Nuclear Data Activities in India 2018-2020

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Introduction

Nuclear reaction cross section is constantly upgraded with precision and accuracy increasing manifold during the past years. Experimental facilities with cutting edge technologies are used to measure the reaction of interest for different projectiles and at several energies. This forms the basic inputs for nuclear reactor design, irradiation of isotopes, accelerator driven systems and other multi-disciplinary fields such as material diagnostics, material science, medical physics, astrophysics. Interaction of matter with a nucleus is continuously engaging field and there are many diverse domains which require to be treated comprehensively. Department of Atomic Energy, India has been actively pursuing many research studies towards the development and use of nuclear reaction data. This report gives an overview of the activities done during 2018 to 2020. The report is organised in five major sections:

A. Activities being pursued in India
B. Summary of nuclear data related activities taken up during 2018 to 2020
C. Gap areas and focus for the future
D. Recommendations
E. Publications

A. Activities being pursued in DAE

Various aspects of nuclear reaction data being pursued in DAE can be grouped under major categories such as

A.1. Nuclear data measurements: Several reaction cross sections have been measured and are also required to be measured in future in order to remove the uncertainties and make improved data available. The different sources of projectiles (neutrons, electrons, protons, heavy ions etc.) in DAE are

i) Research Reactors (Dhruva, AHWR-CF, Apsara-U, KAMINI, FBTR)
ii) Surrogate sources (TIFR, FOTIA)
iii) Accelerators (electron, proton and heavy ion based)
iv) Other in-house facilities

A.2. Contributions to the EXFOR database of IAEA: Several nuclear reactions have been coded in the EXFOR database. This is an on-going activity. Many more reactions are required to be coded.

A.3. Use of processed/ evaluated data for fuel cycle:

i) Processed data is the source of many applications be it reactors, accelerators, astrophysics or even medical physics. Several gap areas where adequate nuclear data is not available have been identified.
ii) Data for fuel cycle analysis: Accurate and extensive set of fission yield data need to be updated using the latest measurements for fissile nuclides (including ^{233U}) and the FP decay data including the average beta particle and gamma ray energies per decay.
iii) Point reactor burnup codes are integral part fuel cycle studies. The spectrum averaged database is constantly updated.

A.4. Evaluation of ENSDF data: Several mass chains are required to be evaluated with respect to nuclear structures.

A.5. Processing of neutron-nuclear cross section data in multi-group format (For advanced reactor being designed in India AHWR, IPWR, CHTR, IMSBR). Though several formatted energy averaged nuclear cross section libraries are available, even today there have been many instances of deviations in predicted parameters. A detailed benchmarking for qualification is another major activity.

- Benchmarking of multi-group formatted data for different reactor types
- Extension of burnup chain for actinides
- Validation of nuclear model codes
- Uncertainty and sensitivity studies
- Sensitivity studies with covariance data
- Processing of data for nuclides not available in international database

A.6. Nuclear reaction data measurements for other applications: Experimentation for measuring reaction cross section and developing facilities and detection mechanisms has reached a high level of maturity in India. Some recent activities are :

i) High precision double differential cross-section data for $^{7}Li(p,n)$, $^{9}Be(p,n)$ reactions for medical and nuclear applications.
ii) Evaluation of cross-sections for 6-10 MeV proton energy for Li to Zn targets particularly (p, p'γ) reaction for PIGE. The validation can be done with the data available for 2-5 MeV proton beam
iii) Proton activation data for full range of targets/ structural and shielding materials used in accelerators for ADS programme. The accuracy of data for *minor* actinides, and other materials which are specific to ADS must be improved for a realistic design, since there are still very large differences among various libraries.
iv) Beta feeding intensity measurement using TAGS
v) Data for astrophysics applications
vi) Production of medical isotopes-effective yield estimation; Influence of resonance shielded data

A.7. Other experiments for generation of nuclear data

A. Fission products yield measurement- Energy dependent fission yields of actinides
B. Fission spectrum measurement for actinides
C. Decay heat measurements for thorium/Uranium based fuel.
D. Accelerators as source of neutrons for cross section measurement

A.8. New experimental facilities for generation of nuclear data

Development Neutron time of flight techniques for neutron cross section measurement: In an endeavour to build better facilities for cross section measurement, a preliminary design of beam tube is being done with neutrons from 30 MeV accelerator

A.9. Other projects/activities

i) Inter code validation and comparison between EMPIRE and TALYS and adopt a single global data library to act as standard from which end users can prepare their own sub-library specific to a particular type of application.
ii) Continued support for EXFOR compilation activities by IAEA-NDS in India keeping in mind long-term requirements.
iii) Nuclear Data Schools and training courses on evaluation techniques.
iv) Maintaining the IAEA mirror site http://www-nds.indcentre.iaea.org.in

B. Summary of nuclear data related activities taken up during 2018 to 2020

B.1. Application of processed data for reactor analysis

1. Nuclear data sensitivity studies for the reactor physics parameters of Indian Pressurised Water Reactor (IPWR):

As part of the design qualification, sensitivity of the reactivity coefficients in IPWR fuel to various multi-group WIMS formatted libraries obtained from IAEA was done. IPWR fuel assembly and core has been simulated with WIMS-ARCH code system using ENDF/B-VII.1, JENDL3gx and JEFF3.1gx nuclear data libraries. The results using JEF3.1gx and JENDL3gx were consistently lower than the design basis value calculated with ENDF/B-VII.1. The overall deviation in the calculated isothermal temperature coefficient for the equilibrium cycle were between -6.7% to -10% from BOC to EOC and that in the power defect was -9.9% to -14%. The saturating fission products load (Xe and Sm) showed a maximum difference of -14% over the entire cycle.

Publication: Nuclear data sensitivity analysis of IPWR fuel assembly and core, Anindita Sarkar, Umasankari Kannan, RPDD/IPWR/63, 2018.

2. Estimation of isotopic contribution to reactivity swings of IPWR:

Energy wise isotopic components of the reactivity swings due to isothermal temperature change and power defect of IPWR has been estimated. The calculation has been done using normalised reaction rate approach. Figure 1 shows the contributions of individual isotopes to the isothermal temperature coefficient. The simulation has given an insight into which isotopes and which energy range is significant for the magnitude and the sign of the reactivity coefficient.

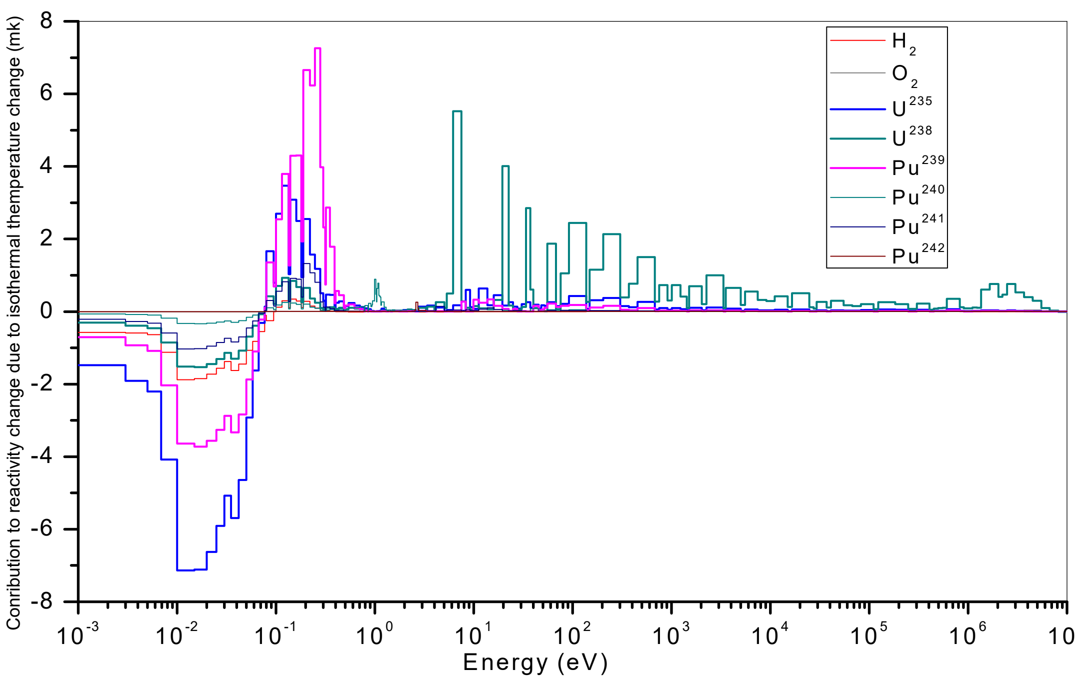

Fig.1 Influence of Isotopic and energy due to different libraries to isothermal temperature coefficient in IPWR

Publication: Anindita Sarkar, Umasankari Kannan, Estimation of isotopic contribution to reactivity in IPWR, June 2019, RPDD/IPWR/88/2019.

3. **Nuclear data sensitivity studies for Thorium based AHWR fuel**
The multi-group cross section sets for thorium-based isotopes are not qualified to the level of isotopes of U-Pu cycle. In another study the uncertainty in the calculated safety parameters due to the use of different nuclear cross section sets for the thorium fuelled AHWR design was quantified. AHWR uses both plutonium and Uranium-233 with thorium in the same fuel assembly. The variation in the calculated moderator temperature coefficient and coolant void reactivity from different data libraries is shown in figures 2a and 2b. These margins are quantified in the design stage as uncertainties.

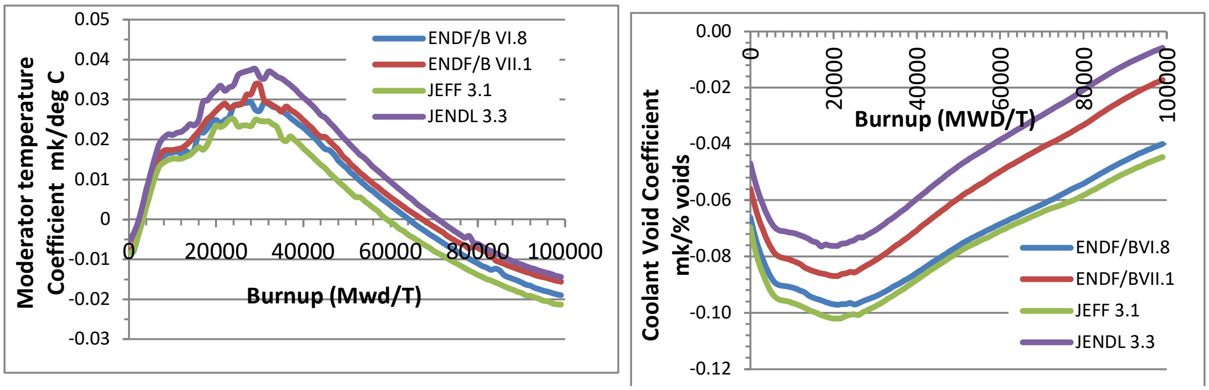

Fig.2a and 2b Sensitivity of moderator temperature coefficient and coolant void reactivity in AHWR due to different processed multi-group libraries

*Publication: Umasankari Kannan, "Perspectives on nuclear data for advanced reactor design and analysis", Journal of Life Cycle Reliability and Safety Engineering, **9**(2), 135-146 DOI 10.1007/s41872-020-00120-5.*

B.2. Processing of cross section for reactor analysis

1. **Generation of continuous energy cross section libraries using NJOY-2016 for thermal reactor applications from ENDF/B VIII.0 and ENDF/B-VI.8 data**
In this work, the evaluated nuclear data in the ENDF/B VIII.0 files are processed using the NJOY code to prepare ACE (A compact ENDF) formatted libraries. The effect of chemical binding of Hydrogen in water, Doppler broadening and up-scattering are considered while processing. The thermal scattering data are given using the scattering law tables [S(α,β)] in the file 7 of ENDF files. The thermal scattering data is processed for moderators at different operating temperatures to account for the lattice binding effects. The scattering cross section of H and Be is presented here.

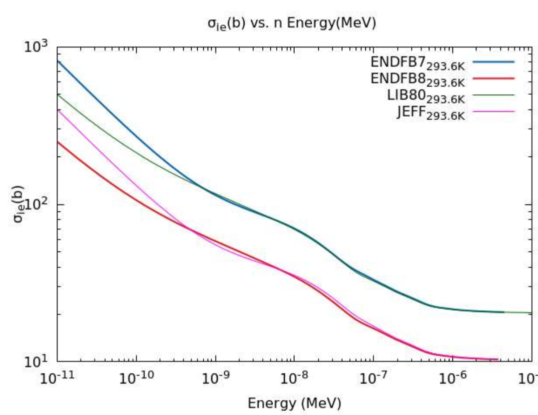

Fig. 3a Inelastic cross section of H in Water

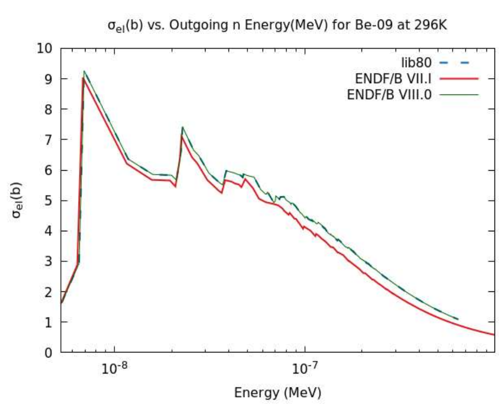

Fig. 3b Elastic cross section of Be

In order to use updated continuous energy cross section for reactor design, a complete set for over 400 nuclides have also been generated from ENDF/B-VI.8 to obtain ACE-formatted data library at 300K. Nuclides that was not present in ENDF/B-VI.8 data library (e.g Mo-98) has been processed separately and added in the new library.

2. Continuous energy data for temperature coefficient calculation:

Unlike the data files of deterministic calculation, where a single file contains cross sectional data at multiple temperatures, ACE formatted files are meant to be valid at a particular temperature for which it is generated. Thus, evaluation of temperature coefficients is impossible by doing temperature interpolation using single ACE formatted data file. One way out is to generate data files at different temperatures at a regular interval and performing k-calculations at those temperatures. For this purpose, major actinides and moderator nuclides have been processed at various temperatures with a temperature gap of 50^{0}C.

3. Estimation of Self-shielding factors for Neutron Activation Analysis from ENDF/B VIII.0 data using NJOY-16

The accurate estimation of self-shielding factors is very important in neutron activation analysis. The self-shielding effects are influenced by the geometry as well as the composition of the sample used for activation. The presence of resonance absorbers leads to dips in the neutron flux spectrum. Theoretical estimation of the self-shielding factors can be performed using the processing code NJOY-16 from the evaluated nuclear data files. The self-shielding factors are generated as a table for several dilutions at the specified temperatures. For the problem at hand, the self-shielding factors can be interpolated. In this study the self-shielding factors are calculated for a cylindrical sample. Additionally, the resonance interference effects are studied using the flux calculator of NJOY-16.

Publication: V. Harikrishnan, Krishna Kumar Yadav and Usha Pal, 'Estimation of self-shielding factors for neutron activation analysis from ENDF/B VIII.0 nuclear data using NJOY16' MTAA (International Conference on Modern Trends in Activation Analysis)-15, Mumbai November 17-22, 2019; page A-19, book of abstracts.

4. Nuclear Data Library development for the fuel cycle studies.

An integral part of fuel cycle studies is the point reactor transmutation and burnup code and its library (falling in class of ORIGEN, ORIGEN-S, FISPACT etc). The spectrum averaged nuclear cross section is required to cater for the fuel cycles specific to India, namely PHWR-220, 540, 700 and AHWR. A fuel cycle code, ADWITA, together with its library has been developed. The code and data library are being updated regularly. Recently the fission yield of fission products from ENDF/B-VIII library has been updated for the use in Indian PHWR reactors. In case of PHWR the prominent nuclides which undergo fission during the time when fuel is inside the core are U^{235}, U^{238}, Pu^{238}, Pu^{239}, Pu^{240}, Pu^{241}, Pu^{242}. Some more isotopes need to be included to account for the use of thorium and Pu based fuel in AHWR. Owing to this, the fission products from four more isotopes, namely, Th^{232}, U^{232}, U^{233} and Am^{241} has been added to the library.

B.3 Nuclear reaction experiments

1. The feasibility of producing ^{99}Mo from Bremsstrahlung photons from electron accelerators through a e-gamma-n route was studied. To this end an experimental set-up was designed to measure the effective cross section of ^{98}Mo(n,γ)^{99}Mo using 10 MeV LINAC of Electron Beam Centre, Kharghar with different moderating arrangements. Experiments were done with HDP moderator. Several foils were activated in the electron LINAC at EBC to measure reaction rates and estimate the reaction cross section. In the first set of the experiments, natural Mo metal foil along with other thermal and resonance flux monitors were irradiated in slightly moderated photo neutron flux. Bare and cadmium covered monitors were used. The activation foils used were thermal flux monitors i.e. Gold, Copper, Manganese, Indium and cobalt and resonance (epithermal) flux monitors namely, Scandium, Silver, Lutetium, Tantalum and Tungsten. The electron Accelerator was operated for 4 hours at 0.24 mA current. Induced gamma activities in the flux monitors were counted using High purity Germanium Gamma detectors. The induced activity was very close to the theoretical estimates**.**

Publications: Kapil Deo, Rajeev Kumar and Umasankari Kannan, "Feasibility study of 99Mo production using electron accelerator-based neutron source", RPDD/EP/156 dated 17th May, 2018.

Kapil Deo and Umasankari Kannan, "Design of an electron accelerator-based photoneutron converter for achieving high neutron flux", RPDD/EP/172 dated 18th June, 2019.

2. Measurements of prompt fission neutrons and gamma ray spectra have been carried out in the fast neutron induced fission of ^{232}Th at an average neutron energy of 2.6 MeV. The experiment was carried out at the Folded Tandem Ion Accelerator (FOTIA) facility, BARC, Mumbai. Quasi monoenergetic neutrons were obtained using the $^{7}Li(p,n)^{7}$ Be reaction by bombarding the proton beam on a natural Li metallic target of thickness ~ 4.0 mg/cm^2. The energy spectrum of neutrons is obtained from the time-of-flight spectrum measured using EJ301 detectors with respect to a twin section trigger fission chamber while the energy spectrum of prompt gamma rays is obtained after unfolding the measured pulse height spectrum using the response matrix of the CeBr3 detectors. The Maxwellian temperature obtained from the prompt fission neutron spectrum was found to be 1.22 ± 0.03 MeV while the average neuton multiplicity deduced from the measurement was 2.049 ± 0.121. The average energy of gamma rays was found to be 0.91 ± 0.04 MeV and the average gamma multiplicity was obtained as 6.55 ± 0.32**.**

Publication: Sukanya De, G. Mishra, R. G. Thomas et al., European Physical Journal A 56, 116 (2020).

3. Neutron spectra were measured in coincidence with evaporated alpha particles produced in the reactions $^{11}B+^{181}Ta,^{197}Au$. The nuclear level density parameter has been extracted for the Os(A≈188) and Pb(A≈204) isotopes by comparing neutron spectra with statistical model prediction. Evidence for collective enhancement has been found for Os nuclei whereas no such enhancement has been seen for Pb nuclei. The energy-dependent enhancement factor has been extracted by simultaneous fitting of the neutron spectra at various excitation energies. Near a temperature of 0.8 MeV, the enhancement starts to fadeout which is lower than the theoretically predicted temperature of 1.4 MeV for ^{187}Os. Also, free energy surface calculation shows that the ^{187}Os nucleus undergoes a transition from collective prolate to noncollective oblate shape close to the temperature of 0.8 MeV, corroborating the early fadeout. No such shape transition is seen for ^{203}Pb.

Publication: G. Mohanto et al. Phys. Rev. C 100, 011602(R) (2019).

4. The $^{54}Mn^*$(surrogate of n+^{53}Mn) and $^{61}Ni^*$(surrogate of n+^{60}Ni) compound systems have been populated at overlapping excitation energies by transfer reactions $^{52}Cr(^{6}Li,\alpha)^{54}Mn^*$ at E_{lab}=33.0 MeV and $^{59}Co(^{6}Li,\alpha)^{61}Ni^*$ at E_{lab}=40.5 MeV, respectively. Measurements were carried out in BARC-TIFR Pelletron Accelerator Facility in Mumbai. The ^{53}Mn(n, xp) cross sections in the equivalent neutron energy range of 8.2–16.4 MeV have been determined within the framework of the surrogate reaction ratio method using ^{60}Ni(n,xp) cross-section values from the literature as a reference. The measured ^{53}Mn(n,xp) cross-section values are found to be consistent with the predictions of the TALYS-1.8 statistical model code using microscopic level densities and results of various evaluated nuclear data libraries: EAF-2010, ROSFOND-2010, and JEFF-3.3 within the experimental uncertainties. *Publication: Ramandeep Gandhi, B. K. Nayak et al., PRC, Phys. Rev. C 100, 054613 (2019).*

5. The ^{59}Ni(n, xp) reaction cross sections have been measured following the surrogate reaction ratio method in the equivalent neutron energy range of 11.9–15.8 MeV by populating the compound nucleus $^{60}Ni^*$ through transfer reaction $^{56}Fe(^6Li, d)$ at E_{lab} = 35.9 MeV. The $^{59}Co(^6Li, \alpha)^{61}Ni^*$ transfer reaction at E_{lab} = 40.5 MeV has been used as the reference reaction. Measurements were carried out using ^6Li beams obtained from BARC-TIFR Pelletron Accelerator Facility in Mumbai. The present experimental data are consistent with the evaluated data library of ROSFOND -2015 but not with TENDL -2015 and ENDF / B - VIII, indicating the need of new evaluations for this reaction of importance to fusion technology. The experiments were carried out by JRF's viz. Ms. J. Pandey and Ms. Bhawna Pandey under the guidance of BARC group.
   Publication: Jyoti Pandey, Bhawna Pandey, A. Pal, S. V. S Suryanarayana et al. PRC, Phys. Rev. C 99, 014611 (2019).

6. The neutron energy spectra in coincidence with both the correlated fission fragments in reaction $^{184}W(^{32}S, F)^{216}Th^*$, E=180 MeV or E^*=71.3 MeV have been measured at various angles with respect to one of the fission fragments using single-sided silicon strip detectors for fission fragments, and an array of 14 liquid scintillators for neutrons at the BARC-TIFR 14UD Pelletron-Linac accelerator facility. Pre- and post-scission neutron multiplicities of 2.68±0.48 and 3.3±0.20, respectively, have been extracted by analyzing angular correlation of the neutron energy spectra using moving source fitting procedure. Comparison with statistical model calculations using JOANNE2 code, and with dynamical model calculations using HICOL code are performed. The experiments were carried out by JRF Prashant N. Patil under the guidance of BARC group.
   Publication: Prashant N. Patil, N. M. Badiger, B. K. Nayak, S. Santra et al. Phys. Rev. C 102, 034618 (2020).

B.4. Code development for processing nuclear data

1. **Development of a code to generate point-wise neutron scattering cross sections in thermal range using incoherent inelastic scattering**
   In the ENDF/B formatted data library, neutron scattering data for the thermal neutron energy range for moderating materials are described by thermal scattering law data. Unfortunately, these data cannot be used in neutron transport theory code directly. A FORTRAN module is required to convert the existing ENDF/B data into double differential form. This will allow to use the ENDF/B data in any neutron transport code, be it a Monte Carlo, or deterministic code. There are different definitions to represent the thermal scattering law data.

   A code has been developed to handle the thermal scattering law data which is used to represent the incoherent inelastic scattering in ENDF/B formatted data library. In incoherent inelastic scattering, both outgoing energy and outgoing direction can vary continuously. This representation is used in all moderating materials. This code also generates point- wise neutron scattering cross sections in thermal range using free gas model. The free gas model can be represented analytically and easy to generate point-wise neutron scattering cross sections.
   Publication: Anek Kumar, Umasankari Kannan and S. Ganesan, "Development of an Assembly-Level Monte Carlo Neutron Transport Code "M3C" for Reactor Physics Calculations", Journal 0f Nuclear Science and Engineering, July 2019, https://doi.org /10.1080/00295639.2019.1645502

2. **A code for calculating the intermediate resonance parameter (lambda factor or Goldstein-Cohen parameter) for deterministic multi-group neutron transport calculation**

Accurate resonance parameters can be as important as the multi-group neutron cross sections themselves in the overall accuracy of a multigroup library. The intermediate resonance (IR) approximation is based on the traditional narrow (NR) and narrow resonance infinite mass (NRIM) approximations. A parameter called Goldstein-Cohen is introduced for interpolation between NR and NRIM approximation. Earlier, a code is developed to solve the slowing down equation using 'equivalent nuclide model'. In the 'Equivalent Nuclide Model', only one fictitious resonant nuclide 'Equivalent Nuclide' is used to do the calculation instead of all actual resonant nuclides.

The code developed is used to calculate slowing down spectra in an infinite homogeneous U-238/H and U-238/H/isotope-i mixtures where i is the non-hydrogen scattering isotope. The lambda factors of non-hydrogen scattering isotopes have been obtained by comparing solutions in mixtures of resonance isotopes with mixtures in which the hydrogen is partially replaced by other isotopes. The slowing down spectrum in an infinite homogeneous U-238/H mixtures at 300K with ratio N(R):N(H)=1:5, which led to σ_0=30 was calculated with the help of developed code. The average absorption cross sections of U-238 in the individual resonance group are estimated. The lambda calculation for different energy group was also performed and the values are found matching with the published lambda values.

B.5. Nuclear data management

1. Biennial EXFOR workshop

In India due emphasis is given to the experimental nuclear data measurement and its EXFOR compilation. The EXFOR coding is implemented through a 3 year fully supported projects awarded by DAE-BRNS to carry out such research activity. The Indian EXFOR workshop is part of such nuclear data initiative. Biennial EXFOR workshop with the active co-operation from the IAEA and DAE-BRNS is organised in India. The workshop is organised in collaborating universities. The workshop includes lectures and exercises on nuclear data related subjects including EXFOR compilation. There has been eight such workshops till now. The last workshop was organised at M. S. University, Vadodara, India during Nov 11 to 13 2019. The contribution from India EXFOR workshop to the EXFOR database is summarized in table-1 and fig-4a and fig-4b.

Table 1 Statistics of EXFOR data compiled in India

Year	Indian Contribution	World Total	Percent Contribution	Cumulative Contribution
2006	10	680	1.5	10
2007	34	720	4.7	44
2008	15	660	2.3	59
2009	63	743	8.5	122
2010	14	482	2.9	136
2011	59	541	10.9	195
2012	19	628	3.0	214
2013	51	460	11.1	265
2014	23	440	5.2	288
2015	49	441	11.1	337
2016	8	401	2.0	345
2017	54	450	12.0	399
2018	23	513	4.5	422
2019	79	565	14.0	501
2020	20	446	4.5	521

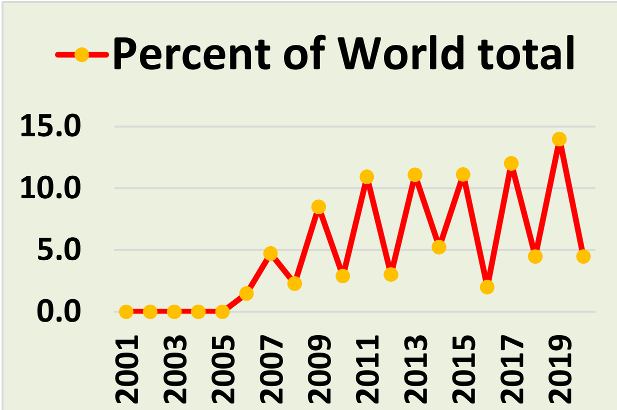

Fig. 4a Contribution of Indian experiments to EXFOR database - Percentage of world total

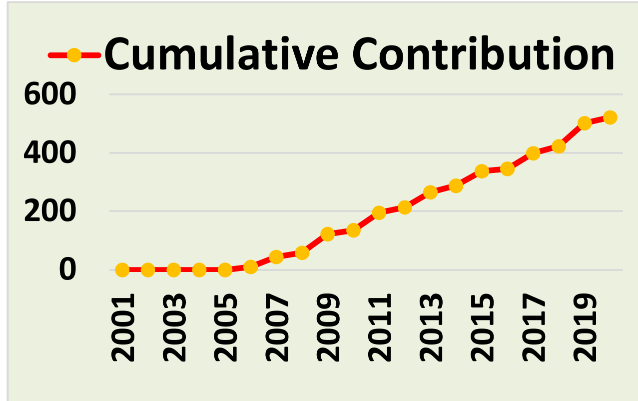

Fig 4b Cumulative contribution of Indian EXFOR workshops to IAEA EXFOR database

2. NRDC meeting in India

A 4- Day Biennial IAEA Technical meeting of International Network Nuclear Reaction Data Centres", NRDC-2018 was organised during May 01-04, 2018, at Global Centre for Nuclear Energy Partnership (GCNEP), Bahadurgarh, India. The meeting was attended by experts from 14 different scientific establishments represented by 16 foreign delegate and 8 Indian participants and observers. The agenda of the meeting consisted of presentation and discussion on Progress Reports and the technical issues pertaining to (a) EXFOR in general, (b) EXFOR Manuals and code dictionary, (c) CINDA, (d) EXFOR compilation needs, (e) EXFOR quality control, (f) EXFOR coding Rules, (g) Software and Dissemination.

C. Gap areas in experimentation, evaluation and use of processed data

A few examples are given below to stress the need for interactions with different agencies and international collaboration. This is not an exhaustive list.

Gap area-1*: Updating cross sections for new materials and re-evaluating existing data*

From detailed analysis and review of experimental data available currently, some deviations in evaluated data were found. Also, there is paucity of data in certain energy domains and certain operating temperatures. A few examples are listed below:

- **Updating thermal scattering data library for a larger number of temperatures and for all types of moderating materials:** Thermal scattering nuclear data in the evaluated nuclear data libraries are given at definite temperatures which cannot be Doppler broadened by any nuclear data processing code like NJOY and PREPRO code systems for desired temperature. For more precise reactor calculation, it is required to generate the thermal scattering law data for finer temperature grid as well as for higher temperatures. Thermal scattering data of Be, BeO, BeF_4 is also required.
- ^{232}Th data still has not converged between evaluated nuclear datasets. Detailed benchmarking and qualification of the new dataset based on ENDF/B-VIII.0
- ***Resonance tabulations for all actinides in WIMSD formatted library***: The actinide beyond Pu do not have resonance tables. If possible, all actinides should be treated as resonant absorbed and temperature - dilution dependent resonance tables should be given.

- Fast neutron induced fission of actinides giving information on mass, charge, kinetic energy distributions of fission fragments and neutron multiplicities
- Validated data for moderating materials like Graphite, Be, BeO
- Other isotopes of interest, Li^6, Li^7, Nat. Gd, Nat. B, Nat. Dy, Zr isotopes
- Need for a qualified data set for high temperature applications. For example, since Lead-Bismuth eutectic is being considered coolant in many reactor types (CHTR) the data should be available up to 1500K.

Gap area-2: Nuclear data evaluation
The gestation period from EXFOR to evaluation requires to be reduced. From measurement to best estimate evaluation, a qualified procedure to be evolved. This is a very voluminous work and has to be colloborative. Benchmarking and beta testing have to be simultaneously done.

Gap area-3: More precise experimental techniques
Better detectors and better instrumentation with cutting edge technologies is the need of the hour.

Gap area-4: Surrogate reactions is a source of required energies and dependence on these have increased. Special attention is required to address the choice of suitable targets. Although the projectile energies can be mono-energetic, the surrogate materials and reactions will have to be chosen carefully for a precise reaction cross section measurement.

Gap area-5: Experimental techniques and instruments
Stability of beams for measurement of cross sections is very important. Availability of clean or pure samples is another important aspect of experimentation.

D. Recommendations

I. Co-ordinated Research Projects to be taken up (IAEA, International projects) for the following activity

a) High temperature evaluated nuclear data in ACE format (for Monte Carlo and ADS applications) and in multi-group format (for High temperature reactors)

- A comprehensive review of the high temperature data for actinides, fission products, structural materials (Similar to the WLUP).
- Generation and testing of evaluated nuclear data for temperatures upto 2000 K
- Generation of multi-group constants for potential application in MSR,HTR,
- Validation and benchmarking of the multi-group dataset

b) Updating the thermal scattering data: The evaluated data files currently have only a few temperatures at larger grid. For a better treatment of thermal phenomena, and for more temperatures and other materials such as BeO and other moderating materials.

c) CRP on update of libraries for point reactor burnup codes: This CRP is required to update the libraries for fuel cycle calculations for several new reactor types. The task may be taken up along the lines of the IAEA-CRP on WIMS Library Update Project (WLUP). The task can involve

- Comprehensive study of existing nuclear data for fuel cycle studies.
- Benchmarking of existing data for new fuel cycles and reactors
- Update of the reactor specific cross sections using lattice codes along with point burnup simulations

Recommending a rigorous validated database for a comprehensive fuel cycle analysis for all types of facilities.

d) A new CRP on evaluated data for Th-U cycle and its applicability to Advanced fuel cycles, MSRs and Gen-IV systems

e) CRP on Surrogate reactions for neutron induced fission cross-sections for unstable actinides targets and also for exploring possible candidate reactions for fusion energy applications.

f) IAEA workshops on evaluation - Focussed themed workshops on

- Fitting of experimental data – New filtering techniques
- Improved nuclear models in evaluation
- Big data management
- Threshold and high energy reactions

II. Collaborative projects for measurement in International facilities like CERN, through coordination by IAEA

a) Formation of working group like CSWEG for continuous interactions with experimenters and users
b) Enhanced beta testing- More participation from different users
c) Recommend new experiments in uncharted regimes
d) Improve the frequency of NRDC and other expert group meetings
e) Mechanism for continuous collaboration and

III. Inter lab and inter code benchmarking for adequacy of resonance energy treatment in thermal reactor systems

Investigation of the adequacy of the resonance energy range of multi-group nuclear datasets for thermal reactor systems

The most prominently used multi-group data for thermal reactor applications is the WIMS format, where the resonance energy range is defined from 4.0 eV to 9.118 keV. Sensitivity studies and benchmarking have established the need for increasing the resonance energy range. Low energy resonances like 0.3 eV of ^{239}Pu, 2.2 eV of ^{233}U are important for Th and U-PU systems. The HELIOS code system has nuclear data library where the resonance energy range is from 1.85 eV to 111.1 keV. This implies that the upper bound of thermal energy limit is also brought down to 1.85 eV. Up-scattering below 4 eV is important for many moderators like H_2O, D_2O, Be and C.

IV. Creation of Joint evaluated IAEA file: International collaborative efforts to reduce the deviations between evaluations. Revisit the files and their use every 10 years and recommend comprehensive evaluated file and disseminate updated data through IAEA-NDS

a) Expand the evaluators domain – The evaluation procedures and methods should be open to larger workforce who can effectively contribute
b) Encourage development of newer methods for evaluation procedures

Summary

Nuclear data measurement and processing for potential applications is time intensive, cost intensive and a huge collaborative effort. Each and every contribution will be required to create a qualified database. The IAEA is connecting agency between different types of experts namely the experimenter, the data analyst, evaluators, experts, potential users and is doing well coordinated job of bringing together all these experts. The focus on the next few years should be on development of mechanism for benchmarking and testing and CRPs is the best way to go ahead,

E. PUBLICATIONS IN 2018-2020

1. Perspectives on nuclear data for advanced reactor design and analysis", Umasankari Kannan, *Journal of Life Cycle Reliability and Safety Engineering*, **9**(2), 135-146 DOI 10.1007/s41872-020-00120-5
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