

SUMMARY

The Technical Committee Meeting (TCM) on Core Physics and Engineering Aspects of Emerging Nuclear Energy Systems for Energy Generation and Transmutation was held 28 November–1 December 2000 at Argonne, Illinois, USA. The meeting was hosted by the Argonne National Laboratory, and was convened by the IAEA on the recommendation of its Technical Working Group on Fast Reactors (TWG-FR).

The objectives of this TCM were threefold: to review the status of R&D activities in the area of hybrid systems for energy generation and transmutation, to discuss specific scientific and technical issues covering the different R&D topics of these systems; and to recommend to the IAEA activities that would be specifically targeted to the needs of the Member States performing R&D in this field.

The TCM had not called for broad overview papers of the various R&D fields. Apart from a rather brief presentation by each delegation of the general issues and the status of the R&D in the respective country, the IAEA had called for in-depth technical papers addressing one or more of the following topics: accelerator driven systems (ADS) concepts, requirements and features of ADS accelerators, target development, experiments and validation, sub-critical core studies, technology of heavy liquid metals, fuel and fuel processes development, and fuel cycle studies.

Forty-five participants from eleven countries and one international organization attended the TCM, and thirty papers were presented.

The status information presented in the delegates' general statements and in some of the papers is as of the time of the TCM. Thus, other later material should also be referenced for more current information. One such source of information is the Web Site of IAEA's project on *Technology Advances in Fast Reactors and Accelerator Driven Systems for Actinide and Long lived Fission Product Transmutation* (<http://www.iaea.org/inis/aws/fnss/>). However, the technical information provided in the papers, representing the bulk of the information presented, remains valid.

1. GENERAL STATEMENTS

1.1. Belgium

Approximately 60% of Belgium's electricity is from nuclear energy. The nuclear installed capacity is nearly 6 GW(e). The total heavy metal used per year is 114 t UO₂ (with an average enrichment of 4.5%) and 8.4 t MOX (with an average Pu enrichment of 6.4%). For the time being, the interest for ADS development, and for partitioning & transmutation (P&T) in Belgium is driven by waste [minor actinides (MA) and long lived fission products (LLFP)] management considerations. The waste management body (ONDRAF/NIRAS) is considering P&T as an option for waste management that deserves to be studied at an international level, and Belgium should play a role in this activity. The SCK•CEN Board of Directors has agreed on the proposal made by SCK•CEN Management to consider the transmutation as a research topic for the backend of the fuel cycle.

Nearly all the Belgian activities in this area are concerned with the development of the MYRRHA concept, a multi-purpose ADS for R&D. The facility should replace the BR2 reactor for material and fuel research, and radioisotopes production, but will be also useful for

transmutation research and medical applications. The main parameters of the accelerator (H^+ or HH^+ cyclotron) are:

$E_p = 350$ MeV, $I_p = 5$ mA. The spallation source is a liquid Pb-Bi windowless target. The sub-critical blanket is made of two zones: an innermost fast spectrum zone made of fast reactor MOX fuel cooled by Pb-Bi, and a thermal spectrum zone made of various in-pile sections located at the fast core periphery.

The ongoing activities are focused on establishing the pre-design of the facility, identifying the R&D needs to support this design, and performing a cost estimate of the project based on the pre-design meeting the scope of application given above. A first evaluation of the overall cost of the facility is around € 100 million (not including the SCK•CEN manpower effort), of which less than half is due to the accelerator. On the basis of a pre-design, SCK•CEN will ask the Belgian Government for funding. SCK•CEN is ready to cover 50% of the project. Other international partners should cover the rest). A more accurate cost estimate of the facility will be available mid 2001.

SCK•CEN participates in the EU 5th Framework Programme (FP5) in activities that are in line with the MYRRHA project (i.e. the projects SPIRE, TECLA, and MUSE, for the time being). Existing collaborations are with CEA (France), PSI (Switzerland), ENEA (Italy), NRC Soreq (Israel), IPUL (Latvia), FZR and FZK (Germany), NRG (Netherlands), IBA, Belgatom, and UCL (Belgium). MYRRHA is not considered to be an ADS demonstration (DEMO) plant, but has its place on the roadmap, as a first step, towards a DEMO.

More specifically, and apart from the activities of SCK•CEN concerning the projects of FP5, the activities related to ADS development presently conducted at SCK•CEN, UCL, IBA, Belgatom and Belgonucléaire are summarized hereafter:

1.1.1. SCK•CEN & UCL

SCK•CEN is willing to design and build an ADS system dedicated to R&D purposes. The applications foreseen in this facility will be on material and fuel research, MA and LLFP transmutation demonstration, ADS technological demonstration, and radioisotope production for medical applications. As complementary applications we foresee neutron and, possibly, proton beam applications for applied physics purposes and also for medical applications. The latter uses depend on establishing collaboration contracts with the academic and medical communities. In support of the MYRRHA project, SCK•CEN is conducting or preparing R&D programmes in the following areas:

- *Basic spallation data:* An experiment is presently conducted at PSI (Switzerland) in collaboration with PSI and NRC Soreq (Israel) consisting in bombarding a solid Pb-Bi target with protons produced by PSI's 590 MeV cyclotron. The parameters to be assessed are the spallation neutron yield (n/p) at 300, 350, 400 and 590 MeV, the spallation neutron energy spectrum, the spallation neutron angular distribution, and the spallation products induced by protons of 300, 350, 400 and 590 MeV.
- *Windowless spallation target design:* An experimental and theoretical programme is presently conducted by SCK•CEN in collaboration with the Department of Thermalhydraulics of the Catholic University of Louvain-la-Neuve (UCL), FZR (Rossendorf, Germany), IPUL (Riga, Latvia), and NRG (Netherlands) aiming at establishing a reliable design of a windowless spallation target made of liquid Pb-Bi. The programme foresees hydraulic experiments with water to simulate the Pb-Bi flow (UCL + FZR) (1999–2000), numerical simulation studies using the FLUENT, FLOW-3D, and STAR-CD CFD (UCL, SCK•CEN, NRG) (1999–2001), confirmation experiments of the MYRRHA windowless spallation module design using a heavy liquid metal (HLM),

namely mercury (SCK•CEN, IPUL, FZR) (2000–2001), and, ultimately, Pb-Bi experiments to be completed in 2001 (SCK•CEN, FZK, ENEA).

- *Validation of the windowless concept in connection with the proton beam vacuum tube:* An experiment called VICE (Vacuum Interface Compatibility Experiment) is presently under preparation at SCK•CEN. The experiment aims at demonstrating the feasibility of connecting a MYRRHA windowless target concept to a vacuum environment compatible with the characteristics of the proton beam delivered by the cyclotron, i.e. the out-gassing of the HLM and the structures, and the assessment of the Pb-Bi evaporation rate in the MYRRHA operating conditions (Pb-Bi at 500°C, vacuum 10^{-6} to 10^{-8} bar).
- *Structural material research:* A research programme is already going on at SCK•CEN based on irradiation in the BR2 MTR, to study the behaviour under high neutron dose irradiation of HT9 (martensitic steel) and EUROFER (reduced activation ferro-martensitic steel, comparable to the F28H Japanese steel) as potential material candidates for MYRRHA structures.
- *THOMOX fuel research:* An R&D proposal in collaboration with TUI (EU) and Belgonucléaire (Belgium) on a comparative study of (Th-Pu)O₂ fuel behaviour as compared to classical MOX or advanced MOX (large grain) is presently under preparation by the partners. The programme foresees fuel fabrication, irradiation in the BR2 MTR under well-characterised conditions (power level, burnup, central temperature, neutron flux), irradiation in stationary and later on in transient conditions, and Post Irradiation Examinations (linear power, burnup, TOP-18 FP, profilometry, SEM, FG punction, gamma scanning, X radiography).

1.1.2. IBA

IBA is a partner of SCK•CEN working on the development of a reliable proton accelerator operating in CW mode and fulfilling the required beam availability specifications for ADS. The present design of the sub-critical core requires the accelerator to deliver a 350 MeV, 5 mA proton beam. This 1.75 MW CW beam has to satisfy a number of requirements, some of which are unique in the world for presently existing accelerators. At this level of power it is compulsory to obtain an extraction efficiency above 99.5% and a very high stability of the beam, but on top of that the ADS application needs a reliability well above that of common accelerators, bringing down the beam trip frequency (trips longer than a few tenths of a second) to below 1 per day. The design principles are based on the following lines of thought:

- Statistics show that the majority of beam trips are due to electric discharges (both from static and RF electric fields). Hence the highest reliability requires minimizing the number of electrostatic devices, which favours a single stage design.
- In order to obtain the very high extraction efficiency, two extraction principles are available: through a septum with well-separated turns, or by stripping.
- The beams are dominated by space charge. Therefore one needs careful transverse and longitudinal matching at injection, and the avoidance of cross talk between adjacent turns (by an enhanced turn separation) if a separated turn structure is required for the extraction mechanism.
- The space charge dominated proton beam needs a 20 mm turn separation at 350 MeV if a septum extraction has to be implemented. This solution requires the combination of a large low-field magnet and of very high RF acceleration voltages for realizing such a large turn separation, and also an electrostatic extraction device.
- In view of what precedes, this solution is not well suited for very high reliability operation. Extraction by stripping does not need separated turns. It may be obtained by the acceleration of H⁻ ions, but the poor stability of these ions makes them extremely sensitive to electromagnetic stripping (and hence beam loss) during acceleration. The use of H⁻

would, therefore, lead to the use of an impracticably large magnetic structure. The other solution is to accelerate 2.5 mA of HH^+ ions up to 700 MeV, where stripping transforms them into 2 protons of 350 MeV each, thus dividing the magnetic rigidity by 2 and thereby allowing extraction. This solution reduces the problems related to space charge since only half the beam current is accelerated. However, the high magnetic rigidity of a 700 MeV HH^+ beam imposes a magnetic structure with a pole radius of almost 7 m, leading to a total diameter of the cyclotron of close to 20 m. The cyclotron would consist of 4 individual magnetic sectors, each of them spanning 45 degrees.

- At the present stage of R&D the last option appears to be the most appropriate one.

1.1.3. BELGATOM

BELGATOM has joined the European Industrial Partnership for developing the ADS DEMO and has expressed its interest in investing some manpower on the ADS topic in the coming years. An implication of BELGATOM in MYRRHA project is expected in the engineering of the auxiliary system and integration of the different components of MYRRHA.

1.1.4. BELGONUCLÉAIRE (BN)

Over the recent years, Belgonucléaire's (BN, Brussels) transmutation activities were mainly concerned with the possibilities of recycling Pu and minor actinides (MA) in presently existing LWRs, either in the homogeneous mode (in an extended MOX fuel) or in the heterogeneous one (MA placed in target pins). A concept was developed, in which target pins loaded with Am, or with Am and Cm, could be put on the corner positions of the MOX assemblies in the core. Attention was paid to the approach to equilibrium in a multiple recycling scheme. The dose rates associated with the fabrication and handling of target pins were determined, and the protections needed were derived. A study considered the incineration of long lived fission products (LLFP), more specifically ^{99}Tc and ^{129}I , in these reactors, but showed that LWRs were not suitable for this. BN also examined, in co-operation with EDF, the multiple recycling of Pu and MA in fast reactors of the EFR (European fast Reactor) type. The following longer term scenario was found to be advantageous: nuclear power plant park consisting of LWRs fuelled with UO_2 (producers of Pu and MA) and of LMFRs of the EFR type without fertile blanket (incinerators of Pu and MA); MA target pins could be loaded on special, moderated positions of the LMFRs, with the objective of reaching 90% burnup, with no further reprocessing of the spent targets. So far, BN did not participate in ADS studies but is intending to participate in the fuel and core design of the MYRRHA project.

1.2. China

The China Institute of Atomic Energy (CIAE) is pursuing a long term scientific and technical project for the development of national, sustainable fission energy, including the construction of an experimental fast reactor (CEFR), the investigation of the MA transmutation capacity of a medium size fast reactor, and basic research in the area of the innovative technical option 'ADS'.

As the first step of the fast reactor engineering development in China, the CEFR is a sodium cooled pool type experimental fast reactor of 65 MW thermal power, matched with a 25 MW turbine generator. By now, the CEFR is just at the beginning of its construction marked by the completion of the reactor building base foundation. It is expected that its first criticality will be at the end of 2005. The second step is to design and build a 300 MW(e) prototype fast breeder reactor (PFBR). It is intended to use the PFBR to burn PWR's MA waste (named

MPFR). With MA content in the fuel of up to 5%, the support ratio of 2 could be reached, provided that acceptable dynamic parameters will be proven.

A five-year programme (Phase 1) of basic research on some physical and technical problems related to ADS is being implemented. A preliminary evaluation of the main performance parameters of the following three ADS has been performed:

- 1 GeV/16 mA accelerator driven fast sub-critical system with a lead spallation target (AD-FBR) could get a support ratio of approximately 4 for the transmutation of MA from PWRs;
- As the previous system, but with a PHWR type sub-critical core (AD-PHWR): the main result is that with the requirement of fissile self-sustainability, U-Pu fuelled sub-critical cores would require a higher beam current compared to the Th-U fuelled case;
- The analysis of a fast-thermal coupled ADS showed that for these systems the required accelerator beam power could be reduced.

Further, some target physics studies have been conducted. The neutron double-differential cross-sections of lead have been measured by irradiating a lead thin target with 1.5 GeV protons at 15°, 30°, 60°, 90°, 120°, and 150°. The theoretical analyses have been performed with the help of the SHIELD code. The results of these calculations agree very well with the experimental data. The neutron yields from lead and tungsten standard thick targets (Ø20 cm × 60 cm) have been calculated with both a code based on the Monte Carlo method and the SHIELD code. The results are rather consistent. Further, for the Ø20 cm × 60 cm target as well as for thin targets containing various materials, the proton energy deposition and the irradiation damage cross-sections have been calculated with the help of the SHIELD code. Based on the results of Phase 1, an ADS based on a 300 MeV/3 mA LINAC, using a 0.9 MW lead spallation source driving a uranium-water sub-critical lattice is the proposed design for an ADS verification facility in China. This will be the work scope for Phase 2 of the Chinese ADS R&D programme, planned up to the year 2007.

1.3. Czech Republic

The Czech Republic has been operating PWRs for more than 15 years with the total installed capacity of 1760 MW(e). Just recently (October 2000), the first block (1000 MW(e)) of a new NPP has been started. The regular operation of this block and the startup of the second one (also 1000 MW(e)) has been planned for the year 2001. This means that the amount of spent fuel that is stored in an intermediate storage at the NPP site will be increasing significantly, and, simultaneously, public awareness of the issues linked to spent fuel management will be increasing, too. Therefore, the Czech Republic is very much interested in the development of innovative nuclear energy systems for energy generation and, more importantly, for the transmutation of the accumulated long lived radwaste. The Czech efforts in that direction are focused on a system with liquid fuel, namely in the form of molten fluorides. The developments so far, as well as the current state of the art of this technology are summarized in the following.

After a preparatory study stage in the first half of the 90s, the national consortium TRANSMUTATION has been established at the end of 1996 by four leading institutions in the nuclear R&D field in the country, namely by the Nuclear Research Institute Rez p.lc (NRI Rez), the Nuclear Physics Institute of the Academy of Sciences, SKODA Nuclear Machinery Ltd., and the Czech Technical University. The consortium proposed an R&D Programme for the ADS with molten fluorides as liquid fuel that was approved and has been supported by

national governmental authorities since the FY 1998. As of 2000, the first 3-year stage of this R&D Programme has been completed. The main results of this first stage are as follows:

(1) Verification, through computational analyses, of an adopted sub-critical blanket design. Preparation of the basis for the experimental validation of the concept, represented by the experimental reactors LR-O in the NRI Rez and VR-I at the Czech Technical University, as well as a series of technological loops with molten fluorides and the FREGAT pyrochemical line

The theoretical and experimental studies of kinetics of the sub-critical blankets started with different composition of fluorides including fissionable component (UF_4).

(2) A second 3-year stage (2001–2003) was proposed on the basis of the results of the first stage of the R&D Programme. The studies in this second stage will concentrate on experimental verification of the design parameters of a demonstration transmuter with liquid fuel (molten fluorides). The proposal has been approved in September 2000 for financial support by both the Government and the Radwaste Repository Agency.

1.4. Germany

In Germany, 19 Nuclear Power Stations are in operation. One large power station — Mülheim Kärlich — is shut down because of formal reasons. The total capacity is about 22 GW(e). The operation of the power stations is extremely satisfactory. About 5 to 6 of the German power stations are regularly among the worldwide top ten leading stations, with respect to the number of MWh produced per year, and with respect to operational availability. The annual amount of TWh's produced in German nuclear power stations is about 160 TWh/a, i.e. about 1/3 of the total German electricity production.

The present two-party coalition German government is supported by the Socialist Party and the Green Party. The government was elected in late in 1998. From the very beginning, it declared the firm intention to phase out nuclear energy. First attempts to stop reprocessing failed. It soon became very clear that an immediate stop of operation of the German nuclear power plants was to be excluded, both because of technical and legal reasons. Lengthy negotiations between the government and the utilities followed. This led to a “consensus agreement” in June 2000. The basic elements of this agreement are as follows:

- For each nuclear power plant the amount of electricity which can be produced from 1 January 2000 on is limited. The operation license will cease when this amount of electricity has been produced. In total, 2600 TW(h)'s will be produced (assuming 160 TW(h)/a, this means continuation of nuclear power plant operation in Germany for another 16 years).
- German government and utilities confirm the high safety standard of German nuclear power plants.
- From 2005 on direct disposal — no reprocessing.
- Gorleben is suited to be a final repository. But further evaluations of other sites will be initiated. Evaluation of Gorleben will be interrupted for at least 3, and for a maximum of 10 years.

Taking into account the existence of the “consensus agreement”, it becomes evident that from the government side there is no strong support for the development of new reactor concepts. However, in the “consensus agreement” it is clearly stated that R&D shall not be regulated but shall stay free. In an “evaluation report” which has been prepared by the leading research centers dealing with nuclear and by the federal ministries in charge of nuclear energy related

issues, it is clearly stated that the international development of foreign innovative concepts shall be followed. In particular P&T is mentioned, since it is considered as a possible alternative in the area of waste management. FZK has prepared a limited programme in this area. Because of limited resources, the programme can only be performed within an international cooperation framework, in particular the European cooperation, but also cooperation with the Russian Federation, Japan and the USA. Some important projects FZK is participating in are the CAPRA/CADRA project, the MEGAPIE project, and the MUSE project. The main elements of the programme are:

- Theoretical studies (thermal hydraulics, neutronics, safety, development of new models, new data etc);
- Development of lead / lead-bismuth-technology (KALLA: 3 lead / bismuth loops, maximum capacity 4 MW / 4 m³/h);
- Target development (participation in the MEGAPIE project, ISTC #559, and international benchmark problems).

All the activities of the FZK programme, as already mentioned, are embedded into international activities, such as the 5th Framework Programme of the EU, or the “Enlarged Technical Working Group” (ETWG), chaired by Professor Rubbia.

1.5. Italy

From 1995 on, a growing interest in ADS concepts has taken place in Italy and has given origin to several basic R&D activities and to an industrial programme involving ENEA (the Italian national research body for energy, environment and new technologies), INFN (the Italian national research institute for nuclear physics), and various industrial partners. Accelerator driven systems, coupling an accelerator with a target and a sub-critical reactor, could simultaneously burn minor actinides and transmute long lived fission products, while producing a consistent amount of electrical energy. As a first step, this interest was confirmed by the national R&D programme TRASCO (Italian acronym standing for waste transmutation – TRAsmutazione SCOrie), funded and started in 1998 under the leadership of INFN for the accelerator, and of ENEA for the sub-critical system. The TRASCO programme was relevant to promote collaboration among groups of different competencies (accelerator, reactor physics, plant design), and will provide important results in support of any related industrial programme. As a matter of fact, parallel to the basic activities of TRASCO, an industrial programme was proposed in two main steps (the first was approved and funded in 1999):

- On-going short term activities in the Italian context to produce a preliminary design of the ADS experimental plant. A reference configuration (named EADF, Energy Amplifier Demonstration Facility, approximately 80 MW(th)) has been proposed and submitted for consideration to the European partners as a contribution to the discussion of the European Road Map for developing an ADS experimental plant. The preliminary design will be completed in the first half of 2001, whilst the main supporting R&D needs were assessed and the realization of the experimental facility CIRCE is in progress, for thermal hydraulic testing of the main subsystems of the experimental plant in lead-bismuth.
- Medium term activities in an European and international context with the aim of performing the detailed engineering design, the realization, and the commissioning of the experimental plant along with all the supporting R&D.

1.6. Japan

In 1988, the Atomic Energy Commission launched the Long Term Programme for Research and Development on Nuclide Partitioning and Transmutation (P&T) Technology (OMEGA Programme). Within the framework of OMEGA, the basic studies and experiments of P&T

concepts have been carried out by the Japan Atomic Energy Research Institute (JAERI), the Japan Nuclear Cycle Development Institute (JNC), and the Central Research Institute of Electric Power Industry (CRIEPI). JAERI is developing concepts of accelerator driven systems (ADS) dedicated to transmutation. The system employs minor actinide (MA)-nitride fuel and lead-bismuth coolant/target. Fabrication and reprocessing technology is being developed for nitride fuel. Development of lead-bismuth coolant/target technology has started recently. Accelerator technology is being developed to construct a powerful proton accelerator under the JAERI-KEK Joint Project. ADS physics and engineering experiments are planned within the framework of the latter. In contrast to JAERI, JNC and CRIEPI are studying transmutation technology using fast breeder reactors (FBRs). JNC is developing a MOX-fuelled sodium cooled FBR. Core design studies, MA-containing MOX fuel fabrication and irradiation experiments are being carried out. Recently, JNC started a feasibility study to evaluate a variety of technology options for commercialized FBR cycle systems. CRIEPI is developing concepts of metal-fuelled sodium cooled FBRs. Fabrication and reprocessing technology is being developed for MA-containing metal fuel.

The Atomic Energy Commission reviewed the progress of the OMEGA Programme in FY 1999. In March 2000, the Commission issued a report entitled Research and Development in Japan on Long Lived Nuclide Partitioning and Transmutation Technology. The review of the status of R&D, the analysis of the effects and significance of P&T technology, and the recommendations for future R&D are contained in this report. It is recommended to continue with the system design work, scenario studies, and basic experiments before embarking on engineering tests for demonstration. It is also recommended to review the progress, to update scenarios, and to reconsider the R&D plans around the year 2005. It must be noted that the importance of active participation in international cooperation projects is underlined in the report.

1.7. Republic of Korea

In the Republic of Korea, an accelerator driven system named HYPER (HYbrid Power Extraction Reactor), is being developed at KAERI within the framework of the national mid and long term nuclear research plan. The basic mission of HYPER is the transmutation of both transuranic elements (TRUs) and long lived fission products (LLFPs). The minimum required sub-criticality of the HYPER core is set to $k_{\text{eff}}=0.97$, and the rated power is 1000 MW(th). Lead-bismuth is used as the coolant due to its benign chemical characteristics. A proton current of 1 GeV impinges on a lead-bismuth target in the core center and generates about 29.3 spallation neutrons per proton. Transmutation of the radioactive nuclides, in general, is based on a closed fuel cycle. Consequently, the related fuel cycle should have a high proliferation-resistance. In HYPER, a simple fuel cycle is assumed, in which spent fuels from commercial reactors are reprocessed with a highly proliferation resistant pyroprocess, such as electro-refining. The TRU fuels contaminated with a significant fraction of lanthanides are incinerated in the HYPER core. The spent fuel of HYPER is also recycled after undergoing a similar proliferation resistant reprocessing. It is claimed that the fuel cycle for the HYPER system has an excellent compatibility with the non-proliferation requirements.

Currently, R&D activities are mainly focused on the core and beam window designs. The HYPER core adopts a hexagonal type fuel array to render the core compact and to achieve a hard neutron energy spectrum. In order to keep the radial power peaking within the design target value of 1.5, the blanket region is divided into three TRU enrichment zones. A low TRU fraction fuel is loaded in the innermost zone and a high TRU fraction fuel in the outermost region. The refueling is to be performed based on the scattered loading with

3 batches for each zone. Target cycle length at the equilibrium state is 1 year with a 75% capacity factor. In order to minimize the reactivity swing of the HYPER core loaded with TRU fuels, two types of burnable absorbers are under investigation, one is B₄C-coated cladding and the other one a mixture of B₄C and ZrH₂. Preliminary studies show that B-10 can be effectively used as a burnable absorber, especially the mixture of B₄C and ZrH₂ can reduce the reactivity swing by a factor of 2, with only a little compromise of the transmutation quality. An important research area is the maximization of the spallation neutron multiplication, in other words, minimization of the proton current. The overall optimization of the core is in progress such that the source multiplication can be maximized.

A unique feature of HYPER is the transmutation of ⁹⁹Tc and ¹²⁹I in a locally thermalized neutron spectrum six fission product containing assemblies are loaded in the middle ring of the core in order to make the support ratio of the fission products similar to that of TRU (4–5). In the fission products target region, metallic ⁹⁹Tc is loaded as a plate in the periphery and NaI or CaI₂ rods are placed in a staggered manner with CaH₂ moderator rods, inside the Tc plate.

Either TRU-Zr metal alloy or (TRU-Zr)-Zr dispersion fuel is being considered as a blanket fuel for the HYPER system. In spite of the successful application of the metal fuel in the conventional LMRs, it is not easy to control the vaporization of americium nuclides in the fabrication process of an alloy type fuel rod. In the case of the dispersion fuel, the particles of TRU-Zr metal alloy are dispersed in the Zr matrix. The cladding material is ferritic-martensitic steel, HT-9. It is expected that the dispersion fuel will generally withstand significantly higher burnup than the alloy fuel.

The proton beam is delivered into the core central region through a beam tube and the target zone is separated by a 2 mm-thick window of 9Cr-2WVTa. Related research work is focused on the lifetime analysis of the window. The target lifetime of window is at least one year. Preliminary results indicate that, as far as the mechanical integrity is concerned, the window seems to withstand 6 mA current of 1 GeV protons. Currently, more detailed analyses are ongoing. In HYPER, a 3-loop heat transport system is adopted in order to keep the coolant speed below the maximum value of 2 m. The p/d (pitch-to-diameter) ratio of the HYPER core is 1.5, and the corresponding Pb-Bi velocity is 1.1 m/s. Instead of wire spacer commonly used for tight lattices, grid spacers are suitable to ensure proper separation of the fuel rod. Thermalhydraulic analyses are being performed with a modified SSC code. In the next phase of R&D, the research will be tuned to the core thermal hydraulic analysis including the target area.

The whole development schedule for the HYPER system is divided into three phases. The basic concept of the system and the key technical issues are identified in Phase 1 (1997–2000). In Phase 2 (2001–2003), core design optimization will be performed to maximize the HYPER economics. Also in phase 2, some experiments will be conducted to confirm the key technical issues. A thermal hydraulic test for Pb-Bi, an irradiation test for the fuel, and a spallation target test are the major experiments that KAERI is considering. In Phase 3 (2004–2007), a conceptual design for the HYPER system will be finished by completing the development of design tools based on the experiments.

1.8. United States of America

In Fiscal Year (FY) 1999, the US Congress directed the Department of Energy (DOE) to develop a “roadmap” to determine the feasibility and life-cycle cost of an accelerator transmutation of waste technology (ATW) system to manage the civilian nuclear spent fuel

from US reactors. The results of this study are published in a report to Congress, "A Roadmap for Developing Accelerator Transmutation of Waste Technology," October 1999. The report addressed issues associated with the question of whether the ATW concept could benefit the waste management system associated with disposing spent fuel from all existing US nuclear plants during their anticipated lifetimes. If successful, the ATW concept transforms plutonium, other long lived actinides, and long lived fission products contained in spent fuel by changing atomic structures. After transmutation, the new less radioactive isotopes can be stored in a permanent repository. By transmutation of long lived radioactive isotopes in spent fuel, the potential exists to: (a) significantly reduce the quantity of fissionable and radioactive material stored in a repository, (b) produce further power from material in spent fuel, and (c) as a result of associated fission product partitioning, permit repository heat management with the potential to significantly enhance the capacity for waste storage in the repository — possibly eliminating the need for additional repositories in the foreseeable future. The US Congress authorized DOE to begin an ATW research programme in FY 2000.

Independent of the ATW research, DOE is also responsible for insuring adequate future tritium production needs of the nation. Although a reactor based option has been chosen as the primary technology to meet the needs, an accelerator production of tritium (APT) backup option exists which to date has developed impressive technology which appears of direct value to demonstrating the ATW objective of benefiting nuclear waste management.

Recognizing the potential synergistic value of ATW and APT, as well as the possibility of supporting the nation's nuclear energy-related education and research infrastructure with an integrated accelerator applications facility, the DOE Office of Nuclear Energy, Science and Technology, under Congressional guidance, established a new Advanced Accelerator Applications (AAA) Programme Office and initiated in FY 2001 an AAA programme whose primary objective will be to conduct scientific and engineering research, development and demonstration to ascertain if cost effective and beneficial transmutation of civilian spent nuclear fuel is possible.

The mission of the AAA programme is to establish a national nuclear technology research capability, a nuclear engineering test bed that can carry out effective transmutation and advanced reactor research and development effort.

The AAA programme will address the following objectives:

- Demonstrate the practical performance of an accelerator driven sub-critical multiplier;
- Demonstrate the viability of transmutation for waste and spent fuel management;
- Enhance the Nation's nuclear science and technology education infrastructure;
- Provide demonstration of a robust backup tritium production capability for national security, if needed.

An accelerator driven test facility (ADTF) would be the key facility for the AAA programme. Existing facilities can support partial completion of the programme objectives, but a new, dedicated facility is required for fully reaching the goals of the AAA programme, which includes demonstrating the safe and efficient coupling of an accelerator, spallation target, and sub-critical multiplier. The ADTF would be designed to perform specific experiments related to the demonstration of waste transmutation, and would have the flexibility to carry out a broad range of experiments for a wide range of advanced nuclear technologies. A target of 2010 has been set as the time frame in which the ADTF would become operational.

This approach will provide the foundation for a new area of nuclear engineering research and technology. AAA research will also contribute to the reinvigoration of nuclear science and

engineering and help train a new generation of nuclear energy experts. Last, this programme will contribute, and support US leadership in advanced nuclear technology through the establishment of a new, first-of-a-kind, world-class research facility.

The research topic areas reflect the fact that the AAA programme integrates the science of spallation-neutron physics with nuclear reactor physics, thermal-hydraulics, heat transfer, and materials into one coupled system that is generally referred to as an accelerator driven sub-critical system. Presently, the research proposed within the AAA programme includes:

- Advanced fuels: R&D on high zirconium/actinide, nitride, mixed oxide, and carbon matrix fuels
- R&D on spent fuel separations (oxide and metal)
- R&D on environmentally acceptable, and cost effective fuel processing (conversion of LWR spent fuel in oxide form to metal fuel for transmutation)
- Sub-critical multiplier neutronics and thermal-hydraulics design
- R&D on coolant compatibility and heat transfer (gas, sodium, and lead/lead-bismuth eutectics)
- Spallation target design, cooling, and the science of coupling spallation neutron sources with sub-critical nuclear lattices
- R&D of new materials which can withstand high energy, high neutron fluence levels, high temperatures, and be compatible with the reference fuel and coolant
- High energy accelerator reliability.

2. SESSION SUMMARIES

2.1. Session 1: Research and development programmes

The research and development session heard papers from the United States of America, Belgium, Italy, Japan, and China. The papers showed a wide variety of concepts, stages of development and mutually interesting basis for potential cooperation.

Presentations from the US gave an overview of a new US Department of Energy Advanced Accelerator Applications (AAA) programme whose research and development plan proposes operation initiation of a 20 to 100 MW proof-of-performance test facility in ten to twelve years.

A Japan Atomic Energy Research Institute representative (JAERI) described current development considerations for ADS under the Japanese OMEGA programme with heavy emphasis on fuels and materials development.

The representative from the China Institute of Atomic Energy described a growing civil-application programme in China with the launching of a joint effort to pursue both fast reactor technology development and ADS basic research.

The Belgium presentation from the SCK•CEN nuclear research center summarized the state of development in their MYRRHA project aimed at having a 20–30 MW ADS facility operational in about eight years utilizing a noteworthy “windowless” target.

The Italian presenter from ENEA, the Italian national research body for energy, the environment, and new technologies, described the parallel efforts of their collaborative TRASCO R&D programme and their national industrial programme supporting the design of an European ADS demonstration plant.

Common themes: country representatives presenting papers in this session appear to have concluded there is sufficient potential value for an ADS to contribute positively to the management of radioactive nuclear waste to warrant recommending enhanced national and international development efforts. Plans were also noted for at least one integrated accelerator/target/multiplier experimental or demonstration facility in the next 8–15 years. Although various combinations of coolant and target material are being considered, lead-bismuth or tungsten targets and some form of liquid metal cooled fast spectrum multiplier facility appear primary in choice for facilities under consideration. In addition, all presenters emphasized the need, and value, for enhancing international collaboration, particularly in the near-term for fuels and materials testing, both in and out-of-pile.

Unique themes: as part of the research and development session, several unique programme features were identified. These include:

- The Belgium paper presented a unique windowless target design concept wherein problems associated with designing a target window to withstand effects associated with intense accelerator beam currents are replaced by problems associated with designing features into the windowless target for required thermal-hydraulic behavior in the presence of spallation products. A significant development and design programme appears underway to firm up the windowless design;
- The Japanese paper identified that serious consideration is now being given in the JAERI programme to use of a nitride based fuel design for the sub-critical multiplier, in which generation of the unwanted ^{14}C isotope is to be suppressed by using nitride fuel enriched with the ^{15}N isotope. Additional reasons given for considering development of nitride fuels include: good thermal conductivity, higher heavy metal density with a resulting harder neutron spectrum than with oxides, as well as other good metallurgical properties. The presenter further noted that the JAERI accelerator driven reference concept includes a dedicated minor actinide consuming reactor system, operating as the second part of a “double-strata, or tier” nuclear fuel cycle system, in which the first strata would be a normal power producing nuclear reactor, utilizing plutonium while producing electricity;
- The Chinese paper noted a major increase in development plans for the Chinese civilian nuclear power programme in which parallel ADS and fast reactor system development programmes are contemplated. Unique to a design under consideration is an accelerator driven sub-critical multiplier system with separate fast and thermal zones incorporated in the single sub-critical assembly;
- US presentations summarized a “multi-mission” ADF programme being pursued in the US, in which demonstration of tritium backup production capability is included as one of the missions anticipated for the ADF.

2.2. Session 2: Concept developments

The concept developments session heard papers: (a) on trade studies performed in the US within the frame of the AAA programme to define and compare possible ATW candidates (lead-bismuth eutectic, sodium, and gas cooled systems were considered), (b) on a two-tiered approach based on the incineration of plutonium in LWRs fuelled with fertile-free fuel and on the transmutation of the remaining LWR spent fuel together with the actinides from the spent fertile-free plutonium fuel in an accelerator driven system; and, finally, (c) on economics aspects of the transmutation fuel cycles.

Important assumptions for the liquid metal cooled systems were: fertile-free fuel; no control rods; power of 840 MW(th). Goal of the optimization was to reach a maximum discharge burn up in order to minimize overall losses during multiple recycling. At the same time, the

reactivity loss per cycle was to be minimized, in order to minimize the required increase in accelerator power needed to compensate the loss of reactivity. It turned out that solutions could be found, but the basic assumptions have a significant feedback on the design, and limit design flexibility. In the discussion the question was raised whether the basic assumptions should or could be modified in order to regain more flexibility; e. g. the significant increase in accelerator power at the end of cycle could be avoided, if control rods would be foreseen.

The studies for gas cooled systems assume TRISO (coated particles) as fuel. This fuel allows extremely high burn up values. Therefore it is proposed to limit the number of reprocessing steps. In general it is proposed to have only one reprocessing, irradiate the fuel and put the irradiated fuel into the repository. Different irradiation schemes were investigated. Thermal as well as fast reactors, and critical as well as sub-critical systems are proposed. Because the fuel is not recycled, the burning of the actinides is not as effective as in the case of multiple recycling. Burning of plutonium is nearly complete, most of the MAs, however, stay in the fuel and are not fissioned. It was argued that, because of the excellent long term retention capabilities of the TRISO fuel, this is acceptable.

The paper on the two-tiered approach presented the results of the studies of a strategy that departs from the assumption made in the US roadmap study, published about one year ago, that both the plutonium and the MAs will be burnt in the ATW systems. As a consequence the number of ATWs needed is rather large. In the strategy presented in the paper on the two-tiered approach it is assumed that the plutonium is separated from the MAs. While the MAs will go directly to the ATW, the plutonium will be inserted in LWRs. Special fertile-free plutonium fuel is proposed for this purpose. The burn up is very high. As a consequence, nearly all the fissile plutonium is being burned in the non-fertile fuel LWRs. The remaining spent fuel from the uranium fuelled LWRs together with the residual plutonium and MA from the spent non-fertile fuel are to be burnt in the ATWs. This two-tiered approach results in a significant reduction of the number of ATWs needed to support the cycle. In the discussion doubts were expressed, that the proposed mode of the plutonium irradiation in LWRs is state of the art, as it was claimed in the presentation. Both the fertile-free fuel and the high burn up were mentioned in this respect. Furthermore, it was emphasized that there are also other candidates, in which the plutonium could be burnt efficiently, e.g. gas cooled reactors. In general, however, there was agreement, that this strategy, namely to burn part of the plutonium in existing reactors, seems reasonable.

Economics aspects of transmutation systems were discussed in the last paper. It became evident that fuel cycle costs of these systems are higher than those of existing LWRs. Increase in burn up and other means could improve the situation. Whether fuel cycle cost similar to those of existing LWRs can be reached seems doubtful.

2.3. Session 3: Fuel and fuel cycle

The fuel and fuel cycle session heard four papers (two from the US, and presented by a Czech representative on work performed in the USA in collaboration with the Massachusetts Institute of Technology, and one from Japan).

The first paper in the session presented the current US plans for developing a fuel for the US ATW concept. In general, the functions and requirements for fuel for any of the systems under consideration by the US programme, in system trade studies, are similar to those for reactor fuels. Some specific requirements for operating conditions are now emerging from evaluation of the various system point designs; although those requirements will change as the concepts evolve, the operating requirements are similar to those for reactor systems for the

respective technologies. Of the many uranium-bearing fuel forms that have been utilized or studied in the past, the US has selected four forms that are favored for specific transmuter technologies. Of those, near term R&D work is concentrated to a metallic dispersion fuel and a nitride pellet fuel. Because there is essentially no experience with non-fertile compositions of the types proposed, R&D activities are necessarily directed toward assessing feasibility of the selected forms. Feasibility information must be obtained by 2005 in order to support ATW programme and technology decisions.

The second paper, presented by J. Laidler of ANL (USA) described the baseline technologies under development for the US ATW programme. The presentation opened with the comment that the US would necessarily re-consider current aspects of US non-proliferation policy if an ATW were actually to be deployed — particularly because chemical separations of TRU constituents from irradiated fuel would be necessary to accomplish transmutation. The overall process scheme for ATW fuel processing was described. Because the head end of the process must prepare TRU from a large quantity of spent LWR fuel, a high-through-put process is required. For this reason, a modification of established aqueous processes is proposed (termed “UREX”, because it is similar to the PUREX process but modified to produce a relatively pure uranium stream rather than a pure plutonium or TRU stream). The UREX process has been demonstrated with simulated LWR fuel, at laboratory scale, to show that a U product that meets low level waste standards can be produced. (This means that uranium can be disposed in the expensive manner of low level waste rather than as high level waste.) The associated pyrometallurgical processes for separation of TRU from fission products, employed at the head-end and for recycle of ATW fuel, have each been demonstrated with simulated fuel and shown to provide sufficient recovery of TRU from rare earth fission products. Technetium and iodine separation has yet not been demonstrated, however. It was noted that americium follows the TRU constituents in all steps of the process.

The third paper was presented by P. Hejzlar (from the Massachusetts Institute of Technology and representing the Czech Republic), and provided a parametric comparison of various fuel types in a once-through, heavy metal cooled reactor fuel cycle. P. Hejzlar noted that while M. Kazimi’s paper, presented in Session 2 (economics paper), addressed one extreme end of the fast reactor fuel cycle (a non-fertile actinide burner employing recycle), his own paper represented the other extreme. Because the analysis in M. Kazimi’s paper indicates that reprocessing is not currently economical for fast reactor systems, a once-through fuel cycle warrants consideration. The Hejzlar paper focused on comparative neutronics, radiotoxicity, and proliferation resistance for four fuel types: metal, carbide, nitride and oxide. Uranium only, uranium-plutonium, and uranium-thorium fuels were investigated. The metal fuels were shown to have the internal conversion ratios, indicating that those fuels would allow the longest core lifetime (based on reactivity considerations). An analysis of a model of a streaming fuel assembly fuelled with metal fuel indicated that it is possible to design a core with a small reactivity burn up swing and a negative void coefficient. The model for the streaming assemblies indicated that radiotoxicity and proliferation resistance of spent fuel are important when assessing comparative attributes of fuel forms that would be disposed in a geologic repository. For a given duty, it was found that the metallic uranium only fuel form (U-10Zr), in general, discharged the smallest amount of actinides to the repository, implying the lowest radiotoxicity, radioactivity, and heat load. However, the plutonium that would be converted in the metal fuel cores would have a significantly lower fraction of ^{240}Pu than the other fuel types, with a “plutonium vector” similar to that of weapons grade plutonium. Thus, the author noted that objectives for radiotoxicity reduction and reduction of proliferation threat are seemingly in conflict. It was further noted that the uranium-thorium-bearing fuels eliminated the concerns over generation of weapons grade plutonium, but added concerns

over generation of ^{233}U . The author also noted that the streaming assembly, metal fuel design that might allow a modular, exportable reactor design with a long core life would also provide a good source of weapons-grade plutonium.

The final paper of the session, presented by M. Saito of the Tokyo Institute of Technology, was an assessment of the radiotoxicity concerns associated with long lived spallation products from various target materials. A review of the spallation product yield curves indicated that one group of spallation products includes rare earth isotopes, some of which are alpha emitters of intermediate half life (i.e. 10^6 to 10^8 years). Those isotopes are likely to have maximal radiotoxicity effect, because the alpha emitters induce the most damage in human tissues when ingested, and because some of those rare earth elements would have longer in-body residence times. Yet, it is for that range of atomic numbers for which spallation yield codes are the least accurate — indicating that the amounts of these spallation products present in a target system are probably underestimated. Calculations of equilibrium radiotoxicity indicated that tin targets have lower toxicity than lead targets, which have lower toxicity than tungsten targets. When compared to fission product radiotoxicity, the spallation product radiotoxicity is only slightly lower than for fission products. Incorporation of further transmutation effects reduces both radiotoxicity values, and increases the discrepancy between the two types of species. However, it was noted that the spallation products tend to have higher melting temperatures, and would be more likely to deposit in cooler areas of target cooling loops outside of the active target area — where the nuclides would be less likely to be transmuted. Calculations indicated that radiotoxicity of the alpha-emitting spallation products is similar to that of the ^{210}Po activation product after about 2 years after irradiation. The means available to reduce spallation product radiotoxicity include selection of beam energy, beam type, and target type. The author noted that this work is only preliminary and would be improved by additional analysis. One participant advised that the work is an indication of the importance of considering the radiotoxicity of all activation and spallation products that might be present in an accelerator driven transmutation system — particularly for species that have not been typically part of the source term for previously-assessed reactor systems.

2.4. Session 4: Sub-critical blanket

The sub-critical blanket development session heard seven papers, underscoring the considerable efforts made with respect to all aspects of sub-critical blanket design. Thus, three papers (Y.H. Kim and W.S. Park, C.H.M. Broeders, and S. Taczanowsky and M. Kopec) presented results of neutronics studies (including methods validation); two papers (F.E. Dunn, and B.W. Spencer and J.J. Sienicki) dealt with thermal hydraulics and thermal mechanical issues; and two papers (W. Eriksson et al., and W. Maschek et al.) discussed safety related parameters of sub-critical blanket designs.

The paper by Y.H. Kim and W.S. Park performed a study of the sensitivity of the neutron source efficiency to the main core design characteristics. It was found that the source efficiency is sensitive to the target design (e.g. target designs with a small proton tube result in a source efficiency decrease, while a small buffer zone around the target increases the neutron source efficiency). The study also quantifies the sensitivity of the neutron source efficiency to the number and distribution of parasitic absorptions in the sub-critical blanket (e.g. fixed absorbers around the source can decrease its efficiency significantly).

The paper by C.H.M. Broeders described the methods development and validation work performed at FZK for ADS neutronics analysis. FZK has established a method based on the coupling of spallation codes with deterministic transport codes (KAPROS/KARBUS code

system). Presently, for transient analyses, the coupling of SAS4A with KAPROS/KARBUS and CITATION/VARIANT-K is under way. FZK is using also the code system LCS (combining the high energy code LAHET with the neutron transport code MCNP4), as well as the code MCNPX (a code that combines directly LAHET and MCNP without any interface utilities or files). Validation work for these tools was made on the basis of both OECD/NEA and IAEA benchmarks.

The study presented in the paper by S. Taczanowsky and M. Kopec was also based on MCNP4. The authors' intention was to contribute to a better understanding of the sub-critical neutron multiplication process and its simulation by the code. This increased understanding is essential for a more targeted definition of validation efforts (both experimental and analytical). One of the effects investigated for various targets was the dependence from neutron generation number i (or time t) of the generation multiplication factor k_i , the number of neutrons, as well as the cumulative energy release. Further, the authors studied the time evolution of the neutron flux and fission rate distributions after a 14 MeV δ -shaped neutron pulse. The results obtained for the relation of the neutron field decay trajectories vs. time and detector position can be explained from physical principles. However, discrepancies observed between the following parameters calculated by MCNP in the KCODE mode: neutron field decay slope, k_{eff} , and life span related to neutron fission (which should be the generation time) could not be fully understood and require further investigations.

The paper by F.E. Dunn presented the results of thermal hydraulics and structural analysis calculations performed on both sodium cooled and lead-bismuth cooled sub-critical blanket designs for which beam interruption transients were assumed. The calculations were performed for base case designs, as well as for designs implementing some mitigation options — either to reduce the strains on the components or to reduce the accelerator trip frequency (e.g. reducing core temperature rise through reduced power or increased coolant flow, reducing the thickness of the subassembly load pads located above core, and assuming two separate accelerators driving the spallation source). The author concludes on allowable beam interruption frequencies for current sodium cooled and lead-bismuth cooled blanket designs. Further, the paper quantifies the effect of design options that could be adopted to accommodate higher beam interruption frequencies.

The paper by B.W. Spencer and J.J. Sienicki* presents a sub-critical, lead-bismuth cooled concept with significantly enhanced natural circulation heat transport characteristics. The enhanced natural circulation is attained through the use of ejector (jet) pumps. The jet pumps are driven by two immersed axial flow coolant pumps. During normal operation, 25% of the core flow is assured by the two immersed axial flow coolant pumps, while 75% of the flow follows the natural convection path via the ejector pump diffusers. With a loss of pumping power and simultaneous failure to scram, the core temperature rise would increase to $\sim 390^\circ\text{C}$ (from 140°C during normal operation), and the peak cladding temperature would reach $\sim 850^\circ\text{C}$ (from 565°C under normal operation). The reactor power would passively shut down owing to negative reactivity feedback, and the heat would be rejected passively via the vessel external cooling to air.

The paper by M. Eriksson et al. discusses safety features of a heavy metal cooled accelerator driven waste burning system. The sub-critical blanket is fuelled with plutonium and minor actinides contained in a zirconium-nitride matrix. Transients started by a step increase in the neutron source and by the loss of coolant flow are analyzed with the help of the SAS4A code.

* The authors did not submit the full paper.

Special attention is paid to the maximum permissible temperature in the nitride fuel with regard to thermal dissociation mechanisms. In general, the results show that the system copes with protected and unprotected loss of flow scenarios. Peak temperatures in fuel, cladding, and coolant are kept within postulated design criteria. The benign behavior is attributed to large pin pitches that allow for large coolant volume fractions, the use of lead/bismuth as thermal bond between fuel and cladding, and to a primary system designed to maintain large amounts of natural circulation flow. It is found that the system is most sensitive to source transients. An increase in source intensity results in high fuel temperatures and this may jeopardize the stability of the nitride fuel. Safety consequences of source transients depend strongly on the particular assumptions involved in the analysis. Above all, the rate and magnitude of the source disturbance and the effectiveness of the heat removal systems may significantly alter the accident scenario. AN important conclusion is the recognition of the importance for early detection and successful termination of source transients. Finally, the authors discuss the technique used in accelerators to perform emergency shut down of the proton beam, and point out the key features that distinguish accelerator beam shut off from the shut down mechanisms used in critical reactors.

The last paper in the session (by W. Maschek et al.) also discussed the safety features of ADS. The fuel retained by these authors in their study is pure transuranics oxide or nitride fuel (“dedicated fuel” without fertile isotopes). These fuels are still to be developed, and programmes are under way to fabricate, characterize, and investigate these innovative fuels. Analyses show that the use of dedicated fuels for a reactor core leads to a strong deterioration of the safety parameters (e.g. the void worth, the Doppler effect or the kinetics parameters, like the neutron generation time, and β_{eff}). In addition, these cores may contain multiple “critical” fuel masses, resulting in a considerable re-criticality potential. Besides the neutronics aspects, these fertile free fuels may also suffer from deteriorated thermal or thermal-hydraulic features, as reduced melting points, reduced thermal conductivities, or even thermal instabilities. On the basis of the current knowledge on these dedicated fuels it is generally concluded that “critical” reactors may not be feasible, because of the aforementioned safety concerns. However, it is claimed that ADSs, due to the sub-criticality of the system, could accommodate such dedicated fuels. These claims have been supported by quite a number of preliminary ADS safety analyses performed in recent years giving some insight into the system behavior of an ADS and generally showing its promising safety features. However, in most of these studies, the ADS mostly used $^{233}\text{U}/^{232}\text{Th}$ fuel, thus relying on the overall good safety parameters of fertile isotopes containing fuel. The results presented in this paper, obtained for fertile-free dedicated transmutation fuel, showed possible safety problems, especially in the case of severe transients. The results are compared with those obtained for $^{233}\text{U}/^{232}\text{Th}$ fuelled ADS, and safety requirements and conditions are formulated which should be fulfilled by cores (critical or sub-critical) fuelled with these innovative dedicated transmutation fuels.

2.5. Session 5: Target developments

The target development session heard three papers.

The first one, entitled MYRRHA: Design of A Windowless Spallation Target for a Prototype ADS, presented by H. Aït Abderrahim (SCK•CEN), addresses the design evolution and simulation studies for the windowless spallation target under development for MYRRHA. The windowless approach is taken because “when you have a feature that you know is going to fail (window), you might as well get rid of it at the start.” The approach is an external lead-bismuth loop with a delivery manifold that fills the target tube with the lead-bismuth,

generating a free surface via a hydraulic jump. The beam strikes this free surface. The first design was shown to be too large for practical application in MYRRHA. The current design is 120 mm diameter. Issues included:

- Elimination of vortex which was unstable via anti-swirl member;
- Elimination of recirculation to minimize hotspot temperature;
- Effect of beam interruption;
- Issue of vapor formation and disposition in beam tube vacuum (where tests with lead-bismuth are planned to address this).

One conclusion was that, while computational fluid dynamics (CFD) codes were valuable tools, they need improvement for the windowless target application:

- Free surface tracking/diffusion;
- Combined with heat transfer;
- Turbulence.

The second paper Analysis of Different Design Options for the Beam Target of the Energy Amplifier Demonstrator Facility was presented by S. Buono. Both window target and windowless target options are being examined for the energy amplifier demonstration facility. The window target module has been developed internally; it is 7 m tall, with lead-bismuth target material cooled by natural circulation. The heat exchanger at the top is bayonet type with oil secondary coolant. Beam power is deposited mainly in the coolant funnel beneath the window which measures 140 mm diameter by 300 mm long. Salient issues discussed in this paper are listed below:

- Stresses in the window were analyzed for beam interruptions and were found to be very high, suggesting a rupture failure mode rather than cracking (leakage).
- The natural circulation cooling is said to handle both beam startup and interruptions by introduction of a flow inverter.
- The windowless target is said to be preferred for scaleup to a full size transmuter with high accelerator current.
- Ansaldo has produced a windowless target design that is virtually interchangeable with the window design for the demo facility; it utilizes primary circuit lead-bismuth coolant as the target material.
- There is also a collaboration with the MYRRHA project on a windowless design using a flow of lead-bismuth perpendicular to the beam axis.

The issue of applicability of CFD codes for heavy liquid metal applications is being addressed through a Benchmark Working Group International Collaboration.

The third paper entitled Experimental and Numerical Studies on Thermal-Hydraulics of Spallation Targets was presented by G. Heusener. The Research Center Karlsruhe (FZK) is integrating its laboratory facilities and numerical codes to evaluate spallation target designs, including the ISTC-project #559 design (IPPE), MEGAPIE (PSI), and the ADS-target (FZK). The FZK lead laboratory (KALLA) consists of numerous static facilities plus three separate loops, the technology development loop, thermalhydraulic loop, and corrosion loop for testing of lead and lead-alloy coolants. Water facilities are also available for flow visualization studies. One result of the numerical analyses presented is a proposal for an improved ISTC-project #559 design that is less vulnerable to lateral beam offset.

2.6. Session 6: Experimental studies

The experimental studies session heard four papers.

The first by M. Hron* described a planned experimental programme to validate the design of a long lived radwaste transmutation system based on molten fluorides fuel.

The second paper by S.E. Chigrinov et al.* presented the “YALINA” experimental setup and the first results of the sub-critical neutronics studies performed at this facility. Experimental ADS validation work using high energy accelerators is expensive. On the other hand, studies have shown that the physics of the coupled target/sub-critical blanket system can be studied simulating the spallation source with the help of a 14 MeV high intensity neutron beam irradiating a lead target. YALINA is the combination of a thermal sub-critical array (10% enriched uranium-polyethylene) with a central neutron source (lead target irradiated by a high intensity (10^{12} ns^{-1}) D-D or D-T continuous or pulsed neutron generator). The author presented first experimental results (reactivity measurements, reaction rate traverses, dynamic characteristics).

The third paper by D.P. Abraham et al. presented the results of an experimental study in which a thermal convection-based test method was used to allow exposure of candidate structural materials (iron-based alloys) to molten lead-bismuth flowing under a temperature gradient. The gradient was deemed essential in evaluating the behavior of the test materials in that should preferential dissolution of components of the test material occur, one would expect dissolution in the hotter regions and deposition in the colder regions, thus promoting material transport. The temperature in the hot leg of the convection harp was $\sim 550^\circ\text{C}$ and the temperature in the cold leg was $\sim 350\text{--}425^\circ\text{C}$. The alloys chosen for the test were AISI S-5, a 2 wt% Si mild steel, and HT-9, a ferritic-martensitic stainless steel. The tests were terminated after sample exposure for ~ 4500 hours. The authors summarized the results of their studies as follows:

- The AISI S-5 alloy was attacked by the molten lead-bismuth. The depth of intergranular penetration ranged up to $40 \mu\text{m}$ in the hot leg and up to $5 \mu\text{m}$ in the cold leg. The test showed that AISI S-5 would not be suitable as a structural material for nuclear reactors using lead-based alloy coolants.
- The HT-9 samples were not attacked by the molten lead-bismuth. The samples appear to have been protected by a stable, adherent, FeCr_2O_4 -type spinel layer. Close control of oxygen activity was not necessary to form the protective spinel. As long as the oxygen activity is well below that needed to form PbO and above that needed to form the spinel, ample leeway exists to form a protective layer on HT-9.
- A comparison of AISI S-5 and HT-9 corrosion behavior clearly demonstrates the important role of chromium in the passivation of HT-9 and EP-823 alloys exposed to molten lead-bismuth.

The section's last paper by X. He et al described a kinetic model developed to estimate the corrosion/precipitation rate in a non-isothermal liquid lead-bismuth eutectic flow loop. This model provides a useful tool for designing and operating liquid lead-bismuth eutectic cooling systems. The model is based on solving the mass transport equation with the assumptions that convective transport dominates in the longitudinal flow direction, and diffusion dominates in the transverse direction. The species concentration at wall is assumed to be determined either by the solubility of species in the liquid lead-bismuth eutectic in the absence of oxygen, or by

* The authors did not submit the full paper.

the reduction reaction of the protective oxide film when active oxygen control is applied. Analyses show that the corrosion/precipitation rate depends on the flow velocity, the species diffusion rate, the oxygen concentration in the liquid lead-bismuth eutectic, as well as the temperature distribution along a loop. Active oxygen control can significantly reduce the corrosion/precipitation of the structural materials. The authors show that the highest corrosion/precipitation does not necessarily locate at places with the highest/lowest temperature. For a material-testing loop being constructed at the Los Alamos National Laboratory, the highest corrosion occurs at the end of the heater zone, while the highest precipitation occurs in the return flow in the recuperator.

3. OVERALL SYNTHESIS, CONCLUSIONS AND RECOMMENDATIONS

High level waste disposal is an element of paramount importance in the discussion of nuclear power generation sustainability. This, and the desire to reduce the quantity of long lived waste material, have stimulated new interest in the transmutation of actinides and some long lived fission products, and in emerging system technologies for energy production with reduced actinide generation. One such system is the combination of a particle accelerator with a subcritical nuclear reactor; another possibility is to reduce the generation of actinides by the introduction of the thorium fuel cycle.

The surmised advantages of accelerator driven systems — apart from their intrinsic low production of long lived radioactive waste, and transmutation capability — are also enhanced safety characteristics and better long term resources utilization (e.g. in connection with thorium fuels). Important R&D programmes are being undertaken by various institutions in many Member States to substantiate these claims and advance the basic knowledge in this innovative area of nuclear energy development.

While long term objectives for developing novel nuclear systems for energy production and transmutation may not be unanimously agreed upon by the different groups participating in this effort, it is clear that the short term goals are similar.

Therefore, quite a few generic R&D areas can be identified that would benefit from international collaboration. The most important technical issues identified and discussed within the framework of the TCM's six sessions were:

- Thermal fatigue (due to beam trips),
- Toxicity of the spallation products,
- The lack of a safety strategy for severe accidents with fertile-free transuranics fuel,
- The lack of data on irradiation damage effects (induced by both proton and neutron irradiation) on the structural properties of the beam window and the adjacent core.

The TCM identified the need and opportunity for collaboration in the following areas:

- Major demonstration facilities, for which international participation should be considered,
- Testing of special effects (e.g. fuels and materials tests, and zero power coupled systems) which offers practical opportunities for dividing up the work, and
- Calculational benchmarks.

Based on the discussions during the TCM, international collaboration was noted several times as being of potential significant value to the various Member States programmes. Potential areas for enhancing collaboration included use of international organizations for coordinating benchmark verification of design analysis codes in areas such as thermal hydraulics and target physics, and for pursuit of establishing criteria for safety related parameters (e.g. sub-

criticality level, fuel composition, reactivity control), and evaluating the design impact of these criteria.

The TCM participants acknowledged the IAEA's role as promoter and facilitator of information exchange and collaborative R&D and encouraged the IAEA to continue using its good offices to promote multi-national collaboration and exchanges. Given the high benefit/cost ratio of data sharing, substantial benefits for the Member States can be achieved through increased IAEA efforts to facilitate the sharing of data. Increased use of dynamic Web-based collaboration and information exchange is considered to be an important means to achieve this goal. It was clearly stated that all the activities in the area of nuclear energy systems for energy generation and transmutation should be tied to the activities of the Technical Working Group on Fast Reactors (TWG-FR).