TM-38228

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Working Material

Meeting Report of

the IAEA Technical Meeting in collaboration with NEA

on

Specific Applications of Research Reactors: Provision of Nuclear Data

International Atomic Energy Agency

Vienna, Austria

12 - 16 October 2009

Reproduced by the IAEA

Vienna, Austria, November 2009

CONTENTS

1. BACKGROUND AND OBJECTIVES

1.1 Background

Research reactors (RRs) have played and continue to play a key role in the development of the peaceful uses of atomic energy. The main applications of most RRs continue to be radioisotope production, neutron beam applications, silicon doping and material irradiation for nuclear systems, as well as teaching and training for human resource development. On the other hand, the last International Conference on Nuclear Data for Science and Technology, held in Nice (France) from 22 to 27 April 2007, has placed special emphasis on atomic and nuclear data needs for basic nuclear physics research, innovative power reactors and fuel cycles (e.g. dedicated reactors for nuclear waste transmutation, accelerator driven systems, Th-U fuel cycle, etc.), as well as efforts to realize fusion reactors (e.g. ITER) and to test materials needed for such facilities, medical applications including radioisotope production, computer simulations of radiation doses to patients and advanced cancer therapies, and analytical techniques adopted for cultural heritage diagnostics and materials composition analysis. RRs continue to occupy a visible and important place in these areas of study and application, along with dedicated accelerator-based neutron sources. For example, some installations like Lohengrin Fission Fragment Separator at Institute Laue-Langevin (ILL) in Grenoble, France, remains a unique place to study fission fragments and their properties from thermal neutron induced fission. Equally, one has to mention the importance of integral measurements performed at RRs to validate evaluated nuclear data libraries used by neutron transport and material evolution codes. In this respect, a new initiative of the NEA Working Party on Evaluation Cooperation has been launched in order to develop methods that combine integral experimental data from RRs and various differential data to examine targeted accuracies for different reactions, isotopes and energy ranges as long as they might affect the integral neutronic parameters used for the design of new nuclear reactors. Finally, some cross-section measurements for short-lived and on-line produced radioactive target nuclei are possible only at RRs because of the high neutron fluxes available.

1.2 Objectives

This IAEA Technical Meeting (TM), held in collaboration with NEA, focused on the specific application of RRs, namely the provision of nuclear data for various purposes. The following subjects were considered with the highest priority:

- Cross-section measurements (e.g. capture, fission, branching ratios, neutron multiplicities, etc.) in thermal, epithermal and fast (fission) neutron energy regions
- Measurements of averaged cross sections in a known neutron energy spectrum
- Measurements of fission fragments (e.g. mass and charge distributions)
- Decay studies of fission fragments and activation products
- Measurements of delayed neutron yields and time characteristics
- Dedicated irradiation experiments for incineration/transmutation studies
- Dedicated integral experiments and modelling to validate evaluated nuclear data libraries

The meeting also aimed at providing a forum to exchange ideas and information through scientific presentations and brainstorming discussions, leading to the following overall objectives: 1) enhancement of RR utilization in Member States for practical applications, 2) increased cooperation between different RR centres and user communities, and 3) promotion and development of specific applications of RRs.

2. WORK DONE DURING THE MEETING

The meeting was attended by 22 participants, from 15 Member States and two international organizations, namely JRC-EC and NEA-OECD. The meeting started off with welcome, opening and introductory remarks by the IAEA senior management representatives from the Physics Section and Nuclear Data Section, both from the Department of Nuclear Sciences and Applications, Division of Physical and Chemical Sciences. Later the welcome address was given by Mr D. Ridikas, the IAEA Scientific Secretary of the meeting. The self presentation of all meeting participants followed afterwards. Ms O. Gritzay (INR, Ukraine) was nominated as a chair person and Mr S. Oberstedt (JRC, EC) was appointed as a *rapporteur* of the meeting. Right after followed a brief presentation by Mr D. Ridikas, the IAEA Scientific Secretary, on specific objectives of the meeting within the ongoing IAEA project D2.01 on Enhancement of Utilization and Applications of Research Reactors.

2.1. Summaries of individual presentations

Later the meeting continued with individual presentations, which can be grouped into the following four main categories:

1. Capture cross section and decay data measurements using neutron beams, NAA, and other dedicated irradiation experiments

2. Fission reaction and fission fragment studies

3. Dedicated integral experiments for cross section and code validation

4. Efforts on code/library development and global benchmarking/validation

The subsequent brief statement on each contribution covers major contents only. The summaries are given according to the presentation order. Copies of all presentations, papers and administrative information were distributed at the end of the meeting to all participants and may be obtained from the Scientific Secretary on request.

O. Gritzay, Institute for Nuclear Research, National Academy of Sciences of Ukraine, presented some of the cross section measurements using the filtered neutron beam technique (FNBT), developed at Kyiv Research Reactor (KRR). She informed that available at KRR wide set of materials, especially high-pure isotopes, enables creation of neutron filters, providing more than ten neutron lines in the energy range from thermal energy to several hundred kilo-electron volts with intensity 10^6-10^8 n/cm^2 s, this allows getting of experimental data with accuracy up to 0.1%. She noted that development of NFBT on base of natural elements as main filter component may be useful for many research reactors.

M. Oshima, Japan Atomic Energy Agency, Japan, presented the results of the development of the new method of excited level analysis incorporating multi-step cascades performed at the JRR-3 Research Reactor. He also described the experimental technique and related data analysis methods for the prompt γ-ray measurements as well as the NAA.

M.-S. Kim, Korea Atomic Energy Research Institute, Korea, introduced the neutron cross-section measurements using a tangential beam tube of HANARO Research Reactor. He also presented the results of the experiments using the neutron transmission and time-of-flight method, the activation and gamma-ray detection method to confirm that the out-of-core neutron irradiation facility of HANARO would be a useful facility to measure the neutron induced reaction cross sections.

S. Hossain, Institute of Nuclear Science & Technology, Bangladesh Atomic Energy Commission, presented experiments based on the utilization of the radial piercing beam port for determination of neutron capture cross sections in thermal region using neutron activation analysis (NAA) technique at 3 MW TRIGA Mark-II. He showed that three experiments have successfully been carried out in determining the neutron capture cross section for the targets W, Ga and Sm using the reactions $^{186}W(n,\gamma)^{187}W$, $^{71}Ga(n,\gamma)^{72}Ga$, $^{151}Sm(n,\gamma)^{152}Sm$ and $^{153}Sm(n,\gamma)^{154}Sm$. The results at new thermal energy region will be useful to observe energy dependence of neutron capture cross sections.

S. Jonah, Centre for Energy Research and Training at Ahmadu Bello University, Nigeria, talked about investigations needed to extend the experimental procedures for the determination of neutron-induced cross section data at Miniature Neutron Source Reactors (MNSR). He also gave the first results of spectrum averaged cross section measurements of the (n,p) reactions on ^{27}Al, ^{28}Si, ^{29}Si, ^{47}Ti, ^{54}Fe, ^{58}Ni and (n,α) reactions on ^{27}Al and ^{30}Si. Additionally, thermal capture cross sections were presented for the target nuclides ^{47}Ca, ^{71}Ga, ^{75}As, ^{94}Zr and ^{238}U.

B. Nyarko, National Nuclear Research Institute, Ghana, talked about the current work on thermal, epithermal and fast neutron cross-section measurements for short-to-medium lived elements the GHARR-1 Research Reactor and Am-Be neutron source. The method of foil activation was implemented using $^{55}Mn(n,\gamma)^{56}Mn$ as a reference reaction. The experimental samples with and without a cadmium cover were irradiated in the isotropic neutron field of the neutron source and the outer irradiation sites of GHARR-1.

A. Letourneau, CEA Saclay, France, presented the measurements performed at the High Flux Reactor of the Laue-Langevin Institute, the experimental set-up, methodology and summary of the obtained cross sections. In particular he talked about the current results of the Mini-INCA project to measure thermal neutron-induced reactions on actinides. He highlighted the experiments performed using two irradiation channels: namely H9 and V4. Both standard NAA and on-line fission rate measurements were adapted for high neutron fluxes available at ILL: high-counting rate α and γ-spectroscopy, mass spectrometry for rare isotopes, and miniature fission-chambers for transmutation studies.

S. Oberstedt, Joint Research Centre (JRC), EU, introduced the recent detector suitability studies being carried out at the JRC-IRMM, which are dedicated to new and accurate measurements of fission γ-ray data in response to the OECD-NEA High Priority data Request List. He talked mainly about the use of recently developed cerium-doped lanthanum halide crystal scintillation detectors to distinguish more accurately, in terms of an improved time-of-flight resolution, between prompt fission γ-rays and neutrons.

O. Serot, CEA Cadarache, France, showed the results of the measurement performed at the High Flux Reactor of the Institute Laue-Langevin (ILL) in Grenoble. He highlighted two types of measurements: the first one concerns the binary fission yields which were measured on the Lohengrin mass spectrometer and the second concerns the ternary fission yields which were measured at the PF1 cold neutron guide installed at ILL. He noted that the results were compared with those obtained from spontaneous fission process which had been measured elsewhere but with the comparable experimental conditions.

K. Rasheed, Bhabha Atomic Research Centre (BARC), India, presented the details of the integral experiment with fast reactor materials carried out in the shielded facility of APSARA Research Reactor. The ratios 'C/E' obtained from various reaction rates provide valuable data for nuclear data validation studies, including neutron transport codes. He reported that from the activity of irradiated gold foils on the incident face of the model. It can be concluded that the neutron flux is fairly constant. This shows the adequacy of the size of the converter assemblies and efficacy of the collimator in ensuring a uniform flux of neutrons. "C/E" values of the experiments with cast iron model were also presented.

A. Kochetkov, SCK•CEN, Mol, Belgium, spoke on the current and future possibilities of measuring of nuclear data for minor actinides in integral and microscopic data experiments at the SCK•CEN Research Reactor installations. He said that the results for the ^{245}Cm isotope recently have been received with statistical uncertainty of the order of 0.9%. The measurements were performed with the well thermalized neutron beams in the large cavity of the graphite moderated BR-1 Research Reactor.

D. Bernard, CEA Cadarache, France, presented the experimental technique and the required calculation neutronic tools based on exact perturbation theory to measure actinide integral cross section of ^{232}Th, ^{233,234}U, ^{237}Np, ^{238,239,240,241,242}Pu, ^{241,243}Am, ^{244,245}Cm in variable neutron energy spectra. The measurements were performed at 50W pool reactor MINERVE located at CEA Cadarache.

B. El Bakkari, CNESTEN/CENM, Morocco, highlighted the neutronic analysis of the current core configuration of the 2-MW TRIGA MARK II Research Reactor in Morocco. Most of the model calculations were validated on the experimental results by experiment benchmarking program. The 3-D continuous energy Monte Carlo code MCNP (version 5) with various data libraries was used to develop a versatile and accurate full-core model of the TRIGA reactor. The model represents a very detailed description of all components of the core with literally no physical approximation.

Y. Mahlers, Institute for Nuclear Research, Ukraine, talked about validation of the ENDF/B-VII library for the WWR-M Research Reactor. He concluded that for all the experiments, neutronics calculation is in good agreement with the available measurements. The maximum absolute deviation of the effective multiplication factor from the results of the measurements is about 0.4% for ENDF/B-VII.0 and 0.6% for ENDF/B-VI.8.

O. Cabellos, Universidad Politécnica de Madrid, Spain, emphasized the impact of transmutation cross-section uncertainties on relevant fuel cycle parameters for a conceptual design of dedicated reactors for nuclear waste transmutation. He talked mainly about recent participation in the elaboration of a proposal entitled: "Accurate Nuclear Data for Nuclear Energy Sustainability" (ANDES), within EURATOM Call FP7-Fission-2009. One of the objectives of ANDES proposal is to improve the inventory codes (ACAB code) to handle the complete set of uncertainty/covariance data to illustrate the potential benefit of generalizing the assessment of simulation results with full uncertainties propagation.

R. Rosa, ENEA, Italy, illustrated the TAPIRO Research Reactor possibilities for reactor experiments with energies up to 1.35 MeV. The reactor is equipped with a homogeneous cylindrical core, SS cladding and a cylindrical copper reflector. All components assembled in a stainless steel tank, are placed inside a near spherical borated concrete shielding system. Channels of various dimensions and with different neutron spectra are distributed around the core and are available for the experiments, including nuclear data measurements.

A. Buijs, McMaster University, Canada, introduced the McMaster Nuclear Reactor (MNR): its design parameters, past and current use and possible future opportunities of this facility for nuclear cross section measurements. He told that at the moment different codes and data libraries may be validated via comparison to integral measurements of reactor properties. Necessary investment and developments were also discussed in order to prepare for a more dedicated scientific project related to the nuclear data measurements.

U. Köster, Institut Laue Langevin (ILL), France, talked about instruments for nuclear data measurements performed at 58 MW high flux reactor of ILL. He presented a number of representative examples and capabilities of existing instruments as the LOHENGRIN mass spectrometer, crystal spectrometer GAMS, and other instruments which use extracted neutron beams with neutron energies ranging from few neV to about 1 eV.

G. Zhigang, China Institute of Atomic Energy, China, presented the updated Chinese Evaluated Nuclear Data Library (CENDL-3.1), which is based on the nuclear data evaluation efforts in recent years, at China Nuclear Data Center (CNDC) in cooperation with China Nuclear Data Coordination Network. For most important nuclei of this library, the benchmark testing and validations have been performed including the comparison with other nuclear data libraries (ENDF, JENDL, BROND, JEFF, et al.). The beta version of CENDL-3.1 is CENDL-3.0 which has been provided for China domestic users at the moment but with the external release of the library planned in the very near future.

M. Pescarini, ENEA, Italy, highlighted the generation of broad-group working cross section libraries for nuclear fission reactor shielding and radiation damage applications in different spectral environments (LWR, SFR, LFR, HTR, etc.), which will allow the use of the three-dimensional (3D) deterministic transport codes in the data validation activities. He emphasized that it is useful not only for the reactor physicist and nuclear engineer data validation interests but also for the nuclear safety authorities and the industrial partners working in the nuclear energy field.

Y. Rugama (NEA-OECD) and V. Pronyaev (NAPC-IAEA) also gave the overview presentations on the ongoing activities relevant to nuclear data and evaluations in their respective organizations.

2.2. Results obtained

All presentations were followed by adequate time for discussions/questions, widely used by the participants and chairs. Further, intermediate summaries and compilations of findings and comments contributed to involving participants into the aims of the meeting and the strengthening of the exchange of knowledge and experience. The following text resumes the outcome of the discussions, observations and intermediate conclusions relevant to the four main topics indicated in the previous section.

2.2.1. *Topic 1: Capture cross section and decay data measurements using neutron beam, NAA, and other dedicated irradiation experiments*

Preparation of physics samples

A crucial point in the measurement of cross-sections is the availability of high quality samples. The meeting participants proposed to establish dedicated databases of:

- Laboratories which have a stock of sample materials that is potentially useful for target preparation (enriched stable and radioactive isotopes, high purity metals and special compounds, etc.)

- Laboratories which have the capability to transform the material into targets useful for nuclear data experiments, i.e. radiochemical purification, physical vapor deposition (evaporation, sputtering), electrodeposition (electrolysis, molecular plating), painting or spraypainting, ion implantation, etc.

- Laboratories which have the capability to characterize the quality of targets (quantification of purity, target thickness, homogeneity, stoichiometry, etc.) by various methods (low level nuclear spectrometry, Rutherford backscattering, etc.).

The above data bases could be grouped according to the capability to handle i) only stable, ii) slightly radioactive or iii) strongly radioactive materials. Already now the existing link to the International Nuclear Target Development Society (http://www.intds.org) should be used and common efforts with this Society should be coordinated.

The participants also recommended establishing a checklist or guidelines how to bring various radioactive or even fissile materials from one laboratory to another for physics experiments. In this regard, a short description of the steps to follow, including the links to access the latest procedures of the specific transport regulations (ADR for road transport, IATA for air transport, and equivalent for railway and marine transport) would be very useful.

Neutron beam design, characterization and nuclear data experiments

The contributions of the meeting gave evidence of a great variety of neutron beams, filtering devices and energy selectors. Therefore,

- A collection of guidelines dedicated to the design of neutron beams, materials of filters, energy selectors and beam characterization methods should be of great advantage for ongoing experiments and newly planned projects. Commonly based methods and share of information would facilitate the experimental data inter-comparison, interpretation and complimentary measurements to be performed.

- The participants enquired to establish the priority list of planned experiments/reactions to be studied using neutron beam techniques, based on information collected from national laboratories including universities, and exchange this information among interested parties

- A dedicated Technical Meeting or Workshop organized by the Agency on this topic is strongly recommended for a preparation of a specific TECDOC

Networking of new techniques

The following networking activities were recommended by the participants:

- Use of Accelerator Mass Spectrometry (AMS) for measuring cross-sections for long-lived reaction products. Establish a list of AMS facilities with the elements they are used to handle.
- Use of monochromatic neutron beams with tunable energy by neutron diffraction on monochromator crystals or with a neutron velocity selector. List of facilities that have such monochromatic neutron beams with their capabilities (neutron energy range, beam size and available flux at sample position) would be very helpful.
- Use of neutron interferometers for measurement of coherent scattering length (and more). Further actions could be overtaken after consulting Mr Helmuth Rauch (AtomInstitut, Austria) and Mr Henry Fischer (ILL, France)
- Use of neutron beams in combination with the high resolution multi γ-ray spectroscopy**.** Indeed, 4π geometry γ-ray spectrometer gives complete cascade information, contributing high accuracy neutron capture measurements.

Again, a dedicated Technical Meeting or Workshop organized by the Agency on the above topic was strongly recommended for a preparation of a specific TECDOC.

2.2.2. Topic 2: Fission reaction and fission fragment studies

The following sub-topics were highlighted during discussions:

a. Prompt fission neutron energy spectra
   i. Reactor calculation on the bases of new prompt neutron spectra measured at thermal energies on ^{235}U in 2008 for three different angles relative to the incident neutron beam (recently evaluated by Mr V. Maslov)
   ii. New measurements of prompt fission neutron spectra are encouraged concentrating on the energy region below 0.5 MeV, which represent the threshold of traditionally employed liquid scintillation detectors. Those measurements should be carried out relative to ^{252}Cf ideally under exactly the same experimental conditions.
   iii. Dedicated experimental techniques should be developed which allows measuring the entire region from low energies to beyond the distribution maximum, e. g. 1.5 MeV, to avoid systematic errors due to normalisation.
   iv. (Velocity-) Filtered beam measurements might be useful to investigate the energy dependence of the spectral shape and fluctuations in the multiplicity due to the influence of the spin of the first resonance, e. g. ^{239}Pu
   v. Reduction of uncertainty at high spectral energies very much recommended
   vi. A better knowledge of the spectral shape at both very low and very high neutron energies will help to select proper neutron emission models

vii. Measurements of prompt neutron spectra and multiplicity at fast-reactor neutron beams

b. Prompt fission gamma-ray spectrum and multiplicity
   i. Measurements of prompt fission gamma-ray spectra and multiplicity in thermal neutron energy range
   ii. Measurements of prompt fission gamma-ray spectra and multiplicity in fast neutron energy range

c. Fission fragment characteristics for the description of neutron and gamma-ray emission by means of Monte-Carlo evaporation codes
   i. Pre-neutron mass and kinetic energy distributions with high mass resolution
   ii. Fission fragment angular distribution as a function of fission-fragment mass
   iii. Spin distribution of primary fission fragments to describe properly the competition of prompt neutron and gamma-ray emission
   iv. Measurements in fast neutron energy range, because no data exist at all

d. Independent and cumulated fission yields for fast-reactor applications
   i. Measurement at fast fission RR beams
   ii. Design of a fast neutron spectrum by means of uranium converters placed in thermal beams at RR (e.g., BR-1 in Belgium)
   iii. Installation of a fission-fragment spectrometer

e. Fission cross-section measurements at thermal energies
   i. Isotopes accessible via (2n,f) relevant for model normalization at thermal energies (^{232}Pa, ^{238}Np, …)
   ii. Fission cross-section measurements at thermal RR including measurements with (velocity-)filtered beams to test the 1/v dependence of the cross-section (Westcott-factor), e. g. ^{245}Cm complementary to previous measurements at a pure thermal beam (BR-1) and at the neutron TOF facility GELINA

In brief, the following remarks resume the above discussion and observations:

- Need to solve the problems related to the prompt fission neutron energy spectrum
- Need for new measurements of prompt gamma-ray spectra relevant for Gen-IV
- Development of new measurement facilities and techniques for fission fragment measurements, in particular in the fast neutron energy region

2.2.3. Topic 3: Dedicated integral experiments for cross section and code validation

Three main subjects were discussed within this topic: a) the methodology sharing, b) the experimental uncertainty assessment and c) the issues related to neutron spectrum unfolding codes.

a) The methodology sharing. Both reactor technology and neutronic calculations are concerned. First of all, facilities should participate in the activities around the OECD IRPhE database. The devices used such as fission chambers, foils, detectors should be available and easily transportable from one facility to another in order to distinguish the statistical and systematic uncertainties of a measured value. VENUS and TAPIRO facilities are open for such collaborations in 2010. Work studies on pile oscillation technique to assess integral nuclear data measurements should be encouraged, namely for cross-section, β_{eff}, Doppler and other parameter measurements in the fast neutron range. The calculation tools, such as ERANOS, and its know-how (training session, summer schools…) should be widely shared and promoted.

b) The experimental uncertainty assessment. Indeed, two components of experimental uncertainty should be provided to perform the validation of the nuclear data. A systematic and statistical covariance matrix should be given by the experimentalists. Experimental benchmarking (such as using the same flux monitoring in different facility) is encouraged (see above bullet (a)).

c) The issues related to neutron spectrum unfolding codes. A dedicated workshop or training course on this topic is requested, in particular for application and use of the SAND code, applied in NAA.

2.2.4. Topic 4: Efforts on code/library development and global benchmarking/validation

The discussion within this topic aimed to define the importance of basic data, codes and integral experiments for model development, evaluation/validation and error propagation issues.

Library development

The compilation and evaluation of nuclear data libraries requires complex and expensive activities. These efforts are justified on the basis of the real benefits to the end users of the data. In this respect, high-quality nuclear data, in particular complete and accurate information, is an essential input required for cost-effective design and safety assessment of present and future nuclear systems, including power reactors.

The differential nuclear measurements in combination with nuclear models are the main source of information of nuclear data evaluators. Those experimental measurements should be preferably reported and included in the EXFOR library. The submitted experimental data have to include details on sources of uncertainty and their possible correlation to improve the assessment of the systematic uncertainties.

The early dissemination of the beta-releases of the ENDF/B data libraries to the potential End-Users (nuclear industry, research centres, universities, etc.) and resulting feedback will permit the possible identification of mistakes for the evaluators.

Processing libraries

The *Quality and Assurance* (Q&A) of processing ENDF/B data libraries is an important task linking evaluators and End-Users. A free-processing procedure including patches of processing codes, reporting bugs and updates when a new ENDF/B file is delivered certainly contributes to this Q&A process. Here, one reviews how the different steps in the data processing phase can support this objective:

i) ENDF Utility Codes (e.g. CHECKR, …) are and should be used as pre-processing codes to guarantee format or physics checking of ENDF/B files. However, it can be noted that additional mistakes can be found in the next code-library processing step.

ii) Processing ENDF/B files (e.g., NJOY, PREPRO, SCAMPI, AMPX…) in PENDF/GENDF libraries is a valuable procedure to compare different ENDF/B files, as well as, to check the algorithms used in those processing codes. Here, the messages (warnings/errors) of these processing codes as well as "the human eye" can be used as to spot inconsistency and error in the ENDF/B files. With those processed PENDF files, INTER code can be used (e.g. spectrum averaged cross sections) to compare with experimental "standard" measurements (e.g. measurements in the Maxwellian averaged neutron spectra).

iii) Besides the standard code-library processing libraries (e.g., ACE, WIMS, MATXS, AMPX, …), the production of fine-group general-purpose cross section libraries (FGENDF) is recommended. These FGENDF files can be used to collapse problem dependent broad-group coupled neutron and photon working cross-section libraries for nuclear rector shielding and radiation damage applications with self-shielding neutron cross sections. These "modern" cross section working libraries for various spectral environments of interest (Gen. III and Gen IV reactors) would permit the use of the 3D deterministic transport codes in the nuclear data validation, with a satisfactory treatment of the neutron leakage.

iv) Finally, processing of decay data (e.g. branching ratios, ...) is also an important task to identify possible mistakes in burn-up calculations.

Global Benchmarking and Validation Experiments

The assessment of the present accuracy of simulation tools and nuclear data can be provided by comparing with well-designed integral experiments. This assessment is based on:

- great number of the integral experiments,
- combination of different simulation tools (deterministic and Monte Carlo) and associated nuclear data.

Well-established international databases containing a comprehensive set of experiments are available for sharing among the specialists. These databases are used for international activities involving validation efforts and for test basic nuclear data evaluations to:

- build confidence in methods and data evaluations,
- assess uncertainties
- define confidence bounds and safety margin

Examples of such experiments can be found in

i) The *International Handbook of Evaluated Criticality Safety Benchmark Experiments* (ICSBEP)

ii) The *Shielding Integral Benchmark Experiment Data Base* (SINBAD)

iii) The *International Reactor Physics Benchmark Experiments Project* (IRPhe)

iv) *Validation Suites*, collection of a number of selected Benchmarks, taken from those Databases (as proposed by R.D. Mosteller- LANL based on ICSBEP) could provide a *general indication* of the overall performance of improvements in computing tools and nuclear data libraries.

The analysis of these and newly planned integral experiments will certainly be useful in assessing the need of new data measurements or evaluations to reach the required accuracy.

Error propagation

At present, the assessment of the accuracy of parameters related to the core performance (e.g, criticality value, …), and fuel cycle parameters (e.g., evolution/transmutation of fuel

inventory, …) due to the uncertainties in the basic nuclear data is a critical issue. This accuracy of calculations can be performed with different *Forward Error Propagation* techniques (ASAP- *Adjoint Sensitivity Analysis Procedure* /FSAP - *Forward Sensitivity Analysis Procedure* /GSAP - *Generalized Sensitivity Analysis Procedure* and/or Monte Carlo) for the systematic introduction of the uncertainty propagation in the simulations.
ENDF/B covariance files (variance/correlations) are needed to perform this uncertainty evaluation. Consequently, an additional effort for further provision of covariance data is also recognized with methodologies to evaluate the uncertainties and covariance matrices from experimental data and nuclear models. In addition, the inclusion of cross-correlations for various reactions-isotopes cannot be avoided and is strongly recommended.

3. SUMMARY OF THE MAIN RECOMMENDATIONS

The TM was highlighted as a success by all participants at the end of the week. Furthermore, the support for the meeting in terms of number and diversity of participants as well as participating Member States, is significant indicator of the success of the broader endeavor – to provide timely practical assistance and support the sharing of Research Reactor (RR) based experience related to the provision of nuclear data, fostering the development of new experimental techniques, establishment of enlarged collaborations, facilitated contacts and formation of networks in nuclear data and evaluation related activities.

It was recognized that the IAEA has undertaken a number of activities through Coordinated Research Projects, Technical Meetings and Workshops, and in some cases also through the Technical Cooperation (TC) projects to assist the Member States in the domain of nuclear data measurements and evaluations. Continuation and expansion, where appropriate, of such activities was desired and encouraged.

Based on the final discussions on current status and future needs in the nuclear data measurements and evaluations using RRs, the participants formulated **the following specific recommendations:**

1. **Finalization and publication of this TM report, including individual papers, as an IAEA TECDOC, titled "Specific Application of RRs in Provision of Nuclear Data"**

2. **Create data base/network of RR facilities with their detailed characteristics performing nuclear data measurements**

3. **Create data base/network of high quality samples/targets used in nuclear data measurements and facilitate/support their share/exchange/transportation**

4. **Preparation of the specific IAEA TECDOC/Guidelines on the potential use of advanced neutron beam techniques and associated detection-instrumentation-Data ACQ systems in the field of nuclear data measurements**

5. **Use of advanced neutron beam techniques and associated detection systems has a huge potential to provide high quality nuclear reaction and decay data. The design and implementation of a dedicated CRP is recommended on the**

development and use of neutron beam based advanced experimental techniques and associated detection systems for the provision of nuclear data.

6. **Encourage and support experimental efforts on**

    a. **measurements of capture cross sections and decay data relevant to production of certain radioisotopes**

    b. **measurements of important inelastic cross sections, e.g. ^{57}Fe, ^{238}U, …**

    c. **measurements of prompt neutron-photon multiplicities and energy spectra for ^{235}U and ^{239}Pu both with thermal and fast incident neutrons**

    d. **fission fragment yield measurements for ^{233,235}U, ^{239,241}Pu and also for ^{232}Th, ^{238}U with fast incident neutrons**

    e. **fission cross section measurements for ^{232}Pa and ^{238}Np in thermal neutron region, and also for other actinides (e.g. ^{245}Cm) at well defined incident neutron energies**

    f. **dedicated integral experiments for reaction rate and cross section validation in well defined and variable neutron energy spectra**

7. **Encourage experimental and evaluation efforts on establishment and/or improvement of covariance matrices relevant to reaction cross sections, propagation of associated uncertainties (reaction rates and decay data) in particular in material evolution/transmutation calculations.**

ANNEX I. BOOK OF ABSTRACTS

3.1. O. Gritzay, INR, Ukraine

Precision Neutron Cross Section Measurements at Reactor Neutron Filtered Beams

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Up-to-date level of scientific and technology development demands the high accuracy of neutron data. This high accuracy of experimental data should allow making progress in development of nuclear simulation codes and in generation of evaluated nuclear data libraries, that are the basis for any transport calculations both for operating and future reactors and for nuclear technologies in medicine, industry, etc.

Now the most difficult situation with the accuracy of neutron cross sections is in the energy range from several keV to several hundred keV. The reason is in the lack of high flux installations in the mentioned energy range. Today a few of such installations are designed and may be used (e.g. in LANL, ORNL), but they are very expensive. There is another way to get the high accuracy neutron cross sections: to use the neutron filtered beam technique at the existing research reactors.

At Kyiv Research Reactor (KRR) the neutron filtered beam technique is used for more than 30 years and its development continues. Now we can measure total neutron cross sections with accuracy 1% and better, neutron scattering cross sections with 3-6% accuracy.

The wide set of natural elements and enriched isotopes are used as components for neutron filters in the Neutron Physics Department (NPD) at KRR:

- Natural elements: Si, Al, V, Sc, S, Mn, Fe, Ti, Mg, Co, Ce, Cr, Rh, Cu, B, Cd, LiF.
- Enriched isotopes: ^{52}Cr (99.3%), ^{54}Fe (99.92%), ^{56}Fe (99.5%), ^{57}Fe (99.1%), ^{58}Ni (99.3%), ^{60}Ni (92.8% – 99.8%), ^{62}Ni (98.04%), ^{80}Se (99.2%), ^{10}B (85%), ^{7}Li (90%).

Availability of such wide set of materials, especially enriched isotopes, allowed to create in the NPD the unique set of neutron filters, providing more than ten neutron lines in the energy range from thermal energy to several hundred kilo-electron-volts, intensity of such lines may reach 10^6 – 10^8 n/cm^{2}s, and this is much more than any other method (time of flight or others) can ensure.

The main purpose of this report is presentation of the neutron measurement techniques, developed at KRR, and demonstration some experimental results, obtained using these techniques.

3.2. M. Oshima, JAEA, Japan

Neutron capture cross-section measurements by high-resolution γ-ray spectroscopy

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Accurate neutron reaction cross-sections of long-lived fission products and minor actinides are required for design of innovative reactor systems including fast breeder reactors and accelerator driven systems. However, accuracy of the experimental data is not sufficient for this request at present. The experimental methods are roughly categorized to neutron activation method and prompt γ-ray spectroscopic one. The former method has a merit of high sensitivity and is used widely, while it is limited to the case of unstable daughter nuclei. On the contrary, the prompt γ-ray spectroscopic method can be applied to all nuclei because all the compound nuclei emit prompt γ-rays. It also covers a wide energy range by combining it with a neutron time-of-flight (TOF) method. Another merit is that, by utilizing high resolution germanium (Ge) detectors, we can distinguish γ-rays from individual nucleus. Even when a sample includes large amount of impurities, one can extract cross sections accurately without their contributions. It is possible to derive neutron capture cross sections by summing all the intensities of ground-state and/or primary transitions[1,2]. In this method one need to identify these transitions based on precise level scheme.

The method of "two-step cascade"[3] has succeeded in constructing a level scheme after neutron capture reactions. We have been developing the new method of excited level analysis incorporating multi-step cascades.[4,5] Multi-step cascades from the ^{62}Ni(n_{cold}, γ) ^{63}Ni reaction were studied via a γ-ray spectroscopy method. With the γ-ray detector array, STELLA, which consists of high-resolution Ge detectors and is installed at the Research Reactor (JRR-3) at Japan Atomic Energy Agency, multiple γ-ray coincident events were accumulated. We could determine the level scheme of ^{63}Ni up to 5.6 MeV. On the basis of the assignment of the ground-state transitions, we could derive the thermal-neutron capture cross section of ^{63}Ni. In this report we describe the experimental and analysis methods for the prompt γ-ray analysis method as well as the activation one in JRR-3.

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3.3. M.-S. Kim, KAERI, Korea

The Out-of-core Neutron Irradiation Facility of HANARO for Measurement of Neutron Cross-section

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The neutron cross-section measurements using a tangential beam tube of HANARO are introduced in this work. In HANARO, a 30 MW research reactor, an out-of-core thermal neutron irradiation facility was designed and constructed using a long beam tube and a fast neutron and gamma-ray filter consisting of thick silicon and bismuth single crystals. It gives a thermal neutron field with a flux over 1×10^9 n/cm^{2}s and a Cd ratio over 150. The shielded irradiation room is spacious enough for various experiments such as studies in the boron neutron capture therapy (BNCT), dynamic neutron radiography, etc. So far, two collimators with neutron beam diameters of 10 and 15 cm can be used. One of the applications of this facility is the measurement of the thermal neutron cross-sections for several nuclides.

In the design stage of the facility, the total neutron cross-sections of the silicon and bismuth crystals were measured by using the neutron transmission and time-of-flight method since the cross-sections of the filter materials were the key parameters for the performance of the facility. For the silicon single crystal, the measured total neutron cross-section showed the good agreement with the calculated value based on the semi-empirical formula. The measured cross-section of the bismuth crystal was much larger than the calculated one. From the result, it was confirmed that several parameters related to the Bragg scattering should be considered in the neutron transmission through the bismuth single crystal.

On the other hand, the thermal neutron capture cross section for the W-180 nucleus was measured by using the activation and gamma-ray detection method. A natural tungsten foil was irradiated in this facility for 5 hours, and finally the capture cross-section was obtained to be a value of 22.6±1.7 b. The measured data for the cross-section for W-180 is very rare, but the production of the W-181 is very important since it is a useful radioisotope neutrino source for the various basic neutrino experiments.

From above results, it was confirmed that the out-of-core neutron irradiation facility of HANARO would be a useful facility to measure the neutron cross-sections.

3.4. S. M. Hossain, BAEC, Bangladesh

Experimental Determination of Neutron Capture Cross Sections at a Rare Thermal Energy Using the BAEC TRIGA Reactor

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Over the past twenty three years, the only research reactor of the country 3 MW TRIGA Mark-II has progressed a variety of tasks in Bangladesh under Bangladesh Atomic Energy Commission (BAEC). These have included isotope production, material research using neutron scattering, materials characterization by neutron radiography, qualitative and quantitative assessment of elements in variety of sample matrices using neutron activation analysis as well as training as centers of excellence in Science & Technology. Since the installation of Triple Axis Spectrometer (TAS) in the radial piercing beam port of the reactor, it has been utilized for material research using neutron scattering technique. Recently, we have opened a new arena by utilizing the radial piercing beam port for determination of neutron capture cross sections at a rare energy in thermal region using neutron activation analysis (NAA) technique. Three experiments have successfully been carried out in determining the neutron capture cross section for the reactions $^{186}W(n,\gamma)^{187}W$, $^{71}Ga(n,\gamma)^{72}Ga$, $^{152}Sm(n,\gamma)^{153}Sm$ and $^{154}Sm(n,\gamma)^{155}Sm$ at the thermal energy of 0.0536 eV. Recent extensive literature review insisted us to claim that there are no experimental neutron capture cross-section data available at our investigated energy. So far, we carried out experiments at 0.0536 eV neutrons for cross section measurements for the first time. The results Published in the reputed journals are a testimony of our claim. Our experimental results were critically compared with the evaluated data quoted in JENDL-3.3 and ENDF/B-VII. The results at new thermal energy region will be useful to observe energy dependence of neutron capture cross section.

3.5. S.A. Jonah, CERT, Nigeria

The Use of Miniature Neutron Source Reactor Facility for the Determination of Neutron-induced Cross section Data

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Abstract

Miniature Neutron Source Reactors are compact low-power nuclear research reactor designed mainly for neutron activation analysis and limited radioisotope production. These facilities have stable neutron flux distribution as a function of operating time as well as a high neutron flux –to-power ratio. Because of proximity of the irradiation channels to the core, the neutron spectral distribution is made up of fast, thermal and epithermal components and therefore careful investigations are needed to evolve experimental procedures for the determination of neutron-induced cross section data. Specifically using an inner irradiation channel, a comparator method relative to the resonance integral of $^{197}Au(n,\gamma)^{198}Au$ reaction was found to be appropriate for the determination of reactor neutron spectrum averaged cross section data for some low and medium mass nuclei. Furthermore, thermal neutron capture cross section data were determined on the basis of Cadmium ratio measured in one of the outer irradiation channel. Results of spectrum averaged cross sections are presented for the (n,p) reaction on ^{27}Al, ^{28}Si, ^{29}Si, ^{47}Ti, ^{54}Fe, ^{58}Ni and (n,α) reactions on ^{27}Al and ^{30}Si. Additionally, thermal capture cross sections are presented for ^{48}Ca, ^{72}Ga, ^{76}As, ^{95}Zr and ^{239}U

3.6. B.J.B. Nyarko, GAEC, Ghana

Cross Section Determination of Short-to-Medium Lived Nuclides in a Low Power Research Reactor and Am-Be Neutron Source

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Thermal and epithermal neutron cross-sections for the ^{197}Au (n, γ) ^{198}Au, ^{75}As (n, γ) ^{76}As, ^{27}Al(n,γ)^{28}Al, ^{51}V(n,γ)^{52}V, ^{127}I(n,γ)^{128}I, ^{154}Sm(n,γ)^{155}Sm and ^{238}U(n,γ)^{239}U reactions were determined by the method of foil activation using ^{55}Mn (n, γ) ^{56}Mn as a reference reaction. The experimental samples with and without a cadmium cover of 1-mm wall thickness were irradiated in the isotropic neutron field of the ^{241}Am-Be neutron source and the outer irradiation sites 7 of Ghana Research Reactor-1 facility whose neutron spectrum shaping factor (α) was found to be (0.037 ± 0.001). The induced activities in the sample were measured by gamma ray spectrometry with a high purity germanium detector. The necessary correction for gamma attenuation, thermal neutrons and resonance neutron self-shielding effects were taken into account during the experimental analysis. By defining cadmium cut-off energy of 0.55eV, samples in the form of wires of negligible thickness and standard solutions of concentration 10ppm were irradiated in both irradiation channels of the ^{241}Am-Be neutron source and the miniature neutron source reactor operating at 3kW to determine the thermal and epithermal neutron cross-sections for the (n, γ) reactions. The samples irradiated in GHARR-1 gave the following results for gold (Au): thermal neutron cross section σ_o = (97.47 ± 2.64) b and resonance integral I_o= (1549.00 ± 1.74) b, and for Arsenic (As): thermal neutron cross section σ_o = (4.28 ± 0.19) b and resonance integral I_o = (61.88 ± 1.07) b. For samples irradiated by the ^{241}Am-Be neutron source, the results obtained for thermal cross-section were: σ_o = (0.198±0.045) b for ^{27}Al(n, γ)^{28}Al, σ_o = (4.626±1.235) b for ^{127}I(n, γ)^{128}I, σ_o = (7.058± 0.984) b for ^{154}Sm(n, γ)^{155}Sm, σ_o = (3.107±0.682) b for ^{238}U(n, γ)^{239}U and σ_o = (5.662±1.025) b for ^{51}V(n, γ)^{52}V. This values obtained compare very well with those found in literature. The work is in progress and we are determining thermal, epithermal and fast neutron cross-sections for short-to-medium lived elements in GHARR-1 and Am-Be neutron source.

3.7. A. Letourneau, CEA Saclay, France

Cross section measurements for thermal neutron-induced reaction on actinides at the ILL reactor

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Within the Mini-INCA project, we measured the thermal neutron-induced capture cross section of ^{232}Th, ^{233}Pa, ^{234}U, ^{237}Np, ^{238}Pu, ^{242}Pu, ^{241}Am, $^{242gs\text{-}m}$Am, ^{243}Am, ^{242}Cm, ^{244}Cm, ^{248}Cm, ^{249}Cf, ^{250}Cf, ^{251}Cf and the neutron-induced fission cross sections of ^{238}Np, $^{242gs\text{-}m}$Am and ^{245}Cm.

Experiments were done using two irradiations channels of the High Flux Reactor of ILL: the H9 and the V4 channels. These two channels offer a diversity of neutron fluxes ranging from pure thermal neutrons to about 15% epithermal neutrons with intensities as high as 1 10^{15} n/cm^2/s.

To determine these cross sections we used activation techniques bases on high-counting rate α and γ-spectroscopy of irradiated samples or mass spectrometry and on-line fission rate measurements with miniature fission-chambers. Thanks to the high neutron fluxes and such techniques it is possible to form short-live isotopes (such as ^{242gs}Am or ^{244m}Am) and to study their decay or their neutron capture or fission cross sections. Fission chambers were also used in the MEGAPIE experiment, the first 1 MW liquid Pb-Bi spallation target, to measure the integral fission cross sections of $^{242gs\text{-}m}$Am and the ^{241}Am branching ratio in the resonance region.

In this paper we will present the measurements performed at ILL, the experimental set-up and techniques and the obtained cross sections.

3.8. S. Oberstedt, IRMM-JRC, EU

Prompt γ-ray emission in nuclear fission

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A particular challenge for the modelling of new generation reactor neutron kinetics is the calculation of the γ-heat deposition e. g. in steel and ceramics reflectors without UO_2 blankets, which is required to be known with an uncertainty as low as 7.5%. The comparison of various benchmark experiments with calculated γ-heating shows a systematic underestimate ranging from 10 to 28% for the main fuel isotopes ^{235}U and ^{239}Pu. This is attributed to deficiencies in γ-ray production data in evaluated nuclear data files [1]. Data found in modern nuclear-data libraries all date back to experiments performed in the early 1970's [2,3,4]. In those experiments NaI scintillation detectors were used as γ-ray spectrometer with an ionisation chamber as fission trigger. In a recent modelling exercise of neutron emission from fission fragments by means of a Monte-Carlo approach [5] the authors achieve a reasonably good description of the average γ-energy released in fission, but they are unable to reproduce the experimentally obtained dependence as a function of the fission-fragment mass. They recommend putting more work in clarifying the competition between neutron and γ-ray emission during fission-fragment de-excitation.

A major difficulty in measuring the competition between neutron and γ-ray emission during fission fragment de-excitation is the suppression of background γ-rays induced by prompt fission neutrons in the γ-detector. A common method is to distinguish between γ-rays and neutrons by their respective different time-of-flight, which however is limited by the timing resolution of the detector (not better than 5 ns for NaI). A promising approach seems to be the use of recently developed cerium-doped lanthanum halide crystal scintillation detectors. We will present results of the characterization of coaxial 1.5” x 1.5” $LaCl_3$:Ce detectors in terms of energy conversion and resolution, linearity, intrinsic efficiency, timing resolution and intrinsic radioactivity in the relevant dynamical range [6]. Additionally, we will discuss a first experiment on fission γ-ray emission at thermal neutron energies using a novel ultra-fast fission event trigger.

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3.9. O. Serot, CEA Cadarashe, France

Binary and Ternary Fission Yield Measurements at the Institut Laue-Langevin

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In spite of the huge amount of fission yield data available in the different evaluated nuclear data libraries, such as JEFF3.1, ENDF/B-VII, JENDL3.3 and others, more accurate data are still needed for both nuclear energy applications and for understanding of the fission process itself. Hence various campaigns of measurements of fission yields were performed at the High Flux Reactor of the Institut Laue-Langevin (ILL) in Grenoble, France.

Two types of measurements were carried out as follow:

- Binary fission yields were measured on the Lohengrin mass spectrometer. Up to now, this mass spectrometer coupled to a high resolution ionization chamber has been used to investigate the mass and isotopic yields of the light mass region for fissioning nuclei from Th to Cf. To complete these measurements, the heavy mass region for the reactions $^{235}U(n_{th},f)$, $^{239}Pu(n_{th},f)$ and $^{241}Pu(n_{th},f)$ have been investigated. For these higher masses, an isotopic separation is no longer possible, so a new experimental method based on gamma spectroscopy was undertaken with the reaction $^{239}Pu(n_{th},f)$ to determine the isotopic yields. These experiments have permitted to reduce considerably the uncertainties.
- Ternary fission yields were measured at the PF1b cold neutron guide installed at ILL. In the frame of a systematic investigation of ^{4}He, ^{6}He and ^{3}H ternary particles, various neutron induced fission reactions were measured covering target nuclei between ^{229}Th and ^{251}Cf. A ΔE-E telescope was used to identify ternary particles and determine both their energy distribution and their emission probability. These results are compared with those obtained from spontaneous fission decays which were measured elsewhere within the same experimental conditions.

3.10. K.K. Rasheed, BARC, India

Fast Reactor Integral Experiments at BARC for Cross-section Evaluation

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Neutron transport through single shield materials were experimentally studied and compared with theoretical calculations so that bias factors (C/E) could be obtained for large size 1000 MWe fast reactors. The shield materials studied were: Carbon steel, SS-316, Graphite, Boron carbide, Sodium, Borated graphite, Nickel and Cast iron. With the experimental data of neutron transmission through different materials and thicknesses, the computational capabilities of Codes with various libraries could be assessed with respect to anisotropy, self-shielding, group structure etc. The experiments were carried out in the Shielding corner of Apsara reactor, BARC, India. The leakage thermal spectrum of Apsara was converted to a fast reactor leakage spectrum by using 0.65 % U-235 depleted Uranium Converter Assemblies (CA).The space between Apsara reactor core and the shielding corner being light water, the neutron flux available for the shielding experiments was attenuated by a factor of 10^{-3} to about 10^{7}n/cm^{2}.s. The flux was increased to 10^{10}n/cm^{2}.s by displacing the water between the core edge and the stainless steel lining of Apsara pool on the shielding corner side by an air-filled aluminium box. The scattered neutron contribution from the surroundings was prevented by use of a collimator on the incident face and surrounding the model by borated concrete blocks of 25 cm thickness on all sides except front and back.

Theoretical calculations of the experimental configurations were carried out using 2D Transport theory Code DORT in X-Y geometry with S8P3 approximation. This Code is based on the Sn method of solving transport equation. The 100 group multi-group neutron cross-section Library DLC-2 (based on ENDF B-IV) was used. A large number of activation detectors (Gold, Copper, Sodium, Indium and Sulfur) and solid state neutron track detectors (SSNTD) like Thorium and Neptunium were used for obtaining the reaction rates which covered the entire energy range from thermal to fast. Bare and cadmium covered detectors were used for separating the epithermal component from the thermal. Threshold detectors were used for the fast range. The measured reaction rates were (E) were then compared with the calculated values (C) and 'C/E' values obtained for the reaction rates as well as the attenuation along the model length.

3.11. A. Kochetkov, SCK*CEN, Belgium

Measurements of Nuclear Data for MA at VENUS-F and BR-1 reactors

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One of the declared advantages of the perspective systems as ADS and Gen.IV fast reactors is the transmutation of the minor actinides (MA) such as neptunium, americium and curium. From the viewpoint of reactor neutronic design, the loading of MA in the core generally affects the physics parameters. The detailed core designs of these advanced designs are difficult so as the reliability of nuclear data of MA, especially for the leads cooled facilities, is not sufficient.

In SCK•CEN (Mol, Belgium) the GUINEVERE project has been lunched. The project consists in coupling a subcritical zero power fast lead core, at the VENUS-F reactor, with a GENEPI external 14 MeV neutron source has to be operated in pulsed and in continuous mode**.** This project aims to investigate the feasibility of ADS on the steps of loading and reactivity monitoring of the subcritical and critical cores. So, as an experimental program dedicated to a critical operation of heavy liquid metal cooled fast reactors used both for the LMFR development and the critical mode operation of ADS has been foreseen. MA fission rates cross sections ratios so as spectral indexes will be measured at 30% enriched uranium lead VENUS-F benchmark core, by means of fission chambers and foils. The measurements will be accompanied with deterministic (ERANOS) and MCNP5 calculations.

The measurements of the thermal neutron-induced fission cross section of a number of Cm isotopes was set up at SCK•CEN in the well thermalized neutron beams provided by the Maxwellian Thermal Spectrum Reference Field in the large cavity of the graphite moderated BR1 reactor. The results for the Cm-245 isotope recently have been received with statistical uncertainty of the order of 0.9%. These results were presented at FISSION 2009 Conference in May. The work is on the way.

3.12.

D. Bernard, CEA Cadarache, France

Integral Experiments in MINERVE Reactor Facility for Nuclear Data Validation

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The oscillation technique in the 50W pool reactor MINERVE located at Cadarache consists in maintaining the reactor criticality while small doped samples (10cm height) are inserted in.

The reactor is supposed critical. A boron chamber indicates the on-line local thermal neutron flux variation during the sample introduction at the center of the reactor (5sec). This electronic signal involves rapidly (1μs) a mechanical movement to a small "automatic" pilot rod which total reactivity worth is about 10pcm (10^{-4} dk/k). The on-line acquisition is performed after reactivity transients when the criticality is observed: the local neutron flux monitored by the boron chamber is then constant.

The pilot rod is made of small glued cadmium sectors on a rotor and a stator. The neutronic shadow effect between the rotor and the stator is used to maintain criticality. The experimental signal of the integral cross section is the angle between the rotor and the stator.

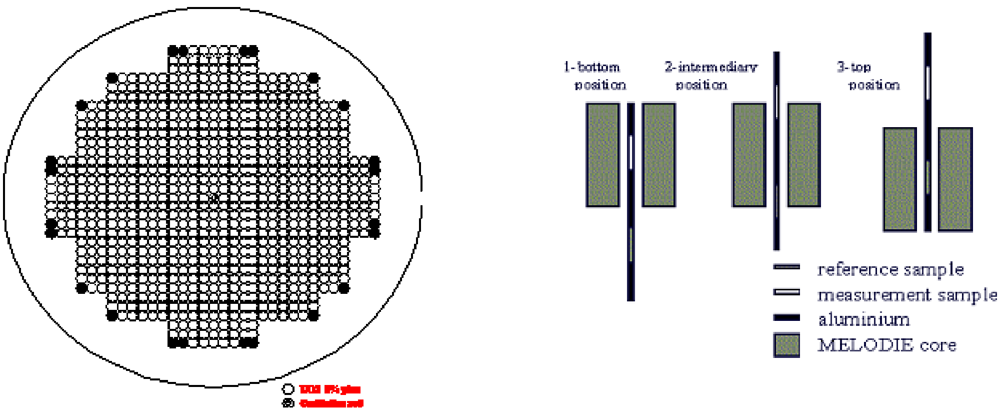

Figure: MINERVE experimental zone

The reactivity of the automatic pilot rod (APR) is linked to the reactivity inserted by the doped sample (SAM) by the following exact perturbation theory:

$$\begin{aligned}\rho_{APR} &= 1/I_F < \phi^*_{APR}, \Delta H_{APR} \phi'_{APR} > \\ &= -\rho_{SAM} = -1/I_F < \phi^*_{SAM}, \Delta H_{SAM} \phi'_{SAM} >\end{aligned}$$

Here, *ΔH* represents the kernel of the Boltzman neutron transport equation for the doped sample, i.e., the self-shielding cross section for pure capturing nuclei and the η value for fissioning nuclei.

This paper will deal with the experimental technique and the required calculation neutronic tools based on exact perturbation theory to measure actinide integral cross section of ^{232}Th, $^{233, 234}U$, ^{237}Np, $^{238,239, 240, 241, 242}Pu$, $^{241, 243}Am$, $^{244, 245}Cm$.

3.13. B. El Bakkari, CNESTEN, Morocco

Benchmark analysis of the 2MW TRIGA MARK II Moroccan research reactor using the MCNP code and the latest nuclear data libraries

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Abstract

A 2-MW TRIGA MARK II research reactor was put in service at « *Centre des Études Nucléaires de la Maâmora – CENM* », Rabat, Morocco in 2007 and it went critical on *May 2007*. The reactor is designed to effectively implement the various fields of basic nuclear research, manpower and training and production of radioisotopes for its uses in agriculture, industry and medicine. This study deals with the neutronic analysis of the current core configuration of the 2-MW TRIGA MARK II research reactor at CENM and validation of the results by benchmarking with the experimental, operational and available Final Safety Analysis Report (FSAR) values. The 3-D continuous energy Monte Carlo code MCNP (version 5) was used to develop a versatile and accurate full-core model of the TRIGA core. The model represents a very detailed description of all components of the core with literally no physical approximation. All fresh fuel and control elements as well as the vicinity of the core were precisely modeled. Continuous energy cross section data from the more recent nuclear data evaluations (ENDF/B-VI.8, ENDF/B-VII.0, JEFF-3.1, and JENDL-3.3) as well as S(α, β) thermal neutron scattering functions from the ENDF/B-VII.8 library were used. The cross section libraries were generated by using the NJOY99 system updated to its more recent patch file "up259". The consistency and accuracy of both the Monte Carlo simulation and neutron transport physics were established by benchmarking the TRIGA experiments. The effective multiplication factor, axial and radial power distribution and peaking factors, reactivity experiments comprising control rod worth, excess of reactivity and shutdown margin were used in the validation process. Differences between neutron cross-section libraries were analyzed and discussed.

Keywords: 2-MW TRIGA MARK II research reactor, validation, MCNP5, ENDF/B-VI.8, ENDF/B-VII.0, JEFF-3.1, JENDL-3.3, NJOY99.

3.14. Y. P. Mahlers, INR, Ukraine

Validation of the ENDF/B-VII library for the WWR-M research reactor in Ukraine

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The initial release of the ENDF/B-VII nuclear data library is validated for the WWR-M research reactor in Ukraine using criticality experiments with various number of fuel assemblies in the core. For neutronics calculation, the MCNP code based on the 3-D Monte-Carlo method is applied. Continuous-energy cross-sections for use with MCNP are calculated with the NJOY code both for ENDF/B-VI and ENDF/B-VII data. Calculated criticality and reaction rates are compared to the corresponding experimental values. Calculation with ENDF/B-VI.8 systematically underestimates criticality while calculation with ENDF/B-VII.0 systematically overestimates it. The main source of such the difference between ENDF/B-VII.0 and ENDF/B-VI.8 is revision of resolved resonance parameter data in the new ^{238}U evaluation, resulting in lower ^{238}U resonance absorption. Nevertheless, for all the experiments, neutronics calculation is in good agreement with the measurements. The maximum absolute deviation of the effective multiplication factor from the results of the measurements is about 0.4% for ENDF/B-VII.0 and 0.6% for ENDF/B-VI.8. The root-mean-square deviation is about 0.2% for ENDF/B-VII.0 and 0.4% for ENDF/B-VI.8, i.e. ENDF/B-VII provides better criticality results than ENDF/B-VI.

3.15. O. Cabellos, Univ. Polit. De Madrid, Spain

Improvements in the Prediction Capability of Codes Used to Design Innovative Reactors

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In the past years, a special emphasis has been invested on nuclear data needs for innovative power reactors and fuel cycles (e.g. dedicated reactors for nuclear waste transmutation, accelerator driven systems). At this respect, the sensitivity studies performed by a well-recognized expert group of OECD/NEA and by a work package of the NUDATRA domain IP-EUROTRANS have identified and independently corroborated a number of highest-priority isotopes/reactions for fast systems and waste minimization technologies. In this direction, we have recently participated in the elaboration of a proposal entitled: "Accurate Nuclear Data for Nuclear Energy Sustainability" (ANDES), within Euratom Call FP7-Fission-2009, to cover three of the main aspects of nuclear data needs: i) improved differential measurements, ii) processing uncertainty/covariance information, and iii) validation with integral experiments.

One of the objectives of ANDES proposal is to improve the inventory codes (ACAB code) to handle the complete set of uncertainty/covariance data (i.e. those of nuclear reactions, radioactive decay and fission yield data) to illustrate the potential benefit of generalizing the assessment of simulation results with full uncertainties propagation. For this, capabilities will be developed both to produce covariance data and to propagate the uncertainties through the inventory calculations. In this paper, our objective is mainly to emphasize the impact of activation cross-section uncertainties on relevant fuel cycle parameters for a conceptual design of dedicated reactors for nuclear waste transmutation (EFIT).

3.16. R. Rosa, ENEA, Italy

The TAPIRO Fast-Neutron Source Reactor as a support to Nuclear Data Assessment

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Abstract

TAPIRO is a fast neutron source reactor operating at CASACCIA Research Center since 1971. The project, entirely developed by ENEA's staff, is based on the general concept of AFSR (Argonne Fast Source Reactor - Idaho Falls). The reactor is equipped with a homogeneous cylindrical core having 6.29 cm as radius and 10.87 cm as height; cladding is provided by stainless steel (0.5 mm thickness) placed on a cylindrical copper reflector having (30 cm as thickness). All components assembled in a stainless steel tank, are placed inside a near spherical borated concrete shielding system having 1.75 m as thickness. Channels of various dimensions and with different neutron spectra are distributed around the core. A large thermal column is manufactured by graphite blocks, suitable to be removed and replaced with experimental assemblies for any research purpose.

The TAPIRO possibilities for reactor experiments with energies up to 1.35 MeV will be illustrated.

3.17. A. Buijs, McMaster Univ., Canada

Research Potential of the McMaster Nuclear Reactor

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This presentation introduces the McMaster Nuclear Reactor (MNR): its design parameters, past and current use and possible future use as a research reactor for nuclear cross section measurements. MNR began operating in April 1959, as the first university-based research reactor in the Commonwealth of Nations, and remains the highest-flux reactor in a university environment in Canada to this day and the only research reactor in Canada with a full containment structure. The reactor is fuelled with low enrichment uranium (19.75% enriched, MTR-type plate fuel), and is cooled and moderated with light water. Current operation is 14 hours per day, five days per week at a thermal power of 3MW. Potential 24-hour operation at a thermal power of 5MW is being investigated.

The reactor is used for a variety of purposes: undergraduate education involves neutron activation analysis (NAA), reactor physics experiments and production of radioisotopes for tracers and counting experiments. Graduate studies use neutron beams for neutron radiography, neutron diffraction, prompt gamma NAA and in-core and hot-cell irradiations for geochronological techniques. Commercial activities include radioisotope production (for the oil and gas industries, biological studies, and cancer treatment) and neutron radiography. The facilities also include a Hot Cell and a high-activity cobalt source and high-level radioisotope laboratories.

The reactor is being considered to receive a larger role in the production of medical isotopes, due in part to the possibly limited operation of the NRU reactor at the Chalk River Laboratories. At the same time, the renewed interest in nuclear power and new reactor designs opens the venue to perform research for validating reactor analysis codes and the nuclear data that are being used in the codes.

For MNR, two types of research are being considered: codes may be validated via comparison to integral measurements of reactor properties. Topics considered here are measurements of the gamma-fields, reactor material cross sections, and reactor properties such as delayed neutron fractions. A second type of research is based on utilization the neutrons from the reactor to measure reaction rates with certain isotopes/materials for which the cross sections are unknown or need to be confirmed or refined. The latter research will potentially require a significant investment in infrastructure and detector equipment.

3.18. <u>U. Köster, ILL, France</u>

Measurement of thermal neutron capture cross-sections of unstable isotopes with the GAMS spectrometer at ILL

<u>Ulli Köster</u>[1], Michael Jentschel[1]

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The crystal spectrometers GAMS4/5 at ILL are mainly used for high resolution nuclear spectroscopy experiments and for the accurate determination of neutron binding energies.

A target is exposed in an in-pile position of the ILL high flux reactor to a thermal neutron flux of about $5 \cdot 10^{14}$ $cm^{-2}s^{-1}$. The prompt and delayed gamma rays emitted after thermal neutron capture are collimated onto one or two bent of flat crystals, usually made from Si or Ge. After Laue diffraction in these crystals the gamma rays are detected with a Ge detector. Rotation of the crystals allows separating gamma rays with a resolution of sub-keV (for gamma rays up to several 100 keV with bent crystals) or even some eV (with flat crystals). Even for higher energy gamma rays bent crystals remain competitive with direct measurements with Ge detectors, not with respect to the energy resolution but with respect to selectivity and sensitivity, i.e. the dynamic range. Thus very weak gamma rays that are usually hidden in Compton background may be detected.

Due to the high neutron flux not only single neutron capture cross-sections can be studied, but the produced radioisotopes may capture one or more neutrons before decaying. Hence, following the grow-in of isotopes produced by double (or multi-) neutron capture allows deducing the thermal neutron capture cross-section of neutron-rich unstable isotopes.

The knowledge of such cross-sections is particularly important for the optimization of the irradiation parameters for radioisotope production, e.g. for medical applications.

We will explain the measurement method in detail, present past experiments, show recent measurements of the $^{36}Cl(n_{th},\gamma)$ and $^{177g}Lu(n_{th},\gamma)$ cross-sections and give an outlook to a series of planned measurements.

3.19. Ge.Zhigang, CIAE, China

The Updated Progress of Chinese Evaluated Nuclear Data Library (CENDL-3.1) and nuclear data evaluation activities in China

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The updated Chinese Evaluated Nuclear Data Library-CENDL-3.1, is a fruit based on the nuclear data evaluated works in recent years, at China Nuclear Data Center (CNDC) in cooperation with China Nuclear Data Coordination Network (CNDCN). CENDL-3.1 contains the evaluated data for reactions with incident neutrons on about 200 nuclides (from ^{3}H to ^{249}Cf) in energy region of 10-5eV-20MeV. All data obtained according to the evaluations of experimental data and theory predictions. For most important nuclei of this library, the benchmark testing and validations have been performed, the comparisons with other nuclear data libraries (ENDF, JENDL, BROND, JEF, et al.) have been done. The testing version of CENDL-3.1 is CENDL-3.0 which has been provided for China domestic users. Follow the using back feed of CENDL-3.0, a lot of improvement has been done. The CENDL3.1 will be provided for all users by ENDF format and released to the world this year.

In the past several years, CNDC and CNDCN also have got a lot of progress in the fields of nuclear data theory study, model and code developments, and nuclear database establishment etc. These progresses will be introduced in this presentation.

3.20. M. Pescarini, ENEA, Italy

Data Processing and Validation Needs of Group-Wise Working Cross Section Libraries for Nuclear Fission Reactor Shielding and Radiation Damage Applications

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Abstract

The generation of broad-group working cross section libraries for nuclear fission reactor shielding and radiation damage applications in different spectral environments (LWR, SFR, LFR, HTR, etc.) is recommended in order to permit the use, in particular, of the three-dimensional (3D) deterministic transport codes in the data validation activities. 3D deterministic transport analyses could be performed, in parallel with similar 3D Monte Carlo analyses, on single-material and engineering neutron shielding benchmark experiments, included in the SINBAD (ORNL/RSICC – OECD/NEA Data Bank) international database. This action, mainly performed by the national research institutes, would be very useful not only for the reactor physicist and nuclear engineer data validation interests but also for the nuclear safety authorities and the industrial organizations.

3.21. Y. Rugama, NEA-OECD

Nuclear Data Activities at NEA: Potential Role of RRs

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Abstract

3.22. V. Pronyaev, IAEA

Nuclear Data Activities at IAEA: Potential Role of RRs

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Abstract

ANNEX II. NEA NUCLEAR DATA HIGH PRIORITY LIST

More detailed information on specific reaction, neutron energy range, precision required, etc. is available from the official NEA-OECD web site at

http://www.nea.fr/html/dbdata/hprl/hprl.pl, November 2009, NEA-OECD

Req.ID	View	Target	Reaction	Quantity	Energy range	Sec.E/Angle	Accuracy	Cov	Field	Date
G 1		14-SI-28	(n,np)	SIG	Threshold-20 MeV	4 pi	20	Y	Fusion	21-SEP-05
H 2		8-O-16	(n,a),(n,abs)	SIG	2 MeV-20 MeV		See details	Y	Fission	21-SEP-05
H 3		94-PU-239	(n,f)	Prompt g-prod	Thermal-Fast	Eg=0-10MeV	7.5	Y	Fission	28-APR-06
H 4		92-U-235	(n,f)	prompt g-prod	Thermal-Fast	Eg=0-10MeV	7.5	Y	Fission	10-MAY-06
H 5		72-HF-0	(n,g)	SIG	0.5-5.0 keV		4	Y	Fission	28-APR-06
G 6		92-U-233	(n,g)	SIG	10 keV-1.0 MeV		9	Y	Fission	28-APR-06
G 7		26-FE-56	(n,xn)	SIG,DDX	7 MeV-20 MeV	1MeV-20MeV	30		Fission,ADS	13-JUL-06
H 8		1-H-2	(n,ela)	dA/dE	0.1 MeV-1 MeV	0-180 Deg	5	Y	Fission	25-JUL-06
G 9		92-U-233	(n,g)	nubar, SIG	Thermal-10 keV		.5	Y	Fission	19-APR-07
G 10		79-AU-197	(n,tot)	SIG	5 keV-200 keV		5		Science,Fusion	18-MAY-07
G 11		94-PU-239	(n,f), (n,g)	SIG,eta, alpha	1 meV-1 eV		1	Y	Fission	09-MAY-07
H 12		92-U-235	(n,g)	SIG, RP	100 eV-1 MeV		3	Y	Fission	29-AUG-07
G 13		24-CR-52	(n,xd), (n,xt)	SIG	Threshold-65 MeV		20	Y	Fusion	23-OCT-07
G 14		94-PU-242	(n,g), (n,tot)	SIG	0.5 eV-2.0 keV		8	Y	Fission	06-JUL-07
H 15		95-AM-241	(n,g), (n,tot)	SIG	Thermal		See details		Fission	08-NOV-07
G 16		95-AM-243	(n,f)	n spectrum	Eth-10 MeV		10		ADS	08-NOV-07
G 17		96-CM-244	(n,f)	n spectrum	Eth-10 MeV		10		ADS	08-NOV-07
H 18		92-U-238	(n,inl)	SIG	65 keV-20 MeV	Emis spec.	See details	Y	Fission	28-MAR-08
H 19		94-PU-238	(n,f)	SIG	9 keV-6 MeV		See details	Y	Fission	31-MAR-08
H 21		95-AM-241	(n,f)	SIG	180 keV-20 MeV		See details	Y	Fission	31-MAR-08
H 22		95-AM-242	(n,f)	SIG	0.5 keV-6 MeV		See details	Y	Fission	31-MAR-08
H 25		96-CM-244	(n,f)	SIG	65 keV-6 MeV		See details	Y	Fission	04-APR-08
H 27		96-CM-245	(n,f)	SIG	0.5 keV-6 MeV		See details	Y	Fission	04-APR-08
H 29		11-NA-23	(n,inl)	SIG	0.5 MeV-1.3 MeV	Emis spec.	See details	Y	Fission	04-APR-08
H 32		94-PU-239	(n,g)	SIG	0.1 eV-1.35 MeV		See details	Y	Fission	04-APR-08
H 33		94-PU-241	(n,g)	SIG	0.1 eV-1.35 MeV		See details	Y	Fission	04-APR-08
H 34		26-FE-56	(n,n')	SIG	0.5 MeV-20 MeV	Emis spec.	See details	Y	Fission	04-APR-08
H 35		94-PU-241	(n,f)	SIG	0.5 eV-1.35 MeV		See details	Y	Fission	04-APR-08
H 36		92-U-238	(n,g)	SIG	20 eV-25 keV		See details	Y	Fission	15-SEP-08
H 37		94-PU-240	(n,f)	SIG	0.5 keV-5 MeV		See details	Y	Fission	15-SEP-08
H 38		94-PU-240	(n,f)	nubar	200 keV-2 MeV		See details	Y	Fission	15-SEP-08
H 39		94-PU-242	(n,f)	SIG	200 keV-20 MeV		See details	Y	Fission	15-SEP-08
H 40		14-SI-28	(n,inl)	SIG	1.4 MeV-6 MeV		See details	Y	Fission	15-SEP-08
H 41		82-PB-206	(n,inl)	SIG	0.5 MeV-6 MeV		See details	Y	Fission	15-SEP-08
H 42		82-PB-207	(n,inl)	SIG	0.5 MeV-6 MeV		See details	Y	Fission	15-SEP-08

ANNEX III. LIST OF RESEARCH REACTORS PROVIDING NUCLEAR DATA MEASUREMENTS

Source: IAEA RR Data Base: http://www.iaea.org/worldatom/rrdb/ , November 2009.

Facilities performing Nuclear Data Measurements

No.	Country	Name	Reactor Type	Thermal Power, kW	Thermal Flux, $n/cm^2/s$	Fast Flux, $n/cm^2/s$	Criticality Date
1	Bangladesh	TRIGA MARK II	TRIGA MARK II	3000	7.5E13	3.8E13	1986-09-14
2	Brazil	IPEN/MB-01	CRIT ASSEMBLY	0	1.0E09	6.0E09	1988-11-09
3	China	MNSR IAE	MNSR	27	1.0E12	1.0E12	1984-03-10
4	China	HWRR-II	HEAVY WATER	15000	2.4E14	5.2E12	1958-09-01
5	China	SPR IAE	POOL	3500	4.0E13	1.1E13	1964-12-20
6	France	MASURCA	CRIT FAST	5		3.0E9	1966-12-01
7	France	EOLE	TANK IN POOL	0	1.0E09		1965-12-02
8	France	HFR	HEAVY WATER	58300	1.5E15		1971-07-01
9	France	MINERVE	POOL	0	1.0E09		1959-09-29
10	Ghana	GHARR-1	MNSR	30	1.0E12	1.2E12	1994-12-17
11	Hungary	NUCL. BUDAPEST RES. REACTOR	TANK WWR	10000	2.5E14	1.0E14	1959-03-25
12	India	APSARA	POOL	1000	1.3E13	1.0E12	1956-08-04
13	Italy	RSV TAPIRO	FAST SOURCE	5		4.0E12	1971-04-04

14	Japan	FCA	CRIT FAST	2		5.0E09	1967-04-29
15	Japan	JRR-3M	POOL	20000	3.0E14	2.0E14	1990-03-22
16	Korea, Republic of	HANARO	POOL	30000	4.5E14	3.0E14	1995-02-08
17	Morocco	MA-R1	TRIGA MARK II	2000	4.4E13	1.8E13	2007-05-02
18	Nigeria	NIRR-0001	MNSR	30	1.0E12	5.0E12	2004-02-03
19	Russian Federation	BFS-1	CRIT ASSEMBLY	0			1961-01-01
20	Russian Federation	BFS-2	CRIT ASSEMBLY	1			1969-01-01
21	Russian Federation	IBR-2	FAST BURST	1500	1.0E13	1.5E14	1977-11-30 Temp.Shut.
22	Russian Federation	SM	PRESS. VESSEL	100000	5.0E15	2.0E15	1961-01-10
23	Switzerland	PROTEUS	CRIT ASSEMBLY	1	5.0E09	5.0E09	1968-01-01
24	Ukraine	WWR-M KIEV	TANK WWR	10000	1.2E14	0.7E14	1960-12-02
25	United States of America	AGN-201 TEXAS A&M UNIV.	HOMOG (S)	0	2.0E08	1.0E08	1957-01-01
26	United States of America	KSU TRIGA MK II	TRIGA MARK II	250	1.0E13	1.2E13	1962-10-16
27	United States of America	NSCR TEXAS A&M UNIV.	TRIGA CONV	1000	2.0E13	2.0E11	1962-01-01

28	United States of America	OSTR, OREGON STATE UNIV.	TRIGA MARK II	1100	1.0E13	5.0E13	1967-03-08
29	United States of America	OSURR OHIO ST. UNIV.	POOL	500	1.5E13	1.0E13	1961-03-16
30	United States of America	UMLR UNIV. MASS. LOWELL	POOL	1000	1.4E13	9.2E12	1975-01-02
31	United States of America	UMRR	POOL, MTR	200	2.0E12	1.0E12	1961-12-11
32	Vietnam	DALAT RESEARCH REACTOR	POOL	500	2.1E13	6.0E12	1963-02-26

ANNEX IV. MONITOR REACTIONS AND MATERIALS FOR NEUTRON SPECTRUM CHARACTERIZATION

Source: Mr Mark A. Kellett, NDS/NAPC-IAEA, M.A.Kellett@iaea.org; November 2009

Recommendations by Frans De Corte (Ghent University, Belgium) for other candidate materials that have suitable capture and threshold reactions

• Au, Zr and Lu; standard materials available as components of k_0-IAEA.

Requests should be addressed to: Mr M. Haji-Saeid, IAEA, Head of Industrial Applications and Chemistry Section, NAPC-IAEA

• Additionally recommended:

1) $^{115}In(n,n')^{115m}In$ [T½ = 4.486 h; Eγ = 336.2 keV, 45.9 %]

~ 0.025 MBq /mg,·φ_f (= $1x10^{11}$) t_{irr} (=5 h)

~ 12000 γ ·s^{-1} / mg,·φ_f (= $1x10^{11}$)·t_{irr} (=5 h)

Notes: a) to be used under Cd-cover; b) not to be used for (n, γ) activation, because of neutron self-shielding problems.

2) $^{64}Zn(n,p)^{64}Cu$ [T½ = 12.70 h; Eγ = 511.0 keV, 35.7 %]

~ 0.0033 MBq /mg,·φ_f (= $1x10^{11}$) t_{irr} (=5 h)

~ 1200 γ·s^{-1} / mg,·φ_f (= $1x10^{11}$) t_{irr} (=5 h)

Notes: a) to be used under Cd-cover; b) low Cu-content = 15 ppm.

3) $^{64}Zn(n,\gamma)^{65}Zn$ [T½ = 244.3 d; Eγ = 1115.5 keV, 50.6 %]

~ 0.0038 MBq /mg,·φ_{th} (= $2x10^{12}$) t_{irr} (=5 h);

~ 0.00038 MBq /mg,·φ_{epi} (= $1x10^{11}$) t_{irr} (=5 h)

~ 1900 γ·s^{-1} / mg,·φ_{th} (= $2x10^{12}$) t_{irr} (=5 h);

~ 190 γ·s^{-1} / mg,·φ_{epi} (= $1x10^{11}$) t_{irr} (=5 h)

4) $^{68}Zn(n,\gamma)^{69m}Zn$ [T½ = 13.76 h; Eγ = 438.6 keV, 94.8 %]

~ 0.054 MBq /mg,·φ_{th} (= $2x10^{12}$) t_{irr} (=5 h);

~ 0.0086 MBq /mg,·φ_{epi} (= $1x10^{11}$) t_{irr} (=5 h)

~ 51000 γ·s^{-1} / mg,·φ_{th} (= $2x10^{12}$)·t_{irr} (=5 h);

~ 8100 γ ·s^{-1} / mg,·φ_{epi} (= $1x10^{11}$)·t_{irr} (=5 h)

5) $^{58}Ni(n,p)^{58}Co$ [T½ = 70.86 d; Eγ = 810.8 keV, 99.5 %]

~ 0.00016 MBq /mg,$\cdot\varphi_f$ (= $1x10^{11}$) t_{irr} (=5 h)

~ 160 $\gamma\cdot s^{-1}$ / mg,$\cdot\varphi_f$ (= $1x10^{11}$) t_{irr} (=5 h)

Note: preferably to be used under Cd-cover.

6) $^{54}Fe(n,p)^{54}Mn$ [T½ = 312.1 d; Eγ = 834.8 keV, 99.98 %]

~ 0.0000024 MBq /mg,$\cdot\varphi_f$ (= $1x10^{11}$) t_{irr} (=5 h)

~ 2.4 $\gamma\cdot s^{-1}$ / mg,$\cdot\varphi_f$ (= $1x10^{11}$)$\cdot t_{irr}$ (=5 h)

Note: preferably to be used under Cd-cover.

7) $^{58}Fe(n,\gamma)^{59}Fe$ [T½ = 44.50 d; Eγ = 1099.3 keV, 56.5 %; Eγ = 1291.6 keV, 43.2 %]

~ 0.00026 MBq /mg,$\cdot\varphi_{th}$ (= $2x10^{12}$) t_{irr} (=5 h);

~ 0.000013 MBq /mg,$\cdot\varphi_{epi}$ (= $1x10^{11}$) t_{irr} (=5 h)

~ 150 $\gamma\cdot s^{-1}$ / mg,$\cdot\varphi_{th}$ (= $2x10^{12}$)$\cdot t_{irr}$ (=5 h); [1099.3 keV]

~ 7.5 $\gamma\cdot s^{-1}$ / mg,$\cdot\varphi_{epi}$ (= $1x10^{11}$)$\cdot t_{irr}$ (=5 h) [1099.3 keV]

8) $^{98}Mo(n,\gamma)^{99}Mo$ [T½ = 65.94 h; Eγ = 140.5 keV (^{99m}Tc; T½ = 6.01 h), 89.06 %]

~ 0.02 MBq /mg,$\cdot\varphi_{th}$ (= $2x10^{12}$) t_{irr} (=5 h);

~ 0.05 MBq /mg,$\cdot\varphi_{epi}$ (= $1x10^{11}$) t_{irr} (=5 h)

~ 18000 $\gamma\cdot s^{-1}$ / mg,$\cdot\varphi_{th}$ (= $2x10^{12}$) t_{irr} (=5 h);

~ 45000 $\gamma\cdot s^{-1}$ / mg,$\cdot\varphi_{epi}$ (= $1x10^{11}$) t_{irr} (=5 h)

9) $^{100}Mo(n,\gamma)^{101}Mo$ [T½ = 14.61 min; Eγ = 306.8 keV (^{101}Tc; T½ = 14.2 min), 88.7 %]

~ 0.24 MBq /mg,$\cdot\varphi_{th}$ (= $2x10^{12}$) t_{irr} (=5 h);

~ 0.23 MBq /mg,$\cdot\varphi_{epi}$ (= $1x10^{11}$) t_{irr} (=5 h)

~ 200000 $\gamma\cdot s^{-1}$ / mg,$\cdot\varphi_{th}$ (= $2x10^{12}$)$\cdot t_{irr}$ (=5 h);

~ 200000 $\gamma\cdot s^{-1}$ / mg,$\cdot\varphi_{epi}$ (= $1x10^{11}$) t_{irr} (=5 h)

Availability of synthetic multi-element standard materials (SMELS)

Peter Vermaercke (SCK, Mol) was contacted during 2006 and reported that SMELS material is available for use in validating the implementation of the k_0 method in a given laboratory and/or research reactor. SMELS material can, in principle, be made available upon written request, but that no new production of the SMELS material is envisaged in the near future.

ANNEX V. LIST OF PARTICIPANTS

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ANNEX VI. MEETING AGENDA

Monday, 12 October 2009

08:30-09:30	**Registration, Gate 1**
09:30-10:30	**Welcome & Opening Remarks** *Mr G. Mank (Section Head, Physics Section, NAPC)* *Mr. D. Abriola (Acting Section Head, Nuclear Data Section, NAPC)* *Ms Y. Rugama (Member of SAC, Nuclear Science Section, NEA)* *Mr V. Pronyaev (Member of SAC, Nuclear Data Section, NAPC)* *Mr D. Ridikas (IAEA Scientific Secretary & Member of SAC, Physics Section, NAPC)* Self introduction of the participants, Election of Chairperson and *Rapporteur* Discussion and Approval of the Agenda, Administrative Arrangements
10:30-11:00	**Objectives of the Meeting (within the IAEA project Enhancement of Utilization and Applications of Research Reactors)** *Mr D. Ridikas*, IAEA
11:00-11:30	Coffee break
11:30-12:15	**Precision Neutron Cross Section Measurements at Reactor Neutron Filtered Beams** *Ms O. Gritzay, INR, Ukraine*
12:15-13:00	**Neutron capture cross-section measurements by high-resolution γ-ray spectroscopy** *Mr M. Oshima, JAEA, Japan*
13:00-14:00	Lunch break
14:00-14:45	**The Out-of-core Neutron Irradiation Facility of HANARO for Measurement of Neutron Cross-section** *Mr M.S. Kim, KAERI, Korea*
14:45-15:30	**Experimental Determination of Neutron Capture Cross Sections at a Rare Thermal Energy Using the BAEC TRIGA Reactor** *Mr S.M. Hossain*, BAEC, Bangladesh
15:30-16:00	Coffee break
16:00-17:00	**Discussion on “RR based neutron beam capabilities for cross section measurements”** All

Tuesday, 13 October 2009

09:00-09:45	**The Use of Miniature Neutron Source Reactor Facility for the Determination of Neutron-induced Cross section Data** *Mr S.A. Jonah, CERT, Nigeria*
09:45-10:30	**Cross Section Determination of Short-to-Medium Lived Nuclides in a Low Power RR and Am-Be Neutron Source** *Mr B.J.B. Nyarko, GAEC, Ghana*
10:30-11:00	Coffee break
11:00-11:45	**Cross section measurements for thermal neutron-induced reaction on actinides at the ILL reactor** Mr A. Letourneau, CEA Saclay, France
11:45-12:45	**Discussion on "RR activation and other methods for cross section measurements"** All
12:45-14:00	Lunch break
14:00-14:45	**Prompt γ-ray emission in nuclear fission** *Mr S. Oberstedt, IRMM-JRC, Belgium/EU*
14:45-15:30	**Binary and Ternary Fission Yield Measurements at the Institut Laue-Langevin** *Mr O. Serot, CEA Cadarache, France*
15:30-16:00	Coffee break
16:00-17:00	**Discussion on "Fission studies and fission fragment measurements at RRs"** All

18:00-	**Hospitality Event** *All*

Wednesday, 14 October 2009

09:00-09:45	**Fast Reactor Integral Experiments at BARC for Cross-section Evaluation** *Mr K.K. Rasheed, BARC, India*
09:45-10:30	**Measurements of Nuclear Data for MA at VENUS-F and BR-1 reactors** *Mr A. Kochetkov, SCK*Mol, Belgium*
10:30-11:00	Coffee break
11:00-11:45	**Integral Experiments in Minerve Reactor Facility for Nuclear Data Validation** Mr D. Bernard, CEA Cadarache, France
11:45-12:45	**Discussion on "Integral RR experiments for Nuclear Data Validation"** All
12:45-14:00	Lunch break
14:00-14:45	**Benchmark analysis of the 2MW TRIGA MARK II Moroccan research reactor using the MCNP code and the latest nuclear data libraries** *Mr B. Bakkari, NSC, Morocco*
14:45-15:30	**Validation of the ENDF/B-VII library for the WWR-M research reactor in Ukraine** *Mr Y.P. Mahlers, INR, Ukraine*
15:30-16:00	Coffee break
16:00-16:45	**Improvements in the Prediction Capability of Codes Used to Design Innovative Reactors** *Mr O. Cabellos, Univ. Polit. Madrid, Spain*
16:45-17:45	**Discussion on "Validation of Different Evaluated Data Files against experimental data originating from RRs"** All

Thursday, 15 October 2009

09:00-09:45	**The TAPIRO Fast-Neutron Source Reactor as a support to Nuclear Data Assessment** *Mr R. Rosa, ENEA, Italy*
09:45-10:30	**Research Potential of the McMaster Nuclear Reactor** *Mr A. Buijs, McMaster Univ., Canada*
10:30-11:00	Coffee break
11:00-11:45	**Measurement of thermal neutron capture cross-sections of unstable isotopes with the GAMS spectrometer at ILL** *Mr U. Koester, ILL, France*
11:45-12:45	**Discussion on "Potential of RRs for provision of Nuclear Data for various applications"** All
12:45-14:00	Lunch break
14:00-14:45	**The Updated Progress of Chinese Evaluated Nuclear Data Library (CENDL-3.1) and nuclear data evaluation activities in China** *Mr Ge Zhigang, CIAE, China*
14:45-15:30	**Nuclear Data Activities at NEA: Potential Role of RRs** *Ms Y. Rugama, NEA, France*
15:30-16:00	Coffee break
16:00-16:45	**Nuclear Data Activities at IAEA: Potential Role of RRs** *Mr V. Pronyaev, IAEA, Austria*
16:45-17:45	**Discussion on "Potential of RRs for provision of Nuclear Data for various applications"** All

Friday, 16 October 2009

09:00-12:30	**Discussion** • **Identify future needs of Nuclear Data and the role RR could play in this respect** • **Identify common issue areas on nuclear data needs, provision, validation by RRs** • **Suggest collaborative research activities in this area that the IAEA could promote and facilitate** *All*
12:45-14:00	Lunch break
14:00-15:30	**Discussion:** • **Formulation of conclusions and recommendations** • **Drafting of the meeting report** *All*
15:30-16:00	Coffee break
16:00-17:00	**Drafting and Finalizing meeting report** **Closing of the meeting** *All*
17:00	End of the meeting