during the regeneration phase, especially by rapid heating, irreversible trapping of a significant fraction of absorbed tritiated species occurs. Heating to ~800 K in vacuum is needed to remove the "permanently" trapped species.

Besides tritium, the induced radioactive waste in the structural materials is of major concern. Culham Laboratory, in collaboration with the British steel industry, has proposed "low-activity" alternatives for existing martensitic and austenitic steels by elemental substitution and impurity control. Their study has shown that

1. elemental substitution and tailoring of the initial composition is required
2. significant component of the remnant gamma dose rate in the low-activity steels arises from the impurities
3. storage period of ~80 yr is required for all ferrous materials to permit decay to the "hands-on" surface dose rate.

ENC-86 TRANSACTIONS

The transactions of the ENC-86 Conference have been published in five volumes by the European Nuclear Society, P.O. Box 2613, CH-3001 Berne, Switzerland.

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