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Technical Meeting on Innovative Fast Reactor Designs with Enhanced Negative Reactivity Feedback Features

International Atomic Energy Agency
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NOTE

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MEETING REPORT

1. Introduction and Background

In order to increase the sustainability of nuclear energy – in particular as far as the use of natural resources and the minimization of the nuclear waste burden - fast neutron reactors with closed fuel cycles must be developed, since they can typically multiply by over a factor 50 the energy production from a given amount of uranium compared to current reactors, and transmute by fission the transuranic elements (plutonium and minor actinides) which are responsible for the highest heat loads and radiotoxicity in the long term. Fast reactors will play 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 reactor's importance for the sustainability of the nuclear option, currently there is worldwide renewed interest in fast reactor technology development, as indicated, e.g., by the outcome of the Generation IV International Forum (GIF) technology review, which concluded with 3 out of 6 innovative systems to be fast reactors (sodium cooled fast reactor, gas cooled fast reactor, and heavy liquid metal cooled fast reactor), plus a potential fast core for a 4th concept, the super-critical water reactor. Currently, fast reactor construction projects are on-going in India (PFBR) and Russian Federation (BN-800), whilst in China the first experimental fast reactor (CEFR) has been recently connected to the grid. Innovative fast reactor programs are carried out, not only in these three countries, but also in Europe (in particular in France), Japan, Republic of Korea and the USA.

One of the most challenging issue of next generation fast reactors is to develop innovative cores with improved safety characheristics (e.g. kinetic and dynamic behaviour), by reducing the coolant void reactivity effect, enhancing negative reactivity feedbacks and adopting passive shutdown systems.

The IAEA, within the framework of its Nuclear Energy Department's *Technical Working Group on Fast Reactors* (*TWG-FR*), assists Member States activities in all technology development areas by providing an umbrella for information exchange [topical Technical Meetings (TMs), Workshops and large Conferences] and international cooperation on R&TD [Coordinated Research Projects (CRPs)], enabling scientists and engineers from research centres, industry and academia to share innovative solutions and best practices. This topical TM is addressing Member States' expressed information exchange needs in the field of innovative fast reactors, with particular emphasis on the development of advanced reactor physics simulation tools, as well as on core designs with enhanced negative reactivity feedback features.

2. Objective of the Technical Meeting

The objective of the TM was to review and discuss the safety characteristics and the performances of the core of innovative fast reactor concepts, as well as to present the on-going R&D activities in the area of core design and advanced simulation tools and methods for fast reactor core physics analysis. The focus was on fast spectrum cores optimized for actinide utilization and transmutation and, in particular, on core designs with enhanced negative reactivity feedback effects.

3. Opening Remarks – Mr. T. Koshy, Section Head of NPTDS

Mr. Koshy, section head of the Nuclear Power Technology Development Section at IAEA, opened the meeting and welcomed the participants.

In his introductory remarks, Mr. Koshy focused attention to the current need of regaining social acceptance of nuclear energy, which is essential for any future development of nuclear power as a large contributor of the world's energy needs. Unless we can persuade people to reaffirm nuclear safety with added safety measures, the Member States cannot fully benefit the enormous potentials and advantages that nuclear energy is able to offer.

In this context, the improvement of safety levels represents a primary objective, since safety of nuclear power plants is historically one of the main components of public concerns on the utilization of nuclear system. Therefore, important efforts have to be devoted to the enhancement of the safety performance of every nuclear energy system, including current and innovative fast reactors concepts. In implementing this action, the IAEA will play a prominent role, supporting and assisting Member States' activities and providing a global framework for knowledge sharing and information exchange.

After the remarks provided by Mr. Koshy, the participants agreed to appoint Mr. B. Merk as Chairman of this TM. Mr. Merk welcomed the participants inviting them to comment on and approve the proposed Agenda (see annex I).

The Agenda of the meeting was approved as proposed by the Secretariat with a small modification concerning the cancellation of the presentation by Mr. Gann who was not able to attend this meeting.

The Chairman invited Mr. Monti to give his presentation on IAEA activities in the field of FRs development and deployment as well as on "Overview of IAEA Activities in support of Fast reactors Development and Deployment and Objectives of this Meeting".

4. Overview of IAEA Activities in Support of Fast Reactors Development and Deployment and Objectives of this Meeting, Mr. S. Monti

Mr. Monti presented in brief the on-going and new coordinated research projects, the forthcoming Topical Technical Meetings in the field of fast reactors and the related technical publications. He also described the IAEA TWG-FR composition, mandate, and main activities.

A summary is provided:

- organize regular Topical Technical Meetings for information exchange;
- organize large conferences on different aspects of FR and related fuel cycles. On this point he announced the IAEA FR13 International Conference on Fast Reactors and Related Fuel Cycles, next edition of the previous FR09 Conference held in Kyoto at the end of 2009. FR13 will be hosted by France in Paris from 3 to 7 March 2013;
- establish a forum for broad exchanges on technical requirements on GENIV FR;
- carry out collaborative research projects (CRP) in the field of FR and ADS; on this topic Mr. Monti summarized outcomes of recently completed CRPs (Analytical and Experimental benchmark Analysis of ADS; Analysis of and lesson learned from the operational experience with Fast reactor equipments and systems; Control Rod Withdrawal and Sodium natural circulation tests performed during the PHENIX end-of-life tests), ongoing CRPs (Benchmark analysis of sodium natural circulation in the upper plenum of the Monju reactor vessel) and, finally, new CRPs to launched in 2012-2013 (Benchmark analysis of an EBR-II shutdown heat removal test; SFR: sodium properties, sodium facility design and safety guidelines; source term for radioactivity release under FR core disruptive accident conditions);
- secure training and education in the field of fast neutron system physics, technology and applications;
- provide support to IAEA Nuclear Safety and Security Department for preparation of fast reactor safety standards, requirements and guides;
- support fast reactor data retrieval and knowledge preservation within the Member States.

He also stressed the importance of continuous collaboration and interaction with INPRO and GIF.

Members of the IAEA Technical Working Group on Fast Reactors include the following:

- Belarus
- China
- Germany
- Italy
- Kazakhstan
- Netherlands
- Switzerland
- UK
- OECD/NEA
- Brazil
- France
- India
- Japan
- Korea
- Russian Federation
- Ukraine
- USA
- European Commission

Observers of the TWG include: Argentina, Spain, Belgium and Sweden

Mr. Monti announced that the next meeting of the TWG-FR will be held at ANL (USA) from 20 - 22 June 2012. He also provided information on forthcoming related TWG-FR technical publications and technical meetings/Workshop.

Mr. Monti concluded his presentation with a short overview of the main safety characteristics and challenging safety issues of LMFRs as well as the motivation and objectives of this TM.

Information on IAEA activities and initiatives related to Fast neutron systems can be found at:

http://www.iaea.org/NuclearPower/FR/

http://www.iaea.org/NuclearPower/Technology/TWG/TWG-FR/

5. Member States Presentations

With exception of the IAEA Technical Leaders, presentations by the Member States were given in alphabetical order, by country name.

5.1 ASTRID Core - Design objectives, design approach, and R&D in support (*Mr. Devictor, Mr. Mignot, CEA - France*)

The presentation focused on the ASTRID project, in particular on the core design objectives which can be summarized as follow:

- natural behaviour favourable for transients of unprotected loss of flow and loss of heat sink, with a target criterion “no sodium boiling for a ULOSSP transient”;
- sodium void effect minimized (and even negative for CFV type core);
- natural behaviour favourable for a complete control rod withdrawal (with no detection) with a target criterion “no fuel fusion”;
- improved performances (cycle length ≈ 480 efpd, high fuel burn up, and breeding gain ≈ 0).

Two core concepts are studied in the frame of the ASTRID project, called SFRv2 and CFV. Part of the R&D in support of the ASTRID core design focuses on materials, static and dynamic mechanical behaviour, simulation tools and development of facilities for qualification, surveillance and protection.

In particular, the R&D efforts are focusing on two innovative lines of protection:

- capability for a stable sodium boiling at the upper part of the subassembly. Such disposition offers additional margin to avoid core degradation;
- design of a third shutdown system without control rod mechanism, called SEPIA. SEPIA is not included at present in the reference design of ASTRID core, since it is necessary to progress in its qualification in order to demonstrate its performance and reliability. Moreover, the specifications for ASTRID third shutdown systems will depend on the potential weaknesses in the safety analysis, and SEPIA’ specifications will then be compared to these ones in order to assess its relevancy. The CEA, with its partners, is also reviewing capabilities of other potential third shutdown systems, for instance SLD, CREED, etc.

5.2 Pre-conceptual design study of ASTRID core *(Mr. Devictor, Mr. Mignot, CEA - France)*

In the framework of the ASTRID project at CEA, core design studies are performed with the support of the industrial partners AREVA and EDF. Pre-conceptual design studies are conducted in accordance with GEN IV reactors criteria, in particular focused on the improvement of the safety features. The enhancement of safety in a sodium-cooled fast reactor requires the review of many aspects of the design and it represents a rather lengthy process. Currently, two concepts of core are under evaluation, one classical derivate from the SFR V2B and one more challenging called CFV (low void effect core) with a large gain on the sodium void effect. In accordance with the ASTRID project requirements, both the two reactor cores are designed for a power of 1500 MW(th).

The SFR V2b core has the following specifications: a very low burn-up reactivity swing (due to a small cycle reactivity loss) and a reduced sodium void effect with regard to past designs such as the EFR (around 2$ minus). Its performances include an average burn-up of 100 GWd/t, and an internal conversion ratio equal to one.

The current CFV design reaches the objective of a negative sodium void worth while maintaining core performances, and that characteristic is checked for larger power (3600MWth). The CFV pre-conceptual design included safety analyses on different transient scenarios like ULOF (Unprotected

Loss of Flow), in order to assess its intrinsic behaviour compared to a more classical design like V2b core.

The analyses performed on this core concept show a favorable behaviour comparing a homogeneous core, and proved the potential capability of the design to enhance the prevention of sever accident scenarios, for instance for loss of flow initiator event, in accordance with the safety requirements. Particularly, the reactivity feedback of the CFV core provides larger grace delays for unprotected transients of loss of flow, and lower neutronic shutdown temperature for loss of heat sink situations.

5.3 Behavior of a heterogeneous annular FBR core during an Unprotected Loss of Flow accident: Analysis of the primary phase with SAS-SFR

(Mr. Massara, EDF - France)

In the framework of a substantial improvement of FBR core safety connected to the development of a new Gen IV reactor type, heterogeneous cores with innovative features are being carefully analysed in France since 2009. At EDF R&D, the main goal is to understand whether a strong reduction of the Na-void worth – possibly attempting a negative value – allows a significant improvement of the core behaviour during an unprotected loss of flow accident (ULOF). Even the physical behaviour of such a core is analysed, before and beyond the (possible) onset of Na boiling. Hence, a heterogeneous design, featuring an annular shape, a Na-plenum with a B4C plate and a stepwise modulation of fissile core heights, was developed at EDF by means of the SDDS methodology, with a cumulative Na-void worth of –1 $.

The behaviour of such a core during the primary phase of a severe accident, initiated by an unprotected loss of flow, was analysed by means of the SAS-SFR code. The study was carried-out at KIT and EDF in the framework of a scientific collaboration on innovative FBRs severe accident analyses. The results show that the reduction of the Na-void worth is very effective, but is not sufficient alone to avoid Na-boiling and, hence, to prevent the core from entering into the primary phase of a severe accident. Nevertheless, the grace time up to boiling onset is significantly enhanced in comparison to a traditional homogeneous core design, and only an extremely low fraction of the fuel (<0.1%) enters into melting at the end of this phase. A sensitivity analysis shows that, due to the inherent neutronic characteristics of such a core, the gagging scheme plays a major role on the core behaviour: indeed, an improved 4-zone gagging scheme, associated with an enhanced control rod driveline expansion feed-back effect, finally prevents the core from entering into sodium boiling.

This major conclusion highlights both the progress already accomplished and the need for more detailed future analyses, particularly concerning: the neutronic burn-up scheme, the modelling of the diagrid effect and the control rod driveline expansion feed-backs, as well as the primary/secondary systems thermal-hydraulics behaviour.

5.4 On the Use of Fine Distributed Moderating Material to Enhance Feedback Coefficients in Fast Reactors

(Mr. B. Merk, HZDR - Germany)

Recently, the use of moderating materials in fuel assemblies for sodium-cooled fast reactors has been investigated. Especially the fine distribution of the moderating material in a layer inside the fuel rod or inside the wire spacer has shown very promising results for the enhancement of the feedback coefficients.

The validity of the HELIOS results has been demonstrated in comparison with MCNP and the transferability of the effect to full core calculations has been shown. The fine distribution of the moderating material is very attractive since it causes a very limited influence on the safety-relevant fuel assembly structure and on the operational parameters like power distribution and end-of-life burn-up distribution. The influence of the moderating materials on systems containing fuels for Minor Actinide transmutation (3% Am, 5% Am, and 2% Np – 2% Am) is examined on the basis of detailed lattice calculations based on 112 energy groups and an unstructured mesh geometry modelling of all fuel assembly details. The influence of the insertion of the Minor Actinides on the fuel temperature and the coolant coefficient has been investigated for a reference case and the possibilities of enhancing the feedback coefficients by the insertion of ZrH moderating material have been analysed. The modifications in the power and burnup distribution due to the introduction of the moderating material ZrH have been studied. Further on, the transmutation potential has been compared for the cases with and without moderating material. Neptunium and Americium is slightly more reduced during burnup of the fuel with moderating material; only the Curium production rises slightly. A detailed analysis demonstrates that the increase of Cm breeding due to the use of moderating material is lower than the additional amount of destroyed Np and Am. Thus, the results demonstrate that even more Am is burnt with the investigated fuel assembly containing moderating material inside the wire wrapper.

Additionally, a first test on the effect of the use of fine distributed moderating material in a lead-cooled fast reactor will be performed. The test is based on preliminary data of the fuel assembly in the GUINEVERE facility. The effect of the moderating material on the neutron spectrum, on the k_{inf}, and on the fuel temperature feedback of the zero power facility is analysed. A literature study is provided for the choice of the ideal moderation compound – yttrium mono hydride with sufficient stability up to ~1300°C.

Over all, the use of fine distributed moderating material has the potential to open the stage for designable feedback coefficients in fast reactors without affecting the operational parameters and core coolability.

5.5 Methodology to Enhance Negative Reactivity

(Ms. Thangavel, IGCAR – India)

There are continuous attempts in the nuclear community to enhance negative reactivity feedbacks in the reactor core to ensure enhanced safety. As the higher magnitude of power coefficient does not necessarily ensure safe shutdown in all accidents, efforts are devoted to find out the methodologies to enhance the negative reactivity which are effective in Unprotected Loss of Flow (ULOF) and Unprotected Transient over Power (UTOP) incidents.

Based on both the experience gained in the design and safety analysis of the 500 MW(e) Prototype Fast Breeder Reactor (PFBR) and the literature analysis, the following methodologies to enhance the negative reactivity have been pointed out: (1) Addition of moderating material; as the addition of moderating material softens the neutron spectrum, it increases the magnitude of prompt negative Doppler feedbacks. This contribution is effective in reactivity transient where power rise is the matter of concern. (2) Increasing the thermal conductivity of the fuel by choosing a suitable alloy material. If thermal conductivity of the fuel increases, the difference in temperature between fuel and coolant decreases, allowing the fuel temperature to increase during ULOF with the enhancement of the negative Doppler feedback. (3) Negative reactivity through changing axial and radial boundary movement worth. The increase in the difference in boundary movement worth increases the negative reactivity effect. (4) Decrease the gap between spacer pads, this closes the gap very fast and initiates flowering as soon as the transient starts. Flowering of sub-assemblies enhances negative reactivity. Methodologies 2 to 4 enhance the negative reactivity when there is a rise in power to flow (P/F) ratio.

5.6 LFR Safety Features through Intrinsic Negative Reactivity Feedbacks

(Mr. G. Grasso, ENEA – Italy)

The design of Lead-cooled Fast Reactors (LFRs), like any other FR in general, must rely on the negative reactivity feedbacks for ensuring a safe intrinsic behavior of the reactor under operational as well as accidental conditions. To maximize the safety features, it is therefore required to cope with the physics peculiarities due to the coolant, facing the drawbacks and exploiting the advantages.

The first fundamental aspect to be considered is the possibility to exclude the coolant boiling scenario. This does not impact on the coolant density coefficient which, for standard designs of large LFRs typically remains positive when related to the active region only; it rather allows relaxing the limits of acceptability for the whole set of reactivity coefficients, by removing the coolant boiling from the list of limiting conditions to be respected during transients analyses.

A second intrinsic aspect of LFRs impacting on reactivity coefficients is the floating of steels in Lead. This can be exploited for passively actuating Control Rods (CRs) by buoyancy, hence positioning withdrawn CRs below the active zone. This has a twofold impact on LFR safety: using the lower diagrid as physical limit for the CRs stroke excludes by design their spurious withdrawal as accident initiator; the typical differential dilation of core support structures and CR drives determines a relative insertion (instead of the usual extraction) of the CRs into the core, incentivizing (instead of acting to the detriment) of the negative coefficient for axial expansion.

A final remark is posed on the necessity of extending the methodology for evaluating the reactivity coefficients, in order to have safety analyses reproduce correctly the physics of feedback mechanisms. As a matter of fact, the possibility of decomposing integral reactivity effects into elementary contributions, each one due to a different phenomenon, hence driven by a different system temperature, allows for a better simulation of the reactivity and anti-reactivity insertions that determine the evolution of reactors during transients, so as to more effectively prove system features or highlight system weaknesses.

5.7 A Safety Design Approach for Sodium-cooled Fast Reactor Core toward Commercialization in Japan

(Mr. Kubo, JAEA – Japan)

In Japan a series of conceptual design studies on sodium-cooled fast reactors (SFRs) toward commercialization has been conducted as subsequent project following the prototype SFR Monju. Those studies have been performed along with the development goals of Generation IV International Forum (GIF) aiming at fulfilment of multiple requirements in the areas of sustainability, economics, safety and reliability, proliferation resistance and physical protection. The design concept of JSFR has been developed to meet the development goals of GIF, and the R&D activities on the JSFR key technologies have been conducted.

Japan has been developing SFR technologies by means of design, construction and operation of the experimental reactors Joyo and Monju, development of core physics and thermal hydraulics tools, validation by using data obtained by critical assemblies and fuel pin bundle thermal hydraulic experiments, development of fuel materials and so on. Basic design technologies for large scale SFR core have been established based on these achievements.

In general fast reactor core is not in the most reactive configuration. Although coolant void reactivity might be positive in the central region of large scale core, coolant boiling can be avoidable against design basis accidents with adequate plant design, in which operation temperature range is sufficient below the coolant boiling temperature and reliable rapid shutdown systems are installed.

The following safety design requirements for SFR core and related systems have been selected based on defence-in-depth concept:

- the reactor core shall have inherent reactivity feedback characteristics with negative power coefficient. This means that positive coolant temperature coefficient is compensated by some negative reactivity components such as the Doppler coefficient.
- operation temperature range shall be sufficient below the coolant boiling temperature in order to avoid coolant boiling against anticipated operational occurrences and design basis accidents.
- if the plant’s state deviates beyond operational conditions, the reactor safe shutdown shall be achieved by automatic insertion of control rods. Two active reactor shutdown systems are installed

- as a design extension condition for prevention, failure of the active reactor shutdown system shall be assumed. Passive shutdown capability shall be provided under such conditions. This can be achieved by means of various design measures which include core reactivity characteristics, core elements and combination of them. JSFR adopts SASS; Self Actuated Shutdown System for the passive shutdown capability.
- as a design extension condition for mitigation, core disruptive accident shall be considered. In order to prevent severe mechanical energy release which might cause containment function failure, core sodium void worth shall be limited certain value and molten fuel discharge capability shall be required. JSFR core design foresees a sodium void worth less than 6 dollars and adopts inner duct structures in fuel assemblies (FAIDUS).

5.8 Conceptual core design study for Japan sodium-cooled fast reactor: Review of sodium void reactivity worth evaluation

(Mr. Ohki, JAEA - Japan)

The conceptual design studies and related R&D on the commercial large-scale Japan sodium-cooled fast reactor (JSFR) was carried out in the framework of the fast reactor cycle technology development (FaCT) project. As a next generation fast breeder reactor (FBR) plant, the JSFR adopts a number of innovative technologies in order to achieve economic competitiveness, enhanced safety and reliability.

The concept of the JSFR core is called “high internal conversion” type. This advanced concept features an economical advantage by achieving high total average discharge burnup (including blanket) and simultaneously a sufficient breeding ratio with a small amount of blanket. Large diameter fuel pins (10.4 mm) are foreseen to increase the internal conversion ratio, and hence to reduce the amount of blanket as much as possible. Moreover, the high internal conversion type core enables to extend the operation cycle length due to its high internal conversion ratio, which is also an economical advantage.

The breeding ratio is intended to have some flexibility and is set at around 1.0 to 1.2 under the design philosophy of using the compatible fuel assemblies among them. The breeding ratio of 1.1 (low-breeding core) is targeted for a smooth transition from LWRs to FBRs. Moreover, a higher breeding ratio of 1.2 (high-breeding core) is also considered to be able to answer to a possible high plutonium demand. Since the plutonium demand will be getting decrease along with the saturation of FBRs deployment, a break-even breeding ratio around 1.0 is enough to sustain FBR fuel cycle eventually.

Regarding the safety requirements, the core design foresees a sodium void reactivity about 6 $ or less, a core height of 100 cm or less, and an average core specific heat about 40 kW/kg-MOX or more. These preliminary characteristic values are chosen to prevent the super-prompt criticality in the initiating phase of the core disruptive accident (CDA), based on the experience gathered in sodium-cooled reactor core safety evaluations. In addition, an innovative fuel subassembly called FAIDUS (Fuel Assembly with Inner Duct Structure) is introduced in FaCT core subassembly design.

Sodium void reactivity strongly depends on the heavy metal composition supplied for fresh fuel fabrication. In particular, fertile nuclides such as minor actinides (MAs) deteriorate the sodium void reactivity because of their large capture cross sections in keV-energy region, and because their threshold fission cross sections in MeV-energy region increase the gradient of an adjoint neutron distribution in the energy space. It is considered to have a large variation in the fuel composition based on the wide variety of the LWR spent fuel composition and the way of recycle them in a FBR fuel cycle system. According to the study on the fuel composition’s change and its effects on the JSFR core design, we found the remarkable correlations between the sodium void reactivity and the other important core characteristics such as Doppler coefficient and the burnup reactivity.

The core neutronic design method has been recently re-established by taking into account of the progress in calculation codes and nuclear data developments. The latest Japanese nuclear data library, JENDL-4.0, has been applied for the effective cross sections calculation with a heterogeneous cell representation. Diffusion calculations have been performed in three-dimensional Tri-Z core representation with a 7 group effective cross section set for burnup calculation, and with an 18-group effective cross section set for reactivity coefficient calculation. The following corrections are necessary to evaluate the nominal design values: neutron transport correction, spatial mesh correction, group collapsing correction, ultra-fine group correction, and the bias factor correction. Currently, efforts are devoted to the verification and validation of the core neutronic design methods as well as developing the uncertainty evaluation methods.

5.9 Evaluation of Sodium Void Effect in the KALIMER-600 TRU Burner Core
(Mr. Kim, KAERI – Republic of Korea)

The safety design philosophy of KALIMER-600, an advanced sodium-cooled fast reactor concept developed in Korea, places high reliance on inherent safety mechanisms, i.e., passive responses to abnormal and emergency conditions, and thereby minimizes the need for engineered safety systems. To that extent, inherent reactivity feedback effects of importance in the KALIMER-600 core which are mainly triggered by coolant voiding, fuel temperature change, thermal expansion of the fuel and cladding, thermal expansion of the core structure, and thermal expansion of the control rod driveline should be properly assessed.

The presentation discussed the study carried out to evaluate the sodium void reactivity effect in the KALIMER-600 TRU burner core using two different calculation tools including the diffusion theory code, DIF3D, and the perturbation theory code, PERT-K. Various scenarios of different void fractions were examined at different locations in the core and fuel temperatures with considering the self-shielding effect, the Doppler effect, the effect of fission products buildup, the effect of varying Zirconium content in the fuel, and the effect of control rods position. Additionally, the uncertainties possibly incurred in applying different calculation models (Tri-Z and Hex-Z models), calculation methods (direct flux calculations, first-order and exact perturbation theory), and broad-group structures were evaluated and discussed.

The resolution into reactivity components regarding contributions from fission, scattering, absorption, and leakage terms was also determined and analysed to attain insight into the physical processes occurring inside the core. The results revealed details concerning the void effect in the KALIMER-600 TRU burner core and therefore could be useful for optimizing the passive safety design of the core under severe coolant loss accident conditions.

5.10 Study of the core compaction effects and its monitoring in sodium cooled fast reactors
(Mr. Zylbersztejn, Chalmer University of Technology - Sweden)

Sodium cooled fast reactors (SFR) received a renewed attention in the past years, in connection with the Generation-IV International Forum (GIF). The SFR is one of the six selected Gen-IV reactor types, and the first Gen-IV reactor to be built will be a SFR. It will be an industrial prototype reactor named ASTRID, built in France by CEA and its industrial partners AREVA and EDF.

Previous experience with the operation of sodium cooled fast reactors in France, such as Phenix (550 MWth), PFR (600 MWth) and Super Phenix (3000 MWth) showed that the processes with significant influence on the reactivity are different from those in light water reactors. While coolant void effects play a significantly smaller role, changes and deformation of the core shape and volume lead to change of the core reactivity. The changes in volume are commonly referred to as “core compaction”. The monitoring of core compaction is a priority for the safety of the installation, as the reactivity coefficient for core compaction is positive.

The presentation discussed three aspects of the core compaction problem in SFRs. The first part was dedicated to the understanding of the technical and physical origins of the different kinds of geometrical deformation of the core, and to the discussion of the importance of the core compactness variations study.

The second part was dedicated to the description of two possible ways of modelling of the core deformation for reactor physics calculations. The calculations were made by mean of the deterministic code ERANOS and the Monte-Carlo code MCNP to access to the variations of reactivity and static neutron flux. In particular the reactivity coefficient for volume variations and the flux on the top of the sub-assemblies were calculated for detection purposes.

The last part was dedicated to the investigation of the possibilities of monitoring core compaction variations from the point of view of the instrumentation, i.e. detector positioning. The French fast reactors are equipped with a cylindrical device on the top of the core (abbreviated as BCC) to hold the control rods and it is also the BCC which will contain the neutron instrumentation. The description of the device, its modelisation, and the study of its neutron behaviour, gave a good estimation of the visibility of the top core neutron flux shape deformations from the detectors positions.

The results of the second and thirds parts were combined to give an order of magnitude of the potential fluctuations in the detectors. One conclusion is that the combination of the information from detectors, situated on two circles, appears to be a feasible way to get an estimation of the detectability of the core geometry deformation. Further studies will be performed to analyse the space-time effects of such geometrical deformation, and to improve to reliability of the identification of these flux shape deformations and neutron noise source in case of core internals vibrations.

5.11 Mechanism of Negative Reactivity Feedback in Nuclear Burning Wave Reactor

(Mr. Fomin, Kharkov Institute of Physics and Technology - Ukraine)

The intrinsic mechanism of negative reactivity feedback in the innovative fast reactor concept based on the self-sustained regime of nuclear burning wave (NBW) was studied. The phenomenon of self-sustained “neutron-fission wave” was discovered and preliminary studied by Lev Feoktistov in 1988. Further, this concept was developed by several groups of investigators using different approaches and different names for this phenomenon: deflagration wave, criticality wave, CANDLE, nuclear burning wave, etc. Lately, the most frequently used name is a Traveling Wave Reactor due to TerraPower and Bill Gates activity.

In addition to many valuable features of the NBW reactor, such as the utilization of depleted uranium and thorium as a fuel, the long-term operation without refuelling and chemical reprocessing of fuel, close fuel cycle and some others, the most important and wonderful features of the NBW regime is a special kind of the negative reactivity feedback inherent to this regime.

The presentation discussed the results of studies on the negative reactivity feedback phenomenon in the NBW reactor with metal fuel of the mixed Th-U-Pu cycle and the Pb-Bi coolant. The results were achieved by solving numerically the non-stationary non-linear diffusion equation of neutron transport together with a set of the burn-up equations for fuel components and the equations of nuclear kinetics for precursor nuclei of delayed neutrons. A notable stability of the NBW regime relative to disturbances of the neutron flux in the system and to possible irregularities of the fuel composition was observed. This stability is conditioned by the above-mentioned negative reactivity feedback which underlies the intrinsic safety of the NBW reactor.

6. Conclusions and recommendations

Mr. Monti opened the discussion session pointing out that the different presentations gave a quite comprehensive overview on how to design a FR core with improved safety characteristics, in particular as far as enhanced reactivity effects features.

Indeed all the possible options are being considered within the different LMFR concepts under development worldwide: reduced core height, increase the pin diameter, introduce moderating material, heterogeneous core structure, etc.

It was also very interesting to have an overview of the different planned programmes for V&V&Q of data and codes to be used for the modelling and simulations of innovative FR cores.

On the other hand, each concept under development has its own design objectives and specific solutions; therefore it may be a difficult task to find common areas of investigation which might be the basis of international collaborations supported by the IAEA.

In order to identify these possible areas of collaboration and also to provide the IAEA with recommendations in the field of the design of fast reactor cores with enhanced reactivity effects features, Mr. Monti invited each delegation to offer their considerations and remarks.

The main remarks, conclusions and recommendations can be summarized as follows:

- The technical meeting was appreciated and well received by all the participants who recognized the importance of the international exchange of information and experience in the field of innovative FR core design;
- The different goals for FRs in the various countries (high breading gain and short doubling time, burning of MA, economic competitiveness, enhanced safety, etc.) reflect in different main technical choices: LFR or SFR, MOX or Metal Fuel; homogeneous or heterogeneous recycling (MA in the core or in devoted targets arranged in the core blanket), etc;
- There are also different safety approaches and goals adopted for the various innovative LMFRs with a different balance between safety and economic competitiveness; the exchange of information and the possible homogenization of the safety approach for LMFR is recognized as a common issue which would deserve attention and action by the IAEA but taking into account the related activities already carried out within Euratom, GIF, etc;
- From the utility perspective there is first the need to demonstrate the affordability of FR for Pu utilization in an economic way, being MA burning a second priority;
- Recognizing the importance of surveillance of the core there is the need of innovative devices and instrumentation also requiring R&D, information processing. International collaboration in this field is welcome;
- Possibility to adopt passive or inherent shut-down systems was presented at this meeting, including some design solutions;
- Indications were received as for common R&D activities: e.g. neutronic and coupled neutronic-thermalhdraulic calculations for various configurations, plant dynamics and system analysis e.g. for ATWS;
- In addition to prevention of core damage, mitigation of core damage consequences is considered an important issue for safety enhancement;
- Despite the different concepts and modelling and simulation tools, there is a common need to reduce the uncertainties in the determination of sensitive reactivity effects. This topic is recommended also for future CRP even as follow-up of previous/on-going research activities e.g. on analysis of BN-600 core and EBR-II;
- Further to safety, one has also to consider the other two issues of security and safeguard which may find suitable solution if afforded at an early stage of the design;
- It was also suggested to verify the international interest in verifying and validating simulation tools to be used for the design of very innovative concepts like travelling wave reactors which are able to address the issue of new specific type of negative reactivity effect.

LIST OF PARTICIPANTS

Name	Institution	Country
N .Devictor	CEA	France
S. Massara	EDF	France
G. Mignot	CEA	France
D. Verwaerde	EDF	France
B. Merk	HZDR	Germany
F. Puente-Espel	GRS	Germany
S. Thangarel	IGCAR	India
G. Grasso	ENEA	Italy
S. Kubo	JAEA	Japan
S. Ohki	JAEA	Japan
S.J. Kim	KAERI	Korea Republic of
A. Hakansson	Uppsala University	Sweden
C.F Hellesen	Uppsala University	Sweden
L. Wallin	Swedish Radiation Safety Authority	Sweden
P. Wolniewicz	Uppsala University	Sweden
F. Zylbersztejn	Chalmers University of Technology	Sweden
S. Fomin	Kharkov Institute of Physics and Technology	Ukraine
T. Koshi (p.t.)	IAEA	
S. Monti	IAEA	
A. Toti	IAEA	