

International Evaluation Co-operation

Volume **

Meeting Nuclear Data Needs for Advanced Reactor System

A report by the Working Party on International Evaluation Co-operation of the NEA Nuclear Science Committee

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Organization for Economic Co-operation and Development

Foreword

The Working Party on International Nuclear Data Evaluation Co-operation (WPEC) was established under the aegis of the OECD/NEA Nuclear Science Committee (NSC) to promote the exchange of information on nuclear data evaluations, validation and related topics. Its aim is also to provide a framework for co-operative activities between the members of the major nuclear data evaluation projects. This includes the possible exchange of scientists in order to encourage co-operation. Requirements for experimental data resulting from this activity are compiled. The WPEC determines common criteria for evaluated nuclear data files with a view to assessing and improving the quality and completeness of evaluated data.

The parties to the project are: ENDF (United States), JEFF/EFF (NEA Data Bank member countries) and JENDL (Japan). Co-operation with evaluation projects of non-OECD countries, specifically the Russian BROND and Chinese CENDL projects, are organized through the Nuclear Data Section of the International Atomic Energy Agency (IAEA).

The following report was issued by WPEC Subgroup 31, whose mission was to utilize the collective knowledge of the international nuclear data measurement community to consider the appropriate resources to address and meet the data needs quantified by WPEC Subgroup 26 for Advanced Reactor Systems. Members of Subgroup 31 performed reviews of uncertainty evaluations by evaluators, of state-of-art experimental techniques, of current experimental situations, and summarized an appropriate path to meet the requirements.

The opinions expressed in this report are those of the authors only and do not necessarily represent the position of any member country or international organization.

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TABLE OF CONTENTS

1. INTRODUCTION
2. REVIEW OF UNCERTAINTY EVALUATION
 - 2.1 Method of SG-26 evaluation
 - 2.2 Method of JENDL-4.0 Evaluation
 - 2.2.1 Resonance Region
 - 2.2.2 Fission Cross Sections above Resonance Region
 - 2.2.3 CCONE Calculations
 - 2.2.4 ν_p and ν_d
3. REVIEW OF EXPERIMENTS
 - 3.1 Current Status of Capture Cross Section Measurements
 - 3.1.1 Prompt γ -ray detection method
 - 3.1.2 Post irradiation activation method
 - 3.1.3 Experimentalist's Review on Measurements of Capture Cross Section for ^{28}Si , ^{206}Pb , ^{241}Am , ^{238}U
 - 3.2 Current status on inelastic cross section measurements
 - 3.3 Experiments on prompt fission neutrons in inverse kinematics
4. RECOMMENDATION OF COLLABORATIVE PATH FORWARD TO MEET THE NEEDS
5. SUMMARY

Meeting Nuclear Data Needs for Advanced Reactor System

1. INTRODUCTION

The nuclear data needs for advanced reactor systems were identified by the activities of WPEC Subgroup 26 (SG-26) “Uncertainty and Target Accuracy Assessment for Innovative Systems Using Recent Covariance Data Evaluations” [1]. The target accuracy assessment was also performed in order to provide quantitatively improvement requirements of nuclear data by each isotope, nuclear reaction and energy range [1]. It was systematically shown that there are significant gaps between the current uncertainties and the target accuracies. To find correct paths to reduce the gaps and find correct paths to meet these nuclear data needs, the WPEC subgroup 31 “Meeting Nuclear Data Needs for Advanced Reactor Systems” was organized. To utilize the collective knowledge of the international nuclear data measurement and evaluation community, the members of the Subgroup 31 (SG-31) were comprised of nuclear data measurement experts, together with evaluation experts.

The uncertainties of nuclear data have been investigated and updated independently under different nuclear data evaluation projects, such as JENDL, JEFF, ENDF, BROND, and MANREAD. Since the accuracy requirements and priorities summarized in SG-26 are strongly depends on the assumed initial uncertainty data, the current evaluations on uncertainty in JENDL, JEFF, ENDF, BROND, and MANREAD are compared with those of SG-26 in Chapter 2 for some selected data. These comparisons recognize existence of discrepancy on accuracy estimation between different evaluations. In section 2.1, outline of SG-26 report are briefly reviewed together with summaries of target accuracies of nuclear data for each reactor type, such as PWR, VHTR, FR, ADMAB. In section 2.2, method of uncertainty evaluation in JENDL-4.0 is reviewed as an example to show current evaluation status on resonance region, unresolved resonance region, calculation using CCONE, and neutron emission multiplicity.

Current status of experimental techniques is reviewed in Chapter 3. Especially, measurement methods on capture cross sections were described in detail in Chapter 3.1, where noticeable advancements on measurement techniques and also possible origins on main uncertainty were identified. Experimentalist's review on measurements are also given on capture cross section for ^{28}Si , ^{206}Pb , ^{241}Am , ^{238}U . Current status on inelastic measurement is briefly described in Chapter 3.2. In Chapter 3.3, recent progress in fission experiments using inverse kinematics is summarized. Although the method is indirect measurement method, it is expected to have important contribution because the method can extend the diversity of fissioning systems.

Based on the reviews and discussions on uncertainty evaluations and recent experimental status, collaborative paths toward to meet the needs are discussed and recommended in Chapter 4. A summary is given in Chapter 5.

The appendix includes the Graphical Summary of Available Measured Data. In the CD in this report, all of the available measured data are included listed by SG-26 reports, and some of the selected data are in this report. This appendix graphically shows available data with their energy range and uncertainty for each nuclear data. This gives comprehensive information of experimental data together with their uncertainties.

References

[1] OECD/NEA WPEC Subgroup 26 Final Report: "Uncertainty and Target Accuracy Assessment for Innovative Systems Using Recent Covariance Data Evaluations", 2008.

2. REVIEW OF UNCERTAINTY EVALUATION

There are various advancements in developing the methodology of covariance evaluation for each nuclear data project, such as WPEC/SG-26, JENDL-4.0, JEFF-3.1.2, ENDF/VII-1, BROND-3, and MANREAD.

Each nuclear data project supplied their evaluated uncertainties for selected nuclear data for SG-31. The evaluated uncertainties were compared for some selected nuclear data in **Table 1**. Although almost all evaluated uncertainties agree within a factor of two, there are exceptions as an example of inelastic cross section for Fe-56. Two evaluation methods (GMA and SYS) in BROND-3 explain the half of the gap. However, JENDL-4.0 gives three times larger uncertainties even compared to BROND-3 (SYS). These discrepancies could be partly explained by reason that the number of experiments for each nuclear data is too limited, and furthermore there are discrepancies between existing experimental data in many cases; it makes statistical treatment difficult. Evaluators need to assume some values for uncertainties based on their own experiences and insights.

In section 2.1, outline of SG-26 report are briefly reviewed together with summaries of target accuracies of nuclear data for each reactor type, such as PWR, VHTR, FR, ADMAB.

In section 2.2, status of evaluation in JENDL-4.0 is briefly reviewed on resonance region, unresolved resonance region, calculation using CCONE, and neutron emission multiplicity.

Table 1 Comparison of accuracy (%) estimation between SG-26, JENDL-4.0, ENDF/B-VII.1, JEFF-3.1.2, and BROND-3

Nuclide	Quantity	Energy range	SG-26	JENDL-4	ENDF/B-VII.1	JEFF-3.1.2	BROND-3 GMA	BROND-3 SYS
Na-23	inel	0.498 - 2.23 MeV	25	17 - 20	8 - 16			7 - 8
Si-28	inel	1.35 - 6.07 MeV	14 - 50		2 - 4	~ 5		
Si-28	capt	6.07 - 19.6 MeV	53			~ 35		
Fe-56	inel	0.498 - 2.23 MeV	16 - 25	15 - 17	7 - 12	~ 2	2	5
Pb-206	inel	1.35 - 2.23 MeV	14		8			10
Pb-207	inel	0.498 - 2.23 MeV	11		9 - 17			10
U-238	inel	0.498 - 6.07 MeV	10 - 20	5 - 11	12 - 21			7
U-238	capt	2.03 - 24.8 keV	3 - 9	7 - 10	3		0.9 - 2.2	7
Pu-238	fiss	0.183 - 1.35 MeV	20	2.4 - 3.0	1.0 - 1.2			
Pu-239	capt	2.03 - 498 keV	7 - 15	5 - 9	7 - 15			4
Pu-240	fiss	0.498 - 1.35 MeV	6	0.8	0.9			1.4
Pu-240	ν	0.498 - 1.35 MeV	4	0.2	0.4			0.7
Pu-241	fiss	454 eV - 1.35 MeV	8 - 20	0.8 - 5				4
Pu-242	fiss	0.498 - 2.23 MeV	19 - 21	2				
Am-241	fiss	2.23 - 6.07 MeV	12	1.4	1.3			1.5
Am-242m	fiss	67.4 keV - 1.35 MeV	17	3	1.5			
Cm-244	fiss	0.498 - 1.35 MeV	50	4	3			3 - 7
Cm-245	fiss	67.4 - 183 keV	47	5	4			3

2.1 Method of SG-26 evaluation

The Working Party on Evaluation Cooperation (WPEC) of the OECD Nuclear Energy Agency Nuclear Science Committee established Subgroup 26 to develop a systematic approach to define data needs for advanced reactor systems and to make a comprehensive study of such needs for Generation-IV (Gen-IV) reactors. The subgroup was established at the end of 2005 [1].

A comprehensive sensitivity and uncertainty study was performed to evaluate the impact of neutron cross-section uncertainty on the most significant integral parameters related to the core and fuel cycle of a wide range of innovative systems, even beyond the Gen-IV range of systems. Results were obtained for the Advanced Breeder Test Reactor (ABTR), the Sodium-cooled Fast Reactor (SFR), the European Fast Reactor (EFR), the Gas-cooled Fast Reactor (GFR), the Lead-cooled Fast Reactor (LFR), the Accelerator Driven Minor Actinide Burner (ADMAB), the Very High Temperature Reactor (VHTR) and a Pressurized Water Reactor with extended burn up (PWR). These systems corresponded to current studies in the Generation-IV initiative, the Global Nuclear Energy Partnership (GNEP), the Advanced Fuel Cycle Initiative (AFCI), and in advanced fuel cycle and Partitioning/Transmutation studies in Japan and Europe.

The covariance data used in the uncertainty analysis was the BOLNA [2] matrix developed in a joint effort of several laboratories contributing to Subgroup 26. The analysis was mostly focused on integral parameter (like k_{eff} , reactivity coefficients, power distributions etc.) uncertainty due to neutron cross-section uncertainties. The integral parameters considered were related to the reactor core performances but also to some important fuel cycle-related parameters, like the transmutation potential, the doses in a waste repository or the neutron source at fuel fabrication.

A compilation of preliminary “Design Target Accuracies” was also put together and a target accuracy assessment was performed in order to provide an indicative quantitative evaluation of nuclear data improvement requirements by isotope, nuclear reaction and energy range, that would meet the Design target accuracies, as compiled in the study. First priorities were formulated on the basis of common needs for fast reactors and, separately, thermal systems.

For what concerns the relevance to the OECD/NEA WPEC Subgroup 31 TABLE I shows a summary of the results obtained over the whole set of fast reactors, and table II and III for the two thermal systems (VHTR and PWR).

Values are given as uncertainty ranges within selected energy intervals and only the most significant values are shown.

From table I several relevant features can be pointed out. Very tight requirements are shown for the σ_{inel} of U-238 (2-3%), Fe-56 (3-6%), Na-23 (4-10%) and even for Pb isotopes. The required accuracies are probably beyond achievable limits with current techniques. These results should be used with precaution. They indicate trends and general priority needs. In fact, these quantitative values have been obtained considering only diagonal (variance) uncertainty values that represent an underestimation of the real uncertainty. Moreover, and certainly more important, the accuracy requirements and priorities are strongly dependent on the assumed initial uncertainty variance-covariance data, in particular in this case the BOLNA matrix. More recent covariance matrix data are now available, and this type of study should be repeated in order to update the results.

In fact more recent studies [3, 4] use a more general method since it accounts for energy correlations in the definition of target accuracies [4] and are based on a very recent release of new covariance data: COMMARA 2.0 [5]. The study of ref [3] was carried out only for minor actinide burners (SFR, and ADMAB). In order to have a more compact way to assess the target accuracy a 7 group energy structure was adopted for this study. The 7 energy group structure is shown in table IV. The target accuracies, obtained with the method of Reference [4], on neutron cross sections needed to meet the design requirements (in this particular case $\pm 0.3\% \Delta k/k$ of uncertainty for the k_{eff}), are given in tables V and VI. The requirements are shown according (rank) to their overall original contribution to the total uncertainty on k_{eff} .

The requirements indicated by this study, and in particular for MA, are very small and quite improbable to be attained with current experimental techniques for differential measurements. The results of this investigation suggest that a careful analysis is still needed in order to define the most appropriate and effective strategy for data uncertainty reduction. Besides a further consolidation of the present covariance data libraries, a strategy of combined use of differential and integral measurements should be further pursued in order to meet future requirements, as proposed, e.g., in Ref. [6].

References

[1] OECD/NEA WPEC Subgroup 26 Final Report: Uncertainty and Target Accuracy Assessment for Innovative Systems Using Recent Covariance Data Evaluations”, 2008.

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Table I. SG26 Summary Target Accuracies for Fast Reactors.

Isotope	Cross-Section	Energy Range	Current Accuracy (%)	Target Accuracy (%)
U238	σ_{inel}	6.07 \div 0.498 MeV	10 \div 20	2 \div 3
	σ_{capt}	24.8 \div 2.04 keV	3 \div 9	1.5 \div 2
Pu241	σ_{fiss}	1.35MeV \div 454eV	8 \div 20	2 \div 3 (SFR,GFR,LFR) 5 \div 8 (ABTR,EFR)
Pu239	σ_{capt}	498 \div 2.04keV	7 \div 15	4 \div 7
Pu240	σ_{fiss}	1.35 \div 0.498 MeV	6	1.5 \div 2
	ν	1.35 \div 0.498 MeV	4	1 \div 3
Pu242	σ_{fiss}	2.23 \div 0.498 MeV	19 \div 21	3 \div 5
Pu238	σ_{fiss}	1.35 \div 0.183 MeV	17	3 \div 5
Am242m	σ_{fiss}	1.35MeV \div 67.4keV	17	3 \div 4
Am241	σ_{fiss}	6.07 \div 2.23 MeV	12	3
Cm244	σ_{fiss}	1.35 \div 0.498 MeV	50	5
Cm245	σ_{fiss}	183 \div 67.4keV	47	7
Fe56	σ_{inel}	2.23 \div 0.498 MeV	16 \div 25	3 \div 6
Na23	σ_{inel}	1.35 \div 0.498 MeV	28	4 \div 10
Pb206	σ_{inel}	2.23 \div 1.35 MeV	14	3
Pb207	σ_{inel}	1.35 \div 0.498 MeV	11	3
Si28	σ_{inel}	6.07 \div 1.35 MeV	14 \div 50	3 \div 6
	σ_{capt}	19.6 \div 6.07 MeV	53	6

Table II. SG26 VHTR: Uncertainty Reduction Requirements Needed to Meet Integral Parameter Target Accuracies

Isotope	Cross-Section	Energy Range	Uncertainty (%)	
			Initial	Required ($\lambda=1$)
U238	σ_{capt}	454 - 22.6 eV	1.7	1.2
C	σ_{inel}	19.6 - 6.07 MeV	30.0	7.1
C	σ_{capt}	19.6 - 6.07 MeV	20.0	7.1
		4 - 0.54 eV	20.0	5.0
Pu239	σ_{capt}	0.54eV - 0.1eV	1.4	0.9
Pu241	σ_{fiss}	454 - 22.6 eV	19.4	6.4
		4 - 0.54 eV	26.8	9.4
		0.54eV - 0.1eV	2.9	1.5
Pu241	σ_{capt}	0.54eV - 0.1eV	6.8	2.4

Table III. SG26 PWR: Uncertainty Reduction Requirements Needed to Meet Integral Parameter Target Accuracies

Isotope	Cross-Section	Energy Range	Uncertainty (%)	
			Initial	Required ($\lambda=1$)
O	σ_{capt}	19.6 - 6.07 MeV	100.0	12.1
		6.07 - 2.23 MeV	100.0	9.9
Pu241	σ_{fiss}	454 - 22.6 eV	19.4	4.7
		4 - 0.54 eV	26.8	7.7
		0.54eV - 0.1eV	2.9	1.7
		0.1eV - thermal	3.3	1.9
Pu239	σ_{capt}	0.54eV - 0.1eV	1.4	1.0
U238	σ_{capt}	24.8 - 9.12 keV	9.4	4.6
		454 - 22.6 eV	1.7	1.4
U238	σ_{inel}	6.07 - 2.23 MeV	14.6	5.1
Pu241	σ_{capt}	0.54eV - 0.1eV	6.8	3.0
Pu240	σ_{capt}	0.1eV - thermal	4.8	3.1
O	σ_{inel}	6.07 - 2.23 MeV	54.9	12.6
		19.6 - 6.07 MeV	84.6	15.6

Table IV. 7 energy groups structure (eV).

Group	Upper Energy	Group	Upper Energy
1	1.96403 10⁷	5	2.03468 10³
2	2.23130 10⁶	6	2.26033 10¹
3	4.97871 10⁵	7	5.40000 10⁻¹
4	6.73795 10⁴		

TABLE V. Nuclear data target accuracy requirements (first 20 major contributors) on a total uncertainty of 300 pcm on K_{eff} for the SFR system

Rank	Cross Sect.	Energy Range (eV)	Current (%)	Required (%)
1	^{240}Pu v Gr. 2	2.23×10^6 to 4.98×10^5	3.2	0.9
2	^{245}Cm σ^{fis} Gr. 3	4.98×10^5 to 6.74×10^4	47.0	3.2
3	^{56}Fe σ^{inel} Gr. 2	2.23×10^6 to 4.98×10^5	11.4	1.8
4	^{56}Fe σ^{el} Gr. 3	4.98×10^5 to 6.74×10^4	8.9	1.7
5	^{245}Cm σ^{fis} Gr. 2	2.23×10^6 to 4.98×10^5	47.4	4.5
6	^{240}Pu v Gr. 3	4.98×10^5 to 6.74×10^4	4.8	1.5
7	^{240}Pu σ^{fis} Gr. 2	2.23×10^6 to 4.98×10^5	2.5	1.1
8	^{238}U σ^{inel} Gr. 2	2.23×10^6 to 4.98×10^5	16.7	2.8
9	^{245}Cm σ^{fis} Gr. 4	6.74×10^4 to 2.03×10^3	15.9	3.7
10	^{56}Fe σ^{cap} Gr. 3	4.98×10^5 to 6.74×10^4	12.2	2.7
11	^{56}Fe σ^{el} Gr. 2	2.23×10^6 to 4.98×10^5	5.8	2.1
12	^{240}Pu v Gr. 1	1.96×10^7 to 2.23×10^6	2.5	1.5
13	^{238}U σ^{inel} Gr. 1	1.96×10^7 to 2.23×10^6	19.4	3.7
14	^{240}Pu σ^{fis} Gr. 3	4.98×10^5 to 6.74×10^4	6.2	2.0
15	^{240}Pu σ^{cap} Gr. 2	2.23×10^6 to 4.98×10^5	33.4	4.1
16	^{242}Pu σ^{cap} Gr. 4	6.74×10^4 to 2.03×10^3	20.2	3.9
17	^{244}Cm σ^{fis} Gr. 2	2.23×10^6 to 4.98×10^5	14.4	3.2
18	^{242}Pu σ^{cap} Gr. 3	4.98×10^5 to 6.74×10^4	28.2	4.6
19	^{240}Pu σ^{cap} Gr. 4	6.74×10^4 to 2.03×10^3	4.5	2.1
20	^{240}Pu v Gr. 4	6.74×10^4 to 2.03×10^3	4.6	2.2

TABLE VI. Nuclear data target accuracy requirements (first 20 major contributors) on a total uncertainty of 300 pcm on K_{eff} for the ADMAB system

Rank	Cross Sect.	Energy Range (eV)	Current (%)	Required (%)
1	$^{245}\text{Cm } \sigma^{\text{fis}} \text{ Gr. 3}$	4.98×10^5 to 6.74×10^4	47.0	1.5
2	$^{244}\text{Cm } \sigma^{\text{fis}} \text{ Gr. 2}$	2.23×10^6 to 4.98×10^5	14.3	1.0
3	$^{245}\text{Cm } \sigma^{\text{fis}} \text{ Gr. 2}$	2.23×10^6 to 4.98×10^5	47.5	2.0
4	$^{244}\text{Cm } \nu \text{ Gr. 2}$	2.23×10^6 to 4.98×10^5	8.7	0.9
5	$^{244}\text{Cm } \sigma^{\text{cap}} \text{ Gr. 4}$	6.74×10^4 to 2.03×10^3	64.4	2.5
6	$^{243}\text{Am } \sigma^{\text{fis}} \text{ Gr. 2}$	2.23×10^6 to 4.98×10^5	8.8	1.0
7	$^{245}\text{Cm } \sigma^{\text{fis}} \text{ Gr. 4}$	6.74×10^4 to 2.03×10^3	14.7	1.6
8	$^{244}\text{Cm } \sigma^{\text{cap}} \text{ Gr. 3}$	4.98×10^5 to 6.74×10^4	48.6	2.7
9	$^{244}\text{Cm } \sigma^{\text{fis}} \text{ Gr. 1}$	1.96×10^7 to 2.23×10^6	19.7	1.8
10	$^{243}\text{Am } \sigma^{\text{inel}} \text{ Gr. 2}$	2.23×10^6 to 4.98×10^5	21.6	2.0
11	$^{243}\text{Am } \sigma^{\text{cap}} \text{ Gr. 2}$	2.23×10^6 to 4.98×10^5	6.8	1.2
12	$^{243}\text{Am } \sigma^{\text{fis}} \text{ Gr. 1}$	1.96×10^7 to 2.23×10^6	12.1	1.6
13	$^{243}\text{Am } \sigma^{\text{cap}} \text{ Gr. 3}$	4.98×10^5 to 6.74×10^4	8.2	1.3
14	$^{244}\text{Cm } \sigma^{\text{cap}} \text{ Gr. 2}$	2.23×10^6 to 4.98×10^5	77.9	4.3
15	$^{244}\text{Cm } \nu \text{ Gr. 1}$	1.96×10^7 to 2.23×10^6	10.1	1.7
16	$^{244}\text{Cm } \sigma^{\text{fis}} \text{ Gr. 3}$	4.98×10^5 to 6.74×10^4	10.0	2.0
17	$^{209}\text{Bi } \sigma^{\text{inel}} \text{ Gr. 2}$	2.23×10^6 to 4.98×10^5	14.0	1.9
18	$^{15}\text{N } \sigma^{\text{elas}} \text{ Gr. 2}$	2.23×10^6 to 4.98×10^5	2.9	0.9
19	$^{245}\text{Cm } \sigma^{\text{fis}} \text{ Gr. 1}$	1.96×10^7 to 2.23×10^6	38.0	4.1
20	$^{244}\text{Cm } \nu \text{ Gr. 3}$	4.98×10^5 to 6.74×10^4	10.0	1.9

2.2 Method of JENDL-4.0 Evaluation

In JENDL-4.0 [Sh11], covariances are given for 95 nuclides, *i.e.*, $^{10,11}\text{B}$, $^{14,15}\text{N}$, ^{16}O , ^{23}Na , ^{48}Ti , $^{52,53}\text{Cr}$, ^{55}Mn , ^{56}Fe , ^{59}Co , $^{58,60}\text{Ni}$, ^{90}Zr , ^{209}Bi , and all actinoids. Except for the actinoids, most of the covariances were taken from JENDL-3.3 [Sh02] or an additional work [Sh07]. Updates were made to ^{23}Na and ^{56}Fe by considering the difference between the evaluated and experimental data. Only the variances of the (n,p) cross section were provided for ^{14}N in the limited energy range. Covariances of the RRP of $^{52,53}\text{Cr}$ and ^{55}Mn were taken from the resonance analyses.[Le10, De10] The rest of this section is devoted to the covariance estimation of actinoids.

2.2.1 Resonance Region

Covariances are given only for RRP. Covariances of URPs are not produced, since URPs are used solely for self-shielding calculations. In some cases, the cross-section covariances calculated from the covariances of RRP are given in order to save file space. For ^{232}Th , $^{233,235,238}\text{U}$ and $^{239,241}\text{Pu}$, the RRP evaluated by the ORNL group with the SAMMY code [La06] were adopted. For $^{233,235,238}\text{U}$ and ^{239}Pu , the covariances of RRP obtained [Le09] with the SAMMY code were partly adopted. However, after the release of JENDL-4.0 in 2010, it was realized that such truncation of covariance matrices led to incorrect cross-section covariances due to strong long-range correlations of the ORNL results. Therefore, updates were made [Ja12] to the resonance region for those nuclides by considering the full matrices. As a result, the MF32 data were completely replaced with the MF33 data, since it takes a very long time to process such full matrices with the NJOY code [Ma99]. **Figure 1** shows the covariance matrix of ^{235}U fission cross sections calculated from RRP covariances. For ^{232}Th and ^{241}Pu , no covariances were available. Therefore, roughly assumed cross-section variances were given for both nuclides in the resonance region.

For $^{236,238}\text{Np}$, the covariances of RRP were taken from the SAMMY calculations. For many other isotopes, only variances of the RRP were given. The variances were taken from the recommendation of Mughabghab [Mu06] and/or from the references on which the parameters were given. If no information was available, uncertainties were assumed tentatively: 0.1 % for resonance energies, 10 - 50 % for resonance widths. In some cases, further cross-section uncertainties were considered to compensate for too small uncertainties calculated from the covariances of RRP by considering the spread of experimental data or the difference between experimental and evaluated data.

2.2.2 Fission Cross Sections above Resonance Region

For $^{233,235,238}\text{U}$ and $^{239,240,241}\text{Pu}$, the covariance matrices were obtained together with the fission cross sections by the simultaneous evaluation. For example, **Fig. 2** shows the uncertainties in the fission cross sections of ^{239}Pu obtained. Since so many experimental data points were considered without correlation among different data sets, the variances obtained were found to be very small. Therefore, the standard deviations were multiplied by a factor of 2.0, and then adopted in JENDL-4.0. This modification might compensate for the neglected correlation.

For other 24 nuclides with many experimental data, the GMA code [Po81] was used to evaluate the fission cross sections and their covariances. The standard deviations obtained were compared with experimental data. If the standard deviations were too small, a suitable scaling factor was applied to them. An example is shown in **Fig. 3**.

2.2.3 CCONE Calculations

For the data evaluated using the CCONE code [Iw07], covariance matrices were estimated using the KALMAN code [Kaw97]. Sensitivities to the model parameters were calculated by the CCONE code using the same

parameter sets as the cross section evaluations. About 50 parameters were used for the KALMAN calculation. The parameters used are those of optical model, level density, fission barrier, γ -ray strength function, and exciton model.

For the capture cross sections of ^{232}Th , $^{233-236}\text{U}$, ^{237}Np and $^{241,243}\text{Am}$, covariance matrices were evaluated using the experimental data sets considered in the cross section evaluations. The statistical and systematical uncertainties of the experimental data were used in the covariance estimations. For the systematical uncertainties, the correlations between the data in the same data set were set to 1.0 and those in the different data sets were assumed 0.8 and 0.4 for the same and different authors, respectively. The experimental uncertainties were modified so that the reduced χ^2 should be close to unity. The difference between the prior cross sections, which were calculated with the CCONE code, and the posterior KALMAN estimates was added to the posterior covariances obtained from KALMAN. This was required, since the covariance estimation was performed after the JENDL-4.0 cross sections were fixed. **Figure 4** shows the capture cross section of ^{237}Np . The evaluated uncertainties are shown by the shaded area. The data of JENDL-3.3 and ENDF/B-VII.0 agree with the JENDL-4.0 within the uncertainty in the lower energy region. The agreement among them becomes worse in the upper energy region, where the experimental data are not available.

For other reaction cross sections, the covariances were evaluated by a simpler procedure. For the cross sections for which experimental data were available, the cross section uncertainties were estimated from the experimental data at selected several energy points. Using the estimated uncertainties, the covariance matrices were calculated using the KALMAN code. **Figure 5** shows the $^{238}\text{U}(n,2n)$ reaction cross section with the evaluated uncertainties. The evaluated uncertainties are around 10 % except for energies near the threshold. For this evaluation, 10 % statistical and systematical uncertainties at 10 and 14 MeV were applied in the KALMAN calculation.

For the data for which no experimental data were available, the covariances were obtained using parameter uncertainties assumed. The parameter uncertainties were estimated from available experimental data on

neighboring nuclides. In most cases, however, the uncertainties thus obtained were increased by a factor of two when they were applied to a reaction for which experimental data were unavailable. A typical uncertainty was several % depending on the sensitivity of the cross section to the parameter.

2.2.4 ν_p and ν_d

For the nuclides having enough experimental data, the ν_p value and its covariance were determined by the least-squares fitting to the experimental data. In some cases, a suitable scaling factor was applied to the variances obtained when they were too small. If no experimental data were available, the systematics recommended by Howerton [Ho77] or Ohsawa [Oh08] were applied to estimate ν_p values. In such cases, the covariances were estimated by fitting a linear function to two data points at 0 MeV and 5 MeV calculated from the systematics. Their uncertainties were assumed to be 5 % for the nuclides lighter than Cm and 10 % for the nuclides heavier than Bk. If possible, the uncertainties were determined from experimental data at the thermal energy.

For the ν_d values, only the variances were given except for $^{235,238}\text{U}$ and ^{239}Pu for which the JENDL-3.3 covariances obtained by the GMA analyses were adopted. Variances were estimated from experimental data if existed, or a standard deviation of 15 % was assumed if no experimental data were available at all.

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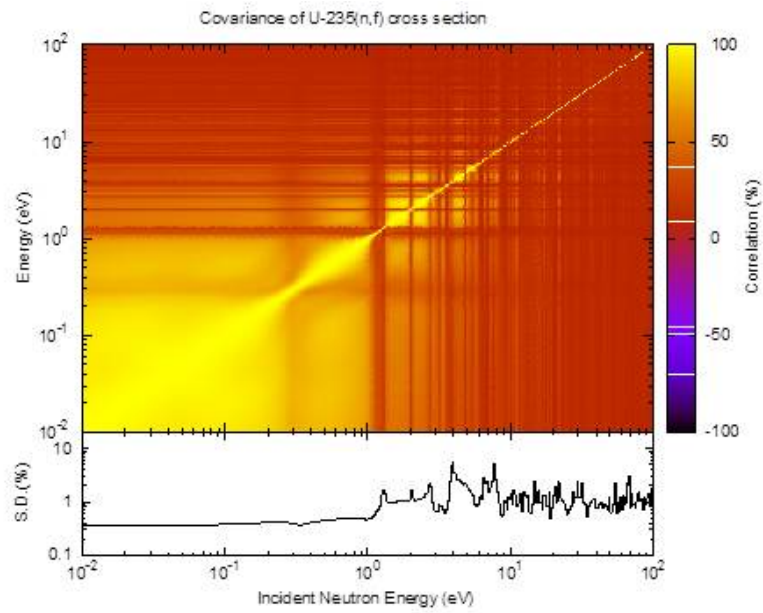


Fig. 1 Covariance matrix of ^{235}U fission cross section

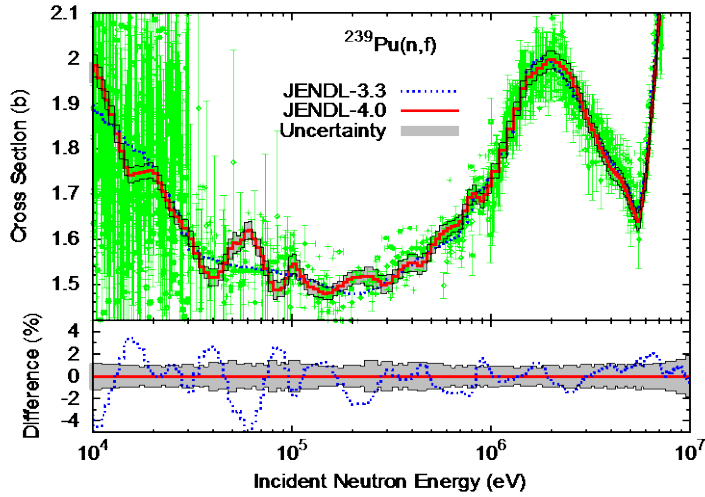


Fig. 2 Fission cross sections of ^{239}Pu . Lower panel shows relative difference of cross sections between JENDL-3.3 and JENDL-4.0.

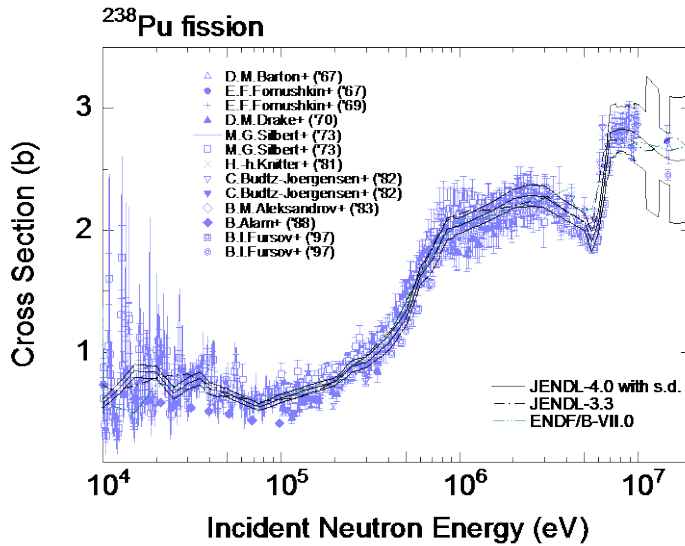


Fig. 3 Fission cross sections of ^{238}Pu . The thin solid lines are the standard deviations of JENDL-4.0.

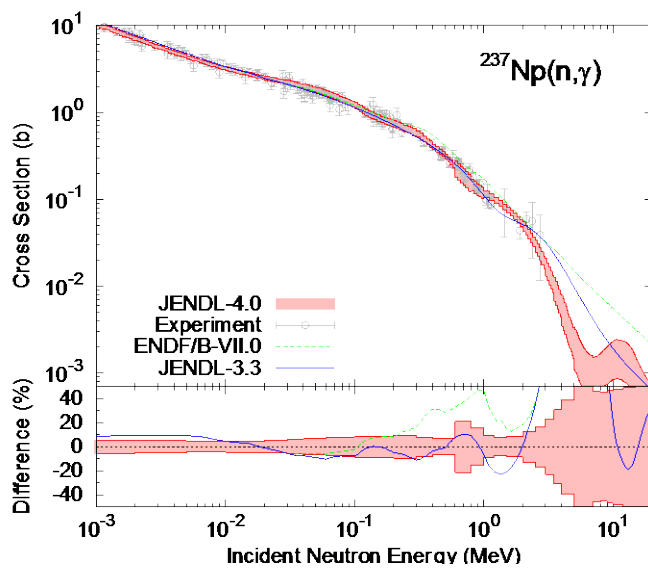


Fig. 4 Capture cross section of ^{237}Np

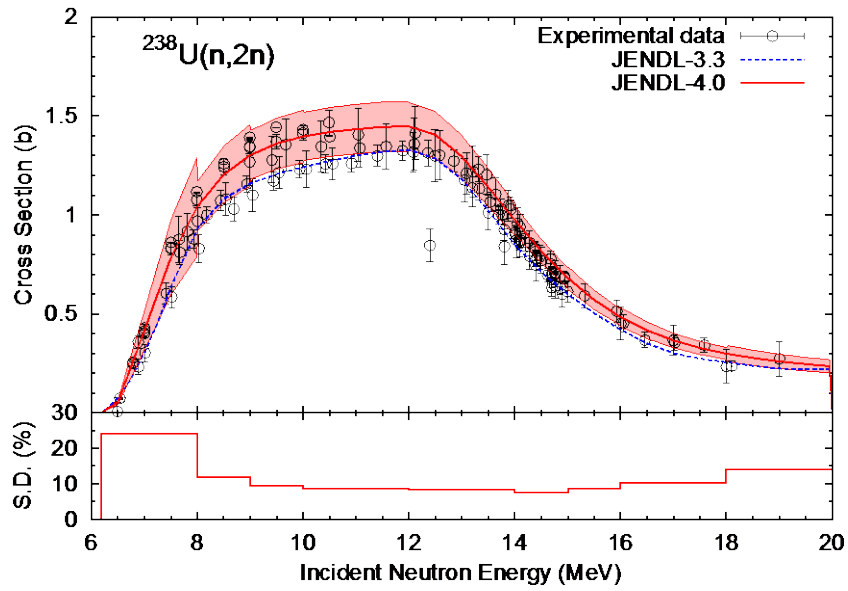


Fig. 5 $^{238}\text{U}(n,2n)$ reaction cross section. In the upper panel, the evaluated uncertainty is shown by a shaded area. The lower panel shows the relative standard deviation of JENDL-4.0.

3. REVIEW OF EXPERIMENTS

Recent advancements on neutron cross section measurements are reviewed. There are noticeable advancements in the experimental techniques, such as utilization of a high intensity spallation neutron beam, and elaborated analysis techniques. Current status on capture cross section measurements is reviewed in Chapter 3.1, and review by experimentalists on the measurements of some selected nuclear data is given in Chapter 3.1.3. Current status on inelastic cross section measurements is briefly reviewed in Chapter 3.2. New experimental approaches using inverse kinematics methods are also described in Chapter 3.3 because of their potential impact especially on fission data for unstable nuclei.

3.1 Current Status of Capture Cross Section Measurements

Neutron induced capture cross section measurements rely either on post-irradiation activation analysis or on the detection of prompt γ -rays emitted in the (n,γ) reaction. To cover a broad energy region, measurements can be performed at a white neutron source applying the time-of-flight (TOF) technique or alternatively by using quasi-mono-energetic neutrons produced by charged particle induced reactions. The choice of the principle and related detection system depends on the reaction to be studied, the energy region of interest, the amount of available sample material and the required accuracy and resolution. Capture cross sections in the resonance region are best derived from results of experiments with a prompt γ -ray detection system that is optimized for TOF measurements. Post-irradiation activation analysis is suited to determine capture cross sections in the continuum region, thermal cross sections and resonance integrals. Recently, the activation method has also been applied to determine cross sections for fast reactor neutrons.

3.1.1 Prompt γ -ray detection method

A prompt γ -ray detection system optimized for TOF-measurements full-fills the following requirements [1, 2]

- the detection efficiency for a capture event is independent of the γ -ray cascade, i.e. independent of the multiplicity of the γ -ray spectrum and the γ -ray energy distribution;
- the sensitivity to neutrons scattered by the sample is low compared to the sensitivity to γ -rays produced by the capture reaction in the sample;
- the detector has a sufficient time resolution;
- for the study of a fissioning nucleus, the γ -rays from neutron capture can be separated from those resulting from neutron fission; and
- in case of a radioactive sample, the prompt γ -rays can be separated from the γ -rays emitted due to the radioactive decay.

Three different principles based on the direct detection of prompt γ -rays can be distinguished: (1) γ -ray spectroscopic (GS), (2) total γ -ray absorption (TA) and (3) total energy detection principle (TE). The main uncertainty for the three principles is related to the normalization of the data and the determination of the background [2].

(1) Capture cross sections based on γ -ray spectroscopic measurements with high resolution γ -ray detectors [3, 4] can be derived from

- the sum of all the partial cross sections of primary transitions depopulating the capture state (GS1);
- the sum of the partial capture cross sections of the transitions feeding the ground state (GS2); or
- the sum of all the observed partial cross sections weighted with the energy of the transition divided by the total γ -ray energy liberated in the capture event (GS3).

The accuracy strongly depends on the complexity of the level scheme of the compound nucleus. They can be determined accurately when the γ -ray transitions of the cascade are well known. Therefore, γ -ray spectroscopic methods are very powerful to determine capture cross section data for light

nuclei or for nuclei with a proton and neutron number close to a magic shell [3]. In case not all γ -ray transitions can be determined the results are biased and only lower limits can be derived [4]. To verify the impact of missing transitions, the principle of γ -ray intensity balance [5] or crossing intensity sum [6] can be applied. The missing contributions can also be based on statistical models to simulate the full γ -ray cascade. Codes that can be used are e.g. DICEBOX [7], DECAYGEN [8] and γ DEX [9]. The γ -ray cascade simulations rely on nuclear level statistical models and nuclear data input (low-lying level scheme, average radiation widths, level densities). The accuracy of the cross section depends on the statistical nature of the γ -ray cascade.

(2) The total γ -ray absorption principle relies on the detection of the energy sum of the γ -rays emitted in a capture event. An ideal detector has a 4π geometry and a 100 % absolute detection efficiency allowing for the detection of the entire electromagnetic cascade. Thus, the energy deposited in the detector is directly proportional to the total energy available in the capture event and independent of the γ -ray cascade.

The first total absorption detectors were large liquid organic scintillation tanks [10,11]. The uncertainty of these systems is limited to 5 % - 10 % and depends on the reaction under study. The limitation is primarily due to corrections that are required to estimate the efficiency to detect a capture event [2, 11]. Organic liquid scintillators (OLS) have extensively been used to determine capture cross sections of fissile material. To separate capture events from fission events different methods have been applied. Some of them using an additional fission chamber in parallel or as an additional measurement to determine correction factors. An extensive list of capture-to-fission ratio measurements for $^{233, 235}\text{U}$ and ^{239}Pu is given in Ref. [2].

Nowadays inorganic detectors are used, which are smaller in size and have a better efficiency. An overview of system that are in use is given in Ref. [2].

Therefore their sensitivity to the ambient background is reduced. However, they still suffer from neutron sensitivity due to (n,γ) reactions in the detection material. Therefore, they are limited to measurements in the resolved resonance region and for nuclei with small scattering to capture ratios. The final accuracy of such systems depends strongly on the reaction under study. Since an ideal detector with a 100 % γ -ray detection efficiency does not exist, a correction is needed when the normalization is performed using a capture reaction which has a different γ -ray cascade from the reaction under study. Such a correction becomes even more important when a constraint is imposed on the multiplicity and energy deposition to reduce the background and when the γ -ray cascade changes from resonance to resonance. Due to an improved understanding of the measurement equipment and techniques through Monte Carlo simulations the detection efficiencies can be determined with better accuracies. However, the final accuracy depends strongly on the statistical nature of the γ -ray cascade, as in the case of organic scintillators. These detectors can also be used to derive capture-to-fission ratio for fissile materials by [12,13].

(3) When the contribution of the fission channel can be neglected, the most accurate capture cross section data can be measured by applying the total energy detection principle (TE) using C_6D_6 detectors combined with the pulse height weighting technique (PHWT). The application of the total energy detection principle requires a γ -ray detector with a relatively low γ -ray detection efficiency which is proportional to the γ -ray energy. Under these conditions the efficiency to detect a capture event is directly proportional to the sum of the energies of the γ -rays emitted in the cascade. This makes the efficiency independent of the γ -ray cascade.

The Moxon-Rae detector achieves approximately the proportionality between the γ -ray energy and detection efficiency by a special design of the detector [14]. However, uncertainties due to imperfect linearity between the detection efficiency and the γ -ray energy are at least 5 % [15,16].

Correction factors in case of the total energy detection principle combined with PHWT are limited compared with all the other principles (GS or TA). This has a strong impact on the accuracy that can be reached. An experimental validation of the total energy detection principle combined with the PHWT for C_6F_6 detectors was performed by Yamamuro et al [17]. Normalization factors derived from the saturated resonances 4.3 eV in ^{181}Ta , 4.9 eV in ^{197}Au and 5.2 eV in ^{109}Ag , were consistent within 2 %. A more extensive performance assessment for a C_6D_6 based system has been carried at the GELINA facility of the EC-JRC-IRMM [2,18,19]. The results in [2,18,19] demonstrate that capture yields with uncertainties better than 2 % can be deduced from thermal energy up the URR when the total energy detection principle in combination with the PHWT is applied. However, such a low uncertainty can only be reached under specific constraints [2].

3.1.2 Post irradiation activation method

Activation method has been widely used to determine thermal neutron capture cross sections using thermal reactor neutrons. The energy dependent capture cross sections measured by neutron time-of-flight technique has been sometimes normalized at the thermal cross sections determined by activation method. Therefore, this technique has important rule on determining neutron capture cross sections and their uncertainties. Recently, activation method using fast reactor neutrons was applied to deduce fast neutron capture cross sections in the case of ^{237}Np [74]. Activation method using accelerator based quasi mono-energetic neutrons was also widely used to determine 25-keV Maxwell averaged cross sections and 14 MeV cross sections. Activation method is in principle independent measurement method of prompt γ -ray detection method. Therefore, cross-check of data measured by these two independent methods is valuable to find an unknown origin of systematic errors.

For measurements of thermal neutron capture cross sections and resonance integrals, the well-known formulation, called as Westcott formulation, has

been used when well moderated reactor neutrons are used for activation. In this case, neutron flux distribution is approximated by Maxwell distribution plus $1/E$ (inverse of neutron energy) components. The effective capture cross section σ is expressed in this case as,

$$\sigma = \sigma_0(g + rs), \quad (1)$$

where σ_0 is the cross section for 2,200 m/sec neutrons and g and s are functions of the temperature T depending on the departure of the cross-section law from the $1/v$ form (for a $1/v$ law, $g=1$ and $s=0$). The r is an epithermal index represents the relative strength of the epithermal $1/E$ component. The s is a quantity proportional to the reduced resonance integrals, that is, resonance integral minus $1/v$ component. By irradiating a sample in two different neutron fields, for example, irradiating it with and without a Cd cover sheet, both quantities σ_0 and resonance integrals are deduced.

The g is called as a Westcott factor that represents the departure of the cross-section law from the $1/v$ form. To deduce the g , energy dependence of capture cross section measured by a TOF method is required. Recent TOF measurements with thin samples are expected to give an accurate g -factor value, since the correction factors for neutron shields and multiple scatterings are small. **Other systematic uncertainties need to be corrected appropriately, such as dead-time and background corrections [73].** When the thermal capture cross section is used for normalization of TOF data, simultaneous analyses of both kinds of data are desired.

In case of an activation method, high neutron flux of reactor can be used for irradiation. Using this advantage, double capture reactions [75] and even triple capture reactions [76] have been utilized to deduce capture cross sections.

In case that neutron capture product nucleus emits gamma rays, Ge detectors have been currently used to determine the amount of product nucleus. The reliability of gamma-ray emission probability is a possible origin of the unrecognized uncertainty, and should be carefully checked. The situation of

reliability on emission probability looks like that on neutron capture cross section [20]. Therefore, precise measurement techniques [77] of gamma-ray emission probability are also important to obtain accurate capture cross sections.

In case that neutron capture product nucleus does not emit gamma rays, mass spectrometry [78] has been used to determine the amount of product nucleus. The method has also been used to determine an amount of sample in a measurement of the thermal neutron capture cross section of radioactive ^{135}Cs [79].

The chemical purity of the sample is the other possible origin of the unrecognized uncertainty. For example, an impurity of ^{238}U was severely investigated in measurement of neutron capture cross section of ^{238}Nu using double neutron capture reaction [75] using a ^{237}Np sample, since effect of contamination of ^{238}U in the sample is significant; ^{238}U is transmuted to ^{239}Np via one neutron capture reaction and a β decay process.

Post-irradiation activation analysis has also been done at quasi-mono energetic neutron beams produced by charged particle induced reactions, where the neutron energy can be controlled by changing the energy of the interacting charged particle and/or the emerging angle. The cross section can be derived from a measurement of the residual activity by γ -ray spectroscopy or by accelerator mass spectrometry (AMS) [21].

3.1.3 Experimentalist's Review on Measurements of Capture Cross Section for ^{28}Si , ^{206}Pb , ^{241}Am , ^{238}U

In this section documented cross section data which can be used to evaluate capture cross sections for ^{28}Si , ^{206}Pb , ^{241}Am and ^{238}U from thermal up to the unresolved resonance region (URR) are discussed. In the search for data results of transmission measurements have also been considered. In the resolved

resonance region (RRR) reliable resonance parameters can only be derived when transmission data are available [2, 22]. Total cross section data provide also important prior information to improve the accuracy of capture cross sections in the URR as shown by Sirakov et al. [23]. Therefore, transmission data are needed to perform a consistent evaluation of the capture cross section in both the RRR and URR with uncertainties that are requested in the conclusions of SG-26.

^{28}Si

Experimental data that can be used for an evaluation of the thermal capture cross section of ^{28}Si are summarized in Table 1. The value recommended by Raman et al. [3] is fully consistent with the cross section derived from the three γ -spectroscopic methods mentioned in the introduction of section 3.1. This cross section is relative to the (332.6 ± 0.6) mb capture cross section in ^1H determined by Cokinos and Melkonian [27]. The capture cross section reported by Islam et al. [26] deviates by more than 20 %. This is partly due to the reference value for $^{14}\text{N}(n,\gamma)$ used in Ref. [26], which is 15% higher compared to the value 68.77 (0.56) mb reported by Belgya [6]. For a full consistent evaluation the coherent scattering length $b_c = 4.106$ (0.006) fm recommended by Koester et al. [28] can also be used.

$\sigma(n_{\text{th}},\gamma)$	Ref				Reference (n, γ)			
Pomerance	24	81	(24)	mb	$^{197}\text{Au}(n, \gamma)$	95		b
Spits and De Boer	25	156	(23)	mb	Al(n, γ)	239	(3.0)	mb
Spits and De Boer	25	163	(57)	mb	Na(n, γ)	534	(5.0)	mb

Spits and De Boer	25	166	(33)	mb	Mn(n, γ)	13.3	(0.2)	b
Islam et al.	26	207	(4)	mb	$^{14}\text{N}(n,\gamma)$	79.8	(1.4)	mb
Raman et al.	3	169	(4)	mb	$^1\text{H}(n,\gamma)$	332.6	(0.6)	mb
GS1		169.4	(3.9)	mb				
GS2		168.0	(3.7)	mb				
GS3		168.1	(3.9)	mb				

Table 1 Results of capture cross section measurements for ^{28}Si at 0.0253 eV.

Capture and transmission measurements have been carried out at GELINA at a 130 m and 400 m flight path, respectively [29]. For the capture measurements a γ -spectroscopic detection system based on BGO-detectors was used. The capture data were normalized to the 1.15 keV resonance of ^{56}Fe . However, no further details about the normalization procedure, e.g. the partial radiation width used for the normalization, were specified. Partial capture cross sections for transitions to the ground state and the first and third excited state have been determined. Resonance areas for ^{28}Si are given for resonance energies up to 4638 keV. Unfortunately the experimental transmission and yields are not available in numerical data. Documented TOF-data resulting from total and capture measurements at ORELA are available in numerical form [30-34]. Transmission measurements on natural samples with different thickness have been carried out at a 47-m, 80-m and 200-m flight path [30-33]. Results of capture measurements at a 40-m station have been reported by Guber et al [34]. The total energy detection principle in combination with the PHWT using C_6D_6 detectors was applied. From these data capture areas and peak cross sections for neutron energies < 700 keV can be deduced with an accuracy < 5 %. However, this level of accuracy can not be reached for the capture cross section between resonances. This is due to the contribution of direct capture and/or possible interference effects.

The ORELA data together with the transmission data of Adib et al. [35] have been used in a resonance shape analysis by Derien et al [36]. In the analysis the impact of a direct capture contribution was also considered. The analysis was based on a Reich-Moore approximation of the R-matrix theory. Hence, interference effects for the capture channel have been neglected. The resulting thermal capture cross sections is consistent with the one recommended by Raman et al. [3]. However, the coherent scattering length is about 5 % smaller compared to the one recommended by Koester et al. [28]. Although the above mentioned data on ^{28}Si and the evaluation of Derien et al. [36] do not cover the energy region of interest in Table 32 of SG-26 (i.e. 6 MeV – 20 MeV), they provide essential data to produce consistent capture cross sections in the high energy region.

^{206}Pb

Experimental capture cross sections at thermal are: $\sigma(n_{\text{th}},\gamma) = 25.5$ (5.0) mb [37], 30.5 (0.7) mb [38], 26.6 (1.2) mb [39] and 28.7 (0.9) mb [40]. These values together with the coherent scattering lengths $b_c = 9.22$ (0.07) fm determined by Ioffe et al. [41] and $b_c = 9.23$ (0.05) Koester and Knopf [42] can be used to derive consistent capture and scattering cross sections at thermal energy.

Transmission measurements for neutron energies between 1 keV and 900 keV have been carried out at a 78 m and 200m station of ORLEA using radiogenic lead samples enriched to 884 % in ^{206}Pb [43-45]. In Ref. 43 capture cross section measurements using Ge-detectors at a 40m station of ORELA are reported. Capture and transmission measurements on a pure ^{206}Pb sample at a 60 m and 25 m station, respectively, have been performed at GELINA by Borella et al. [46]. Capture yields were obtained by applying the total energy

detection principle combined with the PHWT using C₆D₆ detectors. The neutron sensitivity of the set-up was determined by Monte Carlo simulations and verified by experiment. The experiments, data reduction and analysis procedures were carried out following the constraints and recommendations in Ref. [2,18,19]. In Ref. 46 results of a simultaneous analysis of capture and transmission data for neutron energies < 80 keV are given. For neutron energies between 80 keV and 625 keV the experimental yield was analyzed by fixing the neutron widths reported by Horen et al. [44, 45]. From these data capture areas with an accuracy < 5% have been deduced. The thermal data of Refs. [37 – 42] and the TOF-data of Refs [44 - 46], which are available in numerical form in the EXFOR library, can be used to improve the total and capture cross section for neutron induced reactions in ²⁰⁶Pb below 650 keV. From such an evaluation capture areas and peak cross sections for neutron energies < 200 keV can be deduced with an accuracy < 5 %. However, this level of accuracy cannot be reached for the cross section between resonances. Since Borella et al. [46] and Mizumoto et al. [43] have demonstrated that the γ -ray emission spectra are limited to a few cascades, the contribution of direct capture and/or interference effects cannot be excluded.

Capture measurements using C₆D₆ detectors on pure ²⁰⁶Pb have also been reported in Ref. [47]. However, the experimental yield is not available in numerical form. In addition, a weighting for an ideal detection system was applied, i.e. for a 0 MeV discrimination level on the observed energy deposited in the C₆D₆ detector. Therefore, correction factors to account for the finite discrimination level are required, which are very difficult to be determined accurately for neutron capture on ²⁰⁶Pb [2].

Capture measurements on ²³⁸U(n, γ) in the energy region < 200 keV

The status of the thermal capture cross section has been reviewed by Trkov et al. [48]. A list of measured coherent scattering lengths is given in Ref. [49].

These scattering lengths together with results of transmission and capture measurements at ORELA have been used by Derien et al. [50] to determine parameters of individual resonances for ^{238}U below 20 keV. The transmission experiments were carried out at 40 m, 150 m and 200 m station and the capture measurements at a 40 m and 150 m station using an OLS. The capture data of Corvi et al. [51] obtained at GELINA together with the thermal capture cross section of Poenitz et al. [52] have been used to adjust the parameters of the bound state(s). The resulting capture cross section at thermal $\sigma(n_{\text{th}},\gamma) = 2.7$ b is close to the value $\sigma(n_{\text{th}},\gamma) = 2.683$ (0.012) b recommended by Trkov et al. [48].

A list of capture cross section data (including absolute and shape data) for ^{238}U that can be used to determine the average capture cross section in the URR and higher is given by in Ref. [53]. These data have been used to evaluate the average capture cross section up to 2.2 MeV based on a least squares adjustment using the GMA code developed by Poenitz [54]. In Ref. [53] average capture cross sections below 200 keV are recommended with uncertainties between 0.5% and 3.3%. Other evaluations of the capture cross section for ^{238}U below 200 keV are reported by Fröhner [55], Maslov et al. [56] and Courcelle et al. [57]. These evaluations result from a parameterization of the cross section data by the Hauser-Feshbach formalism including fluctuation corrections. The evaluation process includes results of total cross section and in-elastic cross section data.

Unfortunately only one set of capture cross section data in the resonance region, the one reported by Yamamuro et al. [5], was based on the total energy detection principle using C_6D_6 detectors. In addition this data set suffers from a rather large 7.7% normalization uncertainty. Recently, capture cross section experiments have been carried out at GELINA and n_TOF using C_6D_6 detectors combined with the PHWT. A re-evaluation of the cross section data, including the results of these measurements, should result in a capture

cross section for ^{238}U with uncertainties below 2% for neutron energies < 200 keV.

$^{241}\text{Am}(n,\gamma)$

A list of experimental data that can be used to re-evaluate the capture cross section of ^{241}Am is given in Ref. 59. The data include thermal capture cross sections, integral measurements and transmission and capture TOF-data. Transmission measurements in the resonance region have been performed by Adamchuk et al [60], Slaughter et al. [61], Derrien and Lucas [62], Belanova et al. [63] and Kalebin et al. [64]. Results of capture experiments are reported by Weston and Tod [65], Gayther and Thomas [66], Wisshak and Käppeler [67], Vanpraet et al. [68] and Jandel et al. [69]. A comparison of the different data sets reveals large fluctuations between them, as illustrated in Fig.1 and Fig. 2. In Fig 1 the total capture cross section and the capture cross section to the ground state at 0.0253 eV are compared.

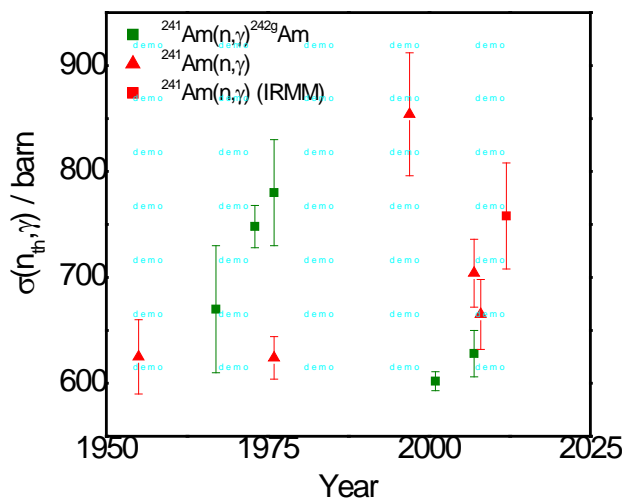


Fig. 1 Results of the total capture cross section and cross section to the ground state at 0.0253 eV.

Fig. 2 plots the ratio of $g\Gamma_n$ values obtained by Derrien and Lucas [62], Kalebin et al. [64] and Weston and Todd [65] relative to those of Jandel et al. [69]. This figure together with the radiation widths in Table 2 show that the $g\Gamma_n$ – values of Jandel et al. [69] are systematically higher compared to those in Ref. [62,64,65] and coincide with a systematic lower radiation width. This observation suggests that the data of Ref. [62,64,65] suffer from a systematic effect due to the sample properties as discussed in Ref. [2]. On the other hand the capture cross section at 0.0253 eV, 665 (33), of Jandel et al. [69] is rather low and could indicate that this data suffers from a systematic effect due to the normalization.

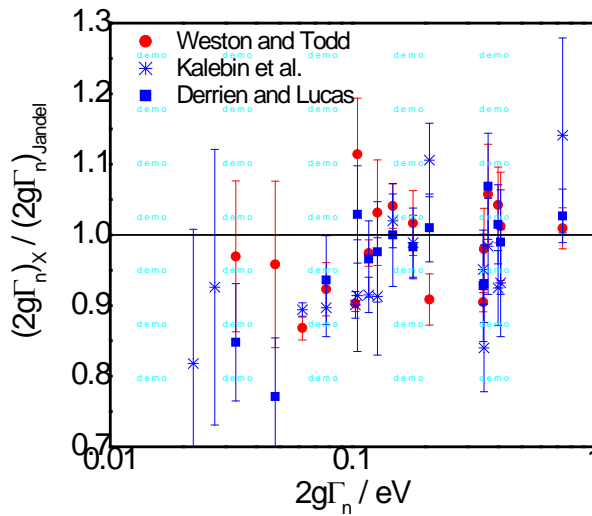


Fig. 2 The ratio of $g\Gamma_n$ -values reported by Derrien and Lucas[62] , Kalebin et al. [64] and Weston and Todd [65] relative to those of Jandel et al. [69].

E_R / eV	Radiation width, $\Gamma_\gamma / \text{meV}$	
	Weston and Todd [65]	Jandel et al. [69]
0.306	46.9 (0.3)	44.4(0.3)
0.574	47.3 (0.3)	43.3 (0.5)
1.272	49.2 (0.3)	45.3 (0.7)

Table 2 Radiation width determined by Weston and Todd [65] and Jandel et al. [69] for three low energy resonances of ^{241}Am .

The accuracy of both the resonance parameters and thermal capture cross section can be improved by re-analyzing the above mentioned TOF-data together with results of transmission and capture measurements performed at GELINA [80] and capture measurements at J_PARC [81] and n_TOF [82]. Although such a re-evaluation of the resonance parameters will not directly contribute to an improved capture cross section in the keV region this data, in particular the transmission data of GELINA, can be used to correct for systematic effects due to sample characteristics and improve the normalization of the capture data of Weston and Todd [65], Gayther and Thomas [66], Wisshak and Käppeler [67], Vanpraet et al. [68] and Jandel et al. [69].

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3.2 Current status on inelastic cross section measurements

Ongoing work

To the best of our knowledge ongoing work for inelastic scattering on major actinides is limited to the collaboration led by the Strasbourg group [1,2,3]. Here ^{235}U , ^{238}U and ^{232}Th have been measured using the $(n,n'\gamma)$ -technique at the IRMM GELINA facility and there are firm plans for measuring ^{233}U with the same setup. Data analysis for the first three isotopes is in progress and final results for ^{235}U and possibly ^{232}Th should be available within about 1 year. For ^{238}U the foreseen completion of the work is in 2 years. The gamma-ray production cross sections should be compared to earlier work, notably that of the Los Alamos group using the GEANIE array (e.g. [4,5]), of the TUNL group [6] and of the Lowell group [7]. The main purpose of these cross sections is as an accurate benchmark for model calculations. In most cases for actinides missing transitions (due to overlap with fission products or natural activity gamma-rays) and a large conversion coefficient for low energy transitions preclude results that are complete up to a certain excitation energy. Here an exception must be made for ^{232}Th where the data are complete up to 774 keV excitation energy. However, the accuracy for the important 49 keV transition is compromised by the above mentioned effects.

Recent and earlier work

1990-present: Recently $^{238}\text{U}(n,n'\gamma)$ data were published by the TUNL group [6]. It is not clear if they have a continuing experimental program. A collaboration between LANL and LLNL carried out measurements for major actinides ($^{235,238}\text{U}$) with the $(n,n'\gamma)$ -technique using the GEANIE array at the LANSCE neutron time-of-flight facility between 1995 and 2005 ([4,5] and references therein). From 1990 to 1999 Subgroup 4 [8] of the OECD-NEA Working Party on Evaluation Cooperation stimulated a considerable experimental effort regarding inelastic scattering on ^{238}U (Baba et al. at Tohoku University [9,10] and references therein, Kornilov et al. at Obninsk [11], Moxon

et al. at IRMM [12] and Plompen et al. also at IRMM [13]). All work for this subgroup focused on the time-of-flight technique ((n,n')-technique) and the detection of the emitted neutrons at one or several well defined angles. The work of Moxon et al. was special in this respect since it involved a filtered beam at the GELINA time-of-flight facility. The other experiments were carried out using quasi mono-energetic neutrons from the ${}^7\text{Li}(p,n)$, ${}^3\text{H}(p,n)$ or ${}^2\text{H}(d,n)$ reactions. In parallel measurements of cross sections were made for ${}^{232}\text{Th}$ by Miura et al. [10,14], Smith and Chiba at ANL [15], for ${}^{235}\text{U}$ by Kornilov and Kagalenko [11] and for ${}^{239}\text{Pu}$ by P. Staples et al. [16] and by G. Yue et al. [17] at the University of Lowell. At the University of Kentucky the spin and parity assignment of the 1414 keV excited state was studied via angular distributions of the emitted gamma-rays produced by inelastic neutron scattering and some unnormalized γ -ray intensities are reported [18].

1950-1990: In the period prior to 1990 the largest number of measurements for inelastic scattering on the major actinides concern (n,n') experiments at pulsed quasi mono-energetic neutron sources. The better known laboratories¹ involved were Bruyères-le-Châtel [19], Argonne National Laboratory [20], the University of Lowell [21], IPPE Obninsk [22], Los Alamos National Laboratory [23], Aldermaston [24], Oak Ridge National Laboratory [25] and the Institute of Atomic Energy in Beijing [26]. Gamma-ray detection or (n,n' γ) experiments were carried out at both pulsed quasi mono-energetic (Dave et al. [27] at Lowell) and a pulsed white neutron source (Olsen et al. at Oak Ridge National Laboratory [28], Voss et al. [29] at the Forschungszentrum für Kernphysik, Karlsruhe). Filtered beam experiments were carried using reactor beams (L.L. Litvinskiy et al., Kiev [30]) and pulsed accelerator sources (R.R. Winters et al., ORNL [31]). The sphere transmission technique, promoted by Bethe and Beyster [32] was applied up to 1963 (Allen et al., LANL [33], M.H. McTaggart and H. Goodfellow AWE [34]). It provides a measurement of the non-elastic cross section with reasonably good accuracy. Part of the reason for its limited use probably lies in the mass of the sphere that is required.

¹ Only one reference is given for each of these laboratories. Please check references therein and other sources (EXFOR) for further work.

Status and outlook

It has been shown in several places that the scatter among the results from the different experiments is relatively large. In part this may reflect progress in measurement technique, such as the optimization of sample geometry, shielding, detector efficiency calibration, method of normalization, and handling of background. In part this reflects the fact that the results shown most often are derived data. In particular, level cross sections and total inelastic cross sections derived from neutron time-of-flight spectra obtained at a number of angles, require important corrections to go from the measurement results to the reported data. For instance, the fission spectrum must be subtracted and elastic scattering must be separated from inelastic scattering in order to report elastic scattering and inelastic scattering differential cross sections. One furthermore has to integrate over angle and obtain the yield of neutrons at low emission energy by an extrapolation to account for the detection threshold. Most measurements do not resolve the states of the ground state rotational band from the elastic scattering peak and therefore report differential scattering data for “pseudo-elastic” scattering implying a sum over the contributions from the $0^+-6^+(8^+)$ states (U-238 and Th-232). Also for states at higher excitation energy the resolution into separate levels is often limited or impossible.

It is felt that the double differential data are nevertheless very valuable and may provide more accurate information than the derived experimental data uncertainties suggest. If used properly it may be that comparison between theory and experiment leads to valuable improvements for modeling of inelastic scattering. It is therefore worthwhile to review the available data in order to check whether the original double differential data are still accessible, what is really reported in the paper, and how the derived data depend on assumptions about the fission spectrum and elastic scattering. Such a review should be wary of the fact that it is difficult to get a comprehensive overview of all pertinent work using the available EXFOR browsers (IAEA/NNDC or NEA). To get a complete overview it is recommended to look at several recent publications (e.g. reports to this technical meeting, P. Young’s report on actinide evaluations for ENDF/B-VII in the Nuclear Data Sheets, the JENDL-4 evaluation report...).

Finally it is of course crucial to stimulate experimentalists to measure this important cross section on the major actinides in the interest of applications. The possible use of new techniques and new facilities should be promoted to allow improvements over past results.

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3.3 Experiments on prompt fission neutrons in inverse kinematics

Introduction

Important progress in fission experiments has been achieved by studying the fission of high energetic projectiles induced by interactions with target nuclei [1,2,3]. This “inverted” kinematical condition allows studying fission of short-lived nuclei that cannot be provided as target material. Fragmentation of relativistic ^{238}U gives access to a large number of projectile fragments with mass number $A \leq 238$ and $Z \leq 92$ [1]. Isotopes of $Z=93$ and isotones with $N=147$, but always $A \leq 238$, are accessible by charge-changing reactions [4]. Transfer reactions of ^{238}U with an energy slightly above the Coulomb barrier produce nuclei in the vicinity of ^{238}U which can also be heavier than $A=238$ [2].

Comparable kinematical conditions may be offered in the future at CERN ISOLDE with postaccelerated radioactive beams produced by spallation of ^{238}U with a diversity of fissioning systems similar to projectile fragmentation [5]. Moreover, in inverse kinematics the possibilities for identifying the fission products are considerably improved by their appreciably higher energies if compared to traditional experiments in normal kinematics. The atomic numbers of all fission fragments can unambiguously be determined in an ionization chamber [6], and their masses can be deduced from the deflection in a dipole magnet [7,8] and a time-of-flight measurement. These new approaches also offer new possibilities for experimental studies on prompt-neutron emission in fission.

Direct detection of prompt neutrons

Neutron detectors with kinematical information have been developed, specifically suited for GANIL energies [9] and for SIS energies [10]. The characteristics of scintillation neutron detectors at energies up to 12 MeV have been studied in ref. [11]. Others provide the neutron multiplicity [12, 13]. The detection of neutrons emitted from fissioning systems at GANIL energies is not much different from experiments in normal kinematics. A major problem, however, results from the large amount of matter provided by the magnetic spectrometer that is needed for identification of the fission products, which leads to substantial neutron absorption and scattering. Thus, direct kinematical neutron detection is not a promising option for this kind of experiments. The neutron detector LAND of GSI has been used specifically in fission experiments (e.g. [14, 15]) with a relativistic ^{238}U beam. It has an active area of 2 m times 2 m, a high detection probability above 90%, a position resolution of 7 cm and a

time resolution of 460 ps [16]. The ALADIN dipole allows free passage of the fission neutrons to the LAND detector that is mounted about 10 m behind the target, because the neutrons are emitted in a narrow cone in forward direction. The conditions will be similar in the future R3B set-up [17], where a new dipole magnet (GLAD) and a new neutron detector will be available. Thus, the complete identification of both fission products in atomic number Z and mass number A can be achieved together with the detection of all neutrons emitted from the fissioning system and from the fission fragments. The kinematical properties can be determined with an uncertainty given by the resolutions in time and position of the LAND detector. However, high multiplicities lead to an overlap of the signals of the individual neutrons.

Indirect determination of mean neutron multiplicities

The mean multiplicity of prompt fission neutrons as a function of fragment mass can also indirectly be determined by measuring the masses of the fission fragments before and after neutron emission, see e.g. [18]. The traditional method consists of a double-energy, double-time-of-flight measurement. In inverse kinematics, the post-neutron mass of one or both fission products can unambiguously be determined with the magnetic spectrometer, e.g. VAMOS or ALADIN, respectively GLAD. The pre-neutron mass can be obtained from a measurement of the emission angles of the fission products. The ratio of the momentum components transversal to the beam direction is proportional to the inverse of the mass ratio. This defines the masses of the fission fragments if the mass of the fissioning system is known. In transfer reactions, the compound nucleus is known from a measurement of the light transfer product. Multi-chance fission, i.e. neutron emission prior to fission, complicates the situation. Application to electromagnetic-induced fission in a heavy target material at relativistic energies suffers from a background of nuclear-induced events that can only partly be suppressed on an even-by-event basis by the condition that all protons of the fissioning system should be present in the fission fragments [1]. This problem will be solved in fission induced by tagged photons in a heavy-ion – electron collider ring [19]. If isotropic neutron emission in the respective fragment frame is assumed, it does not change the emission angle on the average. The resolution is determined by the precision with which the fragment emission angles are determined, by the recoil of the emitted neutrons and by the angular straggling of the projectile in the first section of the target and the fission fragments in the remaining target section. An additional uncertainty originates from the emittance of the primary beam or, if the individual projectiles are

tracked, from the resolution of the dedicated position detectors.

The calibration of the pre-neutron-mass-ratio determination is based on a direct geometrical measurement. The resolution obtained for a single event is not crucial: It is the uncertainty of the mean value of the pre-neutron fragment mass for a given post-neutron fission-product mass that determines the uncertainty of the mean prompt-neutron multiplicity.

Conclusion

Experiments in inverse kinematics extend the diversity of fissioning systems for experimental studies considerably. Measurements on prompt neutrons are feasible. Direct detection of all fission neutrons is possible with relativistic beams, since the large dipole magnet (e.g. ALADIN or GLAD) provides free passage for the neutrons. Indirect determination of mean prompt-neutron multiplicities as a function of the fission-product mass is possible when the mass of the fissioning nucleus is known. This information is available in electromagnetic-induced fission in case of first-chance fission, and it is accessible in transfer-induced fission.

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4. Recommendation of collaborative path forward to meet the needs

To obtain the accurate and reliable nuclear data satisfying the target accuracy, measurements with the precision equivalent to the target accuracy are necessary together with appropriate descriptions of data reduction and uncertainty evaluation in a research report. Furthermore, to verify the nuclear data and their uncertainties, it is not enough to repeat simply the same kind of measurements, but it is recommended to proceed independent measurements, for examples, using different measurement principles, facilities, detectors, samples, experimenters, which may affect the experimental values. Especially, combined use of differential and integral measurements is recommended to achieve the target accuracy. These efforts are indispensable to eliminate unrecognized systematic effects as much as possible.

The difficulty on appropriate descriptions of data reduction and uncertainty evaluation for each experiment is sometimes caused by the limitation of a number of pages on research reports, in many cases, Journal articles. Even in the case that there is a technical report that describes the details of the detectors and/or the facilities, it does not include all of the procedures of data reductions and uncertainty evaluations in details. The procedures depend on a laboratory or even an experimenter. This is one of the negative factors that make the evaluations difficult. The comprehensive and unified guides are desired to be prepared for each experimental method, which describe the details of the data reduction and uncertainty evaluation.

As were discussed in Chapter 3, the difficulties of measurements and the origins of unrecognized systematic effects are different for each reaction and nucleus. However, the collective knowledge of the recognized systematic effect can be used for the case of other nuclear data. Comprehensive knowledge of these systematic effects for each measurement type (capture, fission, inelastic, and ν) will be a valuable base for both experimenters and evaluators, and should be summarized as a series of research reports. The recent review paper on capture in ref. [1] is a good example.

To guarantee independence of experiments, international collaborations are indispensable, paying attention not to be disturbed by “Phase locking effect” pointed out by Lawrence; the phrase means that once one’s measurement agreed with a celebrated expert’s measurement, one stops to doubt his own measurement and misses to find unrecognized systematic effect. In order to demonstrate the effectiveness of such collaborations, an appropriate framework should be established, where serious scientific discussions are possible.

Nuclear data together with their uncertainties should be revised periodically by an appropriate international committee, such as the Committee on Data for Science and Technology (CODATA) [2] in the field of fundamental physical constants.

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5. SUMMARY

To meet the requirement of accurate nuclear data for developing advanced nuclear systems, pertinent efforts of experiments and evaluations are still required and indispensable. As described in Chapter 3, there are striking technical advancements in nuclear data measurement methods. For example, high-intensity-pulsed neutrons generated by spallation reaction at CERN in Europe, LANCE in USA, and J-PARC in Japan become available to obtain high precision neutron TOF data. Finer corrections for traditional techniques became also possible by utilizing recent Monte Carlo simulation techniques or refining existing data reduction codes REFIT, SAMMY etc. New concept of detectors and innovative methods using inverse reactions have also been developed and applied for nuclear data measurements. By utilizing these state-of-art techniques, further improvements of nuclear data accuracy are expected.

It is understood that the experimental result is the best estimate of the value of the measurement, and that all components of uncertainty contribute to the total uncertainty. However, some of systematic effects are sometimes unrecognized and not discussed in published papers. It is worthwhile to notice that only the recognized systematic effects are corrected and taken into account as a part of the total uncertainty. Some recommendations of collaborative path forward to meet the needs were summarized in Chapter 4.

In order to obtain an accurate nuclear data, it is important to measure nuclear data precisely and identify the unrecognized systematic effects as much as possible. The double check experiments are indispensable to verify the results. To guarantee independence of experiments, international collaborations are effective. In order to demonstrate the effectiveness of such collaboration, an appropriate framework should be established, where serious and detailed scientific discussions are possible.

APPENDIX

(the selected reactions are shown in the report paper, the other are stored in CD.)

Graphical presentation of EXFOR data will be inserted here