

# **INDC International Nuclear Data Committee**

Proceedings of the  
**Fifth AASPP Workshop on  
Asian Nuclear Reaction Database Development**

Bhabha Atomic Research Centre, Mumbai, India

22 – 24 September 2014

Edited by

Alok Saxena  
Nuclear Physics Division, Bhabha Atomic Research Centre

February 2015

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**Abstract**

The Fifth AASPP Workshop on Asian Nuclear Reaction Database Development was organized by Nuclear Data Physics Centre of India in cooperation with the IAEA and the Board of Research in Nuclear Sciences, Department of Atomic Energy from 22-24 Sept., 2014 at Bhabha Atomic Research Centre, Mumbai, India. Over seventy participants took part in the workshop representing India, Japan, Republic of Korea, Vietnam, Kazakhstan, Uzbekistan, Mongolia and a representative from the IAEA. The workshop covered the overview of nuclear data activities in different countries and covered the topics related to experiments performed using various facilities, the upcoming and existing accelerators and experimental facilities, EXFOR compilation activities, reactor sensitivity studies to nuclear data, criticality benchmarking studies, nuclear data requirement of nuclear power reactors, theoretical calculations using various codes, and covariances in nuclear data. About thirty five talks were delivered by participants from various countries on these topics. The concluding session had a panel discussion on possible future collaboration involving participants from different countries. The present report gives summary of each presentation.



## Table of Contents

|   |    |
|---|----|
| Agenda  | 9  |
| M. Aikawa et al., Compilation status and research topics in Hokkaido University Nuclear Reaction Data Centre                            | 13 |
| Alok Saxena, An overview of activities of nuclear data physics centre of India (NDPCI)  | 15 |
| Ge Zhigang, Recent nuclear data work in China   | 19 |
| B.K. Nayak, Surrogate reaction methods for neutron induced cross-sections   | 21 |
| Prakash Chandra Rout, Experimental studies on nuclear level density   | 23 |
| V. Semkova and N. Otuka, Neutron-induced activation measurements and EXFOR compilations in the energy range up to 20 MeV                | 27 |
| Lalremruata Bawitlung, Overview of EXFOR activities in India  | 31 |
| Pham Ngoc Son and Vuong Huu Tan, Filtered thermal neutron captured cross sections measurements and decay heat calculations              | 37 |
| Devesh Raj, Decay heat calculations for reactors: Development of a computer code ADWITA   | 41 |
| Gopal Mukherjee, Overview of ENSDF activities in India  | 47 |
| Tran Tuan Anh et al., Total neutron cross section measurements of $^{93}\text{Nb}$ on filtered neutrons beams at Dalat Research Reactor | 51 |
| S. Ganesan, Nuclear data covariances in the Indian context  | 55 |
| E. Radha, ICSBEP criticality benchmarking for nuclear data validations, KAMINI, PURNIMA-II and PURNIMA-I                                | 61 |
| Wang Jimin et al., Evaluation of the $\text{D}(n,2n)\text{p}$ reaction cross section  | 67 |
| Jong Woon Kim and Young-Ouk LEE, Current Status of nTOF Facility construction at KAERI  | 71 |
| S. Acharya et al., Design considerations for an n-TOF facility absed on a 30 MeV RF electronic linac                                    | 73 |
| V.M. Datar, Overview of upcoming INO facility   | 79 |

|  |     |
|--|-----|
| Sailajananda Bhattacharya, Nuclear Physics using VECC cyclotrons: scopes and possibilities   | 81  |
| Amar Sinha, Physics study of D-D/D-T neutron driven experimental subcritical assembly  | 83  |
| P. Singh, An overview of FOTIA and LEHIPA  | 87  |
| P. Sugathan, Overview of accelerator and nuclear physics facilities at IUAC, New Delhi   | 89  |
| A. K. Gupta, An overview of BARC-TIFR Pelletron-Linac Facility   | 93  |
| Sung-Chul Yang et al., EXFOR compilation and nuclear data measurement at KAERI/NDC   | 95  |
| H. Naik, Studies on high energy photon (bremsstrahlung) and neutron induced fission of actinides and pre-actinides                   | 97  |
| Ashok Kumar Jain and Bhoomika Maheshwari, Atlas of nuclear isomers and their systematics   | 103 |
| Xi Tao et al., Theoretical Calculation of $n+{}^6\text{Li}$ Reaction below 20 MeV  | 109 |
| Guinyun Kim et al., Activities for nuclear data measurement using charged particle-, neutron-, and photon-induced reactions in Korea | 113 |
| Guochang Chen et al., Update neutron nuclear data evaluation for ${}^{236,238}\text{Np}$   | 115 |
| M.Odsuren et al., Applications of the photo-nuclear reaction data for activation analysis  | 119 |
| Guochang Chen et al., Recent EXFOR compilation in CNDC   | 125 |
| G. Pandikumar et al., Nuclear data needs for fast reactors   | 129 |
| M.P.S. Fernando, Nuclear data needs of Indian nuclear program  | 131 |
| Umasankari Kannan et al., Sensitivity studies on nuclear data for thorium fuelled Advanced Heavy water Reactor (AHWR)                | 133 |
| S.V. Suryanarayana et al., Statistical model calculations using TALYS code for the study of neutron and $\gamma$ induced reactions   | 139 |
| A. Kumar et al., Study of elastic and inelastic neutron cross sections using time of flight technique                                | 141 |

|  |     |
|--|-----|
| F. Ergashev et al., Inclusion of nuclear data of Uzbekistan authors to the NRDC during 2013-2014 years                                     | 147 |
| N. Burtebayev et al., Experimental and theoretical investigation of scattering of alpha particles from $^{13}\text{C}$ nuclei              | 149 |
| N. Kenzhebayev et al., Joint activities with IAEA on uploading of scientific papers from Kazakhstan and Uzbekistan into the EXFOR database | 153 |
| N. Burtebayev et al., Study of interaction mechanisms of alpha particles with $^{11}\text{B}$ nuclei at low energies                       | 155 |
| Chen Guochang et al., Introduction of the digitization software GDgraph  | 159 |
| Vidya Thakur, Review on compilation work in JCPRG  | 163 |
| Panel Discussion   | 167 |





**5<sup>th</sup> AASPP Workshop on  
Asian Nuclear Reaction Database Development**

September 22-24, 2014.

**Seminar Hall, NUB, Anushaktinagar, Mumbai 400 094**

*(Lecture time includes time for questions & answers)*

**Day 1 (Monday, 22<sup>nd</sup> September, 2014), Session 1**

9.30 – 10.00

**Inaugural Session:**

|                              |  |
|------------------------------|--|
| <b>Welcome address</b>       | <i>P.D. Krishnani</i> , Chairman, Organizing Committee                     |
| <b>Introductory Remarks</b>  | <i>S. Kailas</i> , Ex-Director, Physics Group, BARC                        |
| <b>Theme of the workshop</b> | <i>Alok Saxena</i> , Technical Convener                                    |
| <b>Inaugural Address</b>     | <i>P.K. Vijayan</i> , Director, Reactor Design and Development Group, BARC |
| <b>Vote of Thanks</b>        | <i>Devesh Raj</i> , Co-convener  |

10:00 -10:30 *Mayasuki Aikawa, Hokkaido University, JAPAN*

“Compilation Status and Research Topics in Hokkaido University Nuclear Reaction Data Centre”

10:30-10:50 *Alok Saxena, Nuclear Physics Division, BARC, Mumbai*

“Overview of Nuclear Data Activities of Nuclear Data Physics Centre of India (NDPCI)”

**10:50 – 11:15 High Tea**

11:15 – 11:45 *Ge Zhigang, China Nuclear Data Center, China Institute of Atomic Energy, Beijing, P.R. China*

“Recent Nuclear Data Work in China”

11:45-12:05 *B.K. Nayak, Nuclear Physics Division, BARC, Mumbai*

“Surrogate Reaction Methods for Neutron Induced Cross-sections”

12:05-12:25 *P.C. Rout, Nuclear Physics Division, BARC Mumbai*

“Experimental Studies on Nuclear Level Density”

12:25-12:55 *V. Semkova, Nuclear Data Section, International Atomic Energy Agency, Vienna*

“Neutron-Induced Activation Measurements and EXFOR Compilations in the Energy Range up to 20 Mev”

12:55-13:15 *Lalremruata Bawitlung, Mizoram University, Mizoram*

“Overview of EXFOR Activities in India”

**13:15 – 14:15 Lunch**

## ***Day 1, Session 2***

14:15 – 14:45 ***Pham Ngoc Son, Nuclear Research Institute, Dalat, Vietnam***  
“Filtered Thermal Neutron Captured Cross Sections Measurements and Decay Heat Calculations”

14:45-15:05 ***Devesh Raj, Reactor Physics Design Division, BARC, Mumbai***  
“Decay Heat Calculations for Reactors”

15:05-15:25 ***Gopal Mukherjee, Variable Energy Cyclotron Centre, Calcutta***  
“Overview of ENSDF Activities in India”

15:25-15:55 ***T. T. Anh, Nuclear Research Institute, Dalat, Vietnam***  
“Total Cross Section Measurements of Nb, In and Pr on Filtered Neutron Beams at Dalat Research Reactor”

**15:55 – 16:15      Tea**

## ***Day 1, Session 3***

16:15-16:35 ***S. Ganesan, Reactor Physics Design Division, BARC, Mumbai***  
“Nuclear Data Covariances in the Indian Context”

16:35-16:55 ***E. Radha, Indira Gandhi Centre for Atomic Research, Kalpakkam***  
“ICSBEP Criticality Benchmarking for Nuclear Data Validations, KAMINI, PURNIMA-II and PURNIMA-I”

16:55-17:25 ***Wang Jimin, China Nuclear Data Center, China Institute of Atomic Energy, Beijing, P.R. China***  
“Evaluation of the D(n,2n)p Reaction Cross Section”

## ***Day 2, Session 4 (Tuesday, 23<sup>rd</sup> September, 2014)***

10:00-10:30 ***Jong Woon Kim, Korea Atomic Energy Research Institute, Daejeon, Korea***  
“Current Status of n\_TOF Facility Construction at KAERI”

10:30-10:50 ***S. Acharya, BARC, Mumbai***  
“RF Electron Linacs for Neutron Time-of-Flight Facilities”

10:50-11:10 ***V.M. Datar, Nuclear Physics Division, BARC, Mumbai***  
“Overview of Upcoming INO Facility”

**11:10-11:30      Tea**

## ***Day 2, Session 5***

11:30-11:50 ***Saila Bhattacharya, VECC, Calcutta***  
“Nuclear Physics using VECC Cyclotrons: Scopes and Possibilities”

11:50 -12:10 *Amar Sinha, BARC, Mumbai*  
“Physics Study of D-D/D-T Neutron Driven Experimental Subcritical Assembly “

12:10-12:30 *P. Singh, BARC Mumbai*  
“An Overview of FOTIA and LEHIPA”

12:30-12:50 *S. Sugathan, IUAC, New Delhi*  
“Overview of Nuclear Experimental Facilities and the Accelerator Facilities at IUAC”

12:50-13:10 *A.K. Gupta, BARC, Mumbai*  
“An Overview of BARC-TIFR Pelletron-Linac Facility”

**13:10 – 14:15 Lunch**

### ***Day 2, Session 6***

14:15-14:45 *Sung-Chul Yang, Nuclear Data Center, KAERI, Korea*  
“EXFOR Compilation and Nuclear Data Measurement at KAERI/NDC”

14:45-15:05 *H. Naik, Radio Chemistry Division, BARC, Mumbai*  
“Studies on High Energy Photon (Bremsstrahlung) and Neutron Induced Fission of Actinides and Pre-Actinides”

15:05-15:25 *A.K. Jain, Indian Institute of Technology, Roorkee,*  
“Atlas of Nuclear Isomers and their Systematics”

15:25-15:55 *Tao Xi, China Nuclear Data Center, China Institute of Atomic Energy,  
Beijing, P.R. China*  
“R-matrix Analysis of  $n+{}^6\text{Li}$  Reaction”

**15.55 – 16.15 Tea**

16:30-17:00 *GN Kim, Kyungpook National University, Daegu, Korea*  
“Activities for Nuclear Data Measurement using Charged particle-, Neutron-, and Photon Induced Reactions in Korea”

17: 00-17:30 *Guochang Chen, China Nuclear Data Center, China Institute of  
Atomic Energy, Beijing, P.R. China*  
“Update of Neutron Nuclear Data Evaluation for  ${}^{236,238}\text{Np}$ ”

### ***Day 3, Session 7 (Wednesday, 24<sup>th</sup> September, 2014***

10:00-10:30 *M. Odsuren, Nuclear Research Center, National University of  
Mongolia, Ulaanbaatar, Mongolia*  
“Applications of the Photo-Nuclear Reaction Data for Activation Analysis”

10:30-11:00 **Guochang Chen, China Nuclear Data Center, China Institute of Atomic Energy, Beijing, P.R. China**  
“Recent EXFOR Compilation in CNDC”

11:00-11:20 **G. Pandikumar, Indira Gandhi Centre for Atomic Research, Kalpakkam**  
“Nuclear Data Needs for Fast Reactor”

**11:20-11:40** Tea

### **Day 3, Session 8**

11:40-12:00 **M. P. S. Fernando, Nuclear Power Centre of India Ltd, Mumbai**  
“Nuclear Data Needs for Indian Nuclear Power Program”

12:00-12:20 **Umasankari Kannan, Reactor Physics Design Division, BARC, Mumbai**  
“Sensitivity Studies on Nuclear Data for Thorium Fuelled Advanced Heavy water Reactor (AHWR)”

12:20-12:40 **S.V. Suryanarayana, Nuclear Physics Division, BARC, Mumbai**  
“Statistical Model Calculations using TALYS code for the Study of Neutron and  $\gamma$  Induced Reactions”

12:40-13:00 **Ajay Kumar, BHU, Varanasi**  
“Study of Elastic and Inelastic Neutron Cross sections using Time of Flight Technique”

**13:00-14:00** Lunch

### **Day 3, Session 9**

14:00-14:30 **F. Ergashev, S. Artemov, R. Yarmukhamedov, Institute of Nuclear Physics, Uzbekistan, Academy of Sciences, 702132 Tashkent,**  
“Inclusion of nuclear data of Uzbekistan authors to the NRDC during year 2013-2014”

14:30-15:00 **N.Takibayev, Al-Farabi Kazakh National University, Almaty, Kazakhstan**  
“Joint Activities with IAEA on Uploading of Scientific Papers from Kazakhstan and Uzbekistan into the EXFOR Database”

15:00-15:20 **Yongli Jin, China Nuclear Data Center, China Institute of Atomic Energy, Beijing, P.R. China**  
“Introduction on the Digitization Software GDgraph”

**15:20-15:45** Tea

**15:45 - 16:30 - PANEL DISCUSSION**

# Compilation status and research topics in Hokkaido University Nuclear Reaction Data Centre

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A. Sarsembayeva<sup>2</sup>, B. Zhou<sup>2</sup> and N. Otuka<sup>3</sup>

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## Abstract

Nuclear reaction data are necessary and applicable for many application fields. The nuclear reaction data must be compiled into a database for convenient availability. One such database is the EXFOR database maintained by the International Network of Nuclear Reaction Data Centres (NRDC). As a member of the NRDC, the Hokkaido University Nuclear Reaction Data Centre (JCPRG) compiles charged-particle induced reaction data and contributes about 10 percent of the EXFOR database. In this paper, we show the recent compilation status and related research topics of JCPRG.

## 1. Introduction

Nuclear data are available for many application fields, such as nuclear engineering and medicine, in addition to basic science; e.g., nuclear physics and astrophysics. Thus, it is widely useful and valuable for our lives. Due to such importance, many kinds of nuclear reaction data are obtained experimentally in many institutes worldwide. However, nuclear reaction experiments sometimes require high investment and resources. The nuclear data, therefore, must be stored in a database and opened to the public, free of charge.

A database consisting of such experimental nuclear reaction data is the EXFOR database, maintained by the International Network of Nuclear Reaction Data Centres (NRDC) under the auspices of the International Atomic Energy Agency (IAEA) [1]. In the NRDC, there are thirteen member institutes including five in Asia. There is a new candidate for the NRDC membership: Central Asia Nuclear Reaction Database (CANRDB), started to compile nuclear reaction data obtained in Kazakhstan and Uzbekistan. Under such circumstances, it is worth sharing information on compilation and related activities in each institute, especially in Asia. In this paper, we show the compilation procedure and activities in the Hokkaido University Nuclear Reaction Data Centre (JCPRG).

## 2. Compilation

JCPRG is a member of the NRDC and compiles charged-particle induced nuclear reaction data obtained in Japanese institutes. Papers to be compiled into a database are surveyed in peer-reviewed journals. JCPRG individually performs such surveys of papers in addition to the NRDC survey. If papers are found, we assign entry numbers with E, J, and K which are charged-particle, meson, and photon induced reactions, respectively. We also maintain entries with R, which was compiled by a former member of the NRDC, RIKEN.

Once an entry number is assigned to each paper, we start to place information from the paper into a data input system, Hyper Editor for Nuclear Data Exchange Libraries (HENDEL) [3],

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which was originally developed in JCPRG. The numerical data are received from the authors of the paper or digitized from figure images in the paper using the digitizing software “GSYS”. From Apr. to Sep. 2014, JCPRG submitted twelve new and sixteen revised entries. The contribution of JCPRG is about 10% of the charged-particle data in the EXFOR database, in total.

### **3. Asian Collaboration**

As one of the purposes, JCPRG promotes nuclear data activity in Asia. The activity was supported by the 'R&D' Platform Formation of Nuclear Reaction Data in Asian Countries (2010-2013), the Asia-Africa Science Platform Program, the Japan Society for the Promotion of Science (JSPS) from April 2010 to March 2013. In this respect, annual workshops are held in Japan, China, and Korea, from 2010 to 2012. At the end of the support, we continued the workshops in Kazakhstan in 2013 and India in 2014. The plan for the next workshop, in 2015, is that it will be held in Japan. One of the topics in the workshop is sharing information of compilation among Asian institutes.

### **4. Research Topics**

One of the recent interests in JCPRG is related to nuclear medicine. We are supported by the JSPS Bilateral Program "Measurement and Evaluation of Important Nuclear Data for Diagnosis and Therapy Treatments" for two years from Apr. 2014. The program is in collaboration with the Institute of Nuclear Research of the Hungarian Academy of Sciences (ATOMKI). In Apr. 2014, we performed an experiment at ATOMKI to obtain the cross section of  $^{100}\text{Mo}(p,2n)^{99\text{m}}\text{Tc}$  reaction [4].

### **5. Summary**

Compilation of nuclear reaction data is necessary for applications. JCPRG contributes the compilation as a member of the NRDC. Related to such compilation activity, we conduct and promote research on nuclear data under international and Asian collaboration.

[1] N. Otuka, et al., Nucl. Data Sheets 120, 272, 2014.

[2] N. Otuka and E. Dupont, INDC(NDS)-0661, 2014.

[3] M. Aikawa, et al., INDC(KAS)-001, 2013.

[4] S. Takács, et al, submitted to Nucl. Instr. and Meth. B

# An overview of activities of nuclear data physics centre of India (NDPCI)

Alok Saxena (on behalf of NDPCI)

*Nuclear Physics Division, Bhabha Atomic Research Centre, Trombay, Mumbai-400085 India*

## 1. Introduction

India has a three stage nuclear power programme which requires accurate inputs of nuclear data for design and safe operation of existing as well as for the design of new and innovative reactors. Apart from that nuclear data is required for accelerator shield design, personal dosimetry, radiation safety, production of radioisotopes, radiation damage studies, waste transmutation etc. To cater to various needs of department, the Nuclear Data Physics Centre of India (NDPCI) was formed in 2010-11 to provide a platform for coordinated efforts in all aspects of nuclear data, viz., measurements, analysis, compilation and evaluation involving national laboratories and universities in India. The NDPCI has projects / collaborations with universities and various units of department of atomic energy (DAE) across India involving physicist, radio-chemists, reactor physicists and computer engineers. A number of projects have been awarded under NDPCI to various universities to involve faculties and students in nuclear reactions, nuclear structure and EXFOR compilations. The NDPCI is presently a virtual centre under Board of Research in Nuclear Sciences of DAE and functions through two committees namely Program Implementation Committee and Program Review Committee involving scientists and faculties from various divisions of DAE units and universities. A brief account of NDPCI activities carried out by our researchers is described in this report.

## 2 EXFOR Compilations

NDPCI has been successful in contributing to EXFOR database of IAEA through entries compiled by participants in various domestic workshops on EXFOR compilation as well as through projects given to various universities. The initiative to organize various workshops on EXFOR compilation with the support from faculties of nuclear data section (NDS) of IAEA has been very successful. NDPCI has conducted five domestic EXFOR workshops (2006, 2007, 2009, 2011, 2013) in various parts of country and the next EXFOR workshop is scheduled in Bangalore University in the month of January, 2015. The organizers of these workshops have successfully collected many young and senior participants from the universities and institutes and it is a very efficient way of compiling articles from Indian measurements into EXFOR database. The participation of NDPCI to NRDC (International Network of Nuclear Reaction Data Centres) was officially approved in 2008 NRDC Meeting (Obninsk, Russia). The number of EXFOR entries created in the EXFOR workshops in India is more than over 278 entries which compares well with those compiled by other centres of the network (refer to [https://www-nds.iaea.org/exfor-master/x4compil/exfor\\_input.htm](https://www-nds.iaea.org/exfor-master/x4compil/exfor_input.htm)). The mirror of EXFOR website of NDS-IAEA is maintained by Computer Division of Bhabha Atomic Research Centre (BARC). The mirror website is well synchronized with the primary database in Vienna.

## 3. Nuclear Reaction Experiments in India

The nuclear data experiments in India are performed using various experimental facilities such as

BARC-TIFR pelletron facility, Mumbai, FOTIA at BARC, research reactors like APSARA, DHRUVA and CIRUS in BARC, electron linacs at Khargarh, Navi Mumbai, 14 MeV generator, Pune, Variable Energy Cyclotron, Kolkatta, Inter University Accelerator Centre, New Delhi. More than 35 papers related to nuclear data were published in last two years. Many nuclear data experiments have used  ${}^7\text{Li}(p,n)$  reaction to produce quasi-mono-energetic neutrons. An irradiation facility at pelletron above analyzing magnet was extensively used for many experiments. The measured cross-section data were compared with the predictions of statistical model codes such as EMPIRE and TALYS. The various sources of uncertainties in the measurements were also reported. The attempts are being made to calculate covariance matrices in some of the measurements. A 4 day workshop was organized on covariance in nuclear data during Dec. 16-19, 2013 at Seminar Hall, Homi Bhabha National Institute, Mumbai. The talks included invited tutorials and expert lectures on covariance error matrix and its applications with reference to reactor fuel cycle.

A number of experiments were carried out by surrogate method for neutron induced cross-sections of those actinide nuclei where direct measurements are difficult or impossible due to short half-lives of the target nuclei. Recently, experimental group at Nuclear Physics Division have formulated a new surrogate method known as ‘‘hybrid surrogate ratio method’’, which has been employed by them to determine  ${}^{233}\text{Pa}(n, f)$  cross-section of interest in Th-U fuel cycle for fast neutrons for the first time [1]. This result is very important because of the primary reaction of importance in the thorium cycle. As a continuation, another experiment has been carried out for  ${}^{241}\text{Pu}(n, f)$  cross section data at BARC-TIFR Pelletron facility, Mumbai by surrogate method employing  ${}^{238}\text{U}({}^6\text{Li}, df)$ ,  ${}^{232}\text{Th}({}^6\text{Li}, df)$  reactions, where the half-life of  ${}^{241}\text{Pu}$  is only 14.3 years. The  ${}^{241}\text{Pu}(n, f)$  cross-section data for this measurement were observed to be consistent with the direct measurements, suggesting the applicability of surrogate methods. The surrogate reaction method has also been used to determine neutron-induced fission cross sections of the short-lived minor actinides  ${}^{239}\text{Np}$  and  ${}^{240}\text{Np}$ . We have also measured the fission cross-section for  ${}^{234}\text{Pa}(n, f)$  using hybrid surrogate ratio method. The fission barrier heights corresponding to all the reactions studied above for various isotopes in EMPIRE calculations were obtained from our barrier formula and these calculations describe the experimental results rather well. A two day national workshop on surrogate reactions and its applications was organized at MSU, Baroda during 24-25 Jan, 2013 to involve participation of university in this area of research.

We are also participants in IAEA CRP on prompt fission neutron spectra of actinides which is an important part in the light of our current perspectives on nuclear data physics activities in India [2]. The BARC team in the early sixties had performed several interesting and new studies in neutron induced fission of  ${}^{235}\text{U}$ . In the nineties, the experimental work on fission physics was continued and, for instance, reported in an IAEA Meeting. As a part of our proposal to carry out prompt fission neutron spectra in fast neutron induced fission reaction we have carried out a systematic study of prompt neutron energy spectrum and their multiplicities over an incident energy range of neutron from 2 to 3 MeV for  ${}^{238}\text{U}$  at Folded Tandem Ion Accelerator, B.A.R.C.

The CERN n\_ToF facility is involved in measurement of nuclear data for astrophysics and ADS and BARC has a MOU with CERN for participation in experiments in phase 2. In one of the experiments, where we participated, the measurement of the neutron-induced capture cross section of the fissile isotope  ${}^{235}\text{U}$  was carried out using a fission tagging set-up. This new set combines the n\_TOF  $4\pi$  Total Absorption Calorimeter (TAC) with MicroMegs (MGAS) fission detectors.



A number of experiments were carried out to study light charged particles, heavy ion induced reactions involving fusion-fission, elastic scattering, transfer reaction and nuclear structure measuring mass and angular distributions of fission fragments, neutron and charged particle multiplicity, ER cross-sections. Details about such studies can be found in the Proceedings of Nuclear Physics Symposium for 2012 and 2013 in Volume 57 and 58 and are available online at <http://sympnp.org>. We also have informal collaboration with Legnaro National Laboratory, Italy for experiments on fission dynamics in superheavy region. A number of experiments have been carried out and we have extracted average neutron multiplicity from a spontaneous fission of a superheavy composite system from these measurements for  $^{288}116$ ,  $^{312}124$  and  $^{258}\text{Rf}$  [3].

#### 4. Nuclear Data Theory and Simulations

A number theoretical calculations and simulations were carried out by scientists of NDPCI and some of them are described below:

Neutron induced reaction cross-sections for the production of radio nuclides with intermediate to long half-lives produced in 1-200 MeV neutron induced reactions were calculated using the computer codes EMPIRE-2.19, ALICE-91 & TALYS-1.0 [4]. The target elements chosen for the purpose are the common constituent materials and experimental targets in reactor & accelerator facilities like Al, Fe, Cu, Ni, Ge, Ta, Au, along with preferred targets for Accelerator Driven Sub critical Systems (ADSS), e.g., W, Hg, Bi & Pb for the production of long-lived radioisotopes on the basis of their half-lives & abundances.

A computer code has been indigenously developed to solve the nuclide chain for the thorium fuel cycle. The actinide depletion and formation equations are modelled with continuous energy nuclear data. This code is used to estimate the variation of conversion of pure  $^{232}\text{Th}$  sample into fissile  $^{233}\text{U}$  as a function of neutron energy.

The detailed point wise nuclear data which is required for Monte Carlo calculation are kept in individual tables that are often organized into libraries in the ACE (A Compact ENDF) format.

A continuous energy general geometry Monte Carlo code (M3C code, Monte Carlo Criticality Calculation code) has been developed from scratch which has many advanced features like unionized energy grid approach, explicit sampling of delayed-neutron spectrum, probability table treatment in unresolved resonance range and thermal treatment for light nuclides.

The fission spectrum as given in the multi-group library of WIMS Library Update Project (WLUP) are based in U-Pu systems and is computed with weights of 54%, 8% and 38% for  $^{235}\text{U}$ ,  $^{238}\text{U}$  and  $^{239}\text{Pu}$  respectively. This has been modified for AHWR using the thorium fuel cycle has been computed by averaging over isotopes spectra of  $^{233}\text{U}$ ,  $^{239}\text{Pu}$  and  $^{241}\text{Pu}$  with weights 65%, 27% and 8%, respectively.

A sensitivity study has been done with respect to nuclear data set, modeling uncertainties and manufacturing tolerances such fuel density and enrichments, dimensions etc. has been undertaken for AHWR-LEU fuel. The effect of these uncertainties on the design margins has been quantified. For details please see reference 5.

#### 5 Nuclear Data Evaluation under ENSDF

A workshop on ENSDF Evaluation was organized between Nov. 26 – 29, 2012 at VECC, Kolkata sponsored by Board of Research in Nuclear Science (BRNS), India. The topic covered were such as ENSDF Evaluation Methodology, ENSDF Policies, NSDD network Web, NuDat, bibliographic Data base and XUNDL, nuclear theory for experimentalist and evaluators experimental techniques for gamma-ray spectroscopy. The mass chain  $A = 215$  was taken for

evaluation in the workshop by the trainees mentored by senior evaluators. The evaluation work was pursued after the workshop and it has been successfully completed and published in Nuclear Data Sheets (NDS vol. 114 (2013) p 2023 – 2078).

The work has been done at Indian Institute of Technology, Roorkee to evaluate and update the mass chain A=139, 211, 222 and 224 [6]. Some highlights are given below:

(a) There are total 16 nuclides in this mass chain and 10 nuclides namely  $^{139}\text{Nd}$ ,  $^{139}\text{Sb}$ ,  $^{139}\text{Te}$ ,  $^{139}\text{Xe}$ ,  $^{139}\text{I}$ ,  $^{139}\text{Dy}$ ,  $^{139}\text{Tb}$ ,  $^{139}\text{Gd}$ ,  $^{139}\text{Eu}$  and  $^{139}\text{Sm}$  has been up updated.

(b) For  $^{139}\text{Cs}$ , the individual datasets and adopted data set have been updated. The results of Pandora are being studied for fine tuning the adopted data set.

(c) The evaluation of  $^{139}\text{Ba}$  is also in progress and data from the latest experiment held at BARC is awaited for being included.

In the A=224 mass chain, there are total 10 nuclides and three nuclides namely  $^{224}\text{At}$ ,  $^{224}\text{Bi}$ ,  $^{224}\text{Th}$  have been updated. Out of these three nuclides  $^{224}\text{At}$  has been included in the ENSDF database so that the updated information for this nuclide will be incorporated in the upcoming evaluation by Audi et al. 2011AuZZ. The evaluation of  $^{224}\text{Po}$  and  $^{224}\text{Pa}$  is going on.

## 6. Concluding Remarks

To summarize, the present report presents the highlights of some of the important activities of NDPCI. The NDPCI office is ready at BARC. NDPCI has been identifying and nurturing potential young researchers through the projects and other activities of NDPCI. We continue supporting researchers through funds giving mechanisms to generate basic nuclear data for various applications (Measurements, evaluation and compilation of nuclear data and sensitivity studies for various reactor systems etc) through outreach programmes. More details about NDPCI activities can be found at [https://www-nds.iaea.org/nrdc/nrdc\\_2014/progres/ndpci.pdf](https://www-nds.iaea.org/nrdc/nrdc_2014/progres/ndpci.pdf).

The author thankfully acknowledges all colleagues from various Divisions in BARC, VECC, IGCAR and Universities for useful discussions and inputs for this report.

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# Recent nuclear data work in China

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## **Abstract**

A brief of introduction about China nuclear activities will be introduced in this presentation, which including the view of nuclear data measurement, evaluation activity system of China and the progress of the nuclear data measurements and evaluations during recently years. As the main output of China Nuclear Data Center, The scheme of the new Chinese Evaluated Nuclear Data Library (CENDL) will be introduced also in this article. The new version of CENDL is general purpose evaluated nuclear data file which consists of the neutron reaction sub-library, the activation sub-library, decay data sub-library and fission yield sub-library. CENDL-3.2 can be used for the nuclear engineering, nuclear medicine and nuclear science etc. fields. The CENDL-3.2 is based on the previous version of CENDL and other special purpose libraries established by CNDC, the updated experimental information and new nuclear data evaluation methodologies. The progress of the evaluation methodologies during recently years in China will be introduced in this presentation.



# Surrogate reaction methods for neutron induced cross-sections

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Compound nuclear cross sections for reactions of neutrons and light charged particles with target nuclei across the isotopic chart, taking place at energies from several KeV to tens of MeV, are required for nuclear astrophysics, national security, and nuclear-energy applications [1]. The trans-uranium nuclide produced in the nuclear fuel cycles by successive neutron capture, play prominent role in modeling processes that are relevant to generate energy. The fast-neutron induced reactions have also been proposed for the incineration of actinide materials, notably minor actinide isotopes which are produced in Th-U or U-Pu fuel cycles. Unfortunately, for a large number of reactions the relevant data cannot be directly measured in the laboratory, since the relevant nuclei are often too difficult to produce with currently available experimental techniques or too short-lived to serve as target in present day setups. The Surrogate reaction methods provide access to such nuclear data indirectly.

In recent years, the surrogate reaction methods in various forms have been employed to get indirect estimate of the neutron induced fission reaction cross sections of many compound nuclear systems in actinide region, which are not accessible for direct experimental measurements. We have developed a new experimental technique known as hybrid surrogate ratio method[2] and successfully employed to determine  $^{233}\text{Pa}(n,f)$ ,  $^{234}\text{Pa}(n,f)$ ,  $^{239}\text{Np}(n,f)$ , and  $^{240}\text{Np}(n,f)$  compound nuclear cross sections in the equivalent neutron energy range 10.0 to 16.0 MeV by measuring the ratio of fission decay probabilities in [ $^{232}\text{Th}(^6\text{Li},\alpha)^{234}\text{Pa}/^{232}\text{Th}(^6\text{Li},d)^{236}\text{U}$ ], [ $^{232}\text{Th}(^7\text{Li},\alpha)^{235}\text{Pa}/^{232}\text{Th}(^7\text{Li},t)^{236}\text{U}$ ], [ $^{238}\text{U}(^6\text{Li},\alpha)^{240}\text{Np}/^{238}\text{U}(^6\text{Li},d)^{242}\text{Pu}$ ], and [ $^{238}\text{U}(^7\text{Li},\alpha)^{241}\text{Np}/^{238}\text{U}(^7\text{Li},t)^{242}\text{Pu}$ ] transfer reactions respectively[2-4]. We have also determined  $^{241}\text{Pu}(n,f)$  cross section in the equivalent neutron energy range 11.0 MeV to 18.0 MeV using  $^{238}\text{U}(^6\text{Li},d)^{242}\text{Pu}$  and  $^{232}\text{Th}(^6\text{Li},d)^{236}\text{U}$  transfer reactions employing surrogate ratio method[5].

In this talk, we start with a brief discussion on surrogate reaction methods and present some of the recent results on neutron induced fission cross section measurements carried out by our group and the possibility of extending the measurements for determining (n, $\gamma$ ), (n,2n) and (n,p) reaction cross-sections by surrogate reaction method will also be discussed.

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# Experimental studies on nuclear level density

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## Abstract

The nuclear level density (NLD) is an important physical quantity in the statistical calculation of compound nuclear decay and therefore, useful for both pure and applied research. The variation of the NLD with the mass number (A), excitation energy, angular momentum, parity, isospin, shell effect and paring have been studied both experimentally and theoretically for over many decades. The recent development of the state of the art experimental techniques, namely CERN n-TOF experiment through neutron resonance spectroscopy, Oslo method for study of continuum gamma rays in coincidence with the particles in transfer induced reactions, Mumbai method for the study of continuum particle spectra in coincidence with the particles (ejectiles) in the breakup-fusion reaction, provide opportunities to study many important problems pertaining to the level density. The recent progress in the measurements highlighting various aspect of the nuclear level density is discussed.

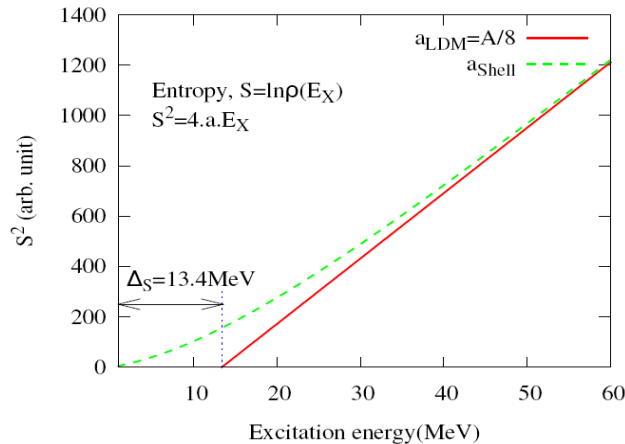
## 1. Introduction to Nuclear Level Density

The NLD is, a fundamental property of the atomic nucleus, defined as the number of energy levels per unit energy at an excitation energy  $E_x$ . It is used for the calculations of reaction rates relevant to nuclear astrophysics, nuclear reactors, and spallation neutron sources. It is also an essential quantity for obtaining the thermodynamical properties of an excited atomic nucleus, namely entropy, temperature and specific heat. The NLD was first calculated by Bethe using a non-interacting Fermi gas model [1,2]. The simplest picture is the equidistant model where the single particle levels are equispaced and non-degenerate. The leading factor in the analytical form is  $\rho(E_x) \sim e^{2\sqrt{a E_x}}$ , where  $a$  is the level density parameter.  $a$  is known as the level density parameter and is given by  $a = \pi^2 g/6$  and  $g$  is the sum of the neutron and proton single particle level density at the Fermi surface. It increases rapidly with excitation energy. The analytic form deduced by Bethe within the frame work of Fermi gas model has been used as the basis for building the most of the phenomenological formula such as constant temperature model (CTM) and Gilbert and Cameron for the fitting of the experimental data or calculating reaction cross section [3-5]. Apart from the thermodynamic approach [1,2] which based on the grand partition function, the other techniques used for the calculating the NLD are: combinatorial method which requires large scale computation for estimating the number of ways the nucleons can be distributed among the available single particle levels and sorting them according to a given  $E_x$  and  $J$ , and spectral distribution method where the energy and angular momentum dependent level density being evaluated from the first few moments of the nuclear Hamiltonian. Recently developed a powerful shell model Monte Carlo (SMMC) method [9], which has been used to calculate the nuclear level density in the presence of correlations. The SMMC method enables calculations in model spaces that are many orders of magnitude larger than those that can be treated by conventional methods. This method was employed to calculate the level densities in nuclei up to  $A \sim 160$  and are in good agreement with the data from various experiments [10].

## 2. Experimental Probes to Study Level Density

The experimental evidences of the nuclear level density are obtained from (a) counting the discrete levels and resonances populated by neutron and charged particle reactions, (b) the inelastic scattering of neutron, proton, alpha and transfer reaction populating the low excitation

energy, (c) the Ericson fluctuation analysis in the compound nuclear reaction which is restricted to light mass nuclei ( $A \leq 60$ ) and (d) the analysis of evaporation spectra. In the first two methods the NLD is obtained by direct counting of the nuclear levels with necessary correction for unresolved and unobserved levels and therefore provide an absolute measure of the nuclear level density. The information of the NLD is limited to near stable nuclei and very low excitation energy (up to binding energy of the nucleon) and angular momentum pertaining to mainly s and p wave. The extrapolated to higher J values is made to estimate the angular momentum summed or total NLD. Recently, CERN has developed a neutron time of flight (n-TOF) facility to investigate nuclear structure at high excitation energies and obtain crucial information on level densities from neutron resonance spectroscopy. Oslo method is used to study the continuum gamma ray spectra following inelastic scattering and transfer reactions, namely ( ${}^3\text{He}, {}^3\text{He}'\gamma$ ) and ( ${}^3\text{He}, {}^4\text{He}'\gamma$ ) and extract the nuclear level density data and gamma ray strength function relevant to nuclear astrophysics [11]. The last method does not provide the absolute value of the NLD, but has been used to study its variation over wide range of the excitation energy and the angular momentum. The excitation energy ( $E_x$ ) and angular momentum (J) dependence of the NLD has been inferred from the statistical model analysis of the measured low energy  $\gamma$ -ray multiplicity gated particle evaporation spectra. An unusual structure has been observed in the angular momentum gated charged particle spectra [12]. The observed enhancement in the extracted NLD in  ${}^{104}\text{Pd}$  with  $E_x$  and J is the first experiment evidence of pairing reentrance in finite hot rotating nuclei [13]. The NLD also depends on the shell effect and collective modes excitation such as rotational and vibrational. The total NLD inferred from various measurements show that on an average the level density parameter  $a$  increases linearly with the mass number of the nucleus as a  $\approx A/8 \text{ MeV}^{-1}$ . This smooth behaviour with respect to mass is due to the liquid drop like properties of the nucleus. However, there is a significant departure from this liquid drop value at shell closures. This departure is the largest for the doubly magic nucleus  ${}^{208}\text{Pb}$ , where the effective  $a$  is as low as  $A/26 \text{ MeV}^{-1}$  at neutron binding energy. This shell effect on the NLD parameter is expected to wash out with excitation energy so that  $a$  approaches its liquid drop value at  $E_x \geq 40 \text{ MeV}$  [14,15] as shown in the figure between entropy square verses excitation energy.

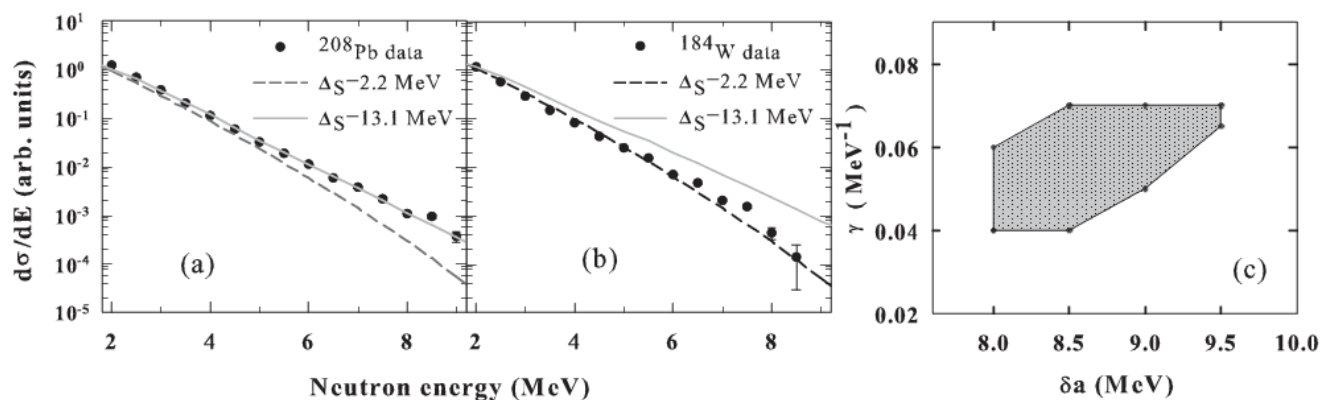


**Figure 1:** Plot,  $S^2$  vs  $E_x$ , shows the washing out of shell effect with excitation energy in the doubly closed shell  ${}^{208}\text{Pb}$  nucleus. The shell correction energy for  ${}^{208}\text{Pb}$  is  $-13.4 \text{ MeV}$ . The solid line is the asymptotic behavior at high excitation energy [14].



### 3. Shell effect and its damping with Ex

The nucleus  $^{208}\text{Pb}$ , formed in the excitation energy range 19 - 23 MeV corresponds to alpha energy from 13.5 - 19.5 MeV, decays predominately by the neutron emission populating the residual nucleus in the Ex  $\sim$  3 - 14 MeV. The statistical model(SM) analysis of the measured neutron spectra shows the expected large shell correction energy ( $\sim$ 13.1MeV) for the nuclei in the vicinity of doubly magic  $^{208}\text{Pb}$  and a small value (2.2 MeV) around  $^{184}\text{W}$  (see figure: 2 (a-b)). An exclusion plot between the damping parameter and the inverse level density parameter  $\delta a$  ( $= A/\tilde{a}$ ) has been made for the first time as shown in Fig. 3(c). It is observed that the acceptable range of  $\delta a$  lies between 8.0 and 9.5 MeV. The shell damping parameter  $\gamma$  constrained to  $(0.060^{+0.010}_{-0.020}) \text{ MeV}^{-1}$  [16].



**Figure 2:** (a) Neutron spectrum from  $^{208}\text{Pb}$  at  $Ex = 20.8$  MeV and solid and dashed lines are the statistical model calculation using shell correction energy ( $\Delta S$ ) 13.1 and 2.2 MeV, respectively, for  $a=A/8.5 \text{ MeV}^{-1}$  and  $\gamma=0.055 \text{ MeV}^{-1}$ . (b) Same as (a) except for  $^{184}\text{W}$  at  $Ex = 20.6$  MeV and (c) the exclusion plot between  $\delta a$  ( $= A/\tilde{a}$ ) and  $\gamma$ . The allowed values of  $a$  and  $\gamma$  are within the contour.

### 4. Summary

The nuclear level density is an important physical quantity for both the basic and applied research. We have understood generic behaviour of the level density as a function of excitation energy and angular momentum within the framework of Fermi gas model. Many experiments reveal correlations which are not included in the Fermi gas model such as pairing, shell effect and also the collective effects. Many correlations are important at low excitation energy for the study of level density are taken care in the shell model Monte Carlo calculations and are limited to the heavy mass nuclei up to  $A \sim 160$ . We have inferred the damping of the nuclear shell effect over a wide excitation energy from an exclusive measurement of the neutron spectra in the  $^7\text{Li}$  breakup followed by fusion of triton with  $^{205}\text{Tl}$ . The measured neutron spectra show a large shell effect in the vicinity of  $^{208}\text{Pb}$ . An exclusion plot between the parameter  $\gamma$  which relates the damping of the nuclear shell effect and the asymptotic nuclear level density parameter  $a$ , has been made for the first time. The precision of the measurement of the damping factor can be improved by using the liquid scintillator detector array for fast neutron measurement and also the Si- strip/CD detector for identification of the light charged particles. In future, the collective enhancement of the nuclear level density and its damping with excitation energy can be addressed using the same experimental technique.

## Acknowledgment

I thank to all my collaborators and members of nuclear reaction group-B for their help during the experiment. Special thanks to V. M. Datar, D. R. Chakrabarty for their valuable discussion during analysis and calculation. We also thank to target laboratory staff for target preparation and Pelletron Linac staff for smooth running of the machine.

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# Neutron-induced activation measurements and EXFOR compilations in the energy range up to 20 MeV

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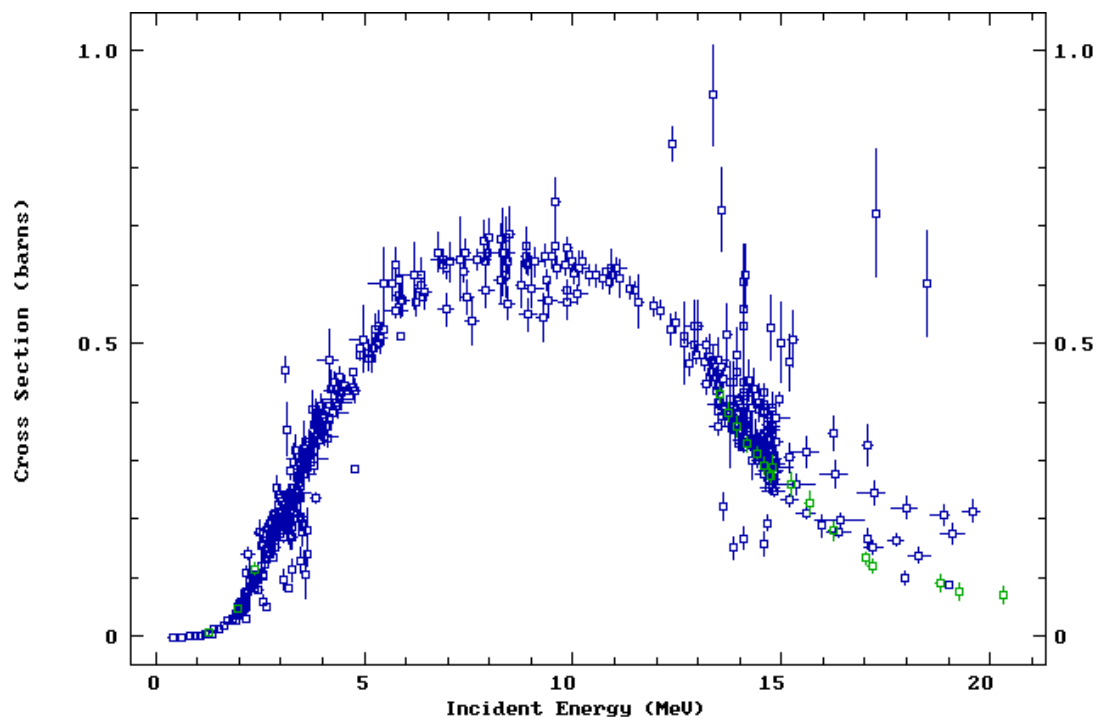
Accurate neutron-induced activation cross-section data are of interest in many fields of science and applications. Such data are needed for calculations and analysis of neutron transport, activation of materials, gas production and radiation damage, dose rates etc. Experimental data provide bases for the parameterization of reaction cross section calculations, and for the assessment of nuclear models and evaluated data libraries. Activation technique in combination with gamma spectrometry is well known and widely used method for neutron-induced reaction cross-section measurements. However, in some cases considerable differences exist between the results from different experiments. A careful consideration of the all factors that may affect each particular measurement is needed in order to obtain reliable data. Measured data are of little value until they are made conveniently available for users and evaluators. The International Network of Nuclear Reaction Data Centres (NRDC) collaborates in collection, compilation and dissemination of experimental nuclear reaction data in the EXFOR data library. In the present work some aspects of the  $^{58}\text{Ni}(n,p)^{58}\text{Co}$  activation cross-section measurements at two different experimental facilities [1,2] and EXFOR compilation files will be presented [3].

## **Irradiation Procedure**

At Institute for Reference Materials and Measurements (IRMM, Geel, Belgium) the  $^3\text{H}(d,n)^4\text{He}$  reaction was used for the production of neutrons in the energy range from 14 to 21 MeV [1] and  $^3\text{H}(p,n)^3\text{He}$  reaction for the production of neutrons from 0.5 to 6 MeV and at the Institute of Experimental Physics (Kossuth University, Debrecen, Hungary) the neutrons in the energy range 13.5 -14.8 MeV were produced via  $^3\text{H}(d,n)^4\text{He}$  reaction [2]. Both reactions as well as  $^2\text{H}(d,n)^3\text{He}$  are considered as sources of monoenergetic neutrons, however low energy neutrons are present above certain energy threshold and also due to neutron production and scattering near the target area. A neutron spectrum unfolding with time-of-flight spectrum as a prior is a solution to make correction for the contribution of the background neutrons to the total measured reaction rate. The method of “spectral indexing” was applied at IRMM for the determination of neutron flux density distribution [4]. It is based on the measurements of the reaction rates induced by the neutrons in monitor reactions with different energy thresholds. Using time-of-flight measurement of neutron spectra the whole energy range is presented in group representation. The flux corresponding to each energy group is determined by method of generalized least squares using the known response characteristics of spectral-index reactions.

The information regarding neutron source reaction and neutron spectrum is included in EXFOR compilations under the keywords INC-SOURCE and INC-SPECT. The experimentalists are also advised to provide numerical data for neutron source spectra [5].

The fluctuations of the neutron source intensity during the irradiations in both experiments were registered by long counter operating in a multichannel acquisition mode and corrections were applied to the studied and monitor reactions

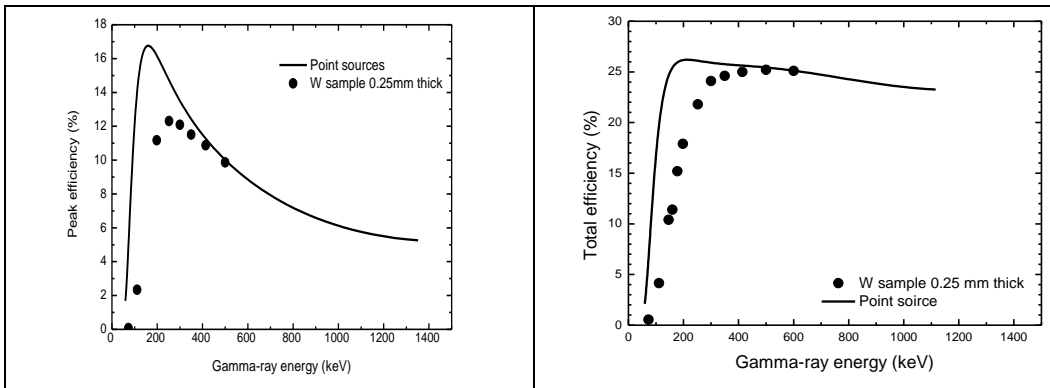


**Figure 1.**  $^{58}\text{Ni}(n,p)^{58}\text{Co}$  reaction cross section. Blue symbols are EXFOR data, green symbols data from [1,2].

### Radioactivity Measurement

HPGe detectors are conventionally used for gamma-rays spectroscopy nowadays. A set of monoenergetic gamma sources was used for the efficiency calibration of the detector in order to avoid coincidence summing effects since most of the measurements was carried out at closed geometry. In addition a model of the detectors was developed for Monte Carlo simulation of the detector response, which allows taking into account the detailed characteristics of the detector and samples (complex shape, sample matrix,  $\gamma$ -ray self-attenuation, volume activity distribution, coincidence summing effects, etc.) and in consequence increases flexibility and accuracy of the measurements [6]. Comparison of detector efficiencies determined by standard sources and MCNP calculations for tungsten sample with 0.25 mm thickness (both for total and peak efficiencies) are presented on Figure 2. Due to the large gamma-ray attenuation in tungsten and the limitation of the self-absorption correction it cannot properly account for the gamma-ray attenuation in the sample and calculated detector response increasingly deviates from the simulations with decreasing of gamma-ray energy.

The decay data and their uncertainties reported by authors are included in EXFOR compilation. Uncertainty propagation of the branching ratios is rather straightforward, however regarding the half-live it depends also on irradiation, cooling and counting time intervals. The compilers are advised to compile this information under the keyword METHOD and code ACTIV if available in the reference. IAEA-NDS also provides an extensive and up to date nuclear structure and decay database through <https://www-nds.iaea.org/livechart/>.



**Figure 2.** Full energy peak (left) and total (right) efficiencies obtained by fitting of data from point standard sources (line) and MCNP calculations for 0.25 thick W sample.

### Data Analysis

Cross sections were calculated using the well-known activation formula. The count rates were corrected for coincidence effects, for  $\gamma$ -ray abundance,  $\gamma$ -ray self-absorption, efficiency of the detector, and measurement geometry, neutron flux fluctuations during the irradiations, and the background neutrons.

Proper reporting of the uncertainties and correlations between partial uncertainties currently are of great interests. EXFOR rules allow compilation of the partial uncertainties and correlations between that in order to provide sufficient information for the construction of covariance matrices.

Example of compilation of discussed experimental details from Ref. 1 in EXFOR entry 22820 is included below.

```

INC-SOURCE (D-T)  $^3\text{H}(d,n)^4\text{He}$ . Deuterons incident on a metal titanium-tritide target (approx.
                2 mg/cm2 thick on a silver backing (0.5 mm thick). This reaction generated
                the primary neutrons with energies in the range 14 to 20.5 MeV. Samples were
                placed at angles from 0 to 75 degrees. The energy scale uncertainty is
                5 keV. Deuteron energies of 1, 2, 3, and 4 MeV were used.
MONITOR (P-T)  $^3\text{H}(p,n)^3\text{He}$  reaction was used to produce neutrons with energies 1 - 3.4 MeV
All relative measurements were ultimately normalized to the primary standard.
(13-AL-27(N,A)11-NA-24,,SIG) Primary standard used directly or indirectly for all
the cross section determinations.
(49-IN-115(N,INL)49-IN-115-M,,SIG) Spectral index component. Very sensitive to
the lowest-energy neutrons.
(13-AL-27(N,P)12-MG-27,,SIG) Secondary standard and spectral index component.
(26-FE-56(N,P)25-MN-56,,SIG) Secondary standard and spectral index component.
(41-NB-93(N,2N)41-NB-92-M,,SIG) Secondary standard and spectral index component.
MONIT-REF (,H.Conde+,R,INDC(SEC)-101,1992) - for the  $^{27}\text{Al}(n,\alpha)^{24}\text{Na}$ 
(,M.Wagner+,B,PH-DAT,13,5,1990) - for  $^{93}\text{Nb}(n,2n)^{92m}\text{Nb}$  reactions
(,A.B.Smith+,R,ANL-NDM-115,1990) - for the  $^{115}\text{In}(n,n')^{115m}\text{In}$  reaction
(,P.L.Rose+,R,BNL-201,1991) ENDF/B-VI - for the  $^{27}\text{Al}(n,p)^{27}\text{Mg}$  and  $^{56}\text{Fe}(n,p)^{56}\text{Mn}$ 
ERR-ANALYS (ERR-T) Total errors for the cross sections were deduced by combining partial
errors from various sources in quadrature. Error sources considered were-
(ERR-S,1,,15.) statistical counting errors;
(ERR-1,2.,3.) uncertainty in detector efficiencies;
(ERR-2,0.01,3.) uncertainties in decay branching factors;
(ERR-3,0.,7.) uncertainty in coincidence summing corrections;
(ERR-4,0.1,5.) uncertainty in the absorption of gamma rays;
(ERR-5,,1.) sample mass error;
(ERR-6,,1.) counting time error;
(ERR-7,0.,12.) uncertainty in low-energy neutron corrections;
(MONIT-ERR,0.5,2.) uncertainties in monitor reactions;
(ERR-8,0.5,2.) neutron irradiation geometry error;
(ERR-9,0.01,4.) uncertainty in decay half-lives
ANALYSIS Cross sections obtained by allowing for complete decay of the isomeric state.

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## Summary

Careful consideration of all factors that may affect particular activation reaction cross section measurement ensures high quality and accuracy of the results. Comprehensive reporting and EXFOR compilations are important for comparison and further analysis of the data from different experiments.

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# Overview of EXFOR activities in India

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## Abstract

Regular EXFOR compilations have been started in India since 2006 through few voluntary compilers from Universities and through EXFOR workshops conducted every two years under the supervision of Nuclear Data Physics Centre of India (NDPCI) and Nuclear Data Section, IAEA. Such efforts have produced a remarkable outcome in terms of managing the nuclear data in the country. A brief history of EXFOR activities, along with the present status and future plan is discussed in brief.

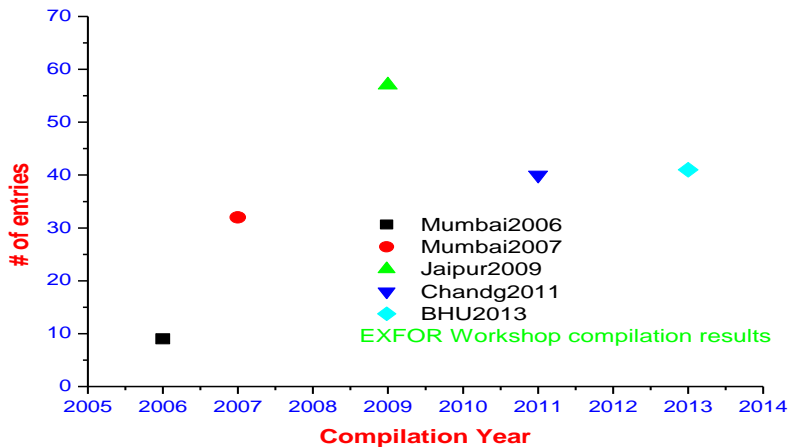
## 1. Brief History of EXFOR activities in India

In India, EXFOR compilation on a regular basis has been started since 2006. Compilation was done by Voluntary compilers and through EXFOR theme meetings. Since the past few years, EXFOR compilations have also been done by Universities through funds given by NDPCI-BRNS. All EXFOR compilations are done under the supervision of NDS, IAEA (Previously with the help of Dr. O.Schwerer, Dr. S.Dunaeva and currently with Dr. N. Otsuka). EXFOR workshops have been conducted successfully since 2006 in an effort to motivate young researchers and students into nuclear data, and Table 1 gives detail about the DAE-BRNS EXFOR workshops in the past.

**Table 1:** DAE-BARC theme meeting on EXFOR compilation of nuclear data

| Sl. No. | Year   | Place                         | Experts from NDS, IAEA          |
|---------|--|-------------------------------|---------------------------------|
| 1       | 4 <sup>th</sup> – 8 <sup>th</sup> September, 2006. | BARC, Mumbai                  | Dr. Otto Schwerer               |
| 2       | 29 <sup>th</sup> – 2 <sup>nd</sup> October, 2007   | BARC, Mumbai                  | Dr. S. Dunaeva                  |
| 3       | 3 – 7 <sup>th</sup> November, 2009                 | Jaipur University, Jaipur     | Dr. S. Dunaeva<br>Dr. N. Otsuka |
| 4       | 4 - 8 <sup>th</sup> April, 2011                    | Chandigarh, Punjab University | Dr. S. Dunaeva<br>Dr. N. Otsuka |
| 5       | 18-22 <sup>nd</sup> February, 2013                 | BHU, Varanasi                 | Dr. S. Dunaeva<br>Dr. N. Otsuka |

The number of EXFOR entries created in the EXFOR workshops in India is more than over 200 entries. The Fig.1 gives the number of EXFOR entries created in each of the EXFOR workshops conducted in the country.



**Fig.1:** Number of EXFOR entries created in the EXFOR workshops in India

For the time being, we adopted Russian EXFOR editor and Japanese Digitizer Gsys. This will continue and will be adopted in the Bangalore workshop which will be held in January 2015.

In the past, few voluntary compilers actively involved in the regular EXFOR compilation, such as

- Dr. Ranjita Mandal from IIT Kharagpur
- Dr. Paresh Prajapati, M.S University
- Dr. M.Bhike BARC
- Mr. Uday and Kalyan from VECC under Dr. Gopal Mukherjee

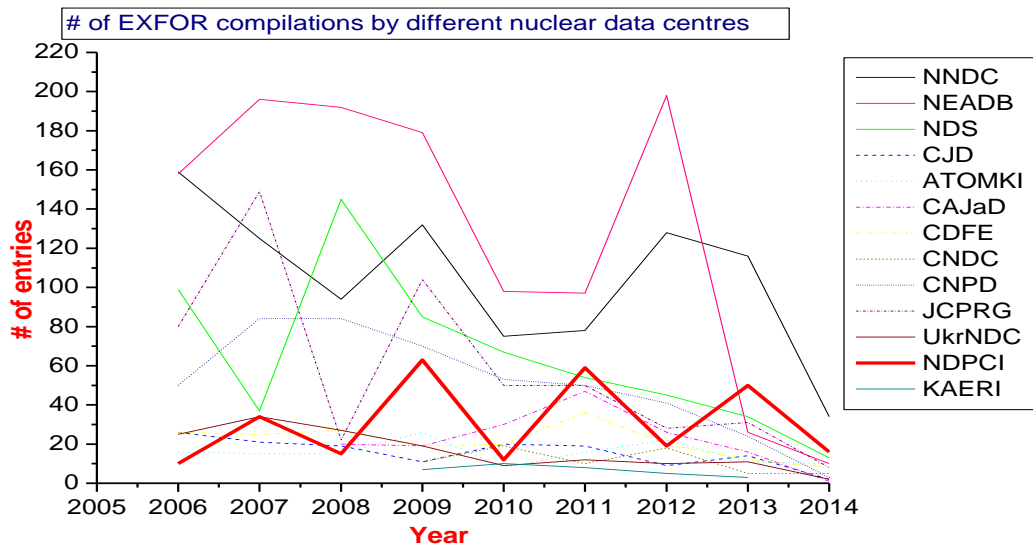
Currently, these compilers no longer engage in regular compilation. Although they still compile their own research work.

## 2. Present Status of EXFOR Activities in India

Currently, all EXFOR activities in India are supervised by NDS, IAEA (Dr. N. Otsuka) right from article assignment to the compilers till finalization of the entry. Dr. B. Lalremruata acts as National coordinator under NDPCI for the EXFOR activity supervising the individual compilers till submission of the entry to NDS, IAEA and finalization. As a result of such efforts, between 2013 and 2014 NRDC meetings, NDPCI transmitted 55 entries. This is larger than NDS, ATOMKI, CNDC, KNDC and UkrNDC (see P2014-06) and also CAJaD and JCPRG (WP2014-02)[1]. The 30th INDC meeting (2-6 June 2014) recognizes this achievement very well as we see in the draft of the recommendation from INDC WG2 the following statement: "We appreciate the significant compilation effort by India by organising a training workshop on a regular basis and suggest continuous support by the NDS".

Therefore NDS may continue to support the Indian workshop (as well as support to the NDPCI regular EXFOR compilation activity). As of now, 270 entries have been transmitted and a few dozen more are in line for transmission. Fig.2 shows the number of entries created by nuclear data centres worldwide.





**Fig.2:** Number of EXFOR entries created by different nuclear data centres

Currently there are two groups who are actively engaged in day to day/regular compilation through funds given by NDPCI, BRNS.

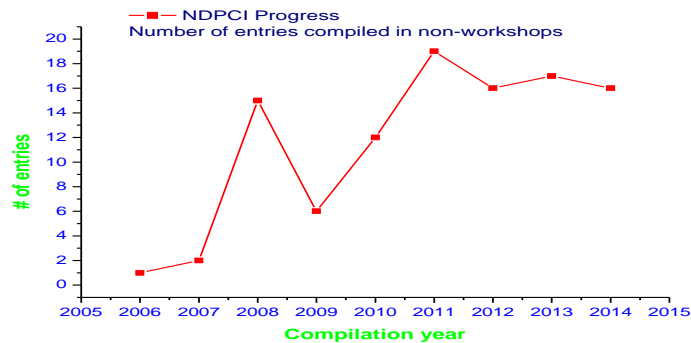
1) NEHU, Shillong:

- Prof. B.Jyrwa
- Ms. Sylvia Badwar
- Ms. Reetuparna Ghosh

2) Mizoram University, Aizawl:

- Dr. B. Lalremruata
- Mr. Lalrinmawia Punte(JRF)
- SRF post will be filled in the first week of October.

These two groups also actively engaged in the nuclear data measurements which are included in the Project in collaboration with BARC scientist. Fig.3 shows the number of EXFOR entries created by regular compilers through non-workshop activities in India.



**Fig.3:** Number of EXFOR entries created from non-workshop activities in India.

### **3. Assessment of Old Neutron Articles missing in EXFOR**

Dr. Vidya Thakur was invited in BARC, in 2014 to check old missing articles through CINDA. Vidya found 30 (mainly neutron through CINDA) articles for compilation during her one month visit at BARC. Assessment was also completed for Pramana. For IPA (Indian J. Pure and Appl. Phys.), scanning has not been done yet for Vol.1-20 of IPA as well as Indian J.Phys. (IJP) till 2002. This scanning can be done only by experts (who know what may be compiled in EXFOR), and Vidya may be invited to BARC again to continue scanning. Out of these 30 articles, we have completed 12 entries (33048, 33049, 33050, 33061, 33062, 33063, 33064, 33065, 33066, 33067, 33068, 33069.). While compiling 33065, we also found missing articles from the reference and these articles will be compiled into EXFOR 1) O. N. Koul, Nuclear Physics 29 (1962) 522, 2) O. N. Koul, Nuclear Physics 33 (1962) 177, 3) O. N. Koul, Nuclear Physics 39 (1962) 325.

### **4. Current Status of New Articles**

As of now, there are 28 new articles published in 2013-2014 (could be more now) for compilation at the next Indian EXFOR workshop. For 19 articles, numerical data were received from the authors and for few of them, numerical data are available in tabular form in the article themselves, thanks to Dr. Alok, Dr. Gopal and Dr. Otsuka. The numerical data provided by the authors help us to avoid compilation of data digitized by non-experienced workshop participants. Most probably, we will compile all the new articles published in 2013-2014 in the Bangalore workshop in Jan, 2015.

### **5. Benefits of EXFOR Workshops in India**

- 1) Each workshop had 40-50 participants where basics in EXFOR are introduced by experts.
- 2) Hands on experience in the EXFOR compilation using the EDITOR and Digitizer is done under the supervision of experts.
- 3) Large numbers of entries are completed in a one week period of time from diverse participants, MSc students, Research Scholars, Young as well as Senior Professors and Scientist.
- 4) Many participants after attending such workshops offer to compile their own research papers.
- 5) Due to this, the number of entries in which numerical data is entered increased since the authors themselves prepare the entry.
- 6) Even in the case where the experimentalists do not compile their article, there are increasing numbers of cases where the authors directly submit their numerical data to NDPCI or NDS directly.

### **6. Future Activities**

- 1) Preparation for the next EXFOR workshop in Bangalore (January 2015)
- 2) Scanning of journals for old articles missing in EXFOR database.
- 3) Probably all new articles will be managed by EXFOR workshop every two years. But regular compilation (and completeness checking of IPA, IJP etc.) needs to be continued to compile all known old articles for a while. Sometimes there are articles reporting large number of data, which are not suitable for compilation during the workshop.
- 4) Awareness and encouragement among Indian participants and researchers to check the EXFOR database and see if all their published data are included in the database. If not,

they should be encourage to inform the NDPCI. Such encouragement would also be helpful in DAE-BRNS symposiums. In the near future, NDPCI will manage the EXFOR compilation by its own, and prepare trans file for entries published by Indian Researchers.

#### **Reference**

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# Filtered thermal neutron captured cross sections measurements and decay heat calculations

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## Abstract

Recently, a pure thermal neutron beam has been developed for neutron capture measurements based on the horizontal channel No.2 of the research reactor at the Nuclear Research Institute, Dalat. The original reactor neutron spectrum is transmitted through an optimal composition of Bi and Si single crystals for delivering a thermal neutron beam with Cadmium ratio ( $R_{cd}$ ) of 420 and neutron flux ( $\Phi_{th}$ ) of  $1.6 \times 10^6$  n/cm<sup>2</sup>.s. This thermal neutron beam has been applied for measurements of capture cross sections for nuclide of <sup>51</sup>V, by the activation method relative to the standard reaction <sup>197</sup>Au(n, $\gamma$ )<sup>198</sup>Au. In addition to the activities of neutron capture cross sections measurements, the study on nuclear decay heat calculations has been also considered to be developed at the Institute. Some results on calculation procedure and decay heat values calculated with update nuclear database for <sup>235</sup>U are introduced in this report.

## I. Introduction

The first object of this works is to measure the thermal neutron capture cross sections on the new thermal filtered neutron beam at the channel No.2 of Dalat research reactor. The measurements were carried out by the activation method, relative to the standard capture cross section of <sup>197</sup>Au. The thermal filtered neutron is tailored by using a combination of 80cm Si and 6cm Bi single crystals. The second object is to calculate decay heat data for fission products from a thermal neutron fission reaction of <sup>235</sup>U. In this study, a computation procedure has been improved for calculating the decay and buildup of fission products following time after a fission reaction or a fission process. The method used in this calculation is numerical analysis, in which the buildup and decay of fission product nuclides are analyzed by exactly analysis of the general solutions of the Bateman's Equations [1] for every full complex decay chain. Based upon the input data of nuclear decay and fission yield data from JENDL4.0 [2], the concentration of each nuclide as a function of cooling time is determined.

## II. Filtered Thermal Neutron Capture Cross Section Measurements

The measurements for neutron capture cross sections of <sup>51</sup>V(n, $\gamma$ )<sup>52</sup>V reaction were performed on the thermal filtered neutron beams at the Dalat reactor. The neutron beams were collimated to 3cm in diameter with neutron flux of  $1.6 \times 10^6$  n/cm<sup>2</sup>.s, and the value of Cadmium ratio  $R_{Cd}(Au)$  is 420. The thermal neutron capture cross sections,  $\langle \sigma_0 \rangle^x$ , for nuclide x can be determined relative to that of <sup>197</sup>Au(n, $\gamma$ )<sup>198</sup>Au standard reaction by the following relations:

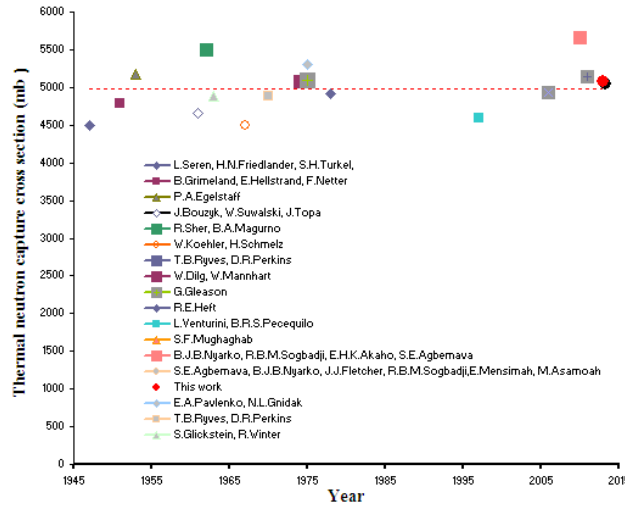
$$\langle \sigma_a \rangle = \frac{C^x f(\lambda, t)^x f_c^x I_\gamma^{Au} \epsilon_\gamma^{Au} N^{Au} \langle \sigma_a \rangle^{Au}}{C^{Au} f(\lambda, t)^{Au} f_c^{Au} I_\gamma^x \epsilon_\gamma^x N^x} \quad (1)$$

$$f(\lambda, t) = \frac{\lambda}{(1 - e^{-\lambda t_1}) e^{-\lambda t_2} (1 - e^{-\lambda t_3})} \quad (2)$$

where C is the net counts of the corresponding gamma peak; the superscript 'x' denotes the nucleus of sample; t<sub>1</sub>, t<sub>2</sub> and t<sub>3</sub> are irradiating, cooling and measuring times, respectively. The

symbol  $\lambda$  denotes the decay constant of the product nucleus,  $\varepsilon_\gamma$  the detection efficiency of detector,  $I_\gamma$  the intensity of interesting  $\gamma$ -ray.  $f_c$  is the correction factor which is account for self-shielding and multiple scattering of neutron in the irradiated sample and standard. The standard thermal neutron capture cross section of  $^{197}\text{Au}$  is adopted from the reference [3]. The correction factors for the neutron self-shielding, multi-scattering were calculated by Monte-Carlo method using MCNP5 code [4].

The result of measurement in this work for thermal neutron capture reaction of  $^{51}\text{V}(n,\gamma)^{52}\text{V}$  is  $\sigma_0 = 5076 \pm 151$  (mb), and the comparison with previous experimental data extracted from EXFOR data base [5] is shown in Fig.1.



**Fig. 1.** The experimental thermal neutron capture cross section of  $^{51}\text{V}$

### III. Decay Heat Calculations

The number of the nuclide  $i^{\text{th}}$  at cooling time  $t$  after a fission burst can be calculated from the following formula:

$$N_i(t) = N_i(0) \exp(-\lambda_i t) + \sum_{j \neq i}^M N_{j \rightarrow i}(t), \quad (3)$$

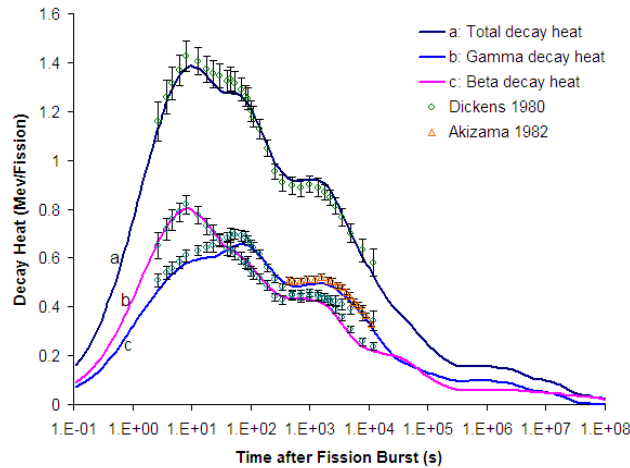
in which,  $N_i(0)$  is equal to the independent fission yield of the nuclide  $i^{\text{th}}$ , and the part of  $N_{j \rightarrow i}(t)$  is the build-up number of nuclide  $i^{\text{th}}$  at cooling time  $t$ , that was formed in the system of decay chains originated from the nuclide  $j^{\text{th}}$ . This term of build-up number can be obtained by analysis the general solutions of the Bateman's equation for every particular linear decay chain. In this work, we developed a numerical algorithm to calculate the term of  $N_{j \rightarrow i}(t)$  in equation (3) directly by using the decay data file and fission yield data from evaluated nuclear structure data libraries. In general case for a linear decay chain, the quantity of  $N_{j \rightarrow i}(t)$  is calculated by using the solution of Bateman's equation as following expression.

$$N_{j \rightarrow i}(t) = \sum_{l=1}^i \prod_{k=l}^{i-1} N_l(0) \left( \sum_{m=l}^i \frac{e^{-\lambda_m t}}{\prod_{\substack{j=l \\ j \neq m}}^i (\lambda_j - \lambda_m)} \right) \quad (4)$$

The computer program called DHP (Decay Heat Power) [6] was applied with the present procedure to perform the above mentioned calculation tasks. In this calculations, all of decay chains and decay modes including  $\beta^-$  decay to ground, first and second isomer states, double  $\beta^-$  decay, electron capture decay to ground and isomer states, alpha and proton decay, delay neutron  $\beta^-$  decay, and internal transitions in the fission product system are taken into consider in the calculated procedure. The summation model for decay heat calculations is as the following function:

$$f(t) = \sum_{i=1}^M \overline{E}_i \lambda_i N_i(t), \quad (5)$$

where:  $M$  denotes the maximum number of FP nuclides;  $E_i = E_{i\beta} + E_{i\gamma}$  stands for the mean energy per decay of the  $i$ th nuclide,  $\lambda_i$  the decay constant,  $N_i(t)$  the corresponding concentration function for cooling time  $t$ .  $f(t)$  is the burst function of decay heat (MeV/Fission/s). The physical quantity equal to  $t*f(t)$  is called decay heat power function (MeV/Fission). In this work, the JENDL FP Decay Data File 2011 [7] and fission yield data file from JENDL 4.0 [2] have been applied for calculations of fission product concentrations as functions of cooling time after thermal neutron fission reactions of  $^{235}\text{U}$ . The results of calculations for  $t*f(t)$  as a function of times after fission burst are shown in Figure 2 in comparison with measured values by Dicken [8], and Akizama [9].



**Fig. 2.** Decay heat of fission products from thermal neutron fission reaction of  $^{235}\text{U}$

#### IV. Summary

In this report, we present the activities of thermal neutron capture cross section measurements and decay heat calculations at the Nuclear Research Institute of VINATOM. The thermal neutron capture cross section of  $^{51}\text{V}(n,\gamma)^{52}\text{V}$  reaction has been measured by using the filtered neutron beam at the Dalat research reactor. The decay heat calculation for fission products from thermal neutron fission of U-235 was performed with the DHP code based on the fission yield and decay data files from JENDL-4.0.

#### Acknowledgement

The authors would like to express their thanks to the Nuclear Data Section of IAEA and the Department of Atomic Energy (DAE) for their valuable supports in financial and organization to this workshop. One of authors was supported by Vietnam National Foundation for Science and

Technology Development (NAFOSTED) under grant number “103.04-2012.59” for his travelling expense to participating in this workshop.

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# Decay Heat Calculations for Reactors: Development of a Computer Code ADWITA

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## Abstract

Estimation of release of energy (decay heat) over an extended period of time after termination of neutron induced fission is necessary for determining the heat removal requirements when the reactor is shutdown, and for fuel storage and transport facilities as well as for accident studies.

A Fuel Cycle Analysis Code, ADWITA (Activation, Decay, Waste Incineration and Transmutation Analysis) which can generate inventory based on irradiation history and calculate radioactivity and decay heat for extended period of cooling, has been written. The method and data involved in Fuel Cycle Analysis Code ADWITA and some results obtained shall also be presented.

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## Decay Heat Generation in a Reactor

The heat generated in an operating reactor is effectively removed such that thermal equilibrium and energy balance is maintained within prescribed limit. When the reactor is shutdown the radioactive materials that remain in the reactor at the time of shut down continue to decay and release energy. This requires effective cooling must be maintained to keep reactor core from damage due to overheating. Decay heat generated in a reactor just after cessation of neutron fission chain reaction (scram) is about 6% of the operating thermal power which comes down to about 0.1% after 40 days. [Fig-4] This means even after cessation of neutron chain reaction the cooling system must perform with efficiency not less than 6% of that at full power.

## Calculation of decay heat in reactors

The method of decay heat estimation relies on the measurements over practical time intervals as well as on calculation for predictions over very long time intervals. Neutron cross sections, fission yields and decay data together with operational history are the basic inputs to such calculations. A code used to calculate decay heat is required to generate isotopic inventory that would be present at the shutdown, based on operational history of the reactor and follow up the decay heat generation over an extended period of time.

The decay heat calculation is done based on standard prescription and algorithm for such calculations. The American National Standard for Decay Heat Power in Light Water Reactors (the ANS standard) ANS-5.1 is one of such standard. Other standards for the decay heat calculation are the JAERI standard (for both LWRs and FBRs), JAERI-M-91-034 and the German standard, DIN 25463.

The decay standard prescribes fission product decay heat power and its uncertainty for a fission pulse and for infinite reactor operation (usually an irradiation of  $10^{13}$ s represents infinite irradiation). Decay heat power from activation products in reactor materials is not specified in the standard and thus the decay heat power is related to the operating power of the reactor only via the fission rate and the recoverable energy per fission during operation. The decay data standard is based on summation calculation and experimental data. Due to scarce experimental data for decay times longer than  $10^5$  seconds the standard largely depends on calculated data for decay times beyond  $10^5$  to  $10^9$  seconds respectively. [1]

The reference [3] describes in detail the methodology of decay heat calculation based on interpolation of decay heat tables according to interpolation parameters such as (a) heat generation rate of the assembly, (b) cycle and cycle times of the assembly, (c) fuel burnup of the assembly, (d) specific power of the fuel, (e) assembly cooling time. The decay standards state explicitly the domain of parameter values to which it can be applied and prescribe correction if the interpolation parameters do not fall in to the applicability domain because of any of the following, like cooling time shorter than that used for making table, operating power larger than used in table, enrichment different from that in the table.

### **Codes for decay heat generation calculation**

The codes for decay heat calculation are used extensively for the safety assessments of all types of nuclear plant, handling of fuel discharges, the design and transport of fuel-storage flasks, long time management of the radioactive waste and thus have far reaching consequence on safety and sustainability of nuclear energy. [2] The codes used for the decay heat calculation for the licensing purposes use the decay data tables according to the regulatory requirement. Such codes are based on realistic decay heat measurements and details of the operational history. The ORNL-LWRARC is such a code. [1,3]

The other kind of decay heat calculation codes are required to generate the inventory at the shutdown based on irradiation history before embarking on decay heat calculation based on radioactive transitions. The example of such codes is ORIGEN and its versions [4, 5], CINDER [6], and DCHAIN [7]. The inventory generation and subsequent decay heat calculation in such codes depends on the reactor specific neutron flux dependent cross section library and a decay data library. ADWITA (Activation Decay Waste Incineration and Transmutation Analysis) is an inventory generation and decay calculation code based on matrix exponential method [4, 5] and series solution of generalised Bateman equation for the transmutation chains through Transmutation Trajectory Analysis (TTA). [8, 9]

### **Cross section data libraries**

Reactor physics design codes usually needs to consider limited number of nuclides which are required for the spectrum and reactivity calculations but decay data codes require reactor specific flux dependent transition rates for exhaustively large number of nuclides which are vital not only for the reactor design calculation but also for the backend process of the fuel cycle. The reactor physics codes usually use about few hundred nuclides explicitly while discharge fuel inventory include few thousands nuclide. The cross section data for the point reactor fuel cycle code used for the generation of inventory and decay data can be obtained from the spectrum calculation code used for the reactor design. The space time averaged cross section data for the nuclides which are explicitly used in spectrum calculation can be obtained for the use in fuel cycle analysis code by flux-volume weighting of self-shielded time dependent microscopic cross section.[10] Table 1 and 2 present capture and fission cross section for some isotopes calculated for PHWR220 library used by code ADWITA. The data for the nuclides, which are not used explicitly in the spectrum codes, can be obtained by weighting the point data with the spectrum specific to the region of the isotope of interest. As example the isotopes belonging to set of fission product can be collapsed with the spectrum specific to the fuel region. Likewise isotopes prevailing in in structural materials or used as soluble poison in coolant or moderator can use spectrum prevailing in those regions. Figure 1 shows a typical BOC (Beginning of Cycle)spectrum prevailing in the average pin, three rings of fuel and in homogenized lattice cell in PHWR220 19 rod cluster geometry as obtained in 172 energy groups WIMS convention. The data for the nuclides which cannot be obtained from the spectrum collapsing because of lack of complete set of data (in the form of ENDF file) but have piece wise spectrum specific measured or calculated data can be obtained by weighting of such piecewise data keeping in to account the difference between the spectrum for which data is reported and the spectrum of interest for which data is to be obtained. The EXFOR (Experimental Nuclear Reaction Data), CINDA (Computer Index of Nuclear Reaction Data) and NSR (Nuclear Science References) databases (all available online at <https://www-nds.iaea.org/>) are rich source of such measured piecewise data.

The fission product yield data is required for the inventory generation codes. The individual and cumulative fission product for fission by different incident energy of neutron is given in MT454 and MT459 of file8 in ENDF/B data file. The fission yield of a nuclide I due to fission of nuclide J is required to be multiplied with the fission cross section of nuclide J to obtain corresponding transition matrix element. This requires the fission product yield must correspond to the neutron spectrum for the system of interest for which fission cross section data has been obtained. The library for code ADWITA uses individual fission yields from MT454. The ENDF/B-VII.0 release includes fission product yields for fission of 31 actinides. As of now, explicit fission yields from fission of 6 actinides, namely  $^{232}\text{Th}$ ,  $^{233}\text{U}$ ,  $^{235}\text{U}$ ,  $^{238}\text{U}$ ,  $^{239}\text{Pu}$  and  $^{241}\text{Pu}$  is used in

## Decay Data Library

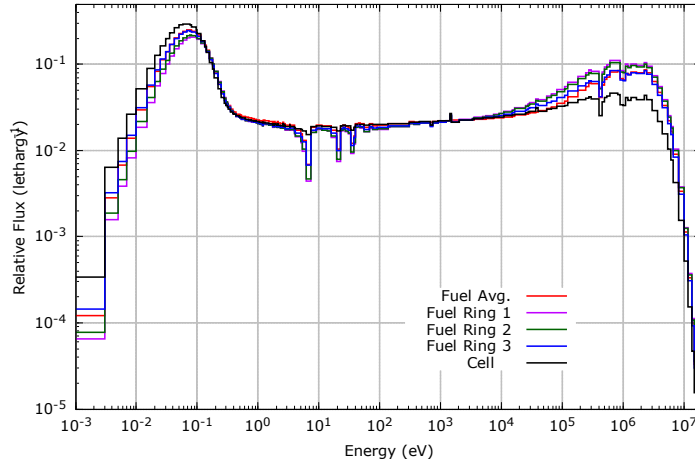
The calculation of transitions and heat generation due to radioactive transitions requires a decay data library with details like decay half-life, branching ratios and energy release (Q value) associated with the decay. Library should also provide isotopic abundances of isotopes and toxicities if output in terms of elemental composition and corresponding toxicities are required. As of now code ADWITA do not support elemental compositions as input and output, nor does it give toxicity. The nuclear wallet card from BNL and MT457 in file8 of ENDF/B-VII.0 release, which is basically ENDF translation of BNL nuclear wallet card database, is the source of decay data file.

Table 1 Some actinide cross section data for PHWR220 library (barn)

| Nuclide | (n,f)    | (n,g)   |
|---------|----------|---------|
| Th-232  | 0.015    | 3.534   |
| U-232   | 26.390   | 20.870  |
| U-233   | 154.150  | 15.650  |
| Pa-233  | 0.048    | 27.480  |
| U-234   | 0.306    | 35.100  |
| U-235   | 150.580  | 27.280  |
| U-236   | 0.136    | 5.340   |
| U-237   | 0.433    | 126.640 |
| U-238   | 0.062    | 1.100   |
| Np-237  | 0.337    | 59.150  |
| Np-239  | 0.374    | 28.600  |
| Pu-238  | 4.930    | 132.290 |
| Pu-239  | 260.300  | 122.160 |
| Pu-240  | 0.360    | 141.830 |
| Pu-241  | 321.260  | 111.465 |
| Pu-242  | 0.262    | 29.800  |
| Pu-242  | 0.262    | 27.570  |
| Am-241  | 2.331    | 214.440 |
| Am-242  | 2105.220 | 407.730 |
| Am-243  | 0.295    | 57.090  |
| Cm-242  | 0.995    | 6.570   |
| Cm-243  | 193.590  | 36.710  |
| Cm-244  | 0.744    | 15.144  |

Table 2 Some LLFP capture cross section in PHWR220 library

Figure1 Typical PHWR-220 Spectrum



| Isotope   | PHWR (Calc.) |
|-----------|--------------|
| 43-Tc-99  | 11.54        |
| 46-Pd-107 | 2.82         |
| 48-Cd-113 | 12393.2      |
| 55-Cs-137 | 0.084        |
| 55-Cs-135 | 3.54         |
| 62-Sm-151 | 3495.27      |
| 63-Eu-155 | 1624.97      |

The code ADWITA can generate inventory based on irradiation or burnup history and continue with the decay of inventory for extended period of time. In one of the input files the name of the cross section file, name of the decay data file together with burnup history is required to be given. In other input file isotopic composition in grams of the nuclides in all three categories, namely activation products, actinides and daughters, and fission products, needs to be given. The code generates three output files, one each for the activation products, actinides and daughters, and fission products, which contain isotopic composition in

The figures 2 through 4 present some typical results for a 19 rod cluster 220 MWe Indian PHWR obtained from the code ADWITA for burnup of one ton natural uranium up to 10 GWD/t with power density 19.77 MW/t and subsequent decay. The nuclide ID in Fig 2 is Z\*10,000+A\*10 of the isotope.

Figure2 Actinide compositions vs. burnup

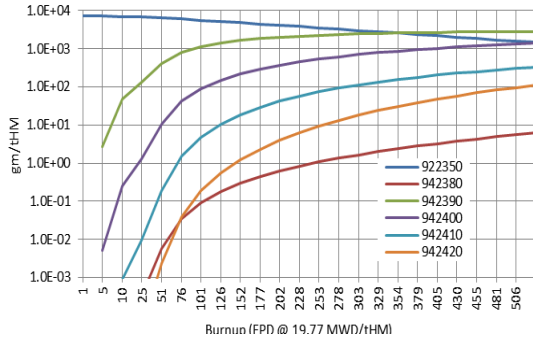


Figure 3 Decay heat generation rate Vs. cooling time

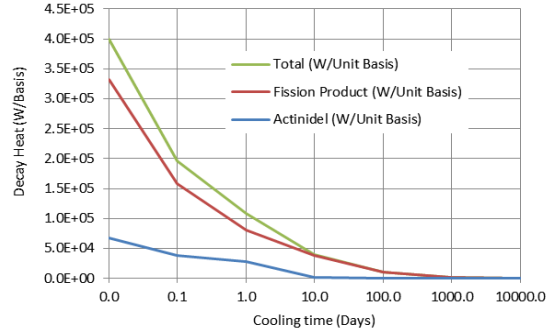
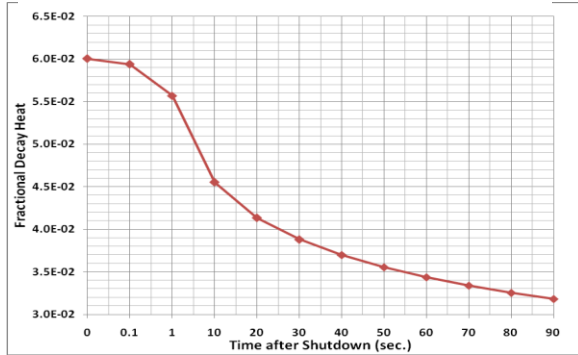


Figure 4 Fractional Decay heat generation after shutdown



The tables 3, 4 and 5 below gives some results obtained from ADWITA and that from reference 15. The table 3 gives the amount of <sup>235</sup>U and <sup>239</sup>Pu at 0, 4000, and 8000 MWD/t burnup as obtained from ADWITA and that given in reference 15 [p 26-27].

Table 3 <sup>235</sup>U and <sup>239</sup>Pu content [15 P 26-27]

| Burnup                   | 0.0 GWD/t | 4.0 GWD/t | 8.0 GWD/t |
|--------------------------|-----------|-----------|-----------|
| <sup>235</sup> U (g/kg)  |           |           |           |
| PPV-Canada               | 7.1138    | 3.7859    | 1.9957    |
| WIMS-Romania             | 7.1137    | 3.7465    | 1.9606    |
| CLU B-India              | 7.1138    | 3.7962    | 2.0195    |
| ADWITA                   | 7.1138    | 3.8290    | 2.0520    |
| <sup>239</sup> Pu (g/kg) |           |           |           |
| PPV-Canada               | 0.0       | 1.9412    | 2.4136    |
| WIMS-Romania             | 0.0       | 2.0949    | 2.7282    |
| CLU B-India              | 0.0       | 1.9309    | 2.4895    |
| ADWITA                   | 0.0       | 2.0811    | 2.6486    |

The table 4 and 5 are the radioactivity and decay heat generated by fission products during cooling up to 1.0E10 sec after a burnup of 7500 MWD/t [15, p102-103]. The difference in result can be due to difference in method, data libraries, and specific power used for the burnup. In case of ADWITA an specific power density of 33.155 MWt/tHM has been used to attain a burnup of 7500 MWD/t. Please refer to reference 15 for the details about quoted results.

Table 4 Radioactivity (Curies/Ton) of Fission Products [15, P102]

| Time (s)  | 0.0E00   | 5.0E00   | 1.0E01   | 1.0E02   | 1.0E03   | 1.0E04   | 1.0E05   | 1.0E06   | 1.0E07   | 1.0E08   | 1.0E10   |
|-----------|----------|----------|----------|----------|----------|----------|----------|----------|----------|----------|----------|
| Argentina | 1.50E+08 | 1.39E+08 | 1.34E+08 | 1.08E+08 | 7.43E+07 | 4.75E+07 | 2.86E+07 | 1.38E+07 | 3.48E+06 | 2.38E+05 | 8.13E+01 |
| Pakistan  | 1.51E+08 | 1.39E+08 | 1.34E+08 | 1.09E+08 | 7.54E+07 | 4.85E+07 | 2.84E+07 | 1.37E+07 | 3.49E+06 | 2.47E+05 | 1.20E+02 |
| ADWITA    | 1.49E+08 | 1.33E+08 | 1.27E+08 | 9.77E+07 | 6.96E+07 | 4.54E+07 | 2.80E+07 | 1.34E+07 | 3.32E+06 | 2.24E+05 | 5.66E+01 |

Table 5 Thermal Power (Watt/Ton) of fission products [15, P 103]

| Time (s)  | 0.0E00   | 5.0E00   | 1.0E01   | 1.0E02   | 1.0E03   | 1.0E04   | 1.0E05   | 1.0E06   | 1.0E07   | 1.0E08   | 1.0E10   |
|-----------|----------|----------|----------|----------|----------|----------|----------|----------|----------|----------|----------|
| Argentina | 1.68E+06 | 1.49E+06 | 1.40E+06 | 1.06E+06 | 6.08E+05 | 2.82E+05 | 1.40E+05 | 6.26E+04 | 1.44E+04 | 9.16E+02 | 1.97E-01 |
| Pakistan  | 1.69E+06 | 1.49E+06 | 1.40E+06 | 1.06E+06 | 6.10E+05 | 2.83E+05 | 1.37E+05 | 6.16E+04 | 1.42E+04 | 8.46E+02 | 2.63E-01 |
| ADWITA    | 1.78E+06 | 1.46E+06 | 1.35E+06 | 8.98E+05 | 5.36E+05 | 2.55E+05 | 1.25E+05 | 5.66E+04 | 1.31E+04 | 7.97E+02 | 1.41E-01 |

It is evident from table 4 and 5 the values obtained from ADWITA are in general smaller than the result quoted in the reference 15. The reason for such occurrence can be either difference in methodology or difference in library and cross section or both. The reading of the reference suggests the methodology as well as data is different for all the three cases. The code ADWITA uses custom library made for the PHWR220 from the most recent data releases while the reference result has apparently used modified LWR cross section library.

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# Overview of ENSDF activities in India

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## Introduction

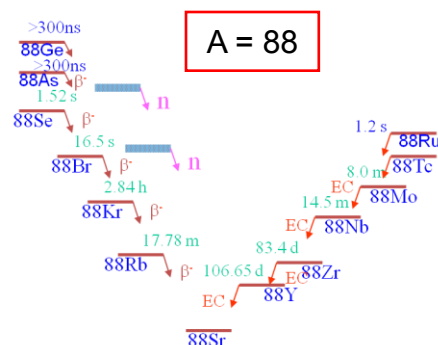
Arguable, the nuclear data evaluation activity is a subject of more than 80 years old and was started by the evaluation of nuclear structure data [1]. Nuclear Structure and Decay Data (NSDD) evaluation in the ENSDF (Evaluated Nuclear Structure and Decay Data File) format is considered as one of the important activities worldwide under the aegis of International Atomic Energy Agency (IAEA). India has now become one of the active members in the Nuclear Structure and Decay Data (NSDD) network and has contributed regularly and significantly for more than 10 years. Before that the Indian contribution to this activity was sparse and irregular.

There are two main data bases for NSDD: (a) The ENSDF (Evaluated Nuclear Structure Data File) database, which contains *evaluated* nuclear structure and decay information for nuclides. The file is updated on a continuous basis and the evaluations are published in the journal Nuclear Data Sheets. The ENSDF evaluation is coordinated by the National Nuclear Data Centre, BNL, USA [2] and the Nuclear Data Section, IAEA, Vienna, Austria [3] and (b) The XUNDL (eXperimental Unevaluated Nuclear Data List) database which contains experimental data *compiled* from recent nuclear structure papers. The XUNDL database has been established in 1998 and is coordinated by McMaster University, Canada.

There are several quantities which are included in the evaluated nuclear structure and decay data file. These include (i) disintegration energies, (ii) decay modes, (iii) excitation energies (iv) radiation and transition probabilities, (v) nuclear level schemes, (vi) spin-parity values of a nuclear state, (vii) magnetic and electric multipole moments, (viii) nuclear band structure and (ix) half-lives. These quantities are very important to the users from several applied disciplines, which include reactor physics, radiation protection, medical science, agriculture, archeology apart from the researches in the field of basic research.

## Mass (A) Chain Evaluation

A complete ENSDF evaluation is the evaluation of different parameters mentioned above with comments on the basis of the adopted values of all the nuclei in a mass parabola as shown in the adjacent figure. That is, these parameters are evaluated for each nucleus and are grouped as isobars i.e all the nuclei under the same mass number and a set of this evaluation is called a mass chain evaluation. There is varied number of nuclei in a particular mass chain. As an example, the  $A = 88$  mass chain contains 13 nuclei.



**Fig 1:** An example of nuclei in a mass (A) chain evaluation

## Datasets in a Mass Chain Evaluation

For every nucleus in a mass chain, there can be several types of datasets, depending on the different ways of producing the ground and the excited states in that particular nucleus. For example, the ground and excited states can be produced by  $\beta$ -(or  $\alpha$ -) decay, by light or heavy-ion induced fusion evaporation reaction, by scattering or transfer reaction, as a fission product, as an isomer decay (IT) etc. Moreover, different reactions in fusion evaporation, scattering and transfer reactions are also put as different datasets. From these individual datasets, an adopted data set is created by the evaluator for each nucleus. An adopted dataset is the most important dataset which contains all the *recommended* levels, gammas etc. for a particular nucleus. The number of datasets can be as high as 50 for a particular nucleus. There can be several published data for each datasets. Therefore, one needs to consult several papers for the evaluation of a particular nucleus. The evaluated data of several such nuclei constitute a mass chain evaluation. This indicates that the mass chain evaluation is a very huge job and must be done very carefully considering that the adopted datasets are used in application in most cases.

## ENSDF File Format

The ENSDF file is created with a standard 80-character format as shown in Fig.2. In this figure, the types of parameters corresponding to different rows and the record length for a particular parameter in that row are shown.

| ENSDF Standard 80-character Formatted Records |       |       |       |           |           |       |       |     |       |       |     |       |       |       |       |        |       |     |   |    |     |    |       |      |     |       |    |   |    |    |    |   |    |   |   |   |  |  |
|---|-------|-------|-------|-----------|-----------|-------|-------|-----|-------|-------|-----|-------|-------|-------|-------|--------|-------|-----|---|----|-----|----|-------|------|-----|-------|----|---|----|----|----|---|----|---|---|---|--|--|
|   | 1     | 2     | 3     | 4         | 5         | 6     | 7     | 8   |       |       |     |       |       |       |       |        |       |     |   |    |     |    |       |      |     |       |    |   |    |    |    |   |    |   |   |   |  |  |
| Record  | 1     | 5     | 6     | 7         | 8         | 9     | 0     | 9   | 0     | 1     | 2   | 9     | 0     | 1     | 2     | 9      | 0     | 1   | 2 | 9  | 0   | 5  | 6     | 0    | 2   | 3     | 4  | 5 | 0  | 4  | 5  | 6 | 7  | 8 | 9 | 0 |  |  |
| IDENT   | NUCID | &     | blank | DSID      |           |       |       |     |       |       |     |       |       |       |       | DSRLET |       |     |   |    | PUB |    |       | DATE |     |       |    |   |    |    |    |   |    |   |   |   |  |  |
| XREF  | NUCID | blank | X     | DSID      |           |       |       |     |       |       |     |       |       |       |       | blank  |       |     |   |    |     |    |       |      |     |       |    |   |    |    |    |   |    |   |   |   |  |  |
| REF   | AAA   | blank | R     | KEYNUMBER | REFERENCE |       |       |     |       |       |     |       |       |       |       |        |       |     |   |    |     |    |       |      |     |       |    |   |    |    |    |   |    |   |   |   |  |  |
| HIST  | NUCID | &     | b     | H         | HTEXT     |       |       |     |       |       |     |       |       |       |       |        |       |     |   |    |     |    |       |      |     |       |    |   |    |    |    |   |    |   |   |   |  |  |
| Q-VALUE                                       | NUCID | blank | Q     | Q-        | DOE       | SN    | DSN   | SP  | DSP   | QA    | DQA | OMIT  |       |       |       |        |       |     |   |    |     |    |       |      |     |       |    |   |    |    |    |   |    |   |   |   |  |  |
| G COMM  | NUCID | &     | †     | #         | CTEXT     |       |       |     |       |       |     |       |       |       |       |        |       |     |   |    |     |    |       |      |     |       |    |   |    |    |    |   |    |   |   |   |  |  |
| F/R COM                                       | NUCID | &     | †     | #         | SYMLAG    | CTEXT |       |     |       |       |     |       |       |       |       |        |       |     |   |    |     |    |       |      |     |       |    |   |    |    |    |   |    |   |   |   |  |  |
| PARENT  | NUCID | blank | P     | S         | E         | DE    | J     |     |       |       |     |       |       |       |       |        |       |     | T |    |     | DT | blank | QP   | DOE | -ION- |    |   |    |    |    |   |    |   |   |   |  |  |
| NORM  | NUCID | blank | N     | S         | NR        | DNR   | NT    | DNT | BR    | DBR   | NB  | DNB   | NP    | DNP   | blank |        |       |     |   |    |     |    |       |      |     |       |    |   |    |    |    |   |    |   |   |   |  |  |
| P NORM  | NUCID | &     | P     | S         | NR*BR     | JNC   | NT*BR | JNC | blank | NB*BR | UNC | NP    | DNP   | blank |       |        |       |     |   |    |     |    |       |      |     |       |    |   |    |    |    |   |    |   |   |   |  |  |
| LEVEL   | NUCID | &     | b     | L         | bl        | E     | DE    | J   |       |       |     |       |       |       |       |        |       |     |   | T  |     |    | DT    | L    |     |       | S  |   |    | DS | MS | Q |    |   |   |   |  |  |
| BETA  | NUCID | &     | b     | B         | bl        | E     | DE    | IB  | DIB   | blank |     |       | LOGFT | DFT   | blank |        |       |     |   |    |     |    |       |      |     |       |    |   |    | UN | Q  |   |    |   |   |   |  |  |
| EC  | NUCID | &     | b     | E         | bl        | E     | DE    | IB  | DIB   | IE    | DIE | LOGFT | DFT   | blank |       |        |       |     |   |    |     |    |       |      |     |       |    |   | UN | Q  |    |   |    |   |   |   |  |  |
| ALPHA   | NUCID | &     | b     | A         | bl        | E     | DE    | IA  | DIA   | HF    | DHF | blank |       |       |       |        |       |     |   |    |     |    |       |      |     |       | bl | Q |    |    |    |   |    |   |   |   |  |  |
| PART  | NUCID | &     | b     | *         | E         | DE    | IP    | DIP | ED    | T     |     |       | DT    | L     |       |        | blank |     |   |    |     |    |       |      |     |       |    |   |    |    |    | C | bl | Q |   |   |  |  |
| GAMMA   | NUCID | &     | b     | G         | bl        | E     | DE    | RI  | DR    | M     |     |       | MR    | DMR   | CC    | DCC    | TI    | DTI | C | bl | Q   |    |       |      |     |       |    |   |    |    |    |   |    |   |   |   |  |  |

- Notes:**
- blank These fields must be blank.
  - & Primary record must have a blank or "1" in this field. Continuation records should have any printable ASCII character except for blank or "1" (one).
  - ! Unique alphanumeric character identifying the source data set.
  - † Allowed characters for this field are C, c, D, d, T, and t.
  - # Character identifying the record being commented on. Allowed characters for this field are N, P, Q, L, G, B, E, A, D, and blank.
  - ‡ Must be blank except for: 1) Particle code for a (delayed-)particle record. 2) Sequence number for normalization and parent records.
  - § Must be blank except when there are multiple parent records then this field should contain an integer relating the parent record to the related normalization record.
  - ¶ Byte 8 must either be blank for a prompt particle radiation or D for a delayed particle radiation. Byte 9 identifies the particle (N, P, D, or T).

**Fig.2:** The ENSDF File format



## Nuclear Structure Data Activities in India

The activities of the nuclear structure data community in India are mainly, (i) Mass (A) chain evaluation, (ii) Horizontal Evaluation, (iii) Decay data evaluation and (iv) Measurements. The Indian community has significant contribution in each of these categories.

The mass chain evaluation is one of the main nuclear data activities in India under Nuclear Data Physics Centre of India (NDPCI). Several groups from all over the country are involved in this activity. There are about 8 different groups from Universities as well as research laboratories contribute to the mass chain evaluation. Since 2005, about 15 mass chain evaluations have been successfully completed and published. A few other mass chain evaluations are currently in progress. A workshop on evaluation of nuclear structure and decay data has been organized recently and was very successful in the way that several participants have attended the lectures and the practical sessions. In the practical sessions, the  $A = 215$  mass chain was started to be evaluated which was subsequently successfully completed and published [5].

The horizontal evaluation means the evaluation of the best values of one or a few selected nuclear parameters for many nuclides irrespective of their mass number. Indian contributions to horizontal evaluations have started long ago and has been continuing [6]. The horizontal evaluations are mainly in the field of rotational and quasi-particle structure of nuclei. One of the recent horizontal evaluations is on Magnetic Rotational bands which is one of the most important topics of current interest in nuclear physics. Another very important horizontal evaluation on nuclear isomers has been completed recently and will be published soon.

Because of the importance of the decay data, special attention is given to the evaluation of the nuclear decay data. In the decay data evaluation, parameters like, (i) Half-lives, (ii) Q-values, (iii) Branching fractions, (iv) Energies, (v) Emission probabilities and (vi) Atomic data, etc are evaluated from the published data for the decay of the nuclei by  $\alpha$ ,  $\beta$ ,  $\gamma$ , electrons (Auger, conversion) and X-rays. The decay data evaluation of  $^{233}\text{U}$  and  $^{229}\text{Th}$  has been completed as a part of the IAEA CRP on "Updated Decay Data Library for Actinides" [7]. The decay data of each of these nuclei contained about 50 excited states and about 200  $\gamma$ -rays along with other related parameters and these nuclei have significance to the Th-U fuel cycle [8].

In the measurement part, Indian researchers have produced significant and important nuclear structure data using the Indian National Gamma Array (INGA) [9]. Besides those, specific measurement as a part of data evaluation has also been carried out [10]. Recently a highly granular setup for the measurement of Total Absorption Spectroscopy (TAS) has been tested using BaF<sub>2</sub> detectors at VECC, Kolkata [11]. This can be used for the TAS measurement to study the  $\beta$ -decay feeding intensities for the nuclei which contribute significantly in the reactor decay heat.

## Summary

The importance of the evaluation of nuclear structure and decay data has been emphasized in this presentation. Some of the details of the ENSDF evaluation have been discussed. India has been included as one of the centres of NSDD network and several evaluators from India have contributed significantly to the ENSDF evaluation. The interest of the researchers from India in the nuclear structure data evaluation include the mass chain evaluations, decay data evaluation, horizontal evaluation as well as specific measurements related to important nuclear data. There have been several significant contributions in these fields from India. The interest in the ENSDF evaluation and measurements among the nuclear physicists in India is growing.

## Acknowledgement

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- [11] G. Mukherjee et al., *EPJ Web of Conferences* **66**, (2014) 11026; DOI: <http://dx.doi.org/10.1051/epjconf/20146611026>

# Total neutron cross section measurements of $^{93}\text{Nb}$ on filtered neutrons beams at Dalat Research Reactor

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## Abstract

The total neutron cross sections of  $^{93}\text{Nb}$  were measured on filtered neutron beams of 24 keV and 133 keV by neutron transmission technique at the Dalat Research Reactor. The purity of these beams is better than 93% and the neutron flux at sample position were  $6.2 \times 10^5$  and  $3.3 \times 10^5$  n.cm<sup>-2</sup>.s<sup>-1</sup>, respectively. A proton recoil spectrometer has been setup to experimental determine these neutron spectra and to measure total neutron cross sections. The measured values of the cross sections were  $8.01 \pm 0.16$  (b) and  $9.57 \pm 0.13$  (b) at the energies of 24 keV and 133 keV, respectively. The obtained results with uncertainties less than 2.0% were compared with evaluated data from ENDF/B-VII and JENDL 4.0 and with data from previous measurements.

## I. Introduction

Neutron beams from nuclear research reactors are ideal facilities for experimental researches on nuclear reaction data [1]. The neutron filter technique was applied for introducing quasi-monoenergetic neutrons of 24 keV, 54 keV, 59 keV, 133 keV and 148 keV at the horizontal channel No. 4 of Dalat Research Reactor in order to implement neutron total and radiative capture reaction cross sections measurements [2]. In this paper, total neutron cross sections of  $^{93}\text{Nb}$  have been measured on filtered neutron beams of 24 keV and 133 keV by a neutron transmission technique based on a LND-281 recoil proton counter.

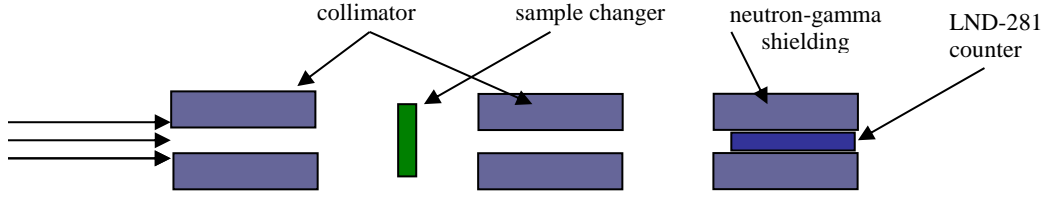
## II. Experimental Setup

The total neutron cross section measurements of  $^{93}\text{Nb}$  was performed on the filtered neutron beams of 24 keV and 133 keV at Dalat Research Reactor. The physical characteristics of these beams are given in Table 1.

**Table 1.** The characteristics of the filtered neutron beams.

| $E_n$<br>(keV) | $\Phi$<br>(n.cm <sup>-2</sup> .s <sup>-1</sup> ) | I<br>(%) | Filter components  |
|----------------|--|----------|--|
| 24             | $6.2 \times 10^5$                                | 98.3     | 0.2g/cm <sup>2</sup> $^{10}\text{B}$ + 20cm $\text{Fe}$ + 30cm $\text{Al}$ + 35g/cm <sup>2</sup> $\text{S}$  |
| 133            | $3.3 \times 10^5$                                | 92.9     | 0.2g/cm <sup>2</sup> $^{10}\text{B}$ + 50g/cm <sup>2</sup> $\text{Cr}$ + 10cm $\text{Ni}$ + 60cm $\text{Si}$ |

The neutron beams were collimated to 10 mm in diameter by using the usual materials of LiF, Cd, B<sub>4</sub>C, Pb and borated paraffin. The LND-281 counter was properly shielded with borated paraffin and lead against scattered neutrons and gamma-rays. The Nb targets were prepared from metallic pellets with diameter of 20 mm and the thicknesses of the samples used in the transmission experiments were 0.04167, 0.08667, 0.14167, 0.22834 and 0.27001 at/b, respectively. The experimental set-up for neutron transmission measurements is shown in Fig. 1.



**Figure. 1.** Neutron transmission system at the beam port of DRR

### III. Data Processing

The total neutron cross sections were measured by neutron transmission technique. For a target of thickness  $N$ , the experimental total neutron cross section is determined by the formula:

$$\sigma_t = \frac{1}{N} \ln \frac{1}{T} \quad (1)$$

where  $N$  is number density (at/b) of the target nuclei and  $T$  is the neutron transmission ratio defined experimentally as:

$$T = \frac{a - a^b}{a_0 - a_0^b} \quad (2)$$

where  $a_0$  and  $a$  are the neutron count rates of the incident and transmitted beams,  $a_0^b$  and  $a^b$  are the corresponding backgrounds.

The statistical error of the transmission  $\Delta T$  was determined by simple error propagation:

$$\frac{\Delta T}{T} = \sqrt{\frac{(\Delta a)^2 + (\Delta a^b)^2}{(a - a^b)^2} + \frac{(\Delta a_0)^2 + (\Delta a_0^b)^2}{(a_0 - a_0^b)^2}} \quad (3)$$

and the error in the cross section  $\Delta \sigma_t$  for each sample thickness is calculated by

$$\Delta \sigma_t = \frac{1}{N} \frac{\Delta T}{T} \quad (4)$$

Experimental cross sections  $\sigma_t$  vary with the thickness of the target due to self-shielding effect and can be expressed by a linear function [2].

$$\sigma_t = \langle \sigma_t \rangle - ax \quad (5)$$

Therefore, the average total cross section  $\langle \sigma_t \rangle$  can be obtained by extrapolating cross sections to zero target thickness.

### IV. Results and Discussion

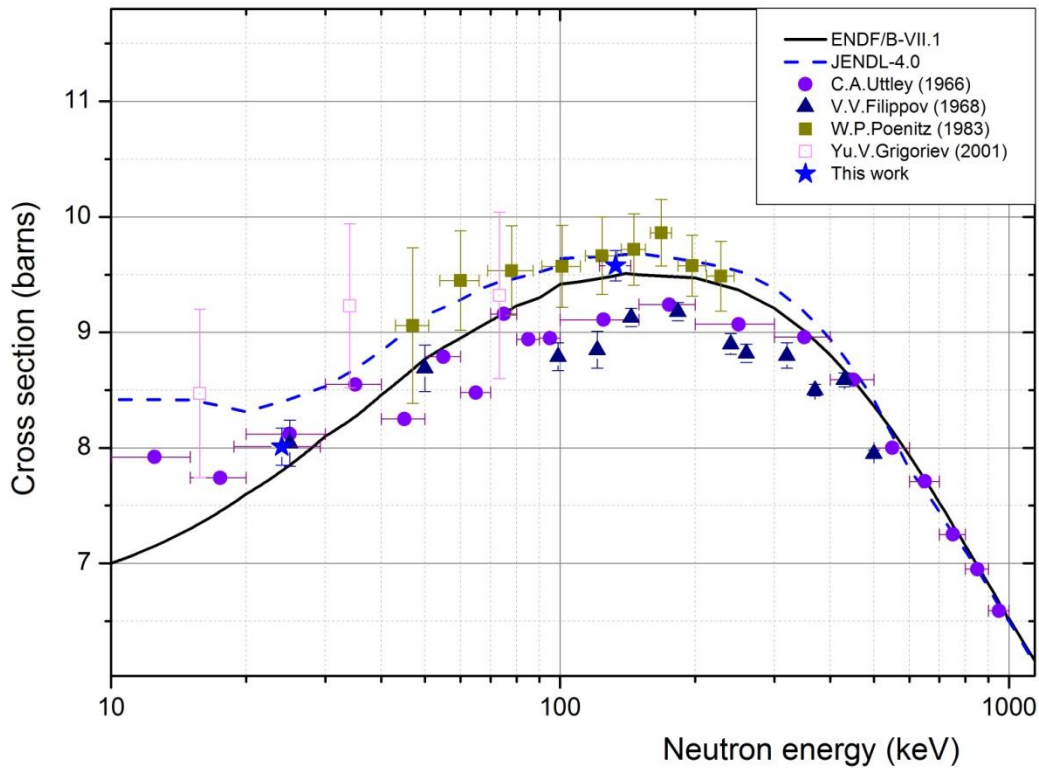
The average total neutron cross sections of  $^{93}\text{Nb}$  have been measured on filtered neutron beams of 24 keV and 133 keV by a neutron transmission technique at the Dalat Research Reactor. The obtained data were given in Table 2. In comparing the evaluated data with the present results at energies of 24 keV and 133 keV, the experimental values differ by 0.8% - 2.7% from ENDF/B VII.0 and by 0.8% - 5% from JENDL 4.0 evaluated cross sections, respectively.

At the energy of 24 keV, the present measured value was 0.4% - 1.4% lower than the previous measurement data of author V.V.Filippov [3] and C.A.Uttley [4]. The comparisons of total neutron cross sections obtained in our work with the values of other authors are shown in Fig. 3.

The statistical errors are about 1-2% and could be further reduced by increasing the number of measuring cycles. The main uncertainties are due to statistics (2%), background (<1%) and the error of sample weight (<0.05%).

**Table 2.** Total neutron cross sections of  $^{93}\text{Nb}$  in keV region (barn).

| Average neutron energy (keV) | ENDF/B VII.0 | JENDL 4.0 | Present         |
|------------------------------|--------------|-----------|-----------------|
| $24 \pm 5.2$                 | 7.80         | 8.42      | $8.01 \pm 0.16$ |
| $133 \pm 10.7$               | 9.49         | 9.67      | $9.57 \pm 0.13$ |



**Figure 3.** Total neutron cross section of  $^{93}\text{Nb}$  in keV region.

## VI Conclusion

The neutron filter technique has been effectively applied to providing mono-energetic neutron beam lines with qualified characteristics for basic research and related applications at the Dalat Nuclear Research Institute. The total neutron cross sections of  $^{93}\text{Nb}$  have been measured at neutron beams of 24 keV and 133 keV by mean of neutron transmission technique using a neutron spectrometer based on a proton-recoil counter. The obtained results with the accuracy less than 2% are in good agreement with those of the other authors and evaluated data.

## **V. Acknowledgments**

The authors would like to express their sincere thanks to the operation staffs of Dalat Nuclear Research Institute for their cooperation concerning to neutron irradiations. This research is funded by Vietnam National Foundation for Science and Technology Development (NAFOSTED) under grant number “103.04-2012.59”.

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# Nuclear data covariances in the Indian context

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## Abstract

The write-up briefly describes the progress achieved thus far, in the Indian context, in the learning curve in the subject area of nuclear data covariances.

## Introduction

The great importance in itself of nuclear data science in Bhabha's 3-stage nuclear programme [1] was institutionally recognized [2] by the Department of Atomic Energy in 2004 due to KAPS over power transient incident and due to several other signals that came simultaneously from operating and post-irradiation experiences on isotopic compositions in a number of cases. A phase transition in the mind set indeed took place in a systemic context towards recognizing the importance of nuclear data science for energy and non-energy applications. In the context of Indian nuclear power programme described in the official website, the author perceives that the importance and motivation in generating covariance error matrices in nuclear data has also been well recognized [3].

## Nuclear Data Physics Experiments in India

It is meaningless to give measured data without a complete description of uncertainties. Under the assumption of quadratic loss, the estimate of the expected values of the nuclear data (say, a reaction cross section) and their covariances together provide a complete description of specification of errors. Scientific data is complete only with the specification of uncertainties in measurements, whether these data are known to have immediate applications or not. NDPCI has concerns on addressing issues of assessment in uncertainties in nuclear data and its impact as part of design and safe operation of nuclear systems.

An illustrative reference list of recently performed nuclear data physics experiments is presented by Alok Saxena in this AASPP-5 event [4]. But covariance analyses were not performed in the analysis of these experiments, except in the one case [5]. It is expected that, in future, more attention will be given to covariance analysis in all the Indian basic nuclear physics experiments.

## EXFOR and Covariances

The activities of the NDPCI in the IAEA EXFOR related activities are briefly described by A. Saxena [4] in these proceedings. It may be noted that there is a strong need and a recognized importance for following up developments in covariances with respect to NDPCI-NRDC requirements [6, 7] of coding of uncertainty data in Indian nuclear data physics experiments into the IAEA EXFOR database. This aspect of data formats for covariances in EXFOR needs nurturing and training in covariance domain expertise to satisfy EXFOR related knowledge, as a mandate of NDPCI as obligation to the NRDC-the international network of nuclear reaction data centers network. The EXFOR coding tasks demand, by design of EXFOR system, a deep technical knowledge on the part of EXFOR compilers but not a judgement about the quality of nuclear data being coded into the EXFOR format. There is awareness in the NDPCI that domain

expertise and human resource development in covariances are required to help young EXFOR compilers in India for doing justice to EXFOR activities of the NDPCI, NRDC.

### **NDPCI Project on Nuclear Data Covariances**

A Phase-1 project in collaboration with the Statistics department in Manipal University, Karnataka (Prof. K.M. Prasad and Prof. S. Nair) on nuclear data covariances was executed successfully. In Phase-1, the emphasis was on a thorough basic understanding of the concept of covariances including assigning uncertainties to experimental data in terms of partial errors and micro correlations, through a study and a detailed discussion of open literature, such as, for instance, in Refs [8-13]. During the Phase-1 stage, the NDPCI has successfully conducted three national theme meetings sponsored by the DAE-BRNS, in 2008 (Manipal University, Karnataka), 2010 (Vel-Tech University, Chennai, Tamilnadu) and 2013 (BARC, Mumbai) on nuclear data covariance. These meetings considerably helped sensitize the topic of covariances. In order to give an idea to the interested reader on the content in these workshops, we mention below some of the topics as covered in these workshops.

- Basic probability, concept of error, error propagation and micro-correlations.
- Linear systems, least squares methods.
- An overview of uncertainties in measurements in nuclear data physics.
- Peele Pertinent Puzzle
- Generalized Least Squares using generalized Inverse
- Examples and exercises: Covariances and error propagation, sandwich error formula.
- Generation of covariance matrix using partial uncertainties and micro-correlations
- Objective Bayesian theory
- Modelling of Nuclear Reactions: Basic Tools and Methods
- Nuclear data sensitivity coefficients in reactor physics: Adjoint flux and exact perturbation theory
- A look at the ENDF formats on covariances
- The Full Bayesian Evaluation Technique - Method and Application
- Student seminars: Exercise on ZPR-6-7 fast reactor assembly; propagation of errors in  $K_{\text{eff}}$  using sandwich formula
- Numerical data of partial errors and micro-correlations in the IAEA EXFOR database
- Total Monte Carlo Approach for uncertainty propagation studies
- A critical look at the current status in India on progress in variance-covariance error matrix in nuclear data physics for advanced energy and non-energy applications.

These NDPCI workshops on covariances were instrumental in our improving basic understanding and in appreciating several difficult issues in the topic of covariances that were otherwise not straightforward to comprehend for beginners and researchers with the traditional background. For instance, the basic concept of nuclear data evaluation, new to the Indian system, such as the one which would provide output covariance data that are consistent with the cross-section evaluation that weights input data with the inverse of its variance-covariance matrix was discussed in these workshops. Also, it was stressed in these workshops that the documentation on Indian nuclear data experiments should attempt to have enough detail to allow full determination of the required input in terms of attributes, partial errors and micro correlation matrices.



Towards the end of Phase-1, a first time covariance analysis of cross-sections for  $^{58}\text{Ni}$  (n, p)  $^{58}\text{Co}$  reaction measured in Mumbai Pelletron accelerator has been attempted as part of a PhD programme [5, 14] during the 2007-'14 period. This attempt is a new initiative in India in nuclear data covariances. In this experiment,  $^7\text{Li}$  (p,n) reactions were used as neutron source in the MeV energy region. We measured the  $^{58}\text{Ni}$  (n, p)  $^{58}\text{Co}$  reaction cross-sections relative to two monitors. The two reference monitors employed are the cross-sections for the formation of  $^{97}\text{Zr}$  fission product in neutron induced fission of  $^{232}\text{Th}$  and  $^{238}\text{U}$ , at effective neutron energies  $E_n = 5.89, 10.11$  and  $15.87$  MeV. We illustrated, in detail [5] the generation and combination of covariance matrices (using partial uncertainties and micro-correlations) in relative measurements at each of the various stages starting from the efficiency calibration of HPGe detector, ratio of  $^{58}\text{Ni}$  (n, p)  $^{58}\text{Co}$  reaction cross-section relative to monitor cross-section and in the process of normalization. We further illustrate the weighted averaging of equivalent data as applicable in relative measurements. The necessary data, the involved attributes and the corresponding table of partial uncertainties as required for compilation in the EXchange-FORmat (EXFOR) database are presented [5]. The attempt is unique in India as it is directed towards describing error propagation, step by step, in the article [5]. The paper [5] and the thesis [14] are expected to be useful for researchers and EXFOR compilers who want to get an idea of the analytic approach and complexities in the error propagation methodology in such activation measurements. This experience in generating a 3 by 3 covariance matrix also helped us appreciate the guidelines in the literature such as the one, for instance, the one as advised by Mannhart [13]: "Of course, such ratio measurements can be normalized by the experimenter, yet it is strongly recommended that the stated covariance matrix does not include this normalization as it is information not based on experimental facts".

A Phase-II DAE-BRNS-NDPCI proposal of project on nuclear data covariances at the Department of Statistics, Manipal University has been evolved. In Phase-2, modern nuclear data evaluation techniques will be further studied as a research and development effort, as a first time effort. The proposal of a Phase-2 project on covariances at Manipal University is currently under peer-review.

Another parallel effort since last 5 months is the just started and on-going study of the use of Kalman filter techniques in nuclear data assimilations by the author in collaboration with Prof. Ms. Jayalakshmi Nair, as part of Master Degree projects being done by Ms. Sana Khan and Mr. Harshavardhan Kadvekar, two students in Master of Engineering, Instrumentation & Control (Batch 2013-2015) at the Vivekanand Education Society's Institute of Technology (VESIT), Mumbai.

### **Nuclear Data Sensitivity Studies for Reactor Applications and Inferences**

Sensitivity studies to nuclear data are an essential part of uncertainty propagation studies. These have been covered in the presentation by Umasankari Kannan [15]. As illustrative examples, we mention a few case studies. Sensitivity studies such as the analysis [16] of coolant void reactivity of Advanced Heavy Water Reactor through isotopic reaction rates, contribution [17] to the isotopic and energy groupwise dependence of fuel temperature coefficient of reactivity in natural uranium fuelled PHWRs and uncertainty analysis for safety parameters in thorium-LEU fuelled Advanced Heavy Water Reactor, are some of the interesting studies. The fission spectra effect in WIMSD conventions due to fissions in isotopes of thorium fuel cycle was studied by a sensitivity study [18] in a recent research paper. Rigorous and exact perturbation theory based tools are in development to address these issues. A need has been felt

in the Indian system to graduate to the required industrial level of being able to generate and use covariances with response functions to obtain error margins due to uncertainties in nuclear data.

### **Nuclear Data Needs of Fast Reactors, Operating Reactors and Integral Nuclear Data Validation Studies**

Pandikumar [19] discussed the demands on the nuclear data needs for fast reactor including covariances and their ENDF/B formats in this AASPP-5 workshop. Fernando [20] discussed the perspectives of a plant operator on the nuclear data needs for the Indian nuclear power program in these AASPP-5 Proceedings. In one of the operating Indian Pressurized Heavy water Reactors, the 220MWe KAPS-1 reactor, an over power transient incident had occurred in 2004 and sent out a strong signal to the Indian nuclear community that generically, whether U-Pu or Th-U fuel cycle, the design manuals of even the operating nuclear systems should be updated every few years as and when improved nuclear data becomes available.

E. Radha [21] has briefly covered the Indian criticality benchmarking activities in these Proceedings. India became a high quality contributor [to the experimental nuclear criticality benchmarks of the International Criticality safety Benchmark Evaluation Project (ICSBEP) of the US-DOE/NEA-DB since 2005. India has, thus far, contributed 3 experimental criticality benchmarks, KAMINI (2005), PURNIMA-II (2007) and PURNIMA-I (2012). These integral benchmarking efforts gave awareness interestingly with a deep insight on the uncertainty in the reactor system characterization that is known as the “benchmark uncertainty”. It was a learning process that was very successful. Generating covariances in measured integral data are expected to be taken up in the future. There is also a need to benchmark operating and higher burnup states, in addition to the existing international practice of benchmarking zero power critical facilities.

A programme of study of propagation of uncertainties in the form of variance-covariances in nuclear data physics in relation to target accuracies and sensitivity studies is of great importance [22] to Indian nuclear programme for energy and non-energy (e.g., medical) applications. As an example, let us consider a commercial fast breeder reactor with  $\text{PuO}_2\text{-UO}_2$  fuel that is expected to have a breeding ratio (BR) of say, 1.2. An uncertainty of 3% due to nuclear data on BR is 0.036. The breeding gain (BG) is proportional to BR-1. Therefore the uncertainty in BG is  $0.036/0.2 = 0.18$ , that is 18%. Energy growth rate is affected, by this alone, by 18%. Consider another situation: The  $^{232}\text{Th}(n,2n)$  reaction though occurs above 6.3MeV. Even in thermal reactors where the flux above 6MeV is just about 0.05%, the reaction produces a few hundreds of ppm of  $^{232}\text{U}$  in thorium fuelled reactors. To systematically assess the uncertainty in calculations of the formation of  $^{232}\text{U}$  in thorium, we need covariances in the energy dependent neutron flux and in the (n, 2n) cross section data, propagating the uncertainty in the covariances in neutron fission spectra [23].

### **Concluding Remarks**

Addressing issues of assessment in uncertainties in nuclear data and its impact as part of design and safe operation is an important task in Bhabha’s 3-stage Indian nuclear power programme. The NDPCI is currently evolving a sound and sustainable programme on nuclear data experiments, EXFOR compilations, evaluations and integral data testing including covariances. In the opinion of the author, the NDPCI is moving positively in its learning curve on covariances and its applications to integral nuclear systems.

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# ICSBEP criticality benchmarking for nuclear data validations, KAMINI, PURNIMA-II and PURNIMA-I

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## Abstract

India has contributed three experimental benchmarks to the International handbook of the International Criticality safety Benchmark Evaluation Project (ICSBEP) of the US-DOE/NEA-DB. This presentation describes the interesting experience in creating these three Indian experimental benchmarks for nuclear data and code validation studies. The concept of definition of benchmark is also reviewed for convenience. Series of sensitivity studies are performed to assess the various uncertainties that arise in knowledge of the description of the actual system.

The International Criticality Safety Benchmark Evaluation Project (ICSBEP) handbook [1] is employed worldwide in advanced reactor physics research and teaching. It is, by design, is a compilation of critical, subcritical, and radiation-transport benchmark experimental data that have been verified, to the extent possible, by reviewing original and subsequently revised documentation, logbooks, internal memos and letters, and by talking with experimenters or individuals who were associated with the experiments or the fabrication of components of the experimental facility. The experimental data have been reduced to a form, called the benchmark specification that facilitates the development of an analytical model of the criticality experiment. Detailed uncertainty analyses are performed and an overall uncertainty is assigned to the benchmark model  $k_{\text{eff}}$  value. This requirement calls for lot of details of the reactor components and parts as it is built. This involves rigorous collection of data from the fabricator, quality assurance personnel and personnel assembling the reactor and performing several sensitivity studies to estimate the total uncertainty due to tolerances and errors in the geometry, composition and densities. When used properly, these data allow criticality safety analysts to validate their analytical techniques and nuclear data evaluators to test data performance without repeating the research and data-reduction steps. Benchmarking essentially involves quality description of the core such that the errors and uncertainties are characterized and in total effect together will not effect  $k_{\text{eff}}$  by more than 1000 pcm (10 mk). The evaluated project is well documented in ICSBEP handbook and DICE (Database for the International Handbook of Evaluated Criticality Safety Benchmark Experiments) is used to find the experiment of interest.

KAMINI (2005): Kalpakkam Mini (KAMINI) Reactor is a U-233 fuelled, light water moderated, natural convection cooled and beryllium oxide reflected low power research reactor which is designed to operate at 30 kW nominal power is located at Indira Gandhi Centre for Atomic Research at Kalpakkam, India. Because of the highly efficient reflector material BeO, it has a very low fuel inventory of approximately 612 g of <sup>233</sup>U in the form of uranium-aluminum alloy plates. An external heat exchanger cools the tank water to control the inlet temperature. Two absorber type safety control plates are used for control as well as emergency shutdown of the reactor. The core-reflector assembly along with experimental fixtures like beam tubes, irradiation tubes, pneumatic fast transfer facility are housed in a stainless steel tank, which is surrounded by a biological shield of concrete. In order to make KAMINI as benchmark all the available data from the records are collected and compiled. Wherever the finer details of

KAMINI were not available or known with poor accuracy, new measurements were carried out wherever possible and best judgement was used with the available data to find the effect of the uncertainties in the prediction of  $k_{\text{eff}}$ . Earlier calculations showed the  $k_{\text{eff}}$  of the reactor as 1.017 and SCP worth as 23.7 mk [2] as against the measured value  $k_{\text{eff}}$  and worth of SCP as 1.0025 and 26.4 mk respectively. This deviation in the predicted value was due to the modelling and the cross section used. The  $k_{\text{eff}}$  and the uncertainty in benchmark of KAMINI in the characterization of the predicted  $k_{\text{eff}}$  was found to be  $0.99119 \pm 0.00025$  (Monte Carlo)  $\pm 0.00586$  (benchmarking) [3]. The present improvement in  $k_{\text{eff}}$  is due to some of the following important improvements such as, fuel density, BeO density, and inclusion of aperture control plate, fuel isotopic composition, and subassembly wise density of fuel, impurities in fuel, aluminum, BeO and zircaloy, inclusion of core cage. The first three factors have the maximum influence on the results.

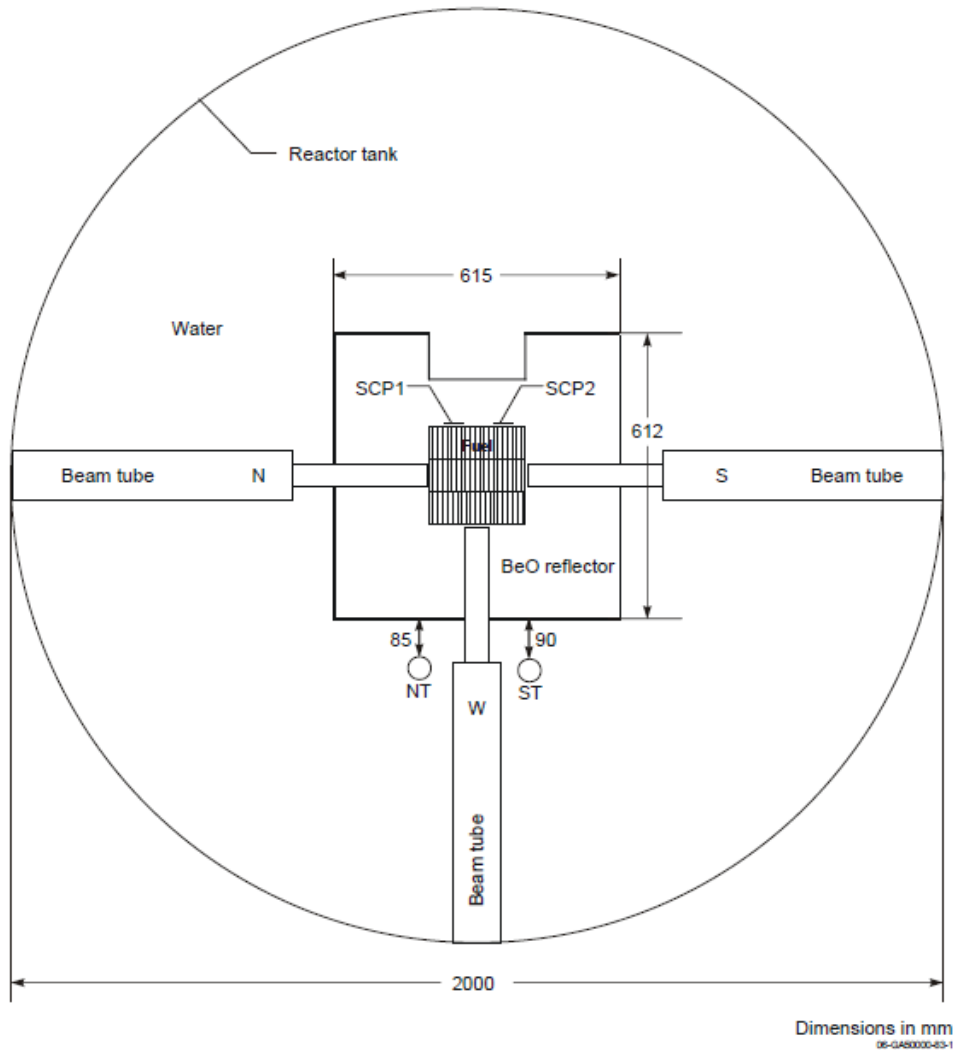
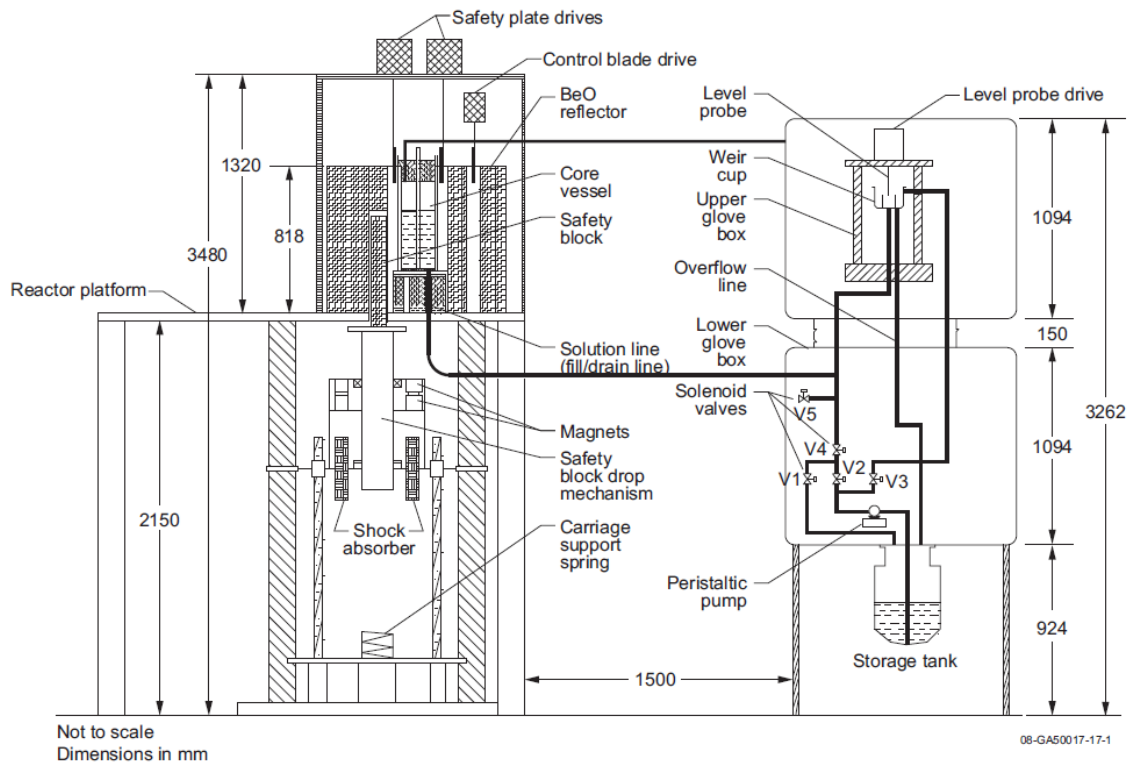


Figure 1. KAMINI Fuel and Reflector Assembly inside the Reactor Tank.

PURNIMA-II (2008): With the experience of benchmarking of KAMINI, PURNIMA-II was taken up for benchmarking. Purnima-II was a  $^{235}\text{U}$  uranyl nitrate solution fuelled, beryllium oxide (BeO) reflected, homogeneous experimental setup at BARC during 1984-86. The entire

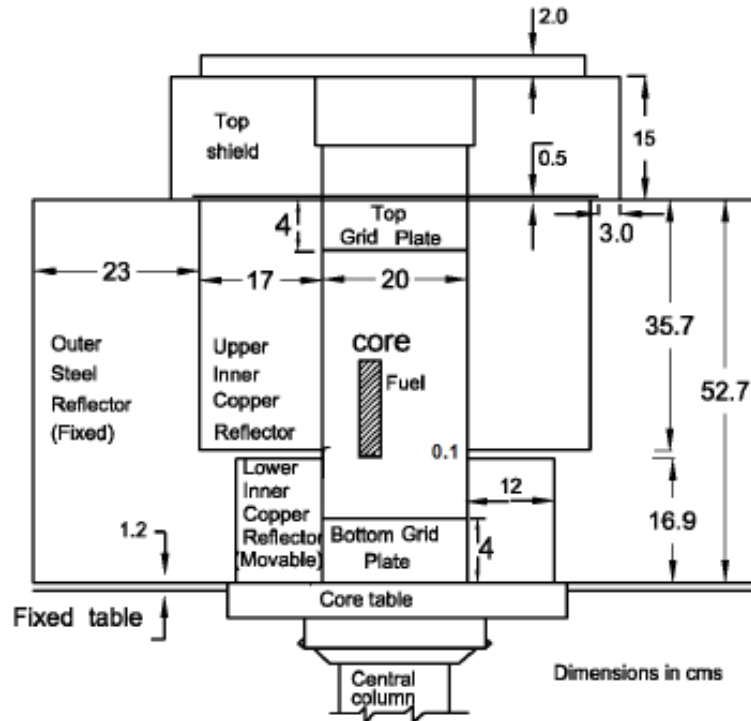
PURNIMA-II set up consists of a reactor assembly and two associated glove boxes for fuel handling operations. Figure 2 presents the schematic elevation view of the reactor assembly and glove boxes. The experiments carried out with this system included measurement of critical mass as a function of solution concentration, reactivity worth of various safety devices, and the measurement of void and temperature coefficients of reactivity. The fissile solution was contained in a cylindrical zircaloy core vessel and was surrounded by 30 cm thick BeO reflector (Fig.2). A pair of cadmium absorber plates operating on opposite sides of the core-vessel screen off reflector returned neutrons for rapid shut-down. The approach to criticality was carried out by plotting the inverse counts versus solution height data. The experiment was carried out for six different concentration of the fuel in the solution. For benchmark analysis highest, lowest and middle concentration of the solution was used. For the remaining three fuel concentrations, the results were extrapolated. The  $k_{\text{eff}}$  and the uncertainty in benchmark of PURNIMA-II in the characterization of the predicted  $k_{\text{eff}}$  for uranium concentration of 116.6 g/l was found to be  $0.9935 \pm 0.0003$  (Monte Carlo)  $\pm 0.00772$  (benchmarking) [4].



**Figure 2** Schematic Elevation Layout of Reactor Assembly and Glove Boxes of PURNIMA-I reactor

PURNIMA-I (2012): Plutonium Reactor for Neutronic Investigations in Multiplying Assemblies was designed, built and operated at BARC, Trombay as India's first zero-energy fast reactor in the early seventies. It was fuelled by plutonium oxide and reflected by 17 cm thick copper followed by shielding of 23 cm mild steel in radial direction (Fig. 3). The fuel pins were contained in an asymmetric hexagon core-vessel of 52 cm height made of 7 mm thick stainless steel. The 4 cm thick top grid plate had tapered holes into which the fuel pins fitted snugly and

were held in position by retaining nuts. The fuel pins are arranged in a triangular array. The central portion of the fuel consists of  $\text{PuO}_2$  pellets with axial molybdenum rods at both the ends. Six safety rods and three control rods made of molybdenum located in vertical channels in the copper reflector close to the core. To shut-down the reactor, the core was driven out of the reflector. It attained criticality with 175 full fuel pins and 4 half fuel pins. This half fuel pin contained 9  $\text{PuO}_2$  and 9 SS pellets stacked alternatively. The  $k_{\text{eff}}$  and the uncertainty in benchmark of PURNIMA-I in the characterization of the predicted  $k_{\text{eff}}$  was found to be  $1.01009 \pm 0.00019$  (Monte Carlo)  $\pm 0.00437$  (benchmarking) [5].



**Figure 3** Elevation view of PURNIMA-I reactor

As a result of our benchmarking, a large portion of the tedious and redundant research and processing of critical experiment data has been successfully eliminated. The necessary steps in criticality safety analyses of validating computer codes with this benchmark critical data has got greatly streamlined, and valuable criticality safety experimental data are preserved. This work of benchmark as part of the ICSBEP has highlighted gaps in data such as impurities and densities and has retrieved lost data, the use of KAMINI, PURNIMA-I and PURNIMA-II benchmarks are on par with other international benchmarks and can be used to help to identify inadequacies and errors in basic nuclear data and cross section processing codes.

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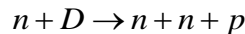
# Evaluation of the D(n,2n)p reaction cross section

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D(n,2n)p is one of the simplest three nucleons reactions without Coulomb interaction, it is very significant for searching after nuclear force and nuclear theory. Deuterium is one of the important fusion fuels, therefore D(n,2n)p reaction cross sections are useful for nuclear power development.

It is difficult to measure the D(n,2n)p reaction cross sections, up to now all the measured data of 13 sets have been collected from threshold to 25MeV. The original data and errors of all the experiments were analyzed. The recommended values of D(n,2n)p reaction cross sections were obtained on basis of least-squares fit with orthogonal polynomial, and were compared with the theoretical calculation and the ENDF/B-VII.1 [1], JEFF-3.2, JENDL-4.0 [2] and CENDL-3.1 [3]. This evaluation is based on the works [4-5].

There are three possible reaction channels of n+D reactions below 30 MeV, elastic scattering, spallation reaction and radioactive reaction. The spallation reaction is as follows:

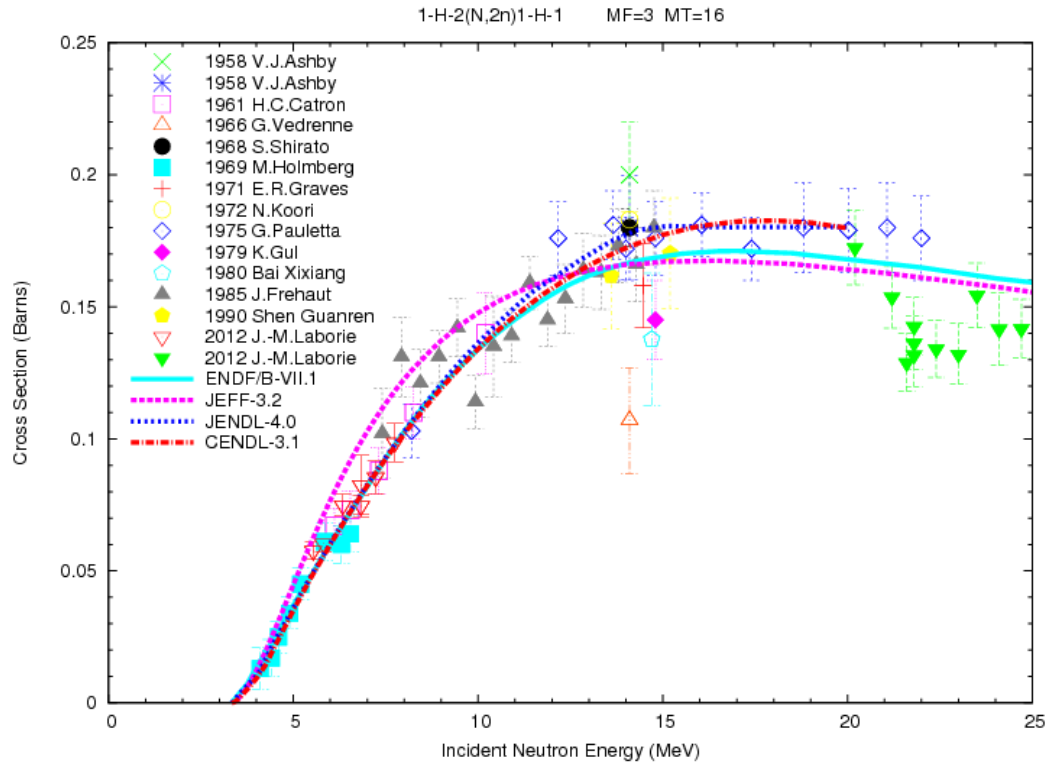


The threshold energy is 3.339 MeV.

The D(n,2n)p reaction cross sections have been measured mostly in the last century 60's to 80's, as shown in the **Fig. 1**. The experimental data have retrieved from EXFOR [6-17], the energy region was from threshold to 22 MeV. The evaluated data in the ENDF/B-VII.1, JENDL-4.0 and CENDL-3.1 have taken from the ENDF/B-VI.4 (released at 1997), JENDL-2 (released at 1983) and CENDL-2.1 [18], respectively. These data have been obtained based on the experimental data at that time, but the data in the ENDF/B-VII.1 (ENDF/B-VI.4) have taken from the theory calculation above 10 MeV. The data of the JEFF-3.2 is the latest evaluation at 2012, based on the modern realistic INOY N-N potentials, but not a good agreement with the measured data. The latest measurement of the D(n,2n)p reaction cross sections were performed by J.M.Laborie [19] at 2012, the energy ranges have been enlarged to 25 MeV.

The experimental data need to be analyzed and the evaluated data need to be recommended again, with the evaluation of the each data library are old, the theoretical results and the measurements are not good agreement, the latest experiment have been done and the fission neutron  $\bar{\nu}$  value has been updated.

The methods used in the measurements can be divided into two categories, the firstly to measure the two neutrons by means of large liquid scintillator [6,7,10,16,19] or TOF [8,14,15,17] and the secondary to measure the proton with the aid of particle identification spectrometer [9,11,12,13].

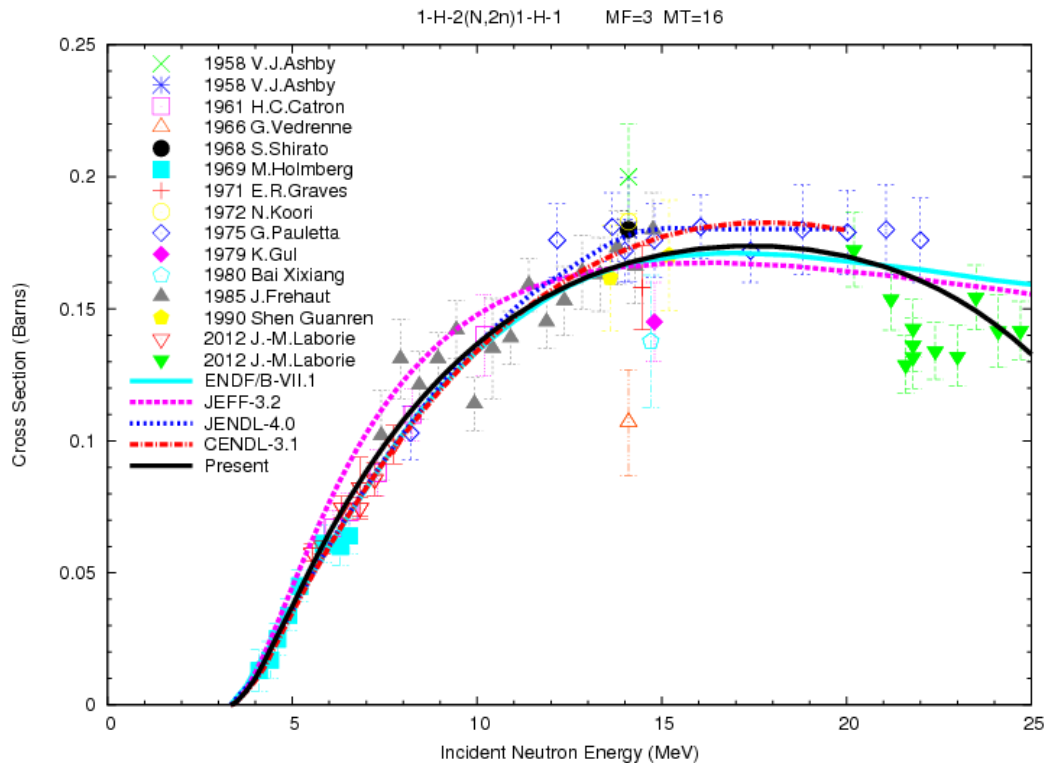


**Fig. 1:** The comparison between experimental and evaluated data of the D(n,2n)p reaction cross sections.

**Table.1:** The recommended value of the D(n,2n)p reaction cross sections.

| En (MeV) | CS (b)   | En (MeV) | CS (b)   | En (MeV) | CS (b)   | En (MeV) | CS (b)   |
|----------|----------|----------|----------|----------|----------|----------|----------|
| 3.34E+00 | 0.00E+00 | 6.50E+00 | 7.72E-02 | 1.18E+01 | 1.54E-01 | 2.10E+01 | 1.66E-01 |
| 3.40E+00 | 3.10E-04 | 7.00E+00 | 8.83E-02 | 1.25E+01 | 1.59E-01 | 2.15E+01 | 1.64E-01 |
| 3.50E+00 | 1.30E-03 | 7.25E+00 | 9.35E-02 | 1.30E+01 | 1.62E-01 | 2.20E+01 | 1.61E-01 |
| 3.75E+00 | 5.10E-03 | 7.50E+00 | 9.85E-02 | 1.35E+01 | 1.65E-01 | 2.25E+01 | 1.57E-01 |
| 4.00E+00 | 1.00E-02 | 8.00E+00 | 1.08E-01 | 1.41E+01 | 1.67E-01 | 2.30E+01 | 1.54E-01 |
| 4.25E+00 | 1.65E-02 | 8.50E+00 | 1.16E-01 | 1.50E+01 | 1.71E-01 | 2.35E+01 | 1.49E-01 |
| 4.50E+00 | 2.36E-02 | 9.00E+00 | 1.24E-01 | 1.60E+01 | 1.73E-01 | 2.40E+01 | 1.44E-01 |
| 5.00E+00 | 3.73E-02 | 9.50E+00 | 1.31E-01 | 1.70E+01 | 1.74E-01 | 2.45E+01 | 1.39E-01 |
| 5.50E+00 | 5.18E-02 | 1.00E+01 | 1.37E-01 | 1.80E+01 | 1.74E-01 | 2.50E+01 | 1.33E-01 |
| 5.60E+00 | 5.46E-02 | 1.05E+01 | 1.42E-01 | 1.90E+01 | 1.72E-01 |          |          |
| 6.00E+00 | 6.51E-02 | 1.10E+01 | 1.47E-01 | 2.00E+01 | 1.70E-01 |          |          |

The original data and errors of all the experiments were analyzed and fitted on basis of the least-squares method with orthogonal polynomial. The recommended values are listed in **Table.1**. **Fig. 2** shows the comparison of the experimental and evaluated data and recommended values of the D(n,2n)p reaction cross sections.



**Fig. 2:** The comparison of the experimental and evaluated data and recommended values of the  $D(n,2n)p$  reaction cross sections.

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# Current status of nTOF facility construction at KAERI

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## **Abstract**

Building Neutron TOF (Time of Flight) experimental facility has been in progress at KAERI, where neutrons are produced by stopping primary electrons from the KAERI super conducting electron accelerator in a liquid lead target. While incident electrons are stopped in the liquid lead target, bremsstrahlung photons are produced. These bremsstrahlung photons produce neutrons through ( $\gamma$ , n) reactions within the target.

This project is being proceeded by two tracks. One is building an nTOF facility and the other is manufacturing a liquid lead photo-neutron source.

Currently, the direction of incident electron beam to the liquid lead target is changed from horizontal to 90° downward direction for easy construction. The simulation results with new configuration give 8% decrease in neutron /photon ratio at collimator direction. However, we can reduce shielding thickness by 80cm in +x and 30cm in +z directions and the neutron flux along the TOF line is slightly higher than before.

In this year, the final version of blueprint for the nTOF facility construction will be ready and some components of the revised photo-neutron source will be manufactured.





# Design considerations for an n-TOF facility based on a 30 MeV RF electron linac

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## Abstract

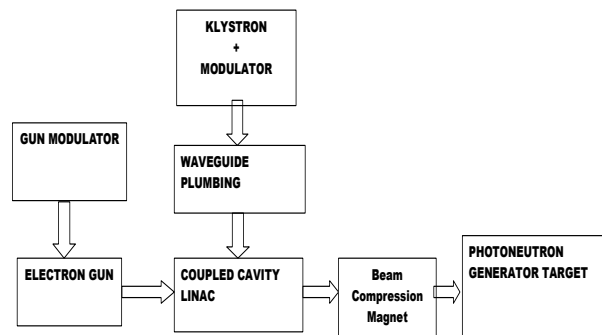
Cross-section measurement of the interaction of neutrons with matter is of fundamental interest for physicists working on nuclear physics, astrophysics and condensed matter. RF Electron Linacs can provide a neutron source for such studies and a proposal is being made for such a facility in India for high resolution neutron spectroscopy. The paper addresses some of the design guidelines to be followed towards realization of this objective.

## Introduction

There are several Neutron Time of Flight(n-TOF) facilities around the world based on RF electron linac and catering to nuclear data generation[1,2,3]. In India, Department of Atomic Energy is pursuing a program of indigenous development of RF Electron Linacs and some linacs have been built. This technology development will be exploited to build a 30 MeV RF Linac to provide a neutron source producing  $10^{13}$  neutrons/sec for an n-TOF facility. In this paper we present our experience in RF Linac Technology and outline some of the design considerations for such a facility.

## Design Objectives

The design goal is to provide a neutron source with a strength of  $10^{12}$ - $10^{13}$  neutrons per second integrated over the total energy spectrum. The pulse-width of the neutron source for energy at 10 eV will be about 500 ns with a moderator and for energy above 100 keV, this value is expected to be less than 5 ns. The resolution required at high energy neutron spectroscopy demands a short pulse (1-2 ns) high current (50-100A) electron beam at a repetition rate of 250 Hz. The schematic of the set-up to illustrate the concept is given is Figure 1.



**Figure 1:** Block Diagram of Neutron Generation Scheme

The electron gun injects a current pulse into the linac structure to accelerate the beam to 30 MeV. The electron gun and the klystron can be pulsed by modulators to achieve desired pulse duration of electron beam. The beam compression magnet is used to shorten the beam pulse and

proportionately increase the current. The beam emerges through a helium-cooled thin Ti6Al4V window to fall on a photoneutron target and generates the neutrons for n-TOF application. We describe some of the design guidelines for different parts of the system.

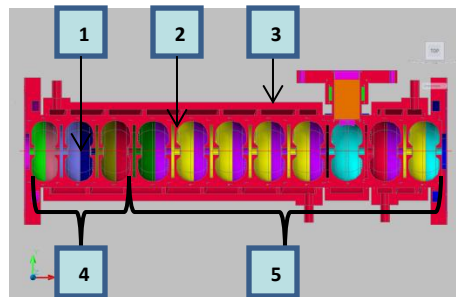
### Electron Gun

The electron gun will provide an electron pulse of energy 50-100 keV which will be first injected to a prebuncher. After the prebuncher, there will be another bunching section with acceleration and two further accelerating sections. The prebuncher cavity will be fed by microwave power of about hundred kilowatt. For short pulse high current operation, the pulse length will range from 10 ns to 50 ns. For long pulse operation the duration will range from 50 ns onwards to 5  $\mu$ s. After the prebuncher, adequate free space will be provided for bunching of the electron beam so that maximum capture of the electrons occurs in the buncher cavity. Cathodes of high emission current density having about 20 mm diameter will be used to realize current up to 20-30A. Ceramic insulation will be used in the triode gun of Pierce geometry and care will be taken to minimize inter-electrode capacitance for better pulse shape. Initially grid-cathode bias will be given to halt electron beam production although the cathode-anode voltage is around 50-100 kV. A trigger pulse of a few kV and suitable duration can be applied between the cathode and grid to obtain the electron beam of required duration.

Our group has designed and operated electron gun in the diode mode with LaB<sub>6</sub> cathode. Line-type modulators have been used to generate pulse durations ranging from 3.5-10  $\mu$ s for linacs. The current extracted from 8 mm dia. cathode is more than 1 ampere. To begin with, the neutron source will be first operated with such long pulses with current in 100-200 mA range at repetition rates of 250 Hz.

### Linac Structure

Presently we are using Standing Wave Coupled Cavity Biperiodic structures operating at 2856 MHz in the linacs designed and developed by us. One of our linacs used for demonstration of industrial applications works regularly at 10 MeV, 3 kW average power at a pulse repetition rate of 300 pps at Electron Beam Centre, Kharghar [4]. As can be seen in Fig2 which is the sketch of a 6 MeV Linac developed by us to act as an X-ray source for cargo-scanning applications, the biperiodic structure has two types of cells i.e. the coupling cell and accelerating cell. The coupling cell which occupies much less space than the accelerating cell has no accelerating field and has a phase difference of  $\pi/2$  with respect to the adjacent accelerating cell. Thus the phase difference between consecutive accelerating cells is  $\pi$  resulting in high shunt impedance with stable operation at the resonant frequency.



**Figure 2:** 1. Accelerating Cavity 2. Coupling Cavity 3. Cooling jacket  
4. 3-Cell bunching Section 5. Accelerating Section

The structure is made of OFHC copper and brazing is done in high vacuum furnaces using Cu-Ag alloy at around  $800^{\circ}\text{C}$ . After the fabrication of the cavities, the electric field is measured by Bead Perturbation method. The resonant frequency, Q of the cavity and shunt impedance are determined using a Vector Network Analyser. Typical shunt impedance and Q values are  $85\text{M}\Omega/\text{m}$  and 12,000 respectively.

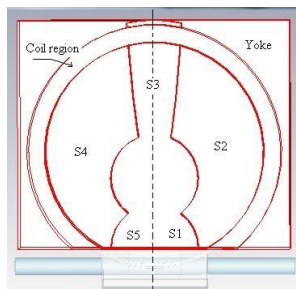
In the 30 MeV Linac, two accelerating sections of 1.5 meter length will be used. Each structure will be driven by TH 2163 Klystrons of 7.5 MW peak power in pulses of 2.5- 7.5 $\mu\text{s}$  duration at repetition rate of 250 pps. In the build-up time of about 750 $\mu\text{s}$ , electric field of 15-18 MV/m is set-up in the accelerating cavity in  $\text{TM}_{010}$  mode. ASTRA, SUPERFISH & CST Studio are being used for cavity design and simulation of beam dynamics in the bunching and accelerating structure. Nearly half of the klystron power is lost in the cavity walls and so it will be provided with cooling to prevent large shift in the resonant frequency of the cavity.

For short pulse high current operation up to 10A, space charge effect will try to expand the beam radially and reduce the phase focusing effect primarily in the bunching section. External magnetic field with the help of solenoids in the drift space after prebuncher and in the buncher section will be provided for radial focusing. In our present cavities, the beam aperture is 10 mm and this may limit our current to 1-2 ampere. In the first stage, experiments will be conducted to find out maximum allowable current in the structure. Simultaneously, studies are going on to improve the structure with larger current, keeping a reasonable shunt impedance. We are also evaluating the options of disc-loaded  $2\pi/3$  mode cavities with apertures of 20-25 mm.

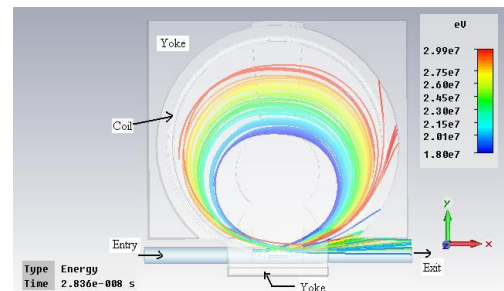
Our present linacs do not use the prebuncher and we find that the current transmission in the accelerating section is 25% of the gun current without any magnetic field. The introduction of prebuncher and the focussing solenoids is expected to increase the transmission to more than 75% at least in the long-pulse small current applications.

### Compression Magnet

Although moderation time becomes a limiting factor for the resolution for neutron spectroscopy below keV level, one can improve resolution at higher neutron energy by going for high current short electron pulse which decides the neutron pulse duration. We intend to compress the electron beam pulse from 10ns to 1 ns boosting the current ten-fold and we plan to pursue the design philosophy of the compression magnet implemented at Gelina[3,5].



**Figure 3:** Five Sector Magnet



**Figure 4.** Beam Trajectory at various energy

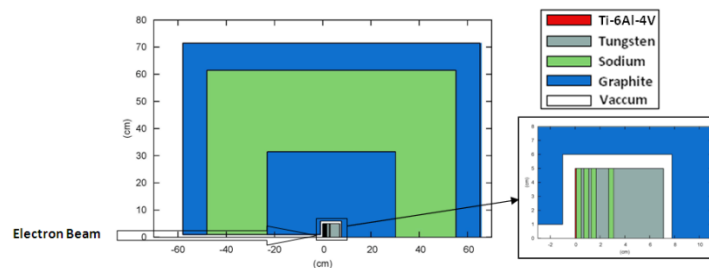
In a linac, electrons come in bunches separated at intervals of the R.F. period. It means that there will be about 30 bunches in the 10 ns current pulse emerging from the linac for an RF

source operating at 2856 MHz. Each bunch will not have same energy due to beam loading effect. In the 10 ns duration, as the beam is taking away energy from the stored energy in the cavity, the stored energy decreases. Since the cavity filling time is of the order of 500 ns, this energy cannot be replenished. Therefore, the bunches coming later experience lesser electric field leading to a variation in energy. When such bunches are subjected to a magnetic field perpendicular to the initial direction, each bunch will have a trajectory of different radius. The magnetic field configuration can be arranged in such a manner that the first bunch and the last bunch exit almost simultaneously.

Simulation work has been carried out for micropulses varying in energy from 18 MeV to 30 MeV in a magnet having five sectors as shown in Figure 3. For our case the field value in sector S1, S3 and S5 is 0.05 Tesla while that in S2, S4 is 0.1 Tesla. Fig. 6 shows the orbit of electrons of different energy in the magnetic field. It is estimated that the magnet will weigh about 25 tons having a size of 3 mts x 3 mts with 35-40 cm height and provision should be made for its mechanical handling.

### Neutron Target

High Z targets like Tantalum, Tungsten and Uranium will be considered as target material for neutron generation. Photoneutrons are produced dominantly by the giant resonance mechanism. It is well known that, the no. of neutrons produced per kilowatt of power slowly increases after an electron energy of 30 MeV and almost saturates beyond 40 MeV[6]. A quick estimate from this reference for 30 MeV electron beam indicates that about 7 kW beam power is required for a Tungsten target to produce  $10^{13}$  neutrons/sec. Much of the beam energy is lost in the target material and water-cooling will be provided to take away the heat. For better thermal management, instead of a target of single piece, a number of pieces separated by thin water channels will be used. MonteCarlo methods using Fluka and Geant4 code will be adopted for target design. A typical target assembly is shown in Figure 5. It consists of four Tungsten plates of thickness 2mm, 2mm, 10 mm and 40 mm, each of cross-section 80 mm x 50 mm arranged one after the other with the coolant flowing between the plates. Here sodium is shown as the coolant as this target is designed for a 30 MeV linac to simulate radiation streaming from Fast Breeder Reactor. The energy spectrum of the neutrons going from the target module to graphite is shown in Figure 6. In the case of n-TOF application, sodium will not be used as the coolant but the calculation shows approximately the number of neutrons expected at this power at 30 MeV.



**Figure 5:** Typical Target Assembly

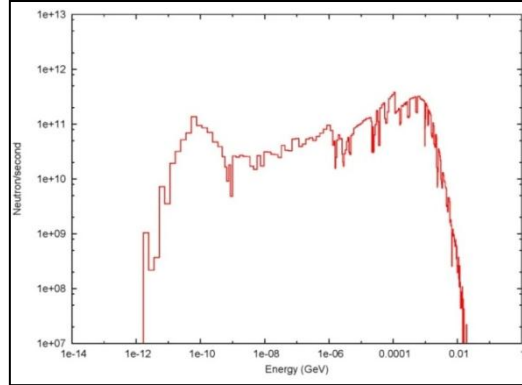


Figure 6: Neutron Energy Spectrum

### Shielding, Collimation & Beam Lines

In such a facility radiation shielding has to be provided for both bremsstrahlung X-rays and neutrons. While X-ray emission from the target is prominently in the forward direction i.e the direction of the incident electron beam, the neutron emission is nearly isotropic. The energy spectrum and angular distribution of photons and neutrons will be taken into account to design the shielding. Concrete, polyethylene and lead will be used as shielding material. Activation of different materials used in the system will be calculated and radiation in public access area will be maintained below 1  $\mu\text{Sv/hr}$ . The layout of the facility will take into consideration the length of the linac, the size of the compression magnet, the size of the neutron source with shielding, location of the power supplies, control system, the detector modules and possible future upgradation.

The length of flight tubes will range from 10 meters to 50 meters. They will be evacuated to few Torr vacuum to prevent neutron scattering in air as well as activation of air due to neutrons. Collimators will be provided at suitable locations for absorption of gamma rays and prevent neutron scattering for better signal-to-noise ratio. For a given neutron energy, the resolution can improve for longer flight path but the neutron flux on the sample decreases. The sample size is smaller than the source size and telescopic design of beam-lines would be made so that sample should see as much of the source as possible for higher neutron flux on the target.

### Conclusion

In the last two decades, indigenous efforts have led to successful development RF Electron Linac technology and the effort is continuing. Discussion with user groups is in progress to understand the nature of experiments, detectors and requirement of space. It is hoped that a neutron source for an n-TOF facility based on a 30 MeV Linac can be made in the next few years.

**ACKNOWLEDGEMENT:** The authors would like to acknowledge fruitful discussions with their colleagues Shri Jayant Mondal, P.K Rai, Dhruva Bhattacharjee, Biswaranjan Nayak, Romesh Chandra and Nishant Chaudhary.

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# Overview of upcoming INO facility

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## **Abstract**

The India based Neutrino Observatory (INO) will be an underground laboratory, at the Bodi Hills in the Theni district of Tamilnadu, housing experiments that will benefit from the low cosmic ray background there. The flagship experiment will make measurements on atmospheric muon neutrinos and antineutrinos using a 50 kiloton Iron Calorimetric (ICAL) detector. As ICAL will be magnetised, with an average B-field of about 1.3 Tesla, it will be able to clearly identify the charge of the muons produced in charged current interaction of muon neutrinos and hence the identity of the primary muon neutrino or antineutrino. One of the key goals of ICAL at INO is to identify the neutrino mass hierarchy, normal or inverted. It will also make precise measurements of some of the parameters in the neutrino mass matrix. Apart from ICAL, the underground laboratory will also house other experiments such as a search for neutrinoless double beta decay using a cryogenic tin bolometer and a cryogenic silicon bolometer for dark matter particles in the low mass (few GeV) range. The current status of the R&D effort including that of the prototype ICAL to be housed at the Inter Institutional Centre for High Energy Physics at Madurai will be presented.





# Nuclear physics using VECC cyclotrons: scopes and possibilities

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## **Abstract**

The energetic ion beams (typically,  $\sim 10$ -60 MeV/nucleon) from K500 superconducting cyclotron (SCC) at the Variable Energy Cyclotron Centre (VECC), Kolkata, is going to open up unique opportunity of working in the frontier areas of intermediate energy nuclear physics, particularly around the Fermi energy domain. New, extensive experimental facilities are being built at VECC to facilitate proper utilisation of the superconducting cyclotron. In recent times, the room temperature cyclotron (VEC) has also been operating very steadily and delivering light ion (up to  ${}^4\text{He}$ ) beams to the users. In this talk, the scope of research with the upcoming facilities and some highlights of the recent VEC experiments will be presented.



# Physics study of D-D/D-T neutron driven experimental subcritical assembly

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## Abstract

An experimental program to design and study external source driven subcritical assembly has been initiated at BARC. This program is aimed at understanding neutronic characteristics of accelerator driven system at low power level. In this series, a zero-power, sub-critical assembly driven by a D-D/D-T neutron generator has been developed. This system is modular in design and it is first in the series of subcritical assemblies being designed. The subcritical core consists of natural uranium fuel with high density polyethylene as moderator and beryllium oxide as reflector. The subcritical core is coupled to Purnima Neutron Generator. Preliminary experiments have been carried out for spatial flux measurement and reactivity estimation using pulsed neutron source (PNS) techniques. Further experiments are being planned to measure the reactivity and other kinetic parameters using noise methods. This facility would also be used for carrying out studies on effect of source importance and measurement of source multiplication factor  $k_s$  and external neutron source efficiency  $\phi^*$  in great details. Some experiments with D-D and D-T neutrons have been presented.

## Introduction

Accelerator driven subcritical systems (ADS) <sup>[1-3]</sup> have sparked interest in scientists worldwide due to their potential for burning actinides and long-lived fission products as well as their superior inherent safety characteristics in terms of avoiding any criticality related (runaway) incidents. ADS can primarily be used for waste transmutation of long-lived fission products and also for power generation. Indian interest in ADS has an additional dimension, which is related to the planned utilization of our large thorium reserves for future nuclear energy generation.

An ADS is basically a subcritical reactor which produces fission without achieving criticality; it uses additional neutrons from an external source – an accelerator producing neutrons by spallation.

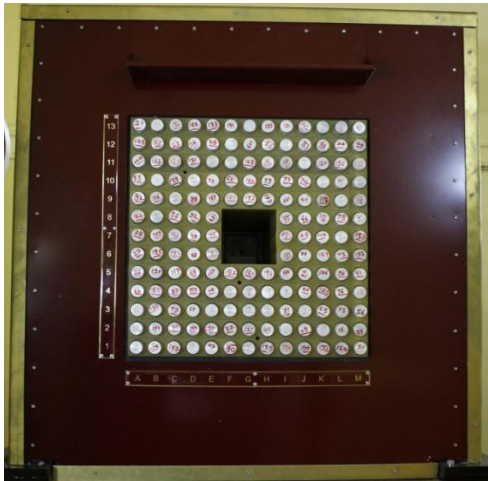
The physics of ADS is quite different from that of critical reactors and it needs to be studied before such systems are put into operation. For example, subcritical neutrons have additional source neutrons (unlike critical reactor where the source is fission neutron only) which have different spatial and energy distribution. This distribution changes from generation to generation which gives rise to the concept of source multiplication factor  $k_s$ . Also, the power in an ADS is very sensitive to the value of  $k_{eff}$  and it is essential to accurately predict this parameter over the entire length of the burnup cycle. In addition to these, the stationary spatial flux distribution and neutron energy spectrum in ADS is different from critical reactor. Similarly, the dynamic response of ADS to transients and perturbations are also quite different from that of the critical reactors. As high energy accelerator suitable for ADS are yet to be developed, the physics of ADS is generally studied using neutrons produced either by D-D / D-T reaction or by other low energy accelerators.

A zero power subcritical assembly driven by D-D/D-T neutron generator has been developed at Purnima, BARC. The subcritical system BRAHMMA (**BeO Reflected And HDPe Moderated Multiplying Assembly**) <sup>[4]</sup> is aimed at understanding neutronic characteristics of accelerator driven system at low power level.

## BRAHMMA Subcritical Core

The modular subcritical assembly consists of metallic natural uranium as fuel and high density polyethylene (HDPE) as moderator in a 13 X 13 square lattice configuration followed by 200 mm of beryllium oxide (BeO) as reflector. The fuel is embedded in high density polyethylene moderator matrix. The whole assembly is surrounded by an outer layer of borated polyethylene and cadmium to isolate the system from scattered neutrons. The central 3X3 positions of the lattice are empty and serve as the cavity for inserting neutron source. The target end of the neutron generator is inserted in this cavity such that the target is located at the centre of the core. Fig.1 shows the subcritical core.

One of the unique features of subcritical core is the use of Beryllium oxide (BeO) as reflector and HDPE as moderator making the assembly a compact and modular system.



**Fig.1.** BRAHMMA Subcritical



**Fig.2.** Subcritical core being coupled to neutron generator

## Purnima Neutron Generator

The coupling with the subcritical core is provided by deuteron accelerator based neutron generator (Fig.2). It is a 300 kV deuteron accelerator. The D<sup>+</sup> ions are produced in an RF ion source, which are extracted, focused, accelerated and bombarded on the target. The target is maintained at ground potential. The deuteron ions impinge on titanium-tritium (TiT) or titanium-deuterium (TiD) targets (with copper backing), providing 14.1 MeV or 2.45 MeV neutrons via T(d,n)<sup>4</sup>He or D(d,n)<sup>3</sup>He reactions respectively.

The neutron generator has both DC and pulsed operation. It has facility for online source strength monitoring using neutron tagging and programmable source modulation.

## Pulsed Neutron Source Techniques

### A. Area Ratio (Sjöstrand)

In this method <sup>[5]</sup>, a series of neutron pulses are injected in the system. The decay of neutrons is observed and a histogram (Fig.3(a)) is plotted. Reactivity is obtained by the ratio of the prompt and delayed areas as

$$\rho \text{ (in \$)} = - A_{\text{prompt}} / A_{\text{delayed}}$$

where  $A_{\text{prompt}}$  is the prompt area and  $A_{\text{delayed}}$  is the delayed area in the PNS histogram.

## B. Slope Fit

The prompt neutron decay constant  $\alpha$  may be obtained from the slope of the prompt part in the PNS histogram as

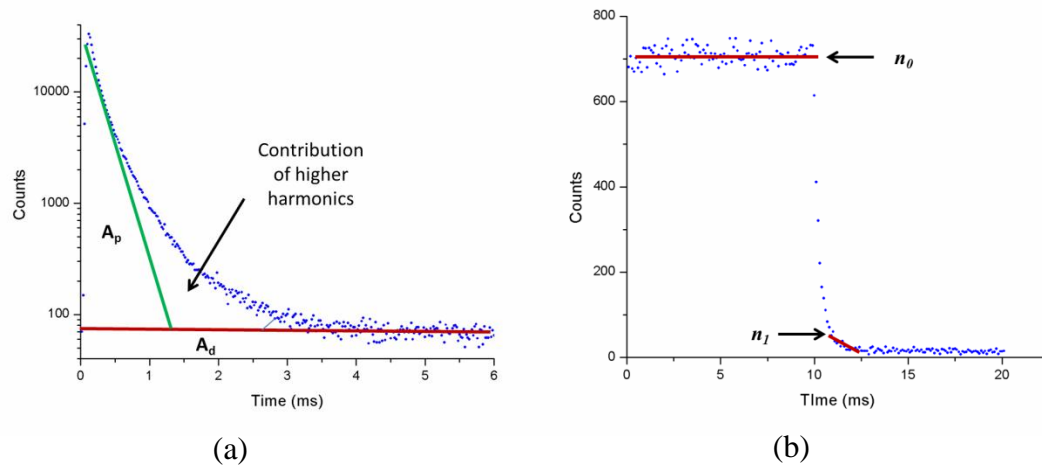
$$\alpha = (\rho - \beta_{eff})/\Lambda$$

where  $\Lambda$  is the mean neutron generation time.

## C. Source Jerk

In this method, the source is operated in a steady state - steady state flux, say  $n_0$ . The source is then switched off (Beam Trip) and the prompt jump of the flux to lower level  $n_1$  is recorded (Fig.3(b)). The reactivity is then obtained as

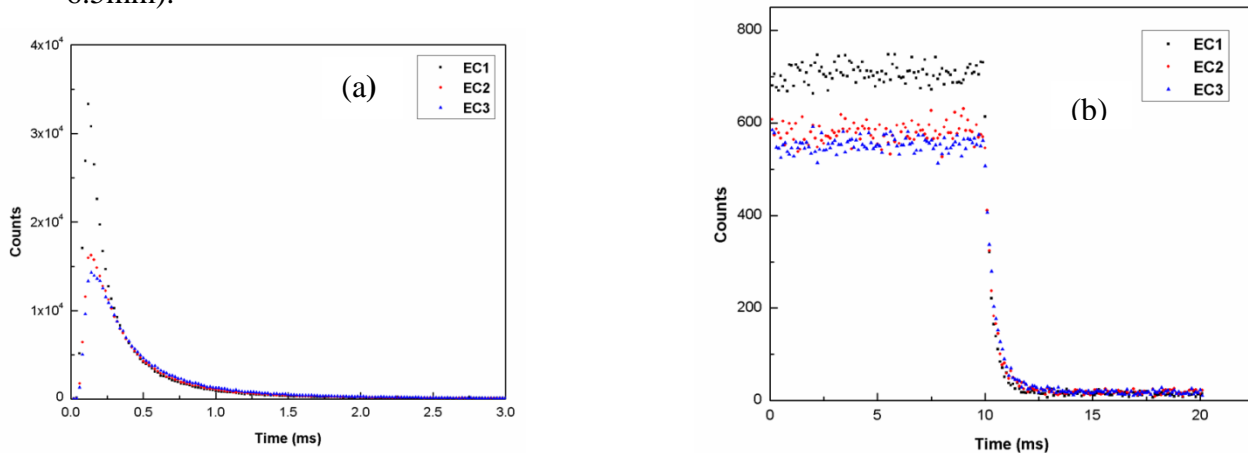
$$\rho \text{ (in \$)} = (n_1 - n_0)/n_1$$



**Fig.3.** (a) PNS histogram (b) Source-Jerk

## Experimental Results

Experiments were carried out for measurement of reactivity using PNS techniques. The neutron counts were measured with a miniature  $^3\text{He}$  detector (Active length: 70 mm, diameter: 6.5mm).



**Fig.4.** (a) PNS histogram (b) Source-Jerk

Measurements were carried out in axial experimental channels EC1, EC2 and EC3. Fig. 4(a) shows the PNS histogram. This histogram is used to obtain reactivity value using Area Ratio method and slope fit method as discussed above. The values of  $\beta_{\text{eff}}$  and  $\Lambda$  are obtained theoretically using Monte Carlo simulations. Fig.4 (b) shows the experimental plot for source jerk method. The reactivity is obtained from the prompt jump data as discussed above. Table 1 summarizes the experimental values obtained.

**Table 1.** Experimental results for reactivity measurement

| Experimental channel | Area Ratio                  | Slope Fit                   | Source Jerk                 |
|----------------------|-----------------------------|-----------------------------|-----------------------------|
| Theoretical          | <b>0.890</b>                |                             |                             |
| EC1                  | <b>0.898</b><br>$\pm 0.002$ | <b>0.887</b><br>$\pm 0.002$ | <b>0.902</b><br>$\pm 0.005$ |
| EC2                  | <b>0.902</b><br>$\pm 0.002$ | <b>0.884</b><br>$\pm 0.002$ | <b>0.890</b><br>$\pm 0.005$ |
| EC3                  | <b>0.893</b><br>$\pm 0.002$ | <b>0.883</b><br>$\pm 0.002$ | <b>0.902</b><br>$\pm 0.005$ |

### Conclusion

A zero power experimental subcritical facility has been developed to study the physics and neutronics of accelerator driven subcritical systems. Preliminary experiments have been carried out and results are in good agreement with theoretically estimated values. This experimental facility is expected to provide new data and would be used as a test bed for testing of new theoretical and experimental concepts. Future plan for upgradation of this facility to higher  $k_{\text{eff}}$  value and design of next generation of zero power fast experimental ADS facility are underway.

### Acknowledgement

The author would like to acknowledge the team members involved in this work, in particular, Mr. Tushar Roy, Dr. Yogesh Kashyap, Dr. Mayank Shukla, Mr. Tarun Patel, Mr. P. S. Srakar, Mr. Nirmal Ray and Ms. Shefali Bajpai.

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# An overview of FOTIA and LEHIPA

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## **Abstract**

A 6 MV Folded Tandem Ion Accelerator (FOTIA) was commissioned in 2000 at BARC. Since then it is in operation and has been used both for basic and applied research. In FOTIA, negative ions extracted from a SNICS-II ion source at 150 keV are injected into low energy accelerating tube where they can be accelerated upto 6 MeV. In the high voltage terminal negative ions are stripped off their electrons to become positive ions and these positive ions are bent by an  $180^\circ$  magnet located inside the high voltage terminal. The beam is then injected into the second accelerating tube mounted in the same column section. In the second accelerating tube ions get additional energy gain of  $nV$  leading to total energy of  $(1+n)V$  MeV. Here  $V$  is the terminal voltage and 'n' is the charge state of the ions. The beam energy analysis is then done using a  $90^\circ$  magnet. An energy resolution of 2 keV is achieved. At present, experiments are done using setups such as General Purpose Scattering Chamber, PIXE and external PIXE systems, Rutherford Back scattering and PIGE. All the nuclear physics experiments are done with  $0^\circ$  line using scattering chamber which has two movable platforms for mounting the detectors. The FOTIA is used very extensively for biology experiments where effects of radiation on live cells are studied. For this, beam is taken out in air through a  $40\ \mu$  thick kapton foil.

We are also building a Low Energy High Intensity Proton Accelerator (LEHIPA) at BARC. It consists of a 50 keV ECR ion source, a 3 MeV CW Radio Frequency Quadrupole (RFQ) and a 20 MeV Drift Tube Linac (DTL). The LEHIPA is designed for 30 mA proton beam. The beam dynamics studies were done carefully and a transmission of  $> 97\%$  is achieved. At these currents and low energies space charge forces become important and beam halo are formed and therefore apertures will be mounted at appropriate locations. A beam dump cum neutron target is being fabricated to handle beam power of 600 kW. A neutron yield in the range of  $10^{15}$  n/sec is expected for 30 mA at 20 MeV and hence proper shielding etc have been planned. The first beam from LEHIPA is expected by end of 2015. Both FOTIA and LEHIPA will be excellent facilities for nuclear data generation at low energies.

In this talk, salient features of these systems and their present status will be discussed.





# Overview of accelerator and nuclear physics facilities at IUAC, New Delhi

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The Inter University Accelerator Centre (IUAC) at New Delhi is a national user facility for accelerator based research programs in India. The accelerator centre equipped with a 15 UD Pelletron accelerator[1] followed by a superconducting booster LINAC provides wide range of stable beams for basic research in the field of low/medium energy nuclear physics, atomic physics, material science, radiation biology and accelerator mass spectrometry. The maximum available beam energy is around 5-6 MeV/amu for projectile mass  $A \sim 60$  amu. For nuclear physics research, most of the experiments are performed near the Coulomb barrier energies. Reactions around Coulomb barrier energies are used to study a wide range of phenomena, namely, decay of excited states in nuclei, shapes and structure of nuclei, dynamics of fusion-fission reactions etc. Existing experimental facilities for nuclear research experiments consists of a general purpose scattering chamber, recoil separators, gamma detector array and large neutron detector array. These facilities have been setup in two experimental halls, Hall I where beams are delivered from Pelletron accelerator alone and Hall II where LINAC beams are being used. In order to improve the intensity and availability of new beams, accelerator facilities are currently being upgraded by the addition of a high current injector systems.

The General Purpose Scattering Chamber (GPSC) installed in Hall I has been extensively used for nuclear reaction studies on elastic/inelastic scattering, fission fragment angular and mass distribution measurements etc. It is a 1.5m diameter vacuum chamber with two independently rotatable arms inside. Charged particle detector telescopes mounted on the arms can be rotated in vacuum to allow measurement of angular distributions of the reaction products. A time of flight(TOF) setup consisting pair of multi wire proportional counters(MWPC) and fast time zero detectors on two arms are used for fission fragment mass distribution studies and other binary reactions in kinematic coincidence. The detector setup provides unique identification of charged particles, their position & energy information. Using the TOF facility and the charged particle telescopes many experiments have been performed in the recent past, probing the dynamics of the fusion-fission process in heavy nuclei by measuring angular and mass distributions of the fission fragments. Recent experimental results shows conclusive evidence of onset of quasi-fission in systems forming the compound nucleus through different entrance channels [2].

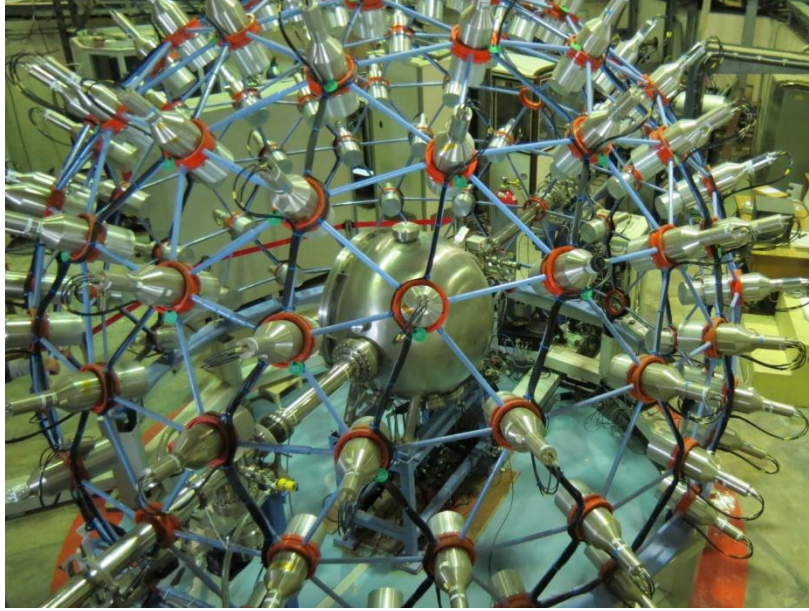
The recoil mass separator facilities are being used for measurement of evaporation residues cross sections in heavy ion induced reactions. The Heavy Ion Reaction Analyzer (HIRA)[3] installed in Hall I is a zero degree recoil mass spectrometer capable of operating at zero degree as well as other angles. Here the fusion evaporation residues(ER) are separated from the primary beam backgrounds and transported to the focal plane of the spectrometer where they are dispersed according their  $m/q$  value. The electromagnetic elements of HIRA is arranged on electric dipole(ED1)-magnetic dipole(MD)-electric dipole(ED2) symmetric ion optics configuration, with magnetic quadrupole doublets placed before and after the electric dipoles. The facility has been used mainly for ER cross section measurement, and multi nucleon transfer reactions around the Coulomb barrier. The operational mode of HIRA can be modified and set for separating secondary beams produced in reaction such as (p,n) (d,n) induced by light ions in

inverse kinematics[4].  $^7\text{Be}$  secondary beam has been extracted in flight and used in few experiments. The Hybrid Recoil Analyzer (HYRA)[5] installed in Hall II is a two stage recoil separator which can be operated in dual mode; gas filled and vacuum mode. It consists of a momentum achromat followed by a recoil spectrometer with the first stage achromat alone operating in gas filled mode or the full system operating in vacuum mode. In the gas filled mode, it operate as gas filled recoil separator for heavy nuclei of mass 200 and above. In vacuum mode the device is optimized for accessing particles around  $A \sim 100$  amu produced in inverse kinematic reactions.

The nuclear spectroscopy research at IUAC was started soon after the installation of 12 HPGe based gamma detector array(GDA)[6]in the Hall I. The array used Anti-Compton shields and other ancillary detectors such as 14 element BGO multiplicity filter, 14 phoswich based charged particle detector array for channel selection etc. Life time measurement was carried out using a recoil distance device. The main research emphasis was on the systematic study of evolution of high spin structures and shape changes in nuclei in the mass region  $A \sim 75,90,120, 130$  and 160. For gamma spectroscopy with LINAC beam, a more efficient array using Compton suppressed Clover detectors was later on setup in Hall II. This facility called Indian National Gamma Array (INGA) [7] has been setup by pooling Clover detectors from various institutions in the country. It can accommodate maximum 24 Clover detectors covering nearly  $4\pi$  geometric solid angle and total photo peak efficiency of  $\sim 5\%$ . The array is optimized for  $\gamma\text{-}\gamma\text{-}\gamma$  and higher fold data. Custom made electronic modules are used for processing the energy & timing signals from Clover detectors and anti-Compton shields. The logic for Compton-suppression, the generation of trigger strobes for data collection and pile-up rejection were also incorporated in this module. The facility has been used by many university users for research in nuclear structure studies using LINAC beams at IUAC.

National Array of Neutron Detectors: Very recently a new experimental facility called National Array of Neutron Detectors (NAND)[8] is being setup especially for the study of fission dynamics by neutron multiplicity measurement in beam Hall II of IUAC. It is a large time of flight facility consisting 100 neutron detectors plus array of fission fragment detectors to detect neutrons in coincidence with fission fragments. The main research focus of the array is on the measurement of pre and post scission neutron multiplicity and the multiplicity distributions of neutrons emitted during the dynamical process of nuclear fission. It will serve as a powerful tool to study the dynamics of the fissioning system and can provide fresh opportunities for experimental study of nuclear fission induced by heavy ions by exploiting the availability of high energy heavier beams from the LINAC accelerator.

The NAND facility at IUAC consists 100 liquid scintillators mounted on a semi-spherical dome structure at a flight distance of 175 cm from the target. Each neutron detector is of type 5" x 5" cylindrical BC501A coupled to a 5" photo multiplier tube (PMT Model Hamamatsu R4144). The mounting structure for the array has been built based on a geo-dome design using metallic (mild steel) hubs and links. The total solid angle coverage of the neutron detector array in this geometry is  $\sim 3.3\%$  of  $4\pi$ .



**Fig. 1:** Neutron detector array facility at Beam Hall II

The target chamber installed at the center of the array is a 100 cm diameter spherical vacuum chamber made out of steel with 4 mm wall thickness. The chamber contains provision for housing multiple targets, array of multi-wire proportional counters and other ancillary charged particle detectors. A photograph of the neutron detector arrays is shown in Figure 1. Custom built NIM electronics modules are used to process the large number of signals from the detector array [9]. Conventional analogue pulse shape discrimination (PSD) technique is used to discriminate the neutrons from gamma rays. This is accomplished electronically by the zero-cross timing technique. The home made single width PSD module contains the integrated electronics for zero-cross detection, time of flight and light output and can process signals from two detectors. To operate the photo multiplier tube (PMT), the high voltage is applied through a home made voltage divider base which contain resistor network and integrated charge sensitive pre-amplifier. For high voltages to detectors, we have built compact high voltage power supply using dc-dc converter that can deliver voltage up to 2000 volt and multi channels of such power supply with remote control has been built.

The fission fragments are detected using low pressure MWPCS mounted inside the target chamber. Two MWPCS have been fabricated based on the conventional wire chamber design using three electrodes, a central foil electrode and two position sensing wire frames [10]. The position information is derived using delay lines chips. The MWPCS have active area of 126 mm x 75 mm and are normally operated with iso-butane gas at typical pressure of the 2-3 Torr. Fast rise time about 3 ns and a position resolution about ~1 mm has been observed with in beam experiments. Two identical MWPCS mounted at the folding angles detect the complimentary fission fragments in coincidence. The data acquisition system for the detector array is based on VME setup using commercial analog to digital converters and time to digital converters. A PCI based fiber optic link VME controller is used to readout all parameters from a single crate using data acquisition software LAMPS [11].

The facility with fully functional 50 neutron detector array has been tested recently in an fusion-fission experiment using  $^{19}\text{F}$  beam on  $^{208}\text{Pb}$  target. Neutrons from 50 detectors were

recorded in coincidence with fission fragment detected by two MWPCs mounted at folding angles. Functionality of all sub systems including the vacuum system, electronics and data acquisition has been checked during the in-beam test and performance found to be good. Currently the implementation of electronics for all 100 detectors are in progress.

The author acknowledges the nuclear physics, accelerator and engineering group members at IUAC for their contributions at various stages.

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# An overview of BARC-TIFR Pelletron-Linac Facility

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## **Abstract**

The 14UD Pelletron Accelerator at Mumbai has recently completed twenty five years of successful operation. The accelerator is primarily used for basic research in the fields of nuclear, atomic, condensed matter and material science. The superconducting Linac booster provides additional acceleration to the ions from Pelletron injector up to A~60 region with E~5 MeV/A. Further, an alternate injector system to the Superconducting LINAC booster is planned as an augmentation programme, comprising of a superconducting ECR ion source, room temperature RFQ and superconducting low-beta cavity resonators. This talk will provide an overview of the recent developmental activities carried out at the Pelletron Accelerator Facility, resulting in enhanced overall performance and uptime of the accelerator. The application oriented programs initiated at Pelletron Accelerator and the current status of the alternate injector system at the Pelletron-Linac facility will also be discussed.



# EXFOR compilation and nuclear data measurement at KAERI/NDC

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## **Abstract**

KAERI/NDC has compiled domestic nuclear reaction data to construct the EXFOR database under the guidance of IAEA/NDS. The produced entries for EXFOR database are a total of 38 and these entries include mainly an experimental data for neutron total cross section, isomeric yield ratio and photo-fission yields by bremsstrahlung beam, the cross section production by charged particle. Since AASPP workshop in 2013, 2 of the total 38 entries entered at the EXFOR DB and one entry was reserved to enter the EXFOR as Compile status.

The production cross-sections of radionuclides from natural yttrium samples were determined by the activation method with the proton beam of 57 MeV at the Korea Multi-purpose Accelerator Complex (KOMAC). The photo-neutron cross-sections for  $^{59}\text{Co}$  were measured by using the end-point bremsstrahlung energies of 55, 60 and 65 MeV at the Pohang Accelerator Laboratory (PAL). The induced activities in the irradiated samples were measured by the HPGe detector and the  $\gamma$ -ray spectrum was analyzed by the Gamma vision program.

The activities for EXFOR compilation and measurement will be presented on the 5<sup>th</sup> AASPP workshop.





# Studies on high energy photon (bremstrahlung) and neutron induced fission of actinides and pre-actinides

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## Abstract

The fission products yields in the bremsstrahlung induced fission of pre-actinides (e.g.  $^{nat}\text{Pb}$  and  $^{209}\text{Bi}$ ) with end-point energies of 45-80 MeV and 2.5 GeV and actinides (e.g.  $^{232}\text{Th}$ ,  $^{238}\text{U}$  and  $^{240}\text{Pu}$ ) with 8-80 MeV were determined based on the off-line gamma ray spectrometric technique and by using the microtron (Mangalore), Khraghar (Navi-Mumbai), ELEBE (Germany), SAPHIR (France) and PAL (South Korea). Similarly, the yields of fission products in the 3.7-18.1 MeV neutron induced fission of actinides (e.g.  $^{232}\text{Th}$  and  $^{238}\text{U}$ ) based on  $^7\text{Li}(p,n)$  reaction neutron at Pelletron facility of TIFR were also determined by using the same technique. The importance of fission products yields data in the high energy photon and neutron induced fission of actinides and pre-actinides relevance to conventional reactors, AHWR and ADSs application as well as the basic understanding of fission phenomenon was discussed.

## Introduction

The yields of fission products in the high energy neutron and bremsstrahlung induced fission of pre-actinides (e.g.  $^{nat}\text{Pb}$ ,  $^{209}\text{Bi}$ ) and actinides ( $^{232}\text{Th}$ ) are needed for decay heat calculation [1] and thus important for the design of Accelerator Driven Sub-critical system (ADSs) [2,3]. In ADSs, the higher-energy (GeV) proton from the accelerator strikes a heavy element like W, Pb or Bi target, which yields a large number of neutrons by spallation reaction. The spallation target becomes a source of neutrons, which can achieve a self-sustaining fission chain in a subcritical core. However, during the spallation processes, along with high-energy neutrons, high-energy photons are also produced, which can cause fission and different types of nuclear reactions of the W, Pb or Bi target. Thus it is also important to measure the yields of the fission products in the high-energy neutron and photon-induced fission of W, Pb and Bi. The purpose of ADSs is not only to generate energy for power production but also for transmutation of long-lived fission products and incineration of the long-lived minor actinides to solve the problem of radioactive waste. Besides ADSs, the advanced heavy-water reactor (AHWR) [4] and fast reactor [5, 6] are important in the recent time for nuclear power generation. Other than these applications, the yields data of fission products are also needed for mass and charge distribution studies, which can provide information about the understanding of nuclear fission process. In view of this, the yields of various fission products in the bremsstrahlung induced fission of pre-actinides (e.g.  $^{nat}\text{Pb}$  and  $^{209}\text{Bi}$ ) with end-point energies of 50-70 MeV and 2.5 GeV and actinides (e.g.  $^{232}\text{Th}$ ,  $^{238}\text{U}$  and  $^{240}\text{Pu}$ ) with end-point energies of 8-80 MeV were determined based on the off-line gamma ray spectrometric technique. Similarly, the yields of fission products in the 3.72-12.52 MeV neutron induced fission of actinides (e.g.  $^{232}\text{Th}$  and  $^{238}\text{U}$ ) were also determined by using the same technique.

## Experimental Details and Calculations

The end-point bremsstrahlung energy was generated by impinging the electron beam on a 0.25 mm Ta or 0.1-1 mm W target. About 12 g of  $^{nat}\text{Pb}$ , 74 g of  $^{209}\text{Bi}$  and 77-114 mg of  $^{232}\text{Th}$  wrapped with Al foil were irradiated for 2-5 h with the end-point bremsstrahlung energies of 45-80 MeV and 2.5 GeV at the Pohang Accelerator Laboratory (PAL), South Korea [7-12]. In the

case of 8 and 10 MeV bremsstrahlung irradiations were carried out by using the microtron at Mangalore and EBC at Khraghar (Navi-Mumbai) in India [13,14]. At ELEBE (Germany), the 12-16 MeV bremsstrahlung induced fission of  $^{232}\text{Th}$  was carried out by using the photon beam generated by impinging the electron beam on a solid graphite beam dump [15]. At SAPHIR (France), the 11.5-17.3 MeV bremsstrahlung induced fission was carried out by using the photon generated by impinging the electron beam on a light water cooled cylindrical tungsten target of 5 cm diameter and 5 mm thickness [16].

In the case of neutron irradiation, the 3.72-12.52 MeV quasi-mono energetic neutron induced fission of actinides (e.g.  $^{232}\text{Th}$  and  $^{238}\text{U}$ ) were carried out by using the neutron beam from the  $^7\text{Li}(p,n)$  reaction with  $E_p=5.6-20$  MeV at Pelletron facility of TIFR [17,18]. The samples irradiated with photon (bremsstrahlung) or neutron beam were then taken for gamma ray counting by using HPGe detector coupled to a PC based 4-16K channel analyzer. From the photo-peak activities of the gamma lines of the fission products, the cumulative yields were determined by using decay equation [7-18]. From the cumulative yields, their mass chain yields (MD) were obtained by using the charge distribution correction [7-18].

## Results and Discussion

The yields of the fission products in the medium to high energy bremsstrahlung and neutron induced fission of pre-actinides (e.g.  $^{\text{nat}}\text{Pb}$ ,  $^{209}\text{Bi}$ ) and actinides (e.g.  $^{232}\text{Th}$ ,  $^{238}\text{U}$ ,  $^{240}\text{Pu}$ ) are important for decay heat calculations and thus design of the advanced reactors such as ADSs [1-3] and fast reactor [5,6]. This is because  $^{\text{nat}}\text{Pb}$  and  $^{209}\text{Bi}$  are used as the spallation source in ADSs, whereas  $^{232}\text{Th}$  and  $^{238}\text{U}$  are used as the breeding materials in AHWR [4] and fast reactor [5,6] for the production of the  $^{233}\text{U}$  and  $^{239}\text{Pu}$ . From India's perspective, which has abundant reserves of thorium, ADSs is relevant because one can also exploit its potential to design hybrid reactor systems that can produce nuclear power with the use of  $^{232}\text{Th}$  as the main fuel.

Typical mass yields distribution of the fission products in the bremsstrahlung induced fission of  $^{209}\text{Bi}$ ,  $^{232}\text{Th}$  and  $^{238}\text{U}$  are shown in Fig. 1-4 [7-16]. It can be seen from Fig. 1-3 that the mass yield distribution in the bremsstrahlung induced fission of pre-actinide (e.g.  $^{209}\text{Bi}$ ) is symmetric in nature, whereas for lighter actinide (e.g.  $^{232}\text{Th}$ ) is asymmetric with triple humped. For heavier actinide (e.g.  $^{238}\text{U}$ ), it is asymmetric with double humped. This is due to the different potential barrier for the pre-actinides, lighter and heavier actinides. It can be seen from Fig. 4 that the yields of fission products around mass number 133-135, 138-140, 143-145 and their complementary products are higher than the other fission products, which is due to effect of even-odd and shell closure proximity. Based on standard I model, the fission fragment around  $A=133-135$  have spherical 82n shell and deformed complementary fragment. Similarly, based on standard II model, the fission fragments around  $A=143-145$  have deformed 86-88n shell and the less deformed complementary fragment. On the other hand from Fig. 5, it can be seen that the FWHM of mass yield distribution increase and average mass ( $\langle A \rangle$ ) decrease with end-point bremsstrahlung energy for pre-actinides, which indicates the role of excitation energy ( $E^*$ ). Similar observations as mentioned above were also obtained from the medium energy neutron induced fission of  $^{232}\text{Th}$  and  $^{238}\text{U}$  [17,18]. The role of excitation energy can very well seen from the increase trend of symmetric products yields and thus the decreasing trend of the peak-to-valley (P/V) ratio in bremsstrahlung, neutron and proton induced fission of actinides (Fig.6).

## Acknowledgement

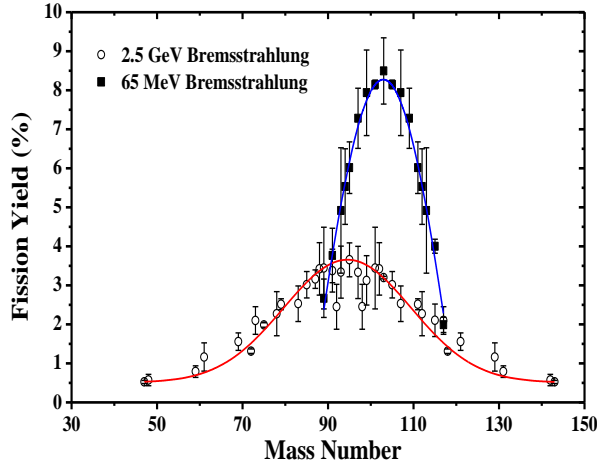
I thanks to Dr. K.L. Ramakumar, Group Director of RC & IG, Dr. S.K. Aggarwal, Associate Group Director of RC & IG, Dr. A. Goswami, Head, RCD, Prof. G. N. Kim Dept. of Phys., Daegu, KNU, Korea, Dr. R. Schwengner, ELBE, Dressden, Germany, Dr. Frédérick Carrel , SAPHIR, CEA, Saclay, Dr. S. Ganesan, BARC for their help in one or other way. I also thanks to the staff of the Microtron, Mangalore, EBC, Kharghar, Navi-Mumbai, PAL, Pohang, South Korea, ELBE, Dressden, Germany, SAPHIR, CEA, Saclay, France and BARC-TIFR Pelletron facility of Mumbai, India for the excellent operation and their support during the experiment.

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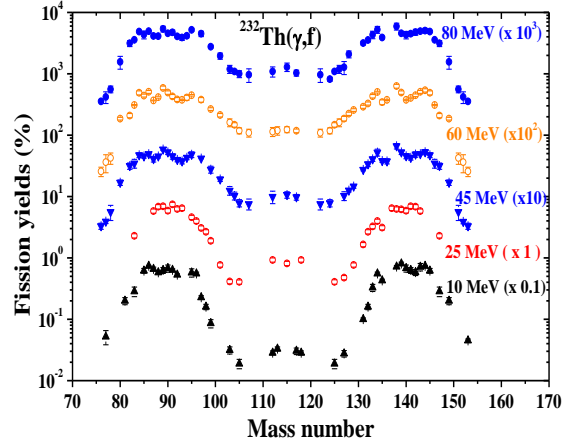
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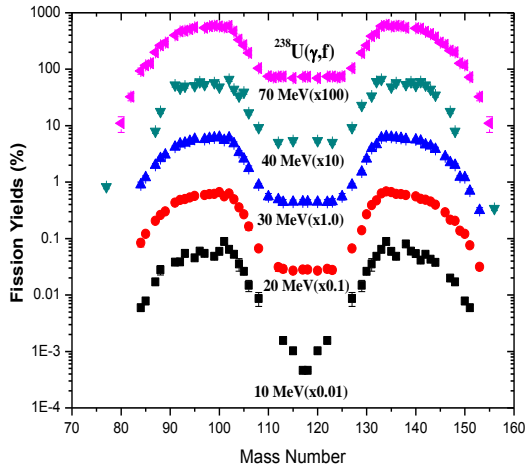
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 Eur. Phys. J. A. **50** (2014) 144.



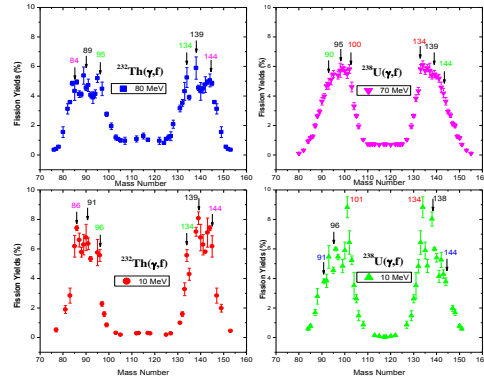
**Fig.1.** Plot of MD vs. A in  $^{209}\text{Bi}(\gamma, f)$



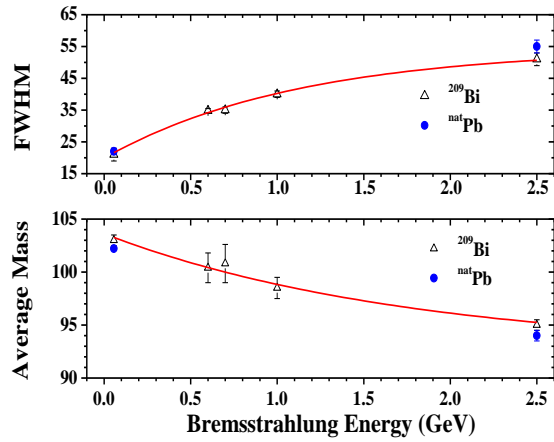
**Fig.2.** Plot of MD vs. A in  $^{232}\text{Th}(\gamma, f)$



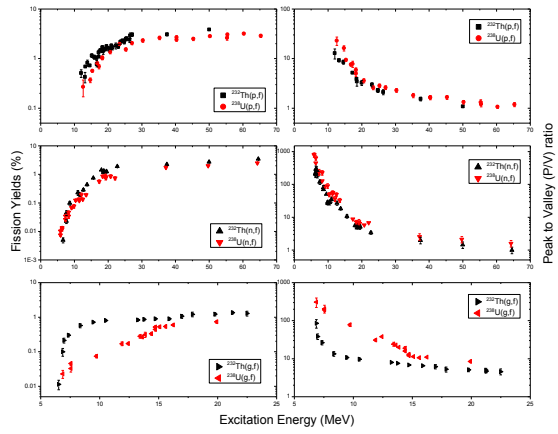
**Fig.3.** Plot of MD vs. A in  $^{238}\text{U}(\gamma, f)$



**Fig.4.** Plot of MD vs. A in  $^{232}\text{Th}^*$  and  $^{238}\text{U}^*$



**Fig.5.** Plot of FWHM &  $\langle A \rangle$  vs. energy P/V vs. excitation energy ( $E^*$ )



**Fig.6.** Plot of symmetric product yields ( $Y_s$ ) and Peak to Valley (P/V) ratio



# Atlas of nuclear isomers and their systematics

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## Abstract

Isomers can be viewed as a separate class of nuclei and offer interesting possibilities to study the behavior of nuclei under varied conditions of excitation energy, spin, life-time and particle configuration. We have completed a horizontal evaluation of nuclear isomers and the resulting data set contains a wealth of information which offers new insights in the nuclear structure of a wide range of configurations, nuclei approaching the drip lines etc. We now have reliable data on approximately 2460 isomers having a half-life  $\geq 10$  ns. A few of the systematics of the properties of nuclear isomers like excitation energy, half-life, spin, abundance etc. will be presented. The data set of semi-magic isomers strongly supports the existence of seniority isomers originating from the higher spin orbitals.

## 1. Introduction

The nuclear isomers are the excited meta-stable nuclear states. Large number of experimental and theoretical studies of isomers have been reported during the past few decades. The isomers provide a unique window into the nuclear structure properties in unusual situations. New isomers are being observed at a rapid pace, as it is now possible to measure the half-lives, ranging from picoseconds to years, in this new age of radioactive beams and modern instrumentation. Nuclear isomers are also known to have many useful applications and some of them promise to have still newer applications.

Our “Atlas of Nuclear Isomers” [1] lists more than 2460 isomers with a lower limit of half-life at 10 ns. It is quite interesting to look for the various regions in the nuclear chart where isomers exist. The answer to the question why, may be answered by a study of their properties and the underlying physics. We have, therefore, tried to systematize their various properties in a number of ways.

## 2. Types of Isomers

On the basis of the hindrance in decay, the nuclear isomers may now be classified into five types. The most common are the spin isomers. The other types are the K-isomer, the fission isomers and the shape isomers [2]. The fifth category, now increasingly recognized, is that of the seniority isomers.

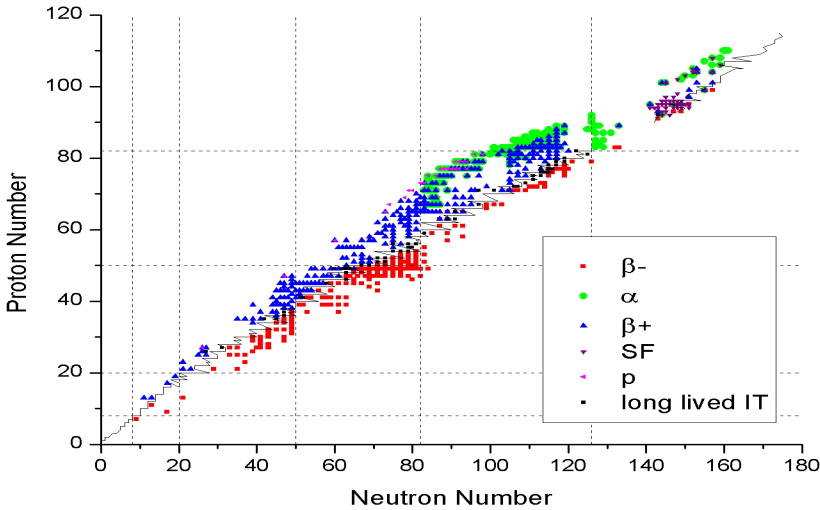
Spin isomers occur due to the difficulty in meeting the spin selection rules in their associated decay. Shell model dictates that most of the spin isomers will occur near the magic numbers. K-isomers exist due to the large change in K, where K is the projection of the total angular momentum on the symmetry axis. K becomes a good quantum number for axially deformed nuclei. When a nucleus is trapped in the secondary minimum at a super-deformed shape in the path of fission, it may decay via spontaneous fission or/and decay back to the first minimum, and is known as fission/shape isomer.

Seniority isomers are mainly found in the semi-magic nuclei, where the seniority becomes a good quantum number due to the dominant role of the intruder orbital in the respective valence space. The matrix elements of the even tensor operator between the same seniority states vanish near the mid-shell, leading to the seniority isomers [3]. For example, the  $10^+$  and the  $27/2^-$  isomers in the Z=50 isotopic and the N=82 isotonic chains behave as seniority isomers after the

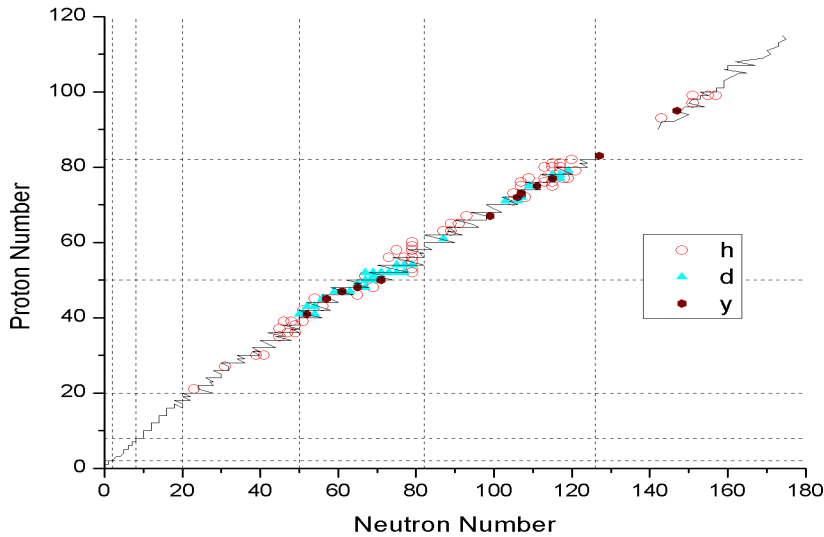
mid-shell, having seniority  $\nu=2$  and 3 configurations in the  $h_{11/2}$  intruder orbital respectively [4, 5].

### 3. Systematics

We present several systematics of the fundamental isomeric properties and the underlying physics. Nuclear isomers mostly decay via gamma decay, known as isomeric transition (IT). They can also decay via other decay modes as shown in Fig. 1, like beta decay ( $\beta$ ), alpha decay ( $\alpha$ ) or spontaneous fission (SF) and proton decay (p).



**Fig. 1:** The nuclear isomer chart with various decay modes, where the long lived IT decay isomers have a half-life cut-off at 1 min.

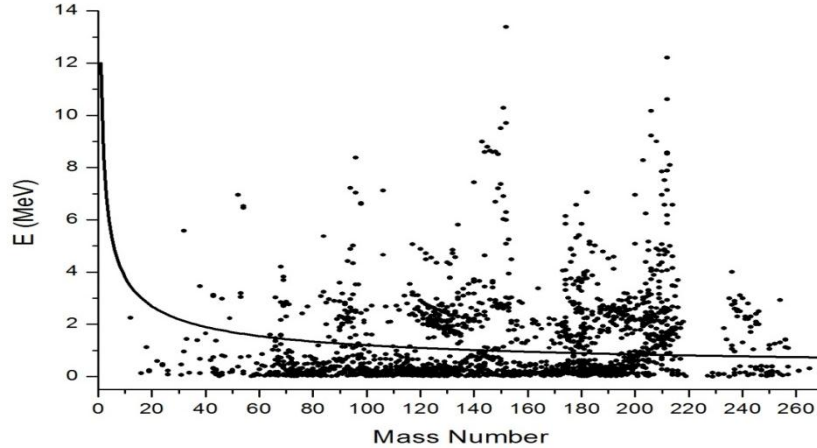


**Fig. 2:** The longest lived isomers having half-lives in  $h$  (hours),  $d$  (days) and  $y$  (years) as a function of  $(Z, N)$ . These isomers lie on the line of stability.

We have plotted the “Nuclear Isomer Chart” with their various decay modes in Fig. 1, along with the long lived IT mode. The dashed lines are drawn at the spherical magic numbers of protons and neutrons respectively. The solid zig-zag line (in grey) represents the beta stability line, where the half-life cut-off of stable isotopes is taken at  $10^{10}$  seconds. The isomeric nuclei, away from the beta stability line, and near to the proton/ neutron drip-line may dominantly decay via  $\beta^+/\beta^-$ -decay. The  $\alpha$ -decay can be seen in heavier nuclei having  $Z > 82$ . One can easily



observe that the SF mode is visible only in trans-actinides, as expected. The line has a gap at  $Z \sim 84-94$ , and  $N \sim 127-137$  corresponding to the  $\alpha$ -decaying radioactive nuclei. It is quite interesting to note that the nuclear isomeric chart also has the same gap, separating the two continents of isomers. We have also plotted the longest-lived isomers in the  $h$  (hours) to  $y$  (years) range, in Fig. 2, which mostly lie on the line of stability.

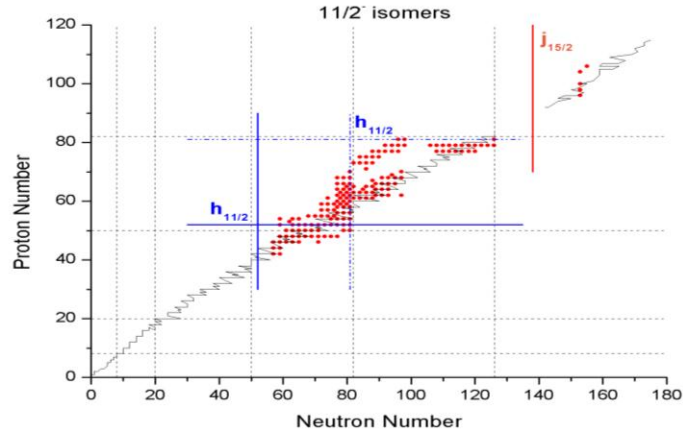


**Fig. 3:** The variation of the isomeric excitation energies with mass number  $A$ . The solid curve represents the pairing gap parameter  $\Delta=12/\sqrt{A}$ , which is observed to pass through the gaps.

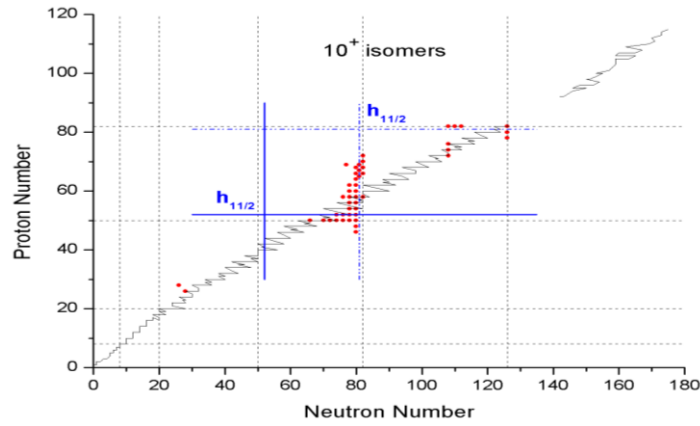
We plot the variation of the isomeric excitation energies with mass number  $A$  in Fig. 3. The solid curve represents the variation of pairing gap  $\Delta=12/\sqrt{A}$ , where  $A$  is the mass number. We find that the solid line passes through the gaps between the zero/one-quasiparticle isomers and the two/three-quasiparticle isomers, which requires the one proton/ neutron pair break up in the even-even and odd- $A$  isomeric nuclei respectively. There are some cases around the spherical magic numbers which fall within the gap. The separation of isomers by the pairing gap is very distinct and useful in classifying their quasiparticle structure.

We have separately plotted the occurrence of high-spin  $11/2^-$ ,  $10^+$ , and  $27/2^-$  isomers in the nuclear chart in Fig. 4, 5 and 6, respectively. We notice that most of the isomers neatly cluster around and in between the particle and hole regions corresponding to the  $h_{11/2}$  orbital, shown by the solid and dashed lines, respectively. The spin  $11/2^-$  originates from the unique-parity  $h_{11/2}$  orbital and low excitation energy suggests that they mostly have a 1-qp configuration. There are five cases far from this cluster, which lie around  $N=153$  with a few hundred keV excitation energy. These cases too have a 1-qp structure, which probably originate from the projection of the next  $j_{15/2}$  intruder orbital, and are possibly the K-isomers.

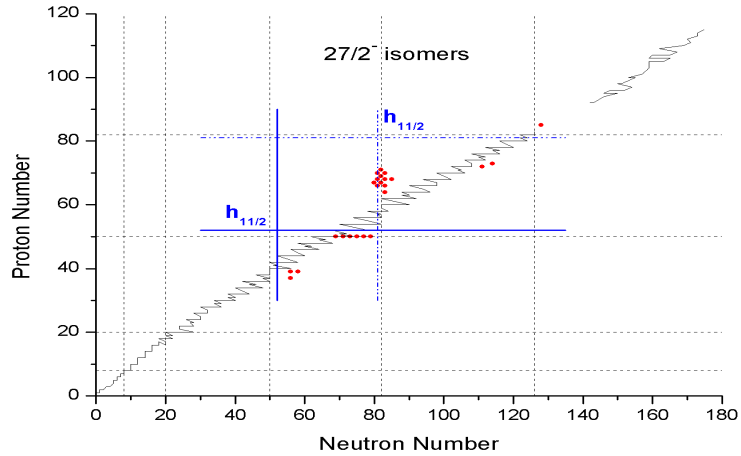
The  $10^+$  and  $27/2^-$  isomers mainly come from the 2-qp and 3-qp structure in the intruder  $h_{11/2}$  orbital except some cases. The two cases in the  $10^+$  isomers lie very low in mass region, having 6.5 MeV of isomeric excitation energy, and should have  $\sim 6$ -qp structure. There are also six more cases, of which four lie on the  $Z=82$  line and the other three lie on the  $N=126$  line, with one common doubly-magic  $^{208}\text{Pb}$  nucleus. Therefore, the  $10^+$  isomers at  $Z=82$ ,  $N=108-112$ , and 126 also have the many-qp structure, but not due to the  $h_{11/2}$  orbital. The  $10^+$  isomer in  $^{208}\text{Pb}$  has a high isomeric excitation energy of  $\sim 5$  MeV, due to the doubly shell-closure. One case in  $27/2^-$  isomer has  $Z=85$  and  $N=128$ , again beyond the scope of the  $h_{11/2}$  orbital. It has excitation energy of 1.3 MeV only, far less than required for core breaking, implying a many-qp structure.



**Fig. 4:** Occurrence of the 11/2<sup>-</sup> isomers throughout the nuclear landscape. The solid and the dashed lines depict the approximate particle and hole occupancies for the respective orbital shown in the figure.

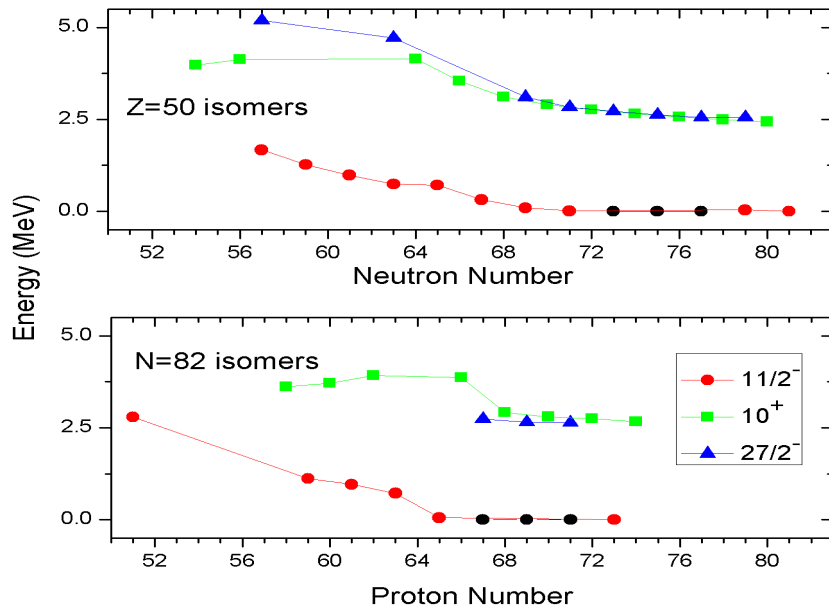


**Fig. 5:** Same as Fig.4 but for the occurrence of the 10<sup>+</sup> isomers.



**Fig. 6:** Same as Fig.4 but for the occurrence of the 27/2<sup>-</sup> isomers.

We have compared the  $11/2^-$ ,  $10^+$  and  $27/2^-$  isomers common to the  $Z=50$  and  $N=82$  chains in Fig. 7. The  $Z=50$  isomers occur due to the valence neutrons while the  $N=82$  isomers occur due to the valence protons. It is quite interesting to note that the experimental energy systematics for all these seniority isomers is almost identical irrespective of the valence proton/neutron space. The charge independent behavior of nucleon-nucleon interaction is also visible, so that the isomers of neutron-rich isotopes behave very similar to those of the neutron-deficient isotones. The relative energy gap between the  $10^+/27/2^-$  isomers and the  $0^+$  ground state/ $11/2^-$  isomer shows an energy transition from  $\sim 4$  MeV to  $\sim 3$  MeV around the middle of the valence space. The  $10^+/27/2^-$  isomers follow each other very closely, after the mid-shell, due to the dominant role of the intruder  $h_{11/2}$  orbital. Large scale shell model calculations are able to reproduce and validate these identical systematics for both the chains [5].



**Fig. 7:** The experimental excitation energy systematics of the  $11/2^-$ ,  $10^+$  and  $27/2^-$  isomers common to the  $Z=50$  isotopic and  $N=82$  isotonic chains.

#### 4. Summary

A few of the systematic features of the isomers have been presented in terms of the fundamental properties like half-life, spin-parity, energy, etc. Some of them are very new, and enable us to explore the physics behind them. Many of these universal and novel features of the isomers may open new perspectives in the field.

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# Theoretical calculation of $n+{}^6\text{Li}$ reaction below 20 MeV

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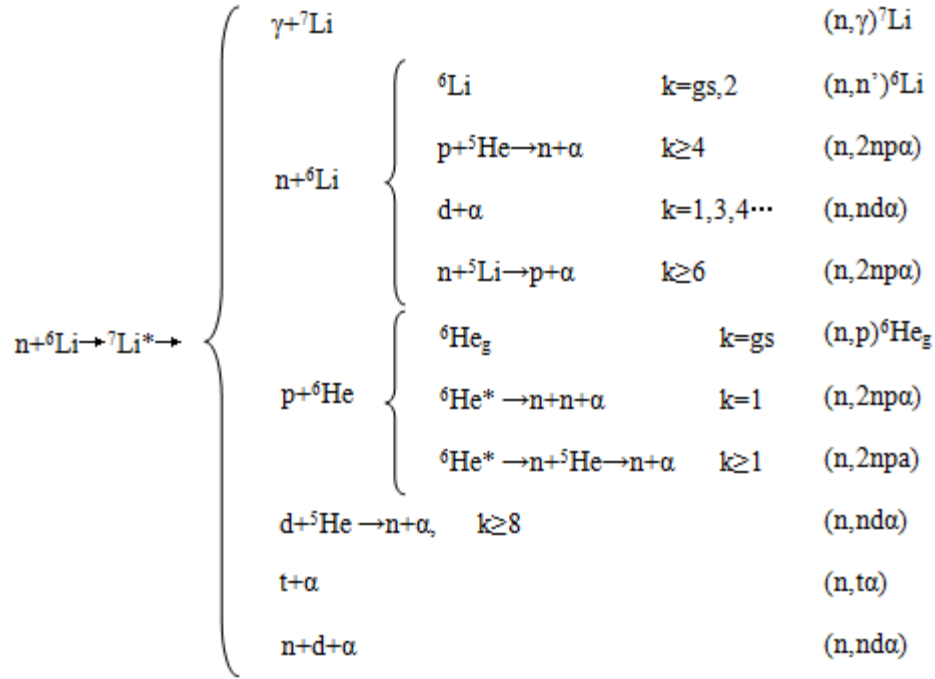
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R-matrix theory is an important theory of light, medium and heavy mass nuclides nuclear reaction in the resonance energy range. Statistical theory cannot describe the resonance. In 1936 G. Breit and E.P. Wigner presented the R matrix theory. In 1958, A.M. Lane and R.G. Thomas [1] published the paper named "R-Matrix Theory of Nuclear Reactions", and this paper is the classic work in R-matrix theory. Base on different assumptions, many R-matrix calculation methods have been developed, such as full R-matrix formalism, Multilevel Breit-Wigner R-matrix formalism, Adler-Adler R-matrix formalism, Eliminated channel R-matrix formalism, Reich-Moore R-matrix method, etc. Many nuclear data evaluations have been done by R-matrix methods. The evaluation of  ${}^6\text{Li}$  in ENDF/B-VII.0 have been done by G. M. Hale, and R. E. Azuma, S. Kunieda, Zhenpeng Chen, Chengjiu Zhu, Qingbiao Shen et al, also did a lot of work on R-matrix theory. EDA, AZURE, and AMUR etc are R-matrix codes for theoretical calculations and nuclear data evaluations.

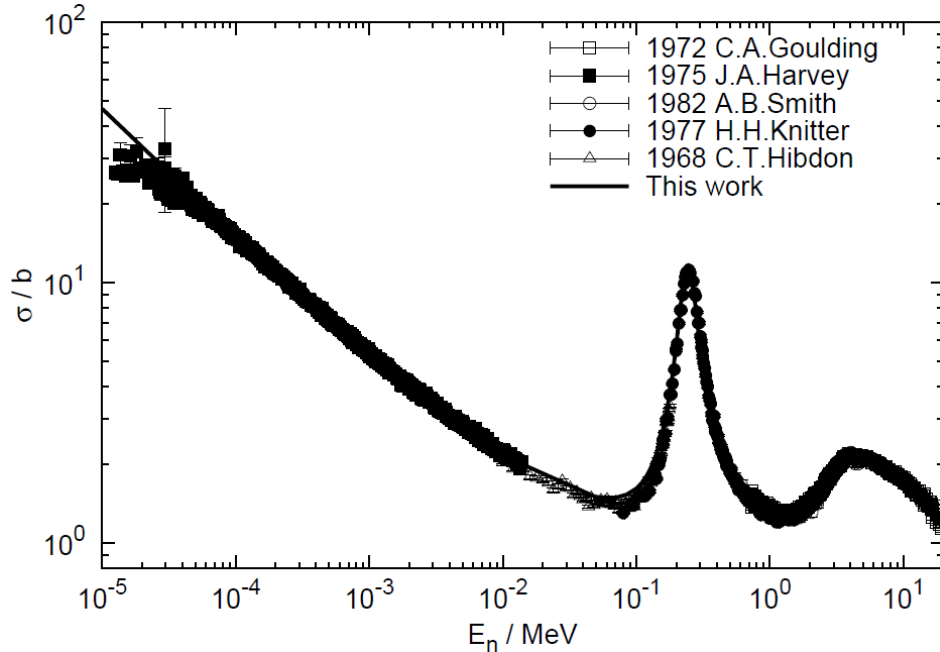
Lithium is widely used in reactor and nuclear energy. Neutron inducing  ${}^6\text{Li}$  can produce tritium, and the cross section of  ${}^6\text{Li}(n,t){}^4\text{He}$  below 1.7 MeV is a standard cross section. The structure of resonance peak is simple. The reaction process of  $n+{}^6\text{Li}$  is complex, and there are many light particles outgoing (**Fig. 1**). The residual nuclei are not stable, and they will continue to decay and emit particles. Because R-matrix theory can only deal with 2-bodys problem, primary emission process can be considered. But there is still a 3-bodys reaction channel, it need be instead with a fake 2-bodys channel.

FDRR [2] is a new R-matrix theoretical code compiled with Fortran language. It is used for light nuclei 2-body reaction, and can calculate cross sections, angular distribution, etc. The code contains 4 calculation methods, diagonal energy shift reduced R-matrix method, multi-level Breit-Wigner R-matrix method, un-diagonal energy shift reduced R-matrix method, and full R-matrix method. There are 3 kinds of parameters, resonance energy levels, energy shifts, and reduced resonance widths.

Multi-level Breit-Wigner R-matrix formalism is used for calculating  $n+{}^6\text{Li}$  reaction in this work. Adjusting the parameters and fitting the experimental data, a set of reduced width parameters has been gained. The calculating results are in good agreement with experimental data. The experimental data are obtained from the EXFOR database.



**Fig. 1:** The process of  $n+{}^6\text{Li}$  reaction.

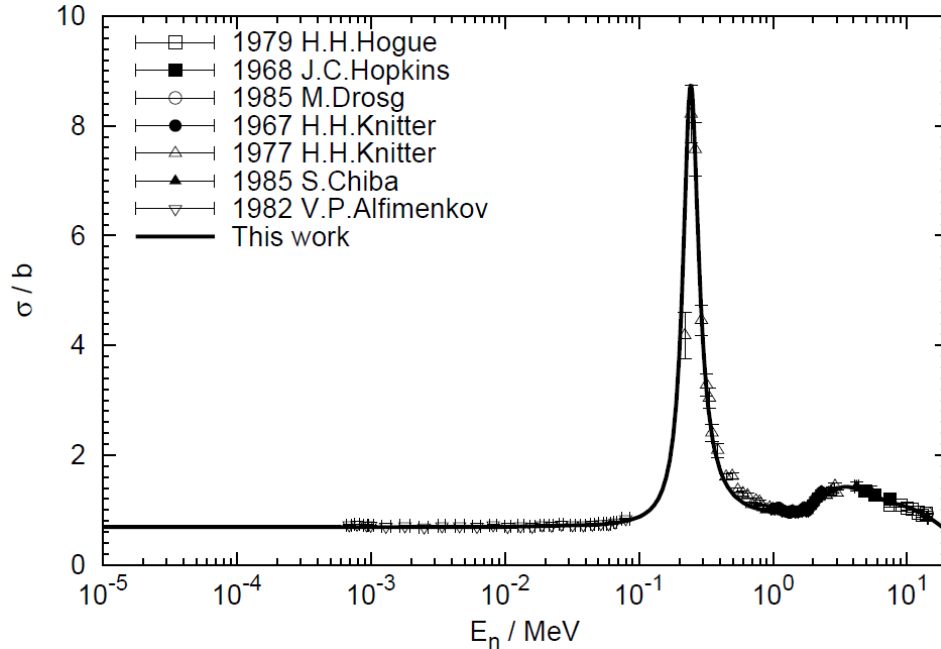


**Fig. 2:** The total cross section of  $n+{}^6\text{Li}$  reaction.

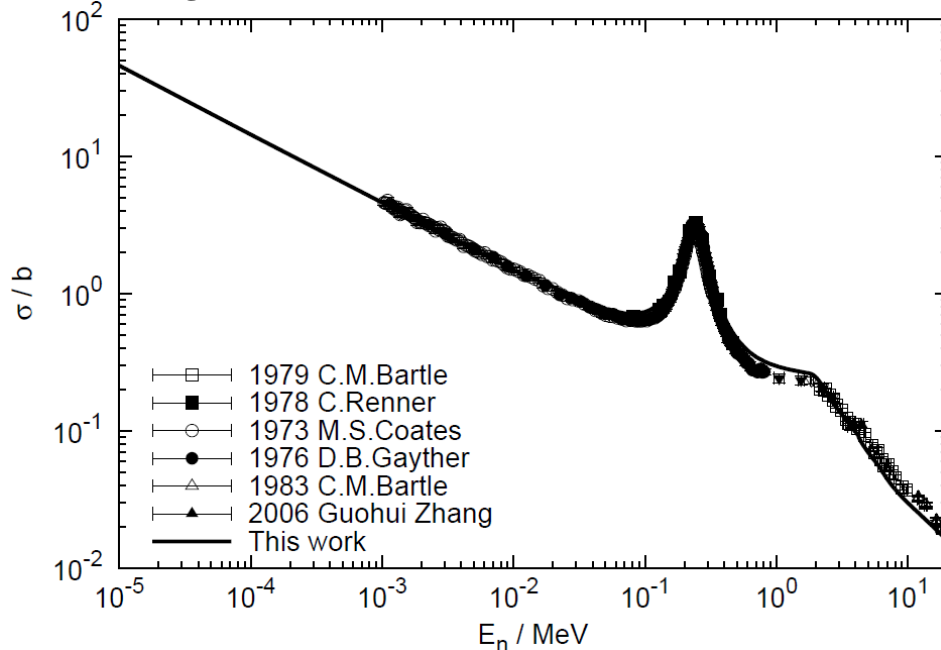
The total cross section of  $n+{}^6\text{Li}$  reaction is calculated in the energy range from 10keV to 20MeV, and well agrees with the experimental data [3-7] (**Fig. 2**).

**Fig. 3** is the result of elastic cross section compared with the experimental data [6,8-13] from EXFOR. In the energy range of unresolved resonance, the calculation of hard sphere scattering is given, and it is in good agreement with the experimental data. In the energy range of the resolved

resonance peak, the result and experimental values are in same good accordance. In high energy range, 3 fake resonance energy level is defined to make the cross section calculated higher and agree with the measured data. **Fig. 4** is the FDRR result of  ${}^6\text{Li}(n,t)$  reaction compared with experimental data[14-19].



**Fig. 3:** The elastic cross section of  $n+{}^6\text{Li}$  reaction.



**Fig. 4:** The cross section of  ${}^6\text{Li}(n,t)$  reaction.

FDRR code can not only gives the cross section, but also gives the angular distribution. Most results of angular distribution are in good agreement with experimental data.

The development of FDRR code provides a new light nucleus evaluation method for CNDC.

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# Activities for nuclear data measurement using charged particle-, neutron-, and photon-induced reactions in Korea

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## Abstract

We report the activities for nuclear data measurements using charged particle-, neutron-, and photon-induced reactions in Korea. Charged particle-induced reaction cross-sections are determined by using the stacked-foil activation technique at the MC-50 cyclotron facility in the Korean Institute of Radiological and Medical Science. Neutron-induced nuclear data were measured by using the pulsed neutron facility, which consists of an electron linear accelerator, a water-cooled Ta target with a water moderator, and a 12 m time-of-flight path. It can be possible to measure the neutron total cross-sections in the neutron energy range from 0.01 eV to few hundreds eV by using the neutron time-of-flight method and also measured the photo-neutron cross-sections by using the bremsstrahlung from the electron linac.

We collaborated with foreign researchers in India, Japan, Russia, China, and Vietnam. We also utilized foreign facilities in India, Japan, Russia, and USA.



# Update neutron nuclear data evaluation for $^{236,238}\text{Np}$

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## 1. Introduction

The nuclear data with high accuracy for actinides play an important role in nuclear technology applications, including reactor design and operation, fuel cycle, estimation of the amount of minor actinides (MAs) in high burnup reactors and to research to transmute the MAs to short half-lived nuclides or stable ones [1].

The nuclides of  $^{236}\text{Np}$  are generated via the  $\alpha$ -decay of  $^{240}\text{Am}$  or  $^{237}\text{Np}(n, 2n)$  and  $^{237}\text{Np}(d, t)$  reactions. And the nuclides of  $^{238}\text{Np}$  are generated via the  $\alpha$ -decay of  $^{242}\text{Am}$  or  $^{237}\text{Np}(n, \gamma)$  and  $^{237}\text{Np}(d, p)$  reactions. In the present work, according to the systematic trend of the total cross section and elastic cross section etc. of different Np isotopes, and based on the neutron optical model parameters (OMP) of  $^{237}\text{Np}$ , a new set of neutron optical model parameters were obtained for  $^{236,238}\text{Np}$ . Based on the new set OMP and the systematic trend of the cross sections of different Np isotopes, a full set of  $^{236,238}\text{Np}$  neutron nuclear data has been updated and improved by theoretical calculation. The present result has significant improvements over the data in CENDL-3.1 [2].

## 2. Reevaluation for $^{236,238}\text{Np}$

After CENDL-3.1 evaluation, there have not new measurements in  $n+^{236,238}\text{Np}$  and only the data in JENDL-4.0 [3] is new evaluated. In the present evaluation, we analyze and conclude the status of CENDL-3.1 and the the systematic trend of reaction cross section of different Np isotopes, modify and update the OMP, direct reaction contribution, inelastic scattering cross section and fission cross section. Other data file is directly adopted from CENDL-3.1.

### 2.1 Update OMP

The optical model parameters are the base of theoretical calculation, which will influence the total cross section, elastic scattering, inelastic scattering and their differential cross sections. In the present work, according to the systematic trend of the total cross section and elastic cross section etc. of different Np isotopes, and based on the neutron optical model parameters of  $^{237}\text{Np}$ , a new set of neutron optical model parameters were obtained for  $^{236,238}\text{Np}$ . The OMP of  $n+^{237}\text{Np}$  reaction is listed in **Table 1**, which are included proton, alpha, deuterium, tritium emission OMP.

The comparison between present results and other evaluation data in  $^{238}\text{Np}(n, \text{tot})$  and  $(n, \text{el})$  reactions is shown in **Fig. 1**. The present result is similar as JENDL-4.0, however the evaluated data of JEFF-3.1 is quite different with other evaluations.

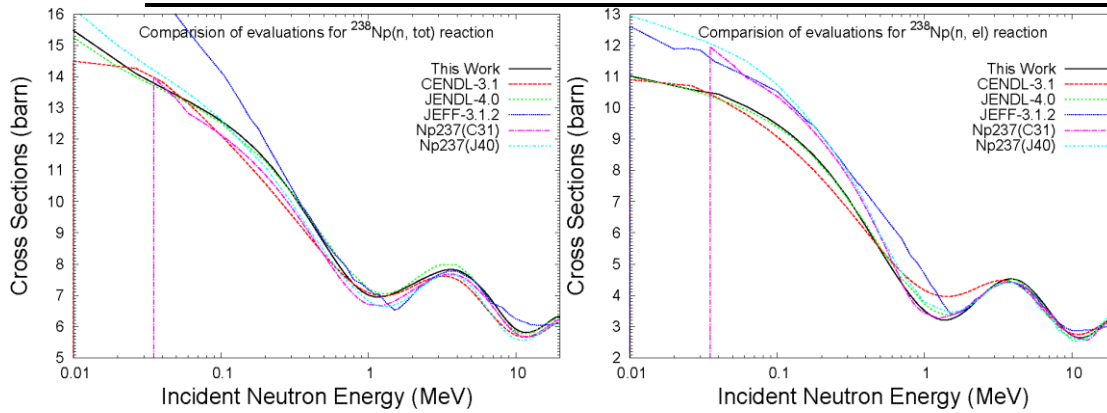
### 2.2 Inelastic Scattering Cross Section

35 discrete levels of  $^{238}\text{Np}$  are taken into account in model calculation for inelastic scattering cross section, and the first collective band of  $^{238}\text{Np}$  are taken into account in direct inelastic scattering calculation using ECIS code [4]. The spin of ground state of is  $6^-$ , and the first excited state is only given the spin information as 1. According to the theoretical calculation results using DWBA method, we assume the parity of the first excited state is “+”. Up to the fourth

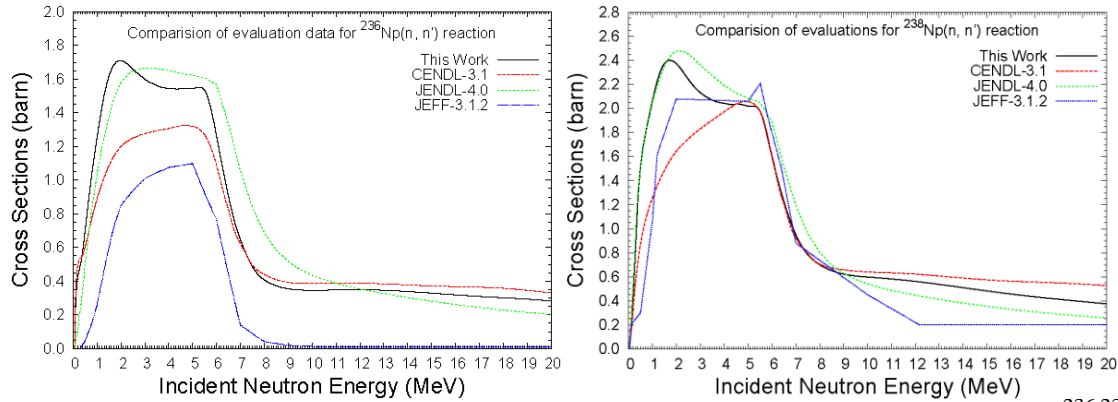
excited states were taken into account in direct inelastic scattering calculation using DWBA method. The calculated results of inelastic scattering cross section of  $^{236,238}\text{Np}$  are shown in **Fig. 2**.

**Table 1** Optical model parameters of  $n+^{237}\text{Np}$

| Parameter | OMP of different emission particals |        |         |           |         |
|-----------|-------------------------------------|--------|---------|-----------|---------|
|           | neutron                             | proton | alpha   | deuterium | tritium |
| AR        | 0.595                               | 0.500  | 0.520   | 0.810     | 0.750   |
| AS        | 0.592                               | 0.510  | 0.490   | 0.680     | 0.750   |
| AVV       | 0.591                               | 0.510  | 0.490   | 0.680     | 0.750   |
| ASO       | 0.565                               | 0.500  | 0.510   | 0.810     | 0.750   |
| XR        | 1.279                               | 1.250  | 1.350   | 1.150     | 1.200   |
| XS        | 1.022                               | 1.250  | 1.350   | 1.340     | 1.200   |
| XV        | 1.222                               | 1.250  | 1.350   | 1.340     | 1.200   |
| XSO       | 1.279                               | 1.250  | 1.350   | 1.150     | 1.200   |
| XC        | 1.250                               | 1.250  | 1.350   | 1.150     | 1.300   |
| UO        | 0.135                               | -2.700 | 0.000   | 0.000     | 0.000   |
| U1        | 0.293                               | 0.220  | 0.000   | 0.000     | 0.000   |
| U2        | -0.009                              | 0.000  | 0.000   | 0.000     | 0.000   |
| VO        | 49.145                              | 54.000 | 151.900 | 81.000    | 165.000 |
| V1        | -0.040                              | -0.320 | -0.170  | -0.220    | -0.170  |
| V2        | -0.015                              | 0.000  | 0.000   | 0.000     | 0.000   |
| V3        | -24.000                             | 24.000 | 50.000  | 0.000     | -6.400  |
| V4        | 0.000                               | 0.400  | 0.000   | 2.000     | 0.000   |
| VSO       | 6.200                               | 6.200  | 2.500   | 7.000     | 2.500   |
| WO        | 9.206                               | 11.800 | 41.700  | 14.400    | 46.000  |
| W1        | 0.037                               | -0.250 | -0.330  | 0.240     | -0.330  |
| W2        | -12.000                             | 12.000 | 44.000  | 0.000     | 0.000   |
| A2S       | 0.700                               |        |         |           |         |
| A2V       | 0.700                               |        |         |           |         |



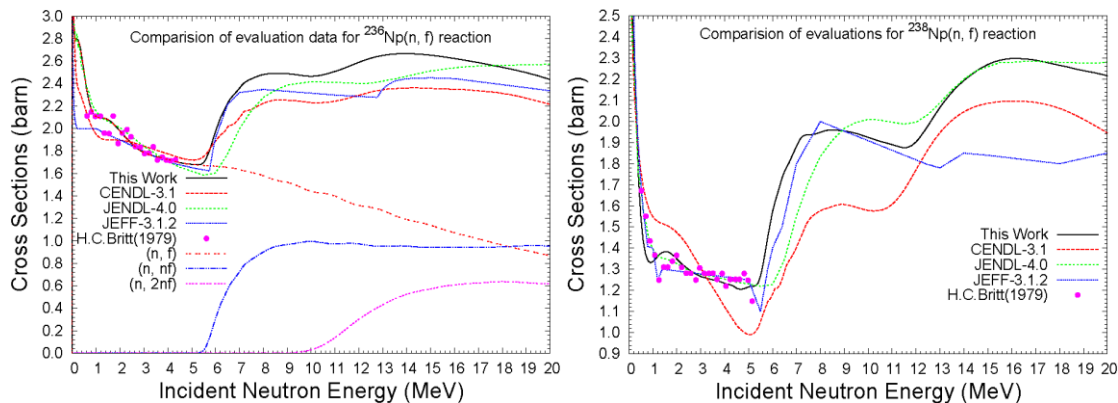
**Fig. 1** Comparison of evaluated data for (n, tot) and (n, el) reaction of  $^{238}\text{Np}$



**Fig. 2** Comparison of evaluated data for inelastic scattering cross section of  $^{236,238}\text{Np}$

### 2.3 Fission Cross Section

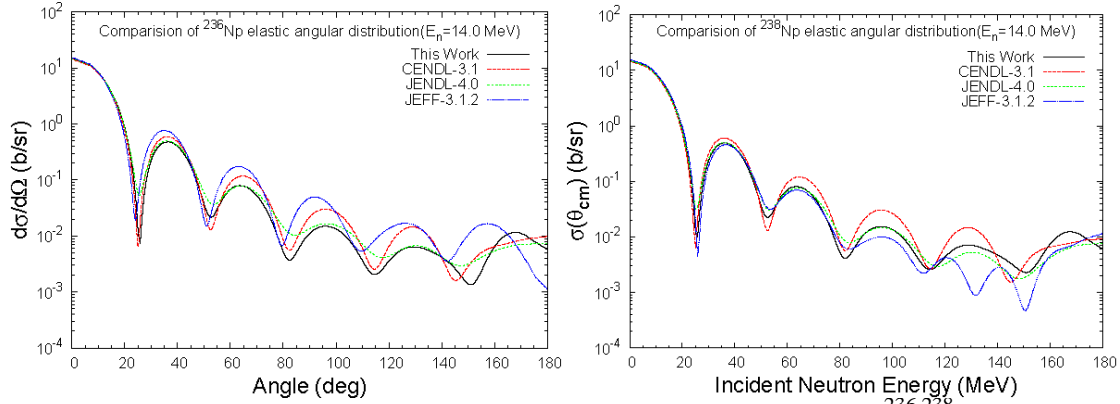
$^{236,238}\text{Np}$  fission cross section is no reaction threshold, so the first fission reaction contribution is decrease with the incident neutron energy. The calculation results of  $^{236,238}\text{Np}$  fission cross section is shown in **Fig. 3**. The present evaluated results agree well with the simulated (n, f) cross sections for exotic actinide nucleithe of H.C.Britt and J.B.Wilhelmy [5].



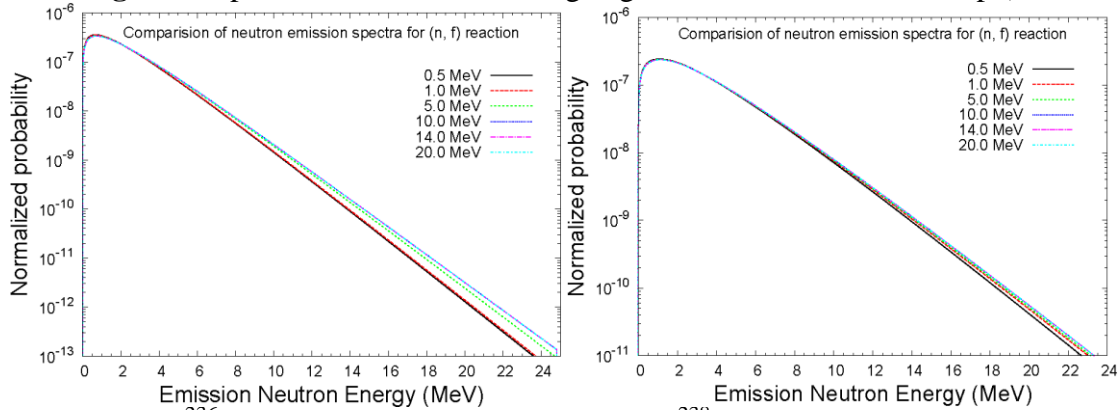
**Fig. 3** Comparison of evaluated data for (n, f) reaction of  $^{236,238}\text{Np}$

### 2.4 Differential Cross Section

Based on the new set of OMP and other theoretical calculated parameters, the differential cross sections for  $^{236,238}\text{Np}$  are obtained using FUNF code [6]. The comparison between present results and other evaluated data in elastic scattering angular distribution at 14 MeV is shown in **Fig. 4**. At forward angle, the evaluated data agree well with each other in shape and amplitude. At backward angle, there are exist discrepancy each other in shape and amplitude. The Madland-Nix method is applied to calculate fission spectra. The comparison of fission spectra with different incident neutron energy is shown in **Fig. 5**. The shape of fission spectra with different incident neutron energy is reasonable.



**Fig. 4** Comparison of elastic scattering angular distribution of  $^{236,238}\text{Np}$  ( $E_n=14$  MeV)



(a)  $^{236}\text{Np}$  fission spectra

(b)  $^{238}\text{Np}$  fission spectra

**Fig. 5** Comparison of fission spectra of  $^{236,238}\text{Np}$

### 3. Conclusion

According to the systematic trend of the total cross section and elastic cross section etc. of different Np isotopes, and based on the neutron optical model parameters of  $^{237}\text{Np}$ , a new set of neutron optical model parameters were obtained for  $^{236,238}\text{Np}$ . Based on the new set OMP and the systematic trend of the cross sections of different Np isotopes, a full set of  $^{236,238}\text{Np}$  neutron nuclear data has been updated and improved by theoretical calculation. The present result has significant improvements over the data in CENDL-3.1 in inelastic scattering cross section, fission cross section, (n, 2n) etc. The present results are more reasonable than the data in CENDL-3.1.

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# Applications of the photo-nuclear reaction data for activation analysis

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## 1. Introduction

In the relative method of activation analysis by continuum wide spectrum gamma-rays the same isotope is usually used for standard reference element and sample material in connection with different dependence of the reaction cross sections on the irradiation beam energy. But, in practice suitable isotopes for reference element are not always available.

So, in this paper, we suggest a new method for photo-activation analysis in which is used the correction factor. This factor takes into account the difference in the photo-nuclear reaction cross section dependence on the gamma-ray energy for standard reference isotope and sample elements. The correction factor is determined by three methods of experimental, theoretical and TALYS evaluation.

Pure metal foils of *Au*, *Cu* and *Mo* were irradiated by bremsstrahlung gamma-rays on the electron cyclic accelerator Microtron MT-22 at the Nuclear Research Center, National University of Mongolia. Gamma spectra of the activated metal foils were measured by HP-Ge detector to obtain element contents in the samples. It was shown that experimental results with correction factors are satisfactorily in agreement with real values of the element contents in the samples.

## 2. Theoretical

### 2.1. The correction factor

The photopeak area is given by following expression

$$S = \frac{N_A m \theta \varepsilon I}{\lambda A} (1 - e^{-\lambda t_i}) e^{-\lambda t_d} (1 - e^{-\lambda t_m}) \int_{E_{th}}^{E_{max}} \Phi(E_\gamma) \sigma(E_\gamma) dE_\gamma, \quad (1)$$

where:  $\sigma(E_\gamma)$  is the photo nuclear reaction cross section;  $E_\gamma$  is the  $\gamma$ -ray energy;  $\phi(E_\gamma)$  is the  $\gamma$ -ray flux;  $\lambda$  is the decay constant of the formed radioactive nucleus;  $\varepsilon$  is the detector efficiency;  $I$  is the gamma ray branch intensity;  $m$  is the target mass;  $N_A$  is the Avogadro's number;  $A$  is the atomic mass;  $\theta$  is the isotope abundance for pure metal sample;  $t_i$ ,  $t_d$  and  $t_m$  are the irradiation, cooling and measurement times, respectively;  $E_{th}$  is the threshold energy;  $E_{max}$  is the end point energy.

The so called integral effective cross section of the regarded nuclear reaction can be defined [1] as

$$\sigma_{eff} = \int_{E_{th}}^{E_{max}} \sigma(E_\gamma) f(E_\gamma) dE_\gamma, \quad (2)$$

where:  $f(E_\gamma)$  is the normalized differential energy spectrum of the bremsstrahlung  $\gamma$ -rays which is determined from the following expression

$$\int_{E=0}^{E_{max}} \phi(E_\gamma) dE_\gamma = \varphi = \varphi \int_{E=0}^{E_{max}} f(E_\gamma) dE_\gamma. \quad (3)$$

Using these definitions the integral in (1) can be expressed as:

$$\int_{E_{th}}^{E_{max}} \sigma(E_\gamma) \phi(E_\gamma) dE_\gamma = \sigma_{eff} \varphi , \quad (4)$$

where:  $\varphi$  is the integral flux density. Then, photopeak areas  $S_1$  and  $S_2$  for the reference and sample isotopes, respectively, are expressed from (1) and (4) as follows:

$$S_1 = \frac{m_1 N_A \theta_1 \varepsilon_1 I_1}{A_1 \lambda_1} \varphi \sigma_{eff1} (1 - e^{-\lambda_1 t_{d1}}) e^{-\lambda_1 t_{d1}} (1 - e^{-\lambda_1 t_{m1}}) \quad \text{and} \quad S_2 = \frac{m_2 N_A \theta_2 \varepsilon_2 I_2}{A_2 \lambda_2} \varphi \sigma_{eff2} (1 - e^{-\lambda_2 t_{d2}}) e^{-\lambda_2 t_{d2}} (1 - e^{-\lambda_2 t_{m2}}) \quad (5)$$

Here:

$$\sigma_{eff1} = \int_{E_{th1}}^{E_{max}} \sigma_1(E_\gamma) f(E_\gamma) dE_\gamma \quad \text{and} \quad \sigma_{eff2} = \int_{E_{th2}}^{E_{max}} \sigma_2(E_\gamma) f(E_\gamma) dE_\gamma . \quad (6)$$

As to the integral effective cross section  $\sigma_{eff}$ , we introduce the correction factor which is expressed by

$$K = \frac{\sigma_{eff1}}{\sigma_{eff2}} = \frac{\int_{E_{th1}}^{E_{max}} \sigma_1(E_\gamma) f(E_\gamma) dE_\gamma}{\int_{E_{th2}}^{E_{max}} \sigma_2(E_\gamma) f(E_\gamma) dE_\gamma} . \quad (7)$$

Then, the sample mass  $m_2$  can be written in the following form:

$$m_2 = \left( \frac{S_2 A_2 \lambda_2}{S_1 A_1 \lambda_1} \right) \left( \frac{\theta_1 \varepsilon_1 I_1}{\theta_2 \varepsilon_2 I_2} \right) \frac{(1 - e^{-\lambda_1 t_{d1}}) e^{-\lambda_1 t_{d1}} (1 - e^{-\lambda_1 t_{m1}})}{(1 - e^{-\lambda_2 t_{d2}}) e^{-\lambda_2 t_{d2}} (1 - e^{-\lambda_2 t_{m2}})} m_1 K . \quad (8)$$

In Ref.[2] the bremsstrahlung  $\gamma$ -rays spectrum  $f(E_\gamma)$  and the photo-nuclear reaction cross section  $\sigma(E_\gamma)$  were separately integrated as independent two integrals. We consider this method is incorrect and suggest a new method to calculate the integrals in (7). The bremsstrahlung  $\gamma$ -ray spectrum is difficult to measure in the wide energy range of 0 to end-point energy  $E_{max}$ . So, to obtain the correction factor  $K$ , we use the Schiff formula for the bremsstrahlung  $\gamma$ -ray energy spectrum.

## 2.2. The Schiff formula

Schiff obtained [3] the energy distribution cross section formula of the radiation from fast electrons in very thin targets by integration of the Bethe-Heitler differential bremsstrahlung cross section [4] over the scattered electron angles and the angles of the radiation:

$$\sigma(E_0, E_\gamma) = \frac{2Z^2 r_e^2}{137 E_\gamma} \left[ \left( 1 + \frac{k^2}{E_0^2} - \frac{2k}{3E_0} \right) \left( \ln a + 1 - \frac{2}{b} \tan^{-1} b \right) + \frac{k}{E_0} \left( \frac{2}{b^2} \ln(1+b^2) + \frac{4(2-b^2)}{3b^3} \tan^{-1} b - \frac{8}{3b^2} + \frac{2}{9} \right) \right] \quad (9)$$

where:

$$b = \frac{2E_0 E Z^{1/3}}{111 \mu E_\gamma} \quad \text{and} \quad \frac{1}{a} = \left( \frac{\mu E_\gamma}{2E_0 E} \right)^2 + \left( \frac{Z^{1/3}}{111} \right)^2 . \quad (10)$$

Here:  $\mu = mc^2$  is the rest energy of an electron ;  $E_0$  is the incident electron energy;  $E$  is the scattered electron energy:  $E_\gamma = E_0 - E$  ;  $Z$  is the scattering atom number,  $k$  is the scattering electron energy and  $r_e$  is the electron radius. In Ref.[5] the normalized differential energy spectrum of the bremsstrahlung  $\gamma$ -rays is obtained as following

$$f(E_\gamma) \propto \frac{1}{16} \times \frac{137 E_\gamma}{2 Z^2 r_e^2} \sigma(E_0, E_\gamma) . \quad (11)$$



So, from (9) and (11) can be obtained the following formula:

$$f(E_\gamma) = \left(1 + \frac{k^2}{E_0^2} - \frac{2k}{3E_0}\right) \left(\ln a + 1 - \frac{2}{b} \tan^{-1} b\right) + \frac{k}{E_0} \left(\frac{2}{b^2} \ln(1+b^2) + \frac{4(2-b^2)}{3b^3} \tan^{-1} b - \frac{8}{3b^2} + \frac{2}{9}\right). \quad (12)$$

### 2.3. The Lorentz-shaped resonance cross section

The energy dependence of the giant resonance absorption cross section for medium and heavy nuclei has often been approximated by a Lorentz-shaped resonance formula:

$$\sigma(E_\gamma) = \frac{\sigma_m \Gamma^2 E_\gamma^2}{(E_m^2 - E_\gamma^2)^2 + \Gamma^2 E_\gamma^2}. \quad (13)$$

Here:  $\sigma_m$  is the maximum cross section at the energy  $E_m$  of the  $\gamma$ -ray;  $\Gamma$  is the total energy width of the giant resonance at the half maximum.

## 3. Experimental Method

The bremsstrahlung gamma-rays were produced on the cyclic accelerator Microtron MT-22 of the Nuclear Research Center, National University of Mongolia, using the Ta 2 mm thick target for 22 MeV electron beam. Samples from natural pure metal thin disks of *Cu*, *Au* and *Mo* were irradiated by bremsstrahlung  $\gamma$ -rays. Thickness of the samples was around 0.2 mm and the diameter was ~1cm. All tight-parked metal foils were covered by 2mm thick *Cd* for shielding from the thermal neutron background. A distance between the front of *Cd*-cover and the *Al*-absorber for electrons was 8 mm. Irradiation time was 130 min. The electron beam current was ~6 $\mu$ A. *Cu* - foils were used as a reference element and for monitoring of gamma-ray flux. *Au* and *Mo* foils were studied to determine a sample mass by relative method.

After irradiation of the metal foils the  $\gamma$ -spectra were measured by 47 cm<sup>3</sup> HP-Ge detector with 2 keV energy resolution at the photopeak of 1332 keV for <sup>60</sup>*Co*. The detector efficiency was determined by <sup>252</sup>*Eu*-source. Measurement time was 600 sec for irradiated metal foils and 1000 sec for background. Half-lives of all studied isotopes were determined to test our experimental methods using the different gamma-peaks for each isotope. The experimental half-lives of all measured isotopes are calculated and the results were in good agreement with reference data. This fact means that our experimental method is reasonable and suitable to use in this measurement.

## 4. Results and Discussions

### 4.1. The correction factor calculations

The correction factor, which expressed by the formula (7), is determined by three methods: experimental, theoretical and TALYS evaluation.

In the experimental correction factor the experimental data [6] of photo-nuclear reaction cross sections for the reference and sample isotopes are used. In this case we were forced to use the Schiff formula of the bremsstrahlung gamma-ray spectrum in connection with difficulty to measure the flux for wide continuum energy spectrum gamma-rays. So, to obtain the experimental correction factor, the bremsstrahlung gamma-ray energy range from 0 to the end point divided by step 0.2 MeV and instead integral in (7) was taken a sum.

As to theoretical correction factor, the Schiff formula for the bremsstrahlung gamma-ray spectrum (12) and the Lorentz formula for the giant resonance cross section (13) are used. The Lorentz formula parameters were taken from [6-8]. In this case, the Schiff and Lorentz formulas

are firstly multiplied and after that the integrals in Eq.(7) are calculated by using the MatLab programming.

The TALYS code evaluation is used in the correction factor calculation as a third method. Calculated correction factors for studied isotopes are given in Table. 1.

#### 4.2. Experimental results and discussions

Mass of the *Au* and *Mo* samples were determined by formula (8) using the peak area from measured gamma-spectra and the calculated correction factors. Mass  $m_1$  and other data of *Cu*-foil, which was close contact with given sample, are used for determination mass  $m_2$  in formula (8) to reduce an uncertainty from gamma-ray flux difference for reference element and samples. Mass determination results of all samples are given in Table. 1 for the experimental, theoretical and TALYS evaluation correction factors, respectively. Relative uncertainties of the mass determination were obtained in comparison with real sample mass and were given in Table. 1, also.

Table 1. The correction factors and mass determination results

| Sample | $E_\gamma$ (keV) | Real mass (g) | By experimental correction factor |                     |                          | By theoretical correction factor |                     |                          | By TALYS code factor |                     |                          |
|--------|------------------|---------------|-----------------------------------|---------------------|--------------------------|----------------------------------|---------------------|--------------------------|----------------------|---------------------|--------------------------|
|        |                  |               | $K_{exp}$                         | Determined mass (g) | Relative uncertainty (%) | $K_{th}$                         | Determined mass (g) | Relative uncertainty (%) | $K_T$                | Determined mass (g) | Relative uncertainty (%) |
| Au-4   | 332.9            | 0.3409        | 6.49                              | 0.3365              | 6.3                      | 8.18                             | 0.2877              | 6.8                      | 7.10                 | 0.3656              | 7.3                      |
| Cu-1   | 1345.8           | 0.0283        |                                   | -                   |                          |                                  | -                   |                          |                      |                     |                          |
| Au-2   | 332.9            | 0.1442        | 6.49                              | 0.1614              | 13.5                     | 8.18                             | 0.1340              | 0.5                      | 7.10                 | 0.1652              | 14.6                     |
| Cu-3   | 1345.8           | 0.0283        |                                   | -                   |                          |                                  | -                   |                          |                      |                     |                          |
| Mo-6   | 739.4            | 0.0817        | 1.77                              | 0.0911              | 6.5                      | 2.16                             | 0.0775              | 4.4                      | 1.78                 | 0.0951              | 16.4                     |
| Cu-7   | 1345.8           | 0.0284        |                                   | -                   |                          |                                  | -                   |                          |                      |                     |                          |
| Mo-8   | 739.4            | 0.0877        | 1.77                              | 0.0979              | 6.6                      | 2.16                             | 0.0833              | 4.2                      | 1.78                 | 0.1022              | 16.5                     |
| Cu-9   | 1345.8           | 0.0287        |                                   | -                   |                          |                                  | -                   |                          |                      |                     |                          |

The relative uncertainties of the sample masses calculated by experimental, theoretical and TALYS evaluation correction factors are acceptable. However some effort to reduce uncertainty is needed.

#### Acknowledgement

The authors are grateful for the support of the "Nuclear Database" Project to Ministry of Education and Sciences of Mongolia. One of the authors, M.O. thanks Dr. N. Otsuka (IAEA) for his checking and assistance on the compilation D entries (D0731-0732; 0736; 0740-0741) of heavy ion reactions data performed on the European facilities for EXFOR library from Mongolia. She also wants to express her thanks to the Bhabha Atomic Research Centre (BARC) in Mumbai for hospitality and travel support from IAEA.

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# Recent EXFOR compilation in CNDC

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## 1. Introduction

The EXFOR library has become the most comprehensive compilation of microscopic experimental nuclear reaction data. It contains cross sections and other nuclear reaction quantities induced by neutron, charged-particle and photon beams, etc. Currently compilation is mandatory for all low and intermediate energy neutron and light charged-particle induced reaction data. Heavy-ion and photon induced reaction data are also additionally compiled on a voluntary basis.

Currently fourteen data centers are participating in the International Network of Nuclear Reaction Data Centres (NRDC) and are collaborating mainly for compilation and exchange of experimental data by using the common Exchange Format (EXFOR Format) under the auspices of the IAEA Nuclear Data Section (NDS).

## 2. EXFOR Compilation in CNDC

Since China joined IAEA at 1984 and China Nuclear Data Center (CNDC) joined NRDC in 1987, we are takes part in scanning Chinese journals and compiling EXFOR entries and collaborating with NRDC. CNDC is one specialized center at NRDC as shown in **Fig. 1**. Which the experiments are carried out by Chinese researcher, the experiments are measured in China and measurements are published in Chinese journals, compilation of bibliographic references (CINDA) to microscopic neutron reaction data and related data published in Chinese, CNDC need to scan and collect measured results and compile these data and information as EXFOR format including neutron and charge particle induced reactions. CNDC are respond more than 9 Chinese journals now such as **Fig. 2** and IAEA assigns EXFOR compilation task.

| Country      | Center                                 | Joined      |
|--------------|--|-------------|
| Russia       | Nucl. Struc. & Nucl. Reac. Data Centre | 1974        |
| Japan        | Japan Nuclear Reaction Data Center     | 1975        |
| Russia       | Centre for Exp. Photonuclear Data      | 1982        |
| <i>China</i> | <i>China Nuclear Data Center</i>       | <i>1987</i> |
| Japan        | Nuclear Data Center                    | 1991        |
| Korea        | Nuclear Data Evaluation Laboratory     | 2000        |
| India        | Nuclear Data Section                   | 2008        |

**Fig. 1:** The specialized center at NRDC

In 1985, IAEA and CNDC hold a working meeting about compilation in EXFOR. Fifteen charged particle EXFOR entries were transmitted to IAEA for NRDC communication at this meeting as shown in **Fig. 3** and started to compile neutron entries at 1989.

- ① Chinese Physics C(ENG/2007;HEN)
- ② Atom. Energy Sci. & Tech.(CHN/1959)
- ③ J. of Nucl. & Radiochemistry(CHN/1979)
- ④ Nuclear Physics Review(CHN/1984)
- ⑤ Nuclear Techniques(CHN/1978;+ENG/1989)
- ⑥ Com. of Nucl. Data Prog.(ENG/1989)
- ⑦ Nuclear Science and Techniques(ENG/1989)
- ⑧ Chinese Physics Letters(ENG/1984)
- ⑨ Chinese Physics B (ENG)
- ⑩ Acta Physica Sinica(ENG/1933)
- ⑪ Conference, Workshop etc.

Fig. 2: List of the responsibility of CNDC

- 01 Li Zhichang+ Yuanzineng Kexue Jishu 3(1977)229
- 02 Liang Qichang+ Yuanzineng Kexue Jishu 1(1977)10
- 03 Mao Zhenlin+ Conf. on Low Energy Nucl. Phys.3(1972)
- 04 Yuan Rongfang+ Chin. J. Nucl. Phys. 3(1981)155
- 05 Jiang Chenglie+ Conf. on Low Energy Nucl. Phys.3(1972)
- 06 Sun Hancheng+ Yuanzineng Kexue Jishu 3(1984)329
- 07 Yan Chen+ Chin. J. Nucl. Phys. 2(1980)137
- 08 Sun Hancheng+ Yuanzineng Kexue Jishu 2(1981)185
- 09 Ma Weiyi+ Chin. J. Nucl. Phys. 2(1980)239
- 10 Shen Wenqen+ High Energ.Phys.Nucl.Phys.1(1977)70
- 11 Tao Zhenlan+ Canadian Nucl Technol.45(1987)
- 12 X.Long+ NST-001 (1985)
- 13 Long Xianguan+ NST-003 (1989)
- 14 Tao Zhenlan+ Chin. J. Nucl. Phys. 3(1981)242
- 15 Tao Zhenlan+ Yuanzineng Kexue Jishu 5(1983)506

Fig. 3: CNDC provided the first 15 charged particle EXFOR entries




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|                                       | No. | Ref.   | Title   | Vol. | Issue | Page    | Lab    | Publsh date | Author              | Status    | Action    | Delay date(n) | Compiler       | Entry No. | Menu |                           |
|---------------------------------------|-----|--------|---|------|-------|---------|--------|-------------|---------------------|-----------|-----------|---------------|----------------|-----------|------|---------------------------|
| <a href="#">修改</a> <a href="#">详细</a> | 1   | J.RCA  | Cross section measurements for gallium in the neutron energy range of 13.5 to 14.6 MeV  | 100  | 4     | 231     | ICPHEU | 201202      | J.Luo               | Allocated | Compiling | 6             | Yousiang ZHANG | 32704     | n    | Assigned this entry & ... |
| <a href="#">修改</a> <a href="#">详细</a> | 2   | J.IFC  | Cross-sections for formation of <sup>99m</sup> Tc through <sup>nat</sup> - <sup>99</sup> Zr( <sup>n</sup> , <sup>p</sup> ) <sup>99m</sup> Tc reaction induced by neutrons at 13.5 and 14.6 MeV  | 81   | 5     | 495     | ICPHEU | 201201      | Yunhua Luo          | Allocated | Compiling | 7             | Yousiang ZHANG | 32702     | n    | Assign this entry it ...  |
| <a href="#">修改</a> <a href="#">详细</a> | 3   | J.IFC  | Cross-sections for <sup>10</sup> B( <sup>n</sup> , <sup>α</sup> ) <sup>7</sup> Li and <sup>10</sup> B( <sup>n</sup> , <sup>α</sup> ) <sup>7</sup> Li reactions on <sup>238</sup> U, under several neutron energy of 14 MeV                      | 81   | 10    | 1563    | ICPHEU | 201204      | Tubin Zhang         | Allocated | Compiling | 4             | Jinin YANG     | 32701     | n    | Assign this entry it ...  |
| <a href="#">修改</a> <a href="#">详细</a> | 4   | J.IFR  | Cross section measurements for <sup>141</sup> Pm( <sup>n</sup> , <sup>α</sup> ) <sup>141</sup> Pu reaction at neutron energies from 13.5 to 14.6 MeV  | 293  | 1     | 261     | ICPHEU | 201204      | Yunhua Luo          | Allocated | Compiling | 4             | Geochang CHEN  | 32703     | n    | It is assigned at Aug ... |
| <a href="#">修改</a> <a href="#">详细</a> | 5   | J.IWH  | <sup>209</sup> Pb( <sup>n</sup> , <sup>p</sup> ) <sup>209</sup> Bi 裂变产物产额测量   | 34   | 2     | 88      | ICPHEU | 201203      | 刘桂芝                 | Allocated | Compiling | 5             | Geochang CHEN  | 32700     | n    | It is assigned at Au ...  |
| <a href="#">修改</a> <a href="#">详细</a> | 6   | J.EPJA | New determination of the astrophysical <sup>12</sup> C( <sup>n</sup> , <sup>p</sup> ) <sup>12</sup> B( <sup>n</sup> , <sup>p</sup> ) factors and reaction rates via the <sup>12</sup> C( <sup>n</sup> , <sup>α</sup> ) <sup>12</sup> B reaction | 48   | 2     | 13      | ICPHEU | 201202      | T.J.Li              | Allocated | Compiling | 6             | Xi TAO         | 30054     | cp   | Assigned this entry ...   |
| <a href="#">修改</a> <a href="#">详细</a> | 7   | J.IWR  | Cross section measurements for <sup>10</sup> B( <sup>n</sup> , <sup>α</sup> ) <sup>7</sup> Li reaction on various isotopes at neutron energies from 13.5 to 14.6 MeV  | 35   | 5     | 445-448 | ICPHEU | 201105      | PU Zhongsheng (濮忠胜) | Allocated | Compiling | 15            | Yousiang ZHANG | 30692     | n    | Finish compile at Ja ...  |
| <a href="#">修改</a> <a href="#">详细</a> | 8   | AST    | 13 非电子俘获所引起的一些反应的核与品质   | 22   | 2     | 250     | ICPHEU | 196602      | 程晓五                 | Allocated | Finalized | 958           | Geochang CHEN  | 20000     | cp   | Finalized this entry ...  |
| <a href="#">修改</a> <a href="#">详细</a> | 9   | JF/CS  | Fusion-fission and quasi-fission competition in the <sup>235</sup> U( <sup>n</sup> , <sup>ff</sup> ) <sup>140</sup> La reaction   | 282  | 1     | 12013   | ICPHEU | 201103      | X.Q.Zhang           | Allocated | Compiling | 17            | Yousiang ZHANG |           | cp   | This is calculation ...   |
| <a href="#">修改</a> <a href="#">详细</a> | 10  | J.EPJA | Level structures in the <sup>115</sup> Sn nucleus   | 47   | 11    | 141     | ICPHEU | 201111      | C.B.Li              |           |           | 9             |                |           | cp   | This paper could't b ...  |

Fig. 4: EXFOR compilation managed Website

Present we have a small group to attend EXFOR compilation work. Everyone respond to scan 2 journals, collect the scanning results of all responsible journals, and assign neutron and charged particle tasks. After that, upload the information such as the assigned entry No., paper in pdf, author, publication date, delayed date, the compiler and the processing of compilation to our EXFOR compilation managed Website as shown in Fig. 4.

Since 2009, CNDC compiled 74 EXFOR entries as shown in Fig. 5 which is included 49 neutron and 25 charged particle entries. Up to now, we still have more than 40 articles should be compiled in 2014. We can find recently the charged particle induced reaction measurement is become more and more.

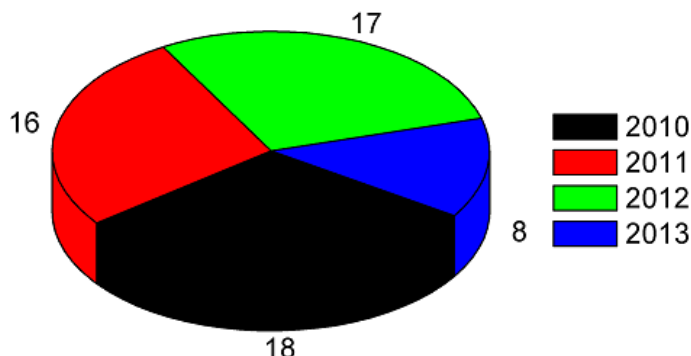


Fig. 5: Compiled EXFOR entries of each year

### 3. Nuclear Data Service

CNDC provides the nuclear data service in China for different institute, school or other requirements. CNDC joined the developing of Chinese basic database and established a “The Database of Nuclear Physics” website as shown in Fig. 6 including experimental data (EXFOR), evaluation data, decay data, nuclear structure, astrophysical data and nuclear data for medical applications for online retrieve and plotting, and the website is “www.nuclear.csdb.cn”. CNDC also established the mirror site of IAEA-NDS at Aug. 31 2013, which is “www-nds.ciae.ac.cn”. And the database of this mirror site is update with IAEA-NDS website at the same time. Up to now, the contents of mirror site includes EXFOR database and evaluation database as shown in Fig. 7. And the contents will be enriched in the future.



Fig. 6: Webpage of “The Database of Nuclear Physics”



Fig. 7: Webpage of IAEA-NDS mirror site

#### 4. Conclusion

The needs for experimental reaction data are always growing. CNDC response to scan, collect and compile the experimental information which are carried out by Chinese researcher, the experiments are measured in China and measurements are published in Chinese journals, and related data published in Chinese. Present CNDC have a small group to attend EXFOR compilation work and construct an EXFOR compilation managed Website for EXFOR compilation organization. CNDC will continue to scan, collect and compile EXFOR data and collaborate with NRDC.



## Nuclear data needs for fast reactors

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### **Abstract**

The nuclear data, i.e., the numerical information about every nuclide - especially those representing the probabilities of various nuclear interactions and of radioactivity - of interest in a nuclear fission reactor are among the most essential inputs to be known a priori, to the best possible accuracy, for the design of nuclear reactor. The nuclides of interest cover not just (1) the fuel nuclides, the containers, the coolant, the moderator (if any), etc., that are initially inserted, but also (2) the actinides, the fission products, etc. that would be produced from the moment the reactor goes into operation and (3) the decay products that are produced even while the reactor is shutdown. The nuclide-list is known to cover a few hundreds. The neutron-nuclear interaction cross-section data, required for a few tens of reactions, very sensitively depend on the nuclide species and the neutron energy. Hence the data requirement significantly varies between thermal and fast reactors. The present talk is intended to touch upon the kinds and forms of nuclear data needed in the design and analysis of fast reactors. The recent variants available in the databases and some inter-comparison results will also be presented.



## Nuclear data needs of Indian nuclear program

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Currently 17 Pressurised Heavy water Reactors (PHWRs), 2 Boiling water reactors (BWRs) and 1 Pressurised water reactor (PWR) are being operated for power production by Nuclear Power Corporation India Limited (NPCIL). For PHWRs, different types of fuel bundles are simulated by the integral transport theory code, CLUB using a combination of collision probability method and interface current technique and employing IAEA supplied 69 /172 group WIMS cross section library based on ENDF-BVI, BVII. Ring power factors are calculated at different burnups and are used to estimate linear heat rating. The two group neutron cross sections of different type of lattices at different core irradiations are also generated by lattice code CLUB. Wherever reactivity devices are present, supercell approach is adopted and the suitable incremental absorption cross sections are obtained using BOXER which is based on 3-D integral transport theory considering two neutron energy groups. Using the appropriate properties for normal lattices and ones affected by reactivity devices, fuel management and core follow up studies are carried out using 3-D diffusion theory based TRIVENI code.

The KAPS-1 power rise transient on March 10, 2004 brought to focus the importance of accurate nuclear data for reactor physics estimation in Indian PHWRs. With IAEA supplied libraries in WIMS format we could satisfactorily resolve the rate of power increase. Stability analysis and sensitivity analysis was carried out for different incore burnup situations resulting from peak flux operation. The quantification of output uncertainties is necessary to adequately establish safety margins of nuclear facilities. The uncertainties in the integral parameters such as reactivity worth and coefficients due to cross section can be assessed using cross section covariance data produced directly from the uncertainties of measurements. Covariance data processing codes and sensitivity analysis tools have to be developed. The part of the uncertainty associated with the nuclear data is estimated by the adjustment method using different covariance evaluations using data sets such as ENDF B/VI, ENDF B/VII, JEF, JENDL etc. Other uncertainties are due to material dimension, density, isotopic composition measurements etc and the sensitivity matrix have to be developed based on measurements.

For shielding design, nuclear heat generation rate, activity calculation, neutron flux related studies, doses in operating and accident conditions, decay heat, radioactivity release, photo neutron estimations accurate nuclear data and processing tools are needed to be augmented.

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2. S.B. Degwekar, "BOXER-3, A 3D Integral Transport theory code for PHWR supercell", Report No. BARC-1295 (1985)



# Sensitivity studies on nuclear data for thorium fuelled Advanced Heavy water Reactor (AHWR)

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## Abstract

Sensitivity studies and uncertainty analyses on safety parameters for reactors are an important analysis tool for qualifying the basic nuclear cross section data. It is also helpful in providing adequate margins at the design stage. An exercise was taken up to quantify the components of the safety coefficients such as coolant void reactivity (CVR), fuel temperature reactivity (FTR) and channel temperature reactivity (CTR) for the First-Of-A-Kind thorium fuelled Advanced Heavy Water Reactor (AHWR) being designed in India.

It has been established that the major contributors are H,  $^{232}\text{Th}$ ,  $^{233}\text{U}$ ,  $^{239}\text{Pu}$  and  $^{240}\text{Pu}$  for the CVR of AHWR-Ref fuel.  $^{239}\text{Pu}$  has a positive contribution and a negative slope with burnup.  $^{232}\text{Th}$  and  $^{233}\text{U}$  have a negative contribution and a positive slope with burnup. H has a positive contribution to the CVR and a positive slope. The sensitivity to channel temperature reactivity (CTR) shows that the major contributors are H and  $^{233}\text{U}$  which have positive components and  $^{232}\text{Th}$ ,  $^{239}\text{Pu}$  and  $^{240}\text{Pu}$  which have negative components. The ratio of the Maxwellian averaged capture to fission cross section of  $^{233}\text{U}$  and  $^{239}\text{Pu}$  has been used to explain the difference in the sign of the contribution to CTR from these isotopes.

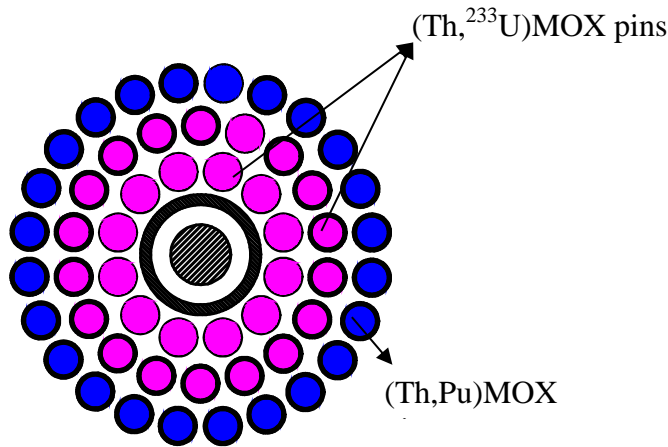
Uncertainty analysis has been taken up for the newer version of AHWR lattice fuelled with (Th,LEU)MOX fuel with respect to processed multi-group data, manufacturing tolerances such as fuel density and enrichment variations. The modelling uncertainties are as high as 25% in the coolant void coefficient. The sensitivities are larger with the use of a larger number of neutron energy groups. The Doppler coefficient of reactivity is uncertain by about 4%. The most sensitive parameter in the uncertainties in manufacturing tolerances is the variations in fuel pellet diameter which amounts to about 11%. Uncertainties in fuel density give rise to a sensitivity of about 7%.

## 1. Introduction

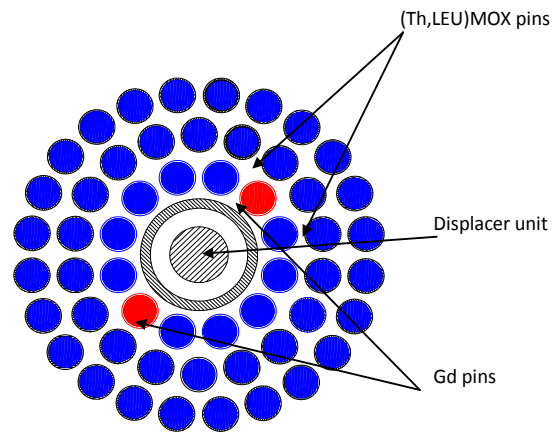
In India, the design on Advanced Heavy water Reactor (AHWR) based on thorium is in its advanced stage of development [1]. It is a first-of-a-kind reactor designed with many passive safety features which are required to be qualified. AHWR is a boiling water cooled reactors and the design objective is to achieve negative coolant void coefficient of reactivity both at normal operating and accidental conditions. The sensitivity to this safety parameter is an important input to the safety studies. In this paper, we discuss several types of sensitivity studies taken up for the integral parameters and reactivity coefficients for the AHWR-reference and the AHWR-LEU variant.

The fuel cluster for the equilibrium core called as the composite cluster consists of 54 pins arranged in an array of 12, 18 and 24 pins. In the AHWR-Ref fuel cluster, inner 30 pins have (Th, $^{233}\text{U}$ )MOX fuel and the outer 24 pins (Th,Pu)MOX fuel. The first array of 12 pins has a  $^{233}\text{U}$  content of 3.0% by weight and the middle 18 pins have 3.75%  $^{233}\text{U}$ . The 24 (Th,Pu)MOX pins in the third array have an axial gradation of 4.0 % Pu in the lower half of the active fuel and 2.5 % Pu in the upper part [2]. For this analysis only 2.5% Pu fuel has been taken. In the AHWR-LEU variant, the cluster uses a differential LEU composition of 30% LEU by weight in the inner 12 pins, 24% LEU by weight in the middle ring of 18 pins and an average of 16% by weight in the outer ring of 24 pins [3]. The LEU composition is assumed to be 19.75%  $^{235}\text{U}$  and 80.25%  $^{238}\text{U}$ . The fuel is composed of a mixture of thorium and LEU as oxides. For these simulations only the

average LEU content of 16% The cross sections of both the fuel assemblies are presented in Fig. 1a and 1b.



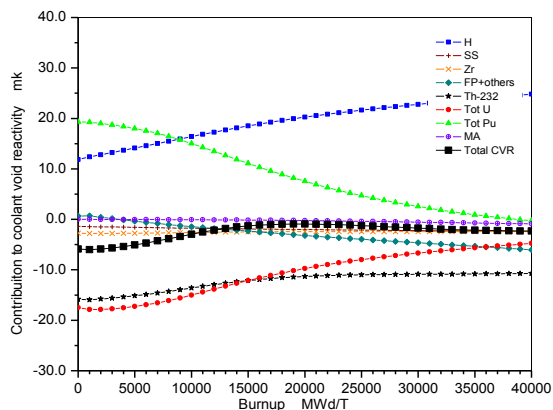
**Fig. 1a** Cross section of AHWR-Ref fuel cluster using  $(\text{Th}, {}^{233}\text{U})\text{MOX}$  and  $(\text{Th}, \text{Pu})\text{MOX}$  fuel



**Fig. 1b** Cross section of AHWR-LEU fuel cluster using Th-LEU MOX fuel

## 2. Method of simulation

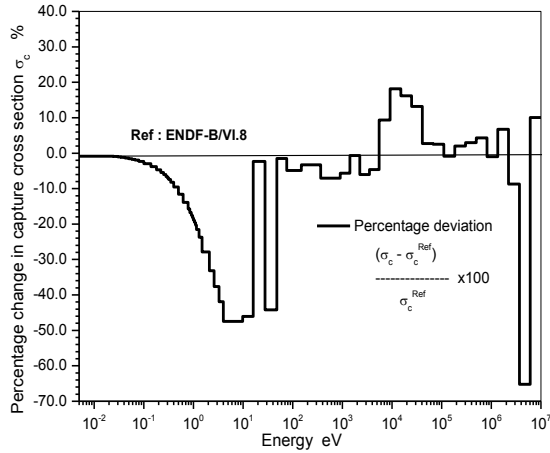
An efficient calculational tool has been developed based on an integral parameter approach developed by Ganda and Greenspan [4], where the reaction rates of all individual isotopes in a lattice is normalized to one fission neutron absorbed. This quantity is a direct measure of the contribution to the feedback coefficient like CVR. The net CVR of the lattice has been quantified through these individual components. The energy wise dependence is also quantified. This method has been extended to find the contributions from the individual energy groups too. The basic lattice calculations have been performed by WIMSD code [5]. The heterogeneous infinite lattice cell calculation followed by a homogeneous leakage calculation. The transport solution has been performed in a 69/172 energy multi-group simulation. The lattice heterogeneities have been modeled in a 2D  $P_{ij}$  collision probability approach. The burnup simulations are done using a critical flux spectrum at operating temperatures. The multi group library used is the ENDF/B-VI.8 WIMS formatted cross section set. Sensitivity studies were performed with other data sets from ENDF/B-VII.0, JENDL-3.2, JEFF-3.1 with both 69 and 172 group libraries [6].



**Fig. 2** Isotopic components of the coolant void reactivity in AHWR-Ref cluster

## 2.1 Sensitivity analysis for the AHWR-Ref fuel cluster

The sensitivities of the individual isotopes to the CVR as a function of burnup is presented in Fig.2. The major contributors to the CVR in the heterogeneous AHWR lattice have been shown to be H,  $^{232}\text{Th}$ ,  $^{233}\text{U}$ ,  $^{239}\text{Pu}$  and  $^{240}\text{Pu}$ . The sign of the CVR depends on the relative contributions of the isotopes in the lattice.  $^{239}\text{Pu}$  has a positive contribution and a negative slope with burnup.  $^{232}\text{Th}$  and  $^{233}\text{U}$  have a negative contribution and a positive slope with burnup. H has a positive contribution to the CVR and a positive slope [7]. More insight was obtained by analyzing the energy wise distribution of the individual components and they are compared for typical in-core burnup in Table 1.



**Fig. 3** Comparison of  $^{232}\text{Th}$  capture cross section in ENDF/B-VI.8 and ENDF/B-VII.0

**Table 1** Energy-wise breakdown of CVR at core average burnup for AHWR-Ref lattice at core average burnup

|   | H      | Th-232  | U-233   | Pu-239 | Pu-240 | Pu-241 | Pu-242 |
|---|--------|---------|---------|--------|--------|--------|--------|
| Fast energy region (10 MeV to 9.118 keV)        | 0.013  | -7.465  | -1.361  | -0.059 | -0.060 | -0.067 | -0.010 |
| Resonance energy region (9.118 keV to 0.625 eV) | 1.300  | -17.078 | -19.426 | -1.006 | -2.559 | -1.445 | -1.054 |
| Thermal energy region (0.625 eV to 0.0 eV)      | 17.197 | 12.402  | 9.868   | 7.278  | 2.768  | 7.249  | 0.036  |

In addition a sensitivity analysis with respect to nuclear cross section data due to the ENDF/B-VII.0 was taken up. The improvements in the thorium fuel cycle and other isotopes data led us to use this data for evaluating the CVR for AHWR which uses thorium. The fission and capture components also were estimated separately. The CVR for the AHWR lattice at EOC was different by 1.7 mk. This difference has been provided in the design margins. From our analysis, it is shown that  $^{232}\text{Th}$  captures and  $^{233}\text{U}$  fissions in ENDF/B-VII.0 are responsible for most of the difference in the calculated CVR in the AHWR-Ref lattice [7]. The difference in the multi-group capture cross section between ENDF/B-VII.0 and ENDF/B-VI.8 is plotted in Fig. 3.

The isotopic breakdown of the CTR shows that the major contributors to the CTR are H and  $^{233}\text{U}$  which have positive components and  $^{232}\text{Th}$ ,  $^{239}\text{Pu}$  and  $^{240}\text{Pu}$  which have negative components. The increase in the CTR with burnup and subsequent decrease is due to the change in  $\eta$  or more precisely  $d\eta/\eta$  [8].

## 2.2 Estimating uncertainties for the AHWR-LEU fuel cluster

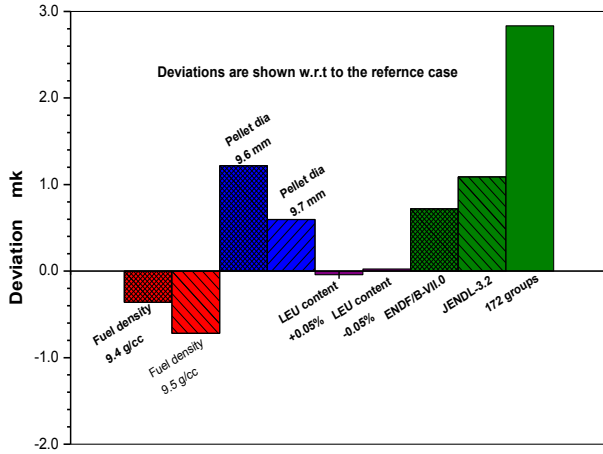
The sensitivity study for the propagation of uncertainties is performed at the lattice level using the WIMSD code. The uncertainties considered in this study are fuel density, fuel pellet dimension and uncertainties in fuel composition. The modelling uncertainties like sensitivity to the use of different nuclear data-sets, use of different methodology, use of more number of energy groups etc. are responsible for estimating the extent of the safety margins. The change in the reactor core state is treated as a perturbation. The parameters compared include the initial infinite multiplication factor and discharge burnup. The core averaged safety parameters calculated are power coefficient, fuel temperature coefficient, coolant void reactivity coefficient, channel temperature coefficient and correspond to an average in-core burnup. The results of this study are presented in Table 2. The range of uncertainty due to each parameter has been quantified. This parameter has been calculated considering all the uncertainties such as increase in the number of energy groups, use of different multi group datasets, manufacturing tolerances including enrichments, Gd content and fuel dimensions.

**Table 2** Estimation of uncertainty ranges in calculated design parameters

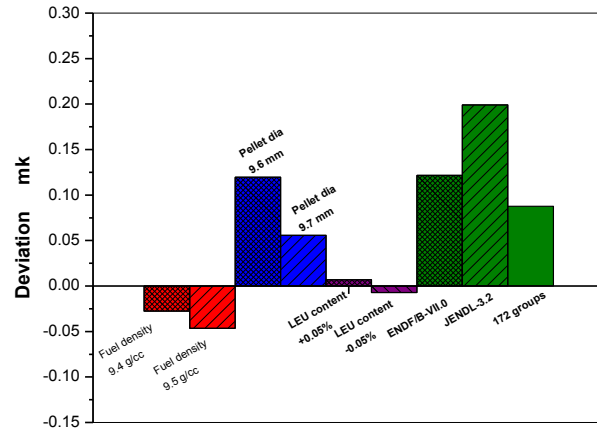
| Calculated parameter   | Range of uncertainty % |
|--|------------------------|
| Moderator temperature reactivity                                 | -25.89 to +2.5         |
| Channel temperature reactivity                                   | -18.23 to +3.82        |
| Fuel temperature reactivity                                      | -3.69 to +0.88         |
| Reactivity due to change in power from hot-standby to 100% power | -6.91 to +4.23         |
| Reactivity due to loss of coolant from operating voids           | -27.11 to +6.89        |



The deviation in the fuel temperature coefficient and coolant void reactivity coefficient due all the different uncertainties are presented in a graphical form in Figures 4a and 4b. The simulation of voids requires a more rigorous multi-scale modelling. It is seen that variations in the multi-group nuclear data set accounts for 10% deviation in the calculated coolant void reactivity. With



**Fig. 4a** Uncertainty in coolant void reactivity in AHWR-LEU fuel cluster



**Fig. 4b** Uncertainty in fuel temperature reactivity in AHWR-LEU fuel cluster

the present uncertainty analysis for the Th-LEU fuelled AHWR we have been able to quantify the safety margins. Fuel temperature coefficient is least sensitive and about 4%. The channel temperature coefficient comprises of coolant temperature coefficient and this sensitive by about 18%. The absorption in hydrogen is an important parameter which is very sensitive to both thermal spectrum (below 4.0 eV) and also relative ratio of fuel-to-coolant atoms. The most sensitive parameter is the coolant void reactivity and it ranges from -27% to 6.9%.

### 3. Summary and Conclusions

Sensitivity studies provide inputs for quantifying the design margins for safety studies. The coolant void reactivity is very sensitive to nuclear data set and the modelling. The multi-group cross section sets show significant deviations for e.g. -50% in thermal regions for  $\sigma_c$  of  $^{232}\text{Th}$ , +60% in  $\sigma_f$  of  $^{239}\text{Pu}$  at ~1000 eV and  $\sigma_f$  of  $^{235}\text{U}$  varies from -15% to +10%. The effect of these cross sections on the safety parameters has been quantified. The uncertainty studies are required to be taken to level higher where covariances can be established. It is important to analyse the uncertainties in a more rigorous manner. It is recommended that the evaluated and multi-group data for the isotopes of the thorium cycle are required to converge to more high accuracy.

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# Statistical model Calculations using TALYS code for the study of neutron and $\gamma$ induced reactions

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## Abstract

TALYS-1.4 and EMPIRE are nuclear reaction codes and are essential for advanced modeling of nuclear reactions such as direct, compound, pre-equilibrium, fission reactions and to estimate their cross sections. These codes incorporate various nuclear models for the optical model, level densities, direct reactions, compound reactions, pre-equilibrium reactions, fission. They incorporate a large nuclear structure database, phenomenological expressions for parameters of these models depending on projectile energy and target mass. These codes simulate nuclear reactions involving projectiles such as neutrons, photons, protons, deuterons, tritons, <sup>3</sup>He and alpha-particles. TALYS code is useful for projectiles in the 1 keV - 200 MeV energy range and for target nuclides of mass  $A \geq 12$ . Cross sections can be calculated using default values of input parameters for a gross estimation, through a simplified input file listing only the projectile, target element symbols, their mass numbers, and incident projectile energy. However, detailed calculations can be done by using keywords in the input file for selecting several options for various physical models and their parameters in order to fit the experimental data. The keywords enable to select various quantities in output such as total and partial cross sections, energy spectra, angular distributions, double differential spectra, excitation functions for residual nuclide production, including isomeric cross sections.

Nuclear data of neutron and gamma induced reactions are of academic interest and also essential for many important applications. Especially, precision measurements of cross section data are needed for validating the nuclear reaction model codes and are of vital need for applications in reactor, dosimetry, medical and other fields. In this talk we discuss TALYS calculations for nuclear data applications. We present calculations for a few cases of neutron and gamma induced reactions and some of the experimental measurements performed by our group, such as <sup>96</sup>Zr( $\gamma$ , n)<sup>95</sup>Zr, <sup>93</sup>Nb( $\gamma$ , n)<sup>92</sup>Nb, <sup>100</sup>Mo( $\gamma$ , n)<sup>99</sup>Mo, <sup>238</sup>U( $\gamma$ , n)<sup>237</sup>U at  $E_\gamma$  (bremsstrahlung) = 10 and 12.5 MeV and the neutron capture reactions <sup>238</sup>U(n, $\gamma$ )<sup>239</sup>U, <sup>232</sup>Th(n, $\gamma$ )<sup>233</sup>Th at various neutron energies. The experimentally determined cross-sections are compared with the TALYS calculations and also the evaluated data of ENDF-VIIB and JENDL-4.0 libraries, these details will be discussed.

Our collaborators covering many experiments and calculations, include Drs. Paresh Prajapati, Rita Crasta, Vikas Mullik, B.S. Shiva Shankar, K.S. Jagadeesan, S.V. Thackery, S. Ganesan and A. Goswami



# Study of elastic and inelastic neutron cross sections using time of flight technique

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## Abstract

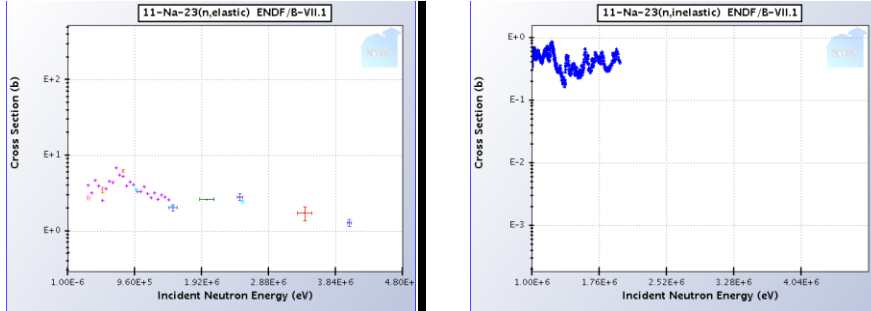
Elastic neutron scattering angular distributions from <sup>23</sup>Na has been measured for incident neutron energies between 1.0 and 4.0 MeV at the University of Kentucky using neutron time-of-flight technique. The cross sections obtained are important for applications in nuclear reactor development and other areas, and they are in a energy region in which existing data are very sparse. These cross sections were obtained by normalizing Na angular distributions to the well-known *np* cross sections.

## 1. Introduction

Neutron cross-section standards are important in the measurement and evaluation of all other neutron reaction cross-sections. Not many cross-sections can be defined as absolute - most cross-sections are measured relative to the cross-section standards for normalization to absolute values. Previous evaluations of the neutron cross-section standards were completed in 1987 and disseminated as both NEANDC/INDC and ENDF/B standards. R-matrix model fits for the light elements and non-model least-squares fits for the heavy elements were the basis of the combined fits for all of the data. Some important reactions and constants are not standards, but assist greatly in the determination of the standard cross-sections and reduce their uncertainties. High precision neutron scattering data has become increasingly important in such diverse areas as the development of nuclear reactors and accelerator systems, astrophysics and space system design, radiation therapy and isotope production, and for shielding considerations [1].

For nuclear reactors alone, well determined cross sections are needed for computer modeling calculations for reactor design and safety, for heat transfer properties and shielding, and Na cross sections in particular are needed for proliferation resistant thorium fast reactors. Elastic and inelastic neutron cross sections are especially important for use in transport codes and energy loss calculations, and for the inelastic neutron channel, the resulting  $\gamma$ -rays can lead to heating of reactor materials. Knowledge of the (*n,n*) and (*n,n'*) channels are often important for deducing the (*n,p*) and (*n, $\alpha$* ) cross sections which are also important to reactor design. For <sup>23</sup>Na, the existing inelastic neutron scattering cross sections are known to approximately 30% in the 2-6 MeV region and the desired uncertainties are of the order of 12-13% [2].

The focus of the research presented here is elastic neutron differential scattering cross sections measured between 1 and 4 MeV on  $^{23}\text{Na}$  as the existing  $^{23}\text{Na}$  neutron experimental differential cross section measurements don't fill the energy region (specially 1-10 MeV) completely as shown in Fig. 1.



**Fig. 1.** Existing  $^{23}\text{Na}$  elastic and inelastic experimental neutron cross sections are shown in the left and right panels respectively.

## 2. Experimental Methods and Analysis

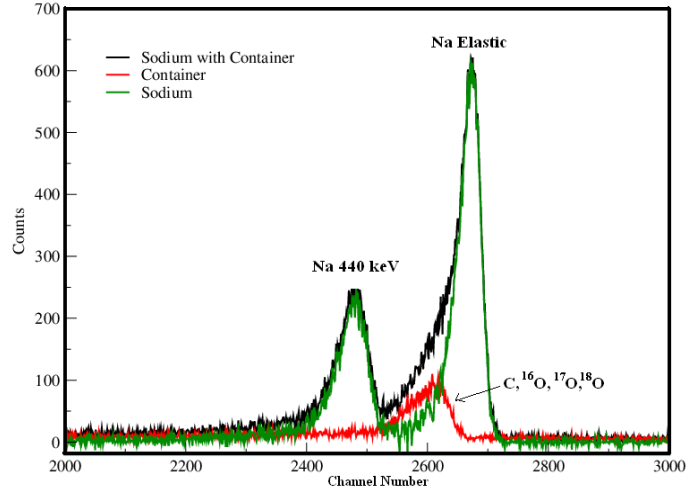
### Neutron TOF technique

Measurements were performed using the 7 MV CN Van de Graaff located at the University of Kentucky Nuclear Structure Laboratory. The rf discharge ion source is coupled to a 1.875 MHz pulsing and  $\sim 1$  ns bunching system to produce a pulsed proton beam necessary for neutron time-of-flight (TOF) measurements. The beam was used as stop and signal from detector was used as a start for different energies. The  $^3\text{H}(p,n)$  reaction was used to produce a nearly monoenergetic neutron beam in the forward direction emerging from the gas cell of length 3.0 cm and at half atmosphere of gas pressure. The distance of the scattering sample from the centre of gass cell was 7.0 cm. The shielded neutron detector was mounted at distances of 2.2 to 3.8 m from the scattering samples. The FWHM energy resolution was approximately 80 keV. Neutrons were detected in a 11.5 cm diameter x 2.5 cm thick C6D6 liquid scintillator, which permitted pulse-shape rejection of  $\gamma$ -ray events. Pulse shape discrimination was used for both the monitor and main scintillators to eliminate  $\gamma$ -ray induced events. This detector was properly collimated and shielded and located in a detector carriage that can be rotated though angles  $0^0$  to  $150^0$ . Neutron production from the gas cell was monitored with a small NE213 scintillator; yields from this detector were used to normalize all angular distributions.

Measurements were made at  $10^0$  intervals from  $30^0$  to  $150^0$ . The scattering yields were extracted from the TOF spectra using a peak fitting program that models each peak with two Gaussians and an exponential tail and incorporates kinematic constraints into peak positions and peak shapes. Yields were corrected for neutron attenuation and multiple scattering in the sample using Monte Carlo methods. Absolute normalization of the cross sections were determined by measuring n-p scattering [3-4].

Cylindrical sodium sample with a diameter of 2.60 cm, a height of 2.67 cm and a mass of 16.26 g was used for the neutron scattering measurements. The metallic sodium sample was heat sealed inside a thin polypropylene container to reduce oxidation and for safety considerations. Neutron scattering from samples of a blank Polypropylene container, Carbon, and Polyethylene were measured to assist with background subtractions, cross section normalization, and physical data corrections. Example TOF spectra from the main detector of the  $^{23}\text{Na}$  sample plus

container, the Polypropylene container, and resulting container subtracted  $^{23}\text{Na}$  spectrum is shown in Fig. 2. Peak yields were extracted from the TOF spectra using a program that positions peaks to maintain energy separations fixed by kinematics, and then fits them with an asymmetric form that accounts for the time and energy spreads in the experiments.



**Fig. 2.** Neutron TOF spectrum showing the resulting Na spectrum once the container spectrum is subtracted from the Na with container spectrum.

### Main $\text{C}_6\text{D}_6$ detector efficiency

$$eff(E_n) = \frac{Y_{3H(p,n)}(\theta)}{Y_{FM} \frac{d\sigma}{d\Omega_{3H(p,n)}}(\theta)} \quad (1)$$

where  $eff(E_n)$  is the relative detector efficiency at neutron energy  $E_n$  at an angle  $\theta$ ,  $Y_{3H(p,n)}$  is the main detector yield,  $Y_{FM}$  is the monitor yield, and  $d\sigma/d\Omega$  is the source reaction cross section at an angle  $\theta$ . The relative deviation  $d\sigma/d\Omega_{3H}(\theta)$  is important for Na cross sections determination and is estimated to be 3% [4].

### Cross Section estimation

The uncorrected and un-normalized relative angular distribution of the scattered neutrons  $W(\theta)$  is then determined by the following expression:

$$W_{Na,lab}(\theta) = \frac{Y_{Na}(\theta)}{Y_{FM} eff(E_{n\ for\ Na}) N_{Na}} \quad (2)$$

Where,  $Y_{Na}$  is the yield of the peak of interest,  $Y_{FM}$  is the forward monitor yield,  $eff(E_{n\ for\ Na})$  is the efficiency of the main  $\text{C}_6\text{D}_6$  detector at the energy of the scattered neutron, and  $N_{Na}$  is the number of Na nucleus in the scattering sample. Similarly,  $W_{H,lab}$ , is deduced for  $np$  scattering from the polyethylene sample, which is used as the neutron cross section standard since the  $np$  total cross sections are known to about 3% in the 2-4 MeV region. The neutron scattering

differential cross sections for Na are then determined by forming the ratio of  $W_{Na,lab}$  and  $W_{H,lab}$  and scaling to the evaluated  $np$  cross sections [4],

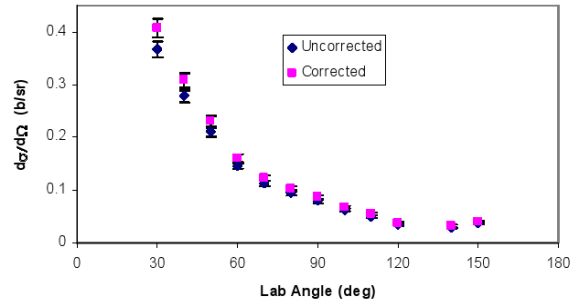
$$\frac{d\sigma}{d\Omega_{Na}}(\theta) = \left( \frac{W_{Na,lab}(\theta)}{W_{H,lab}(\theta)} \right) \left( \frac{\sigma_{Tot}^{n-H}}{4\pi} 4\cos(\theta_L) \right) \quad (3)$$

The last factors in Eq. (3) are the angle-integrated elastic cross section for  ${}^1\text{H}(n,n)$ , which is isotropic in the center-of-mass frame, and transformed to the laboratory frame.

### Attenuation and multiple scattering corrections

Cross sections determined via Eq. (3) is corrected for neutron attenuation and neutron multiple scattering in the sample by Monte Carlo code MULCAT [5]. Starting from experimental data (values from Eq. (3)), the perturbed angular distribution is calculated and the resulting change in the estimated angular yield used to back-calculate the unperturbed cross section. The procedure is iterated until the process stabilizes. A typical example of  ${}^{23}\text{Na}(n,n)$  angular distribution at  $E_n = 3.4$  MeV, before and after corrections, is shown in Fig 3. Corrections range from 10-16% of the cross section.

The MULCAT code performs Monte Carlo analyses of neutron interaction in a single nuclide cylinder, irradiated radially by neutrons produced by a charged particle accelerator. Nuclear data which must be specified as input consist of the energy dependent total cross section and the angular variation at single energy of either the radiative capture cross section or the cross sections for scattering elastically and inelastically to a single discrete level. The calculation uses Legendre polynomial expansions to represent the angular dependence. The iteration consists of systematically changing the angular cross section tabulation, fitting with a new set of Legendre coefficients, and running another Monte Carlo calculation. Other data, such as the inelastic energy level in the sample, sample size and density and properties of the neutron producing target is also needed to be specified for this calculation.



**Fig. 3.** Corrected and un-corrected elastic neutron scattering differential elastic cross sections for Na at  $E_n = 3.4$  MeV.

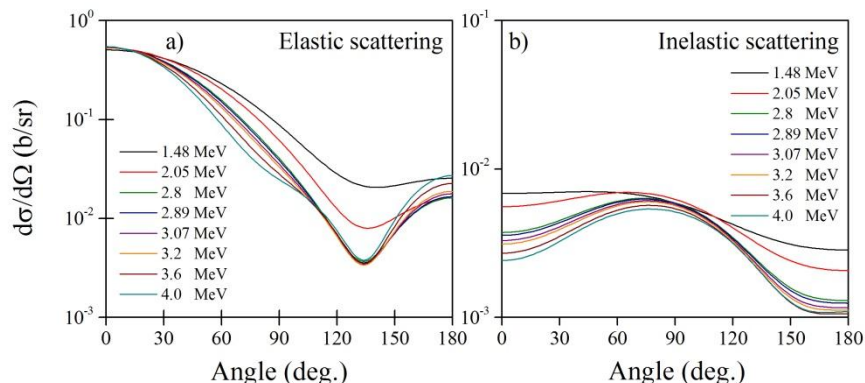
### Talys calculations

For the elastic and inelastic cross section measured for the different incident energies we performed theoretical simulations using the Talys-1.4 code [6]. This code treats  $n$ ,  $\gamma$ ,  $p$ ,  $d$ ,  $t$ ,  $h$ , and  $\alpha$  as projectiles and ejectiles and is applicable for a wide range of energy from 1 keV to 200 MeV, as well as target mass numbers from 12 to 339. The optical model with a default set of parameters is used for the present calculations. In addition to the Talys calculations, we have



also retrieved available data for the studied reaction from Evaluated Nuclear Data File (ENDF/B-VII.1) [7].

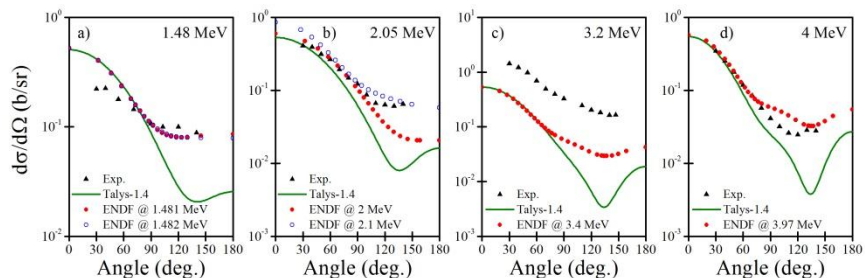
We present the results of elastic (panel a) and inelastic scattering (panel b) of neutron differential cross sections at different incident energies calculated using the Talys code in Fig.4.



**Fig. 4** Talys-1.4 results of (a) elastic and (b) inelastic neutron scattering differential cross sections at different incident energies.

### 3. Results and Conclusions

In Fig. 5, the measured elastic scattering differential cross sections is compared with the Talys-1.4 calculation and available ENDF/B-VII.1 data. For the incident energies of this experiment in ENDF database there are no data. However the data available to the nearest energies of 1.481, 1.482, 2, 2.1, 3.4 and 3.97 MeV are considered here for comparison.



**Fig. 5.** Measured (solid triangles) elastic scattering differential cross sections are compared with the Talys-1.4 results (solid line) and ENDF/B-VII.1 data (solid and/or open circles).

In Fig. 5(a), the angular distributions of measured values (solid triangles) and Talys-1.4 results (solid line) are presented for 1.48 MeV however the ENDF data is for 1.481 (solid circles) and 1.482 MeV (open circles). The measured values are in good agreement with the Talys-1.4 results as well as with the ENDF dataset. Similar agreement is seen for 2.05 MeV and 4 MeV also as shown in panel (b) and (d) respectively. In panel (c), it is seen that the measured values are found to differ with Talys-1.4 and ENDF (3.4 MeV) results atleast by an order of magnitude. A critical analysis is needed at 3.2 MeV to explore the possibility of resonance as the variation of cross section at this energy differs greatly w.r.t. rest energies studied. The present data also has its importance due to the reduction in the error associated with it. The total error associated with

the present data is 11%, in which main contribution comes from the hydrogen cross section which is used for normalization purpose for Sodium cross section and contributes almost 3% as available till date. This was one of the main goal of this measurement to reduce the magnitude of error associated with the measured elastic cross section of  $^{23}\text{Na}$ .

Model calculations in case of  $^{23}\text{Na}$  are very critical due to the overlapping resonance structure and coupling to direct processes. Also as  $^{23}\text{Na}$  is strongly deformed nucleus, so it is also required to include this effect in the calculations. The contribution above than angle  $90^\circ$ , in all energy regions as shown in Fig. 4, is showing deviation with model calculations, indicating necessary corrections in model calculations.

These measured data along with angular distributions at various incident neutron energies with high accuracy level and reduced errors will provide important information for nuclear reactor design and safety considerations.

### Acknowledgements

This work was supported in part by the Department of Energy through the NEUP program and by the Cowan Physics Fund at the University of Dallas. We acknowledge with appreciation the many contributions of accelerator engineer Harvey Baber. We also acknowledge the grant received through UGC Networking scheme, from UGC, Govt of India. The author AK also acknowledge to Department of Science and Technology (DST), Govt. of India for financial support through SR/FTP/PS-068/2010.

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## Inclusion of nuclear data of Uzbekistan authors to the NRDC during 2013-2014 years

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### Abstract

Uzbekistan's first step to join to IAEA was made in 1995 by International Nuclear Information System (INIS) and after that Uzbekistan became a member of INIS. The group organized in INP AS RUz (Tashkent) was called "Uzbekistan national INIS center". Liaison officer of this group is Dr. Makhtuba Kadirova and deputy liaison officer is Dr. M. Salimov. Uzbekistan national INIS center provides a comprehensive information reference service for the specialists in nuclear science and technology, including scientific publication in Uzbekistan (<http://www.inp.uz/node/306>).

Since the time INP was founded, a number of nuclear experiments have been made and the nuclear reaction data, induced by charged-particle reactions obtained. During the years, the obtained experimental data have been published in domestic and international journals. In order to save those data for future needs, they needed to be added to some database. Those works have been performed by members of Nuclear Reaction Data Centres (NRDC) of the world. At the time of those works were carrying out, some problems appeared related to the correspondence with domestic authors of the articles. To solve those problems, the members of the centers decided to contact with scientists of domestic institutes and request them to contact with authors, collect experimental data from them.

The next step toward to join to IAEA was participation in the 4<sup>th</sup> Asian Nuclear Reaction Database Development Workshop (ANRDDW) which was held in Almaty, Kazakhstan (23-25 October, 2013). Researchers from Uzbekistan participated in this workshop and set active collaboration with members of CA-NRDB (Central Asian Nuclear Reaction Data Base). The new EXFOR entries prepared by the Kazakhstan and Uzbekistan compile group are shown in Table 1.

The nuclear scientists of Uzbekistan suggested including the data on astrophysical S-factor obtained for the nuclear astrophysical reactions at low energies in the future.

**Table 1.** New EXFOR entries prepared by the Kazakhstan+Uzbekistan group

| Entry # | First author   | Article                | Lab.    | Status      |
|---------|----------------|------------------------|---------|-------------|
| 31737   | S.R.Palvanov   | J,PAN,77,35,2014       | 4UZ UZB | PRELIM.3163 |
| 31738   | Yu.N.Koblik    | J,PAN,77,39,2014       | 4UZ UZB | PRELIM.3163 |
| 31741   | G.A.Abdullaeva | J,NESE,3,72,2013       | 4UZ UZB | Compiled.   |
| D0712   | N.Burtebayev   | J,NP/A,909,20,2013     | 4KASKAZ | TRANS.D093  |
| D0726   | A.Vasidov      | J,IZU,1981,(3),93,1981 | 4UZ UZB | in EXFOR    |
| G0041   | S.R.Palvanov   | J,PAN,77,35,2014       | 4UZ UZB | PRELIM.G029 |



# Experimental and theoretical investigation of scattering of alpha particles from $^{13}\text{C}$ nuclei

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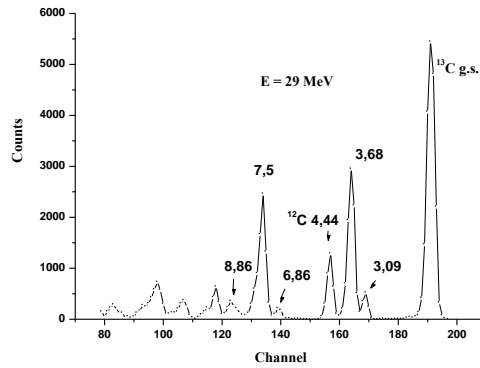
Email: nburtebayev@yandex.ru

$^{13}\text{C}$  is a good example of a “normal” nucleus well described by the shell model. Its level scheme is reliably determined up to the excitation energies  $\sim 10$  MeV (see e.g. [1]). However, some new approaches such as the hypothesis [2] of the  $\alpha$ -particle condensation suggest that cluster states with an enhanced radius can appear. The famous Hoyle state ( $0_2^+$ ,  $E^* = 7.65$  MeV) in  $^{12}\text{C}$  was considered as the most probable candidate for having such structure. It was also expected [3] that the analogues of the Hoyle state would reveal themselves in some neighboring nuclei, e.g., the  $1/2^-$  ( $E^* = 8.86$  MeV) state in  $^{13}\text{C}$ . The analysis [4] of the  $^{13}\text{C} + \alpha$  scattering data measured at  $E(\alpha) = 388$  MeV [5] really demonstrated a considerable enhancement of the radius of this particular state. However, the method of extracting the radii used in Ref. [4] (the Modified diffraction model, MDM [6]) may not quite adequate at high energies ( $\geq 100$  MeV) when nuclei are too transparent. Besides, the existence in  $^{13}\text{C}$  of some states with enhanced dimensions but of different structure was discussed as well. Thus, a neutron halo was identified in the first excited state 3.09 MeV ( $1/2^+$ ) by two independent and complementary methods [7,8]. Consequently, new measurements especially at lower energies are highly desirable.

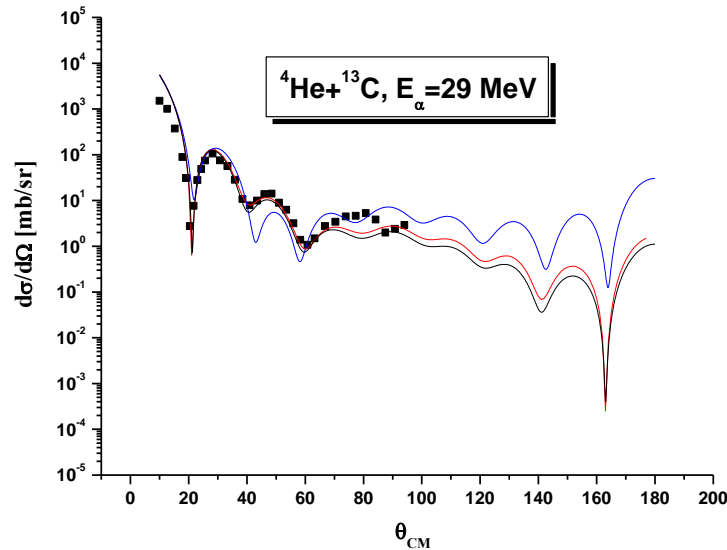
In this paper we studied the elastic  $\alpha + ^{13}\text{C}$  scattering at  $E(\alpha) = 29$  MeV using optical model. Differential cross sections of elastic and inelastic  $\alpha + ^{13}\text{C}$  scattering were analyzed within the framework.

The experiment was carried out at the isochronous cyclotron U-150M in Institute of Nuclear Physics (Almaty, Kazakhstan) at energy  $E_\alpha = 29$  MeV. Sets of  $\Delta E$ - $E$  telescopes for detection of the alpha-particles were used in the experiments. A self-supporting  $^{13}\text{C}$  target ( $0.4$  mg/cm<sup>2</sup>) with the 84% enrichment was used (Fig.1).

Angular distributions of  $\alpha$ -particles elastically and inelastically scattered from  $^{13}\text{C}$  nuclei at energy  $E_{\text{lab}} = 29$  MeV: ground state,  $1/2^-$  (8.86 MeV) and  $1/2^+$  (3.09 MeV) were measured in the angular range  $\theta_{\text{lab}} = 10^\circ - 80^\circ$  with increments of  $10^\circ - 20^\circ$ . Energy resolution of the detector at small angles is  $\sim 290$  keV, and at large angles is  $\sim 350$  keV. The experimental data at this energy showed a well-developed diffraction scattering pattern.



**Figure 1** - A sample spectrum ( $\theta = 32^\circ$ ) for the  $\alpha + {}^{13}\text{C}$  scattering at  $E(\alpha) = 29$  MeV.



**Figure 2** - Differential cross sections for elastic scattering of alpha particles on  ${}^{13}\text{C}$  at energies  $E = 29$  MeV. Solid red and black lines are the calculations within the framework of optical model, solid blue line is calculations using double folding model.

In addition to our experimental data at  $E_{\text{lab}} = 29$  MeV,  $\alpha + {}^{13}\text{C}$  elastic scattering was analyzed at different energies from literature; 65 MeV [9], 54.1 and 48.7 MeV [10], 35 MeV [11] and 26.6 MeV [12]. The theoretical predictions were performed using both empirical Woods-Saxon (2 sets of potentials) and double folding optical model potentials (Fig.2). The comparison between the experimental data and the theoretical predictions is fairly good overall the whole angular range. We managed from obtaining physically reasonable parameters for the interaction potentials.

Analysis of data on inelastic scattering (excited states of 3.09 and 8.86 MeV) is planned to done with the use of the obtained parameters of optical potentials in the following papers. This work is supported by grant of MES of the Republic of Kazakhstan # 0601 GF.

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## Joint activities with IAEA on uploading of scientific papers from Kazakhstan and Uzbekistan into the EXFOR database

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More than a year passed since Kazakhstan joined the international network of nuclear reactions data centers (NRDC). A Central Asian centre for nuclear reactions (CANRDB, Central Asia Nuclear Reaction Database) has been established at al-Farabi Kazakh National University, and a group of experts there is actively working on expansion of the database, further development of the specialized software, and fostering partnership with international nuclear physicists. There are also on-going activities aimed on training, searching of the published nuclear data obtained earlier by scientists from Central Asia to incorporate their results in the database. The main objective of CA-NRDB is the development and formation in Kazakhstan of open and user-friendly database on nuclear reactions with further incorporation of this database in the international network of nuclear databases under the International Atomic Energy Agency (IAEA). We note that such a database is created in the entire Central Asian region for the first time.

The CANRDB team started its work from compilation of research article in November 2013 and since then eleven articles were downloaded into the EXFOR database. All the articles were published in 2013-2014 and relate to experiments on nuclear reactions; the authors are from Kazakhstan and Uzbekistan. Below we present a list of the articles incorporated in the database by the CANRDB team.

| <b>Entry #</b> | <b>First author</b> | <b>Article</b>         | <b>Lab.</b> | <b>Status</b> |
|----------------|---------------------|------------------------|-------------|---------------|
| 31737          | S.R.Palvanov        | J,PAN,77,35,2014       | 4UZ UZB     | PRELIM.3163   |
| 31738          | Yu.N.Koblik         | J,PAN,77,39,2014       | 4UZ UZB     | PRELIM.3163   |
| 31741          | G.A.Abdullaeva      | J,NESE,3,72,2013       | 4UZ UZB     | Compiled.     |
| D0711          | Sh.Hamada           | J,PR/C,87,024311,2013  | 4KASKAZ     | in EXFOR      |
| D0712          | N.Burtebayev        | J,NP/A,909,20,2013     | 4KASKAZ     | TRANS.D093    |
| D0723          | V.T.Gkadun          | J,IZK,1980,(4),82,1980 | 4KASKAZ     | in EXFOR      |
| D0725          | I.N.Khaustov        | J,IZK,1990,(2),3,1990  | 4KASKAZ     | in EXFOR      |
| D0726          | A.Vasidov           | J,IZU,1981,(3),93,1981 | 4UZ UZB     | in EXFOR      |
| D0727          | S.Muhammedov        | J,IZU,1985,(5),81,1985 | 4UZ UZB     | in EXFOR      |
| D0728          | V.V. Dyachkov       | IET,56,521,2013        | 4KASKAZ     | in EXFOR      |
| G0041          | S.R.Palvanov        | J,PAN,77,35,2014       | 4UZ TSK     | PRELIM.G029   |

As one can see from the table above, six articles have already been posted and are available at the related IAEA web-site. Such uploading process takes from one to four months. Several articles published in previous years in local journals are now processed for incorporation in the EXFOR database.

CANRDB team has participated in the technical meeting of NRDC under the auspices of the IAEA held in Smolenice (Slovakia) on May 6-9, 2014. Participation in this meeting was of particular importance for the CANRDB team and allowed to present our activities among experienced colleagues from other centers with nuclear data bases. This was the first step

towards joining the NRDC network. Important issues related to preparation of some valuable articles published in local journals in the past, digitization of data and data incorporation to the EXFOR database were discussed at the meeting. We plan to continue such work together with Dr. Naohiko Otsuka during his visit to Almaty. There will also be arranged a training for the CANRDB team members on how to upload articles into the EXFOR database.

Establishing the database which got the name *Central Asia Nuclear Reactions Database* (CA-NRDB) it was important to identify its distinguishing features. Location of Kazakhstan and local activities back in USSR times have predetermined one of such features: Kazakhstan was a land of many testing sites including Semipalatinsk Nuclear Test Site. Decontamination, rehabilitation and radioactive waste issues there became the priority in the state agenda.

Specialists of al-Farabi Kazakh National University and National Nuclear center, nuclear physicists, radiochemists, biologists and environmentalists took active part in numerous research and activities there. Research outcomes are published by IAEA and a variety of international organizations. One of the divisions of the CA-NRDB will be devoted to radio-ecology issues.

Another distinguishing feature is stipulated by multiethnicity of Kazakhstan: the CA-NRDB will be created in three languages – Kazakh, English and Russian. The database is designed to support educational activities, scientific research and technology development. An extensive electronic library will also be created to incorporate textbooks, presentations, lectures, scientific papers, etc. The English-language version of the CA-NRDB is designed for incorporation into the international network of the nuclear data bases under IAEA. The core of the CA-NRDB would be the development of the database for nuclear physics and nuclear astrophysics.

Currently, the small team of the CA-NRDB consists of two full-time employee and 4 part-time assistants (graduate students majoring in nuclear physics). Since the turnover among the part-time junior employees is quite high, it is a bit challenging to assure stable operation and efficient qualification improvement of the staff; we work towards mitigation of such factors.

In spite of some difficulties, the CA-NRDB team has got considerable progress. The own website has been designed, vast data have been collected and proceeded, particularly for the educational part of the resource, we organized publishing of own pre-prints. A pilot version of the CA-NRDB has been launched at the University's web portal, the structure and main components have been shaped.

Efficiency and usefulness of a nuclear reactions database is directly related to the organization of a high-speed and multi-channel search engine. This important technical and optimization task implies development of new information methods and our team is doing that in cooperation with foreign scientists and specialists.

The research objects are published papers and works on nuclear physics and nuclear reactions, experiments at particle accelerators, descriptions of experimental facilities and experiments, computer searching engines, calculational and analytical software packages.

A project proposal for the state funding application for 2015 - 2017 has been prepared in cooperation with foreign scientists and specialists. Such competition for funding through the Ministry of Education and Science has been announced in Kazakhstan for the first time. We look forward to our fruitful cooperation with the CA-NRDB team in Hokkaido University.

Our center maintains friendly cooperative relations with many data centers in different countries. We are planning to officially incorporate the CANRDB into the International Network of NRDC. In this regard, the team is planning to upload all the noticeable domestic papers into the EXFOR by the next technical meeting to be held in Vienna on April 2015.

# Study of interaction mechanisms of alpha particles with $^{11}\text{B}$ nuclei at low energies

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Experimental angular distributions of alpha particles scattered from  $^{11}\text{B}$  nuclei were measured with extracted beam of isochronous cyclotron U-150M of Institute of nuclear physics (INP, Almaty, Kazakhstan) at energies  $E=29, 40, 50$  MeV.

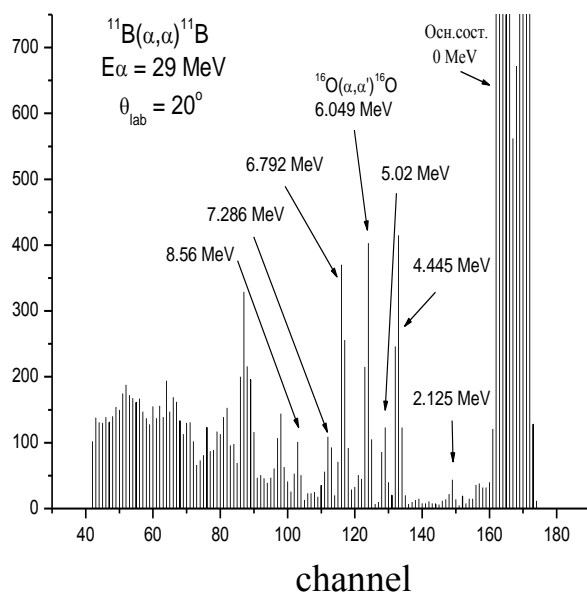
Thin metallic foils made of boron-11 isotope were used as targets. Target thickness was determined at INP UKP-2-1 linear accelerator. For this purpose  $^{27}\text{Al}(p,\gamma)^{28}\text{S}$  reaction yield curves in the region of  $E_p=992$  keV resonance [1] were measured using aluminum foil and sputtered target. The shift of the resonance in  $^{27}\text{Al}(p,\gamma)^{28}\text{S}$  reaction, caused by the protons energy loss as passing through boron film, was 62,2 keV that corresponds to target thickness of  $320 \mu\text{g}/\text{cm}^2$  (Figure 1). This method allowed determining the target thickness with an uncertainty not worse than 5%.

Nuclear reactions products were registered and identified using standard  $\Delta E-E$  method realized on the basis of PC/AT personal computer. ORTEC silicon semiconductor detectors were employed as counters. Angular distributions of differential cross sections of registered alpha particles were measured in angular range of  $\theta_{\text{lab}}=10^0-170^0$  with an increment  $1^0-2^0$  for the forward hemisphere and  $3^0-5^0$  for backward hemisphere. The registration system solid angle was  $4,22 \cdot 10^{-5}$  steradian. The detector energy resolution at small angles was within (290) keV, at large angles it was within (350) keV, and was mainly due to cyclotron beam energy spread and target thickness.

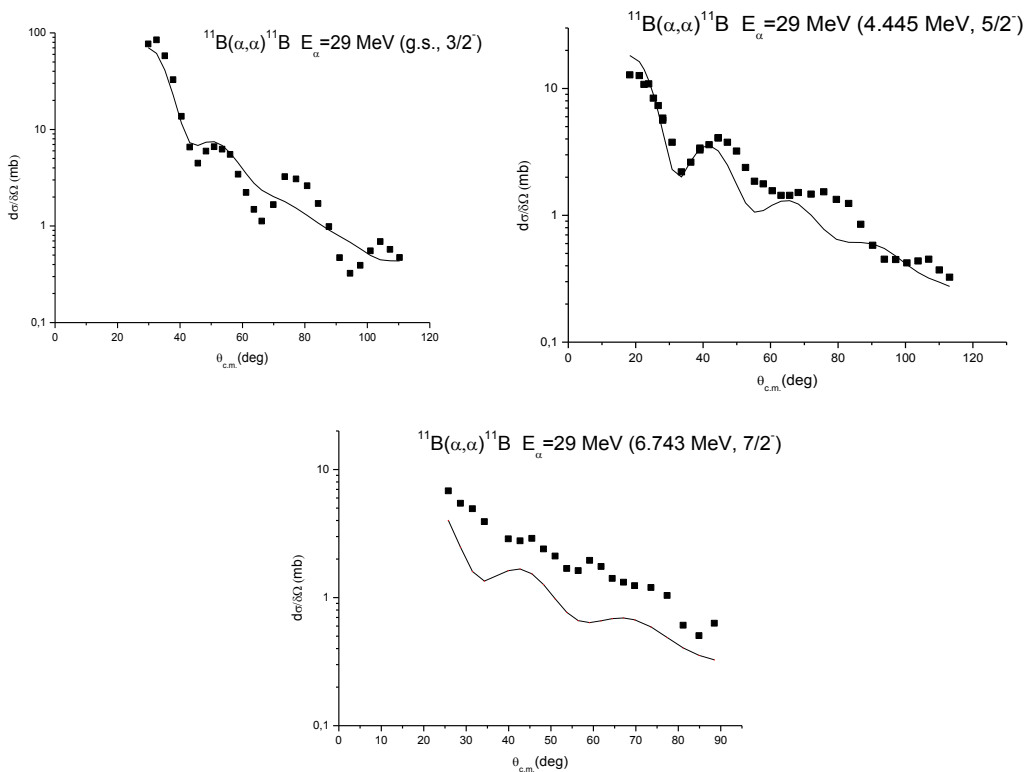
The experimental data on alpha particles elastic and inelastic scattering leading to 4,445 MeV ( $5/2^-$ ) and 6,743 MeV ( $7/2^-$ ) states were analyzed within the framework of coupled channels method (Figure 2). The analysis was made with the employment of experimental data on  $\alpha+^{11}\text{B}$  scattering at  $E_\alpha=40$  and 50 MeV [2] in order to obtain reliable optical potential parameters. We obtained  $\beta_2$  and  $\beta_4$  deformation parameters which are in good agreement with literature data.

Poor agreement of theoretical calculations for 6,743 MeV ( $7/2^-$ ) is probably due to necessity of more accurate determination of higher order deformation parameters

The results of current work will be used in future works for describing further excitation levels e.g. 8.56 MeV ( $3/2^-$ ) which according to predictions may have enhanced radius.



**Figure 1** – Energy spectrum of alpha particles scattered from  $^{11}\text{B}$  measured at INP U-150M accelerator at  $E_\alpha=29$  MeV.



**Figure 2** – Differential cross sections of elastic and inelastic scattering  $\alpha+^{11}\text{B}$  at  $E_\alpha=29$  MeV leading to 4,445 MeV ( $5/2^-$ ) and 6,743 MeV ( $7/2^-$ ) states (points) and theoretical calculations using coupled channels methods (curves).

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# Introduction of the digitization software GDgraph

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## 1. Introduction

The evaluators and experimenters always desire to have full and latest experimental data sets. However, the data are often published as figures without any numerical values for some publications or journals. Furthermore, the quality of figures is not always good enough, especially for some figures scanned from the hard copy of old publications. On the other hand, the researchers would like to retrieve the data directly from EXFOR database. Digitization of figures is only one method to obtain the numerical data and correlative uncertainty, when there are only figures available from publications. Therefore we need a digitization tool to fit the requirements from evaluation, measurement and EXFOR compilation in CNDC.

Before 2000, there have no common software to digitize experimental and evaluated data. And the quality of digitization results can not fit the requirements of evaluation and measurement using the traditional coordinate paper or rule. The end of twenty century, the PC was developed so quickly that to develop a software for digitization purpose become possible. Since 1997, CNDC devotes to design and develop a software for digitization. Four years later, the first version of digitization software GDGraph was developed using VC++ and released in CNDC. Although, the functions of the 1<sup>st</sup> version of GDGraph is fit the basic requirements of digitization only, in which can digitize one group data excluding data error, BMP image format only, and it can not randomly delete digitizing data points. However, it obtains higher quality digitizing results and efficiency than the traditional method.

Five years late, we collected some feedback information and suggestion on update of this software. The 2<sup>nd</sup> version of GDGraph software was released at 2006, in which the whole software was re-written using Perl to obtain more comfortable conditions for programming and updating. The version 3.0, 4.0 and 5.0 of GDGraph is released at 2011, 2012 and 2013, respectively. And the manual of GDgraph-5.0 is released in English at 2013.

## 2. Main Feature of GDGraph5.0

Main features of GDGraph5.0 is listed below:

- (1) Operating system: WindowsXP or the higher version of Windows.
- (2) Intuitive and light GUI: Provide Chinese and English version GUI.
- (3) Supports image format: such as PNG, GIF, BMP and JPEG etc.
- (4) The image can be automatically fit to the GUI window, and zoom-in or zoom-out manually together with the digitizing X, Y axis.
- (5) Allow to rotate the image and set a rotation angle with degree unit.
- (6) The maximum digitizing data group number is three, and the color, size and shape of each group can be defined by user using "Settings" function.
- (7) Randomly add the digitizing point and move it by mouse or cursor keys.
- (8) Output data can be saved as a data file or at clipboard.
- (9) Import data function is enable to reuse the former digitizing data or compare with other data group easily.
- (10) X, Y axes: Select or set a unit for X, Y data by user is available. Allow to digitize X, Y error with symmetry or asymmetry mode and move it using mouse or cursor keys, and set a

fix value with relative (%) or absolute mode.

(11) Magnifying glass function: It magnify the local area of the image, and the window size can be set from 200% to 800% and move it by mouse. The partial image in the magnifying glass window can be magnified 2 or 4 times.

(12) Setting the color, size and shape of digitizing point, the background with or without gridding lines, output digitizing numerical data format is available.

(13) Project function: It is used to save image, digitizing results with other settings as a project file \*.GDP for checking and modification in the future.

(14) Remark function is applied to keep some marks and memo text for checking, modification and memory by user.

### 3. Basic Functions of GDGraph5.0

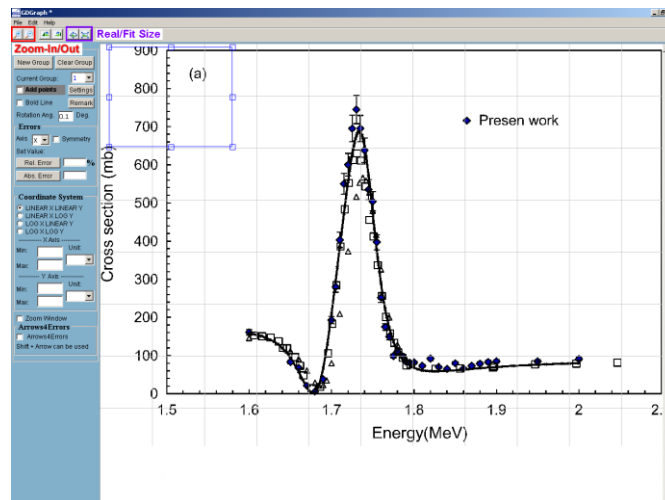
#### 3.1 Installing GDGraph5.0

Recently GDGraph5.0 can be used in WINDOWS operation system only such as WindowsXp or the higher versions. To download the installation file “GDGraph5.0.msi” from the website as “www.nuclear.nsd.c.cn/gdgraph” or “https://www-nds.iaea.org/nrdc/nrdc\_sft”, and the English manual also can be download from “https://www-nds.iaea.org/publications/nds/iaea-nds-0216”. On the other hand, the manual is included in the software also.

Double clicks “GDGraph5.0.msi” file to start installing GDGraph in your computer. Click “Next” button to continue install. The “Custom” mode of installation allows to choose the installation directory, and the “Install” mode use the default installation directory as “C:\Program Files\CNDC\GDGraph”. You can find the short cut execution link at Windows start program menu as “CNDC” and have a shortcut link at desktop also.

#### 3.2 Loading an image file

There have some options to load an image. One is using “Load Graph File” from the “File” menu in the menu bar. Select an image file (as PNG, GIF, BMP or JPEG etc.) from a file dialog box in a new window and the image is loaded with original size as default. Another way is directly copy an image using clipboard from other file such as MS Word, PDF, etc. If an image file is successfully loaded, the image is displayed on the main panel as shown in **Fig. 1**. It is allowed to rotate with an setting angle, zoom in, zoom out, auto fit to the GUI window and revert the image size.

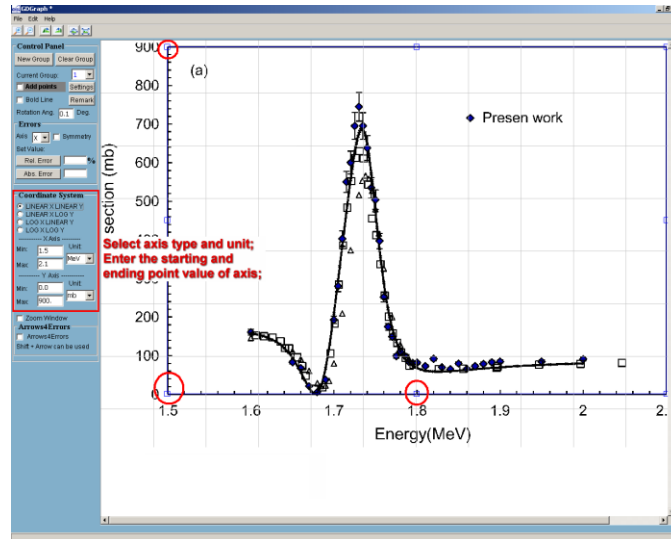


**Fig. 1:** An image loads on the main panel.



### 3.3 Setting axes

Use cross lines to set the X-Y axis, and adjust the square symbols of the starting, middle and ending positions of X-Y axis to fit the image one. Then select the X-Y axis type as Linear-Linear, Linear-Log, Log-Linear or Log-Log. According to the cross line positions, fill in the starting and ending value of X-Y axes, and select or fill in a unit for X-Y axes at “Coordinate System” in “Control Panel” for identifying the digitizing data in output data file, respectively. When we set or adjust the positions of axis, there exist an orthogonal cross lines to assist confirming the position of axes. When you finish setting the axis, you will see a window as shown in **Fig. 2**.

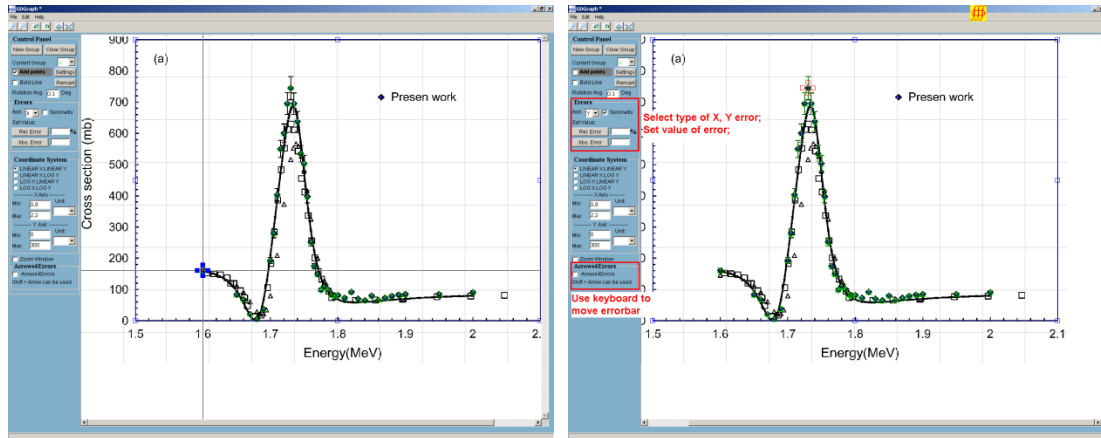


**Fig. 2:** Window after setting axes with squares.

### 3.4 Digitizing the data and error

Activate the check box “Add points” to start digitizing data points. If need to digitize more than one group points, please select which group you want to digitize or modify the original digitizing results. If you click on the image in the data input mode, a digitizing data point is added on the image. Continue to click the image until all the data points are added. When you finish adding the data points, you will see a window as shown in part (a) in **Fig. 3**.

The function of digitizing data point error is available with asymmetric or symmetric now. First of all, disable the “Add points” mode. Then, to select each error mode for X/Y with asymmetric or symmetric. The default error mode of X/Y is asymmetric. To set a symmetric error for X or Y, first select X-axis or Y-axis at “Axis” list box in “Errors” part, then click “Symmetric” check box. After that, click a data point to activate it which will appear 4 red square symbols around it. In this mode, it realize to move data point by mouse or cursor keys or set a fix value as X-Y error. The left and right square symbols represent X-Err-/X-Err+, and the bottom and top symbols represent Y-Err-/Y-Err+. If you obtain the information of X-Y uncertainty from paper or other ways, you can directly fill in a fix value as X-Y uncertainty with relative (% , in percent) or absolute mode. You can directly use mouse to pull one of four red squares to obtain X-Y error. On the other hand, you can apply “Arrows4Errors” function to realize using cursor keys to move one of four red squares to proper position. After inputting errors, you will see a window as shown in part (b) in **Fig. 3**. After digitizing one point, we can use “Pageup” or “Pagedown” key to activate the previous or next data point.



(a) Reading the data points (b) Reading the data errors  
**Fig. 3:** Window after reading the data points and errors.

### 3.5 Outputting the digitizing result

Select “Save Data File” from the “File” menu or directly use “Ctrl+C” to copy all digitizing results to clipboard and paste to other applications. The output data is in exponent or float format, and the number of digits can be set at “Settings” function. Each data can be set as 11 columns to fit EXFOR format requirements. The output file contains the information of each group No., number of data points, name of each column and digitizing data. Each group output contains X, Y, Y-Err+, Y-Err-, X-Err-, and X-Err+ as shown in **Fig. 4**.

| 1  | 2            | 3           | 4           | 5           | 6           | 7           | 8           | 9 | 10 | 11 | 12 | 13 | 14 | 15 | 16 | 17 | 18 | 19 | 20 | 21 | 22 | 23 | 24 | 25 |  |
|----|--------------|-------------|-------------|-------------|-------------|-------------|-------------|---|----|----|----|----|----|----|----|----|----|----|----|----|----|----|----|----|--|
| !  | ##           | X (MeV)     | Y (mb)      | dY+         | dY-         | dX+         |             |   |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |  |
| 1  | #####group 1 |             |             |             |             |             |             |   |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |  |
| 2  | ##           |             |             |             |             |             |             |   |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |  |
| 3  |              | 1.03415E+00 | 5.34411E+01 | 2.63429E+00 | 2.63429E+00 | 0.00000E+00 | 0.00000E+00 |   |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |  |
| 4  |              | 1.15063E+00 | 2.73492E+01 | 3.13606E+00 | 3.13606E+00 | 0.00000E+00 | 0.00000E+00 |   |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |  |
| 5  |              | 1.17425E+00 | 2.24569E+01 | 2.38340E+00 | 2.38340E+00 | 0.00000E+00 | 0.00000E+00 |   |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |  |
| 6  |              | 1.19746E+00 | 6.65120E+00 | 2.50884E+00 | 2.50884E+00 | 0.00000E+00 | 0.00000E+00 |   |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |  |
| 7  |              | 1.22109E+00 | 1.63351E+00 | 1.63075E+00 | 1.63075E+00 | 0.00000E+00 | 0.00000E+00 |   |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |  |
| 8  |              | 1.24430E+00 | 1.22961E+01 | 2.88517E+00 | 2.88517E+00 | 0.00000E+00 | 0.00000E+00 |   |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |  |
| 9  |              | 1.26751E+00 | 6.41037E+01 | 3.13606E+00 | 3.13606E+00 | 0.00000E+00 | 0.00000E+00 |   |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |  |
| 10 |              | 1.27932E+00 | 9.30809E+01 | 4.51592E+00 | 4.51592E+00 | 0.00000E+00 | 0.00000E+00 |   |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |  |
| 11 |              | 1.29073E+00 | 1.34100E+02 | 6.77388E+00 | 6.77388E+00 | 0.00000E+00 | 0.00000E+00 |   |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |  |
| 12 |              | 1.30294E+00 | 1.83274E+02 | 9.15728E+00 | 9.15728E+00 | 0.00000E+00 | 0.00000E+00 |   |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |  |
| 13 |              | 1.31435E+00 | 1.99958E+02 | 1.00354E+01 | 1.00354E+01 | 0.00000E+00 | 0.00000E+00 |   |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |  |
| 14 |              | 1.32616E+00 | 2.31695E+02 | 1.15407E+01 | 1.15407E+01 | 0.00000E+00 | 0.00000E+00 |   |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |  |
| 15 |              | 1.33780E+00 | 2.48260E+02 | 1.20897E+01 | 1.20897E+01 | 0.00000E+00 | 0.00000E+00 |   |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |  |
| 16 |              | 1.34937E+00 | 2.31569E+02 | 1.16661E+01 | 1.16661E+01 | 0.00000E+00 | 0.00000E+00 |   |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |  |
| 17 |              | 1.36118E+00 | 2.12627E+02 | 1.07880E+01 | 1.07880E+01 | 0.00000E+00 | 0.00000E+00 |   |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |  |
| 18 |              | 1.37258E+00 | 1.78256E+02 | 8.90640E+00 | 8.90640E+00 | 0.00000E+00 | 0.00000E+00 |   |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |  |
| 19 |              | 1.38480E+00 | 1.67719E+02 | 8.53007E+00 | 8.53007E+00 | 0.00000E+00 | 0.00000E+00 |   |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |  |
| 20 |              | 1.39539E+00 | 1.32093E+02 | 6.77388E+00 | 6.77388E+00 | 0.00000E+00 | 0.00000E+00 |   |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |  |
| 21 |              | 1.40802E+00 | 8.37982E+01 | 4.26504E+00 | 4.26504E+00 | 0.00000E+00 | 0.00000E+00 |   |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |  |
| 22 |              | 1.41860E+00 | 5.79571E+01 | 2.88517E+00 | 2.88517E+00 | 0.00000E+00 | 0.00000E+00 |   |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |  |
| 23 |              | 1.43082E+00 | 4.94270E+01 | 2.50884E+00 | 2.50884E+00 | 0.00000E+00 | 0.00000E+00 |   |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |  |
| 24 |              | 1.44304E+00 | 3.27432E+01 | 1.63075E+00 | 1.63075E+00 | 0.00000E+00 | 0.00000E+00 |   |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |  |
| 25 |              | 1.45404E+00 | 3.60047E+01 | 2.13252E+00 | 2.13252E+00 | 0.00000E+00 | 0.00000E+00 |   |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |    |  |

**Fig. 4:** An example of outputting numerical data file.

### 4. Conclusion

Since 1997, the digitization software GDgraph is developed to fit the requirements of evaluation, measurement and EXFOR compilation. From the mold software to present version 5.0, GDgraph is mainly fit the requirements, although there are some aspects need to modify and add some new functions also.

# Review on compilation work in JCPRG

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## Abstract

This document summarizes the review on the compilation status of JCPRG in Hokaido University and introduction of HENDEL editor. This paper also gives the description of Exchange Format for the transmission of experimental nuclear reaction data between National and International Nuclear Data Centres for the benefit of nuclear users in all countries maintained by the International Network of Nuclear Reaction Data Centres (NRDC) coordinated by IAEA Nuclear Data Section.

## Introduction

Development of a knowledge base and providing accurate description of the basic nuclear interactions is a fundamental and central part of the evolution of nuclear science and technology. It is a part of the inevitable necessity of knowledge sharing on nuclear data for various applications in nuclear physics.

“Nuclear data” is a technical term that stands for quantitative results of scientific investigations of nuclear properties of matter. These numerical data describe quantitatively the physical properties of atomic nuclei and the fundamental physical relationships governing their interactions, there by characterizing the physical processes underlying all nuclear technologies. Generation and proper use of nuclear data inevitably constitute important components of “basic science” and “base technology efforts” behind nuclear systems. Nuclear reaction data have been a crucial resource in nuclear technology (e.g., fission, fusion energy and medical diagnostics) as well as science (e.g., nuclear physics, astro-physics, nuclear chemistry and earth science). Generation and proper use of nuclear data comprising measurement and evaluation of recommended values of accurate nuclear data belong to the cutting-edge science and form an important component of basic nuclear physics. Various reaction models have been developed based on nuclear theory and phenomenology, and have been verified by experimental nuclear reaction data and utilized for revision of evaluated nuclear reaction data.

Hokkaido University Nuclear Reaction Data Centre (JCPRG) compiles and accumulates charged-particle nuclear reaction data obtained in Japanese facilities in their own data format (NRDF:Nuclear Reaction Data File) which are distributed through the internet [1]. A part of compiled files is translated to the EXFOR format for the transmission of the experimental nuclear reaction data between national and international nuclear data centers for the benefits of nuclear data users in all countries [2].

## Compilation of New and Old Papers

Total number of new entries compiled in fiscal year 2013 is 25. Total number of old entries compiled is 35. Therefore total number of entries compiled is 60.

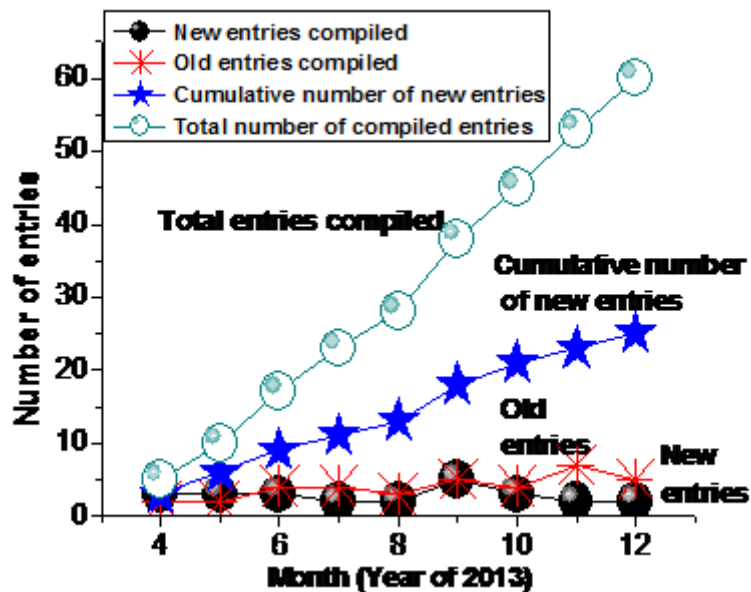


Figure1: The graph of number of entries compiled in fiscal year 2013.

### RIKEN Papers Compiled in 2013

The JCPRG-RIKEN nuclear data project started in 2010 to distribute the new-fangled RIBF experimental data to all over the world via the EXFOR database. With the progress of globalization for RIBF experiments, collaborative relationship for the compilation progress with world experimentalists has been built successfully. Main accelerators in RIBF are a heavy-ion linac (RILAC), AVF cyclotron (AVF), ring cyclotron (RRC). In addition, newly developed ring cyclotron and superconducting ring cyclotron, are available and make possible to perform various kinds of experiments by using unstable beams [3].

Number of papers compiled in 2013 by JCPRG group that contained the RIBF data are 10. All eleven entries had already been transmitted [4].

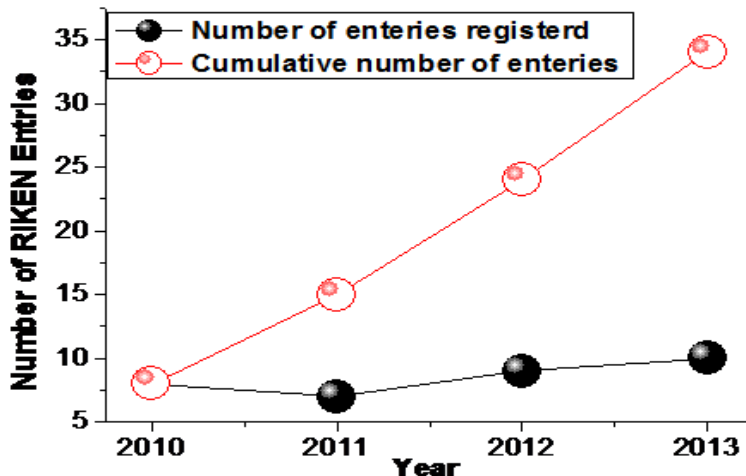
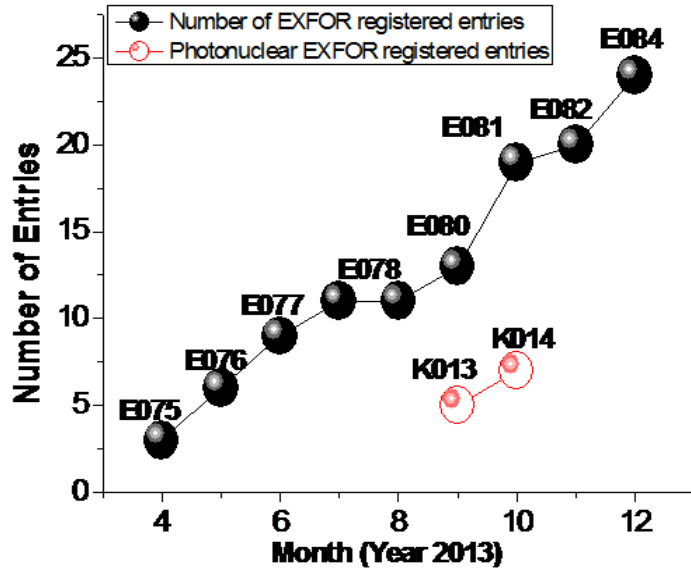


Figure2: The graph of number of RIKEN entries compiled in fiscal year 2010-13.

## Transmitted Papers

The graph shows the number of entries transmitted in fiscal year 2013. This graph shows the variation of number of entries transmitted in each month of year 2013.



**Figure3:** The graph of number of entries transmitted in fiscal year 2013.

The solid sphere indicate number of entries transmitted in every year and empty sphere shows the photonuclear EXFOR entries registered in this year.

## Acknowledgment

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## References

- [1] <http://www-nds.iaea.org/exfor-master>
- [2] <http://www.jcprg.org/>
- [3] <http://www.nishina.riken.jp/RIBF/accelerator/overview.html>
- [4] <http://www.jcprg.org/riken/http://www.jcprg.org/>



## Panel Discussion

In the concluding session, we had panel discussion where panellists included one representative from each country and was moderated by Dr. S. Kailas. The participants were also asked to present their views about the future AASPP workshops. Some of the highlights of the panel discussions are as follows:

Dr. S. Ganesan pointed out that Dr. Kato and Dr. Aikawa started the AASPP event in 2010 in Sapporo. It was unanimously decided that the AASPP meetings should continue as they provide the ideal platform for the scientists working on nuclear data in these countries to meet and share their experiences and it was decided that next AASPP workshop will be hosted by Faculty of Science, Hokkaido University, Sapporo. There was a suggestion about exploring the possibility to organize a school on the broad theme Nuclear Data under the AASPP banner. It was suggested to host a Website page for AASPP workshops which is now already implemented by Dr. N. Otsuka of the IAEA. Dr. N. Takibayev emphasized about the need of cooperation between the neighbouring countries on nuclear data and sharing of research facilities. There should be joint research in the field of nuclear physics. Dr. Krishnani suggested that yearly AASPP news e-letter can be released where collaborating institute can report whatever they think worth reporting. He also suggested about possible collaboration on computer programs for nuclear data evaluation. It was also suggested to bring out the proceedings of this workshop which is being brought out as the present INDC(IND) report. Dr. Alok Saxena thanked all the participants, invited speakers, organizing and advisory committee members for the active involvement and also the DAE-BRNS and the IAEA for supporting the workshop. The help and support of Dr. Otsuka of IAEA during the various stages of organization of this workshop are gratefully acknowledged.







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