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Verification of the OREST (HAMMER-ORIGEN) Depletion Program System Using Post-Irradiation Analyses of Fuel Assemblies 168, 170, 171 and 176 from the Obrigheim Reactor

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Abstract

Recyling calculations for UO_2 and mixed oxide (MOX) fuel are being carried out in conjunction with Project SR 316 entitled "Safety Analysis for Thermal Recycling in the Federal Republic of Germany." On the basis of the calculated nuclide inventories in spent fuel assemblies it is then possible to treat safety issues related to thermal recycling. The reliable determination of fission material quantities and their composition is therefore of particular importance.

The OREST program system [1] is used to calculate the nuclide inventories in spent UO_2 and MOX fuel assemblies.

This new program system represents a consistent combination of the well-known programs HAMMER [2] (neutron spectrum) and ORIGEN [3] (depletion calculation), with which a large number of boundary conditions such as reactor and fuel assembly type, burnup and fuel composition can be considered. It makes it possible to carry out the recycling calculations in conjunction with Project SR 316 in a physically satisfactory way.

The boundary conditions defined in SR 316 range from high burnups of as much as 55 GWd/tU to MOX charges with 3.4 wt% fissionable plutonium to the recycling of reprocessed uranium in UO_2 and MOX fuel assemblies, and therefore exceed the scope of application of the data bases of the common depletion codes ORIGEN, ORIGEN-2 and KORIGEN. Especially problematic in this case would be MOX depletion calculations for up to 40 GWd/tHM, since the only library available for mixed oxide (ORIGEN-2) is established only for 2.9 wt% Pu-fiss and 33 GWd/tHM for a Westinghouse reactor. Furthermore, no data base exists that could adequately describe the high burnups of over 50 GWd/tU or the reprocessed uranium inventory in UO_2 and MOX.

The above problems are avoided if, as in OREST, the depletion code is accompanied by a spectral code, which determines the neutron spectrum in the actual fuel mixture at the beginning of and during burnup and carries out resonance treatment for the most important uranium and transuranium isotopes.

In this report the reliability of the OREST system for UO_2 burnup in pressurized water reactors is first demonstrated. Post-irradiation analyses of five UO_2 fuel assemblies from the Obrigheim Pressurized Water Reactor (KWO) with an initial enrichment of 3.13 wt% U-235 and a mean burnup rate of 28.4 GWd/tU are used as a yardstick.

These fuel assemblies were reprocessed in 1977-78 in the Karlsruhe Reprocessing Plant (WAK),* whereby half a KWO** fuel assembly was measured separately in each WAK campaign. The post-irradiation analyses for a number of fissile material and fission product isotopes were carried out by four independent institutes within the framework of the "Isotopic Correlation Experiment" (ICE) [4]. This makes 40 measurement points available per isotope. These ICE measured data are compared with the OREST calculation results. The report also lists the results obtained at Kernforschungszentrum Karlsruhe with the KfK depletion code system "HAMMER-KORIGEN."

The good agreement between experiment and calculation, also with respect to the KfK results, proves the reliability of OREST for UO_2 fuel in pressurized water reactors. Additional calculations for other reactor types and other fuels (MOX) in progress.

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* WAK = Wiederaufarbeitungsanlage Karlsruhe.

** KWO = Kernkraftwerk Obrigheim [Obrigheim Nuclear Power Plant].

1. OREST DEPLETION CALCULATIONS

The OREST program system is made up of the depletion code ORIGEN [3] and the cell code HAMMER [2], which are coupled by means of several interface modules for the purpose of continuous calculation of fuel temperatures, spectral indices, uranium and transuranium cross sections, recoverable energy per fission, and above all for the purpose of data transfer.

With the HAMMER cell code [2] it is possible in the OREST system to consider the reactor data and fuel assembly geometries that apply to the Obrigheim PWR.

Thus the depletion of the measured KWO fuel assemblies 168, 170, 171, 172 and 176 can largely be simulated mathematically.

1.1 OREST Input Data

The reactor, fuel assembly and operating data given in Table 1 are used as the input data. They are based on the KWO design data and the fuel assembly data for a 14 x 14 rod lattice with 16 control rod guide tubes [8].

The moderation effect of the 16 water-filled control rod guide tubes is described by an effective pitch, which was calculated using the Monte Carlo code KENO-IV [7]. As expected, this effective pitch, at 1.475 cm, was somewhat above the geometric pitch of 1.43 cm. The 'cold' rod and pellet geometries change somewhat during burnup due to pellet swelling and downward creep of the fuel cladding. The conclusion is reached in [9] that there is mechanical contact between cladding and fuel during the power period. Measurements after burnup showed finally a definite reduction of the radial gap to one third. At the same time, slightly reduced cladding diameters were found.

Table 1 therefore compares the "cold" geometric data with the "hot" geometric data used in this research, taking into account gap reduction, cladding shrinkage and control rod guide tubes.

The coolant density is calculated analytically in OREST from the coolant pressure and temperature based on [11]. In all computer runs it yields the value of 0.7283 g/cm^3 and is a decisive input quantity for the burnup result along with the effective pitch. The high value of 0.789 g/cm^3 given in the analogous KfK calculations [12] could not be reproduced; it would correspond to an operating pressure of about 500 bar. The homogeneous boron poisoning in the reactor coolant was assumed to be 450 ppm in analogy with the KfK calculations.

The KWO fuel assemblies 168, 170, 171, 172 and 176 were used in Cycles 3 and 4, were unloaded in the 5th cycle and finally irradiated again in the 6th cycle. The corresponding reactor diagram in Figure 1, which was simulated in the calculations, is taken from Ref. [6] and was evaluated graphically in full-load and zero-load days. The resulting power history is shown in Table 2.

The burnups attained per fuel assembly and cycle were obtained from the operator [5]. The mean specific powers (MW/THM) per cycle and fuel assembly that were calculated from the full-load days of Table 2 and from these data are entered in Table 3.

Whereas in Cycle 6 practically all fuel assemblies exhibited the same powers, the values in Cycle 3 vary from 22 to 35 MW/THM and in Cycle 4 from 25 to 40 MW/THM.

In the dissolution process in WAK, half of a fuel assembly was processed and analyzed in each case. The correlation between these batches and the fuel assemblies is shown in Table 4. Column 2 in this table corresponds to the burnup measurements per batch; Column 3 gives the burnup measurement/operator data factors with which the batch burnups were attained in the OREST calculations with a given power history.

Fig. 1. Reactor operation diagram for Obrigheim, Cycles 3, 4, 6

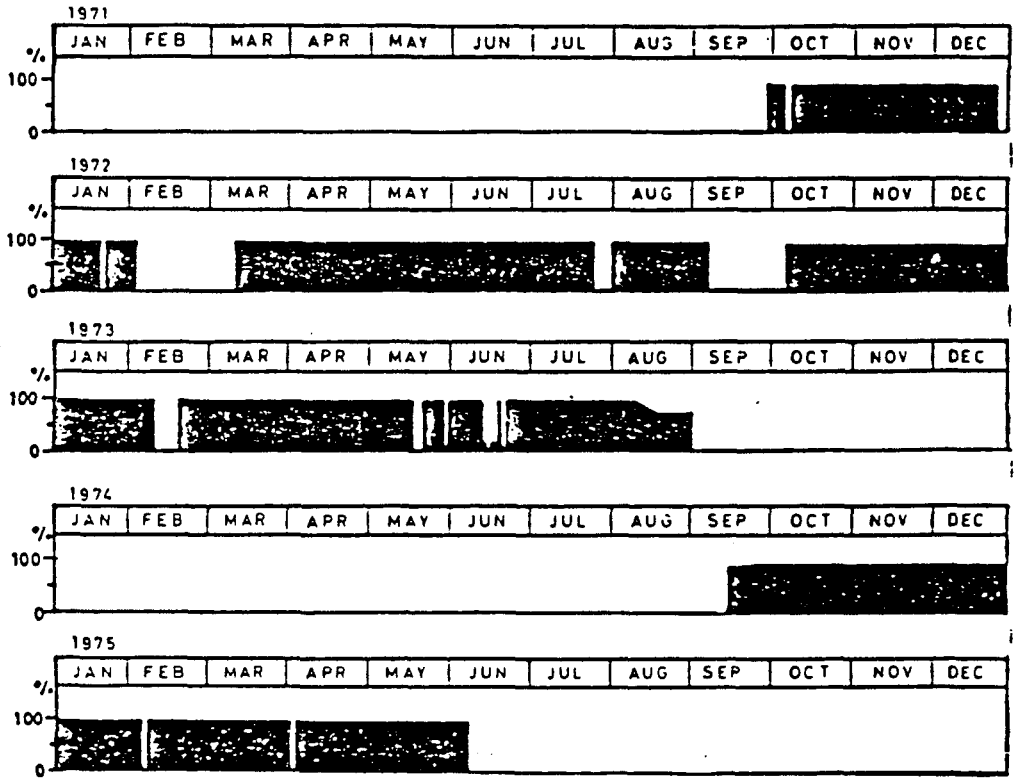


Table 1

OREST Input Data, KWO fuel assemblies 14 x 14 - 16

Pellet diameter	cold:	0.913 cm	hot:	0.925 cm
Cladding, inside	cold:	0.930 cm	hot:	0.930 cm
Cladding, outside	cold:	1.074 cm	hot:	1.071 cm
Lattice pitch		1.430 cm	effective:	1.475 cm

Coolant pressure:	151 bar
Coolant temperature:	299 °C
Void content:	0 %
Resulting coolant density:	0.7283 g/cm
Boron:	450 ppm

Fuel temperature:	depending on power drawn
Cladding temperature:	" " " "

Fuel density:	10.2 g/ccm cold without dishing
	10.05 g/cm with dishing vol. 1.5%
	9.742 g/cm effective hot
Enrichment:	3.13 wt % U-235
	0.03 wt % U-234

Table 2

OREST Input Data

Power history of fuel assemblies 168, 170, 171, 172, 176, in accordance with a graphic interpretation of the diagram in Fig. 1

Days Cycle 3	Operating State	Days Cycle 4	Operating State	Days Cycle 6	Operating State
4.9	FL[1]	124.2	FL	127.8	FL
2.4	ZL[2]	8.5	ZL	2.4	ZL
79.1	FL	90.1	FL	54.8	FL
3.7	ZL	2.4	ZL	1.2	ZL
18.3	FL	9.7	FL	65.7	FL
1.2	ZL	1.2	ZL		
12.2	FL	13.4	FL		
37.7	ZL	4.9	ZL		
136.5	FL	1.2	FL		
6.8	ZL	2.4	ZL		
37.0	FL	70.6	FL		
29.2	ZL				
369	total	328	total	252	total
288	FL total	309	FL total	248	FL total
81	ZL total	19	ZL total	4	ZL total
		Cycle 5			
		377	ZL		

[1] FL = full load

[2] ZL = zero load

Table 3

OREST Input Data

Spec. power and final burnups according to operator data for fuel assemblies 168, 170, 171, 172 and 176

	Cycle 3 MW/tHM	Cycle 4 MW/tHM	Cycle 5 MW/tHM	Cycle 6 MW/tHM	Final Burnup GWd/tHM
BE-168	29.95	40.17	0	36.20	30.018
BE-170	22.41	40.12	0	35.94	27.764
BE-171	30.14	40.31	0	35.96	30.052
BE-172	35.41	25.45	0	35.96	26.979
BE-176	32.26	36.96	0	36.03	29.647

The specific powers were calculated from the full-load days and the operator's burnup data

Table 4

Connection between WAK Batches and KWO Fuel Elements

Batch No.	Burnup Measurement [10] GWd/THM	Measurement/ Operator Data	Fuel Assembly Matching as per [12]
No. 86	28.40	0.9461	No. 168
87	30.30	1.0094	168
88	28.27	0.9407	171
89	29.04	0.9663	171
90	29.52	0.9957	176
91	27.99	0.9441	176
92	26.54	0.9837	172
93	29.24	1.0838	172
94	25.93	0.9339	170
95	28.08	1.0114	170

1.2 Computation Sequence

OREST calculations were carried out for each of the 10 batches with the power histories as shown in Table 2.

The UO_2 fuel temperature important for the Doppler broadening of neutron capture resonances (U-238, Pu-240) is not rigidly specified in OREST, in contrast to [10], but is continuously re-determined from rod geometry, heavy metal density and power drawn during burnup using a rod temperature program and material data as given in [13]. Figure 2 shows the relationship between rod power and mean fuel temperature as calculated by OREST on the basis of the KWO input data. Fuel temperatures, neutron spectra and actinide cross sections were tracked about 6-7 times per batch.

1.3 OREST Computation Results

A selection of the calculated nuclide concentrations of the most important heavy metal isotopes is shown in Table 5.

The values are given in grams per ton of charged uranium. They correspond to the time of fuel unloading after KWO Cycle 6, since the experimentally determined nuclide concentrations are also related to this point in time.

Fig. 2. Fuel temperature and rod power for KWO

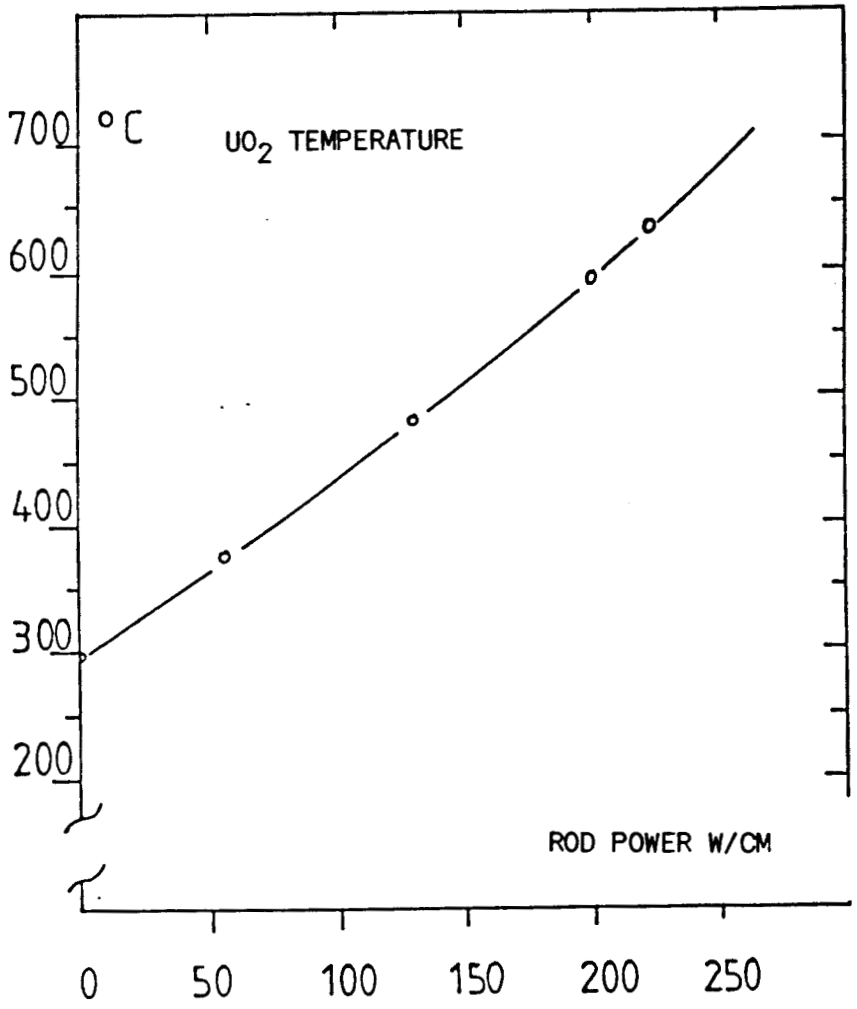


Table 5

Calculated nuclide concentrations of Batch 86-95 in gram/ton uranium

Element/Isotope		Batch No. 86	87	88	89	90	91	92	93	94	95
Uranium	235	9.66(3)*	8.84(3)	9.71(3)	9.37(3)	9.17(3)	9.84(3)	1.05(4)	9.28(3)	1.08(4)	9.79(3)
	236	3.58(3)	3.68(3)	3.57(3)	3.61(3)	3.64(3)	3.55(3)	3.46(3)	3.63(3)	3.42(3)	3.56(3)
Neptunium	237	3.68(2)	3.99(2)	3.66(2)	3.79(2)	3.86(2)	3.60(2)	3.34(2)	3.81(2)	3.26(2)	3.63(2)
Plutonium	238	1.01(2)	1.17(2)	1.00(2)	1.07(2)	1.11(2)	9.84(1)	8.71(1)	1.09(2)	8.13(1)	9.79(1)
	239	5.07(3)	5.15(3)	5.07(3)	5.10(3)	5.10(3)	5.05(3)	4.99(3)	5.11(3)	4.99(3)	5.09(3)
	240	2.18(3)	2.32(3)	2.17(3)	2.23(3)	2.26(3)	2.15(3)	2.05(3)	2.25(3)	2.01(3)	2.16(3)
	241	1.13(3)	1.19(3)	1.12(3)	1.15(3)	1.17(3)	1.11(3)	1.04(3)	1.15(3)	1.01(3)	1.12(3)
	242	4.24(2)	4.93(2)	4.19(2)	4.47(2)	4.65(2)	4.09(2)	3.60(2)	4.55(2)	3.42(2)	4.14(2)
Americium	241	3.86(1)	3.98(1)	3.88(1)	3.94(1)	3.93(1)	3.83(1)	3.47(1)	3.65(1)	3.36(1)	3.54(1)
	243	6.16(1)	7.52(1)	6.07(1)	6.63(1)	7.01(1)	5.87(1)	4.90(1)	6.74(1)	4.54(1)	5.95(1)
Curium	242	1.79(1)	2.03(1)	1.78(1)	1.87(1)	1.96(1)	1.76(1)	1.65(1)	2.01(1)	1.45(1)	1.72(1)
	244	1.56(1)	2.09(1)	1.53(1)	1.73(1)	1.87(1)	1.46(1)	1.14(1)	1.79(1)	1.03(1)	1.49(1)

* 9.66(3) corresponds to $9.66*10^3$ gram/ton uranium

2. ICE POST-IRRADIATION ANALYSES, COMPARISON BETWEEN CALCULATION AND MEASUREMENT

In conjunction with the Isotopic Correlation Experiment (ICE), the nuclide composition of the KWO fuel assemblies 168, 170, 171, 172 and 176, which in the 3rd, 4th and 6th KWO cycles reached burnups ranging between 26 and 30 GWd/THM, was analyzed by the following four laboratories independently from one another in 1977-78: Europäisches Institut für Transurane (TUI) [European Institute of Transuranium Elements], Institut für Radiochemie (IRCH) [Institute of Radiochemistry], Wiederaufarbeitungsanlage Karlsruhe (WAK) [Karlsruhe Reprocessing Plant], and the International Atomic Energy Agency (IAEA).

In reprocessing at WAK it was possible in each case for half a fuel assembly to be processed as a batch, so that because of the 4 laboratories there were a maximum of 40 measuring points for nuclide concentrations.

The comparisons refer to the measured data published in the KfK reports [10] and [12].

2.1 Experimental Burnup Determination

The present radiochemical methods (Nd-148, Cs-137) for burnup determination involve inaccuracies of about +/- 4%, according to [12], since along with measurement uncertainties the input already involved assumptions about the final burnup, in order to be able to recalculate the desired burnup value via an effective fission yield per fission and an effective energy per fission.

According to [10] the burnup determination for all WAK batches was performed by only one laboratory using the Nd-148 method. The values determined agree with the operator data to within 3 to 4%. A more precise burnup determination would certainly be desirable, in order to facilitate a more precise comparison of calculated and measured nuclide inventories.

2.2 Measurement Results

The measurement results were taken from the KfK references [10] and [12], since the original measurements were not available. This means that many data are determined graphically from the figures published in these reports.

The analyses, which were carried out three years after the fuel unloading date, were calculated back to the unloading date by the institutes. According to [12], these corrections can bring about uncertainties, especially with Pu-238 and the americium and curium isotopes. Some of the measurements are given in IMA (initial metal atom) [10]. The factor for conversion to ORIGEN moles that we determined is 4.203×10^3.

Table 6 gives the measured nuclide concentrations per ton uranium used in the reactor for important isotopes from uranium to curium. The values for uranium and plutonium, converted from [10] into grams, correspond to mean values for four measuring points each (see figures in appendix). Americium and curium values are derived graphically from the figures in [12]. In this case there was only one measuring point per batch and isotope.

Figures A1 to A6 in the appendix based on [12] give the measured nuclide concentrations as a function of burnup. At the same time, isotope-specific measuring uncertainties become clear.

Table 6

ICE measurement results for batches 86 to 95, relative to the time of unloading, in gram/ton uranium

Element	Isotope	Batch No. 86	87	88	89	90	91	92	93	94	95
Uranium[1])	235	9710	8981	9831	9482	9134	9874	10594	9065	10945	9917
	236	3733	3836	3729	3746	3807	3694	3623	3737	3590	3711
Plutonium[1])	238	119	150	114	115	124	111	117	120	94	113
	239	5078	4937	4880	4965	4933	4923	4716	4870	4808	4944
	240	2026	2066	1924	1997	2038	1926	1810	1979	1798	1963
	241	1103	1142	1050	1102	1124	1061	979	1082	977	1074
	242	409	449	374	402	437	370	328	416	310	379
Americium[2])	241	57	74	107	112	119	109	108	112	105	120
	243	20.5	31.2	39.0	43.4	48.8	39.5	34.6	41.5	32.2	22.0
Curium[2])	242	20.2	22.5	19.5	20.7	21.9	18.5	19.6	24.0	17.2	20.4
	244	15.9	19.2	14.8	17.0	19.3	14.2	11.6	16.8	10.3	14.6

1) according to [10], p. 53, IMA data converted to gram/THM, 4 measuring points per batch

2) according to [12], p. 92 and p. 93, read graphically, 1 measuring point per batch

2.3 Comparison between Calculation and Measurement

Table 7 gives the differences between the measured results and the calculated values. The last column contains the mean value of the respective OREST deviations. Negative values mean that the calculated values are below the measured results.

The OREST calculation results are plotted in Figures A1 to A6 in the appendix as a function of burnup for nuclides U-235, U-236, Pu-238, Pu-239, Pu-240, Pu-241, Pu-242, Am-241, Am-243, Cm-242 and Cm-244.

Very good agreement is achieved in the case of uranium 235 with a deviation of only 1%. Calculated and measured values coincide for all practical purposes, as shown by Fig. 1 in the appendix.

The underestimation of U-236 by 4% is slight; however, compared with a measurement uncertainty [12] of 1%, production rates are apparent that are somewhat too low. The calculated values for U-236 are plotted graphically against the measured results in Fig. A 1.

For Pu-238 an average of about 14% lower concentrations are calculated. As shown by Fig. A 2, the measurement uncertainty is of about the same order of magnitude. The extent to which these Pu-238 differences, which were also found in KfK verification calculations, are due to the measurement techniques (see [12]) cannot be evaluated here.

The concentration of bred Pu-239 as determined by OREST is still more or less within the range of measurement accuracy of 3% (according to [12]), although with a slight excess (+3.4%). However, Table 6 and Figure A 2 shows that there is a definite overestimation, which because of subsequent activation processes has an effect on the higher plutonium isotopes Pu-240 (+12%, Fig. A 3), Pu-241 (+5%, Fig. A 4) and Pu-242 (+9%, Fig. A 4). In contrast, the KfK verification calculations showed lower concentrations, as also shown by Figures A 3 and A 4, so that the measured values lie between the two burnup calculations.

Surprising differences between calculation and measurement were found for the americium isotopes Am-241 and Am-243, whereby KORIGEN and OREST calculations yield relatively similar results (Figures A 5). The variation between measured values is great. Per batch there is additionally only one point of one institute available.

The reason for the excessively high Am-241 measurements and the excessively low Am-243 analysis values, comparatively, is to be found in measurement difficulties, according to [12]. Subsequent precision measurements of pellets from KWO-BE-210 and 124 did yield values that are in agreement with KfK depletion calculations.

Table 7

Percentage deviations of OREST calculation results from the measured values (calculation-experiment/experiment)*100

Element	Isotope	Batch No. 86	87	88	89	90	91	92	93	94	95	Mittel
Uranium	235	-0.5 %	-1.6 %	-1.2 %	-1.2 %	0.4 %	-0.3 %	-0.9 %	2.4 %	-1.3 %	-1.3 %	-0.6 %
	236	-4.1	-4.1	-4.3	-3.6	-4.4	-3.9	-4.5	-2.9	-4.7	-4.1	-4.1
Plutonium	238	-15.1	-22.0	-12.3	-7.0	-10.5	-11.4	-25.6	-9.2	-13.5	-13.4	-14
	239	-0.2	4.3	3.9	2.7	3.4	2.6	5.8	4.9	3.8	3.0	3.4
	240	7.6	12.3	12.8	11.7	10.9	11.6	13.3	13.7	11.8	10.0	11.6
	241	2.4	4.2	6.7	4.4	4.1	4.6	6.2	6.3	3.4	4.3	4.7
	242	3.7	9.8	12.0	11.2	6.4	10.5	9.8	9.4	10.3	9.2	9.2
Americium	241*	-32	-46	-64	-63	-67	-65	-68	-67	-68	-71	-61(?)
	243*	200(?)	141(?)	56	53	44	49	42	62	41	170(?)	86(?)
Curium	242*	-11	-10	-9	-10	-11	-5	-16	-16	-16	-16	-12
	244*	-1.9	8.9	3.4	1.8	-3.1	2.8	-1.7	6.5	0.0	2.1	2.0

* only 1 measuring point per batch; very uncertain analytical values except for Cm-244

Within the framework of measurement accuracy, the curium isotopes Cm-242 and Cm-244 were determined very effectively. Although the Cm-244 build-up depends in highly non-linear fashion on burnup, the OREST results exhibit constant accuracy (see Figures A 6).

The following table, Table 8, gives a brief overview of the calculation accuracies achieved using the depletion codes OREST and KORIGEN independently of one another. The last column reproduces the measurement uncertainties listed in [12].

The OREST-KORIGEN deviations from the experiment are plotted graphically in Figure 3.

Table 8 Comparison between experiment and calculation as regards nuclide concentrations in fuel assemblies 168 through 176

Isotope	Deviation *) OREST	Deviation *) KORIGEN [12]	Measurement Uncertainty [12]
U- 235	-0.6 %	-1.9 %	±2 %
236	-4.1 %	-1.9 %	±1 %
Pu- 238	-14.0 %	-23.3 %	±15 %
239	3.4 %	0.4 %	±3 %
240	11.6 %	-5.1 %	±3 %
241	4.7 %	-6.0 %	±4 %
242	9.2 %	-9.0 %	±4 %
Cm-242	-12 %	ca -30 %	large, up to a factor of 2
244	2 %	ca -23 %	±20 %

*) Deviation in percent equivalent to (calculation-experiment)/experiment*100

FIG. 3 Comparison between calculation and measurement for KWO fuel assemblies 168, 170, 171, 172, 176

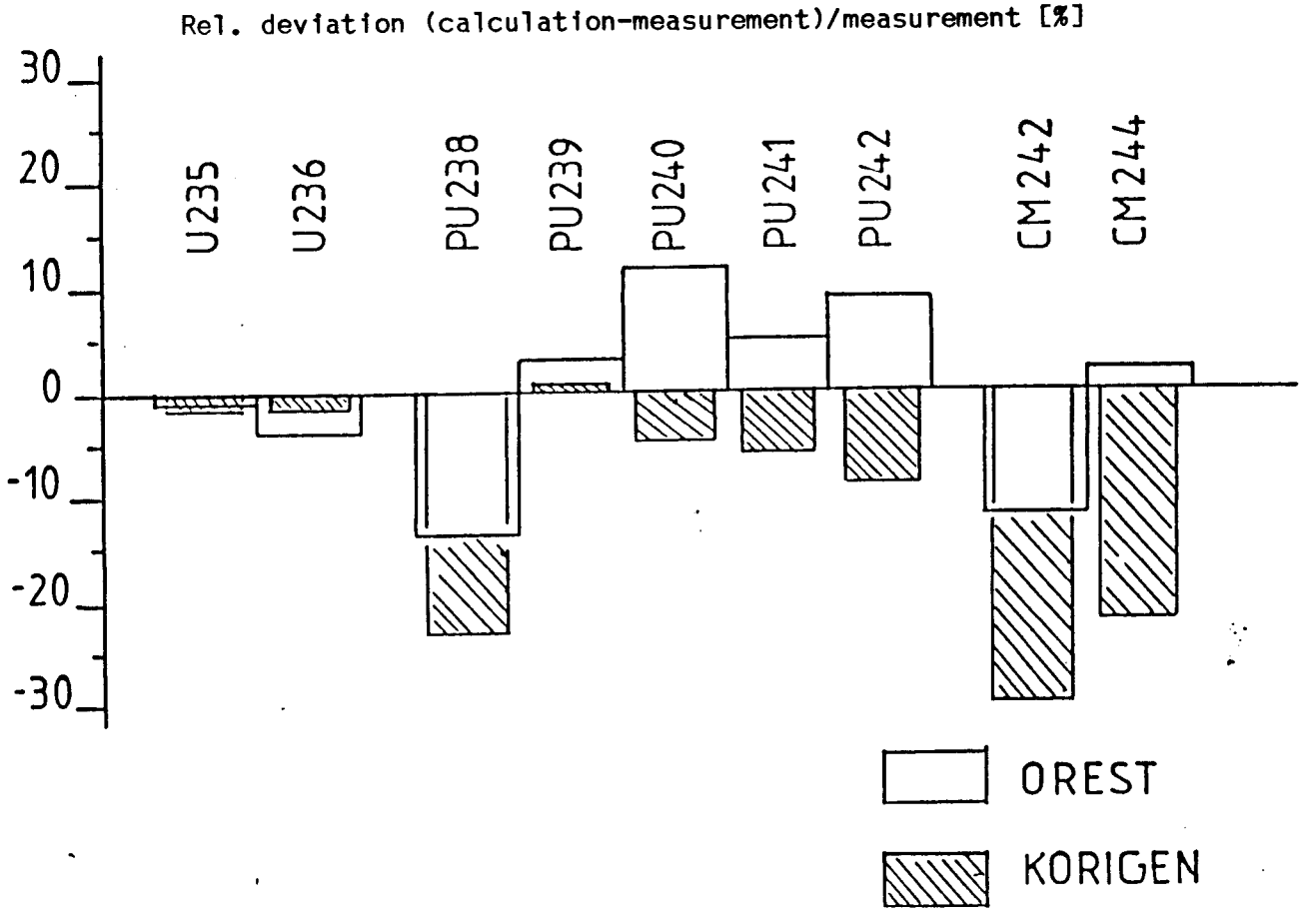
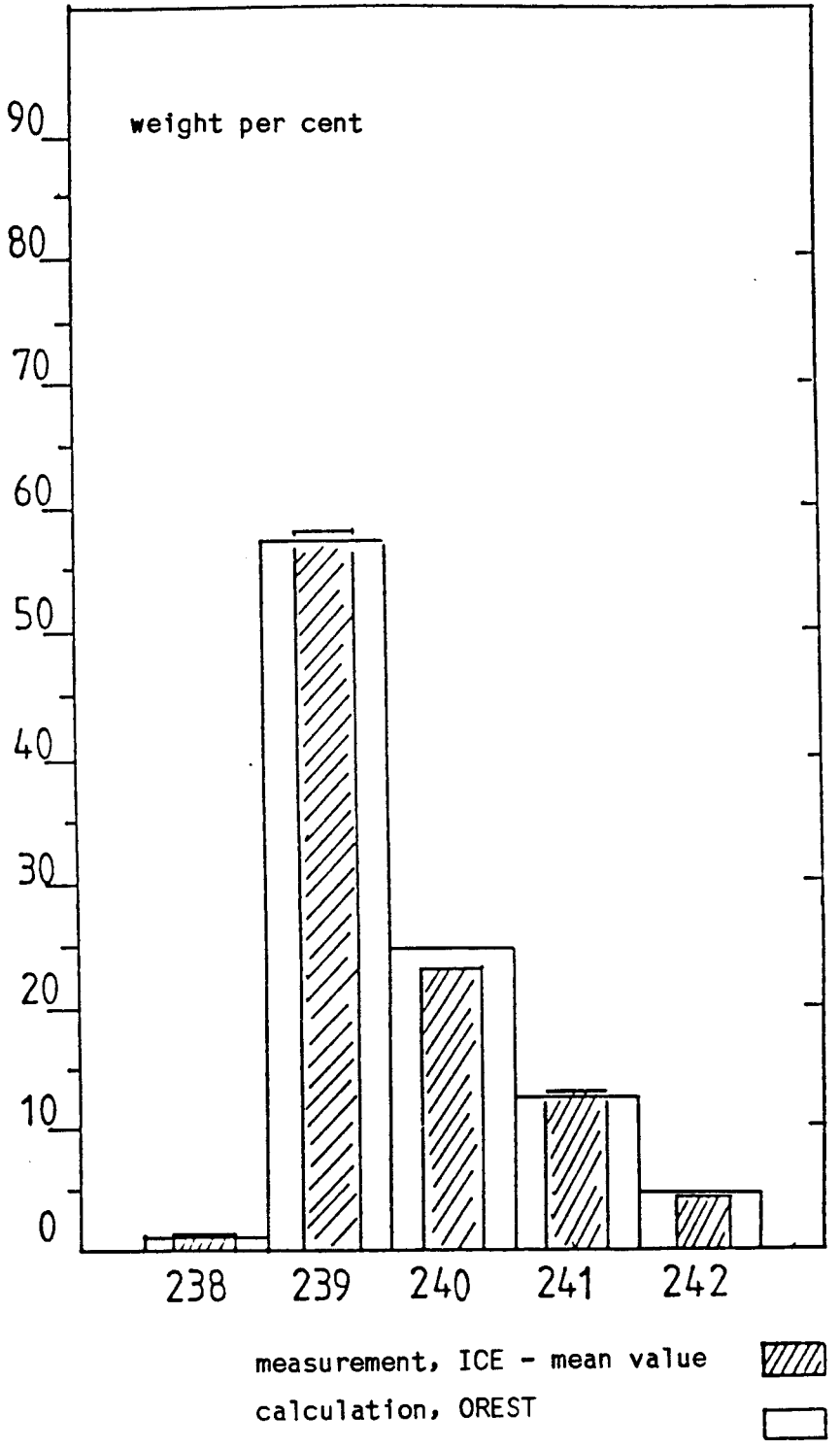


Fig. 4 Plutonium isotope distribution



The deviations from uranium to curium that go beyond the uncertainty range of the measurements are below 12% (Pu-240), but are systematic in nature, as shown by Table 7. The cause may lie in the following:

- in the HAMMER base data. In OREST the cell code results are adopted without external re-calculation (as in [10]). Here we would like a review based on the most recent core data;

- in real deviations from the model assumption of an extended rod lattice due to specific fuel assembly positions in the core and local perturbations in the vicinity of the fuel assembly pieces and top and bottom fittings. Such effects can only be considered at considerable additional expense.

A summary of the calculated and measured values for the Pu isotopes according to Tables 4 and 5 yields the Pu vector that applies to a burnup of about 28 GWd/THM, as given in Table 9.

Table 9 Plutonium vector with an average burnup of 28.3 GWd/THM, comparison between OREST and experiment

Isotope	OREST	EXPERIMENT
Pu-238	1.14 %	1.40 %
Pu-239	57.04 %	58.17 %
Pu-240	24.49 %	23.16 %
Pu-241	12.58 %	12.68 %
Pu-242	4.75 %	4.59 %
Total amt.	8.893 kg	8.423 kg

The total amount of Pu determined by OREST differs from the experiment by 6%. Figure 4 shows a graphic comparison of these plutonium vectors. The agreement by isotopes between the calculated and measured distributions is apparent.

In summary we can state that the post-irradiation analyses of the KWO fuel assemblies 168 to 176 can be reproduced effectively using the OREST program system.

Differences from the analysis values that go beyond the range of measurement inaccuracies remain below 12% (Pu-240). A comparison of the experimental and calculated plutonium vectors for an average burnup of 28.3 GWd/THM shows good agreement. The entire Pu amount is overestimated by OREST by 6%.

The concentrations of the radiation-relevant isotopes Pu-238, Cm-242 and Cm-244 are accurately determined in the burnup range from 26 to 30 GWd/THM that was the subject of the research, even in comparison with other verification calculations.

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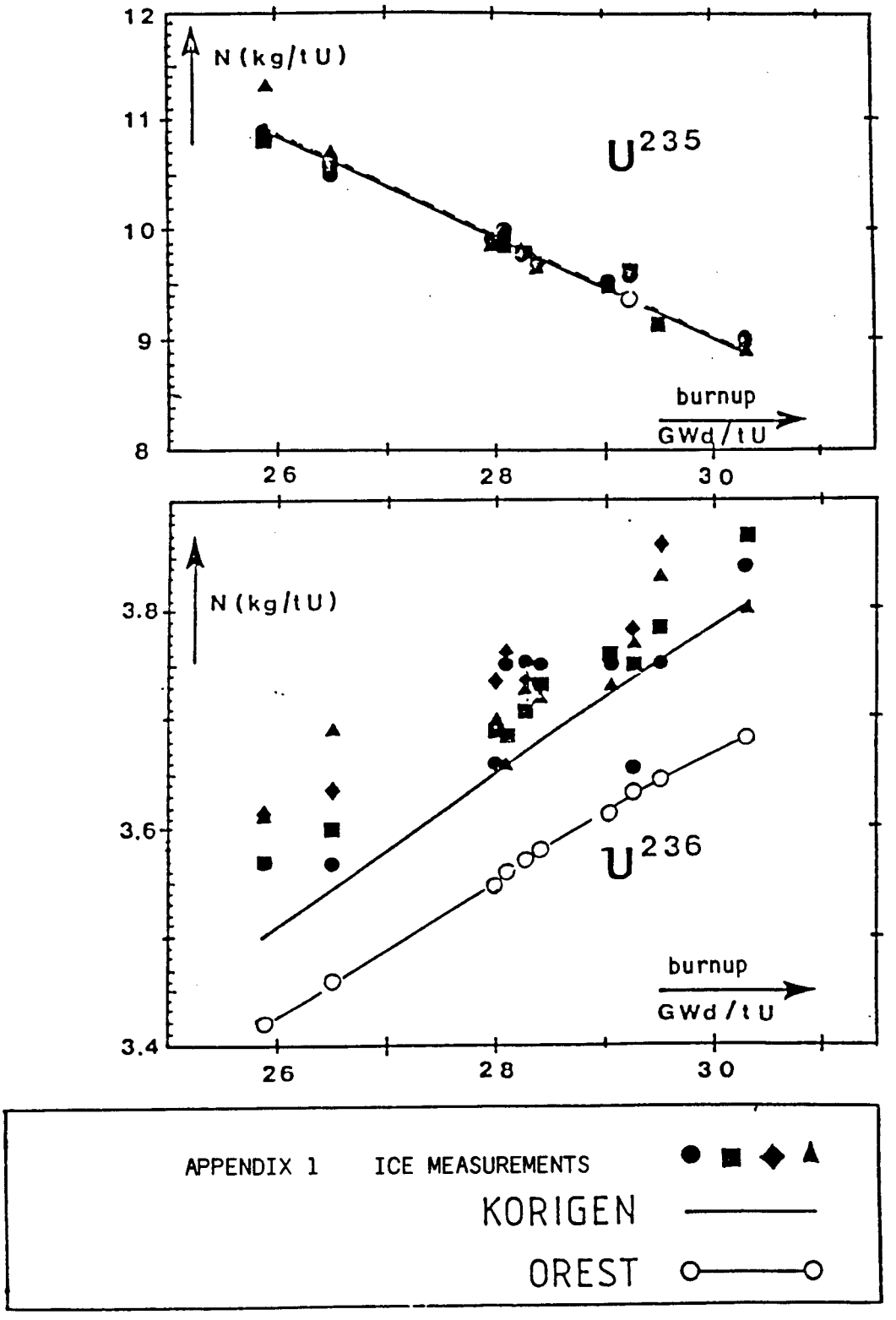
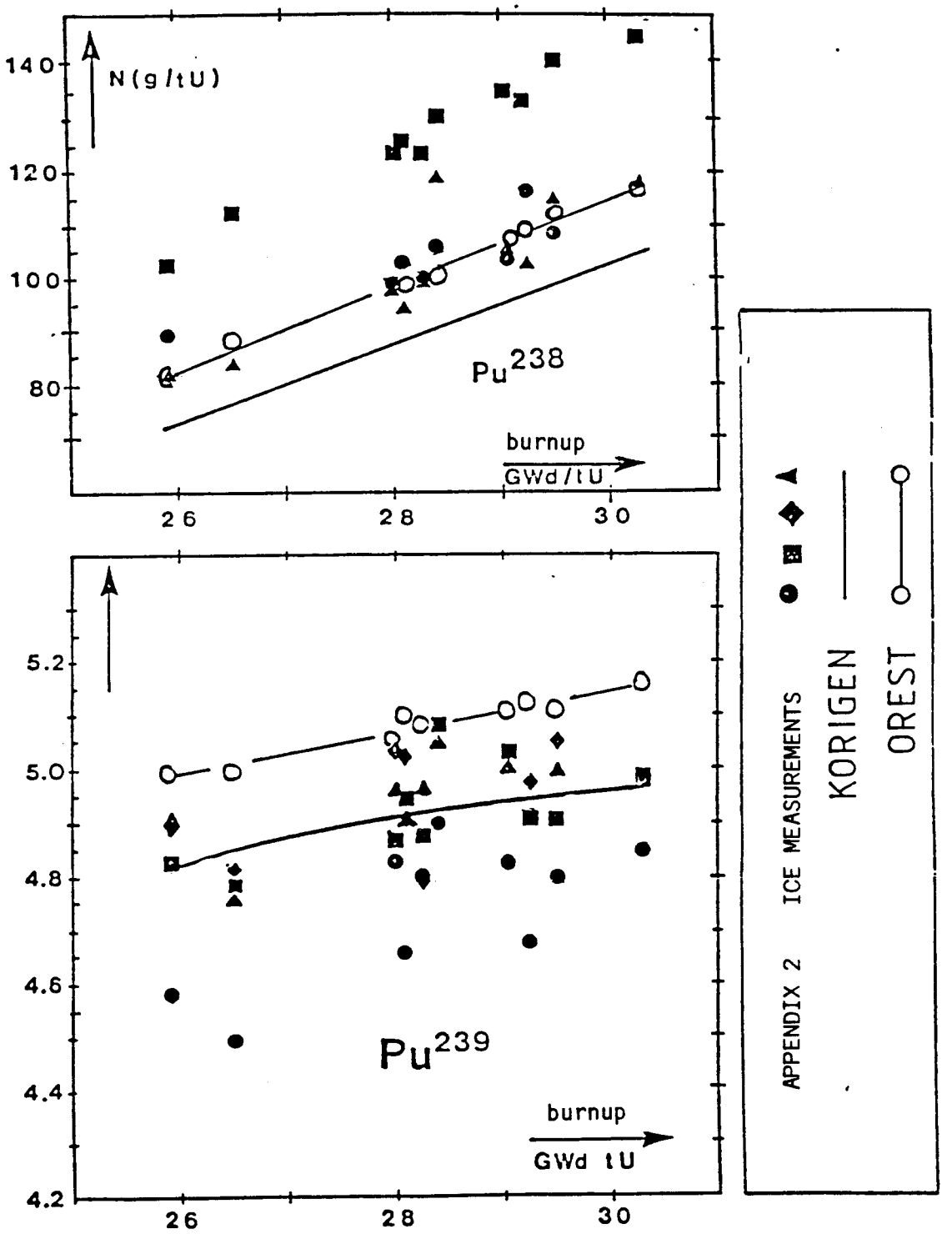
Appendix: Comparison of calculated and experimental nuclide concentrations of uranium, plutonium, americium and curium isotopes

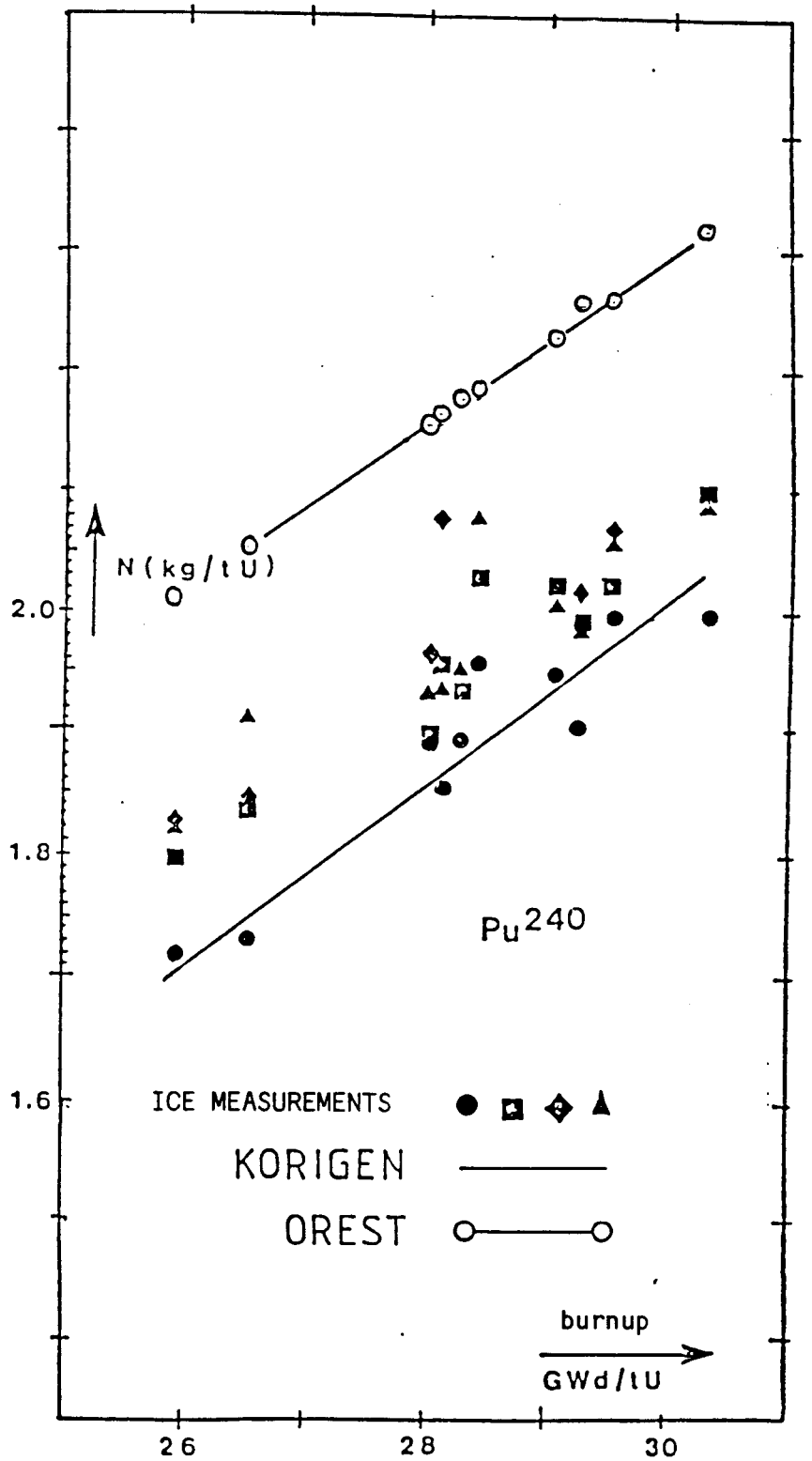
Figure A 1: U-235 nuclide concentrations (kg/tU)
U-236 nuclide concentrations (kg/tU)
Figure A 2: Pu-238 nuclide concentrations (kg/tU)
Pu-239 nuclide concentrations (kg/tU)
Figure A 3: Pu-240 nuclide concentrations (kg/tU)
Figure A 4: Pu-241 nuclide concentrations (kg/tU)
Figure A 5: Pu-242 nuclide concentrations (kg/tU)
Am-241 nuclide concentrations (kg/tU)
Am-243 nuclide concentrations (kg/tU)
Figure A 6: Cm-242 nuclide concentrations (kg/tU)
Cm-244 nuclide concentrations (kg/tU)

The figures are taken from [12].

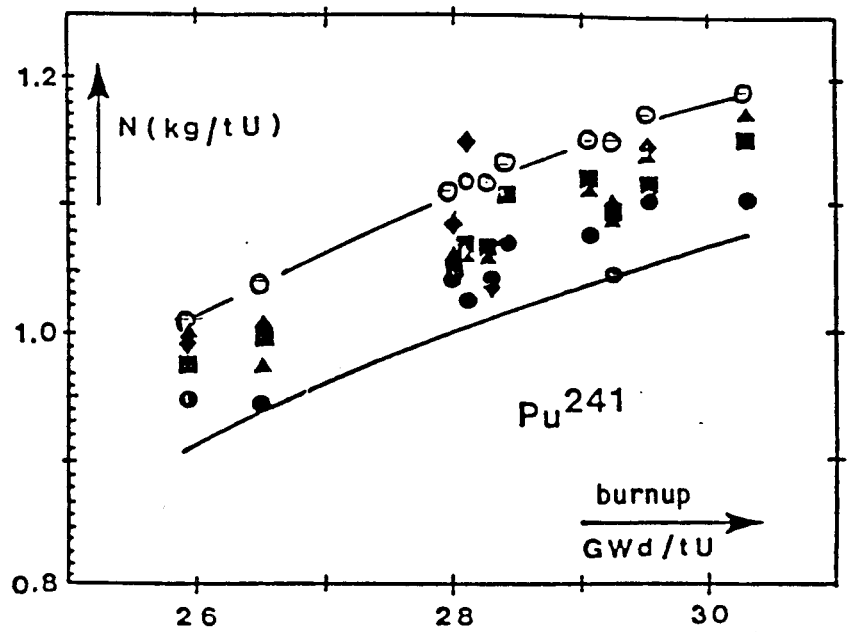
The OREST calculation results from this report are also plotted.

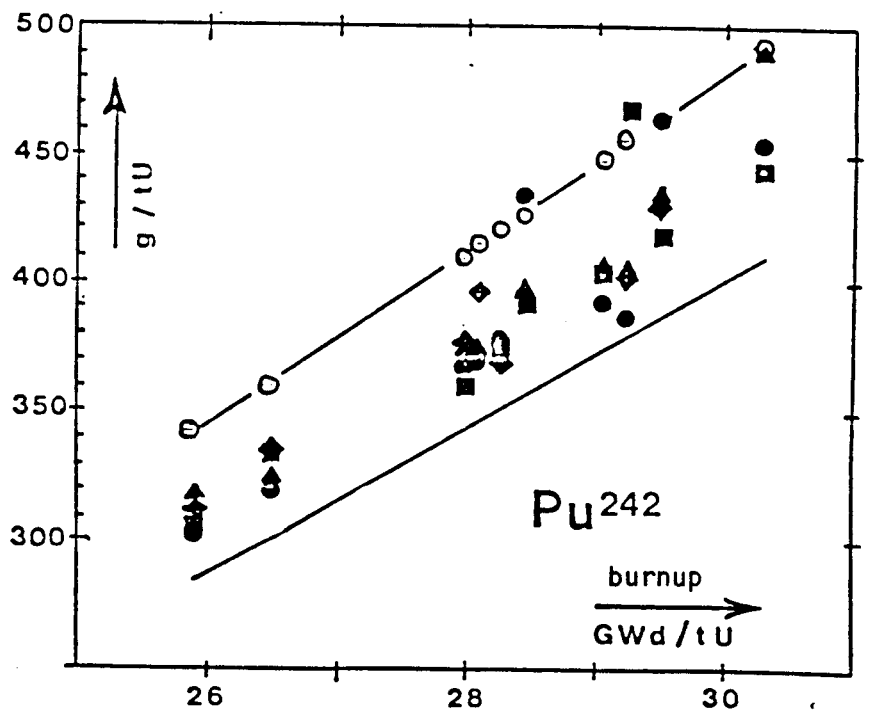

N (kg/t U)
U235
burnup
GWd/t U
12
11
10
9
8
26
28
30
N (kg/t U)
U236
burnup
GWd/t U
3.8
3.6
3.4
26
28
30
APPENDIX 1 ICE MEASUREMENTS
KORIGEN
OREST


N(g/tU)
140
120
100
80
Pu238
burnup
GWd/tU
26
28
30
5.2
5.0
4.8
4.6
4.4
4.2
Pu239
burnup
GWd tU
26
28
30
APPENDIX 2
ICE MEASUREMENTS
KORIGEN
OREST


N (kg/t U)
2.0
1.8
1.6
Pu240
ICE MEASUREMENTS
KORIGEN
OREST
burnup
GWd/tU
26
28
30

APPENDIX 3

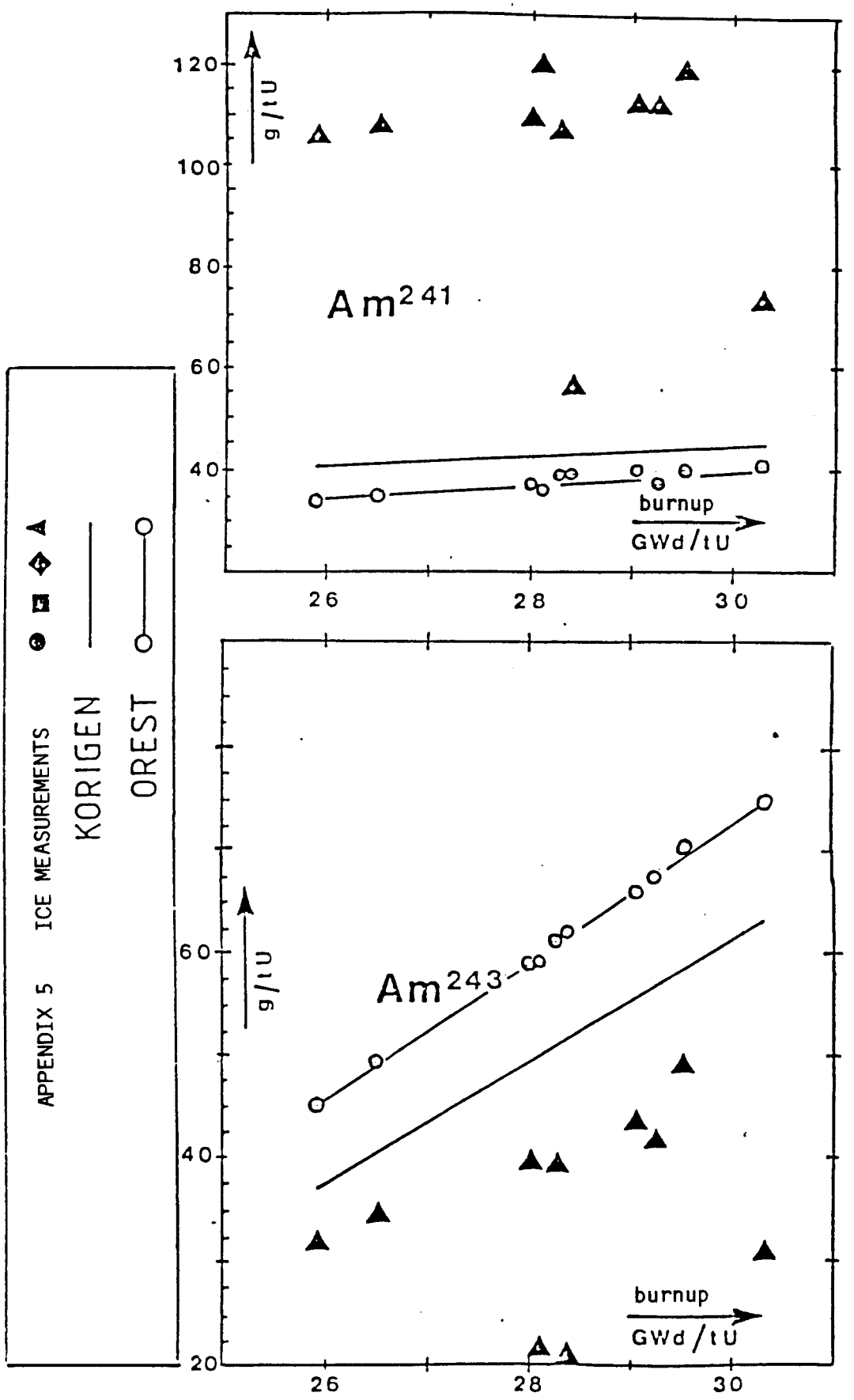


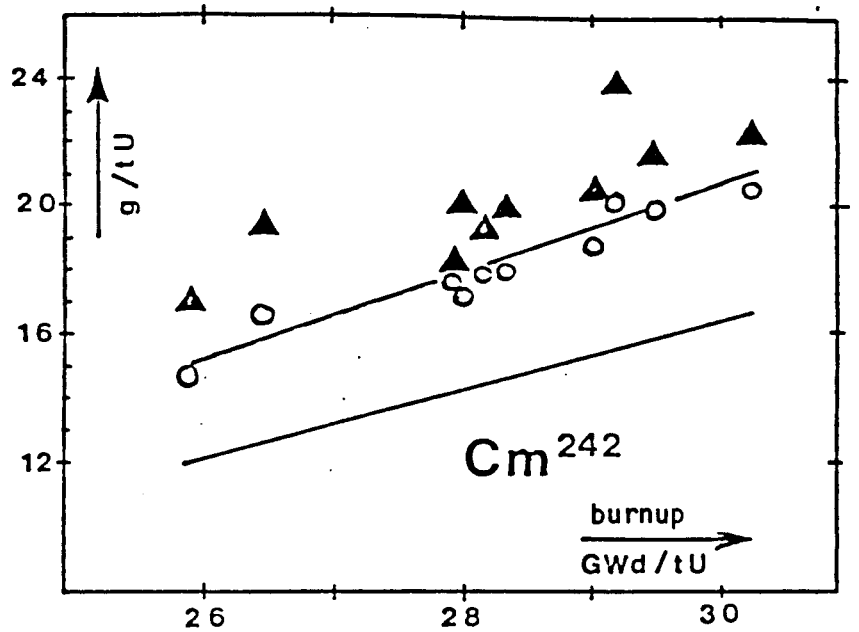


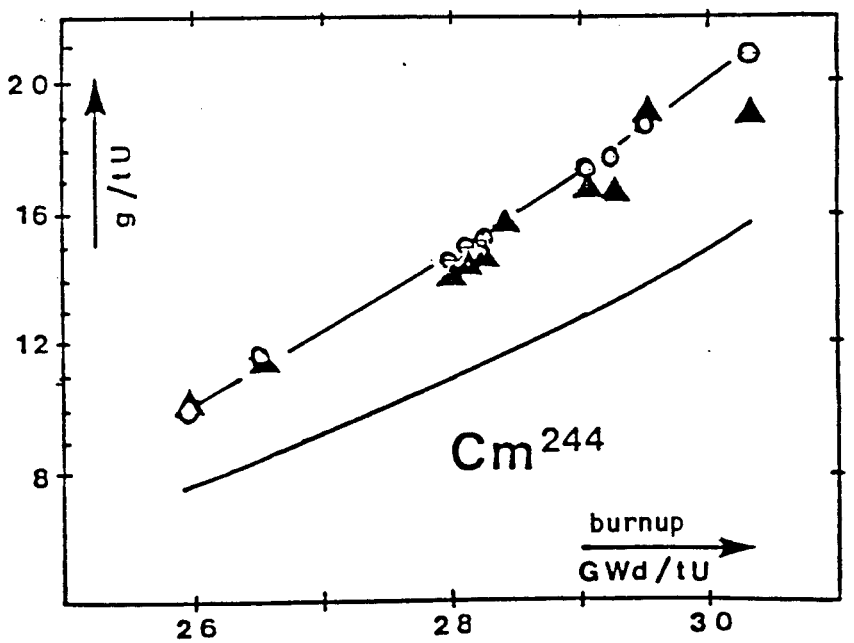
APPENDIX 4 ICE MEASUREMENTS

KORIGEN

OREST


APPENDIX 5
ICE MEASUREMENTS
KORIGEN
OREST
Am241
Am243
g/tU
burnup
GWd/tU
120
100
80
60
40
20
26
28
30





APPENDIX 6 ICE MEASUREMENTS

KORIGEN

OREST