KfK 2917
April 1980

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ISSN 0303-4003

Abstract

A reliable prediction of the in-pile and out-of-pile physics characteristics of nuclear fuel is one of the objectives of present-day reactor physics. The paper describes the main production paths of important actinides for light water and fast breeder reactors. The accuracy of recent nuclear data is examined by comparisons of theoretical predictions with the results from post-irradiation analysis of nuclear fuel from power reactors, and partly with results obtained in zero-power facilities. A world-wide comparison of nuclear data to be used in large fast power reactor burn-up and long term considerations is presented. The needs for further improvement of nuclear data are discussed.

Kerndatenanforderungen für die Analyse des zeitlichen Verhaltens von Aktiniden-Isotopen in Kernreaktoren

Zusammenfassung

Die zuverlässige Vorhersage der physikalischen Charakteristika von Kernbrennstoff innerhalb und außerhalb des Reaktors ist eine der Aufgaben der heutigen Reaktorphysik. Der Bericht beschreibt die Hauptwege für die Erzeugung einiger wichtiger Aktiniden in Leichtwasserreaktoren und schnellen Brütern. Die Genauigkeit von Kerndaten wird überprüft durch Vergleiche von theoretischen Vorhersagen mit Ergebnissen aus der Analyse von Nachbestrahlungsuntersuchungen an Brennstoffen aus Leistungsreaktoren und auch an einigen Ergebnissen aus Nulleistungsanordnungen. Ein weltweiter Vergleich von Kerndaten zur Verfolgung von Langzeitanalysen in großen schnellen Reaktoren wird dargestellt. Anforderungen zur weiteren Verbesserung von Kerndaten werden diskutiert.

NUCLEAR DATA NEEDS FOR THE ANALYSIS OF GENERATION AND BURN-UP OF ACTINIDE ISOTOPES IN NUCLEAR REACTORS

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Introduction

A reliable prediction of the in-pile and out-of-pile characteristics of nuclear fuel is one of the objectives of presentday reactor physics investigations. From the nuclear data point of view, the cross-sections of the dominating actinide isotopes as ^{235}U, ^{238}U, and ^{239}Pu, have been investigated over the energy range of interest for more than two decades. Especially the fast reactor development was accompanied by cross-section measurements in many laboratories. Even today the nuclear data do not match the accuracy requirements of fast reactor designers, so that many laboratories still adjust their data files to a large variety of integral experiments. In 1975 a first international specialists' meeting[1] showed large discrepancies in nuclear data and corresponding group constants for nearly all of the secondary actinide isotopes (all other actinide nuclides except the main isotopes of U and Pu). Though the accuracy requirements for these secondary isotopes are not as stringent as those for the main nuclides, improvements were clearly necessary. The usual tests of nuclear data in critical or subcritical zero power facilities is concentrated on the start-up conditions of a power reactor. The change in isotopic concentration of the fuel during burn-up, especially the build-up of secondary actinides usually is checked by post irradiation examination of spent fuel. Unfortunately, the information from these experiments is often regarded as commercial. A large effort has been spent on nuclear data measurements and evaluations especially for the cross-sections of the secondary actinide isotopes in the seventies. At many conferences the status and the needs for further improvement have been described. In May 1978, on a symposium on nuclear data problems in thermal reactor application[2], an already satisfactory accuracy of most of the cross-sections, important for in-pile and out-of-pile investigations in thermal reactors, has been reported. At Harwell in September 1978, a broad review of the nuclear data status for reactor applications has been given. For instance, in Ref. 3 the nuclear data needs for the analysis of the out-of-pile stages of various nuclear fuel cycles have been discussed. From that discussion it transpired that further improvement of nuclear data with respect to their status in 1978 was very unlikely to reduce any of the technical difficulties in fuel handling. Proper updating of the data, used in the various laboratories, has been required. At Brookhaven in November 1978[4], at Cadarache in May 1979[5] the status of the cross-sections of the secondary actinides has been summarized. Additional information can be obtained from the International Conference on fast reactor physics in Aix-en-Provence in September 1979[6]. Admittedly, only little new information can be reported in this paper. It will concentrate on the check of actinide nuclear data with special emphasis on the secondary actinide isotopes in LWRs and FBRs. Further needs for nuclear data improvements will be deduced from these tests.

Main production paths of important secondary actinides in LWRs and LMFBRs

In the generation chain of the actinides the paths of the main nuclides are well known. In order to determine the importance of nuclear data improvements of secondary actinides, the main production paths of ^{237}Np, ^{238}Pu, ^{243}Am and ^{244}Cm in PWRs and LMFBRs are listed in Table I[7].

Table I. Main Production Paths for Important Secondary Actinides. [%]

ISOTOPE	Formation	PWR 32000 MWd/t	LMFBR 85000 MWd/t	
			LWR-Pu	EQUIL.
^{237}Np	(n,γ) from ^{236}U	80	14	10
	(n,2n) from ^{238}U	20	86	90
^{238}Pu	(n,γ) from ^{237}Np	91	15	29
	(α) from ^{242}Cm	∿ 9	73	56
	(n,2n) from ^{239}Pu	< 0.1	12	15
^{243}Am	(n,γ) from ^{243}Pu (including $^{241}Pu \rightarrow {}^{241}Am \rightarrow {}^{242g}Am \rightarrow {}^{242}Pu$	99.4	95.4	97.3
	(n,γ) from ^{242m}Am	0.6	4.6	2.7
^{244}Cm	(n,γ) from ^{243}Am	99.6	99.6	99.8
	(n,γ) from ^{243}Cm	0.4	0.4	0.2

^{237}Np is important because via neutron absorption ^{238}Np is generated which decays in about 2 days to ^{238}Pu with its strong α-decay. This isotope causes special difficulties in reprocessing of spent fuel by radiolysis, and also in refabrication of reprocessed fuel. ^{243}Am is the main nuclide generating ^{244}Am by neutron capture, which decays in about 10 hours to ^{244}Cm. ^{244}Cm is, like ^{242}Cm a very strong α- and

neutron emitter.

As can be seen from Table I, the main difference between thermal and fast system is that (n,2n) processes, especially on ^{238}U, play a more important role in fast than in thermal systems. As a result the production path of ^{237}Np and therefore the production path of ^{238}Pu are changed. Also it has to be recognized that there are great differences between a fast reactor fuelled by LWR-plutonium and one operating in its equilibrium cycle.

As can be deduced from Table I, neutron capture in ^{242}Am and in ^{243}Cm are relatively unimportant. Besides the neutron reaction data of the higher plutonium isotopes also those of ^{241}Am are important, because this isotope contributes to the reactivity balance in a fast reactor. Clearly, the branching ratio leading to the ground and isomeric states of ^{242}Am have to be known sufficiently well (the accuracy requirements can be looked up e.g. in Ref. 1). Furthermore, neutron capture in ^{236}U, ^{237}Np and ^{243}Am are rather essential in leading to nuclides which are of great concern in out-of-pile fuel cycle analyses. Along with this statement goes the importance of the fission cross-sections of these isotopes. Additionally, (n,2n) processes on ^{238}U and ^{239}Pu are of concern. The requirement for accurate data for these processes is well covered by the requirements of fast reactor physics.

In the following sections we examine the accuracy of present data and methods against results obtained from post-irradiation experiments in power reactors and partly with results obtained in zero-power facilities.

Test of actinide nuclear data for LWR fuel cycle analysis

A thorough comparison of theoretical predictions with experimental results for the isotopic composition of spent fuel is very complex. The complete power-history of an operating plant has to be known. In particular the local variations of reaction rates in time around the irradiation position have to be considered very carefully. This is especially important if the isotopic compositions of higher actinides, which are rather sensitive to the neutron flux level and its variation, have to be determined. This is due to the competition between neutron reactions and decay. In order to have some conclusive results with respect to nuclear data uncertainties, some of the calculational complexities are removed by prescribing the experimental power density or flux density at the irradiation position as a function of time. If at least the dominating effective fission cross-section (i.e. of ^{235}U) is correct, from the local power density the local flux can be deduced sufficiently accurate. Often either ratios of nuclide concentrations or other relative figures (e.g. isotopic abundances), which are usually given by chemical experiments, are investigated, by which calculational uncertainties are somewhat decreased. In Table II some results from post-irradiation analyses of BWR- and PWR-fuel are given.[8,9] The figures give the deviations between theory and experiment in percent. Some of the nuclide concentrations are measured against burn-up (B), others are measured against depletion of ^{235}U (D^5), the last line gives the isotopic ratios against uranium (U). GARIGLIANO and GUNDREMMINGEN are BWRs of 150 MWe and 237 MWe, respectively. TRINO and OBRIGHEIM are PWRs of 250 MWe and 283 MWe, respectively. This comparison shows differences between theory and experiment of several percent, if empirical information is used. In the last line only the burn-up has been modified arbitrarily by 1 %, no adjustment of group constants was done. Even if empirical information is used in theory, the agreement between "theory" and experiment is sometimes not satisfactory. This can be seen in the figures for the German BWR plant GUNDREMMINGEN. The burn-up "adjustment" for TRINO gives large differences for the higher Pu and transplutonium isotopes.

Deviations sometimes are reduced by adjusting the group constants to the experiments. Darrouzet et al[7] 1978 compared theoretical results with unadjusted and adjusted nuclear data for isotopic ratios in the Ardennes PWR power plant with chemical mass determinations, shown in Table III. In adjusting the group constant data to experiments, the differences can be reduced to a few percent.

Often irradiation experiments are interpreted by use of the Oak Ridge code ORIGEN[10], which is a fundamental mode burn-up and irradiation program and in its original version it uses time independent one group

Table II. Burn-up Analysis of LWR - Spent Fuel

Reactor	^{238}Pu	^{240}Pu	^{241}Pu	^{242}Pu	^{241}Am	^{242}Am	^{243}Am	^{242}Cm	^{244}Cm
GARIGLIANO+)		D^5:0.5	D^5:0.8	D^5:1.6					
GUNDREMMINGEN+)		B:3.0	B:4.6	D^5:9.4			B:3.0	B:6.9	
OBRIGHEIM+)	B:2.3		D^5:1.0	D^5:1.0			D^5:7.0	D^5:1.3	D^5:2.8
TRINO+)	B:3.1		B:1.2	B:2.2			B:6.2		
TRINO*)		U:1.8	U:1.0	U:7.0	U:21.0	U:24.0	U:28.0	U:8.0	U:1.0

+) Theoretical prediction by using empirical information from measurements on many samples.[8]

*) Theoretical prediction with adjustment due to changing the burn-up by 1 %.[9]

Table III. Comparison of Theoretical and Experimental Isotope Ratios for Ardennes-PWR Power Plant

Isotope Ratio	Pre-Adjustment $\frac{E-C}{C}$ [%]	Post-Adjustment $\frac{E-C}{C}$ [%]
^{232}U / ^{238}U	+ 6	+ 7
^{236}U / ^{238}U	+ 7	+0.4
^{237}Np / ^{238}U	-10	- 4
^{238}Pu / ^{239}Pu	+16	- 3
^{242}Pu / ^{239}Pu	-10	- 5
^{242}Cm / ^{239}Pu	+12	+ 4
^{244}Cm / ^{239}Pu	+53	- 3

cross-sections. Part of these data were adjusted. Because this code is widely used all over the world, it is of interest to check its ability with more refined methods and with experiment. Part of the information on the revised version of ORIGEN, called ORIGEN-2 has been reported in 1978[11]. ORIGEN-2 uses time dependent group constants and has been checked with a special option of the CITATION diffusion code. Parallel to ORIGEN-2 at Karlsruhe we have developed a similar code, named KORIGEN[12]. Many of the nuclear data have been adopted from the ORIGEN-2 version, while some are based on the KEDAK nuclear data file.

In order to improve the calculational basis for out-of-pile investigations (i.e. considering more than 1000 nuclides consistently also during reactor life) a code-system, named HAMKOR, has been established[13]. With this system the neutronics of LWR-cells can be determined for any burn-up state and for any light-water lattice. HAMKOR is based on an improved version of HAMMER for the static calculations, and coupled to the burn-up and irradiation program KORIGEN. Nuclear data for the important isotopes are taken from the latest available data on the KEDAK-file, status 1979. Further sophistication and generalization of this system are underway at Karlsruhe; this includes also coupling of the burn-up routine to other cell codes and to global whole-core neutronic codes as well as the adoption of any more accurate nuclear data. Table IV gives a comparison of various methods with experiments. The experimental results, on which the figures of Table IV are based, were taken from post-irradiation analysis of the US-ROBINSON reactor[14].

The last column of Table IV shows the trend of deviations of ORIGEN-2 results in comparison to more refined calculations, which are not described in detail ("literature values" in Ref. 11).

As can be seen, the ORIGEN and KORIGEN versions show surprisingly good results up to ^{241}Am, both in comparison with each other and with experiment. The original ORIGEN version gives such relatively good agreement because of data adjustment. The good agreement of ORIGEN-2, which is mainly based on ENDFB/IV, may be fortuitious, as may be concluded from the last column. The large deviations in nuclide concentrations above ^{241}Am can easily be traced back to insufficient data in ENDFB/IV.

HAMKOR-results show large deviations from experiment for ^{238}Pu, ^{242}Am, and ^{242}Cm. Work is underway to reduce the about 5 % deviation of the ^{235}U concentration from experiment by careful investigation of all thermal power contributors, but the agreement principally is limited by the uncertainty in the burn-up determination. All other relative concentrations (they are given as ratio of the specific nuclide concentration to the concentration of the corresponding element) are sufficiently well predicted (remember that the power history has been prescribed). First of all, the nuclides with larger deviations have very low isotopic concentrations ($^{238}Pu \sim 1.6$ a/o, $^{242}Am \sim 0.3$ a/o, $^{242}Cm \sim 3.3$ a/o). The statistical errors of the relative mass determinations are about 1 % for ^{238}Pu, larger about 10 % for ^{242}Am and larger about 5 % for ^{242}Cm (these figures were communicated to the author by L. Koch, Transuranium Institute Karlsruhe). Now, various groups have undertaken an interlaboratory comparison to check the experimental accuracy of post-irradiation analysis. Preliminary results show an unsatisfactory discrepancy (e.g. for the same sample a factor of two in the ^{242}Cm content). Therefore, more effort has to be spent to assess reliable uncertainty margins. If possible, measurements, based on different methods, have to be applied to reduce systematic errors. The situation seems to be comparable to that of differential cross-section measurements some years ago, when the statistical error of a certain measure-

Table IV. Deviations (in percent) of Isotopic Compositions [a/o] between Theory and Experiment for the US-ROBINSON PWR at discharge. $(\frac{E-C}{E}$ [%])

NUCLIDE	HAMKOR (1979)	KORIGEN (1978)	ORIGEN-2 (1978)	ORIGEN (1973)	Trend of deviations for ORIGEN-2 (PWR, BWR)
^{234}U	–	-7.1	0.	7.1	up to 30 % overpred.
^{235}U	+4.9	8.4	10.4	5.8	
^{236}U	-2.9	-2.0	0.	-1.4	
^{238}U	-0.03	-0.1	-0.06	-0.03	
^{238}Pu	20.0	5.1	-3.8	15.4	up to 40 % underpred.
^{239}Pu	0.05	1.3	-1.3	-1.7	
^{240}Pu	-3.8	-1.3	8.0	4.2	up to 15 % overpred.
^{241}Pu	5.5	-4.4	-8.7	-3.6	$\pm \sim$ 15 %
^{242}Pu	-1.3	-11.6	-2.8	1.7	$\pm \sim$ 15 %
^{241}Am	-3.4	13.6	10.6	-5.7	up to 40 % underpred.
^{242m}Am	22.8	46.5	-31.4	factors	
^{243}Am	5.4	-23.1	-17.5	8.8	$\pm \sim$ 5 - 10 %
^{242}Cm	-30.5	17.4	26.2	70.0	
^{243}Cm	-4.1	27.3	74.4	88.4	
^{244}Cm	-3.1	-5.3	-2.0	-7.6	up to 30 % underpred.

ment was much smaller than the discrepancy to the results of another experimentalist. In case of ^{242}Cm, comparison of HAMKOR with other post-irradiation experiments shows satisfactory agreement.

From Table IV we conclude that for most of the isotopes of interest in LWR fuel cycle analysis the nuclear data presently are accurate enough. The important exceptions are for ^{238}Pu and ^{242}Cm (with respect to HAMKOR analysis). Further investigation of these discrepancies is required. The assessment of more reliable uncertainty margins in post-irradiation experiments is necessary. Additional comparisons of theory with spent fuel experiments have to be performed to get a more transparent view of the data status. These are underway at Karlsruhe, especially for German PWRs and for PWRs with recycled plutonium fuel elements. Presently no further needs for nuclear data measurements can be made.

It should be mentioned that the spent fuel analysis is of great concern in nuclear safeguards investigations. There it is a key issue to detect reliably and early any diversion of fissile material in spent fuel. The essential plutonium content can e.g. be determined by means of the so-called isotopic correlation technique.[15] At the present time, more theoretical exploration of this method has to be undertaken. For the experimental results, better and more reliable accuracies have to be achieved, as already mentioned earlier. Up to now the technique of isotopic correlation can only be used as a supporting measure in safeguards analysis. In addition, there is no unique concept of safeguarding nuclear material, so that a request for more accurate nuclear data in this field is unlikely to be made now, see also Ref. 16.

Test of actinide nuclear data for fast reactor fuel cycle analysis

It is well known that the nuclear data for fast reactors are not of the same sufficiently good quality as for thermal reactors. This is reflected in Table V, where one group cross-sections for fast reactors are compared. These data have been provided in 1976 to the author by US and UK members of the NEACRP. A very wide spread in capture and fission data is observed. The data status is that of about 1973 and older.

Table V. Comparison of ORIGEN (US) and FISPIN (UK) One Group Cross-Sections for Fast Reactors

Isotope	ORIGEN (1973)		FISPIN (1973)	
	Capture	Fission	Capture	Fission
^{237}Np	0.76	0.36	1.87	0.34
^{238}Pu	0.22	1.38	0.44	1.15
^{239}Pu	0.5	1.85	0.51	1.82
^{240}Pu	0.41	0.35	0.59	0.38
^{241}Pu	0.43	2.49	0.59	2.64
^{242}Pu	0.34	0.28	0.38	0.30
^{241}Am	0.99	0.46	1.91	0.40
^{242}Am	0.4	1.83	0.1	3.33
^{243}Am	0.55	0.27	1.7	0.19
^{242}Cm	0.38	0.42	0.5	1.26
^{243}Cm	0.4	0.32	0.1	3.14
^{244}Cm	0.37	0.41	0.48	0.55

The discrepancies in nuclear data also show as large differencies in isotopic compositions, radiation and heat production in spent fast reactor fuel. A comparison of ORIGEN-type calculations with experimental results on the fast test reactor RAPSODIE is given in Table VI[17]. The used data are a mixture of nuclear data from various origins, some of them taken from the 1975 Karlsruhe meeting on data for actinides[1] and some of more recent communications up to 1977.

Table VI. Comparison of Calculations (ORIGEN type) with Experimental Results for RAPSODIE $(\frac{C - E}{E}$ [%])

ISOTOPE	BURN-UP a/o		
	1.126	4.035	1.089
^{234}U	-4	-4	-9
^{235}U	-0.1	-0.6	-0.1
^{236}U	-10	-12	-20
^{238}U	-0.6	+0.3	-0.2
^{238}Pu	-99	+2	-99
^{239}Pu	+1.5	+3	+1.6
^{240}Pu	0.09	-0.01	-0.2
^{241}Pu	+2	+5.6	+2
^{242}Pu	-9	+5	-11
^{241}Am	+39	+15	-44
^{242}Am		-96	
^{243}Am		-68	
^{242}Cm	-47	-47	-47

Even larger discrepancies can be observed for ^{238}Pu, and the americium and curium isotopes as in similar comparisons for thermal system (see Table IV).

To improve the situation, some experiments have been performed in fast critical facilities. Table VII comprises C/E ratios for fission rate ratios and $\sigma_a/\sigma_f = 1+\alpha$ obtained in the ZEBRA and SNEAK assemblies, which were already reported in 1977[18]. At the Aix-en-Provence conference in 1979 additional results were reported by Sanders et al for ^{241}Am and ^{243}Am[19]. The nuclear data for ^{241}Am were based on UK-evaluations by Lynn et al, and for ^{243}Am solely on nuclear model calculations. The agreement is satisfactory.

At the Aix-en-Provence conference in 1979 the successful in-pile measurements in PHENIX on a large variety of samples have been reported by Giacommetti[20]. The results are not yet available. After adjustment, good agreement was found with experiment. From this

analysis it can be concluded that in France and in institutions, associated with the CEA, a fairly good knowledge of the present accuracies of differential nuclear data for actinides, applied to fast power reactor experiments, exists.

Table VII. Comparison of Theoretical and Experimental Results in Fast Critical Experiments (C/E)

ISOTOPE I	ZEBRA 1975 $\sigma_f^I/\sigma_f(^{239}Pu)$	SNEAK 1975 $\sigma_f^I/\sigma_f(^{239}Pu)$	SNEAK 1975 $1+\sigma_c/\sigma_f(^{239}Pu)$
^{238}U	1.04 ± 4 %	0.95 ± 2.2 %	0.98
^{240}Pu	1.003± 5 %	0.94 ± 1.5 %	1.27
^{241}Pu	1.05 ± 3 %	1.05 ± 1.5 %	1.03
^{242}Pu	1.23 ± 5 %		
^{241}Am	1.26 ± 4 %	1.40 ± 2 %	1.95
^{243}Am	0.88 ± 4 %		
^{244}Cm	1.35 ± 8 %		

ZEBRA 1979

Isotope	σ for Cm Production	C/E	Remarks
^{241}Am	1.28±0.1	0.88±0.07	PFR Spectrum
^{243}Am	1.32±0.2	1.20±0.18	
^{241}Am	1.48±0.2	0.84±0.06	CDFR Spectrum
^{243}Am	1.99±0.2	0.87±0.09	

In 1979, the author acted as focus within an activity of the NEACRP to compare one group data for actinides (fission and capture), as used in the various laboratories. This type of benchmark comparison was based on the spectrum (for collapsing) of the NEACRP-benchmark for a 1000 MWe fast reactor.[21] The results of the benchmark were reported many times, e.g. by Lesage at the Gatlinburg Conference 1978.[22]

Tables VIII, IX and X show the one group constants provided by France (FRA), Germany (GER), Japan, UK and USA (ENDFB/IV and ENDFB/V). For completeness, values from the USSR are included, which are not strictly comparable, because they are not based on the NEACRP benchmark, but rather on a more simple benchmark (often quoted as BAKER-benchmark) of a large fast reactor; therefore due to spectrum differences the one group constants may differ to those of the NEACRP benchmark. In the tables A means: adjusted in CARNAVAL IV, the French fast reactor group constant set, M means a modification of data with respect to the UK set FGL5.

At first sight and compared with Table V, considerable improvements have been obtained in recent years. There are no longer large discrepancies as before for actinide nuclei important in in-pile and out-of-pile fuel cycle investigations. However, the differences in capture data for ^{232}Th, ^{233}Pa, $^{233,234}U$, ^{237}Np, $^{238,240,241,242}Pu$, $^{242,243}Am$ and $^{242,243,244}Cm$ are not so small that they can be neglected.

First, one has to repeat the checks already made in fast reactors, with the more recent data available now. However, it is not justified to apply the group constants, taken e.g. from Table VIII to Table X, for smaller test facilities as RAPSODIE with different neutron spectra, because most of the cross-

Table VIII. Comparison of One Group Constants for the NEACRP-LMFBR Benchmark: ^{232}Th to ^{237}Np

	CAPTURE [b]							FISSION [b]						
COUNTRY	FRA	GER	JAPAN	UK	USA		USSR*)	FRA	GER	JAPAN	UK	USA		USSR*)
BASIS	CARN IV	KEDAK	JAERI	FGL5	ENDF/B IV	ENDF/B V		CARN IV	KEDAK	JAERI	FGL5	ENDF/B IV	ENDF/B V	
^{232}Th	0.59	-	-	0.43	0.43	0.41	-	0.0097	-	-	0.0094	0.0091	0.0096	-
^{233}Pa	1.33	-	-	1.13	1.18	1.18	-	0.22	-	-	0.64	0.062	0.062	-
^{233}U	0.33	-	-	0.27	0.28	0.29	-	2.81	-	-	2.85	2.90	2.84	-
^{234}U	0.35	-	-	0.61	0.63	0.66	0.68	0.3	-	-	0.29	0.29	0.32	0.34
^{235}U	0.59^{A}	0.6	0.64	0.53	0.62	0.62	-	1.94^{A}	2.0	2.10	1.98	2.02	2.0	-
^{236}U	0.51	-	-	0.59	0.60	0.61	0.62	0.092	-	-	0.088	0.089	0.099	0.11
^{238}U	0.29^{A}	0.31	0.31	0.29	0.31	0.31	-	0.040^{A}	0.041	0.045	0.043	0.040	0.040	0.041
^{237}Np	1.44	1.64	-	1.95	1.68	1.86	1.8	0.33	0.32	-	0.31	0.31	0.32	0.32

*) not strictly comparable, see text.

Table IX. Comparison of One Group Constants for the NEACRP-LMFBR Benchmark: ^{238}Pu to ^{242}Pu

	CAPTURE [b]							FISSION [b]						
COUNTRY	FRA	GER	JAPAN	UK	USA		USSR*)	FRA	GER	JAPAN	UK	USA		USSR*)
BASIS	CARN IV	KEDAK	JAERI	FGL5	ENDF/B IV	ENDF/B V		CARN IV	KEDAK	JAERI	FGL5	ENDF/B IV	ENDF/B V	
^{238}Pu	0.54[A]	0.68	0.91	0.45	0.48	0.80	0.90	0.84[A]	1.03	1.12	1.13	1.15	1.14	1.16
^{239}Pu	0.57[A]	0.57	0.61	0.55	0.56	0.57	-	1.81[A]	1.87	1.88	1.83	1.84	1.86	-
^{240}Pu	0.55[A]	0.57	0.62	0.63	0.58	0.61	-	0.33[A]	0.36	0.37	0.35	0.36	0.36	-
^{241}Pu	0.5[A]	0.5	0.55	0.62	0.51	0.50	-	2.53[A]	2.54	2.61	2.69	2.61	2.63	-
^{242}Pu	0.63	0.5	0.41	0.39[M]	0.39	0.48	-	0.22	0.24	0.28	0.22	0.27	0.25	-

*) not strictly comparable, see text.

Table X. Comparison of One Group Constants for the NEACRP-LMFBR Benchmark: ^{241}Am to ^{244}Cm

	CAPTURE [b]							FISSION [b]						
COUNTRY	FRA	GER	JAPAN	UK	USA		USSR*)	FRA	GER	JAPAN	UK	USA		USSR*)
BASIS	CARN. IV	KEDAK	JAERI	FGL5	ENDF/B IV	ENDF/B V		CARN IV	KEDAK	JAERI	FGL5	ENDF/B IV	ENDF/B V	
^{241}Am	2.02[A]	1.93	1.69	2.01[M]	1.37	1.89	1.90	0.29	0.26	0.30	0.31	0.41	0.28	0.31
^{242}Am	0.7	0.46	-	0.11	-	0.097	0.42	3.7	3.86	-	3.33	-	3.61	3.2
^{243}Am	1.6	1.53	1.65	1.73[M]	0.86	1.20	1.8	0.2	0.2	0.23	0.19	0.17	0.22	0.20
^{242}Cm	0.59	-	-	0.51	-	-	0.46	2.05	-	-	1.23	-	-	0.16
^{243}Cm	0.5	0.18	-	0.10	-	0.27	0.39	3.39	2.46	-	2.89	-	2.77	2.5
^{244}Cm	0.85	0.65	0.66	0.49[M]	0.53	0.91	0.98	0.45	0.43	0.43	0.38	0.52	0.40	0.42

*) not strictly comparable, see text.

sections are strongly energy dependent, as can be seen from Table VII: the differences in group constants, obtained with a PFR- and a CDFR spectrum, respectively, are relatively large. Therefore, again coupling of modern data libraries to whole-core diffusion and burn-up codes (including the long-term decay phase for out-of-pile investigations), is necessary. This has been done already in Karlsruhe[23], the application of this system is underway now.

Conclusion

For thermal reactors, the present accuracy of nuclear data for the analysis of fuel cycle aspects, probably is sufficient. Further checks to integral experiments, especially on commercial power plants, have to be performed to support this statement. Updating of all data libraries including group constant libraries to the 1979 or later status is required.

For fast reactors, a careful evaluation by tracing back the observed differences between theoretical and experimental results to the basic nuclear data is required first. Additional comparisons with results from integral experiments in power reactors will show the quality of the data.

Only after this exercise for thermal and fast systems requests for new differential data measurements may be formulated.

Acknowledgment

The author is grateful to those members of the NEACRP, who provided the material presented in this paper. Helpful discussions with U. Fischer, B. Goel, M. Marzo and H.W. Wiese are gratefully acknowledged, also the careful reading of the manuscript by A. Rowe.

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