SUMMARY

1. INTRODUCTION

The Advisory Group Meeting (AGM) on "Design and Performance of Reactor and Sub-critical Blanket Systems with Lead and Lead-Bismuth as Coolant and/or Target Material" was held in Moscow, Russian Federation from 23-27 October 2000. This AGM was convened by the International Atomic Energy Agency (IAEA) and was hosted by the Research and Development Institute of Power Engineering (RDIPE) in Moscow, in cooperation with the Institute of Physics and Power Engineering (IPPE) in Obninsk, and Minatom. Its purpose was to provide a forum for international information exchange on all the topics relevant to lead (Pb) and lead-bismuth (Pb-Bi) cooled critical and sub-critical reactors. In addition, the AGM aimed at: (1) finding ways and means to improve international co-operation efforts in this area; (2) obtaining advice from the Member States with regard to the activities to be implemented in this area by the IAEA, in order to best meet their needs; and (3) laying out the plans for an effective co-ordination and support of the R&D activities in this area.

Attendance at the AGM included sixteen participants from eight countries (Belgium, China, India, Italy, Japan, Kazakhstan, the Republic of Korea, the Russian Federation) and one international organization (IAEA). Presentations and/or statements with regard to the status of the respective R&D programmes were made by all participants.

An important result achieved at the AGM was extended information exchange on the recent R&D developments in the following fields: critical and sub-critical concepts, coolant properties, experimental and analytical validation work. The fast reactor concept BREST-OD-300 (Russia), as well as various conceptual designs of heavy liquid metal cooled fast reactors pursued in Japan were presented. Research and development work on hybrid (accelerator driven) sub-critical systems were presented by the Republic of Korea (HYPER), India (fast-thermal hybrid accelerator driven sub-critical system ADS), Italy (European "energy amplifier demonstration project"), Belgium (development of an ADS R&D experimental facility), China (study on ADS Pb/Pb-Bi spallation target physics), Sweden (investigations of void coefficients in sub-critical lead-bismuth cooled blankets), and Kazakhstan (basic nuclear data for ADS).

Important results were further achieved during the visit to the IPPE in Obninsk on 25 October 2000. An extensive scientific and technical exchange covering the following topics was held during this visit: spallation targets, sub-critical target/blanket systems, multi-purpose small fast power reactors with lead-bismuth cooling (modular design), lead-bismuth thermal-hydraulics, and steel corrosion resistance in lead-bismuth. Of particular interest were the visits in two IPPE laboratories: the experimental facilities for heavy liquid metal coolant corrosion tests, and the experimental facilities for heavy liquid metal coolant (HLMC) technology.

2. STATUS OF DESIGN DEVELOPMENT AND PERFORMANCE

2.1. Heavy metal cooled fast reactors

2.1.1. Japan

The conceptual design study on various types of HLMC reactor plant concepts was conducted in JNC. Through preliminary design study, JNC identified some technical features on each concept and roughly evaluated economical competitiveness based on the total weight of NSSS. Finally, Pb-Bi cooled medium tank type reactor was selected as a most promising concept. JNC insisted that lower melting point of Pb-Bi coolant makes it more attractive than Pb coolant and the amount of Bi resources seems not to be a significant issue based on roughly but conservative estimation. JNC is now performing design study on a medium sized and complete natural circulation cooled reactor with Pb-Bi coolant. This concept takes advantage of superior natural circulation characteristics of Pb-Bi and seeks simplification of NSSS and BOP designs by employing passive features.

2.1.2. Russia

The conceptual design of a fast reactor BREST with (U, Pu)N fuel and lead coolant has been developed such as to consistently implement the principles of natural safety without a sufficient deviation from materials and technology which were proven in defense and civil nuclear power facilities. Designs for reactors with the power in the range of 300 MWe (BREST-300) to 1200 MWe (BREST-1200) were optimised and proved to meet safety and non-proliferation requirements. The use of lead coolant in the BREST reactor designs allows the following design features: single vessel or pool type arrangement without a metal vessel, two circuits in the main and emergency cooling systems; no special system to wash off coolant from fuel assemblies during refuelling; improved passive control and protection features with threshold response; simpler design of steam generators; and simpler rooms in the cooling circuits and other nuclear power plant constructions.

Neutronic experiments were performed at U-Pu-Pb critical assemblies to validate reactor physics and revise the nuclear data. Experimental work on lead coolant technology and in-pile testing of nitride fuel to study the corrosion of nitride fuel and structural materials in lead have been conducted and are being conducted. The BREST-300 engineering design is now in progress. Specific projects on R&D of closed nuclear fuel cycle technology, and development and construction of a demonstration power plant as well, have been suggested at the international collaboration level.

Mononitride (U, Pu)N fuel was chosen for BREST-300 due to favorable properties such as its high actinide density and high thermal conductivity. The use of lead bonding allows operating under so called cold fuel conditions, i.e., with maximum temperatures below one third of the melting point. For the BREST design, a moderate average burnup of 6% average is foreseen. Two fabrication processes - carbothermic reduction of oxides and direct nitriding of metals, were developed at the Bochvar Institute (VNIINM). Oxygen and carbon impurities resulting from the carbothermic reduction method could be surpressed down to 0.2% weight fraction. The metal nitriding method is still preferred due to the possibility of obtaining pellets with densities above 90% after sintering. The lead bonding concept was validated by heating lead filled fuel pins up to 800^{o}C inner cladding temperatures for 2000 hours. No chemical reaction between lead and cladding was observed.

The irradiation experience of nitride fuels in Russia encompasses 15 years of operation of a UN core in BR-10. For (U, Pu)N, the highest burnup reached was 6%. A test of the high temperature performance was made by irradiating He bonded (U, Pu)N at 1045 W/cm up to sign of liquid metal or nitrogen gas formation was seen in subsequent destructive examinations. (Pu, Zr)N fuel was also fabricated and irradiated successfully up to 8% burnup in BOR-60. On-site pyroprecessing of spent BREST fuel is foreseen. KCl and LiCl were used for testing the performance on UN, 99% recovery efficiency was achieved, even though no recovery from the electrolyte was performed. One still has to decide whether to directly recover U and Pu metal from the cathode or to use a liquid process for the purpose. The issue of PuN decomposition was raised in the discussion, but it was assured that no observation of metal or gas formation in Russian nitride irradiation had been made. It appears that the moderate burnup projected for the BREST fuel justifies the high pellet density (as compared to the west/European recommendation of 85% TD), but laboratory tests of the high temperature stability of the BREST fuel would be required to ensure that the fuel performs as expected under accident conditions.

The sodium cooled loop type reactor BOR-60 has been designed to accommodate one experimental channel in the fifth row of its fuel core for testing BREST fuel. The experimental loop consists of a mockup of the fuel assembly containing 4 complete fuel rods. The fuel assembly is cooled independently from the reactor core primary system by a flow of liquid lead. The lead is pumped, and is cooled by the sodium that flows inside the reactor through a double wall arrangement. The oxygen content is constantly monitored and controlled, as well as the eventual release of fission gases (Xe and Kr) revealing a pin failure. The auxiliary systems have been already mounted and fully tested while the loop will be installed in the reactor in January 2001 after delivering of the structural materials to be tested in it. The experiment is planned to be run for 2 years reaching 4% burnup in the uranium-plutonium nitrate fuel with an average spectrum of 234 keV and flux of 10^{15} n/cm^2 s. For comparison, the maximum and average discharge fuel burnup foreseen in the BREST-OD-300 reactor will be 10% and 6%, respectively. After this two-year experiment no other plans have been made for the BOR-60 reactor experimental channel and it could be in principle available to the international community for a new experiment. Many laboratories have already showed a strong interest in this loop. The lead loop will not be available because it will be completely dismantled to be analyzed after the irradiation.

The small fast reactor SVBR-75/100 design was developed on the basis of reactor operation experience obtained from Pb-Bi cooled nuclear submarines. It was shown that such a reactor concept that meets safety and nonproliferation requirements, could have multi-purpose uses, e.g., as nuclear power desalination plants in developing countries, and as part of large modular nuclear power plants in developed countries.

2.2. Accelerator Driven Sub-critical System

2.2.1. Belgium

The ADS facility was proposed in the MYRRHA project by SCK•CEN. The nuclear waste, and particularly the minor actinides and long-lived fission products, are one of the issues affecting the acceptance by the public of nuclear energy. Because it is controlled by an external source, the ADS is seen as a promising tool towards the transmutation of those materials. MYRRHA is intended to be a first step towards an ADS-DEMO facility. The MYRRHA concept aims at providing an irradiation facility for ADS related research: structural materials, fuel, liquid metals, sub-critical reactor physics, and subsequently on applications such as the transmutation of waste. The main originality of the MYRRHA concept resides in its windowless spallation source. The objective of a fast neutron flux (~ 10^{15} n/cm^2 s) on a low power machine (30 MW) leads to a very compact core and spallation source. The project is presently in a pre-design phase. The purpose in this phase is to identify the main technical issues, to present a machine configuration and provide realistic cost estimates for the construction. International cooperation has already been set up on particular topics and it is the will of SCK•CEN to further broaden this cooperation in the future. The pre-design phase is expected to be completed in spring 2001. The detail design, engineering, safety studies, construction and licensing will start in 2008 as final objective for the first operation.

2.2.2. India

India considers the Pb-Bi coolant as an essential part of the ADS design. The fluid dynamics and heat transfer characteristics of Pb-Bi are being investigated now. Two types of cooling methods are considered - natural convection and other by gas injection. India also has interest in benchmarking computational studies in heavy metal coolant systems which have been done so far. For this objective, some experiments have been planned with heavy metal coolant loops with mercury in the near future and with Pb-Bi in the years to come. In this program, the international collaboration is invited. The area of priority will be to gain experience and data on properties of structural steels which are compatible with heavy metal coolants. This is required for the construction of system equipment and the running of heavy metal coolant process systems. India is inclined to join such activities on partnership basis.

2.2.3. Italy

The Italian ADS program is divided into two subprograms, which strongly interact one with each other and are based on the C. Rubbia design of an Energy Amplifier. The first is a fundamental R&D program named TRASCO (from the Italian acronym of Waste Transmutation) and the second is an industrial program related to system design. The TRASCO program is further divided in two parts: the first part, lead by INFN, is devoted to the development of a 1 GeV, 30 MeV proton linac accelerator based on RFQ, DTL, and super-conducting cavities technology. The second part is devoted to the sub-critical system. In this framework the CHEOPE and LECOR facilities were built at ENEA’s Brasimone site. They will be used for testing the basic technology related to Pb-Bi coolant (loading/unloading operations, oxygen control, instrumentation, and filtering) and the compatibility of structural materials. Experiments are also planned for the MEGAPIE and the MYRRHA projects. The TRASCO program has been already extended, and ENEA is building the first large Pb-Bi pool facility in the world, CIRCE, that is currently being installed in the old PEC reactor facility of Brasimone and will be put into operation in the second half of 2001. It consists of a 9 m high and 1.2 m large pool filled with 100 t of Pb-Bi where a series of tests will be done devoted not only to test basic phenomenology (such as gas injection for natural circulation enhancement, oxygen monitoring and control in pool conditions), but also component mock-ups of the demonstration facility proposed by the Italian community. The use of the PEC facility will be also available for international collaborations. The system design studies which are performed by Ansaldo, CRS4, ENEA and INFN, aim at completing a pre-engineering design of a 80 MW Pb-Bi ADS demonstration facility which will be the Italian proposal for a European demonstrator to be built in the framework of a European collaboration.

2.2.4. Kazakhstan

The Institute of Nuclear Physics (INP) in Republic of Kazakhstan has been performing some research for the multi-purpose ADS as well as for fast reactors in terms of nuclear data and material science. Some new material data on physical and mechanical properties of irradiated structural materials of BN-350 have been obtained. New approaches and models applicable to the calculation of spallation residues and fission fragments have been developed.

2.2.5. *Japan*

The ADS core design performed by JAERI was analyzed in terms of void worths as function of pin pitch. It was shown that Pb-Bi coolant always yields lower void worths, with a transmutation from positive to negative worths for pin pitches exceeding 1.5 times pin diameters. It was emphasized that Monte Carlo methods need to be used in order to obtain correct void worth values. In the Sing Sing core design performed at RIT, boron carbide is introduced into the core in order to minimize neutron capture rates on the minor actinides. Hence helium production in the fuel can be surpressed, and reactivity losses are mitigated. The price paid is an increase in void worths that become positive even using Pb-Bi coolant and large pin pitches. The advantage of Pb-Bi over sodium is still clear though, allowing operating the core sub-criticality with a k-eigenvalue 3000 pcm higher for the Pb-Bi cooled version. There is a wellknown uncertainty in the inelastic cross sections evaluations of Pb void worths, and is as such very significant. Uncertainties concerning the minor actinides cross sections could also affect these evaluations. It was pointed out that the scenario of having local voids in the center of the core, induced by, e.g., the release of fission gas from fuel pins should be investigated.

2.2.6. *Republic of Korea*

A conceptual design of HYPER system and related R&Ds are being performed in KAERI, aiming at the transmutation of nuclear waste and the energy production. Some major design features of the Pb-Bi cooled HYPER system have been developed through the investigation of various transmutation system concepts in terms of the transmutation capability, safety and the proliferation resistance of related fuel cycles.

The core is designed to produce 1000 MW_{th} with reflector assemblies filled with liquid lead being located at the core perimeter. Burnable absorbers are designed to surpress a large reactivity swing resulting from a relatively small amount of fertile in the HYPER core. Either TRU-Zr alloy or (TRU-Zr)-Zr dispersion metallic fuel is under consideration, based on the pyroprocessing process. From the trade-off studies for fuel fabrication, a dispersion metallic fuel is being considered more favorable for the HYPER system in terms of high discharge burnup. TRU and fission products (FP) will be loaded separately for incineration. TRU is loaded as a fuel to drive the system and FPs are loaded as a target (FP assembly) to be irradiated in a special region of the reactor core. Long-lived FPs such as Tc-99 and I-129 will be incinerated using a localized thermal neutron flux obtained by inserting some moderators in the FP assembly. FP assembly (target) configurations are different for Tc and I, respectively.

A three loops system with Pb-Bi coolant is chosen for the preliminary design of HYPER cooling system. The Pb-Bi core coolant is also used as a spallation target. The beam target design is optimized for stable and safe operation of the target and for reasonable lifetime of the beam window. Beam diameter and window thickness are determined based on the simulation results.

There are few experimental data available for justifying the TRU-Zr metallic fuel for the HYPER system. In addition, the development of Pb-Bi coolant/target technologies is very challengeable in the HYPER system design. Experiments to solve the key technical issues related with TRU-Zr fuel and Pb-Bi coolant/target will be performed from 2001 to 2003 and the conceptual design of the HYPER system will be completed by 2006.

2.2.7. *Russia*

The test loop TS-1, the first target complex in the world, has been designed by IPPE and will be operated for testing a liquid metal Pb-Bi cooled target in the proton beam of the linear accelerator in the Los Alamos National Laboratory (LANL).

3. CONCLUSIONS

The AGM stressed that nuclear energy is a realistic solution to satisfy the energy demand, considering the limited resources of fossil fuel, its uneven distribution in the world and the impact of its use on the planet, and the expected doubling of the world population in the 21st century and tripling of the electricity demand (especially in the developing countries). However, the AGM concluded that the development of innovative nuclear technologies must be pursued meeting the following requirements: (a) deterministic exclusion of any severe accident; (b) proliferation resistance; (c) cost competitiveness with alternative energy sources; (d) sustainable fuel supply; and (e) solution of the radioactive waste management problem.

4. RECOMMENDATIONS

On the basis of the technical presentations and discussions, both at RDIPE and during the visit at IPPE, the AGM has identified the following areas of mutual interest:

- Corrosion resistance, embrittlement and radiation ageing of structural materials in lead, as well as in lead-bismuth coolants;
- Technology of advanced fuels, such as nitride plutonium-uranium fuel, including their irradiation testing in fast research reactors;
- Fuel reprocessing techniques that would support the non-proliferation regime and would be unsuitable for the production of weapons-grade materials;
- Radioactive waste treatment techniques, including transmutation of minor actinides and long-lived fission products in fast reactors and/or ADS;
- Nuclear data libraries, codes, and experimental validation studies.

Considering the importance of development work in the area of innovative nuclear technologies, the participants in the AGM suggested that wide bilateral and multilateral collaborations on the above mentioned issues would greatly benefit national efforts. It was stated that the best way to carry out these activities would be under IAEA co-ordination. At the same time, it was proposed that the ways and means of such collaborations could be determined by the participating Member States to best meet their specific interests.

More specifically, the following R&D topics that would benefit from an international collaboration under IAEA aegis were identified:

1. Updating of the heavy metal nuclear data [a co-ordinated research project (CRP) could be initiated];
2. Fast reactor and ADS fuels and material development;
3. CRP on computational fluid dynamics methods and computer code validation;
4. Nitride fuel experiments in BOR-60 [a CRP could be initiated];
5. Information exchange on the operational experiences of heavy liquid metal coolant (HLMC) experimental facilities as well as reactors, and discuss the creation of a reliability database for HLMC components.