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WORKING MATERIAL

Technical Meeting on
Liquid Metal Reactor Concepts: Core Design and Structural Materials

Conducted within the framework of IAEA Nuclear Energy Department's

Technical Working Group on Fast Reactors (TWG-FR)

IAEA HEADQUARTERS, VIENNA

12 – 14 June 2013

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Building M, Room M0E100

MEETING REPORT

1. Background information

In order to guarantee the long term sustainability of nuclear power, innovative fast neutron systems operated in closed fuel cycles have to be developed. These systems have the potentiality to fully exploit the energy potential of natural resources (Uranium and Thorium), as well as to transmute the transuranic elements which are responsible for the highest heat load and radiotoxicity of the long term nuclear wastes. Fast neutron systems will play therefore an increasingly important role in the future, and help to ensure that nuclear energy remains a sustainable long-term option in the world's overall energy mix.

In recognition of the fast reactors' importance for the sustainability of the nuclear option, currently there is worldwide renewed interest in fast reactors technology development, as indicated, e.g., by the outcomes of recent scenario studies of the International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) and of the Generation IV International Forum (GIF) technology review, which concluded with 4 out of 6 innovative systems to be fast reactors, with special emphasis on liquid metal cooled fast reactors (LMFR), i.e.: sodium cooled fast reactor (SFR) and heavy liquid metal cooled fast reactor (LFR). Currently, LMFR construction projects are on-going in India (PFBR) and Russian Federation (BN-800), while in China the first experimental sodium-cooled fast reactor (CEFR) has been recently connected to the grid. Innovative SFR and LFR concepts are under development also in China, France, European Union, India, Japan, Republic of Korea, Russian Federation and USA.

Within these programmes, significant R&D effort has been devoting to the design of innovative reactor cores characterized by intrinsic safety features (enhanced negative reactivity feedbacks, reduced coolant void reactivity effects, etc.), high performances (in terms of cycles length, high fuel burnup, breeding gain, etc.), and minor actinides (MA) transmutation capability.

In this framework, the development, characterization, testing and qualification of high performance in-core structural materials represents one of the most challenging aspect faced by the development of innovative LMFRs, generally characterized by severe operating conditions (high neutron flux, liquid metal coolant, high temperatures, etc.). In this area, current R&D activities are aimed at the development of structural materials that feature high resistance to swelling, irradiation creep, and irradiation embrittlement, which represent major issues in the choice of materials for the reactor core components.

The International Atomic Energy Agency (IAEA), within the framework of its Nuclear Energy Department's *Technical Working Group on Fast Reactors* (*TWG-FR*), encourages technical cooperation and promotes collaborative R&D projects among Member States with fast reactors programmes, with the general aim of catalyzing and accelerating technology advance in this field. The regular organization of topical technical meetings, workshops/seminars and international conferences is a significant part of this effort.

This topical Technical Meeting (TM) on "Liquid metal reactor concept: core design and structural materials" addresses Member States' expressed need of information exchange on innovative LMFR core designs and related structural materials development and qualification.

2. Objectives of the meeting

The objective of the TM on "Liquid metal reactor concept: core design and structural materials" was to present and discuss innovative liquid metal fast reactor (LMFR) core designs with special focus on the choice, development, testing and qualification of advanced reactor core structural materials. Main results arising from national and international R&D programmes and projects in the field were reviewed, and new activities to be carried out under the IAEA aegis were identified on the basis of the analysis of current research and technology gaps.

3. Summary Report

3.1 Opening Session

The Scientific Secretary of the technical meeting Mr. S. Monti welcomed the meeting participants and thanked them for the participation. The list of participants is reported in Annex I.

After a brief self-introduction, meeting participants agreed to appoint Mr. K. Tucek from the European Commission as meeting Chairman and approved the meeting Agenda, which is reported in Annex II.

Recent IAEA Achievements in the Field of Fast Neutron Systems and Scope and Objectives of the Meeting

Mr. Monti provided the meeting participants with an overview of context, scope and main objectives of the technical meeting.

The first part of his presentation focused on the IAEA activities on FR and ADS technology and on the recent achievements in the field. In particular, the status of recently completed, on-going and planned CRPs was summarized. The recent initiatives on FR safety like the joint IAEA-GIF workshop on SFR safety design criteria were mentioned during the presentation.

Mr. Monti also summarized the statistics and main highlights of the *International Conference on Fast Reactors and Related Fuels Cycles – FR13*, held in March 2013 in Paris, with special emphasis on the main achievements in the area of fast reactor materials development presented at the conference.

Finally, Mr. Monti presented the IAEA activities on nuclear knowledge management, in particular the FR-KOS system developed by the NKM section. The overall picture of the IAEA activities was completed with a presentation of the recent IAEA publications "*Status of Fast Reactor Research and Technology Development*", the IAEA Nuclear Energy Series "*Liquid Metal Coolants for Fast Reactors Cooled by Sodium, Lead and Lead-Bismuth Eutectic*" and the new booklet "*Status of Innovative FR Designs*", which is currently under preparation.

3.2 Presentations by Member States Representatives

Activities conducted in the domain of fast reactors, with a specific focus on core design and material related topics, have been presented and discussed by experts from Belgium, China, Germany, India, Italy, Republic of Korea, Netherlands, Sweden - as well as IAEA and EC.

Presentations focused on energy policy strategies, role of liquid metal cooled fast reactors, and on technical topics of fuel and core design, structural materials, as well as related safety and economic aspects.

Liquid metal reactor concepts considered in Europe and China employ MOX fuels, while metallic fuels are considered for future fast reactor designs in India and Republic of Korea, the latter concept also involving Pu and MA recycling. The ELECTRA training reactor concept will employ (Pu,Zr)N fuel. In China, R&D on fusion-fission hybrid reactors is also on-going.

Increased emphasis is being given to the refinement and improvement of suitable computational modelling tools and their qualification domains, as well as on material databases to enhance the confidence in the safety demonstration of innovative reactor designs. This involves the improvement of the fidelity for assessment of safety margins to fuel and component damage as well to impairment of functionality or damage to systems important to safety. Further efforts have been also reinitiated to improve the knowledge about the phenomenology and possible consequences of hypothetical severe accidents, which is still partly incomplete or missing. In this respect, the attention is specifically dedicated to the integrity assessment of materials forming confinement barriers (such as the fuel pin cladding), which are subject to design specific environmental conditions (irradiation damage, temperatures, physico-chemical interactions) and mechanical loads. In support to the development of risk-informed approach to the design of next generation systems, critical data gaps - underlying uncertainties and their impact on assessment results - need to be also comprehensively identified and evaluated.

As for the fuel pin behaviour, key issues are the understanding of the prevalent phenomena (FCMI, FCCI, JOG formation, embrittlement) in relevant design and environmental conditions. Combinations of different environmental effects and loads (e.g. liquid metal assisted cracking, creep-fatigue, stress-corrosion cracking, etc.) need to be particularly considered. To support the safety demonstration, dedicated facilities need to be set-up, fuel & material irradiations (including fuel pins with coating materials) need to be performed in representative conditions, possibly also including accidental conditions. In particular, as high burn-up fuels are considered for economy reasons, this calls for specific design challenges with respect to fuel and core design, as well as for safety demonstration. Adequate modifications of technologies at back end of fuel cycle, reprocessing and waste management might also be required.

The issue of scalability of experimental results obtained at specific conditions to real plant or design conditions needs to be considered (cf. clad swelling-to-dose-rate dependence). In general, the availability of irradiation capacities might be a bottleneck for timely qualification of fuels and materials.

As performance assessment of mitigation measures in Level 4 of the Defence-in-Depth might be required to form a part of the safety demonstration irrespective of how robust the design or provisions for prevention are, the efficiency of severe accident mitigation measures and required safety systems, particularly those ensuring the fulfilment of fundamental safety functions, might need to be further demonstrated in the prototypical conditions.

A summary of the specific contributions from representatives of the Members States is reported here below.

Belgium
Mr. Schuurmans from Belgium discussed several aspects of structural materials for the MYRRHA reactor core. MYRRHA is a flexible fast spectrum irradiation facility able to operate in sub-critical and critical mode for material and fuel developments for GEN IV and fusion reactors, as well as for radioisotopes production. The presentation focused on the role of materials research for the development of the MYRRHA project, including materials selection for the principal components (reactor vessel, core support structures, wrappers, cladding and target window), materials properties evaluation and R&D programme which includes identification of materials issues, assessment of materials properties and development of testing infrastructure. Several considerations and main results of experimental investigations on key aspects such as liquid metal corrosion, liquid metal embrittlement and irradiation effects were presented.

China
Chinese Academy of Sciences (CAS) launched an engineering project to develop ADS system and lead-based reactors named China LEAd-based Reactor (CLEAR) series. The Institute of Nuclear Energy Safety Technology (INEST) will be responsible for the CLEAR design and R&D. In this programme, CAS plans to develop the lead-based reactors through 3 phases which are 10MW(th) lead-bismuth cooled research reactor (CLEAR-I) to be built in 2010s, 100MW(th) lead-based experimental reactor (CLEAR-II) to be built in 2020s, 1000MW(th) lead-based demonstration reactor (CLEAR-III) to be built in 2030s. As a pre-testing facility, a lead-based zero power reactor (CLEAR-0) is required to be built before CLEAR-I construction and operation. The lead-based reactor new conceptual design including hydrogen production, tritium production for fusion energy and thorium utilization is also on-going. Lead-lithium cooled fusion reactor blanket design and lead-lithium experimental loops were developed more than 10 years.

CLEAR series reactor conceptual design has been finished and detailed engineering design for CLEAR-I is underway. The R&D activities for CLEAR reactor including design and safety software, key components, structural materials, lead-based experimental loops and neutronics experimental platform are developing. Series of liquid lead-based experimental loops named DRAGON (Lead-Lithium) and KYLIN (Lead-Bismuth) have already been built or on constructing to performed experiments investigating the structure material corrosion issues and the thermal-hydraulic properties of lead-based coolant. The Highly Intensified D-T Neutron Generator HINEG for neutron experiment and software validation will be constructed. Series advanced reactor design software and nuclear library have been developed for lead-alloy cooled reactor, including CAD-based Multi-Functional 4D Neutronics Simulation System (VisualBus), Monte Carlo Automatic Modeling Program for Radiation Transport Simulation (MCAM), Super Monte Carlo Simulation Program (SuperMC), Nuclear Radiation Virtual Simulation and Assessment Program (RVIS), Neutronics and Thernal-hydraulics Coupling Safety analysis Program (NTC) and Hybrid Evaluated Nuclear Data Library (HENDL). China Low Activation Martensitic (CLAM) steel is being developed and tested in lead-based coolant and neutron irradiation environment are in progress to validate the probability using for CLEAR reactor. These experiment devices and the software will support CLEAR design and construction.

In this contribution, the CLEAR series reactor design and R&D activities including development of design and safety analysis software, structural materials, lead-based experiment loops, key components and high intensified neutron source were presented.

Germany

Due to the French project ASTRID, the European CP-ESFR project, and the MYRRHA/FASTEF project, the research work on fast reactors has got a new push in Europe. Following this trend, the Institute of Resource Ecology at the Helmholtz-Zentrum Dresden-Rossendorf decided to start several studies dedicated to fast reactor technology, among them the extension to FR of the well validated LWR core simulator DYN3D. Three of these studies were presented and discussed:

I. Modeling of SFR cores using the Serpent-DYN3D code sequence

DYN3D is an advanced multi-group nodal diffusion code originally developed for 3D steady-state, burnup, and transient analysis of Light Water Reactor (LWR) systems. Currently, the DYN3D code is being extended for the analysis of Sodium-cooled Fast Reactor (SFR) cores. Thermo-physical properties of sodium were included into the thermal-hydraulics module database. The development of thermo-mechanical model is planned for the near future. The main objective of this study is to demonstrate the feasibility of using DYN3D for the modeling of SFR cores. For this purpose, a 2D model of the mixed oxide fuel (MOX) European Sodium Fast Reactor (ESFR) core was considered. 2D full core depletion calculations were performed using Serpent-DYN3D code sequence and the results were compared with those obtained from a 2D full core Monte Carlo solution. Very good agreement between the codes was observed for the core integral parameters and power distribution at beginning of life (BOL) and end of life (EOL).

II. Core shielding assessment for the design of FASTEF-MYRRHA

In the frame of the FP7 European project Central Design Team (CDT), an extensive simulation study has been done to assess the main shielding problems in view of the construction of the MYRRHA accelerator-driven system at SCK•CEN. A method based on the combined use of the two Monte Carlo codes MCNPX and FLUKA has been developed, with the goal to characterize realistic neutron fields around the core barrel and build complex source terms, to be used in detailed analyses of the radiation fields due to the system in operation, and of the coupled residual radiation. The results evidenced a powerful way to analyze the shielding and activation problems, with direct and clear implications on the design solutions.

III. Neutron shielding studies on an advanced Molten Salt Fast Reactor (MSFR) design

In a Molten Salt Reactor (MSR), the fuel is dissolved in a fluoride salt coolant. Prior MSRs were mainly considered as thermal-neutron-spectrum graphite-moderated concepts. Nowadays, the R&D has focused on the development of fast-spectrum MSR concepts (MSFR). In MSFR, the neutron spectrum is significantly harder. Fast neutrons are born close to the core vessel wall. Hence, the shielding becomes more challenging than in thermal MSR concepts. In order to assess the main shielding problems of MSFR, a 2D model of advanced MSFR was analyzed using the HELIOS unstructured mesh neutron transport code. A very high neutron flux was demonstrated in the core vessel walls. In a case of core vessel failure, the outer vessel has to fulfill a safety function. Therefore, the advanced MSFR geometry was optimized in order to reduce the fluence on the outer vessel. It was shown that the outer vessel can be shielded by a well blanketed system, which keeps the fluence within the limit of 1020 n/cm2 for a reasonable operation of 80 years.

India

Techno-economic implications for achieving high burnup targets with MOX fuels were presented by the Indian delegate Mr. Clement Ravi Chandar Sowrinathan. He explained the fuel and material development programme on-going in India and highlighted irradiation experiences with different fuels – mixed carbide, mixed oxide and metal fuels – that are currently under irradiation in the FBTR. Various issues related to high burnup were pointed out, as instance the economic advantages, concerns regarding penalties on the fuel performance due to lowering of melting point and thermal conductivity of the fuel, high fission gas release and JOG formation at higher burnup. Concerning the pin integrity, new structural materials like D9I are required for burnups higher than 150 GWd/t. Also, with respect to core design aspects, the need for extra fuel SA for meeting the loss of reactivity was indicated. The compensation in the control rod worth in the form of enhanced enrichment requirements was brought out for burnup up to 200 GWD/t. The effect of the increased burnup on the reprocessing of the fuel was also discussed, from the point of view of specific activity of the fuel and its decay heat. It was shown that both the issues increase but they are not linearly varying with the burnup. Regarding the fuel cycle cost (FCC), it was shown that for MOX fuel, FCC initially decreases with burnup but saturates around 150- 200 GWd/t, then it starts increasing. In the light of the above findings, it was informed that India wants to limit the MOX fuel burnup to maximum 200 GWd/t and restrict the construction of MOX fuelled FBR to two or four units only. However, there is an active consideration as far as imported MOX fuelled FBR. In the future, India looks towards the launch of a series of metal fuelled 1000 MW(e) FBRs, from the considerations of high breeding ratio and high burnup capability of the metal fuel. Towards this, some irradiation programmes with metal fuel pins are already on-going. Also, some essential R&D activities to be pursued by India were also indicated.

Italy

The Lead-cooled Fast Reactor – one of the six concepts identified by the Generation-IV International Forum – is presently considered in Europe as the shortest-term alternative to the Sodium-cooled Fast Reactor (the only Fast Reactor technology which has accumulated operational experience thus far), thanks to the significant extension of its technological base in the past years.

Nevertheless, a main issue challenges the development of LFR projects: the compatibility of structural materials in a harsh environment – hard spectrum neutron irradiation and corrosive/erosive action of lead. A thorough R&D campaign has identified viable solutions for short-term implementation, as well as promising possibilities for the long-term, following a general approach: selecting a base material suitable to resist the stresses (also neutron-induced) along with a strategy tailored to the surface protection of the latter so as to face the attack of lead.

The presentation provided an overall picture of the main achievements that led to the selection of the candidate materials and protections for the Advanced LFR European Demonstrator (ALFRED) core, discussing the interaction between the selection of these materials and the design of the ALFRED core. The status of the qualification of these materials was also presented, together with a perspective view of the remaining R&D progresses envisioned for the design of the industrial scale LFR (the European LFR –

ELFR), in the overall framework of the European LFR technology chain, discussing the potentialities of the LFR concept which depend on the availability of innovative materials.

Republic of Korea

The national plan for SFR development was authorized by KAEC in 2008 and updated in 2011 in order to refine the plan and to consider the available budget. The key milestones include specific design of a prototype SFR by 2017, design approval by 2020, and construction of a prototype SFR by 2028. The on-site spent fuel storage is foreseen to be started its limit around by 2016. The objectives of the prototype SFR are to test and demonstrate the performance of TRU metal fuel required for a commercial SFR, and to demonstrate the TRU transmutation capability as a burner reactor.

Prototype SFR is initially loaded with uranium metal fuel (U core) and it will evolve into LTRU (TRU from LWR spent fuel) core and finally will evolve into MTRU (LTRU and recycled TRU) core.

Unchanged core structural geometry from U core to MTRU core, which does not violating all core design constraints, was found. Performances of each core stage were examined and also reactivity coefficient and shutdown margin of U core were evaluated. Tight pressure drop constraint which is conservative constraint to strongly ensure natural circulation for decay heat removal, lowers power density. Trade-off studies with relaxed pressure drop constraint are on-going to increase the core performance and major feature of core design will be finalized this year.

Advanced cladding materials for high burnup fuel are being developed and ferritic/martensitic steels (9~12% Cr) are preferred candidate as fuel cladding materials. Grade 92 basis alloy was designed and manufactured, and the test shows superior high temperature mechanical property to the conventional material. Processing technologies related to tube making process are developed to enhance high temperature mechanical property. HT9 cladding tube was manufactured using domestic technology as a preliminary stage prior to the manufacture of advanced FMS cladding tube and out-of pile mechanical properties were evaluated. Advanced cladding tube is being developed and prepared for irradiation test.

The Netherlands

The following topics were discussed in the presentation.

- NRG contribution to Matter and Matisse projects;
  NRG is contributing in Matter to develop fracture toughness testing guidelines and verification of the guidelines by round robin tests. NRG is also contributing in Matisse to investigate the mechanisms of crack initiation and propagation under constant and cyclic loading.

- Practical challenges measuring Liquid Metal Embrittlement (LME);
  Challenges in this area are: control of the environment concerning O_2 content which is solved using a control system achieved from KIT, measuring crack propagation – which is achieved using measurement bridges and an indirect measurement system using two displacement transducers outside the tank; perform both the precrack and the test in liquid metal in order to have full wetting of the crack tip.

- Description of LMET: 3 tank system of which one tank for melting, one for preconditioning and one for final conditioning and testing. Tanks are placed in a hydraulic materials testing machine and is designed to be placed in hot cell.

- Possibilities with LMET: fracture toughness (K1C and J1C), fatigue crack propagation, elastic fatigue, tensile up to 400 degree C

- Results:
  Fatigue crack propagation and fracture toughness tests are performed and evaluated. Wetting is assured and results are repeatable. Fracture surfaces of both T91 in contact with Ar and with LBE are similar as equivalent surfaces found in literature.

Sweden

The main issues related to both structural materials, modeling and design of the European Lead-Cooled TRAining reactor (ELECTRA), currently under development in Sweden and promoted by the Royal Institute of Technology (KTH), Chalmers University of Technology, and Uppsala University, were presented and discussed.

ELECTRA is a 0.5 MW fast reactor cooled by natural circulation of lead, intended mainly for education and training purposes. It is aimed at serving also as a test bed for Lead-cooled Fast Reactor technology, being a facility for R&D on fuel recycles and manufacture, and on fast reactor dynamics.
The reactor and the complementary fuel cycle center (ELECTRA-FCC) is planned to be built in Oskarshamn, Sweden.
The small core size, necessarily sought in order to allow a 100 % natural circulation cooling, is achieved thanks to the application of U-free (Pu,Zr)N inert matrix fuel, which permits to design a critical core with less than 400 fuel pins having an active height of 30 cm due to the high Pu density it can provide.

The thermal conductivity for (Pu0.4,Zr0.6)N was measured by VNIINM, resulting of the order of 14 W m^{1} K^{-1} in the ELECTRA fuel nominal operating temperature range, and high temperature stability tests were carried out under vacuum, Ar and N.
(Pu,Zr)N fuel with 88 % density and less than 0.3 wt. % O and C impurities was produced within a collaboration between KTH and JAEA by heat treatment of a mixture of ZrN fabricated from metallic Zr, and PuN fabricated from PuO_2. The process optimization is currently ongoing, and two fuel pins are planned to be fabricated, irradiated and qualified at the Paul Scherrer Institut (PSI) in Switzerland, where the CONFIRM project has been successfully concluded. Moreover, a Pu fabrication laboratory is under commissioning at Chalmers, and the relicensing of a Pu conversion facility commissioned and operated in Studsvik is under discussion for ELECTRA fuel fabrication.

The current reference material selected for the fuel cladding tubes of ELECTRA is 15-15Ti, surface-alloyed with FeCrAlY using the GESA method. This material provides a combination of high creep rupture strength and excellent corrosion resistance, while its swelling performance is acceptable for the End of Life dose of less than 40 dpa expected in the ELECTRA cladding.

For heat exchangers and above core structures, bulk Fe10CrAl-RE alloys are under consideration. These alloys are developed in close collaboration with Sandvik. 10000 hours tests in stagnant lead have shown that a very thin (< 200 nm) and stable alumina scale is forming on these steels at a temperature of 550 °C and an oxygen concentration of 0.1 ppm weight in the lead. Thermal aging tests reveal that the considered alloys are not affected by spinodal decomposition. Their relatively poor creep resistance may be improved by introduction of oxide dispersions, relying on gas atomization techniques currently used for industrial production of Fe20Cr5Al-ODS steels by Sandvik.

ELECTRA core consists of a single fuel assembly with an active height of 30 cm and hexagon flat-to-flat of 28.2 cm. Twelve absorber elements (rotating drums) are placed outside the core assembly to improve the neutron economy. Their design has been optimized on the basis of the Larson-Miller parameter and transient analyses outcomes indicating that the clad integrity is ensured by a maximum reactivity insertion of 1.7 $ at a rate of 1 $ s^{-1}.

The reactor belongs to the class of reflector-dominant fast systems, thus the neutron lifetime is extremely sensitive to the core surrounding regions, arising the need of a reliable method to calculate such a parameter, which is considerably important for both operation and safety.

Another challenge in transient modeling is pointed out also concerning the axial expansion reactivity feedback in SAS4A/SASSYS-1, as an inconsistency in the built-in model has been found, bringing significant errors in the case of small cores. Discussion with Argonne National Laboratory code development team has been initiated.

Finally, the need of including the capability of properly treating natural circulation in safety codes is stressed.

European Commission
To support the drafting, development, implementation and monitoring of European energy policy, the Institute for Energy and Transport of the European Commission's Joint Research Centre conducts pre-competitive research in the areas of experimental qualification of advanced materials as well as simulation and modelling of reactor safety, integrity of systems, structures & components as well as material properties. The work covers assessments, design optimisation, and characterisation of advanced material properties and degradation mechanisms for improvements to the safety of current and advanced reactor systems, the latter envisaged to meet the EU's long-term energy needs while respecting enhanced safety, sustainability, and economic aspects.

JRC/IET focuses on pre-normative research with respect to material characterisation, component testing and assessment procedures. Integration of component testing, micro-structural analysis and modelling from physics-based to engineering-based methods is central. Emphasis is also put on material and system performance in the relevant environments and conditions, incl. in prototypical operational transient and accidental conditions. The current research topics include:

- Pre-normative R&D for non-standard test (e.g. small punch and segmented mandrel tests) and materials data management;
- Design codes and standards (incl. RCC-MRx);
- Development and application of models at different length-scales and for different applications, such as for fuel-clad chemical and mechanical interactions as well as fracture assessment of welds;
- Thermal fatigue, creep, and creep-fatigue;
- Stress-corrosion cracking and other environmental effects in dedicated tests loops for BWR, PWR, and supercritical water reactors;
- Advanced micro-structural characterisation methods;
- Residual stress measurements using neutron and synchrotron X-ray diffraction;
- Materials testing for among others LFR, SFR, and process heat applications.

The research is linked and contributes to related EERA Joint Programme Nuclear Materials (EERA JPNM), EURATOM Framework Programme projects, Generation IV International Forum (GIF), European Sustainable Nuclear Industrial Initiative (ESNII), International Atomic Energy Agency as well as OECD's Nuclear Energy Agency (OECD NEA) activities.

After the presentations given by representatives from Member States, Mr. Monti, on behalf of Mr. Zeman from the IAEA Department of Nuclear Science and Applications (NA), presented the NA departmental activities related to nuclear materials for advanced reactor systems, in particular on-going CRPs and recent publications in the field.

3.3 Conclusions and recommendations

The following comments and recommendations were pointed out during the discussions:

- Representatives from different Member States expressed their interest in establishing international collaboration in the area of fast reactor core design and structural materials;
- In view of future activities devoted to the development and qualification of materials and safety demonstration of fast reactor cores, the need of standardization of testing procedures at international level was clearly recognized by meeting participants;
- It would be beneficial to develop IAEA guidelines for the systematization of the management of experimental data (for instance for structural materials properties), which are an important resource for further safety assessments and R&D efforts. In this context, the importance of keeping databases updated was also underlined;
- It was proposed to perform an update of the handbook on heavy liquid metals (published by the OECD-NEA) under the auspices of the IAEA in order to include the contribution from the non-OECD-NEA Countries. In case of implementation, the IAEA will search for collaboration with the OECD-NEA;
- A large number of participants expressed the need for the international harmonization of code and standards for structural materials and components design for fast reactors;
- The need for guidelines for qualification and in service inspection methods for core materials was highlighted. In this area, in the frame of the programme&budget 2014-2015 the IAEA is planning to hold a technical meeting specifically devoted to ISI&R in fast reactors.

In conclusion the following recommendations on future IAEA activities were expressed by the participants:

- Standardisation of data for improved safety;
- Behaviour of structural materials in HLM environments (incl. environmental effects – liquid metal embrittlement, stress-corrosion cracking, corrosion fatigue and creep, irradiation, etc.);
- Benchmark on fuel performance modelling;
- Benchmark on material performance modelling (at relevant high temperatures and doses);
- Benchmark to support qualification of coupled neutronic-thermal hydraulic codes, possibly on the basis of available experimental data;
- In-service inspection in liquid metal environments;
- Guidelines for systematic collection of experimental data and for testing procedures;
- Establishment of common design safety criteria for all the fast reactor concepts (Note: for GENIV SFR there are already on-going GIF-IAEA activities).

ANNEX I: List of Participants

Belgium	Mr	Schuurmans Paul	SCK•CEN
China	Mr	Bai Yunqing	Chinese Academy of Sciences, Institute of Nuclear Energy Safety
China	Mr	Li Chunjing	Chinese Academy of Sciences, Institute of Nuclear Energy Safety
China	Mr	Wu Yican	INEST
Germany	Mr	Rachamin Reuven	HZDR
India	Mr	Sowrinathan Clement, Ravi Chandar	IGCAR
Italy	Mr	Angiolini Massimo	ENEA
Italy	Mr	Grasso Giacomo	ENEA
Korea, Republic of	Mr	Bae MooHoon	KINS
Korea, Republic of	Mr	Chang Jinwook	KAERI
Korea, Republic of	Mr	Shin An-Dong	KINS
Netherlands	Ms	Jong Monica	NRG
Sweden	Ms	Bortot Sara	KTH
European Union	Mr	Tucek Kamil	JRC Petten
IAEA	Mr	Monti Stefano	
IAEA	Mr	Toti Antonio	

ANNEX II: Meeting Agenda

WEDNESDAY, 12 JUNE 2013

Time	Topic	Speaker
	Opening Session	
14:00 – 15:00	• Welcome	**Mr S. Monti** NPTDS, IAEA
	• Opening Remarks	**TBD** NENP, IAEA
	• Self-introduction of the participants • Appointment of the Meeting Chair	All Meeting Participants
	• Chairperson's remarks	Meeting Chair
	• Discussion and Adoption of the Agenda	Meeting Chair
15:00 – 15:30	• Recent IAEA achievements in the field of fast neutron systems and presentation of scope and objectives of the meeting	**Mr S. Monti** NPTDS, IAEA
	Presentations by experts from the IAEA Member States	
15:30 – 16:00	• Materials aspects of the MYRRHA core	**Mr P. Schuurmans** SCK.CEN, Belgium
16:00 – 16:30	*Coffee Break*	
	Presentations by experts from the IAEA Member States, cont.	
16:30 – 17:00	• Development plan and R&D status of China lead-based reactor (CLEAR)	**Mr Y. Wu** INEST & CAS, China
17:00 – 17:30	• Tools and applications for core design and shielding in fast reactors	**Mr R. Rachamin** HZDR, Germany
17:30 – 18:00	• Wrap-up of the first day meeting	**Chairman**
18:00	*End of Day 1*	

THURSDAY, 13 JUNE 2013

Time	Topic	Speaker
Presentations by experts from the IAEA Member States, continued		
09:30 – 10:00	• Achieving high burn targets with MOX fuels: techno-economic implications	**Mr S. C. Ravi Chandar** IGCAR, India
10:00 – 10:30	• The materials challenges for lead-cooled fast reactors core design	**Mr G. Grasso** ENEA, Italy
10:30 – 11:00	*Coffee Break*	
Presentations by experts from the IAEA Member States, continued		
11:00 – 11:30	• A core concept of PGSFR and plan for core structural material tests	**Mr J. Chang** KAERI, RoK
11:30 – 12:00	• Liquid metal embrittlement testing in heavy metals: development and first fracture mechanics test results	**Ms M. Jong** NRG, Netherlands
12:00 – 12:30	• Structural materials for the design of ELECTRA	**Ms S. Bortot** KTH, Sweden
12:30 – 14:00	*Lunch Break*	
14:00 - 14:30	• Liquid metal reactor core design and structural material research by the Institute for Energy and Transport at the European Commission's Joint Research Centre	**Mr K. Tucek** EC-JRC Petten
14:30 – 15:00	• Title TBD	**Mr Monti on behalf of Mr A. Zeman** NAPC, IAEA
15:00 – 15:30	• General discussion	Chair + All meeting participants
15:30 – 16:00	*Coffee Break*	
Identification of research and technology gaps to be covered through new R&D initiatives to be carried out under the aegis of the IAEA		
16:00 – 17:00	• Proposals from the participants	All meeting participants
17:00 – 17:30	• Discussion and wrap-up of the second day meeting	Chair + All meeting participants
17:30	*End of Day 2*	
18:00	*Cocktail reception organized by the IAEA*	

FRIDAY, 14 JUNE 2013

Time	**Topic**	**Speaker**
	Closing Session	
09:30 – 10:00	• Conclusions and recommendations of the meeting	Chair + All Meeting Participants
10:00 – 10:30	• Drafting of the meeting report	**Mr S. Monti** with the support of all meeting participants
10:30 – 11:00	*Coffee Break*	
11:00 – 11:30	• Finalization of the meeting report	**Mr S. Monti** with the support of all meeting participants
11:30 – 12:00	• Closing remarks	Meeting Chairman and IAEA
12:00	*End of the Technical Meeting*	