KfK 4461
Januar 1988

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PWA 46/88

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ISSN 0303-4003

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Abstract

The Joint Evaluated File JEF-1 is used as a basis to generate group constants for fuel cycle analyses of PWRs. This group constant set is applied to the NEACRP-benchmark on "Recycling of Reprossed Uranium" and to postirradiation experiments in the KWO/PWR. The results in k_∞ for the NEACRP-benchmark almost coincide with the average of the solutions submitted to the benchmark. The calculated end-of-life nuclide concentrations are in good agreement with experimental results obtained for a KWO/PWR configuration with the exception of the EOL concentration of Cm242; a careful check shows that very probably the experimental values are too low; this conclusion is consistent with the overall findings.

As another example, JEF-1 nuclear data are used to analyse a tight lattice configuration of advanced PWRs. This APWR is being investigated in a joint cooperation between the Karlsruhe Nuclear Research Center, SIEMENS/KRAFTWERKUNION, and the Paul Scherrer Institute PSI at Würenlingen, Switzerland. A special aspect is addressed in this paper, i.e. the reliable determination of the reactivity change upon voiding a tight lattice. It is shown that the use of consistent weighting spectra for the normal (water-in) and the voided (water-out) configuration drastically reduces the void reactivity change compared to that case, when the same weighting spectrum (e.g. that of the water-in configuration) for both the normal and the voided state of an APWR is used.

Qualifikation der JEF-1 Kerndatenbibliothek für die Analyse des Abbrandverhaltens von Druckwasserreaktoren

Zusammenfassung

Die Kerndatenbibliothek JEF-1 wird als Basis zur Gruppenkonstantenerstellung für die physikalische Analyse des Brennstoffzyklus von Druckwasserreaktoren zugrunde gelegt. Dieser Gruppenkonstantensatz wird auf das Benchmark "Rezyklierung von aufgearbeitetem Uran" und auf Nachbestrahlungsexperimente im KWO-Druckwasserreaktor angewendet. Für das NEACRP-Benchmark stimmen die mit diesen Daten ermittelten Werte für k_∞ praktisch mit dem Mittelwert aller eingereichten Lösungen überein. Die berechneten Konzentrationen von Nukliden bei Entladung aus dem Reaktor stimmen gut mit Meßwerten überein mit Ausnahme der Nuklidkonzentration von Cm242; eine sorgfältige Prüfung dieses Ergebnisses ergibt, daß mit großer Wahrscheinlichkeit die Meßwerte zu niedrig sind.

Als weiteres Beispiel wurde der JEF-1 Gruppensatz zur Analyse eines Fortgeschrittenen Druckwasserreaktors eingesetzt; diese Untersuchung erfolgte im Rahmen der Kooperation Kernforschungszentrum Karlsruhe, Siemens/Kraftwerkunion und dem schweizer Paul Scherrer Institut in Würenlingen. Speziell wird in diesem Bericht die verläßliche Bestimmung der Reaktivitätsänderung bei Kühlmittelverlust dieses sehr engen Gitters untersucht. Es wird gezeigt, daß die Verwendung konsistenter Wichtungspektren zur Bildung von Gruppenkonstanten sowohl für den Normalfall als auch für die vom Kühlmittel entleerte Reaktorkonfiguration die Reaktivitätsänderung drastisch reduziert verglichen mit dem Ergebnis, das man bei Verwendung nur eines Wichtungsspektrums (z.B. für den Normalfall) erhält.

Content

1. Introduction

A first test of the Joint European Nuclear Data File JEF-1 was already performed on light water moderated criticals and on start-up conditions of power reactors of the PWR-type /1/. In general, good agreement with experimental results and with benchmark calculations could be achieved. It was not clear whether the JEF-1 nuclear data can also describe burnup conditions in PWRs satisfactorily well: This problem is treated in this paper.

In a first study /2/, JEF-1 group constants had been applied to describe the burn-up behaviour of PWR-fuel as requested in the NEACR*-Benchmark on "Recycling of Reprocessed Uranium" /3/ and, as a second example, JEF-1 was applied to a power plant lattice of a German PWR plant, KWO** /4/, for which many experiments had been performed to determine the nuclide concentrations at end of life (EOL) by postirradiation analyses of the fuel. In /2/ especially the comparison of the calculated nuclide concentrations with experimental results showed differences, which could be either due to the fundamental data of JEF-1 and their processing to group constans via the Karlsruhe version of the processing system NJOY /5/, or the use of the group constants within the Karlsruhe reactor analysis code system KARBUS. Therefore an intensive investigation of the nuclear data, the processing of the data with NJOY, and the use of the group constants in the Karlsruhe code system was performed. The results of these investigations are described in /6/ and /7/. In Ref. /7/, an inconsistency in the use of group constants in the Karlsruhe calculational scheme could be removed. After intensive testing of the final calculational scheme, the JEF-1 group constant library was again applied to the NEACRP-benchmark on "Recycling of Reprocessed Uranium" and to a KWO-PWR configuration. The new results are discussed in this paper.

As another example, a tight lattice PWR configuration (APWR) /8/ was analysed: This benchmark is still under investigation by NEACRP. The results of the improved calculational tools are compared in this report with those of an earlier study on a reliable determination of the reactivity change following a loss of coolant from this reactor, performed in 1987 /9/.

* NEACRP = NEA-Committee on Reactor Physics

** Kernkraftwerk Obrigheim

For some important nuclear reactions, a comparison of the JEF-1 data with those from KEDAK-4 /10/ is given.

2. Test of JEF-1 Group Constants on the NEACRP-Benchmark on "Recycling of Reprocessed Uranium"

In 1983, NEACRP has posted a "Benchmark on Recycling of Reprocessed Uranium" to compare the effects of numerical methods and nuclear data relevant to the formation and to the depletion of uranium 236 in thermal reactors.

2.1 Definition of the benchmark

The first test example was for a typical PWR pin-cell for 33 GWd/t burnup; this example (case 1 a) should give the reference data without U236 initially.

Simplified assumptions were made:
The single cell is given in cylindrical geometry without leakage, i.e. the cell is infinitely long in axial direction and periodic boundary conditions are prescribed radially. The boron concentration is fixed.

The material properties and the number densities are as follows:

Fuel: UO_2
Diameter of the pin: 0.819 cm;
Mean temperature: 700° C
Number densities for 3.52 % enriched uranium fuel:
U 234: $7.24 \cdot 10^{18}$ cm^{-3}
U 235: $7.55 \cdot 10^{20}$ cm^{-3}
U 238: $2.22 \cdot 10^{22}$ cm^{-3}

Clad: Zirconium
Number density: $4.325 \cdot 10^{22}$ cm^{-3}
Temperature: 400 °C
Inside diameter: 0.836 cm
Outside diameter: 0.99 cm

Coolant/Moderator: H_2O

Number density: $2.387 \cdot 10^{22}$ cm^{-3}

Temperature: 306 °C

Natural boron concentration: $1.78 \cdot 10^{19}$ cm^{-3}

($B10 : 3.21 \cdot 10^{18}$ cm^{-3})

Cell diameter: 1.422 cm

Mean power: 32 MW/ton HM

(power per ton of heavy material)

The energy released by fission also is prescribed:

U 235: 201.7 MeV/fission
U 238: 205 MeV/fission
Pu 239: 210 MeV/fission
Pu 241: 212.4 MeV/fission

Requested are: k_∞ as function of burnup in steps of 3 GWd/tonHM, and the isotopic concentration of the uranium and plutonium isotopes, in addition also of Np 237 at end of life (EOL).

2.2 The burnup dependence of k_∞ with the JEF-1 data set

Tab. 1 shows the variation of k_∞ with burnup for different data sets: JEF-1/87 was prepared in 1987; it was used with small inconsistencies in the reactor code package as described in /7/. JEF-1/88 gives the result of the present (consistent) group constant set. The column "NEACRP-Average" shows the average of the submitted solutions to the benchmark. The results, obtained with KEDAK-4 nuclear data, are given for comparison. The additional data sets, starting from a consistent "mixed JEF/KEDAK group library" contain data for some plutonium isotopes from KEDAK-4 instead of JEF-1.

As can be seen, the consistent treatment in the present final data set compared to the previous data set JEF-1/87 lowers the values for k_∞ by about 0.5 % and less. It can be noted that the results of both the JEF-1 and KEDAK-4 group sets almost coincide with the averaged NEARCP solution.

The last four columns of Tab. 1 contain results for k_∞, which were calculated with group constant sets having nuclear data partly from KEDAK-4. In the mixed JEF/KEDAK group constant set the fundamental cross sections for following nuclides are taken from KEDAK-4: Th232, Np237, Pu239, Pu241, Pu242, Am241, Am242m and Cm244; all other data are derived from JEF-1. In the subsequent column the Pu241 data from KEDAK-4 have been replaced by JEF-1 data: differences in k_∞ can only appear for higher burnup of the fuel. It can be seen that at a burnup value of 33 GWd/tonHM the data for Pu241 from JEF-1 give about a 1 % higher reactivity compared with the results of the mixed data set. In the next column, additionally to the replacement of Pu241 also the data for Pu239 are taken from JEF: k_∞ is reduced slightly (about 0.2 %, and less for higher burnup). The additional replacement of the KEDAK-data for Pu242 by JEF-1 data gives negligible changes in k_∞ (last column). In the final data set the data for Am241 and Cm244 (except those for Am242m which were recently re-evaluated at Karlsruhe /16/) are replaced by JEF-1 data. Here it was expected that practically no changes would be observed, because for these isotopes the data in JEF-1 are comparable with those from KEDAK-4.

In Fig. 1, a graphical comparison of the k_∞ dependence on burnup for the final data set with the averaged results of the NEACRP contributions and with the results from KEDAK-4 is shown.
The k_∞-curve, obtained with the final JEF-1 group set, is only slightly higher than the averaged results of the all contributions. What should be noticed: the basic JEF-data 1987 give results at the upper boundary of the interval, which contains all the contributions to the benchmark; the consistent use of JEF-1 in the reactor codes lowers this solution practically to the average of all contributions. For rememberence: this effect is due to a consitent treatment of the group data especially for the low energy resonances of the heavy nuclides. As a final remark it should be noticed that KEDAK-4 data give values for k_∞ which are slightly lower than the NEACRP-average for higher burnup.

2.3 Sensitivity of the calculated EOL-concentrations for the NEACRP benchmark with respect to nuclear data sets

In Tab. 2 the nuclide concentrations at EOL are summarized for all data sets. Comparing the results obtained with the group set JEF-1/87 with those of

JEF-1/88, it can be seen that JEF-1/88 gives major changes in the EOL-concentrations for the higher Pu-, the Am- and the Cm-isotopes: this is due to an improved treatment of the low energy resonances for the heavy isotopes and the consistent use of the corresponding group constants in the reactor codes. Surprising is the very high isotopic concentration of Cm242; this observation will be discussed in chapter 3.3, where the calculated isotopic concentrations are compared to experimental results.

The last four columns of Tab. 2 give results for the EOL-concentrations calculated with mixed JEF-1/KEDAK-4 group sets. In comparing the results of JEF-1/88 with those in the column which is labelled "Mixed JEF/KEDAK Data", about a 2 % change in the U235 concentration is observed. Larger effects can be seen for the EOL concentration of Pu241 and of all nuclides with higher mass number. Great changes in the nuclide concentrations of all higher Pu-isotopes as well as those of the Am and Cm isotopes occur then, when in the mixed JEF/KEDAK set the data of Pu241, which are taken from KEDAK-4, are replaced by those from JEF-1. To trace the origin of these large changes, in Fig. 2 an Fig. 3 the differences in the group cross sections between KEDAK-4 and JEF-1 for Pu241 are shown for neutron fission and neutron capture, respectively: Large differences exist between these two data sets in the thermal energy range: for the fission cross section KEDAK-data are higher than JEF-1 data except for energies around 0.1 eV and around some eV; the capture cross section in KEDAK is also higher up to energies about 0.1 eV; around 1 eV the KEDAK-values are lower than the JEF-1 values (in the dip of the group-cross section curve).

The additional replacements of the KEDAK-4 data by JEF-1 data in the MIXED JEF/KEDAK Set give only minor changes.

In summary, this sensitivity study shows very clearly that large differences in the individual cross section between KEDAK-4 and JEF-1 for the isotopes in question exist, but nevertheless compensating effects work in such a way that the k_∞-values, calculated with KEDAK-4 and with JEF-1, almost coincide.

2.4 Conclusion on the quality of JEF-1 nuclear data by application to the benchmark on "Recycling of Reprocessed Uranium"

JEF-1 gives very satisfactory results for k_∞ and the nuclide concentrations at EOL (33 MWd/tonHM), if the group data are consistently used in reactor calculations. This then confirms other tests for zero burnup, but the excellent agreement with the "average" solution for higher burnup values as well in k_∞ as in the nuclide concentrations gives confidence in the JEF-1 data for thermal reactor application. There is one exception: the Cm242 EOL concentration is surprisingly high compared to the results obtained with JEF-1/87 and KEDAK-4.

In the following chapter it will be shown, how the calculated EOL nuclide concentrations compare to experimental results from postirradiation experiments.

3. Comparison of Calculated Nuclide Concentrations at 30.16 GWd/tonHM Burnup with Experimental Results of Postirradiation Analyses

Within the framework of the "Isotopic Correlation Experiment (ICE)" many measurements were performed on irradiated fuel from the PWR plant Obrigheim (KWO) in the years 1977/78 /11/. The measurements were performed independently at the laboratories of the European Transuranium Institute (TUI), together with the IAEA. Analysed were those KWO-fuel elements, which had reached various burnup values in the 3rd, 4th and 6th KWO-cycle; two batches of each fuel element could be analysed. Theoretically the experimental results were used to check the accuracy of the KORIGEN code of KfK /12/. For minor actinides, a comprehensive review of the KfK-analyses shows the status of calculating the burnup behaviour and the isotopic concentrations of irradiated nuclear fuel /13/. For the test of JEF-1 data, the Karlsruhe burnup code system KARBUS /14/, was used.

3.1 Some remarks to the calculational procedure / 13 /

The time intervals between spectrum recalculations are chosen in accordance with the spectral variations caused by the soluble boron content in the coolant, and by fission product and transuranium isotope generation. These variations are larger at the beginning than towards the end of each cycle; the time steps are chosen accordingly. For the calculation of the neutron spectrum in the unit cell all important nuclei are considered, as usual. The time dependence of the soluble boron is taken into account properly, which is especially important in nuclear data checks. To avoid any ambiguity caused by theoretical methods near boundaries, water gaps or control rods, the experimentally measured power density at the position of the fuel sample to be analysed, is used. Thus a better deduction of nuclear data deficiencies in the postirradiation analysis for the various nuclide concentrations is possible. The accuracy of the theoretical analysis then depends mainly on the accuracy of the measured time dependent pin power density, usually about 2 - 3 %, and the nuclear data uncertainties, assuming the methods for the cell calculations to be correct. In the calculations, often not the (time dependent) pinpower density is taken from experimental results, but instead the measured burnup is used in the theoretical investigation; then accordingly the pin power has to be normalized to give the experimental burnup. The burnup value is the most dominant parameter for the amount of fission products and actinides in spent fuel, especially of all transuranium isotopes. In experiment, various methods are used to determine the burnup of spent fuel from a sample. Usually one can assume an uncertainty of 2 - 5 %

In conclusion, if nuclear data and theoretical methods are assumed to be correct, the predicted nuclide concentrations are as uncertain as the experimental burnup value i.e. up to 5 %. This is not a very high accuracy, but it is at least sufficient for the investigations of actinide isotopes. Clearly this uncertainty has to be compared with the uncertainty of the measurements for the determination of the nuclide concentrations; this is discussed subsequently.

3.2 Some remarks on the accuracy of the experimental results /13/

Many parameters influence the accuracy of the experimental results for nuclide concentrations in reactor fuel. Generally, the kind of material is important with respect to the used method of analysis, whether the sample contains fuel powder, whether it is a solution, or whether the fuel is oxide, carbide or nitrate. The accuracy is increased, if the material to be analysed is a pure material. High burnup fuel, either from an initial reactor loading with enriched uranium or mixed Pu/U oxide requires special attention in the analytical techniques. The homogeneity of the sample is of prime importance. In addition, different techniques need different amounts of sample material to perform a reliable measurement. A careful investigation of possible perturbations of a measurement for a specific isotope by other constituents of the sample is requested.

The following experimental techniques are used to analyse nuclear fuel: alpha counting, alpha spectrometry, gamma spectrometry, spectrophotometry, thermal ionization, mass spectrometry, titrimetry, X-ray fluorescence, isotope dilution mass spectrometry (IDMS). These methods yield different accuracies in applying them to element or isotope analysis in nuclear material. Any measured result should be given only together with a reliable error analysis. Unfortunately, this clear request is not fulfillyed in most of the quoted experimental results. For uranium and plutonium isotopes, de Bièvre et al. have assessed the achievable uncertainty components in fissile element and isotope assay in destructive analyses of nuclear material /15/. The uncertainties are quoted in the upper part of Tab. 3 for U and Pu elements in spent fuel and for Pu isotopes from pure plutonium material. The accuracy of experimental results in an isotopic analysis of spent fuel is estimated from the spread in the results obtained in various laboratories for alpha spectrometry and isotopic dilution measurements. The estimate for IDMS might be slightly optimistic; however, the experience, obtained in KfK, is promising.

3.3 Comparison of the calculated EOL nuclide concentrations at 30.16 GWd/tonHM with experiments

In Fig. 4 to Fig. 9 the calculated isotopic concentrations for the important heavy isotopes are compared to experimental results for a KWO-PWR configuration at about 30 GWd/tonHM burnup; JEF-1/88 is the cross section basis.

With only one exception, all isotopic concentrations agree satisfactorily well with the experimental results. The expection is Cm242 as already seen in section 2.3. The fact that the concentration for Am241 is in agreement with experiment, leaves only two explanations for this discrepancy: (a) the buildup of Cm242 with its short half life of 163 days via the 16 hours state of Am242 and the corresponding branching ratios from Am241 to Am242/Am242m is not correctly decribed, or (b) that the experimental results, as discussed in section 3.2 and Tab. 3, are low by about 50 %. The error possibility (a) was checked: no mistake could be found in the nuclide chain and in the branching ratios. The good agreement of the experiments with the calculation in Ref. /2/, using KARBUS is accidental: The data for fission and capture cross sections of Cm242 in this calculation were not taken from KEDAK-4; they are too high by orders of magnitude, so that these reactions can compete with the radioactive decay. From Fig. 8 can be seen that the Am241 concentration, calculated by KORIGEN, is lower by 50 % compared to the bulk of experiments and to the JEF-result, which inturn leads to a lower concentration of Cm242 by almost the same 50 %. This allows the following conclusion: Because the agreement between experiment and theory for the Am241 isotopic concentration is improved by the JEF-1 set in comparison with KORIGEN, then the JEF-result for the Cm242 concentration must be higher than the experimental result: Very probably the measured concentrations of Cm242 are too low. This explanation is consistent with the overall findings.

3.4 Conclusion on the quality of JEF-1 nuclear data drawn from a comparison of the EOL nuclide concentrations with experiments

The calculated EOL nuclide concentrations of a PWR are in excellent agreement with experimental results for a burnup of 30.16 GWd/tonHM with the exception of Cm242. In this case very probably the experimental results, which scatter

usually by about 50 %, are too low. Minor differences exist for the EOL concentrations of U235 and Pu239: Either the cross sections for these isotopes are not yet accurate enough, or the calculational procedure is to be improved further. It is hoped that with the release of JEF-2 these small deviations will vanish eventually.

4. Application of JEF-1 Group Sets to Tight Lattice APWRs

In Ref /17/ a review on the present state of the theoretical and engineering developments of an advanced PWR (APWR) and of related safety aspects was given. As far as the calculational accuracy is concerned, most of the main physics parameters as k_∞ or k_{eff}, the reaction rate ratios, the conversion ratio, the Doppler coefficient and the flux distribution are calculated fairly well when the results are compared to experiments, done e.g. in the PROTEUS assembly at the Paul Scherrer Institute (PSI) in Würenlingen, Switzerland. An uncertainty is still left for the theoretical determination of the void reactivity effect. Although in general the total void reactivity effect can be predicted, sometimes there is no consistency with the main contributions to this reactivity effect as e.g. expressed in reaction rate ratios: This is due to compensating effects in the void reactivity, which always has to be calculated with great sophistication.

At Karlsruhe, preliminary calculations for reaction rate ratios, measured in the PROTEUS lattice, were performed; these calculations were based on the JEF-1/87 group constant set. Satisfactory results were obtained in comparison with experiment for almost all reactor parameters /18/. These investigations will be repeated with the final data set for APWR applications, based on JEF-1.

With respect to the determination of the void reactivity effect, the still preliminary results of the NEACRP/HCLWR benchmark /19/ show non-acceptable deviations between the various contributions. In Ref. /9/ and in Ref. /18/ it was shown that for a reliable determination of the void reactivity effect in an APWR lattice, realistic weighting spectra for both the normal (water-in) and the voided (water-out) configuration have to be used. After internal pre-publication and distribution of the results, these findings were confirmed by other scientists; moreover, with this prescription good agreement of the calculated results with experiments was found /20/.

Specification of the benchmark

Lattice cell:	hexagonal with a cylindrical fuel rod; axially infinitely extended.
Moderator/fuel volume ratio:	0.6
Cell pitch:	1.0883 cm
Fuel:	Mixed PuO_2/UO_2
Pu-fissile:	8.0 %
Temperature:	900 K
Cladding:	Stainless Steel
Outer diameter:	0.95 cm
Clad thickness:	0.065 cm
Temperature:	600 K
Moderator:	H_2O
Temperature:	600 K
Linear Power:	160 W/cm

Number densities (x 10^{24} cm^{-3})

Fuel:

U235:	$6.094 \cdot 10^{-5}$
U238:	$2.025 \cdot 10^{-2}$
Pu239:	$1.563 \cdot 10^{-3}$
Pu240:	$6.872 \cdot 10^{-4}$
Pu241:	$2.765 \cdot 10^{-4}$
Pu242:	$2.108 \cdot 10^{-4}$
O16:	$4.610 \cdot 10^{-2}$

Cladding:

Fe (natural):	$4.831 \cdot 10^{-2}$
Cr (natural):	$1.570 \cdot 10^{-2}$
Ni (natural):	$7.648 \cdot 10^{-3}$
Mn55:	$1.486 \cdot 10^{-3}$

Moderator:

H:	$4.744 \cdot 10^{-2}$
O16:	$2.372 \cdot 10^{-2}$

With the present data set on JEF-1 nuclear data basis the drastic change in the void reactivity effect was found again as in Ref. /19/, when for the voided (water-out) configuration a special group set (taken from Ref. /7/) is used.

As expected from the PWR analysis in chapter 2, the k_∞ of the normal (water-in) configuration was decreased by about 0.5 %. For the voided (water-out) configuration, some slight changes were obtained compared to the results in /9/, which are due to the consistent treatment of the resonance region in the reactor calculations, and to an enlarged σ_0-grid. The absolute difference between the Δk_{void}, calculated with a group set using an APWR-weighting spectrum for both the normal and the voided configuration, and those sets generated with the APWR-weighting spectra for the water-in and for the water-out case, amounts to

$$\delta\,(\Delta k_{void}) = -0.016$$

This means that the calculational procedure with an extra weighting spectrum for the voided (water-out) configuration reduces the void reactivity effect remarkably and has always to be considered, if systems of strongly differring neutron spectra are investigated.

5. General Conclusion on the Quality of the JEF-1 Nuclear Data Library for Pressurized Water Reactor Analysis

Group constant sets, based on the JOINT Evaluated File JEF-1, are shown to describe fairly well the burnup behaviour of present days Pressurized Water Reactors up to a burnup of about 30 GWd/tonHM. This proof was obtained:

(a) by a comparison of the calculated burnup dependent k_∞-values and nuclide concentrations at EOL with the average of all solutions submitted to the international benchmark on "Recycling of Reprocessed Uranium": For this benchmark the calculated k_∞-values almost coincide with this average; the nuclide concentrations at EOL are also consistent with the average.

(b) by a comparison of the EOL nuclide concentrations with experimental results from a postirradiation analysis of fuel from the German KWO/PWR. The calculated EOL concentrations agree satisfactorily well with the measured values; only the EOL nuclide concentration for Cm242 are discrepant. A careful check of the results leads to the conclusion that very probably the experimental results are too low.

(c) JEF-1 group constant sets were applied to determine the reactivity change upon voiding an advanced PWR. Two group sets were established: one with the weighting spectrum for the normal (water-in) and one for the voided (water-out) configuration. In using these two group constant sets, the value for the reactivity change, i.e. Δk_{void}, is reduced drastically compared to that Δk_{void} which is calculated with a group set containing only the weighting spectrum of the water-in configuration. This result was checked in the SWISS PROTEUS critical assembly: as reported, good agreement of calculations on JEF-1 basis with experiment was found.

Acknowledgement

The author would like to thank the IAEA and KfK for their financial support; Prof. Keßler is thanked for making my stay at KfK possible. The sincere support and scientific guidance of Dr. H. Küsters is especially acknowledged. Mr. C.H. Broeders is thanked for his great help in using KARBUS, Dr. H.W. Wiese for stimulating discussions and a careful reading of the report. Mrs. S. Bereszko is thanked for her effective typing.

6. Tables and Figures

Burnup (GWd/tonHM)	JEF-1/87	JEF-1/88 Final	NEACRP Average	KEDAK-4	Mixed JEF/KEDAK Data Set	Repl. of Pu241 by JEF-Data	Addit. Repl. of Pu239 by JEF-Data	Addit. Repl. of Pu242 by JEF-Data
0	1.281	1.278	1.269	1.268	1.278	1.278	1.278	1.278
3	1.202	1.199	1.191	1.191	1.200	1.200	1.199	1.199
6	1.166	1.161	1.155	1.154	1.162	1.163	1.161	1.161
9	1.134	1.128	1.122	1.120	1.128	1.129	1.128	1.128
12	1.104	1.098	1.092	1.090	1.097	1.100	1.098	1.098
15	1.077	1.071	1.066	1.063	1.069	1.073	1.071	1.071
18	1.052	1.046	1.041	1.038	1.043	1.048	1.046	1.046
21	1.028	1.023	1.019	1.015	1.018	1.025	1.023	1.023
24	1.006	1.002	0.998	0.993	0.996	1.004	1.002	1.002
27	0.985	0.982	0.978	0.973	0.975	0.984	0.982	0.982
30	0.965	0.963	0.959	0.953	0.955	0.965	0.963	0.963
33	0.947	0.945	0.941	0.935	0.936	0.947	0.945	0.945

Table 1: The Variation of k_∞ with Burnup for Various Data Sets up to 33 GWd/tHM

Isotopic EOL Concentrations	JEF-1/87	JEF-1/88 Final	NEACRP Average	KEDAK-4	Mixed JEF/KEDAK Data Set	Repl. of Pu241 by JEF-Data	Addit. Repl. of Pu239 by JEF-Data	Addit. Repl. of Pu242 by JEF-Data
U234(10^{18})	4.197	4.221	4.336	4.224	4.202	4.219	4.221	4.221
U235(10^{20})	2.143	2.182	2.210	2.141	2.146	2.178	2.182	2.182
U236(10^{19})	9.213	9.237	9.239	9.365	9.277	9.241	9.237	9.237
U238(10^{22})	2.159	2.159	2.159	2.159	2.159	2.159	2.159	2.159
Pu238(10^{18})	3.627	3.769	3.629	4.131	3.848	3.779	3.768	3.768
Pu239(10^{20})	1.507	1.520	1.481	1.548	1.522	1.526	1.520	1.520
Pu240(10^{19})	5.675	5.083	5.379	5.042	5.117	5.086	5.083	5.083
Pu241(10^{19})	3.132	3.410	3.410	3.420	4.008	3.415	3.410	3.410
Pu242(10^{19})	0.956	1.030	1.109	1.084	1.063	1.029	1.028	1.030
Np237(10^{19})	1.129	1.175	1.018	1.252	1.182	1.176	1.175	1.175
Np239(10^{18})	2.012	2.014	2.018	1.999	2.044	2.017	2.014	2.014
Am241(10^{18})	0.950	1.041		1.041	1.186	1.042	1.044	1.044
Am242g(10^{15})	2.069	2.288		2.264	2.644	2.289	2.284	2.284
Am242m(10^{10})	2.073	2.376		2.353	2.660	2.374	2.372	2.372
Am243(10^{18})	1.688	1.936		2.174	1.996	1.955	1.952	1.935
Cm242(10^{17})	1.750	2.945		1.967	3.345	2.944	2.939	2.939
Cm244(10^{17})	5.550	6.174		6.995	6.299	6.209	6.191	6.140

Table 2: Nuclide Concentrations at EOL for the Various Data Sets and Comparison with the Average Solution of the NEACRP Contributions and with KEDAK-4

U, Pu Elements in spent fuel: 0.5 - 1 %

Pu - isotopes from pure plutonium material

Pu238 (2 %) Pu239 (0.1 - 0.15 %)
Pu240 (0.3 %) Pu241 (0.3 - 1 %)
Pu242 (0.5 %)

Spent ful isotopic analysis (based on exp. intercomparisons)

Nuclide	- spectrometry (%)	IDMS (%)
Pu236	20 - 30	
Pu238	5 - 15	
Pu242	< 5	
Am241	large scatter (50)	5
Am243	20	5
Cm242	large scatter (50)	
Cm244	5	5

Table 3: Estimated Experimental Accuracy of Postirradiation Techniques

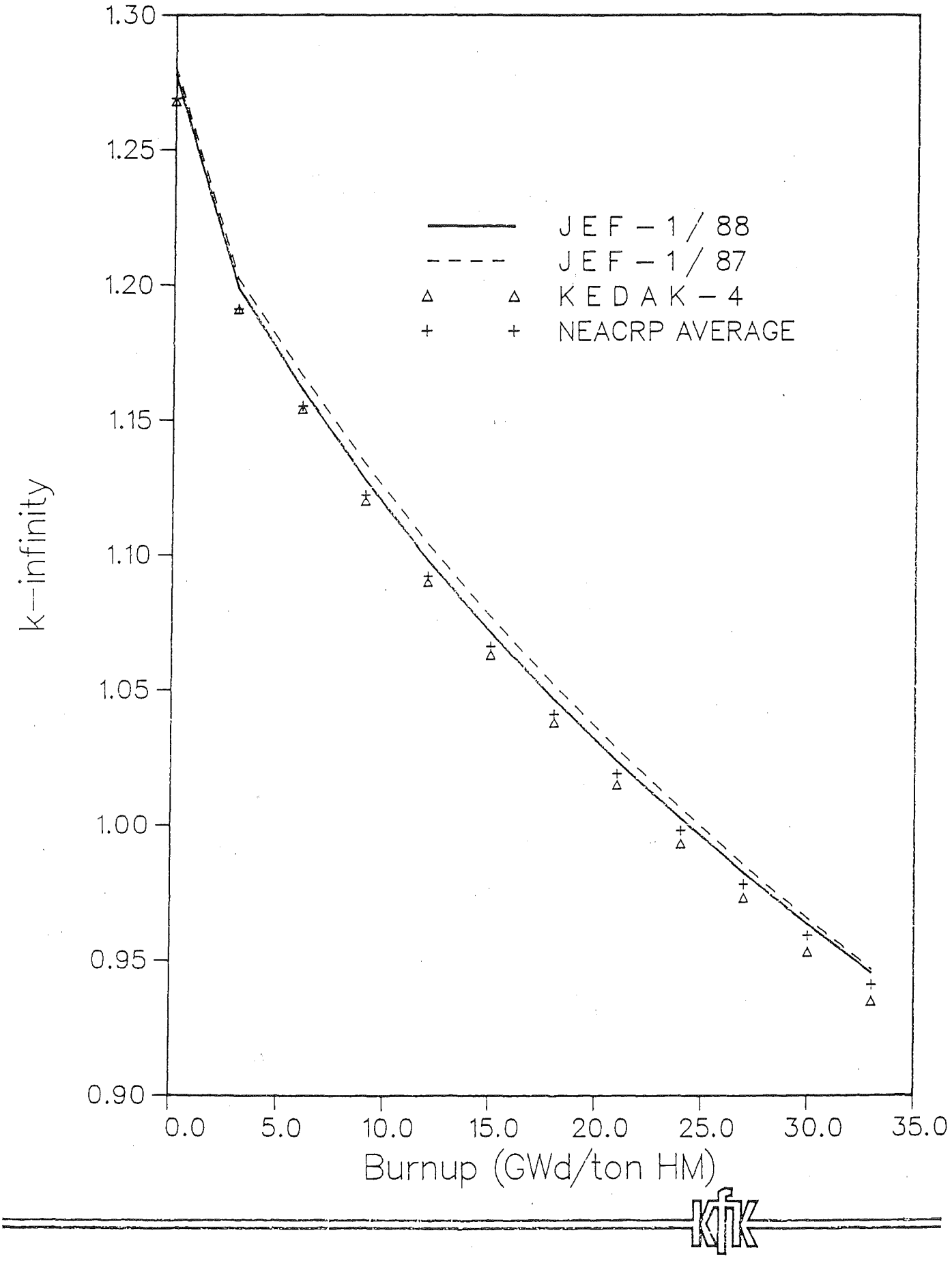


Fig. 1: Burnup Dependence of k_∞ for the NEACRP-Benchmark on "Recycling of Reprocessed Uranium" (Exercise 1a)

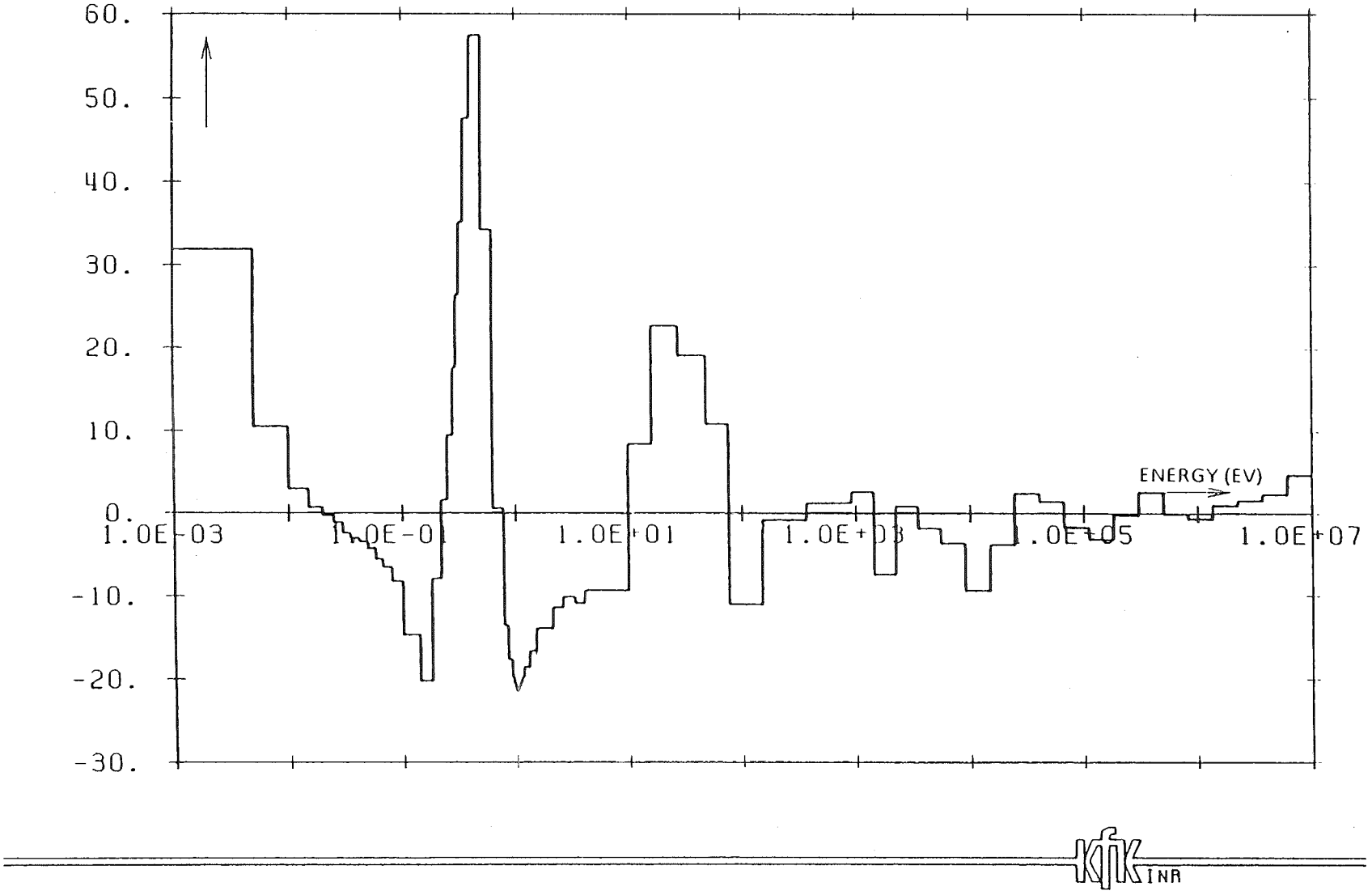


Fig. 2: Difference in % for the Fission Cross Section of Pu241

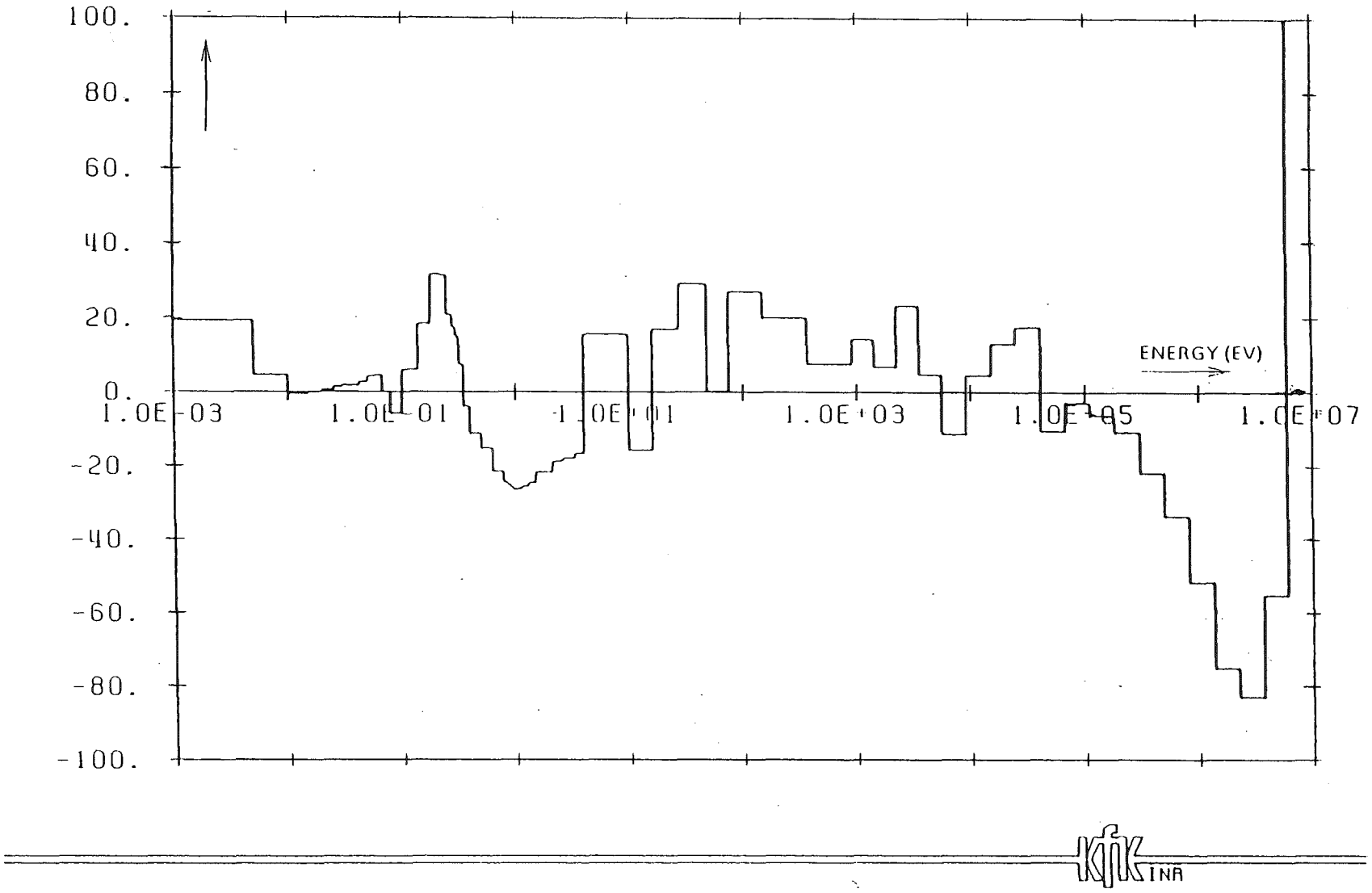


Fig. 3: Difference in % for the Capture Cross Section of Pu241

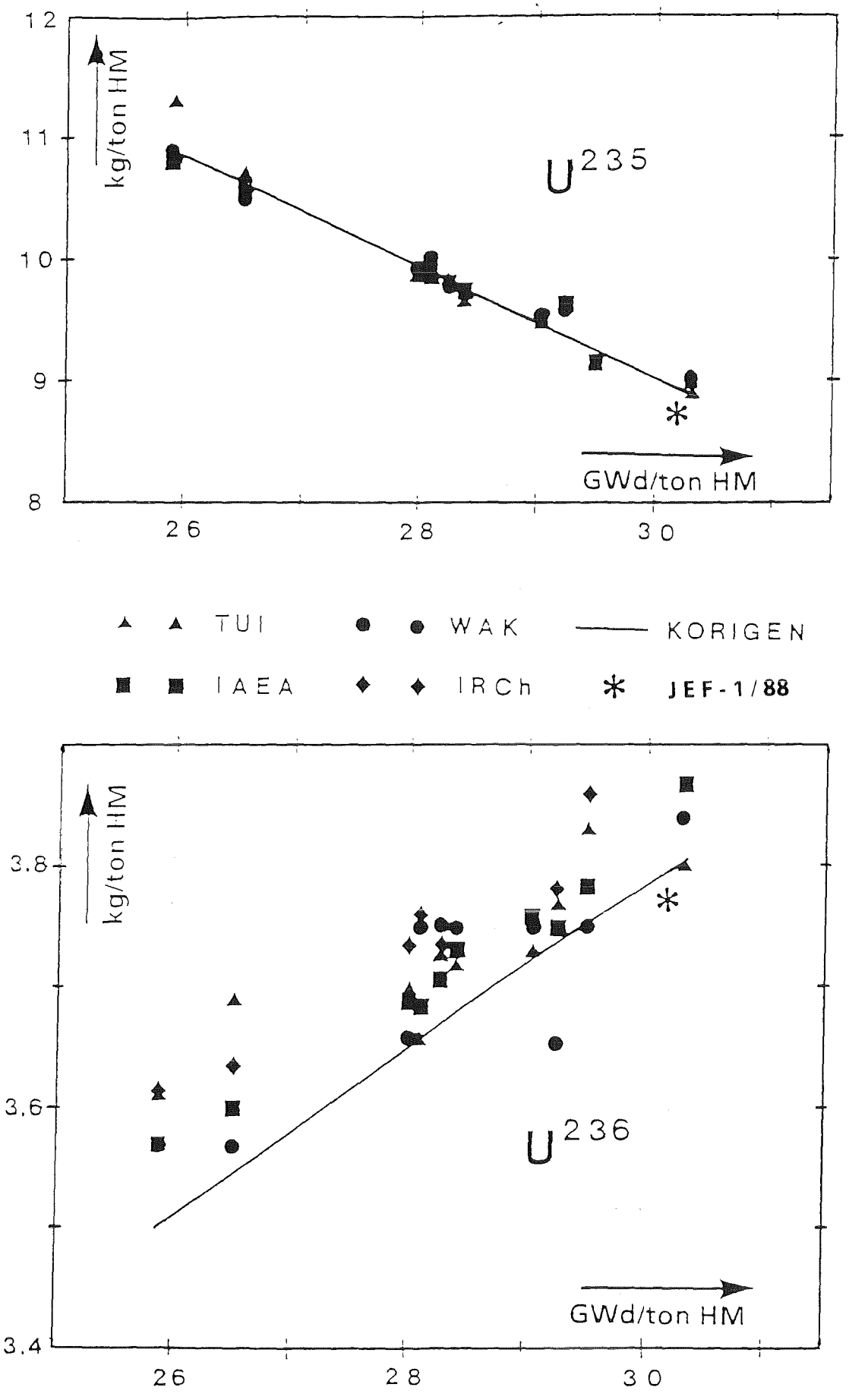
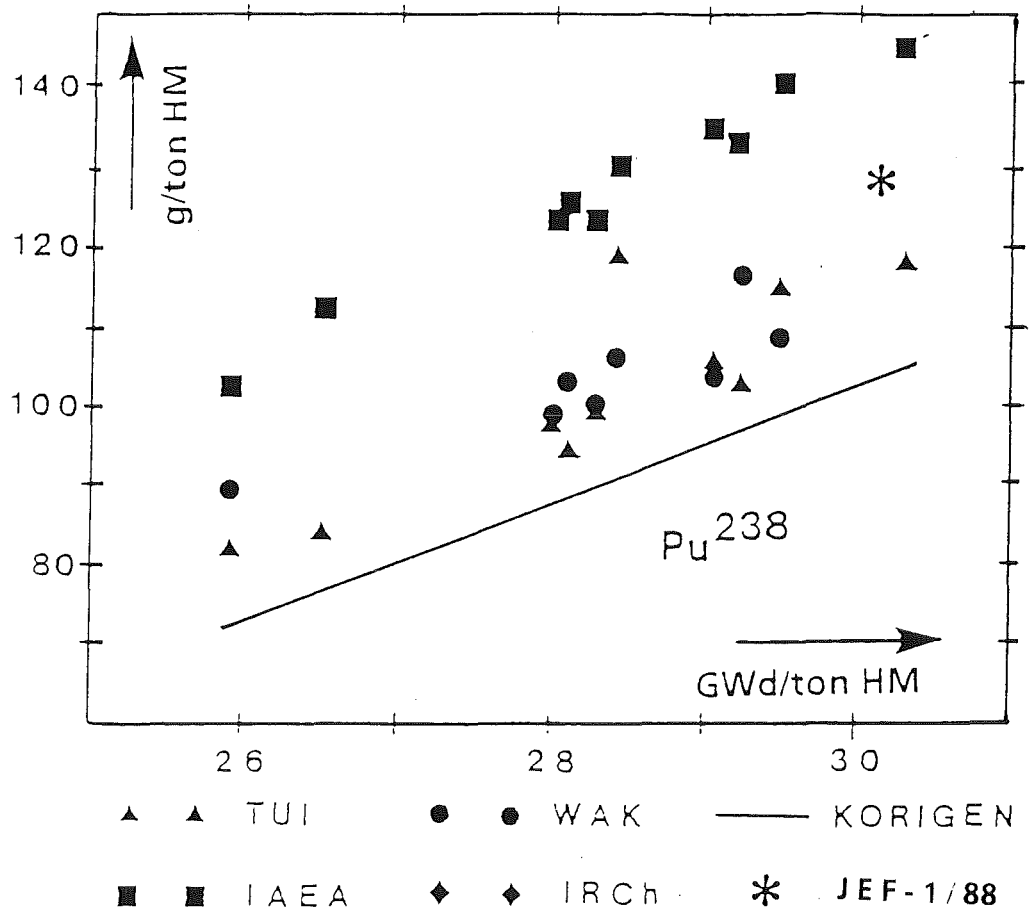


Fig. 4: Burnup Dependent Isotopic Concentrations for U235 and U236



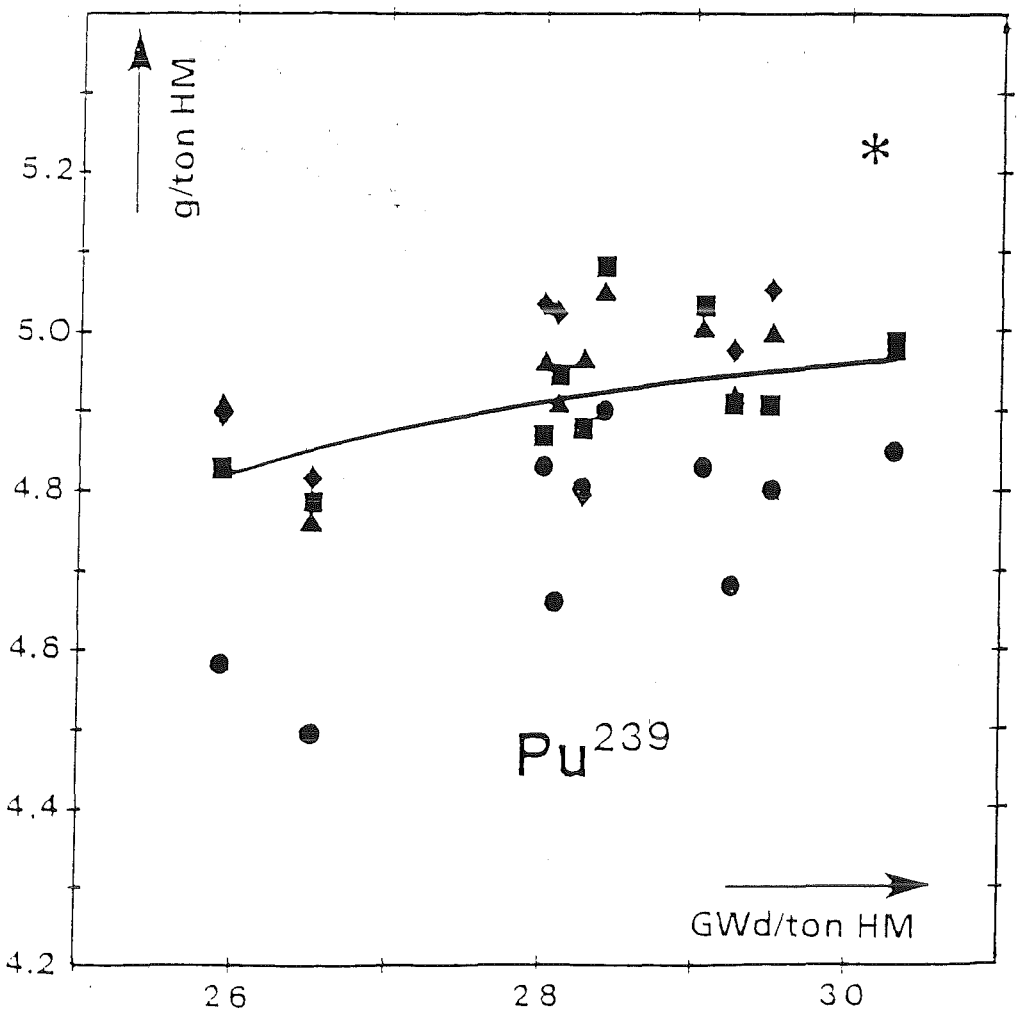
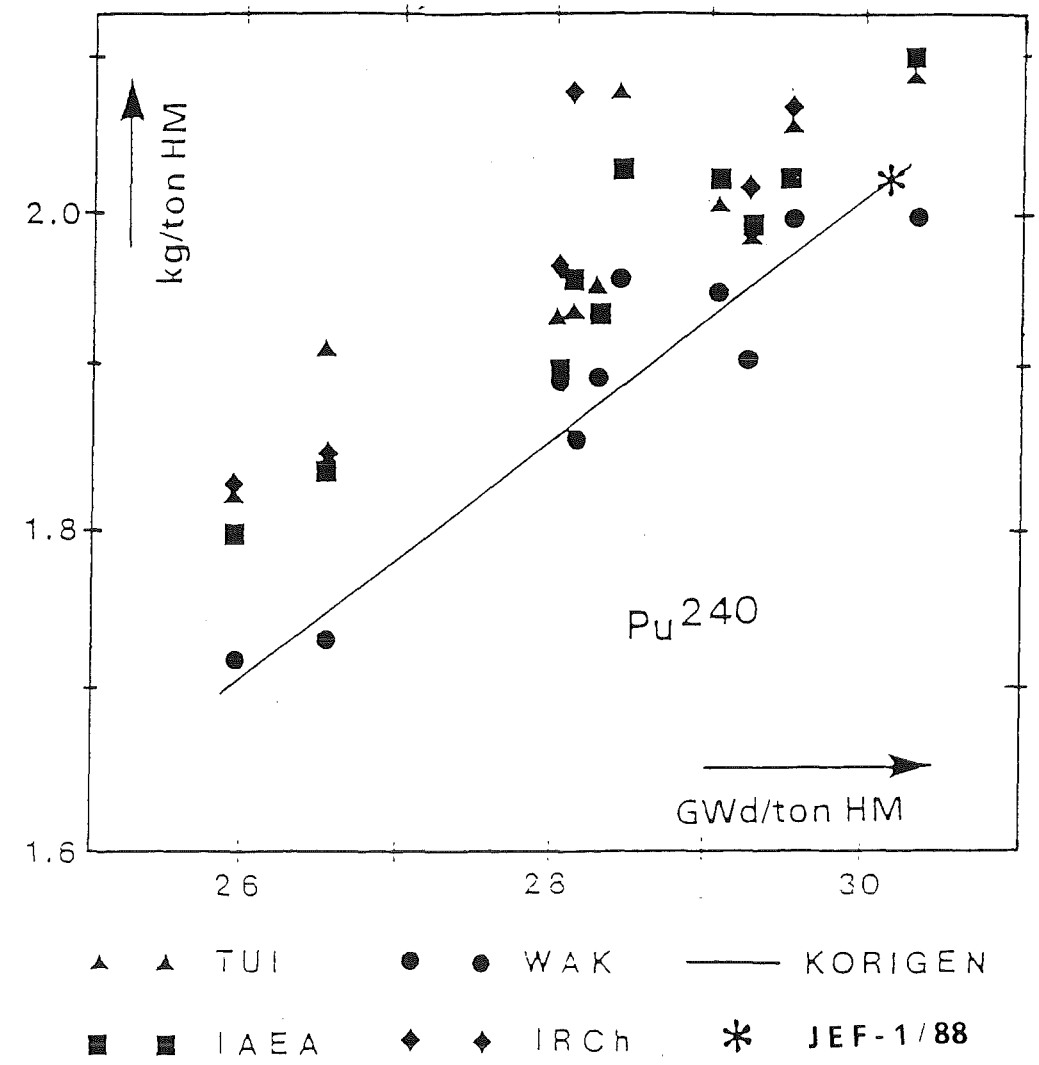


Fig. 5: Burnup Dependent Isotopic Concentrations for Pu238 and Pu239



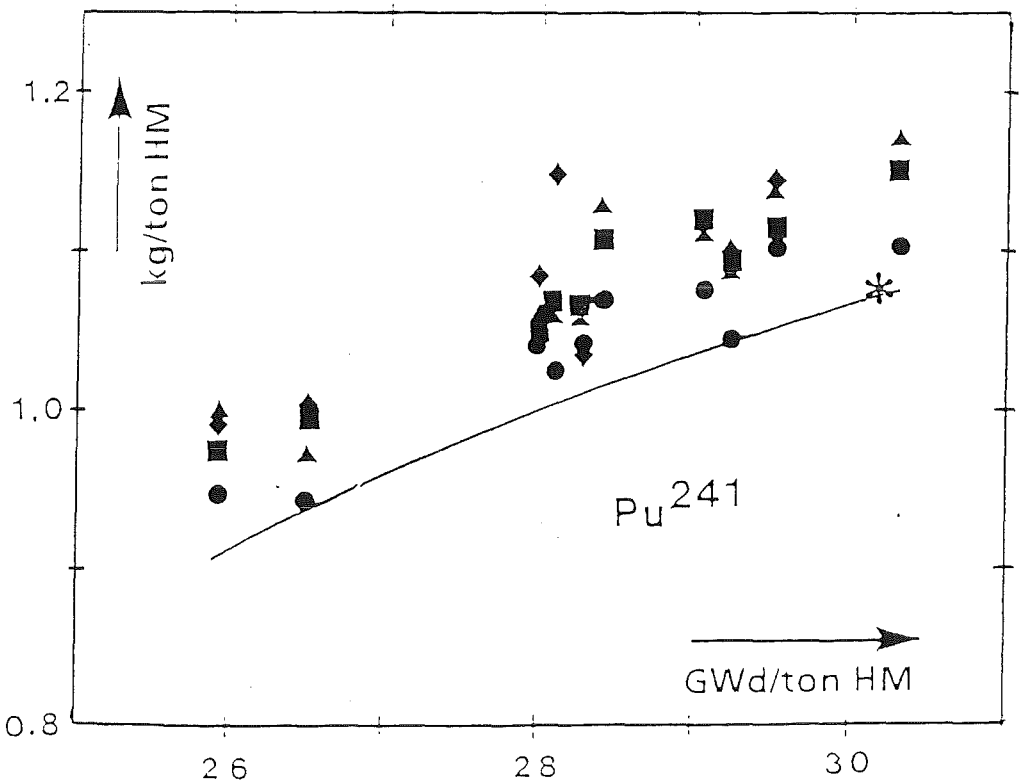


Fig. 6: Burnup Dependent Isotopic Concentrations for Pu240 and Pu241

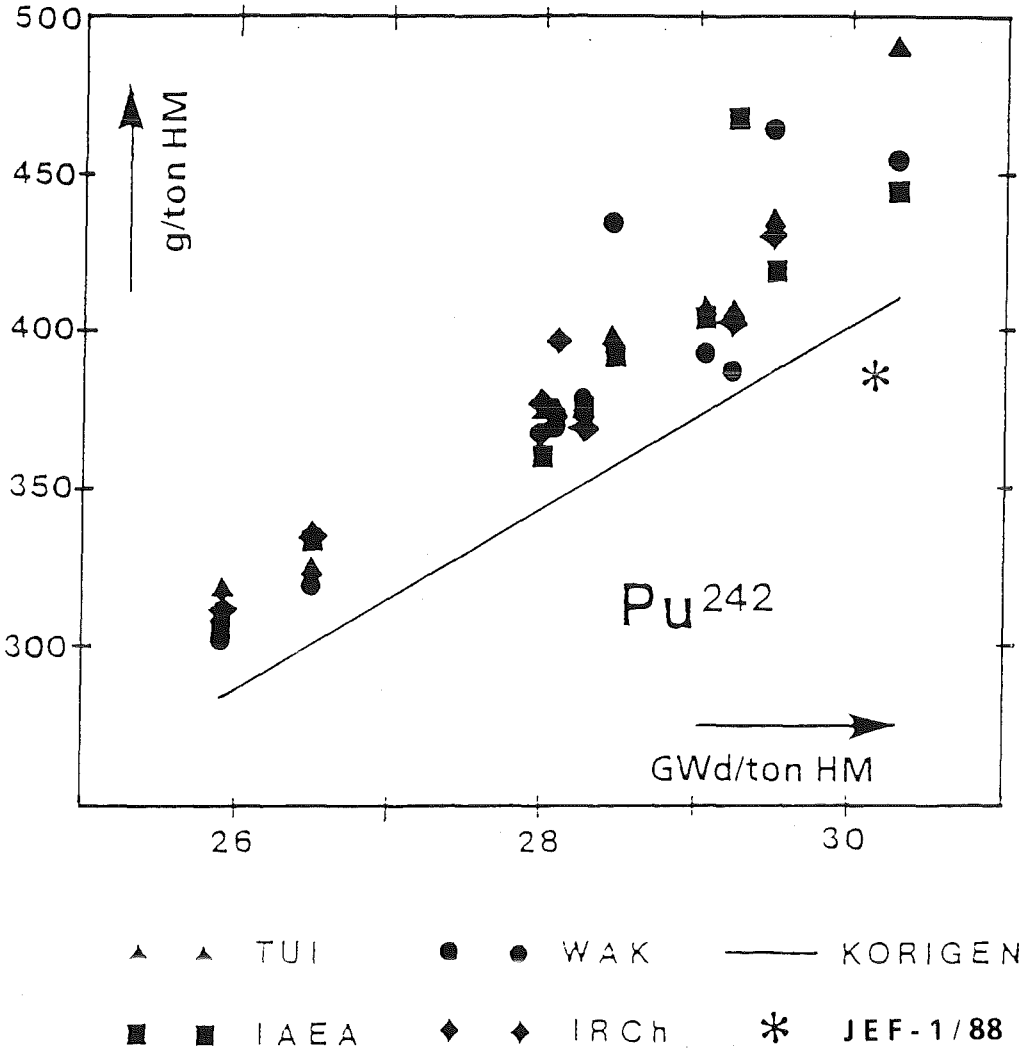
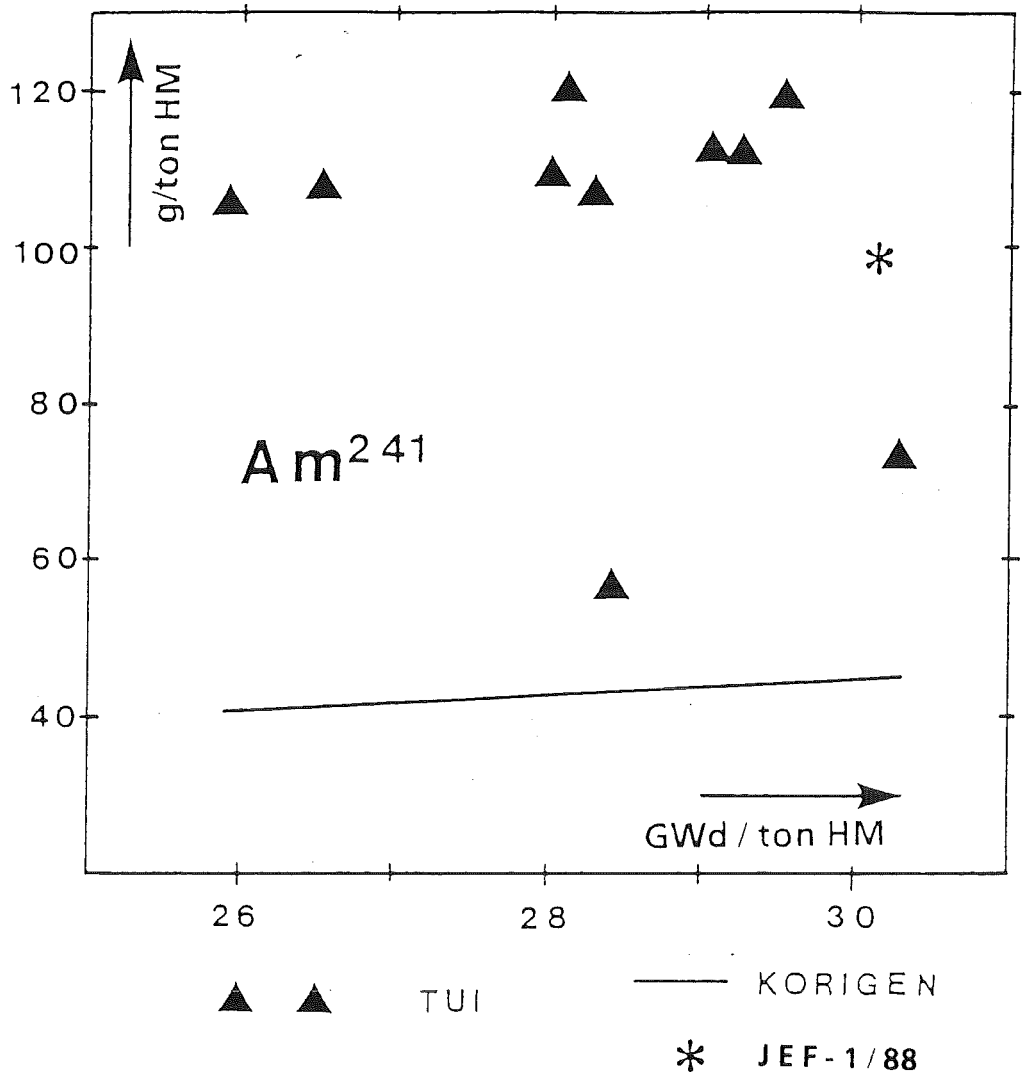


Fig. 7: Burnup Dependent Isotopic Concentration for Pu242



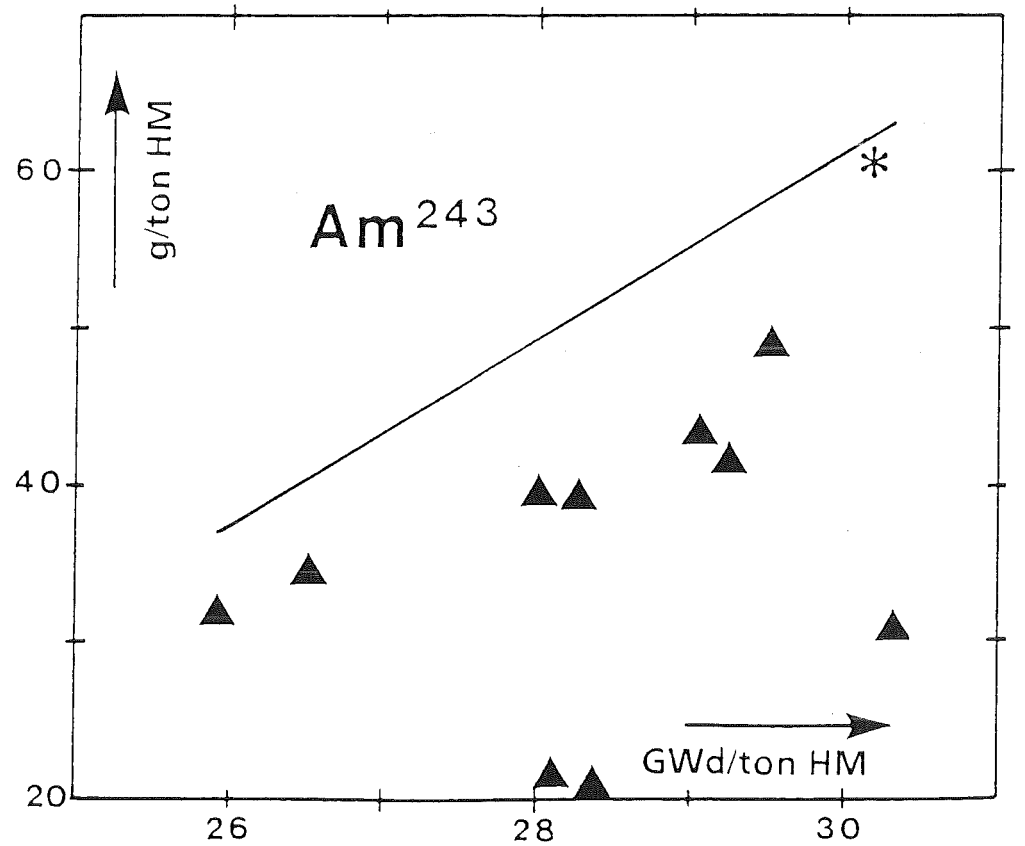


Fig. 8: Burnup Dependent Isotopic Concentrations for Am241and Am243

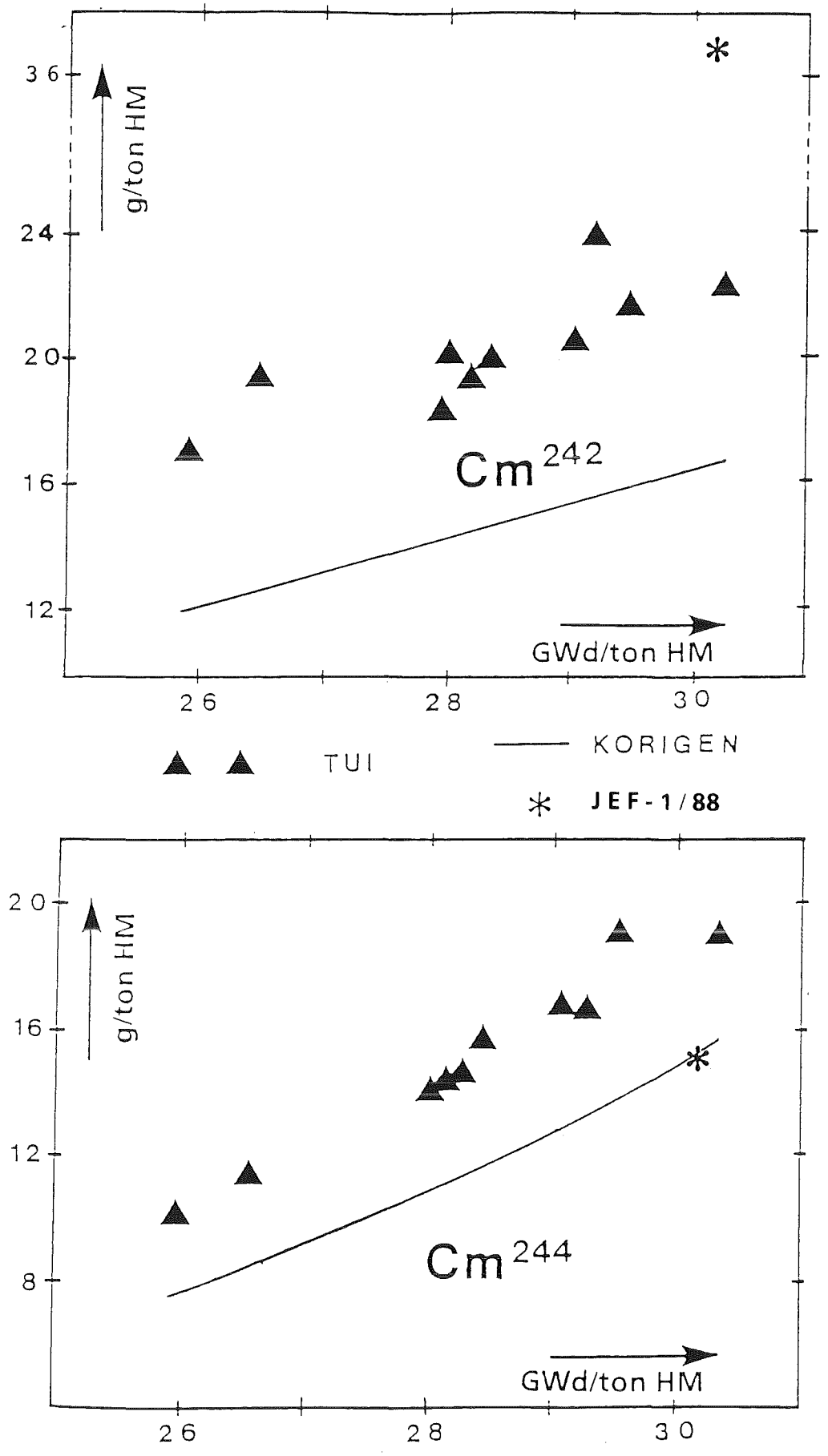


Fig. 9: Burnup Dependent Isotopic Concentrations for Cm242 and Cm244

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