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STANDARD PROBLEM EXERCISE ON CRITICALITY CODES FOR LARGE ARRAYS OF PACKAGES OF FISSILE MATERIALS

by a
CSNI Working Group

FINAL REPORT
August 1984

Edited and Published by
Oak Ridge National Laboratory
Oak Ridge, Tennessee 37830 U.S.A.
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Martin Marietta Energy Systems, Inc.
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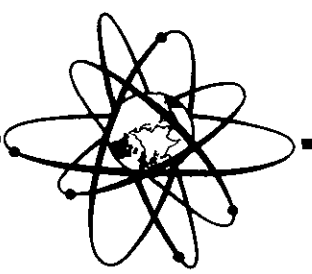

COMMITTEE ON THE SAFETY OF NUCLEAR INSTALLATIONS
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by a CSNI Group of Experts
on Nuclear Criticality Safety Computations

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Committee on the Safety of Nuclear Installations
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I. INTRODUCTION

In May 1982 a report on a "Standard Problem Exercise on Criticality Codes for Spent Fuel Transport Containers" was issued by a CSNI Working Group. This report presented the results of a study which was intended to demonstrate whether a number of computational methods could be validated for use in evaluating the criticality safety of transport containers for spent reactor fuel. The impending completion and expected success of this study was noted by an International Atomic Energy Agency (IAEA) group which was reviewing the criticality safety aspects of the Agency's regulations for the safe transport of radioactive materials.

In a letter from the IAEA to the OECD Nuclear Energy Agency, a request was made for the CSNI Working Group to consider extending their study to cover calculations for arrays of transport packages.

This report presents the results obtained by the CSNI Working Group for large finite and infinite arrays of fissile material units. The current work consists of five different fissile material forms which have several thicknesses of water between the fissile units and a thick water reflector around the outside of the finite arrays. These were intended to be hypothetical transport packages with the importance of interspersed moderation being the parameter studied.

In an effort to study and validate only one parameter at a time, the effects of other packaging and structural materials are not considered in this report.

The initial effort of the current exercise was similar to that of a previous standard problem exercise reported in CSNI Report No. 71; that is, experimentally critical systems were chosen to be used in validating the computational procedures for systems with physical and material properties similar to that which one might find in Fissile Class II packages.

Due to the lack of experiments with large ($\geq 5^3$) arrays of fissile materials (in which each unit is a substantial fraction of a critical mass), it is not possible to validate the calculations as a function of array size. This leaves, without an easy solution, the problem of needing to extrapolate from the data which are available. There are two difficulties which cause concern about such extrapolations. First, for several experiments which have been performed in which the array size was varied, the bias in the calculations has been a reasonably strong function of the array size. Secondly, with a finite number of neutrons per generation for Monte Carlo calculations of very large arrays, a question must be raised regarding the adequacy of sampling. These two concerns were addressed by this study.

Another problem addressed by this study involved the effect of commingling Fissile Class II packages with different fissile material contents. The current IAEA regulations have as one of their criteria the condition that five times the number of packages allowed in a shipment would be subcritical. This provides a margin of safety in case two more shipments are placed together by a transportation organization. The question which needed study was whether the safety margins thought to be in place would be reduced if the multiple shipments which were placed together consisted of different fissile materials. Hypothetical arrangements for commingling of different types of fissile materials were studied in an effort to understand whether commingling would reduce the safety margins.

While our study allowed us to make a recommendation regarding some of the above-mentioned problems, it became clear that additional critical experiments are needed to satisfactorily provide the complete guidance needed to assure safe and economical transport of Fissile Class II packages.

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II. OBJECTIVE OF THE EXERCISE

The objective of this exercise was to establish the validity of criticality safety computational methods for computing arrays of fissile transport packages which are defined as Fissile Class II under the IAEA Regulations for the Safe Transport of Radioactive Materials. This initial work has been directed toward examining the validity of criticality computer programs under conditions of varying amounts of hydrogenous moderators between the fissile units. Future work will need to address other issues such as the effects of structural material neutron poisons, other package components, and additional work on the intermingling of unlike packages.

The problems chosen consist of experiments with materials which might typically be placed in a Fissile Class II package. Specifically, the problems involve:

1. highly enriched uranium metal
2. highly enriched uranyl nitrate
3. 5% enriched uranium oxide
4. 5% enriched uranium oxide with H/U = 20
5. plutonium oxide.

Calculations were also made on experimentally critical arrays of units resembling the hypothetical packages of Problems 1-4 with the objective of providing a basis for judging the validity of the computer program and the associated cross-section libraries. No appropriate experimental data were found to compare with Problem 5.

The problems chosen for study consisted of 8x8x8 (8^3) arrays of fissile material units with varying amounts of water as interspersed moderation and a thick water reflector. The 8^3 array size was chosen because this number of units (512) was thought to be a reasonable "large" number of packages which would be used in Fissile Class II shipments.

In addition to the study of 8^3 arrays for various fissile materials, another study was made to determine whether adequate neutron sampling was occurring for large arrays. Calculations were made by the participants for arrays of size 4^3, 12^3, and 16^3. The results of these calculations were compared for consistency, and for evidence of inadequate sampling, with the results obtained for the 8^3 calculations.

A study was also made to evaluate the effect on k_{eff} which would result from commingling two different fissile material packages. The object was to determine if the k_{eff} resulting from commingling two different fissile material packages would be greater than the k_{eff} for an array of the same size which consisted of either of the two packages alone.

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III. PROBLEM DESCRIPTIONS

Simulated Transport Package Problems 1-5 consisted of 8x8x8 (8^3) arrays of fissile material surrounded by a thick water reflector at the array boundary. The fissile material was spherical in shape and was surrounded by a spherical shell of water with varying thicknesses of water. A complete description of all problems is given in Appendix A. The following is a short summary of the problem specifications.

1.	Material	^{235}U Metal
	Density	18.76 g/cm^3
	Radius of Fissile Material	6.242 cm
	Thicknesses of Water Shell	a) 0.0 cm
		b) 2.54 cm
		c) 10.16 cm
2.	Material	^{235}U Nitrate Solution
	Density	1.07892 g/cm^3
	H/U	426
	Radius of Fissile Material	12.0 cm
	Thicknesses of Water Shell	a) 0.0 cm
		b) 2.0 cm
		c) 2.8 cm
3.	Material	U(5% ^{235}U)O_2 Powder
	Density	5.0 g/cm^3
	H/U	0
	Radius of Fissile Material	20.5 cm
	Thicknesses of Water Shell	a) 0.0 cm
		b) 2.2 cm
		c) 4.0 cm
4.	Material	U(5% ^{235}U)O_2 Powder
	Density	2.2 g/cm^3
	H/U	20
	Radius of Fissile Material	14 cm
	Thicknesses of Water Shell	a) 0.0 cm
		b) 4.0 cm
5.	Material	PuO_2 (25% ^{240}Pu, 12% ^{241}Pu, 3% ^{242}Pu)
	Density	5.0 g/cm^3
	H/U	0
	Radius of Fissile Material	10.7 cm
	Thicknesses of Water Shell	a) 0.0 cm
		b) 2.2 cm
		c) 6.0 cm

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Experimentally critical* systems which were similar to the arrays of simulated transport packages are given below. Problems I-IV follow.

I.	Material	U(93.2% ^{235}U) Metal Cylinders
	Density	18.73 g/cm^3
	Radius of Fissile Material	5.753 cm
	Height	10.765 cm
	Moderator	Plexiglas box surrounding each cylinder outside dimensions 21.4x21.4x20.7 cm high wall thickness 2.38 cm
	a) Array Size	2x2x2
	Paraffin Refl Thickness	0.0 cm
	b) Array Size	2x2x2
	Paraffin Refl Thickness	15.2 cm
	c) Array Size	3x3x3
	Paraffin Refl Thickness	0.0 cm
II.	Material	Uranyl (92.6% ^{235}U) Nitrate Solution Cylinders
	Density	1.083 g/cm^3
	$H/^{235}U$	441
	Radius of Fissile Material	19.04 cm
	Height	17.77 cm
	Moderator	Plexiglas cylinder with wall and end thicknesses of 0.64 cm
	Array Size	3x3x3
	No Reflector	
III.	Material	U(4.46% $^{235}U)_3O_8$-H_2O
	Density	4.47 g/cm^3
	H/U	0.77
	Cube Fissile Material	15.3 cm on each side
	Moderator Cube	0.923-cm-thick plexiglas surrounding fissile material
	Array Size	5x4x5
	Reflector	Thick plexiglas reflector on all sides
	Experimental k_{eff}	1.014
IV.	Material	U(4.89% $^{235}U)_3O_8$-$(C_{17}H_{35}CO_2)_3C_3H_5$
	H/U	9.86
	Cube Fissile Material	40.64x40.64x58.17
	Array Size	1x1x1
	Reflector Cube	Thick H_2O or paraffin reflector on all sides

*Note that Problem 3 had an experimental k_{eff} = 1.014 with no gap (i.e., criticality occurred with a small gap remaining between the two halves of the system).

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IV. RESULTS

EXPERIMENTALLY CRITICAL ARRAY CALCULATIONS

The experiments (Problems I-IV) chosen for this exercise were selected to be as similar as possible to the hypothetical class II fissile packages which were to be studied. Data for arrays as large as 8x8x8 (8^3) were not available; hence, the data chosen were not as complete as we would have desired. We were unable to obtain experimental data for arrays of plutonium oxide.

Table 1 presents the results obtained for the experimentally critical systems studied. The results can be observed to be in reasonably good agreement with the expected values with the exception of the EIR results. The EIR calculations were performed with a new computational procedure which cannot yet completely treat the geometries present in these experimental systems. The rather good agreement between experimental and calculated results obtained with the majority of the methods provides considerable confidence that the methods will perform satisfactorily for calculations on the "hypothetical problems" chosen.

8^3 ARRAYS OF SIMULATED FISSILE CLASS II PACKAGES

The multiplication factor for 8x8x8 arrays of simulated Fissile Class II packages were each computed with three values of water moderation between the units (except for Problem 4 where only two values were computed). The water moderation was chosen to represent no moderation, optimal moderation, and overmoderation. The simple moderation model chosen was recognized not to be a complete study of moderation effects. In actual analysis of such systems it is necessary to establish optimum moderation; therefore, these studies cover only a portion of the range of application which would be needed to establish the safety of a package.

The results in almost every case are in agreement with the results obtained for the calculations of the experimentally critical systems. That is, if a negative/positive bias (i.e., the computed k_{eff} of the experimentally measured system is lower/higher than the experimental k_{eff}) is observed, then the results for the simulated package are less/greater than the expected values based on a concensus of the results from all participants for the simulated package calculations. A summary of the results is given in Table 2. The data are shown graphically in Appendix B.

The EIR results, which were not in good agreement with the experimentally measured systems, produced excellent results for the 8^3 arrays. The method employed by EIR was demonstrated to adequately compute k_{eff} for systems which meet the geometrical qualifications of the method.

LARGER ARRAY CALCULATIONS

In studying the results received for the 8^3 array problems, the problem coordinators became concerned about the convergence pattern of several of the Monte Carlo calculations. The question was raised as to whether there was a risk associated with computing an array with 500 to 5000 units while using only 100 to 500 neutrons per batch (or stage). Since in this situation it is quite possible that not every unit would be sampled during the processing of one batch, the concern was that an incorrect value of k_{eff} might be calculated.

To study this effect it was agreed that we would compute arrays of 12^3 and 16^3 to observe if this produced evidence of inadequate sampling. Based on the results obtained, several participants decided to go the other direction in array size and computed a 4^3 array.

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Table 1. Results of experimentally critical systems ($k_{eff} \pm 1\sigma$)

Computer Program	SPHERE	MORET	MORET	KENO	KENO-IV	MORSE-K	KENO	KENO
Cross Sections		Hansen-Roach	APOLLO	SCALE 27 GP	Hansen-Roach	RSYST	SCALE 27 GP	GAM-THERMOS 123 GP
Case No.	EIR Switzerland	CEA France	CEA France	ORNL U.S.	GRS Germany	PTB Germany	ENEA Italy	ENEA Italy
1.1	0.9269	0.984 ± 0.005	0.985 ± 0.005	1.001 ± 0.004	0.995 ± 0.005	1.001 ± 0.007	0.997 ± 0.004	0.995 ± 0.004
1.2	1.0350	0.990 ± 0.009	0.998 ± 0.009	1.005 ± 0.005	1.000 ± 0.005	1.005 ± 0.013	0.999 ± 0.005	1.004 ± 0.004
1.3	0.9636	0.983 ± 0.005	0.975 ± 0.005	0.993 ± 0.005	0.993 ± 0.005		0.989 ± 0.004	1.007 ± 0.005
2	0.8799	1.005 ± 0.005	1.004 ± 0.005	1.002 ± 0.005	1.005 ± 0.006		1.019 ± 0.005	
3(b [∞)	1.3026		1.335 ± 0.005	1.315 ± 0.004	1.302 ± 0.005*		1.296 ± 0.004	1.318 ± 0.003
3	1.053		1.029 ± 0.006	1.002 ± 0.005	1.024 ± 0.005*		1.012 ± 0.005	1.009 ± 0.004
4(b [∞)	1.4702		1.466	1.450	1.443 ± 0.003*	1.499 ± 0.010	1.455 ± 0.008	1.456 ± 0.003
4	1.0056		0.996 ± 0.005	0.986 ± 0.003	0.991 ± 0.005*	0.999 ± 0.009	0.987 ± 0.004	0.990 ± 0.006

*GAMTEC-II cross sections

**Multi-KENO computer program

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Table 1 (continued)

Case No.	Computer Program / Cross Sections	KENO 27-Group GAM GATHER ENEA Italy	MONK UKNDL SRD Italy	KENO SCALE 27 GP Sweden	KENO Hansen-Roach BN Belgium	MONK UKNDL SRD U.K.	KENO-IV MGCL JAERI Japan	KENO Jr MGCL JAERI Japan
1.1		0.983 ± 0.005	1.016 ± 0.007	1.000 ± 0.005	1.000 ± 0.005	0.992 ± 0.007	1.004 ± 0.003	0.979 ± 0.002
1.2		0.985 ± 0.005	1.019 ± 0.007	0.995 ± 0.006	1.006 ± 0.008	0.995 ± 0.008	1.011 ± 0.002	1.002 ± 0.002
1.3		0.987 ± 0.005	1.010 ± 0.007	0.995 ± 0.004	1.009 ± 0.004	0.989 ± 0.007	1.002 ± 0.003	0.990 ± 0.002
2		0.984 ± 0.0006	1.022 ± 0.008	0.997 ± 0.005	1.010 ± 0.005	1.002 ± 0.007	0.974 ± 0.003	1.010 ± 0.003
3(b ∞)		1.297 ± 0.005	1.311 ± 0.009	1.288 ± 0.004	1.355 ± 0.004	1.290 ± 0.005	1.338 ± 0.002	1.325 ± 0.002
3		1.021 ± 0.005	0.999 ± 0.008	0.999 ± 0.004	1.033 ± 0.006	1.010 ± 0.006	1.033 ± 0.002*	
4(b ,∞)		1.419 ± 0.003	1.485 ± 0.008	1.450	1.457 ± 0.004	1.464 ± 0.004	1.464 ± 0.002	1.456 ± 0.003
4		0.970 ± 0.005	1.015 ± 0.008	0.991 ± 0.006	0.972 ± 0.008	1.016 ± 0.007	0.977 ± 0.002	0.984 ± 0.003

*Multi-KENO computer program

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Table 2. Summary of results for simulated transport packages

	Computer Program	SPHERE	MORET	MORET	TRIMARAN	KENO	KENO	KENO	MORSE-K	KENO
	Cross Sections		Hansen-Roach	APOLLO Hansen-Roach	APOLLO	SCALE 27 GP	Hansen-Roach	GAMTEC-II	RSYST	SCALE 27 GP
Case No.	H_2O Thickness	EIR Switzerland	CEA France	CEA France	CEA France	ORNL U.S.	GRS Germany	GRS Germany	PTB Germany	ENEA Italy
1.a	0	0.9975	0.968 ± 0.005	0.977 ± 0.007		0.997 ± 0.004	0.970 ± 0.005	1.016 ± 0.005	0.957 ± 0.009	0.980 ± 0.004
1.b	2.54	1.0238	0.991 ± 0.007	0.989 ± 0.007	0.991 ± 0.003	0.993 ± 0.004	0.989 ± 0.005	1.024 ± 0.005	0.971 ± 0.009	0.989 ± 0.005
1.c	10.16	0.9988	0.965 ± 0.007	0.978 ± 0.007		0.975 ± 0.007	0.968 ± 0.004	0.997 ± 0.005	0.988 ± 0.006	0.978 ± 0.005
1.a (b ∞)	0	2.3205								
1.b (b ∞)	2.54	1.8298	1.807 ± 0.005	1.820 ± 0.005	1.729 ± 0.002	1.819 ± 0.003	1.818 ± 0.003	1.793 ± 0.003	1.799 ± 0.009	1.825 ± 0.004
1.c (b ∞)	10.16	1.0239								
2.a	0	0.9992	0.973 ± 0.005	0.976 ± 0.007		0.973 ± 0.004	0.973 ± 0.005	0.982 ± 0.005	0.929 ± 0.007	0.997 ± 0.006
2.b	2.0	0.9893	0.974 ± 0.007	0.968 ± 0.007	0.983 ± 0.003	0.974 ± 0.004	0.969 ± 0.005	0.973 ± 0.005	0.947 ± 0.007	0.971 ± 0.004
2.c	2.8	0.9734	0.964 ± 0.005	0.966 ± 0.007		0.958 ± 0.005	0.964 ± 0.005	0.967 ± 0.004	0.930 ± 0.006	0.955 ± 0.006
2.a (b ∞)	0	1.6737								
2.b (b ∞)	2.0	1.3964	1.414 ± 0.006	1.412 ± 0.007	1.369 ± 0.001	1.386 ± 0.003	1.409 ± 0.003	1.382 ± 0.003	1.359 ± 0.013	1.391 ± 0.004
2.c (b ∞)	2.8	1.2700								
3.a	0	0.7676	0.733 ± 0.005	0.740 ± 0.007		0.720 ± 0.003	0.726 ± 0.005	0.754 ± 0.005	0.655 ± 0.007	0.727 ± 0.004
3.b	2.2	0.9941	0.999 ± 0.005	0.991 ± 0.007		0.975 ± 0.003	0.990 ± 0.004	0.954 ± 0.004	0.924 ± 0.011	0.983 ± 0.004
3.c	4.0	0.8715	0.861 ± 0.005	0.864 ± 0.007		0.845 ± 0.004	0.856 ± 0.004	0.822 ± 0.004	0.817 ± 0.008	0.845 ± 0.004
3.a (b ∞)	0	0.8399								
3.b (b ∞)	2.2	1.3228	1.318 ± 0.005	1.323 ± 0.005		1.293 ± 0.002	1.321 ± 0.004	1.240 ± 0.004	1.300 ± 0.011	1.306 ± 0.004
3.c (b ∞)	4.0	1.0470								
4.a	0	0.9962	0.971 ± 0.005	0.952 ± 0.007	0.969 ± 0.002	0.967 ± 0.003	0.947 ± 0.005	0.962 ± 0.004	0.968 ± 0.007	0.971 ± 0.004
4.b	4.0	0.9151	0.897 ± 0.007	0.890 ± 0.007		0.898 ± 0.004	0.875 ± 0.004	0.896 ± 0.004	0.915 ± 0.008	0.899 ± 0.005
4.a (b ∞)	0	1.4401	1.412 ± 0.002	1.426 ± 0.007	1.41 ± 0.001	1.398 ± 0.002	1.407 ± 0.002	1.399 ± 0.003	1.458 ± 0.012	1.426 ± 0.003
4.b (b ∞)	4.0	1.0415								
5.a	0	1.0337		0.976 ± 0.007		0.996 ± 0.004	0.998 ± 0.005	0.960 ± 0.005	0.953 ± 0.012	1.001 ± 0.005
5.b	2.2	1.0318		0.977 ± 0.007	0.980 ± 0.003	1.003 ± 0.003	0.993 ± 0.005	0.962 ± 0.005	0.960 ± 0.010	1.008 ± 0.005
5.c	6.0	0.9371		0.893 ± 0.007		0.910 ± 0.004	0.917 ± 0.005	0.878 ± 0.005	0.897 ± 0.006	0.909 ± 0.005
5.a (b ∞)	0	2.7678			1.769 ± 0.002					1.861 ± 0.003
5.b (b ∞)	2.2	1.8580		1.797 ± 0.005		1.838 ± 0.003	1.841 ± 0.003	1.760 ± 0.003	1.827 ± 0.010	
5.c (b ∞)	6.0	1.1105								

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Table 2 (continued)

	Computer Program	KENO	MONK	KENO	KENO	KENO	MONK	KENO-IV	KENO Jr
	Cross Sections	GAM-THERMOS 123 GP	UKNDL	GAM GATHER 200 Groups	SCALE 27 GP	Hansen-Roach	UKNDL	MGCL	MGCL
Case *No.*	H_2O *Thickness*	ENEA Italy	ENEA Italy	ENEA Italy	Sweden	BN Belgium	SRD U.K.	JAERI Japan	JAERI Japan
1.a	0	0.978 ± 0.005	0.994 ± 0.007		0.968 ± 0.005	0.970 ± 0.005	0.975 ± 0.008	0.988 ± 0.002	0.975 ± 0.002
1.b	2.54	0.988 ± 0.005	1.012 ± 0.007	0.997 ± 0.005	0.998 ± 0.006	0.984 ± 0.005	0.996 ± 0.008	1.002 ± 0.002	1.001 ± 0.003
1.c	10.16	0.966 ± 0.004	1.012 ± 0.007		0.973 ± 0.007	0.986 ± 0.007	1.007 ± 0.008	0.979 ± 0.003	0.994 ± 0.003
1.a (b ∞)	0								
1.b (b ∞)	2.54	1.8032	1.807 ± 0.010	1.81251	1.808 ± 0.004		1.797 ± 0.005		
1.c (b ∞)	10.16								
2.a	0		0.979 ± 0.008		0.986 ± 0.005	0.964 ± 0.009	0.990 ± 0.008	0.959 ± 0.002	0.989 ± 0.003
2.b	2.0		0.979 ± 0.008	0.978 ± 0.005	0.978 ± 0.006	0.953 ± 0.007	0.984 ± 0.007	0.955 ± 0.003	0.985 ± 0.003
2.c	2.8		0.969 ± 0.008		0.964 ± 0.006	0.962 ± 0.013	0.962 ± 0.007	0.940 ± 0.003	0.968 ± 0.003
2.a (b ∞)	0								
2.b (b ∞)	2.0	1.3849	1.396 ± 0.008	1.41139	1.388 ± 0.005		1.396 ± 0.004		
2.c (b ∞)	2.8								
3.a	0		0.720 ± 0.006		0.711 ± 0.005	0.724 ± 0.008	0.720 ± 0.007	0.756 ± 0.002	0.737 ± 0.002
3.b	2.2		0.983 ± 0.007		0.978 ± 0.004	0.993 ± 0.005	0.964 ± 0.007	1.011 ± 0.003	1.002 ± 0.002
3.c	4.0		0.854 ± 0.007		0.844 ± 0.005	0.857 ± 0.007	0.837 ± 0.007	0.877 ± 0.002	0.867 ± 0.002
3.a (b ∞)	0								
3.b (b ∞)	2.2	1.3174	1.282 ± 0.007		1.291 ± 0.005	1.318 ± 0.007	1.279 ± 0.005	1.344 ± 0.002	1.323 ± 0.002
3.c (b ∞)	4.0								
4.a	0		0.999 ± 0.008		0.964 ± 0.005	0.941 ± 0.006	0.995 ± 0.007	0.959 ± 0.002	0.978 ± 0.003
4.b	4.0		0.913 ± 0.008		0.899 ± 0.006	0.873 ± 0.008	0.905 ± 0.007	0.886 ± 0.002	0.910 ± 0.003
4.a (b ∞)	0	1.4230	1.454 ± 0.009		1.409 ± 0.003	1.405 ± 0.004	1.415 ± 0.003	1.432 ± 0.001	1.429 ± 0.003
4.b (b ∞)	4.0								
5.a	0		0.979 ± 0.007		0.989 ± 0.005	0.993 ± 0.008	1.000 ± 0.007	1.006 ± 0.003	1.006 ± 0.003
5.b	2.2		0.996 ± 0.007	1.004 ± 0.005	0.999 ± 0.006	0.987 ± 0.006	0.998 ± 0.008	1.009 ± 0.003	1.015 ± 0.002
5.c	6.0		0.913 ± 0.009		0.913 ± 0.009	0.891 ± 0.009	0.927 ± 0.008	0.911 ± 0.003	0.921 ± 0.003
5.a (b ∞)	0	1.8709	1.873 ± 0.006	1.83949	1.825 ± 0.004	1.798 ± 0.005	1.817 ± 0.005	1.874 ± 0.002	1.868 ± 0.002
5.b (b ∞)	2.2								
5.c (b ∞)	6.0								

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Based on the collective results observed from these calculations, it was concluded that no evidence was observed which indicated that one would have problems with inadequate sampling in arrays of identical units. Based on some limited research by one of the participants, it appears that, if there is a problem, it would more likely occur in arrays of size 4^3 to 8^3. Since the use of 300-500 neutrons per batch would assure reasonable sampling in these systems, it is suggested that there is no cause for concern regarding sampling in uniform arrays.

The results obtained by the participants for these problems are given in Table 3. The reader will observe that excellent agreement was obtained even though the number of neutrons per batch was varied over a wide range.

MIXED ARRAYS

One of the questions raised by the IAEA's request for study was whether the commingling of two unlike arrays would produce a k_{eff} which would be higher than two like arrays.

This concern is raised because of the interpretation and use of current IAEA regulations concerning Fissile Class II units. The regulations were designed to provide for safety in case more than one shipment arrived at the same location during transportation. An interpretation also allows a single shipment to be made up of unlike units as long as a summation of their individual TI (transport index) does not exceed the allowable number.

The members of the Working Group considered which units of the five used in the 8^3 array study, when commingled, would be most likely to produce a higher k_{eff} than an array of like units. The units chosen were the bare sphere of 235-uranyl nitrate solution (Problem 2.a) and the sphere of dry uranium (5% ^{235}U) oxide powder (Problem 3.b). The commingling was to be in the following fashion.

i. an alternating ("chequerboard") arrangement of 2.a- and 3.b-type packages;

ii. a 6x6x6 core of 3.b-type packages surrounded by a one-package-deep shell of 2.a-type packages;

iii. a 6x6x6 core of 2.a-type packages surrounded by a one-package-deep shell of 3.b-type packages;

iv. any other arrangement and type of packages to be chosen by each participant.

The results of our commingling study are given in Table 4. Not every participant solved every problem; however, a sufficient number of results are available to reach a conclusion about this study. The exercises, i, ii, and iii, demonstrate that commingling of these particular units in the fashions described does not increase k_{eff} beyond that of an 8^3 array of either unit type alone. However, results presented by the UK representative which used package type 5.a and package type 2.b in an arrangement similar to that described in i and ii demonstrate that for this type of simulated package a significantly higher k_{eff} can be obtained through commingling. Independent calculations by other countries have subsequently confirmed the UK results.

The full consequences of our findings can only be determined by additional research on this topic. Many factors could affect the validity of our findings. We probably have not uncovered the commingling situation which produces the greatest increase in k_{eff}. On the other hand, we have not considered the effect of the steel which is present in most typical packages and which could negate the increase in k_{eff} due to commingling.

The Working Group considered whether the summation rule was valid when making up a single shipment. While there was at least one strongly dissenting opinion, the Working Group felt that in most practical situations the use of the summation rule in making up a single shipment should be allowed. The Working Group, however, recommended that the effect of commingling of multiple shipments should be studied further.

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Table 3. Results for simulated transport packages with varying array sizes

	MONK 100 neutrons/batch	MONK 200 neutrons/batch	KENO-IV 100 neutrons/batch	KENO Jr. 100 neutrons/batch	MORET	KENO-IV 250 neutrons/batch	KENO-IV 1000 neutrons/batch	KENO-IV	KENO-IV Hansen-Roach	KENO Hansen-Roach	KENO SCALE 27 GP
Array Size	UK	UK	Japan	Japan	France	U.S.	U.S.	Italy	GRS Germany	Belgium	Sweden
1-b Arrays											
4^3	0.933 ± 0.007					0.932 ± 0.005	0.931 ± 0.003	0.922 ± 0.005		0.926 ± 0.005	
6^3								0.962 ± 0.004			
8^3	0.996 ± 0.008					1.000 ± 0.003	1.000 ± 0.003	0.989 ± 0.004	0.999 ± 0.005	0.984 ± 0.005	
12^3	1.083 ± 0.008	1.071 ± 0.007				1.058 ± 0.002	1.061 ± 0.003	1.050 ± 0.005	1.061 ± 0.005		1.064 ± 0.0047
16^3	1.118 ± 0.008	1.127 ± 0.007				1.122 ± 0.003	1.124 ± 0.003	1.112 ± 0.005	1.116 ± 0.004	1.119 ± 0.005	1.115 ± 0.0046
∞^3	1.797 ± 0.005							1.825 ± 0.004		1.818 ± 0.003	
2-b Arrays											
4^3	0.876 ± 0.008		0.845 ± 0.003	0.876 ± 0.003	0.858 ± 0.005	0.870 ± 0.006		0.877 ± 0.005			
6^3			0.908 ± 0.003	0.933 ± 0.003	0.916 ± 0.005			0.914 ± 0.005			
8^3	0.981 ± 0.006		0.955 ± 0.003	0.987 ± 0.003	0.968 ± 0.007	0.974 ± 0.004		0.977 ± 0.006	0.969 ± 0.005		
12^3	1.067 ± 0.007	1.068 ± 0.007	1.042 ± 0.003	1.069 ± 0.003	1.066 ± 0.005	1.059 ± 0.005		1.066 ± 0.006	1.058 ± 0.005		1.056 ± 0.0041
16^3	1.115 ± 0.007	1.129 ± 0.006	1.103 ± 0.002	1.119 ± 0.003	1.119 ± 0.005	1.123 ± 0.004		1.116 ± 0.005	1.117 ± 0.004		1.109 ± 0.0047
∞^3	1.396 ± 0.004	1.396 ± 0.004	1.385 ± 0.002	1.389 ± 0.002	1.412 ± 0.005	1.386 ± 0.003		1.391 ± 0.004			

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Table 4. Commingled arrays

Array Configuration	KENO Hansen-Roach Belgium	MORET France	KENO 27-Group U.S.	MONK U.K.	KENO 27-Group Italy	KENO-IV 137-Group MGCL Japan	KENO-Jr. 137-Group MGCL Japan	KENO-IV Hansen-Roach GRS Germany	KENO SCALE 27-Group Sweden
8^3 2a	0.964 ± 0.009	0.973 ± 0.005	0.973 ± 0.004	0.990 ± 0.008	0.980 ± 0.005	0.959 ± 0.002	0.989 ± 0.003	0.973 ± 0.005	0.986 ± 0.005
8^3 3b	0.993 ± 0.005	0.999 ± 0.005	0.975 ± 0.003	0.964 ± 0.007	0.977 ± 0.004	1.011 ± 0.003	1.002 ± 0.002	0.990 ± 0.004	0.978 ± 0.004
8^3 2b			0.974 ± 0.004	0.981 ± 0.006	0.978 ± 0.005				0.978 ± 0.006
8^3 5a			0.996 ± 0.004	1.000 ± 0.007	1.005 ± 0.005				0.989 ± 0.005
8^3 alternating 2a + 3b	0.979 ± 0.005	0.986 ± 0.005	0.979 ± 0.003	0.984 ± 0.007	0.989 ± 0.005	0.986 ± 0.002	0.992 ± 0.003	0.985 ± 0.005	0.988 ± 0.007
8^3 3b around 2a	0.998 ± 0.005	0.997 ± 0.005		0.993 ± 0.007	1.007 ± 0.005	0.999 ± 0.002	1.014 ± 0.003		1.012 ± 0.005
8^3 2a around 3b	0.974 ± 0.005	0.970 ± 0.007	0.963 ± 0.003	0.972 ± 0.007	0.981 ± 0.005	0.977 ± 0.003	0.982 ± 0.003	0.969 ± 0.005	0.980 ± 0.008
8^3 2b around 5a			1.021 ± 0.003	1.023 ± 0.008	1.034 ± 0.005				
8^3 5a around 2b				1.027 ± 0.007	1.042 ± 0.004				
8^3 alternating 2b + 5a			1.028 ± 0.003	1.034 ± 0.0081	1.035 ± 0.005				
8^3 3a								0.726 ± 0.005	
8^3 3c								0.856 ± 0.004	
8^3 alternating 3a + 3c								1.021 ± 0.003	
8^3 2 x 3a around 3c								0.886 ± 0.005	

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V. PRINCIPAL CONCLUSIONS AND RECOMMENDATIONS FROM THE STUDY

The main conclusion reached was that the criticality codes studied can satisfactorily handle the largest arrays likely to be of practical interest, providing precautions are taken to ensure that:

— the computed value of k_{eff} has converged sufficiently, and

— there has been sufficient neutron sampling of all parts of the array.*

It was also found that k_{eff} for such arrays may be significantly lower than k_{∞} for the corresponding (hypothetical) infinite array. Thus, use of k_{∞} values in regulating actual loadings may impose a severe penalty on the size of arrays accepted for transport.

It was recommended that besides the margins usually allowed on such calculated k_{eff} values (e.g., 3-sigma in statistical approaches, plus an allowance for the difference between theoretical and experimental neutronic cross-section data), a specific, supplementary allowance should be made when treating larger arrays to ensure subcriticality and to allow for the uncertainties resulting from the lack of experimental data for large arrays.

MIXED ARRAYS OF FISSILE MATERIAL PACKAGES

It appears impossible to give a general demonstration of the validity of the summation rule for criticality transport indices ($T.I._c$) when several types of packages are stacked together.** Given the practical importance of the question, members of the Working Group made calculations for a few mixed arrays composed of two package types. The package types used were selected from those defined for the large array study as being most likely to show a higher k_{eff} for the mixed array than for an array of the same size of either type alone. (This was intended to examine whether in some cases the summation rule was non-conservative.)

It was concluded that the summation rule is an acceptable tool for determining the allowable composition of a shipment of unlike packages (with $T.I._c > 0$) if the following two conditions are met:

i. the calculations for each array are performed with optimal moderation between the fissile units, and

ii. the conclusions and recommendations in the first two paragraphs of this section are adhered to.

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*This precaution may be particularly important for arrays of "loosely-coupled" packages (i.e., between which neutron interaction is small) and for arrays of different package types (see next section). Undersampling may lead to an (non-conservative) underestimate of k_{eff}. Specifically, care must be taken to ensure that a sufficient number of neutron histories per code iteration are followed, particularly for large non-uniform arrays. The final spatial distributions of the fission reaction rates should also be checked to confirm that adequate sampling has occurred.

**To ensure the criticality safety of a mixed array, the rule is that the number of packages stacked together must be limited so that $\Sigma(T.I._c)_i < 50$, where $(T.I._c)_i$ is the criticality transport index for package type i.

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The Group did observe one case involving two simulated packages with $T.I._c > 0$, which when mixed produced a higher k_{eff} than was observed for an array of the same size of either type alone. While this does not necessarily indicate that the summation rule can be violated for practical packages with $T.I._c > 0$, it does provide an incentive for greater vigilance and perhaps additional research regarding the possibility that the safety margin might be less than expected in the actual transport environment when several shipments are combined or come together.

Further, several theoretical calculations were made which appeared to demonstrate that an unsafe condition could arise if a shipment of packages with $T.I._c = 0$ was mingled with a shipment of packages with $T.I._c > 0$.

UNREFLECTED ARRAYS OF PACKAGES CONTAINING FISSILE MATERIALS IN SOLUTION

The Working Group reviewed the data suggesting that criticality calculations for unreflected arrays of packages containing fissile materials in solution may not be conservative. Additional information was obtained, and the original calculations were repeated by several members of the Group.

No intrinsic code deficiencies were identified that would preclude making accurate calculations for such situations. It was found important to ensure that neutron reflection from concrete and steel structures (even those a few meters away from the array) is modeled adequately. There remain several experimental results which cannot be satisfactorily calculated, and additional study will be needed to resolve this deficiency.

GENERAL CONCLUSIONS FROM THE WORKING GROUP'S STUDIES; FUTURE WORK NEEDED

The exercises performed by the Working Group were based on theoretical package designs. In consequence, and because in general it is difficult to extrapolate experimental data to permit comparison to model cases, the Group is reticent about suggesting explicit general recommendations on technical grounds about "acceptable" computed criticality margins for transport loadings on the basis of the results of their joint study.

This would require evaluating how accurately the codes allow for the effects of all package structural materials, neutron poisons, shielding, etc., in the calculation. However, a wide variety of materials is used in real package designs, including cadmium, wood, cellotex, borated styrofoam, ice, corrugated paper, borated plaster, steel, lead, phenol, and copper. (In their computational exercise, the Working Group deliberately avoided attempting to address in detail the effects of actual packaging materials. It was considered that the numerous additional calculations needed to study the effects of such materials would not add a great deal to the value of the exercise, which had been performed to demonstrate that the codes can make accurate calculations for large arrays. In any case, one of the main packaging parameters affecting array criticality is interspersed neutron moderation, and the Group had already studied this particular parameter by introducing water shells in the model packages used.)

It was generally agreed that more work in individual countries is necessary before it will be useful to perform any further joint comparative study on materials effects.

ADDITIONAL CONSIDERATIONS REGARDING SAFETY MARGINS AND VALIDATION OF CRITICALITY CALCULATIONS FOR FISSILE MATERIAL LOADINGS

The Working Group found it impossible for several reasons to propose on more general grounds a simple, satisfactory, all-encompassing solution to the interlinked problems of criticality acceptance

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criteria for transport loads and validation procedures for criticality codes themselves. It was felt that any general advice would best consist of a discussion of important technical elements in establishing safety standards for criticality control.

First, even if general rules could be devised, they would probably have to be very restrictive. Setting a single acceptance value for k_{eff} (0.95 for instance) can be very costly when low enriched materials are involved, because it can lead to random and unduly large safety factors. In some cases when the critical mass of a material is well known from experiment, one can probably ensure that accumulations of the material remain safe simply by limiting them tc less than this amount.

Second, it is well known that there may be a systematic bias between calculated and experimental criticality values. However, as noted above, real package designs involve a very large number of combinations of packaging geometries and materials. An enormous number of critical experiments would be required to calibrate the key parameters (e.g., average energy of fission-causing neutrons) against the many variables involved.

Third, safety factors are inherently subjective, while safety calculation methods must be demonstratively conservative. Criticality calculations based on Monte Carlo approaches have an inherent uncertainty associated with them due to the statistical nature of the method. As a result, even if one allows a 3-sigma uncertainty band on the accuracy of the results calculated, there is still a 0.2% probability that the true k_{eff} lies outside this band.

It was noted that while regulations do vary between countries, a general "least common denominator" acceptance criterion for accumulation of fissile materials is that the corresponding computed k_{eff} should be less than 0.95 (after allowance has been made for all known code biases plus three times the uncertainties in code predictions and experimental values). While such a rule has the apparent virtue of simplicity for universal application, it presents certain disadvantages in practice.

For instance, packages with k_{∞} less than 1 (say 0.99) will always be subcritical. However, severe restrictions on the arrangement and compactness of any real arrays of such packages are necessary if k_{eff} is to be kept to less than 0.95. In contrast, under strictly controlled situations the maximum allowed k_{eff} for handling and storing metallic uranium may in practice be even as high as 0.97. In one country, the limit on k_{eff} is 0.95 for spent fuel in pond storage, but 0.98 for fresh fuel in dry storage. In another country the rule applied in some situations is that no more than three-fourths of a critical mass should be allowed to collect (unless, perhaps, neutron poisons are present). (It is also considered important to know the manner in which the fissile material can accumulate; whether quickly and undetectably, or in a slow, obvious manner.) In the same country, safety arguments are sometimes based on a low probability of accident occurrences (when, for instance, subcriticality cannot be guaranteed for all conceivable situations).

The criticality requirements in current international transport regulations are already strict; each package must be subcritical, five times the allowable number of packages in a stack must be subcritical, and twice the same number must still be subcritical even when the packages are assumed to be damaged and flooded in an accident. As suggested by the Working Group's exercises, perhaps the biggest problem lies not in proving that criticality codes can accurately deal with different loading situations, but rather in identifying the calculation bias arising from the many materials involved in real cases. In this regard, one country has conducted over 800 benchmark calculations to establish the bias that should be allowed in many cases of interest. (The Working Group suggests caution in this area; experiments tend to simplify the real situation, which raises the question of how to combine the biases established by different experiments when dealing with more complex real situations.)

In particular, the Working Group noted that there is a definite need for more experiments to improve the neutronic cross-section data available for some steels, and for lead in the form of both internal partitions and high-leakage external reflectors. On the basis of the experimental evidence

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examined in the Group's exercise, there is a need for more experiments on low-enriched, low H/U uranium oxide. The Working Group would like to emphasize the importance of a free exchange of all experimental data applicable to the criticality safety of the transport of fissile materials.

The Working Group strongly recommends that one should validate codes in a systematic fashion, taking great care in extrapolating data, and using a step-by-step approach in studying the influence of each parameter. Such care is necessary to characterize adequately the biases and uncertainties that arise in criticality calculations for the reasons discussed above.

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APPENDICES

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Appendix A

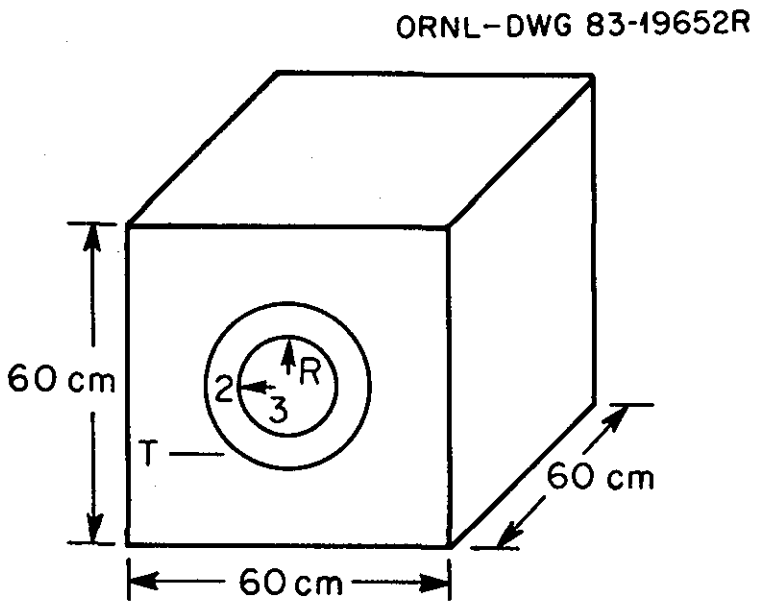
Detailed Specifications

Listing of Benchmark Problems

94230022

Simulated Transport Package 1

Uranium Metal



Unit Package Description

1. Void
2. Water (thickness "T") with density = 0.9982 g/cm^3
   P_H = 0.066742 x 10^{24} atoms/cm^3
   P_{oxy} = 0.33371 x 10^{24} atoms/cm^3
3. Fissile Material: Sphere (Radius R = 6.242 cm)
   $P_{^{235}U}$ = 0.048021 x 10^{24} atoms/cm^3

Array Description

512 units (8 x 8 x 8) with 30.48-cm-thick external water reflector

Water Thicknesses (T)

T = 0.0 cm
T = 2.54 cm
T = 10.16 cm

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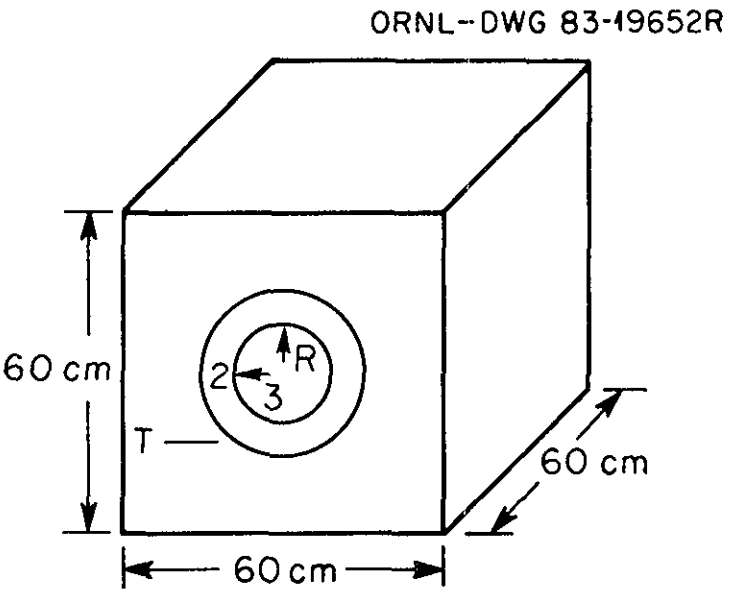
Simulated Transport Package 2

Uranium Nitrate



Unit Package Description

1. Void
2. Water (thickness "T") with density = 0.9982 g/cm^3
   P_H = 0.066742 x 10^{24} atoms/cm^3
   P_{oxy} = 0.33371 x 10^{24} atoms/cm^3
3. Fissile Material: Sphere (Radius R = 12.0 cm)
   $P_{^{235}U}$ = 0.0001537
   P_N = 0.0003075
   P_H = 0.0654579
   P_{oxy} = 0.0339588

Array Description

512 units (8 x 8 x 8) with 30.48-cm-thick external water reflector

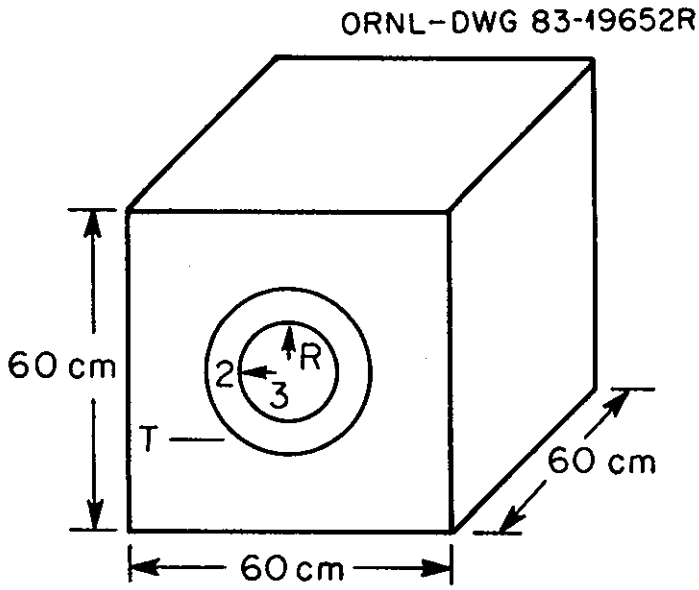
Water Thicknesses (T)

T = 0.0 cm
T = 2.0 cm
T = 2.8 cm

9423002

Simulated Transport Package 3

Dry 5% Enriched Uranium Oxide



Unit Package Description

1. Air
   $P_N = 0.00004325 \times 10^{24}$ atoms/cm^3
   $P_{oxy} = 0.00001081 \times 10^{24}$ atoms/cm^3
2. Water (thickness "T") with density 1.0 g/cm^3
3. Fissile Material: Sphere (Radius R = 20.5 cm)
   $P_{^{235}U} = 0.0005648 \times 10^{24}$ atoms/cm^3
   $P_{^{238}U} = 0.0105952 \times 10^{24}$ atoms/cm^3
   $P_{oxy} = 0.02232 \times 10^{24}$ atoms/cm^3

Array Description

512 units (8 x 8 x 8) with 20.0-cm-thick external water reflector

Water Thicknesses (T)

T = 0.0 cm
T = 2.2 cm
T = 4.0 cm

94230025

Simulated Transport Package 4

Damp 5% Enriched Uranium Oxide

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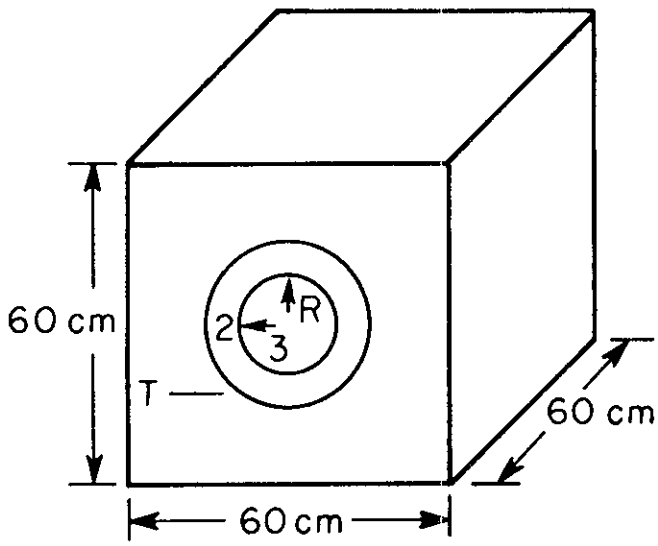

Unit Package Description

1. Air
   $P_N = 0.00004325 \times 10^{24}$ atoms/cm^3
   $P_{oxy} = 0.00001081 \times 10^{24}$ atoms/cm^3
2. Water (thickness "T") with density 1.0 g/cm^3
3. Fissile Material:
   $P_{^{235}U} = 0.000149 \times 10^{24}$ atoms/cm^3
   $P_{^{238}U} = 0.002796 \times 10^{24}$ atoms/cm^3
   $P_{oxy} = 0.03534 \times 10^{24}$ atoms/cm^3
   $P_H = 0.0589 \times 10^{24}$ atoms/cm^3

Array Description

512 units (8 x 8 x 8) with 20.0-cm-thick external water reflector

Water Thicknesses (T)

T = 0.0 cm
T = 4.0 cm

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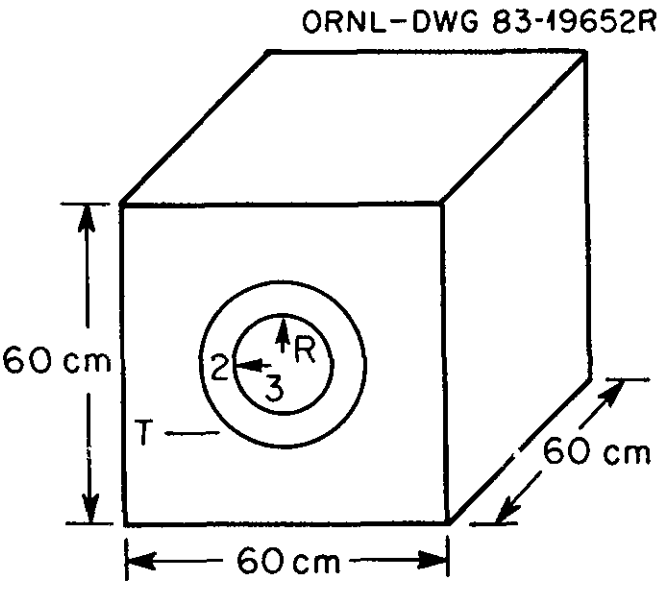
Simulated Transport Package 5

Dry Plutonium Oxide



Unit Package Description

1. Void
2. Water (thickness "T") with density 1.0 g/cm^3
   $P_H = 0.06686 \times 10^{24}$ atoms/cm^3
   $P_{oxy} = 0.03343 \times 10^{24}$ atoms/cm^3
3. Fissile Material: Sphere (Radius R = 10.7 cm)

   $P_{^{239}Pu} = 0.0066689 \times 10^{24}$ atoms/cm^3
   $P_{^{240}Pu} = 0.0027671 \times 10^{24}$ atoms/cm^3
   $P_{^{241}Pu} = 0.0013227 \times 10^{24}$ atoms/cm^3
   $P_{^{242}Pu} = 0.00032931 \times 10^{24}$ atoms/cm^3
   $P_{oxy} = 0.022176 \times 10^{24}$ atoms/cm^3

Array Description

512 units (8 x 8 x 8) with 20.0-cm-thick external water reflector

Water Thicknesses (T)

T = 0.0 cm
T = 2.2 cm
T = 6.0 cm

940330027

Experimental Problem 1

Arrays of Uranium Metal

Plexiglas Moderator: (Methacrylate Plastic)
(60.5 wt % C, 8.3 wt % H) density = 1.18 g/cm ^{3}
Cuboid dimensions:
Outside base = 21.4 x 21.4 cm
Outside height = 20.7 cm
Wall thickness = 2.38 cm

Paraffin Reflector: ($C_{25}H_{52}$), density = 0.93 g/cm^3

Reflector Location: Reflector spaced from peripheral uranium cylinders a distance equal to one-half the surface separation between cylinders in the array, i.e., at peripheral "cell" boundaries.

Uranium Cylinders: Uranium metal cylinders
diam., cm 11.506
height, cm 10.765
mass, kgU 20.960
Uranium isotopic content:
93.2 wt % ^{235}U, 5.6 wt % ^{238}U,
1.0 wt % ^{234}U, 0.2 wt % ^{236}U.

Table - Critical Arrays of Uranium Cylinders Centered in Plexiglas Cuboids

Problem	Array n_x,n_y,n_z	Paraffin refl. thickness, cm	U-cylinder center separation,* cm	
			Horizontal	Vertical
1.1	2x2x2	0	21.745	21.004
1.2		15.2	27.953	27.212
1.3	3x3x3	0	27.795	27.054

*Error of separation of cylinders in unreflected arrays is ±0.13 cm;
and ±0.026 cm in the reflected array.

Reference: J. T. Thomas, "Critical Three-Dimensional Arrays of U(93.2)-Metal Cylinders," *Nucl. Sci. Eng.* **52**, 350-359 (1973).

94230028

Experimental Problem 2

Arrays of Uranyl Nitrate

Containers:

Methacrylate plastic (Plexiglas, density 1.18 g/cm^3)
Nominal 0.64-cm-thick wall; outside height = 19.05 cm and outside diam. = 20.32 cm
Volume of solution per container 5 liters ± 0.3 cm^3

Aqueous Uranyl Nitrate Solution:

$UO_2(NO_3)_2$, sp. gr. = 1.083, 0.05845 gU/g-soln., 0.0633 gU/cm^3.
Total nitrate corresponds to N:U = 2.006
Uranium isotopic content:
92.6 wt % ^{235}U,
5.9 wt % ^{238}U, 1.0 wt % ^{234}U,
and 0.5 wt % ^{236}U
H/^{235}U = 441

Table - 27 Unit Unreflected Critical Array

Problem	Array	Container avg. tare wt.(kg)	Unit surface separation (horizontal and vertical) (cm)
2.	3x3x3	1.280	2.41 ± 0.13

Reference: J. T. Thomas, "Critical Three-Dimensional Arrays of Neutron Interacting Units," ORNL/TM-719 (October 1963).

94030029

Experimental Problem 3

4.46% Enriched Uranium Oxide, H/X = 0.77

Title: 100 units of $U(4.46)_3$ O_8-H_2O (H/U = 0.77), separated by 0.923-cm-thick Plexiglas plates and fully reflected by Plexiglas

Geometry: Array geometry shown in Figures 1 and 2. The "Core Box" or standard fuel unit is a cubical box, 15.3 cm on each side. This is an external dimension which includes twice the 0.15-cm-thick aluminum box wall. Material and atomic compositions are given in Tables A1 and A2.

Reflecting Walls: The reflector consists of plain Plexiglas (designated "P") and Plexiglas containing Tris (designated "T"). The distribution of the two types of Plexiglas and the reflector dimensions are shown in Figures 1 and 2.

Critical Configuration: The measured critical configuration includes a 1.05-cm air gap as shown in Figure 1. The reactivity worth of the gap has been calculated to be approximately 1.4% $\Delta k/k$.

Problem Exercise: Problem 3.1 An infinite array of the "Core Box" units separated by 0.923-cm-thick Plexiglas plates

Problem 3.2 The configuration as shown in Figures 1 and 2 with the exception that the air gap be omitted.

Reference: R. E. Rothe, I. Oh, G. R. Goebel, "Critical Experiments with Interstitially-Moderated Arrays of Low-Enriched Uranium Oxide," NUREG/CR-1071 (RFP-3008) (1980).

94230030

Experimental Problem 4

Damp 4.89% Enriched Uranium Oxide

Title: Mixture of $U(4.89)_3O_8$ - $(C_{17}H_{35}CO_2)_3C_3H_5$ (H/U = 9.86), reflected by H_2O and paraffin

Geometry: The fuel mixture occupies a rectangular parallelepiped of dimensions 40.64 x 40.64 x 58.17 cm^3. Assembly is reflected on the bottom and sides by 18 cm of H_2O and on the top by 15.74 cm of paraffin as shown in Figure 1. Atomic compositions are given in Table A3.

Critical Configurations: The dimensions of the critical assembly were obtained by interpolation between two assemblies, one subcritical and the other supercritical, differing only by one 10.16 x 10.16 x 10.16 cm^3 unit of the fuel material. This implies a maximum critical mass uncertainty of approximately 1%.

Problem Exercise: Problem 4.1 An infinite medium of the fuel mixture

Problem 4.2 The configuration as shown in Figure 1.

Reference: D. F. Cronin, "Critical Mass Studies, Part X," ORNL-2968 (September 1960).

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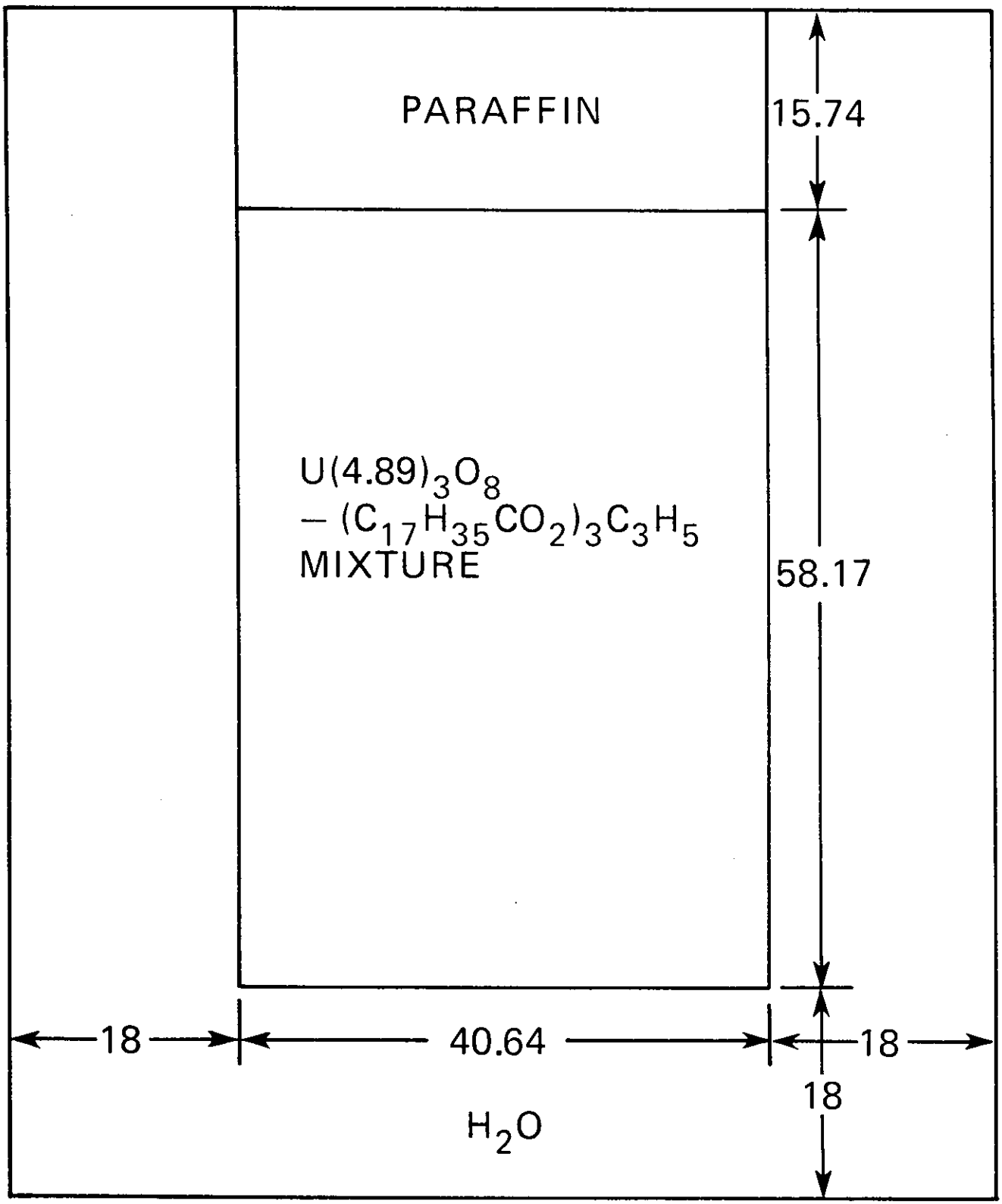
ORNL–DWG 83-19653



DIMENSIONS IN cm

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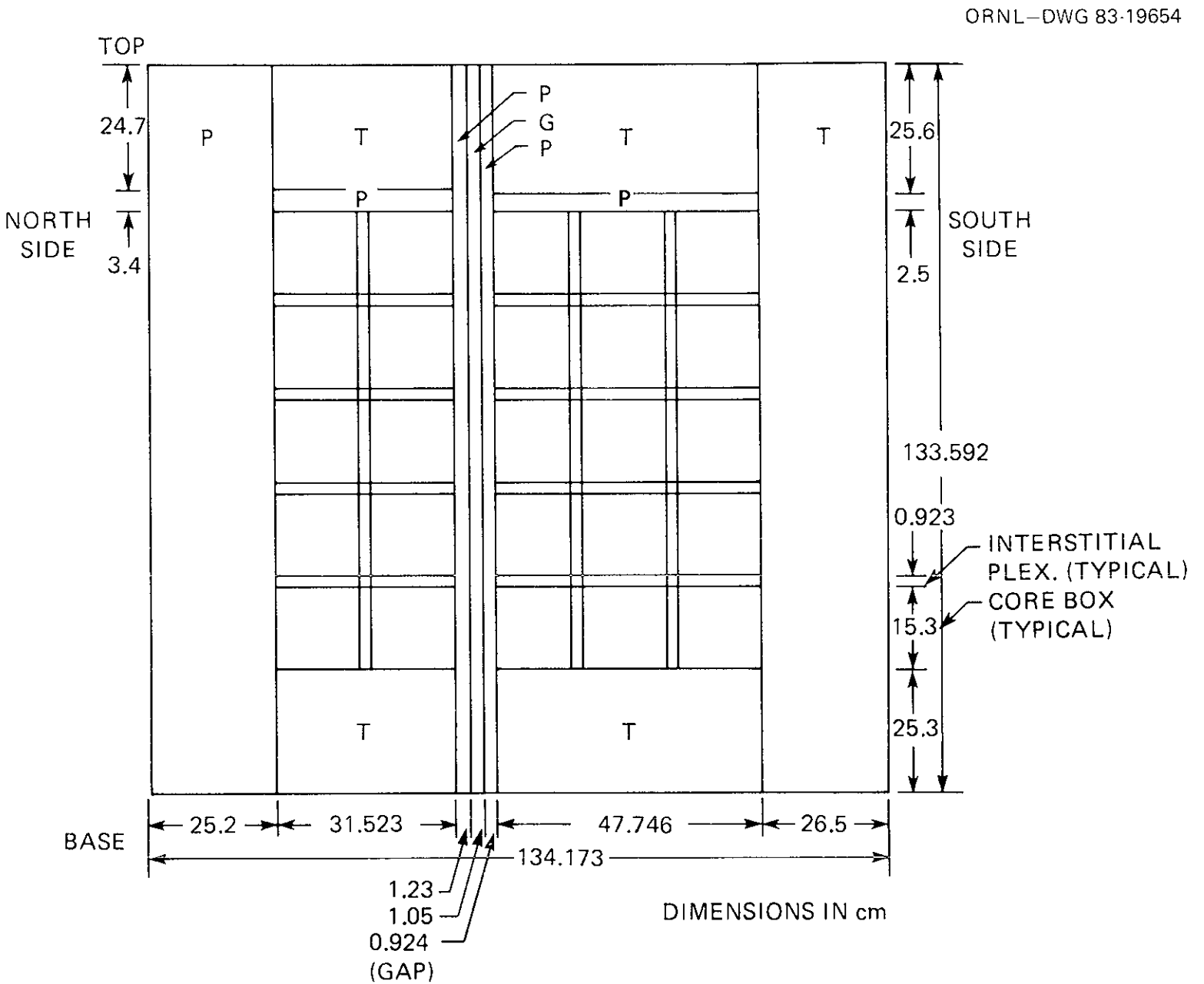
ORNL–DWG 83-19654



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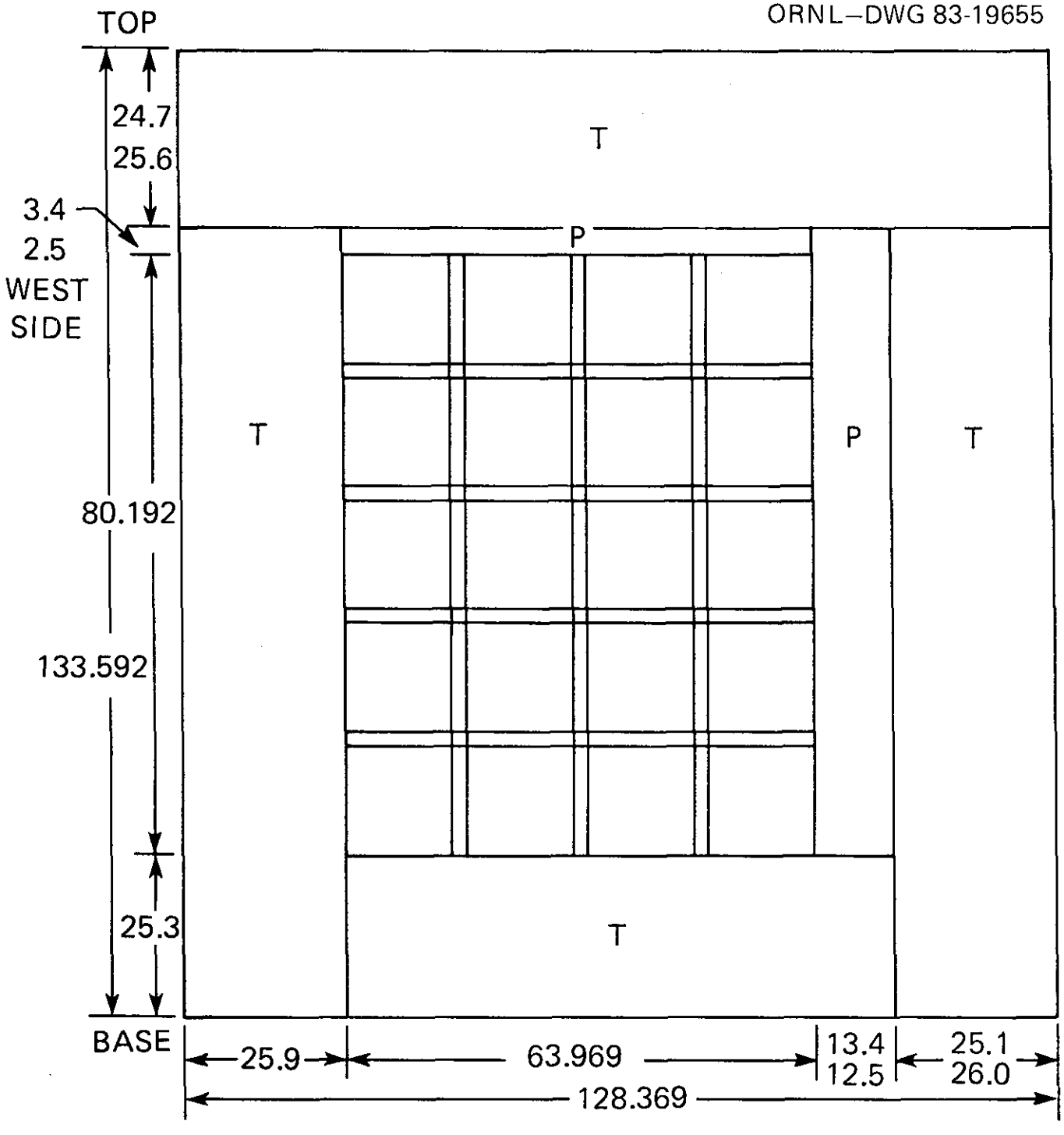

PAIRS OF DIMENSIONS READ: NORTH TABLE DIMENSION
SOUTH TABLE DIMENSION

DIMENSIONS IN cm

94230034

Table A1. Material compositions

	U_3O_8	Core Box H_2O	Synthetics	Core + Reflector P(Plain Plex.)	Reflector T(Plex. + "Tris")
			Densities (g/cm^3)		
	4.47	9.304×10^{-2}	1.778×10^{-2}	1.185	1.284
			Weight Percentages		
Hydrogen	-	11.19	13.13	7.83	7.16
Carbon	-	-	81.66	59.49	52.03
Oxygen	15.51	88.81	3.4	32.48	29.82
Phosphorous	-	-	-	-	1.02
Chlorine	-	-	1.29	-	1.81
Calcium	-	-	0.35	-	-
Bromine	-	-	-	-	7.10
Uranium[a]	84.49	-	-	-	-
	100.00	100.00	99.83	99.8	98.94

[a]Uranium Enrichment, wt % ^{234}U:0.03, ^{235}U:4.46,^{236}U:0.08, ^{238}U:95.43

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Table A2. Atomic Concentrations $10^{24}/cm^3$

Nuclide (Symbol)	Core Box (U_3O_8 + H_2O + Synthetics)	Aluminum Box	Plain Plexiglas(P)	Plexiglas and Tris(T)
Hydrogen (H)	7.617×10^{-3}	-	5.679×10^{-2}	5.493×10^{-2}
Carbon (C)	7.280×10^{-4}	-	3.549×10^{-2}	3.350×10^{-2}
Oxygen (O)	2.871×10^{-2}	-	1.420×10^{-2}	1.441×10^{-2}
Aluminum (Al)	-	6.024×10^{-2}	-	-
Phosphorous (P)	-	-	-	2.547×10^{-4}
Chlorine (Cl)	3.896×10^{-6}	-	-	3.948×10^{-4}
Calcium (Ca)	9.351×10^{-7}	-	-	-
Bromine (Br)	-	-	-	6.871×10^{-4}
^{234}U	2.925×10^{-6}	-	-	-
^{235}U	4.330×10^{-4}	-	-	-
^{236}U	7.734×10^{-6}	-	-	-
^{238}U	9.148×10^{-3}	-	-	-

Table A3. Atomic compositions $10^{24}/cm^3$

Nuclide (Symbol)	Fuel Mixture $U(4.89)_3O_8 + (C_{17}H_{35}CO_2)_3C_3H_5$	H_2O	Paraffin
Hydrogen (H)	3.314×10^{-2}	6.676×10^{-2}	8.258×10^{-2}
Carbon (C)	1.7173×10^{-2}	-	3.970×10^{-2}
Oxygen (O)	1.0766×10^{-2}	3.338×10^{-2}	-
^{234}U	6.8×10^{-7}	-	-
^{235}U	1.6628×10^{-4}	-	-
^{238}U	3.1926×10^{-3}	-	-

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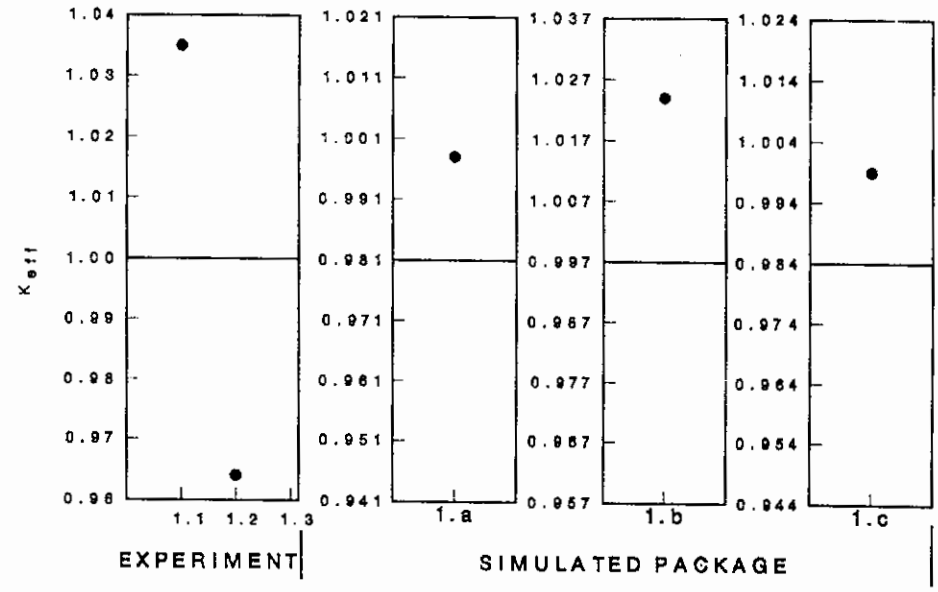
Appendix B

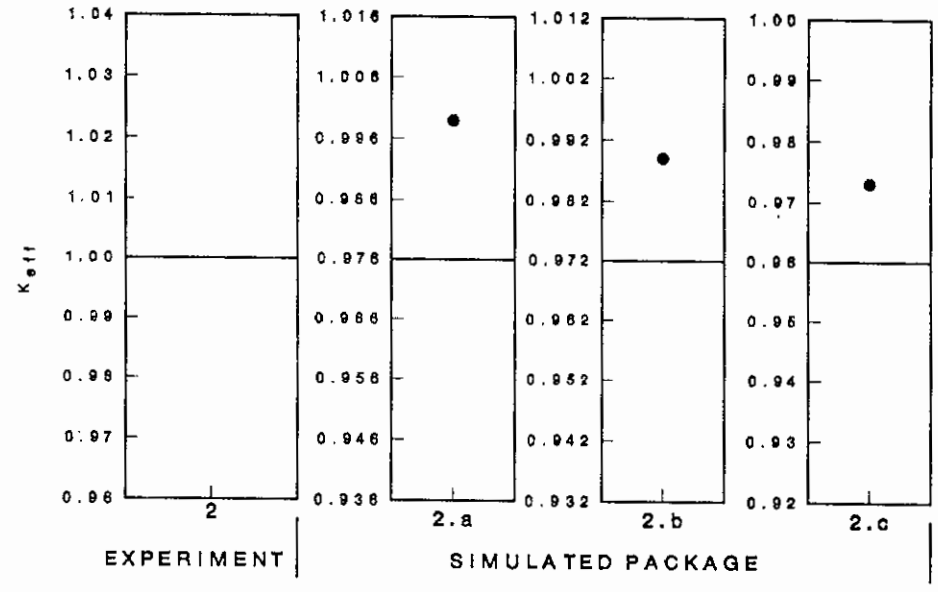
Graphical Presentation of Results

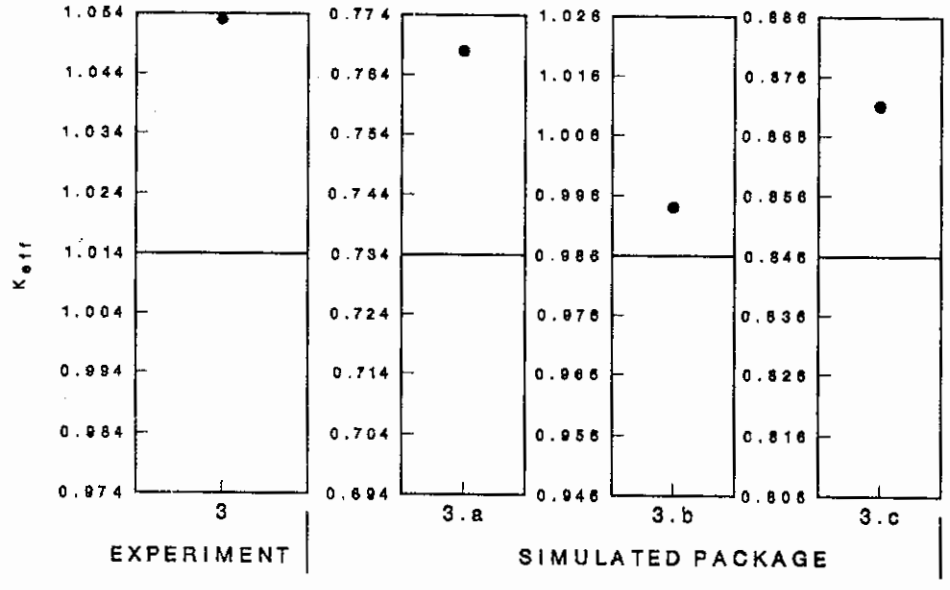
Shown in Tables 1 and 2

94230037

EIR Results Program SPHERE ORNL-DWG 84C-12727







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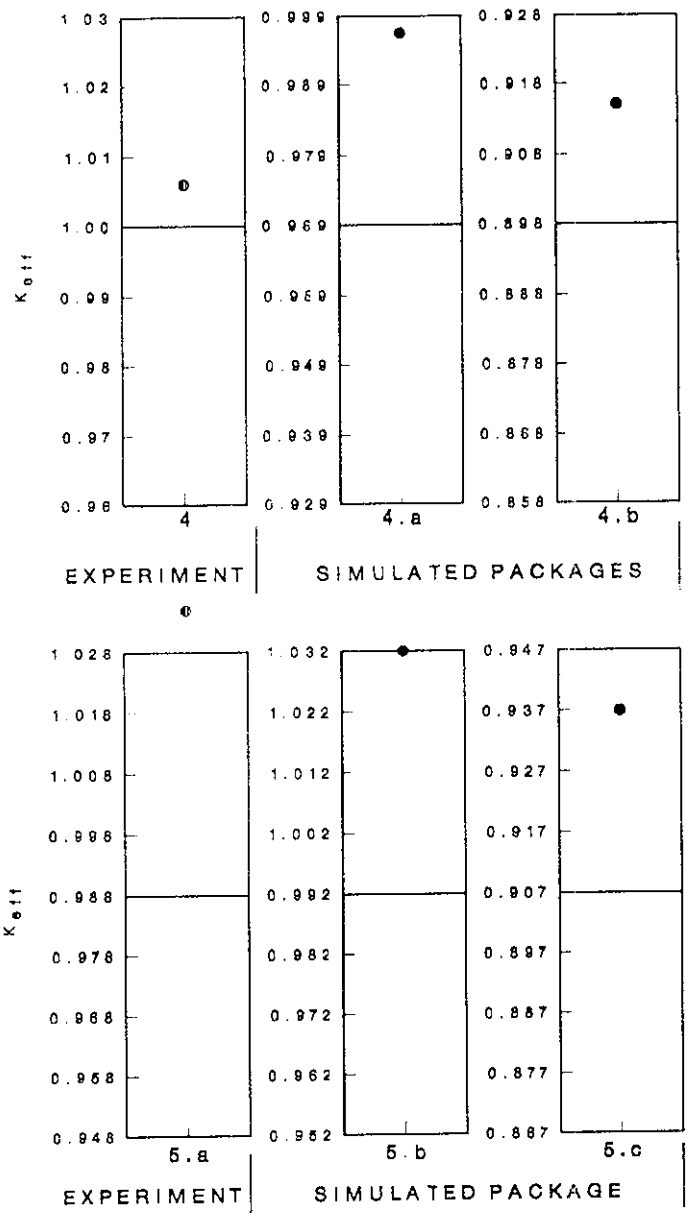
EIR Results Program SPHERE (continued)

ORNL-DWG 84C-8751



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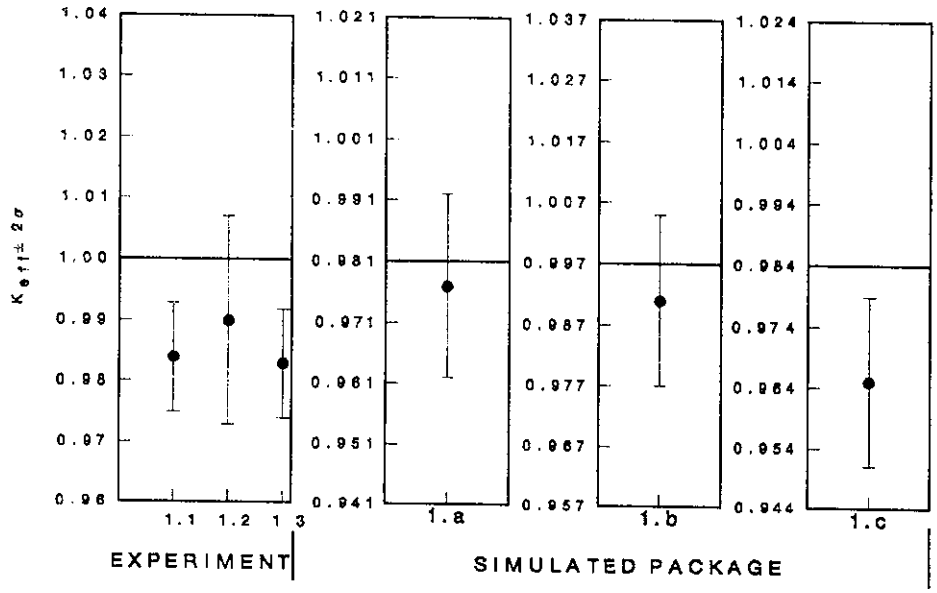
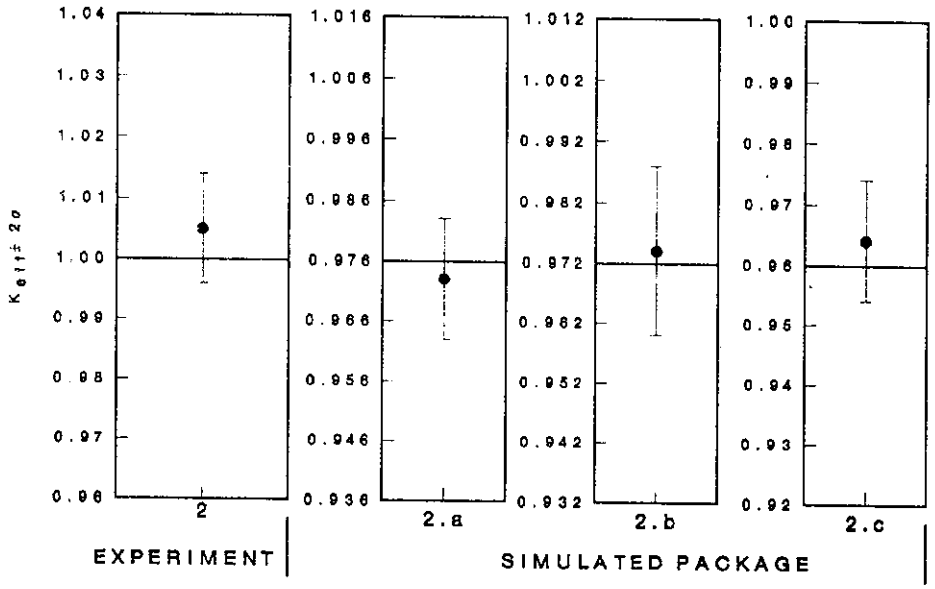
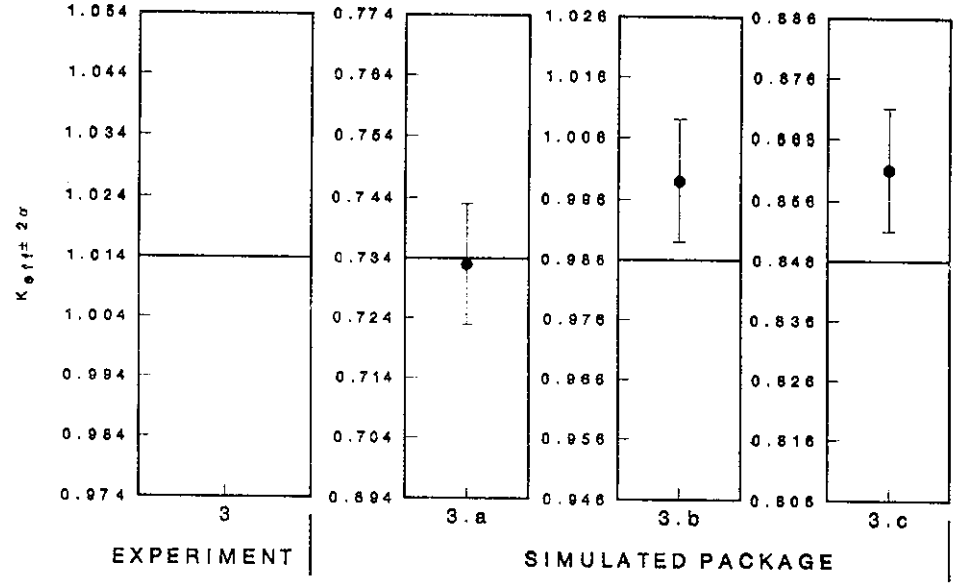
CEA Results Program MORET Hansen-Roach Cross Sections ORNL-DWG 84C-12728







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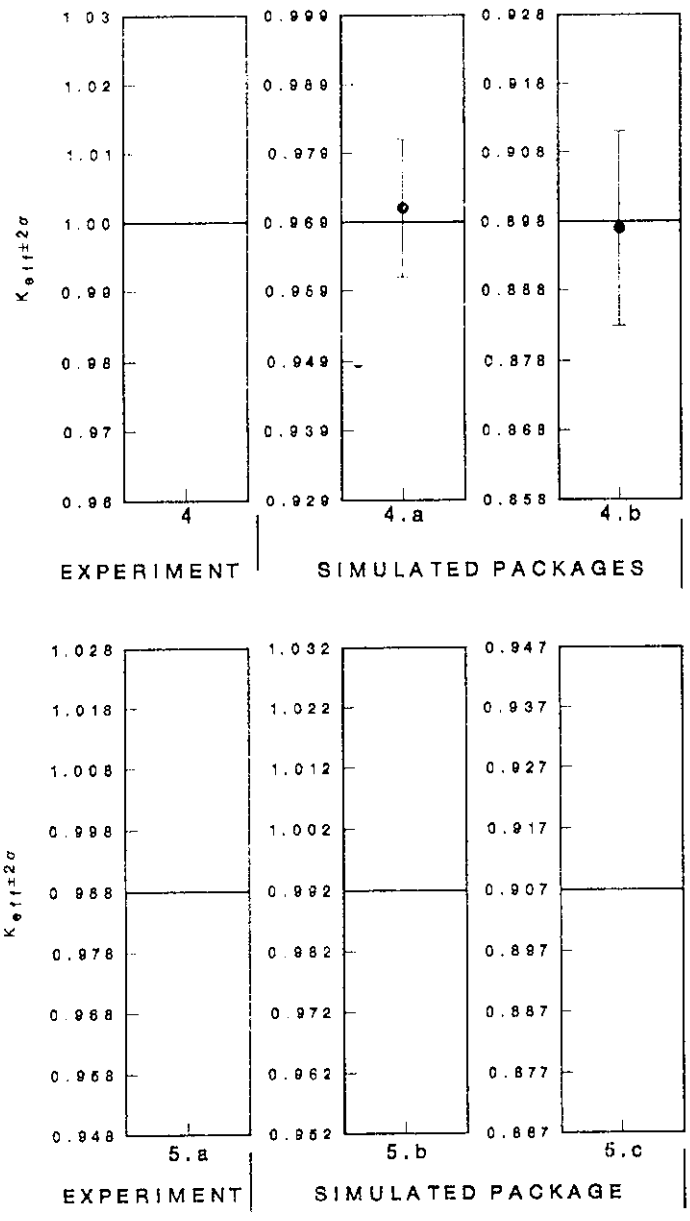
CEA Results Program MORET Hansen-Roach Cross Sections (continued)

ORNL-DWG 84C-8752



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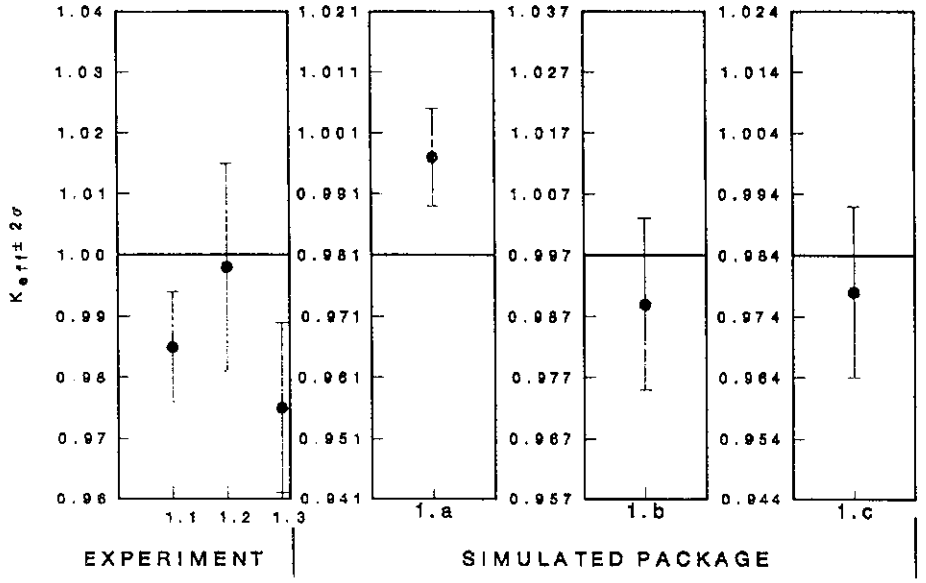
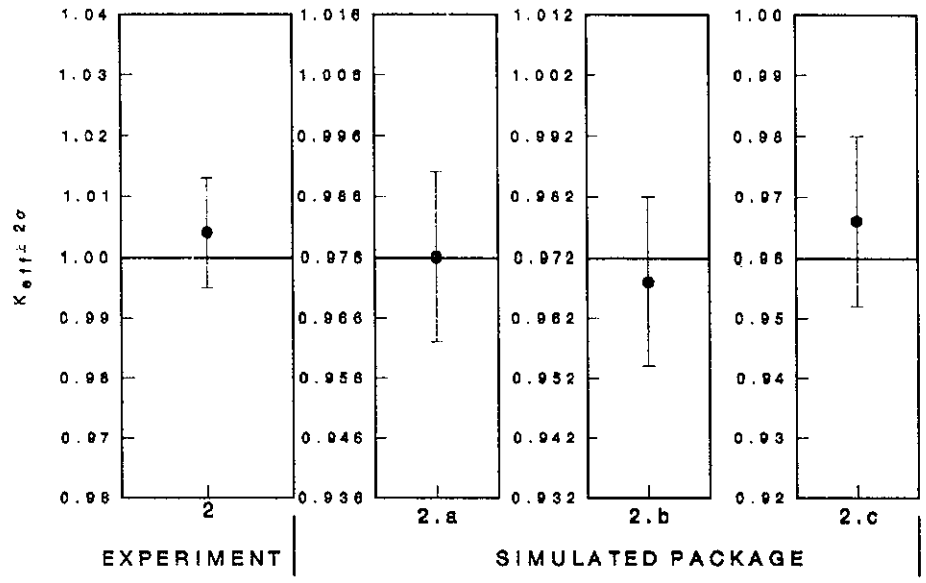
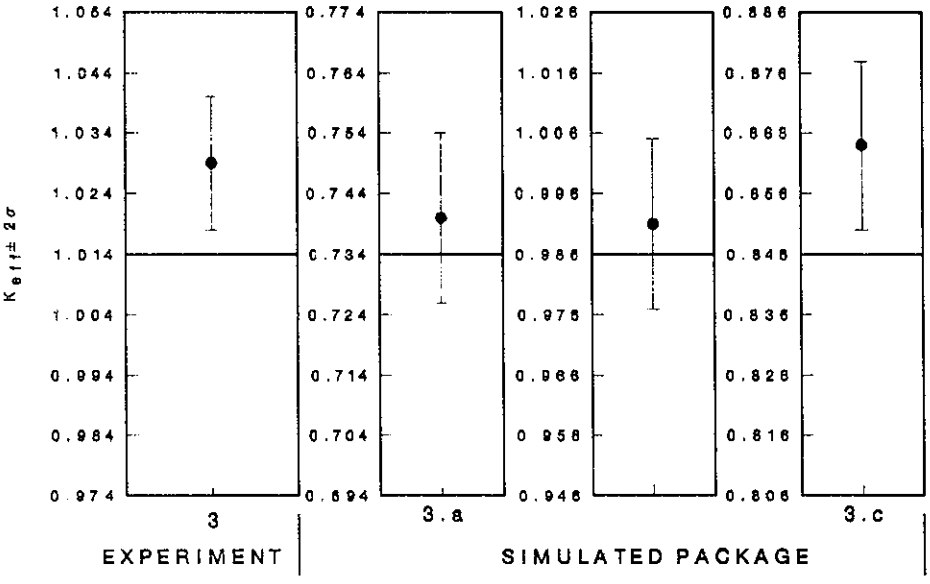
CEA Results Program MORET APOLLO Cross Sections ORNL-DWG 84C-12729







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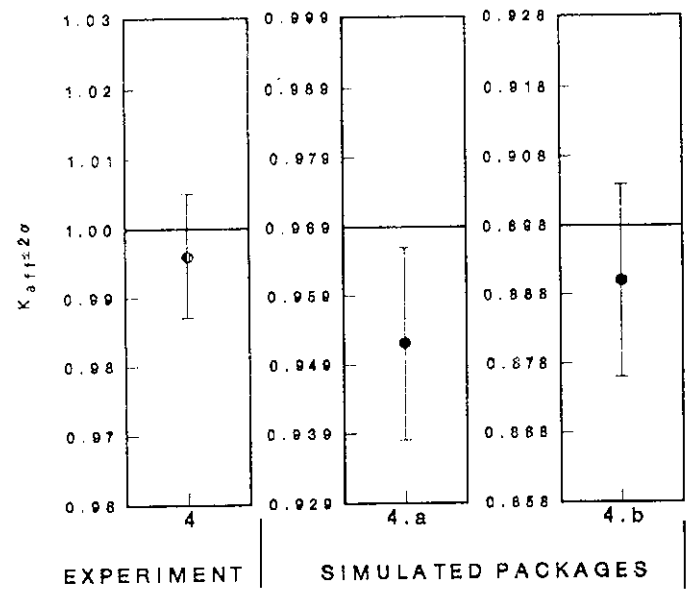
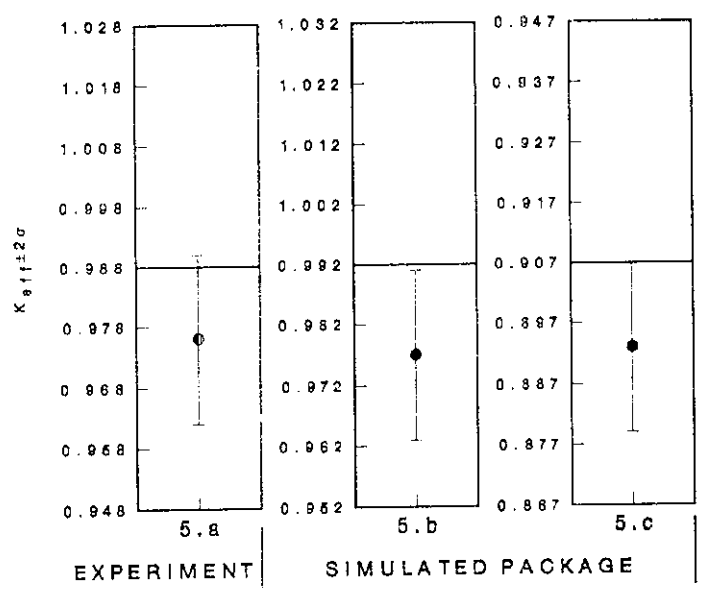
CEA Results Program MORET APOLLO Cross Sections (continued)

ORNL-DWG 84C-8753





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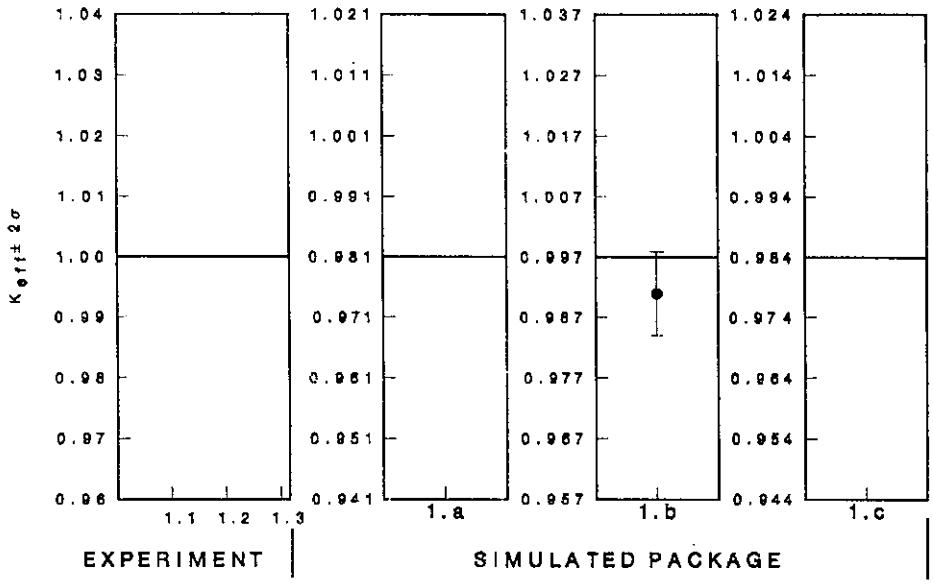
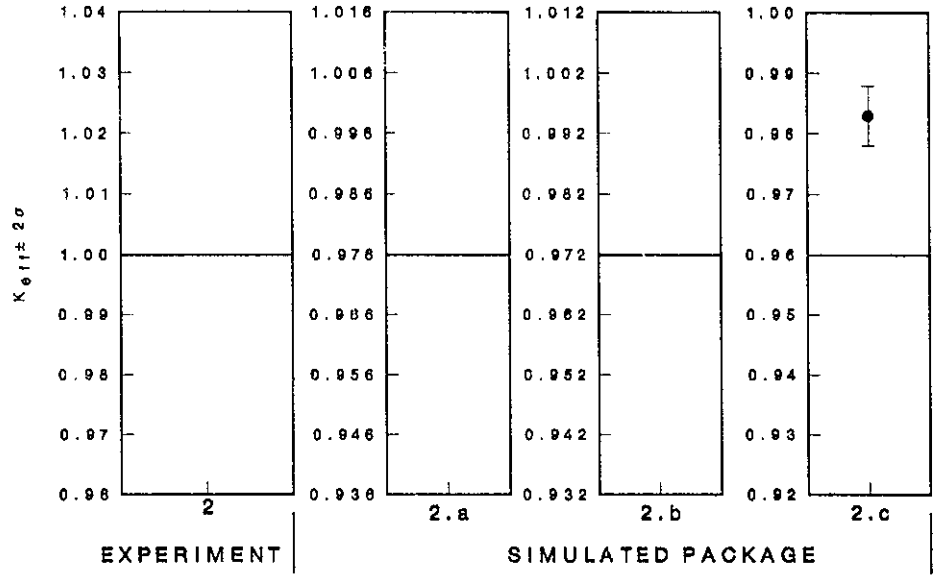
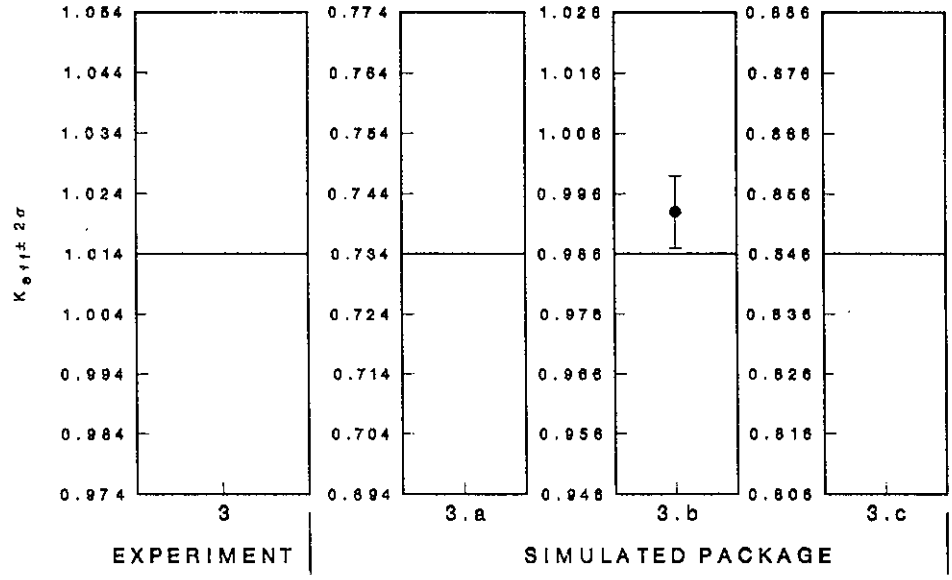
CEA Results Program TRIMARAN APOLLO Cross Sections ORNL-DWG 84C-12730







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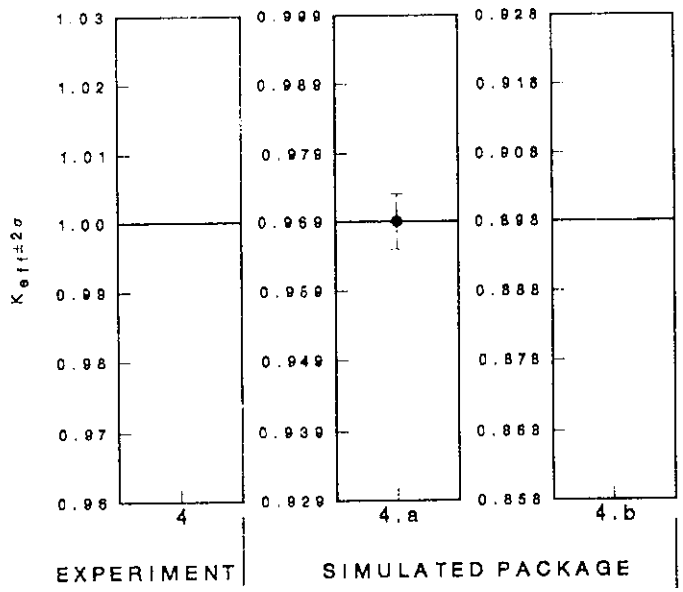
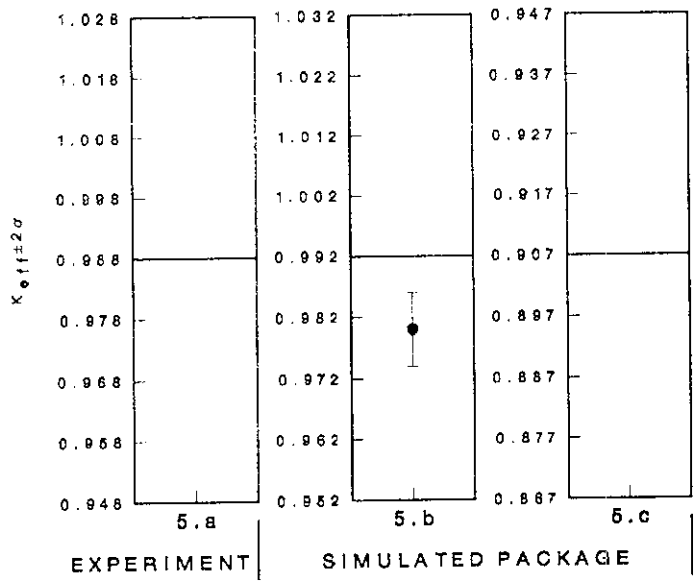
CEA Results Program TRIMARAN APOLLO Cross Sections (continued)

ORNL-DWG 84C-8754

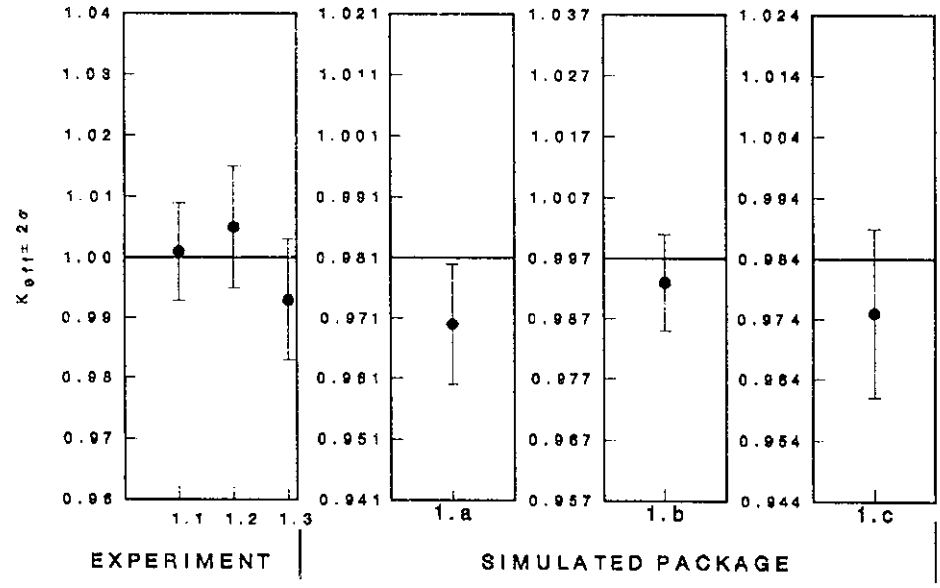


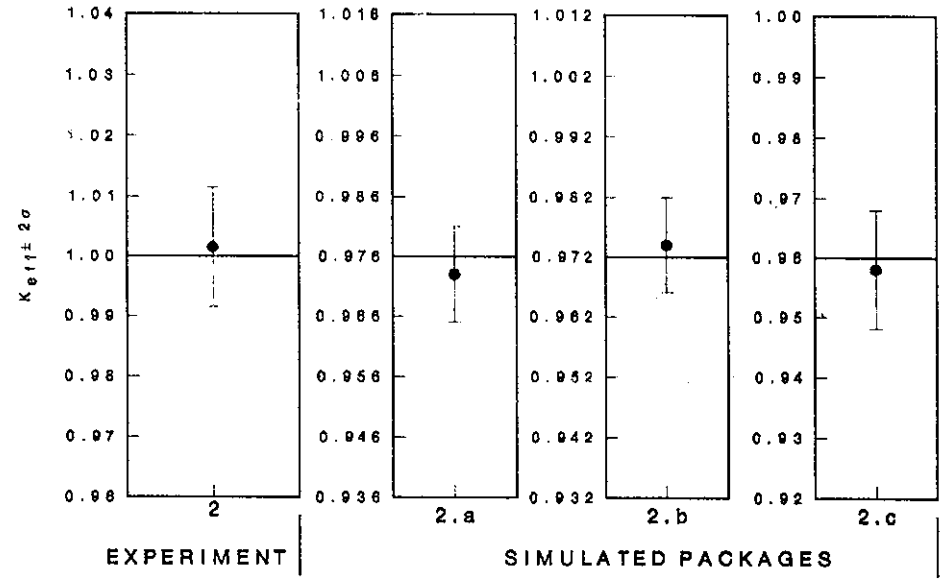


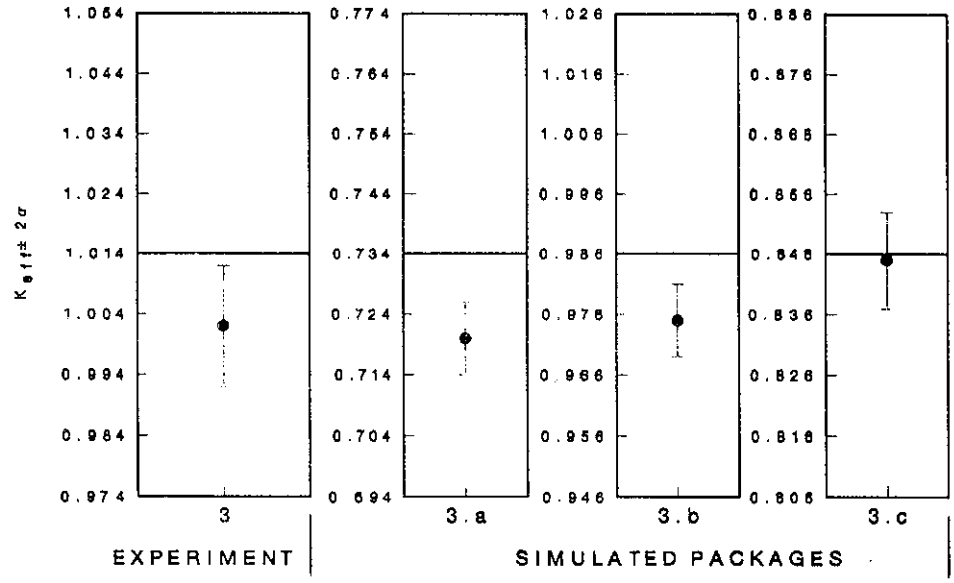
9420045

ORNL Results Program KENO SCALE 27 GP Cross Sections

ORNL-DWG 84C-12731







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ORNL Results Program KENO SCALE 27 GP Cross Sections (continued)

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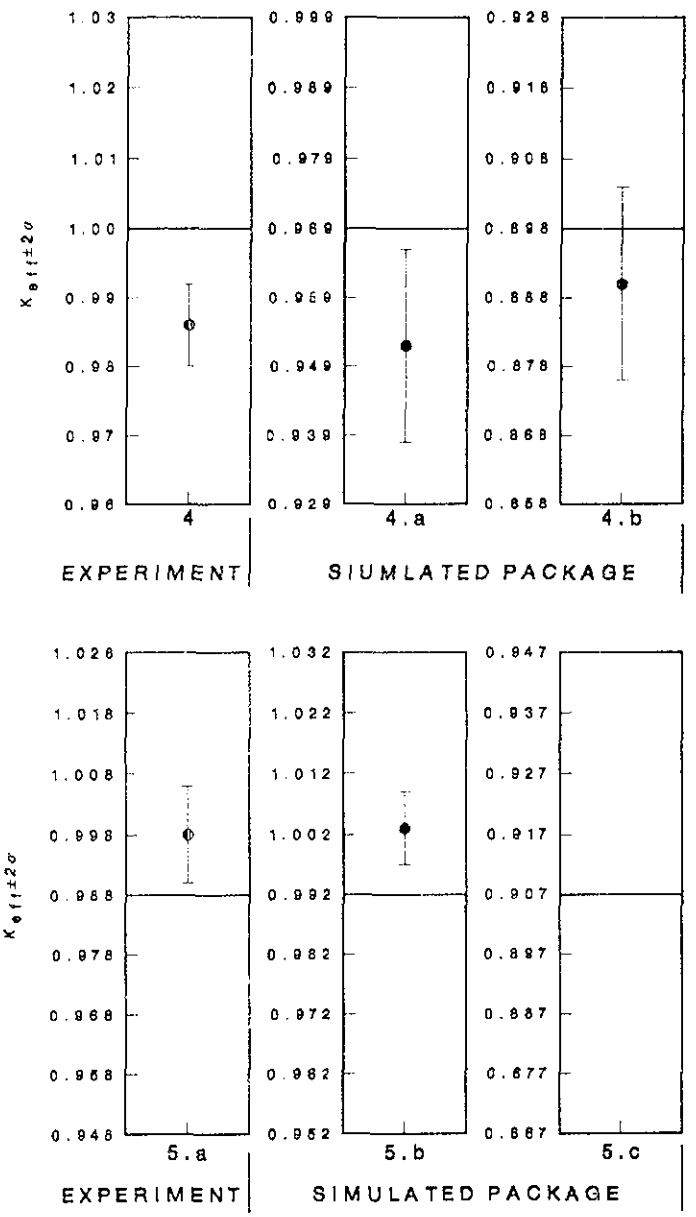

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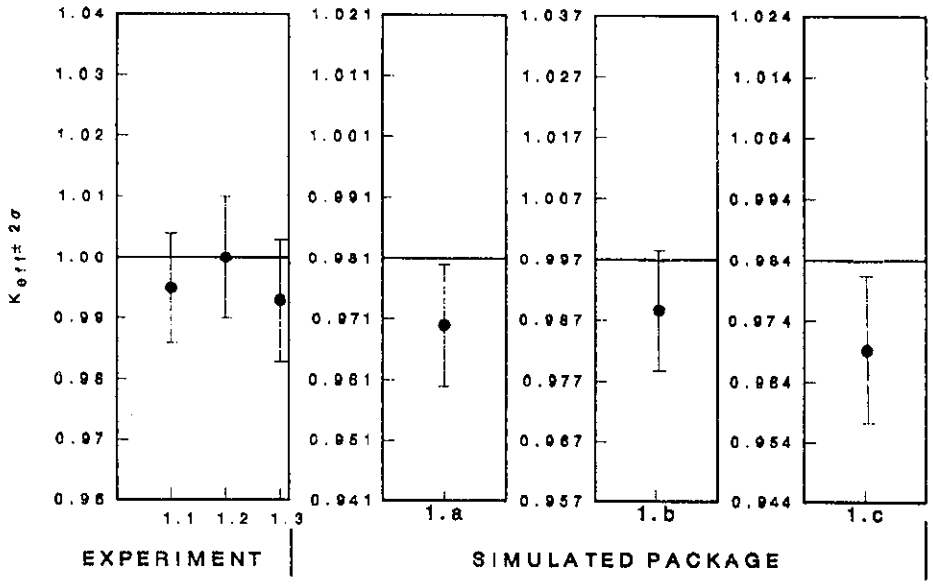
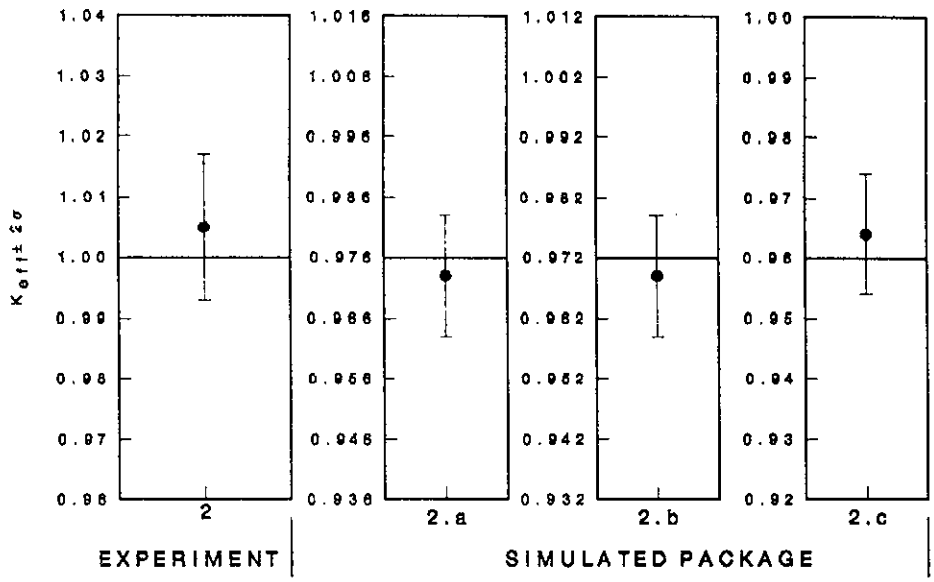
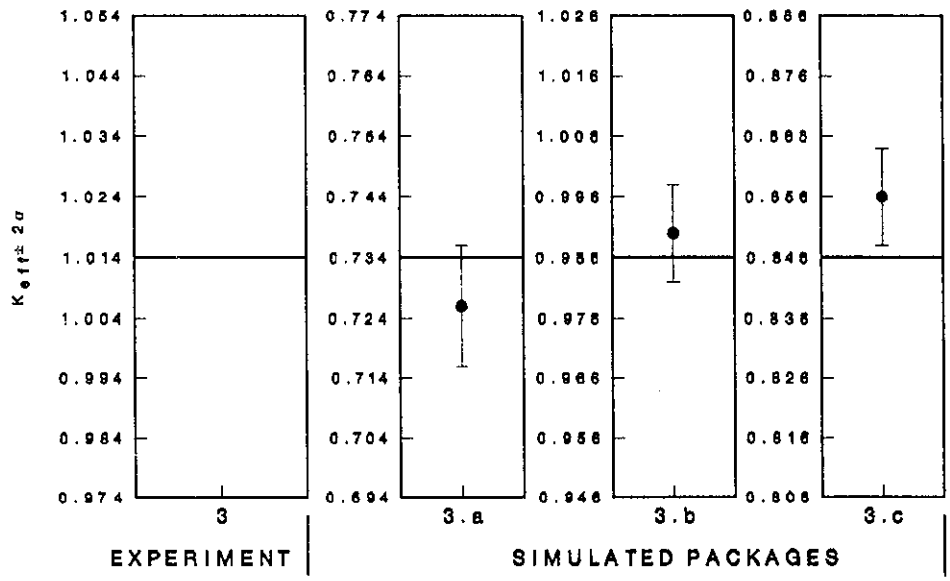
GRS Results Program KENO Hansen-Roach Cross Sections

ORNL-DWG 84C-12732







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GRS Results Program KENO Hansen-Roach Cross Sections (continued)

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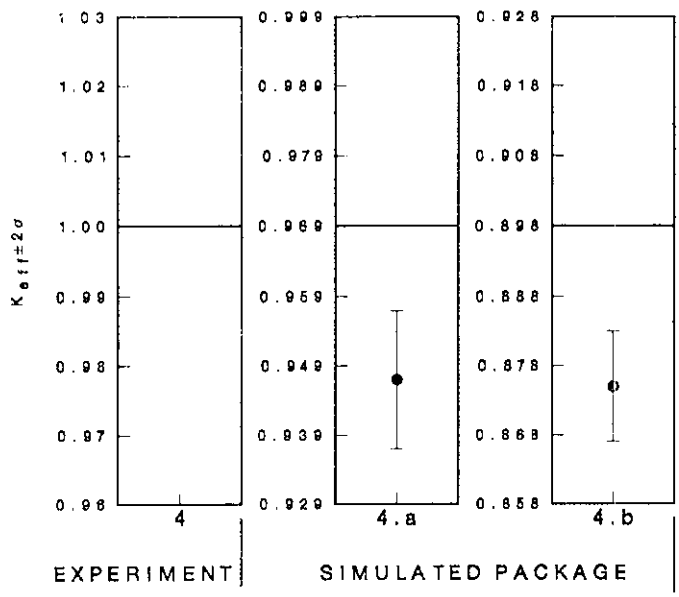

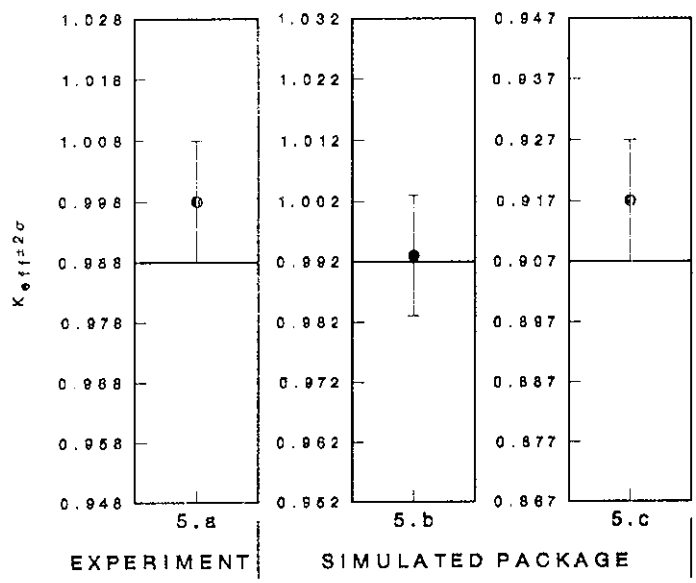

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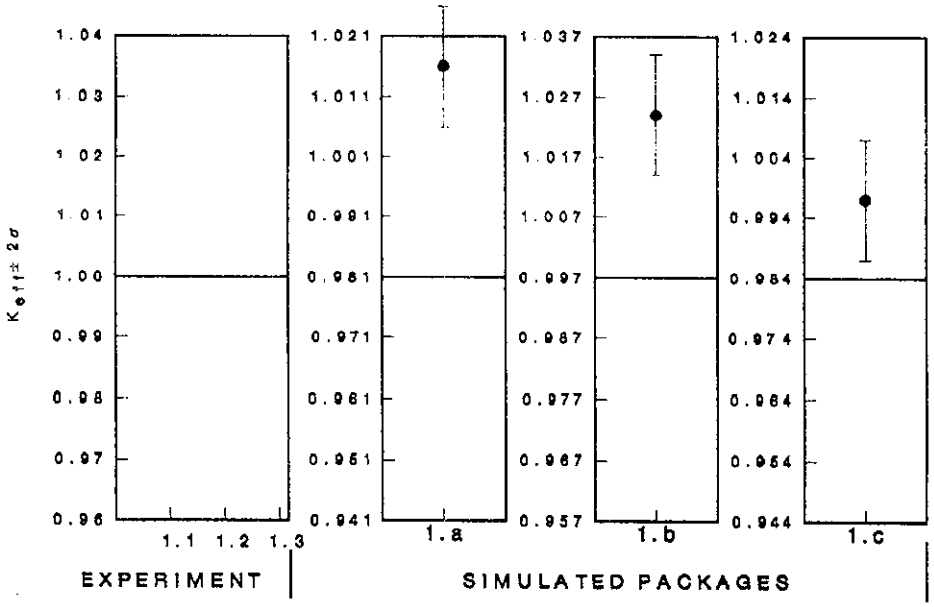
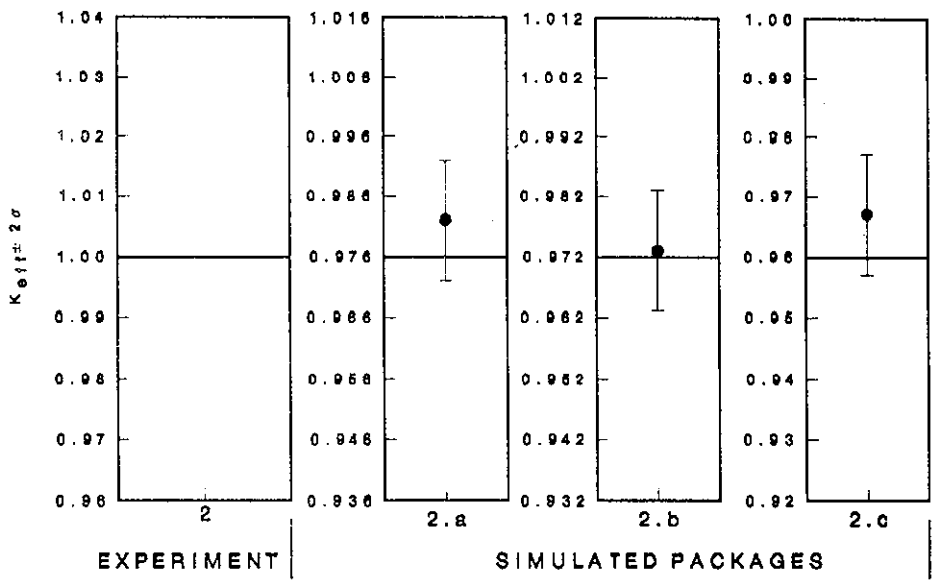
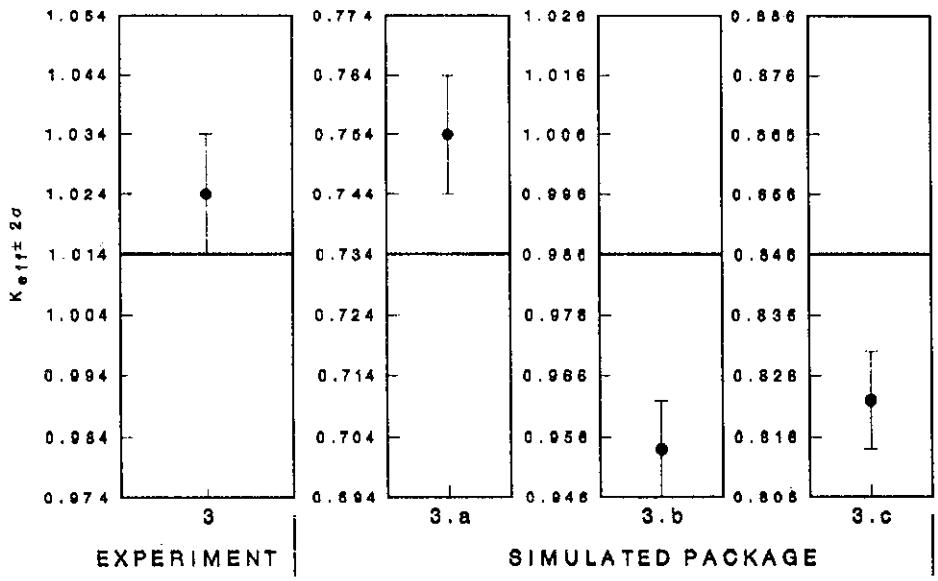
GRS Results Program KENO GAMTEC-II Cross Sections ORNL-DWG 84C-12733







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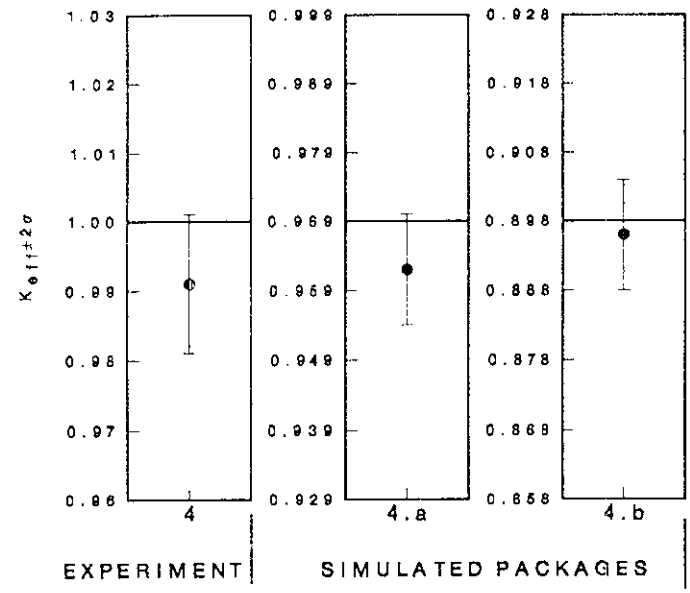
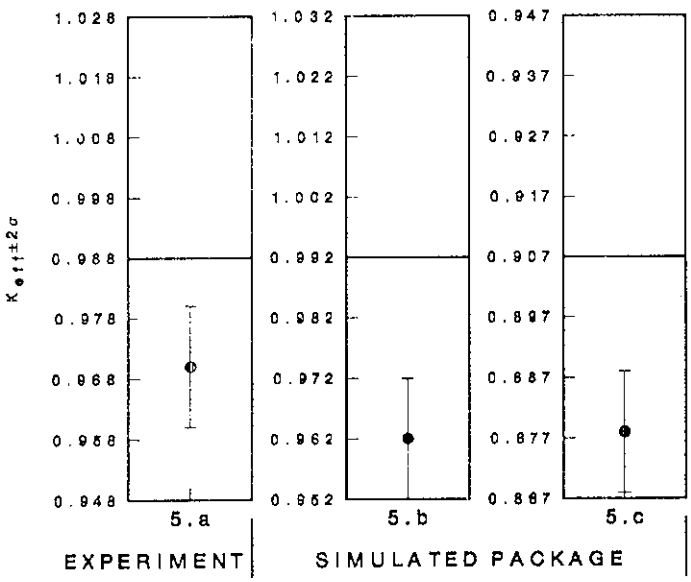
GRS Results Program KENO GAMTEC-II Cross Sections (continued)

ORNL-DWG 84C-8757





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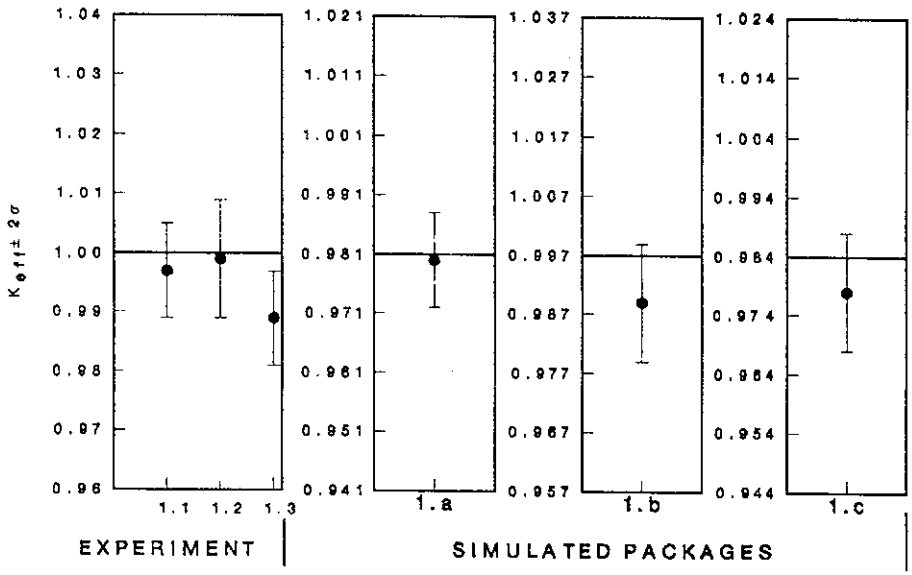
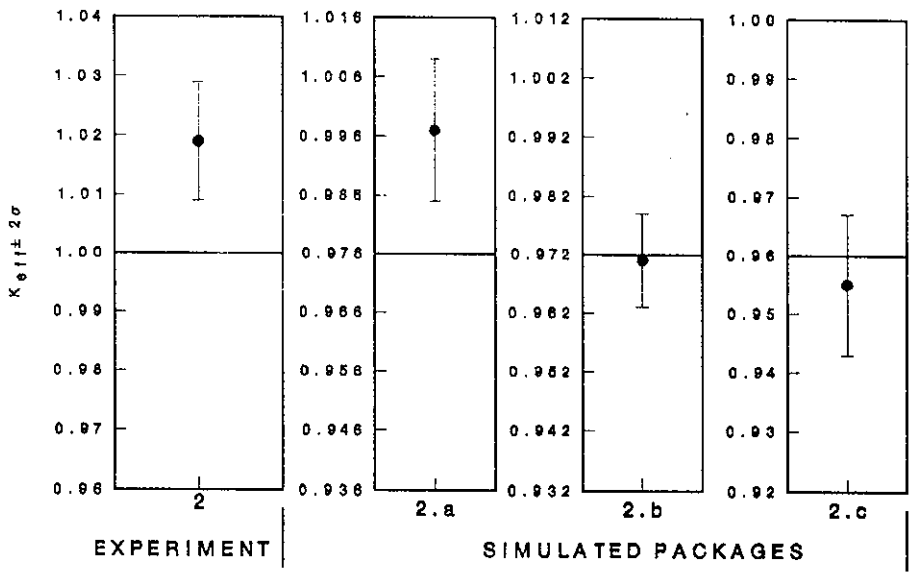
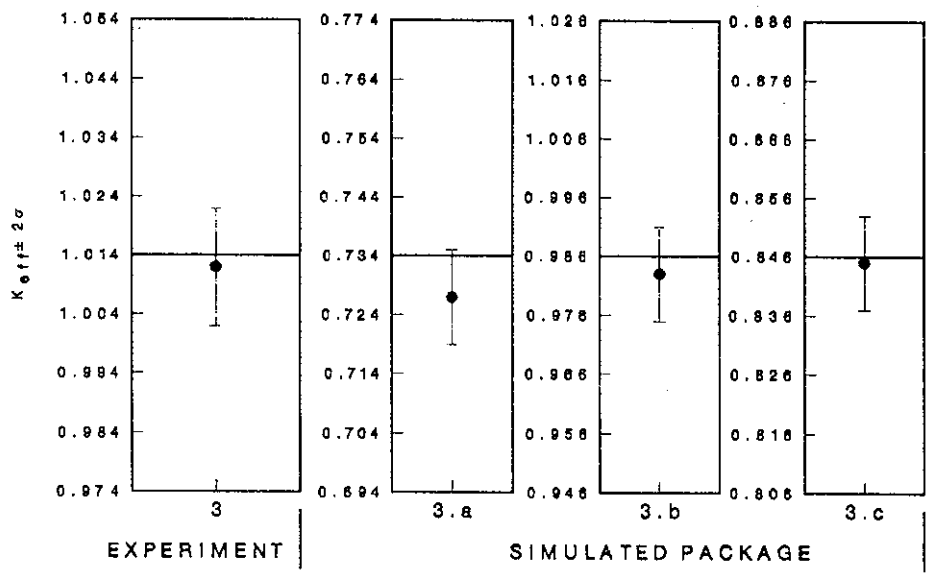
ENEA Results Program KENO SCALE Cross Sections

ORNL-DWG 84C-12735







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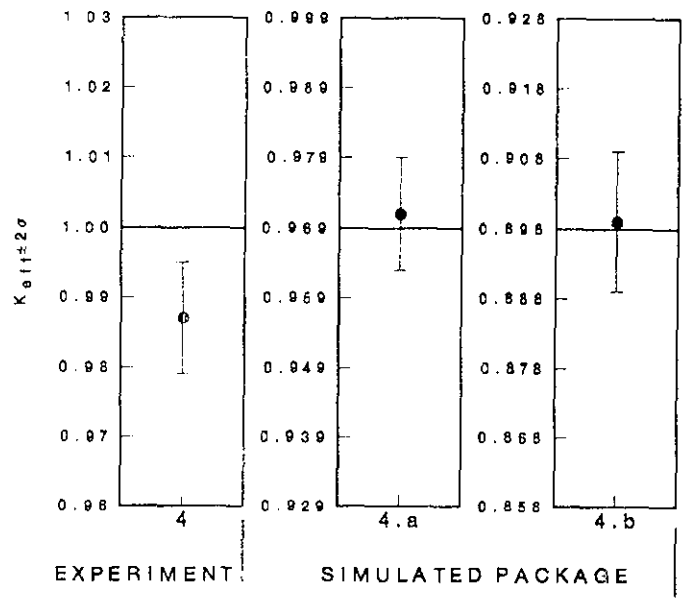
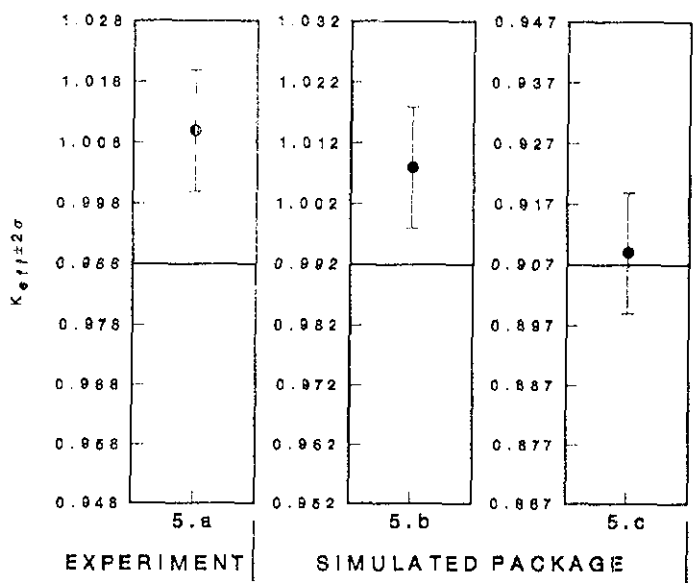
ENEA Results Program KENO SCALE Cross Sections (continued)

ORNL-DWG 84C-8759





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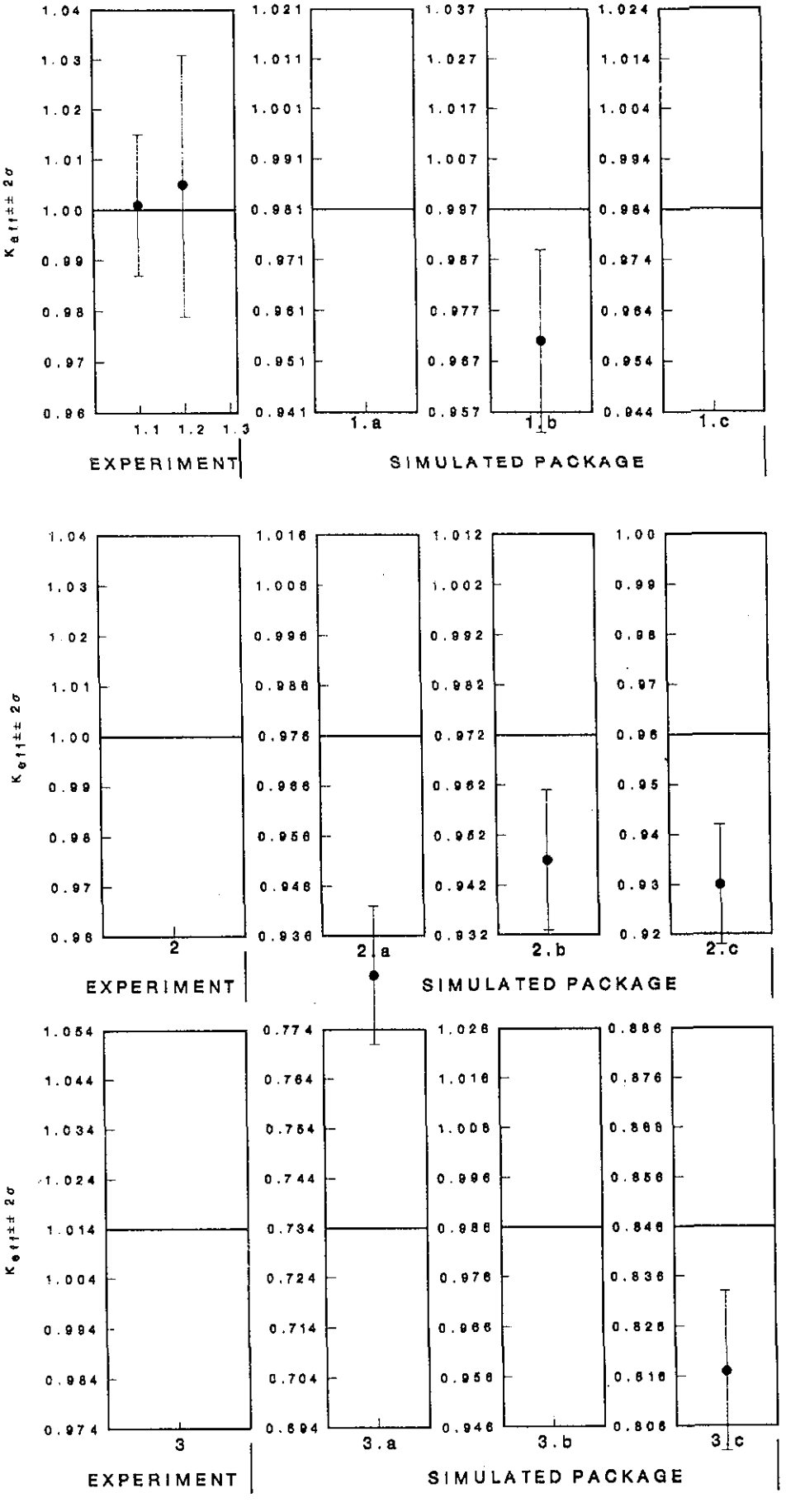
PTB Results Program MORSE-K RSYST Cross Sections

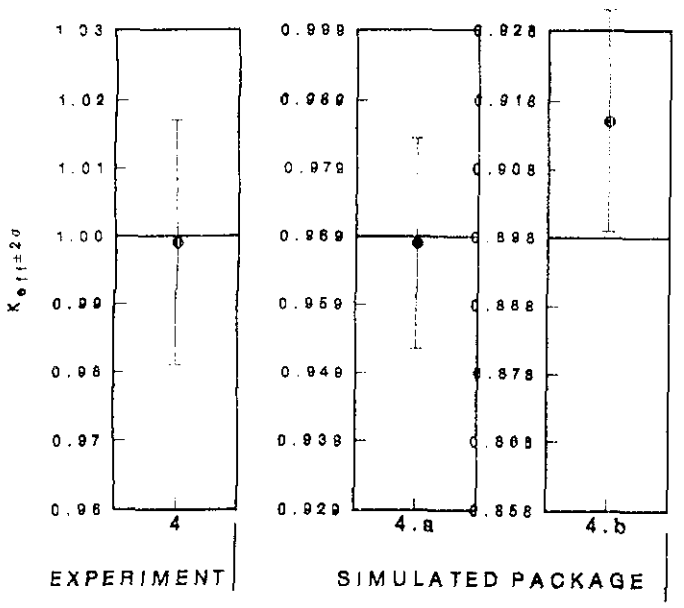
ORNL-DWG 84C-12734

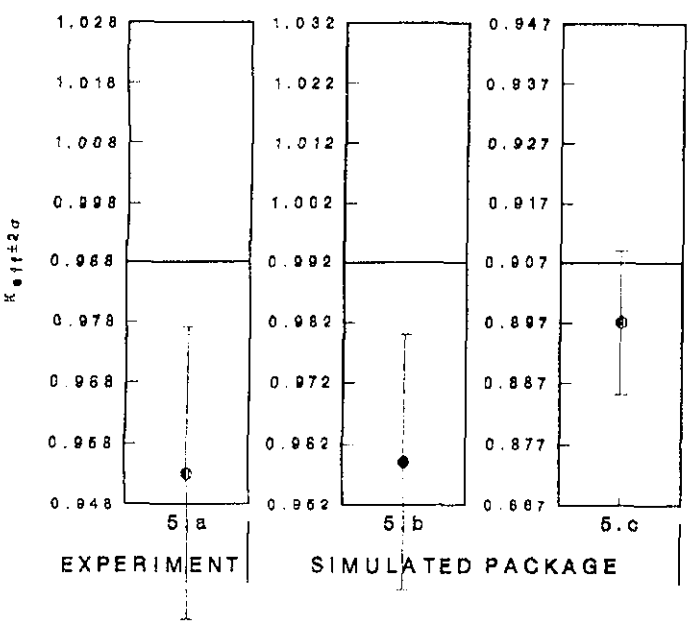


94030054

PTB Results Program MORSE-K RSYST Cross Sections (continued)

ORNL-DWG 84C-8758





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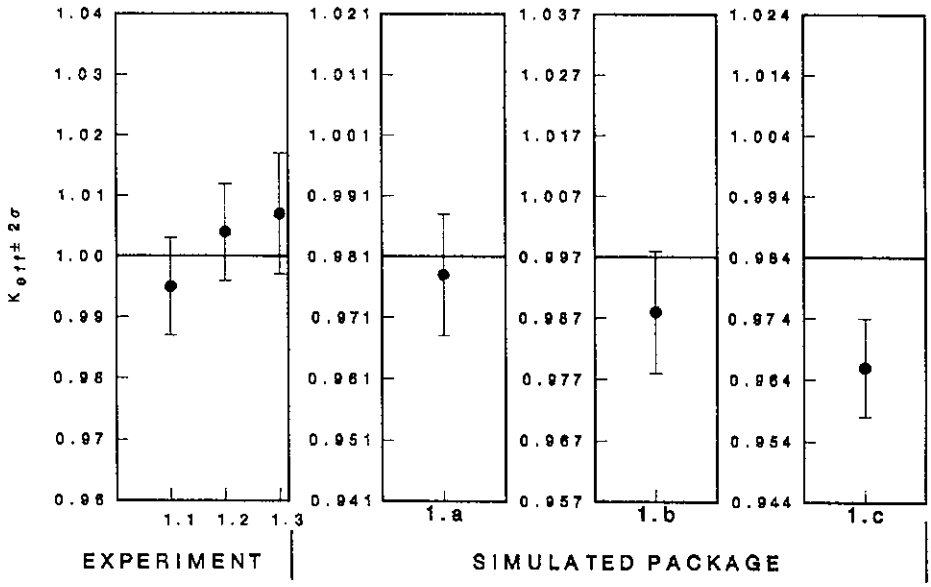
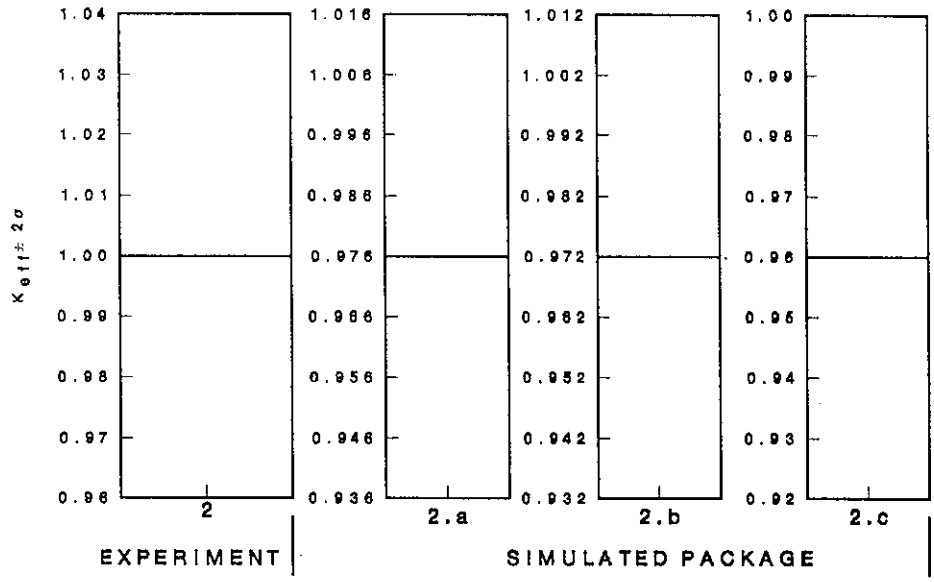
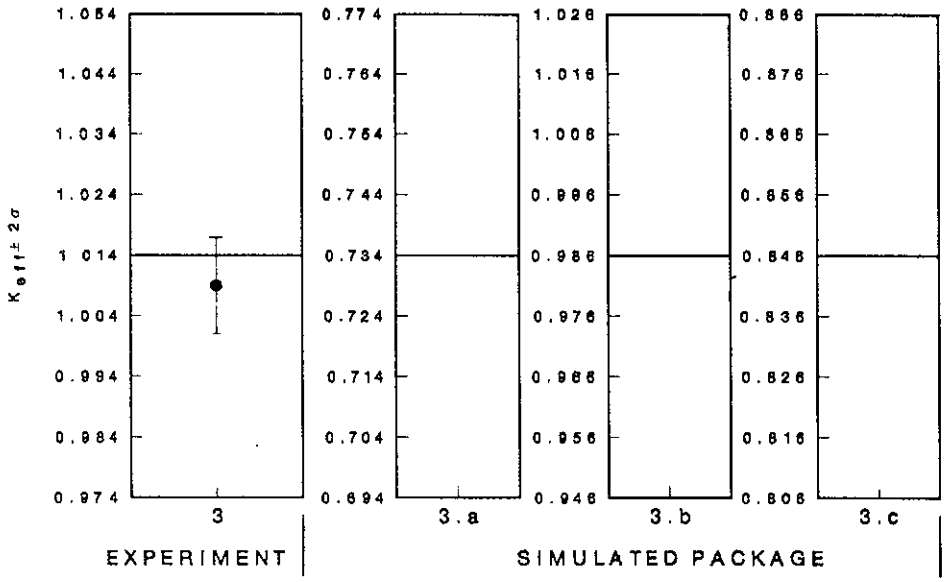
ENEA Results Program KENO GAM-THERMOS Cross Sections

ORNL-DWG 84C-12736



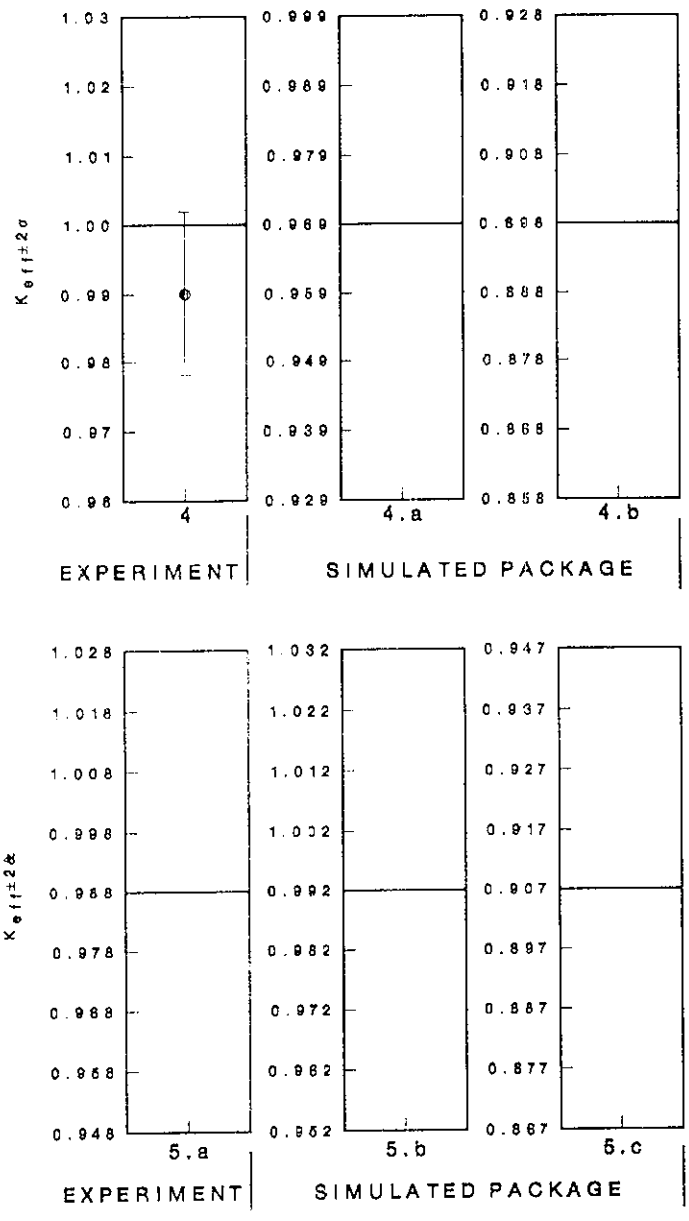




94230056

ENEA Results Program KENO GAM-THERMOS Cross Sections (continued)

ORNL-DWG 84C-8760



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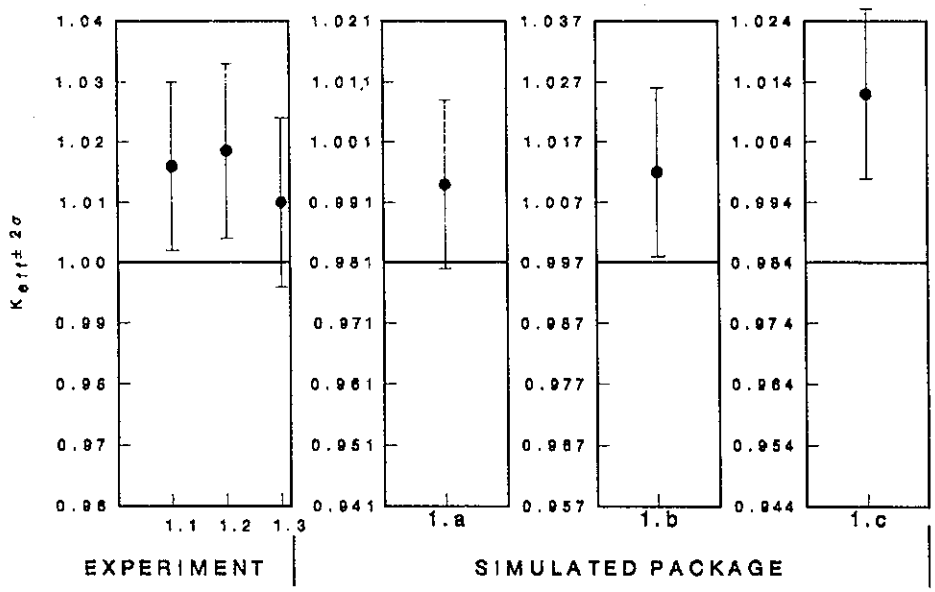
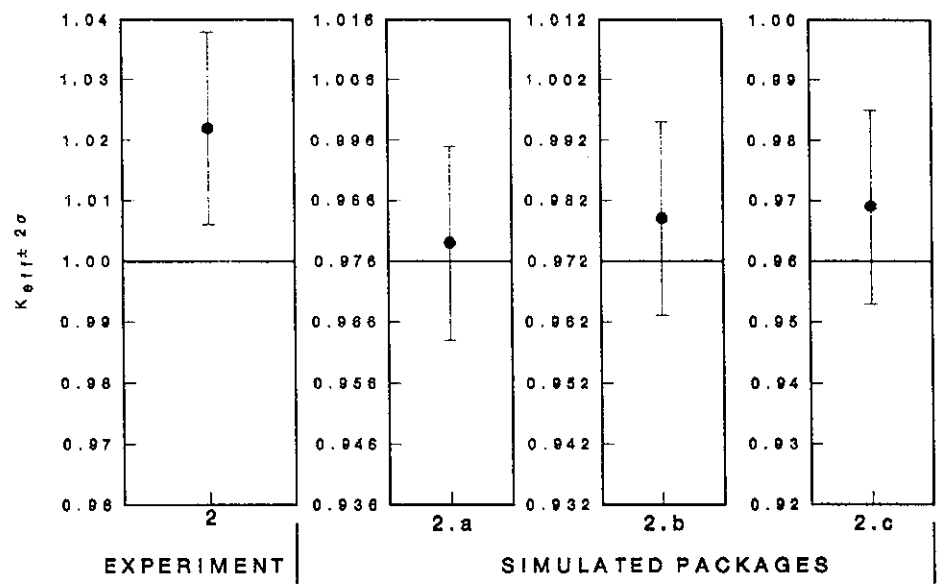
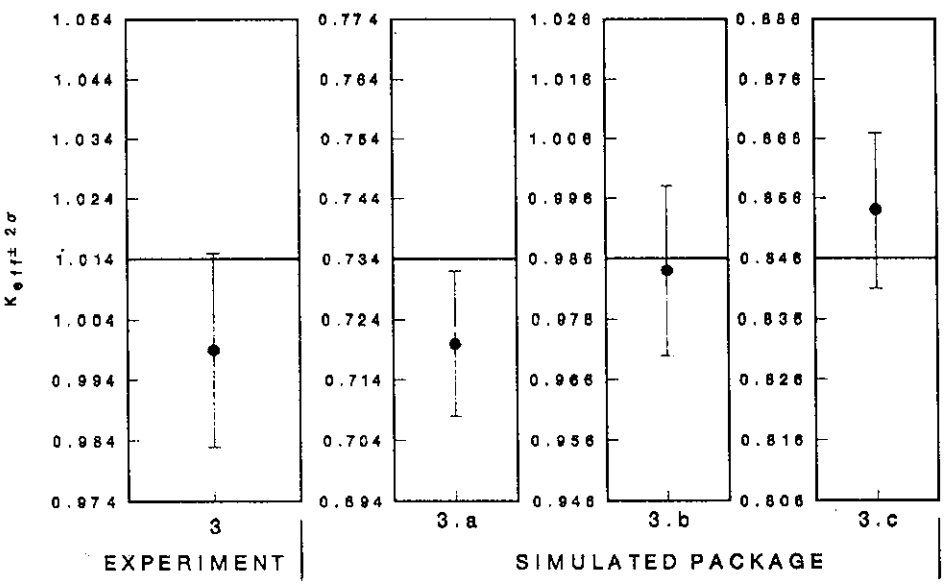
ENEA Results Program MONK UKNDL Cross Sections ORNL-DWG 84C-12737







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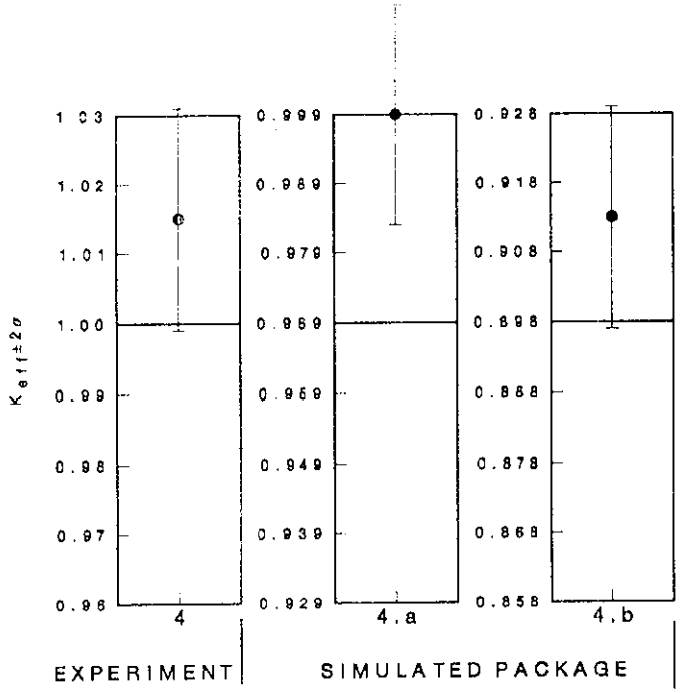
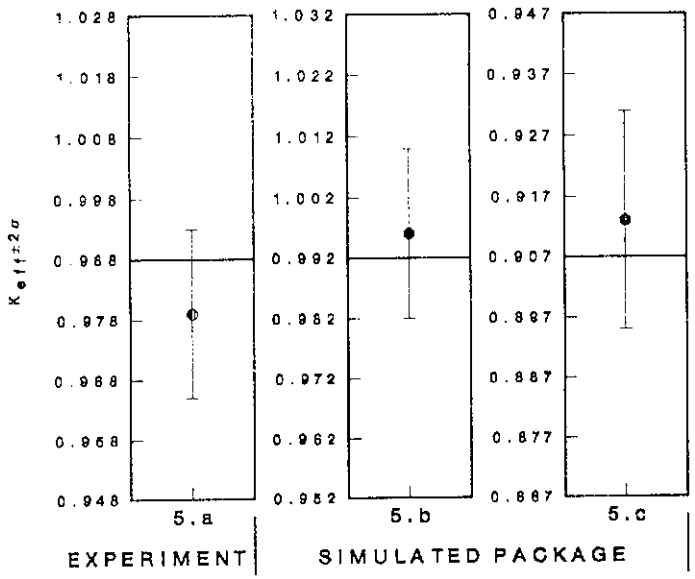
ENEA Results Program MONK UKNDL Cross Sections (continued)

ORNL-DWG 84C-8761

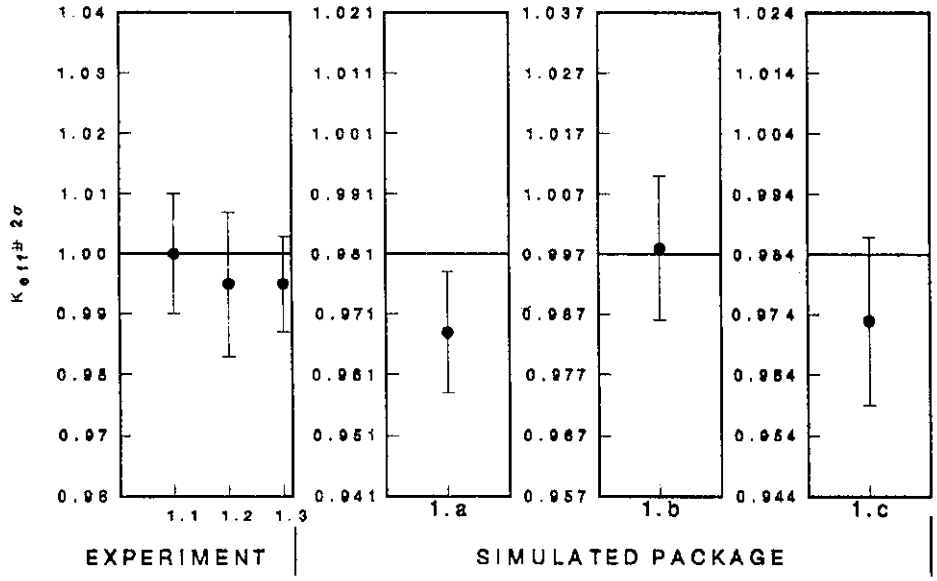


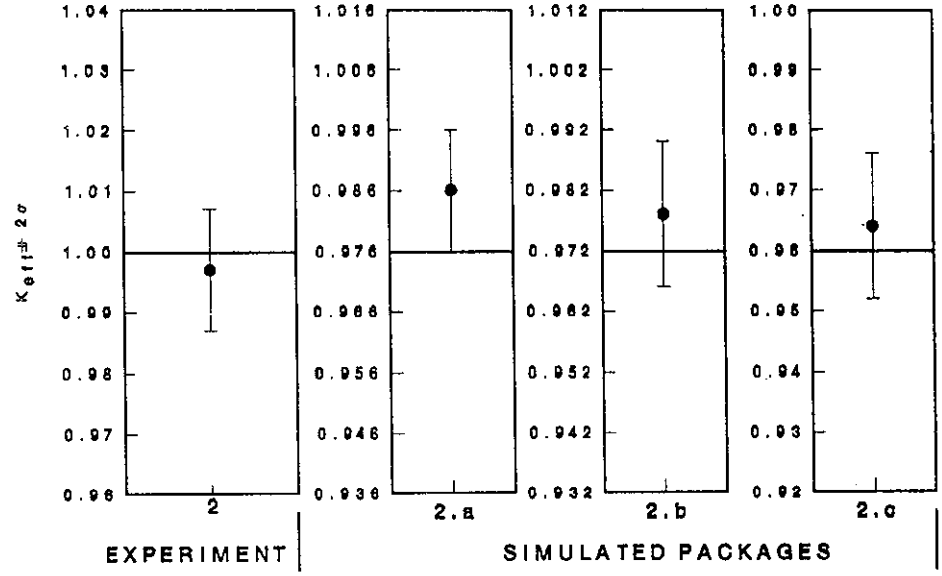


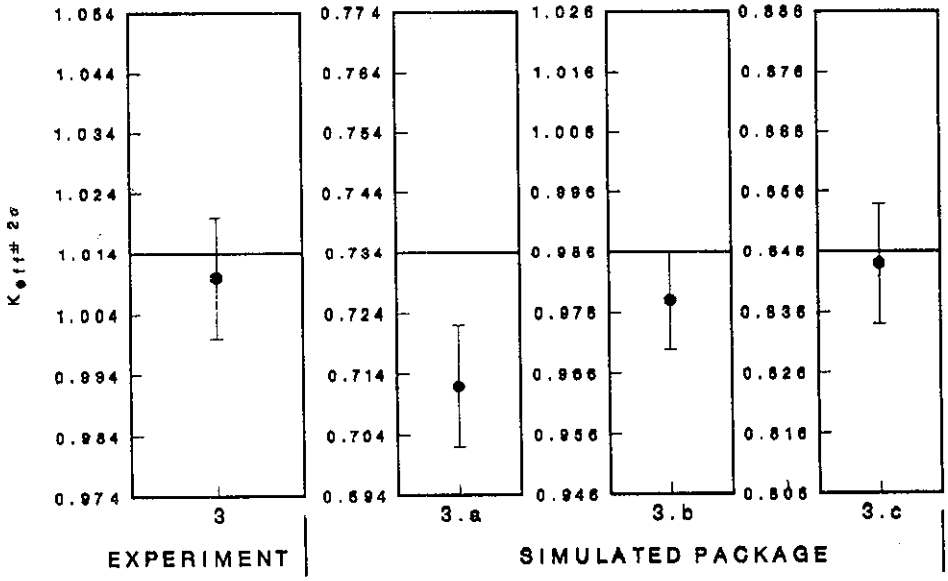
9430059

Sweden Results Program KENO SCALE Cross Sections

ORNL-DWG 84C-12739



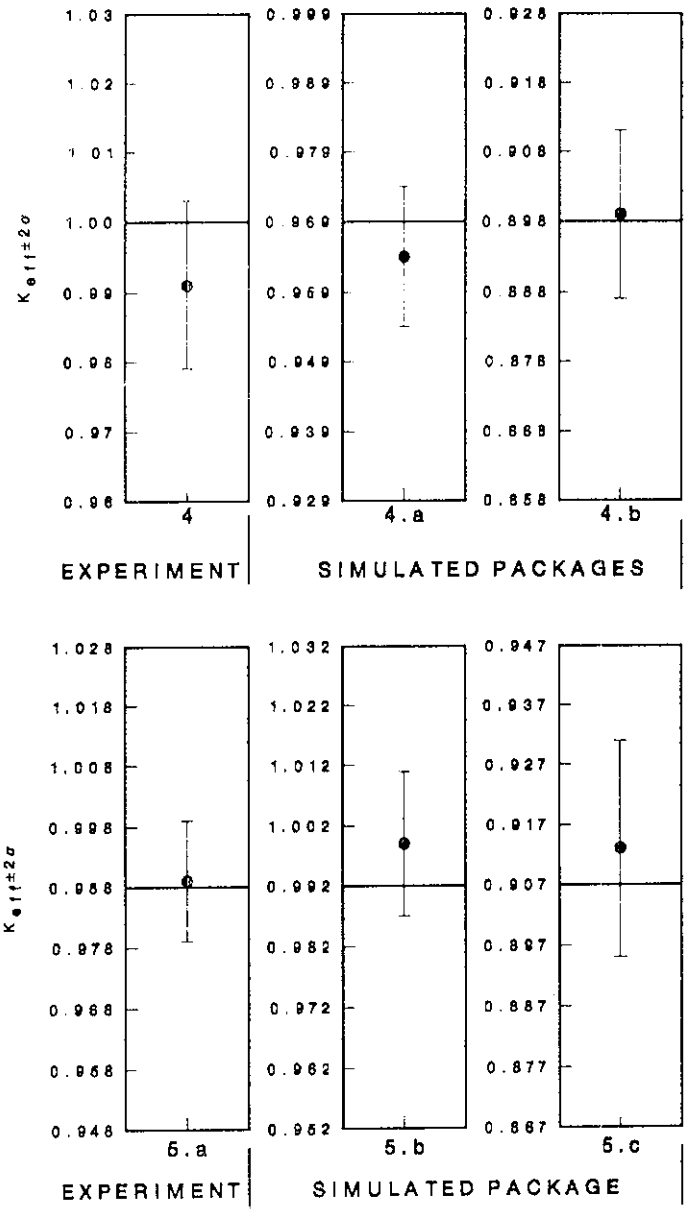




9400060

Sweden Results Program KENO SCALE Cross Sections (continued)

ORNL-DWG 84C-8783



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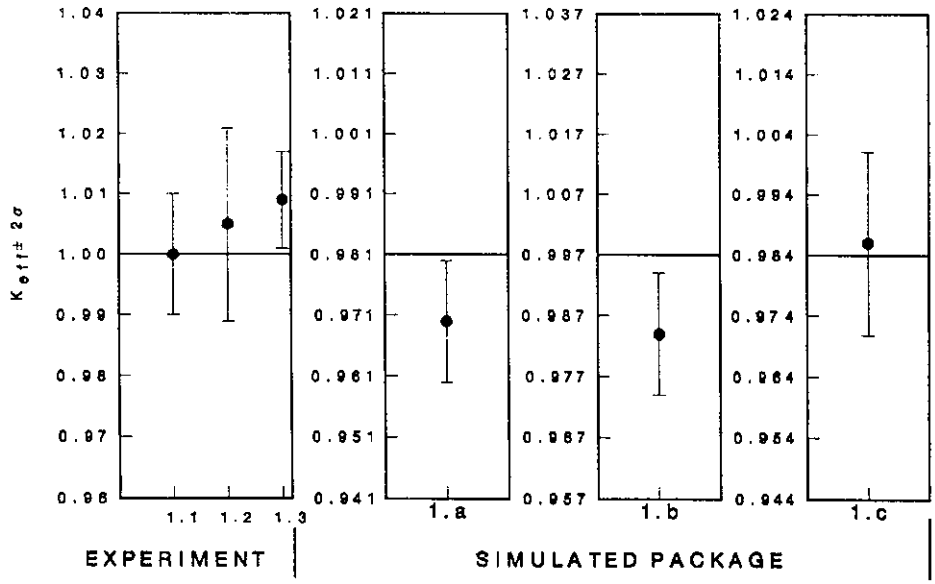
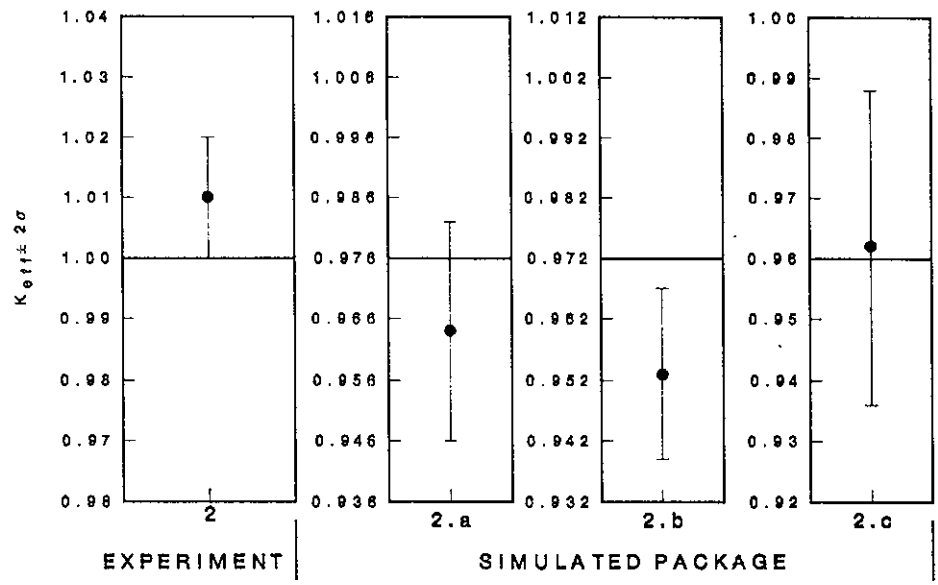
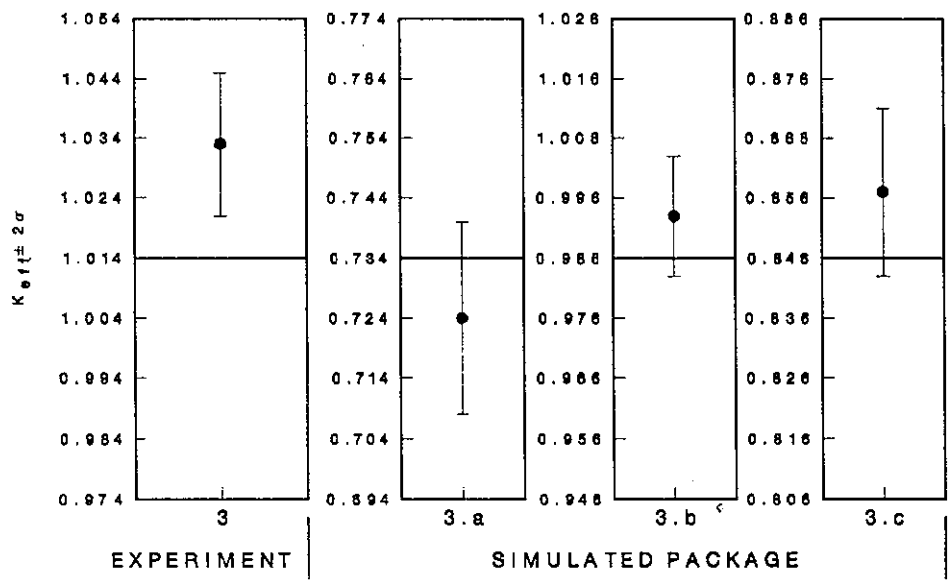
BN Results Program KENO Hansen-Roach Cross Sections ORNL-DWG 84C-12740



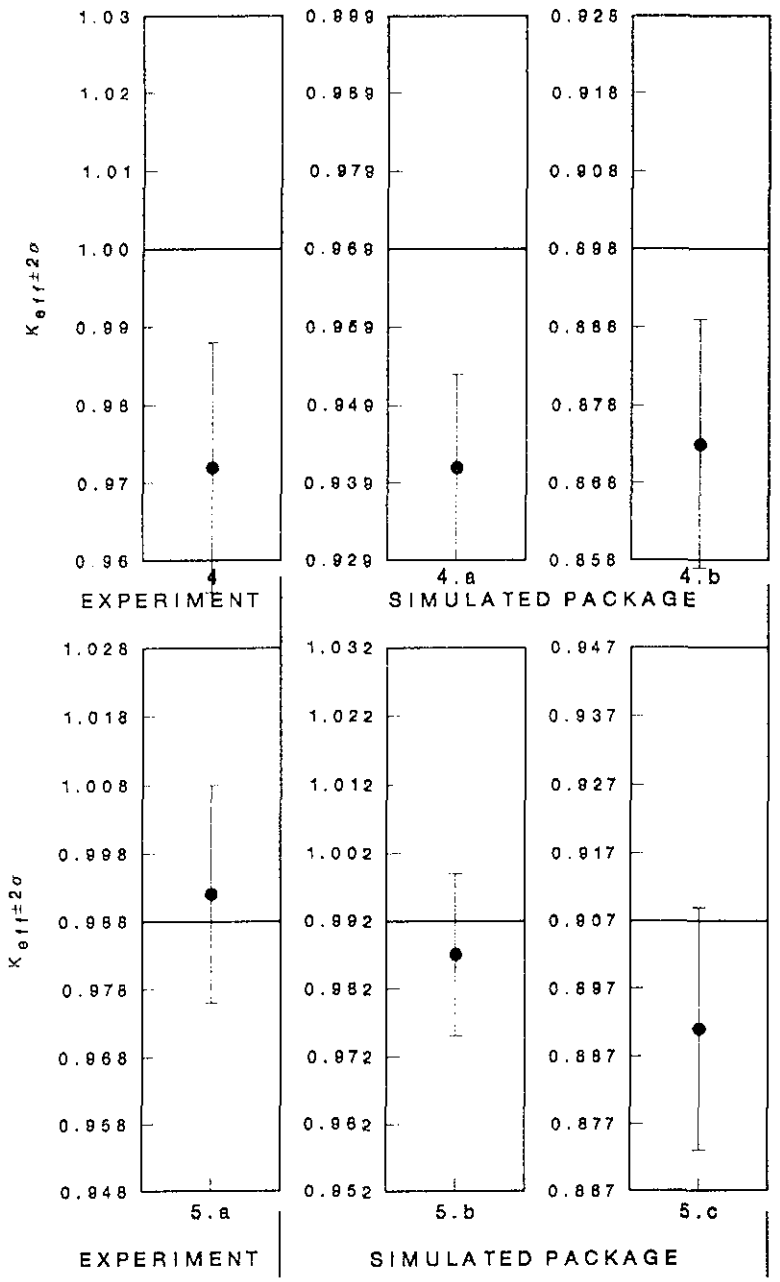




94030062

BN Results Program KENO Hansen-Roach Cross Sections (continued)

ORNL-DWG 84C-8764



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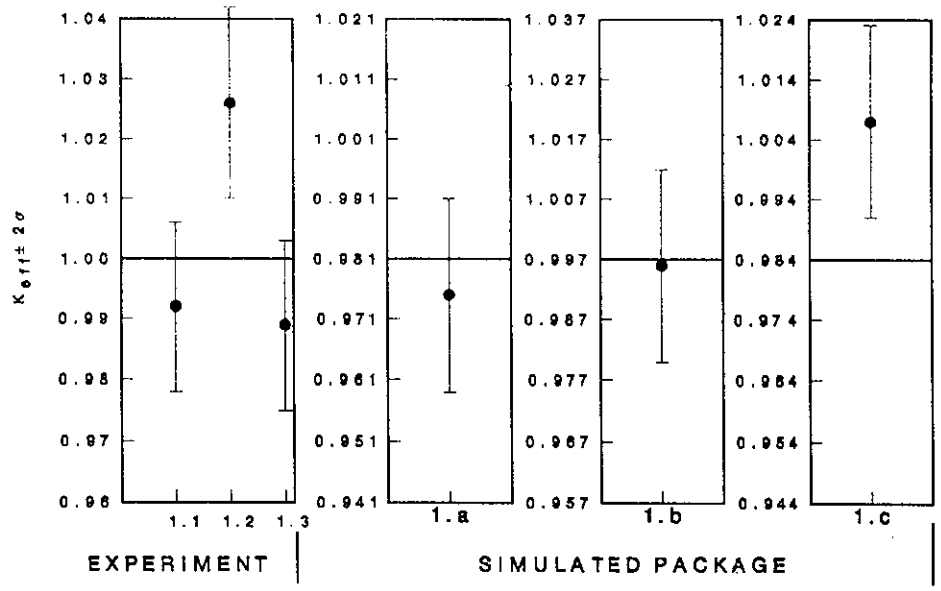
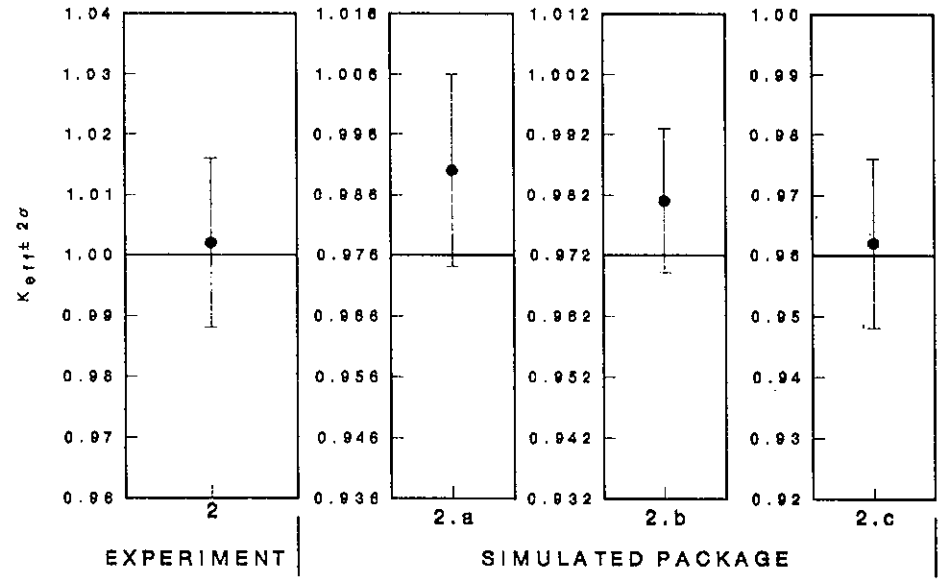
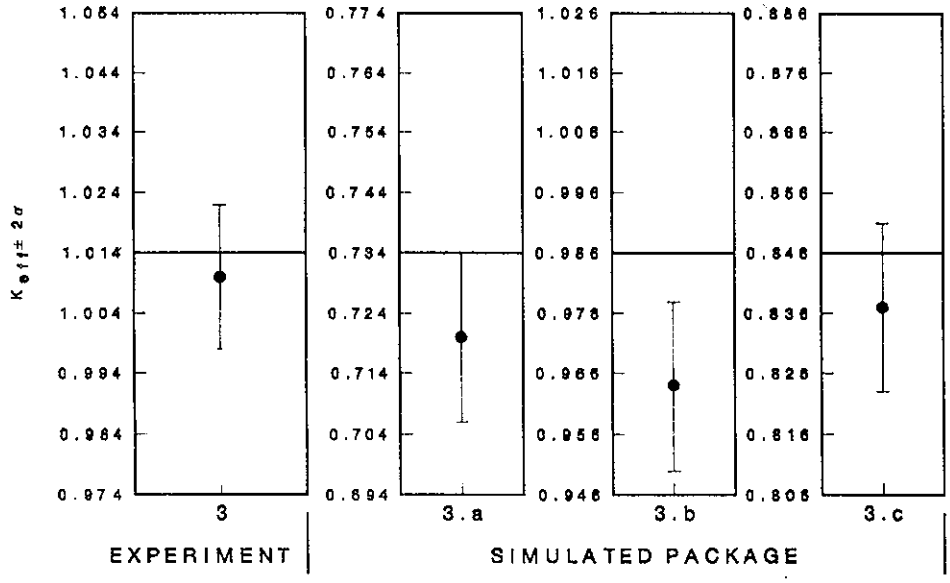
SRD Results Program MONK UKNDL Cross Sections ORNL-DWG 84C-12741







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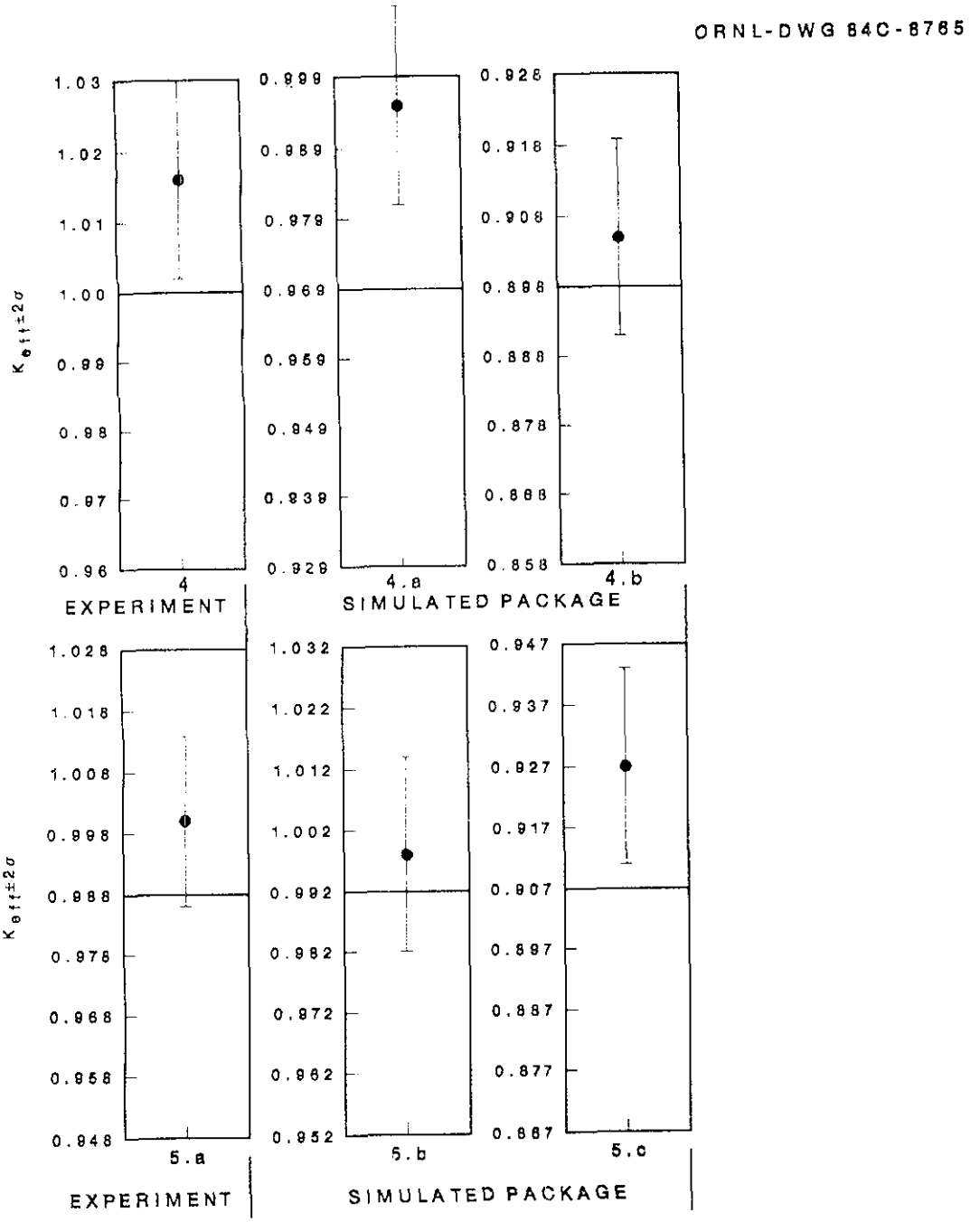
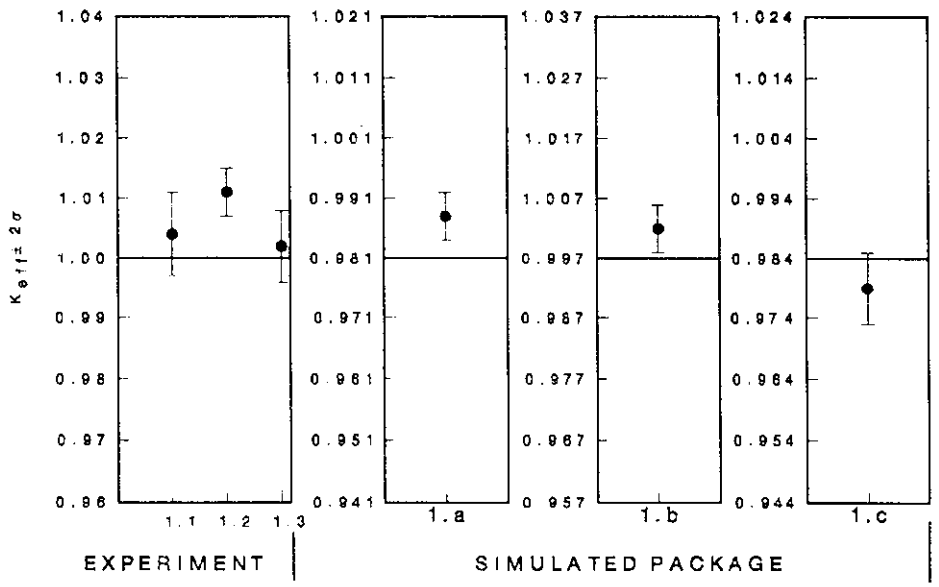
SRD Results Program MONK UKNDL Cross Sections (continued)

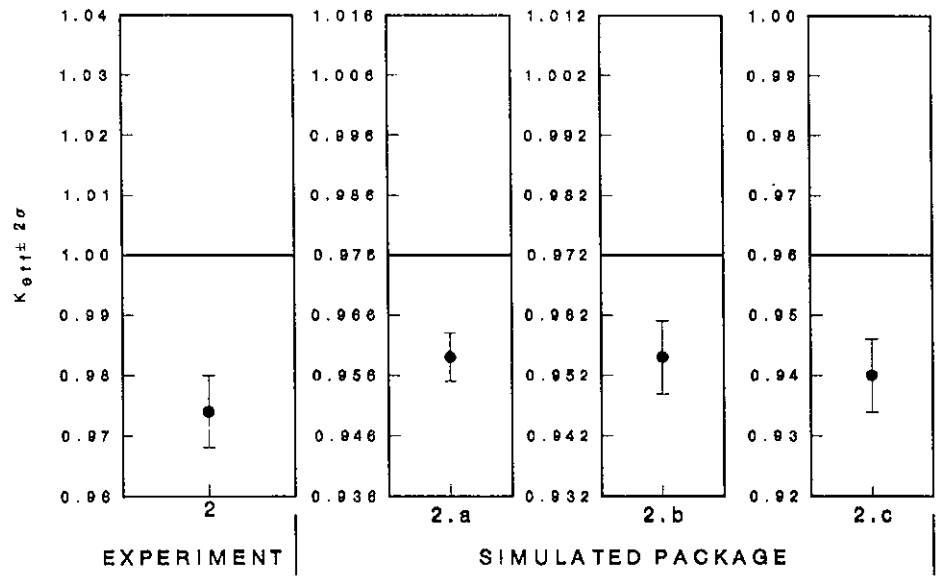


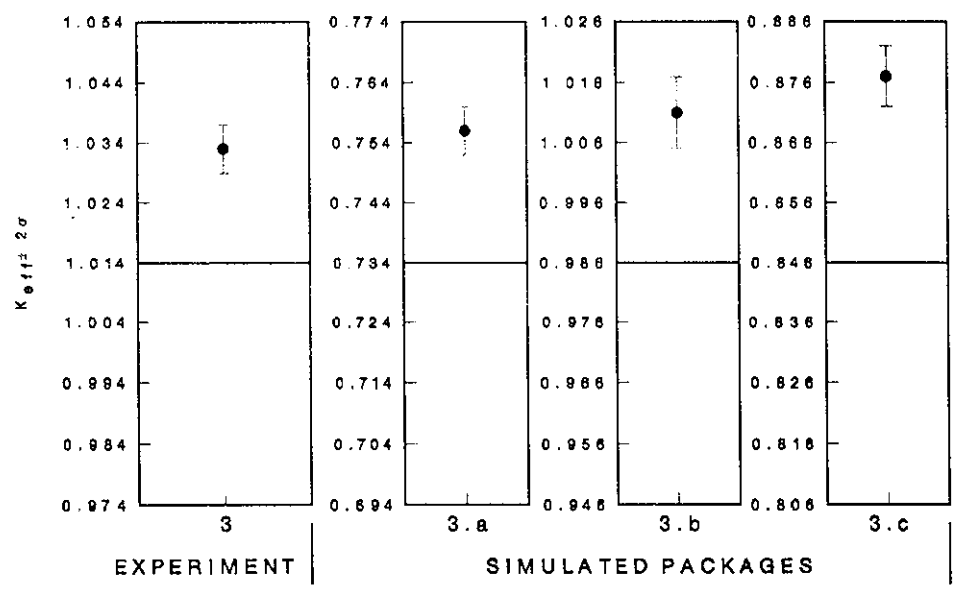
9430065

JAERI Results Programs KENO (.) and MULTIKENO (x) MGCL Cross Sections

ORNL-DWG 84C-12742



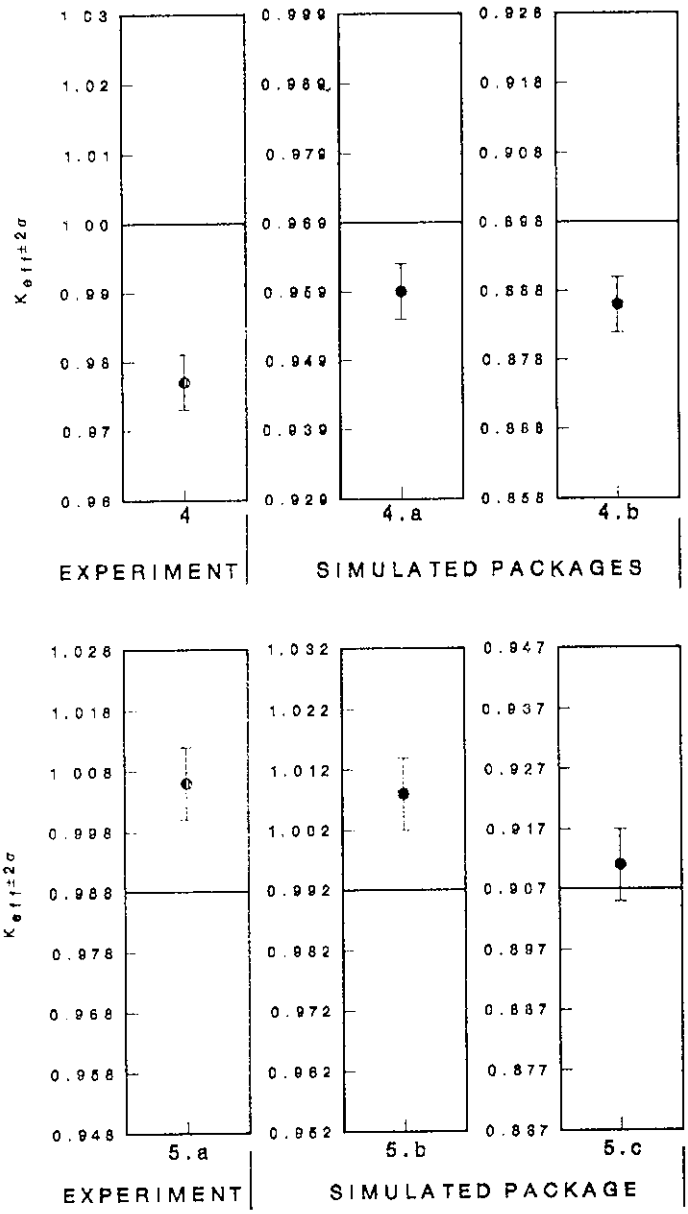




94030066

JAERI Results Programs KENO (.) and MULTIKENO (x) MGCL Cross Sections (continued)

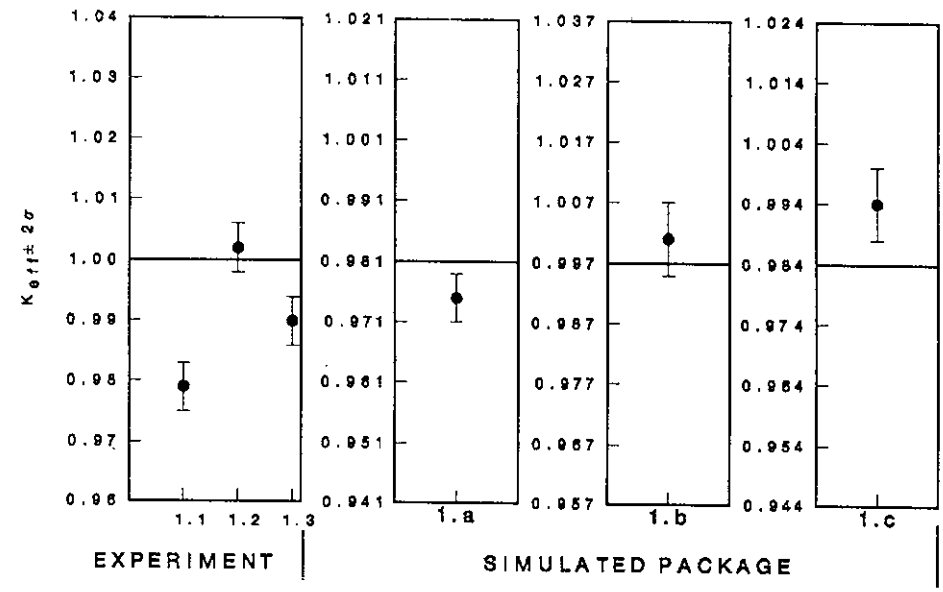
ORNL-DWG 84C-8766

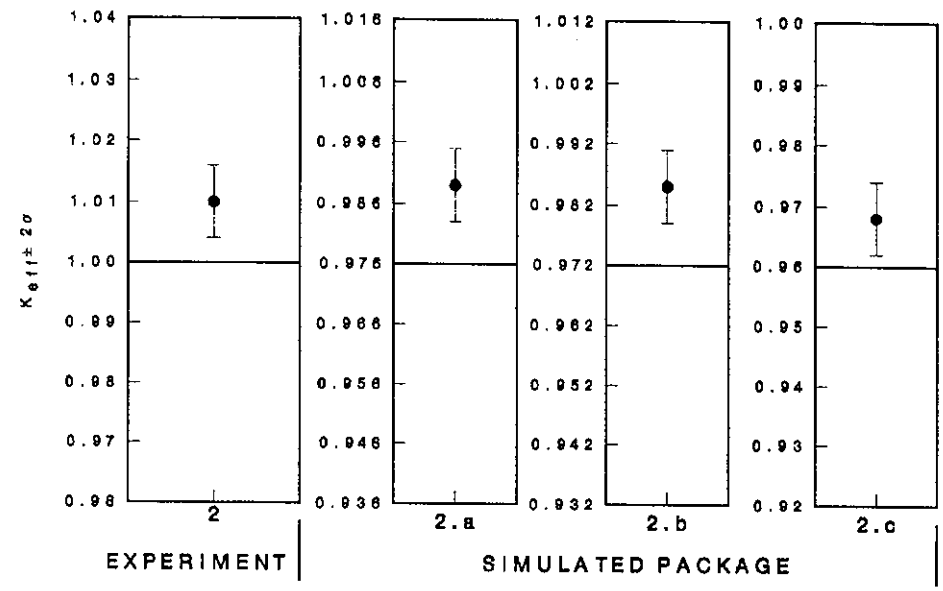


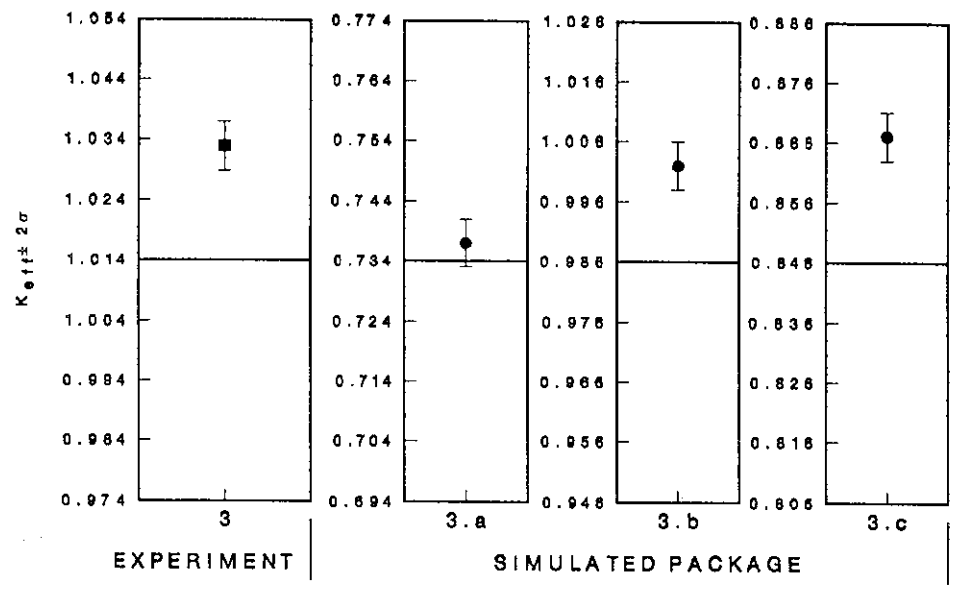
94[illegible]067

JAERI Results Programs KENO Jr (.) and MULTIKENO (x) MGCL Cross Sections

ORNL-DWG 84C-12743



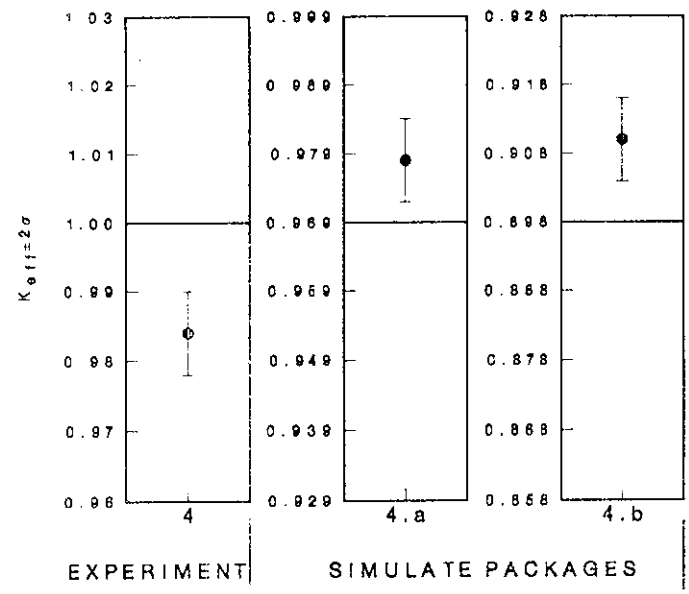


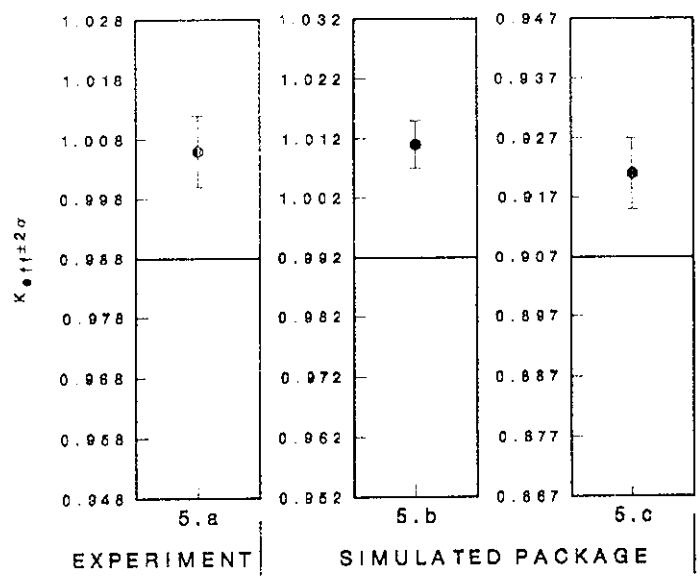


94030068

JAERI Results Programs KENO Jr (.) and MULTIKENO (x) MGCL Cross Sections (continued)

ORNL-DWG 84C-8787

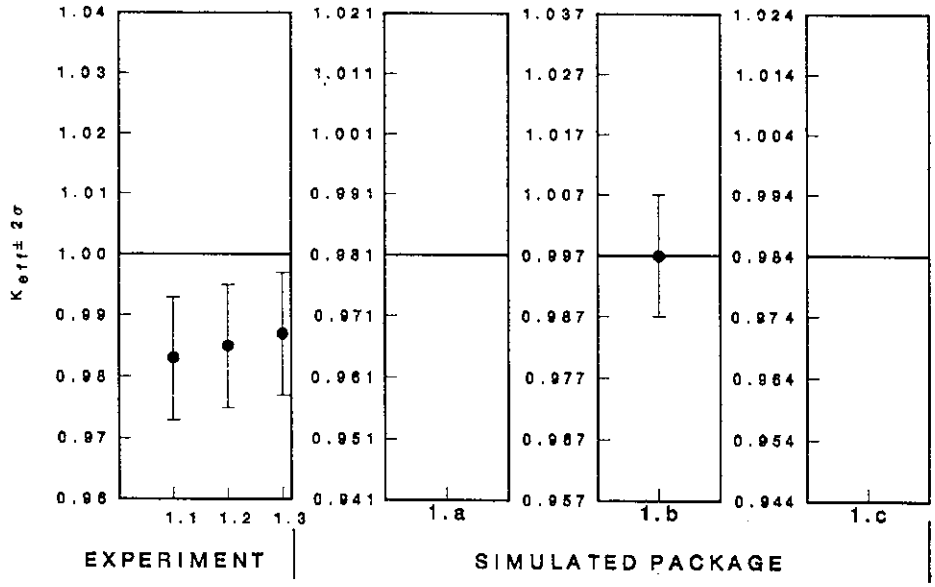


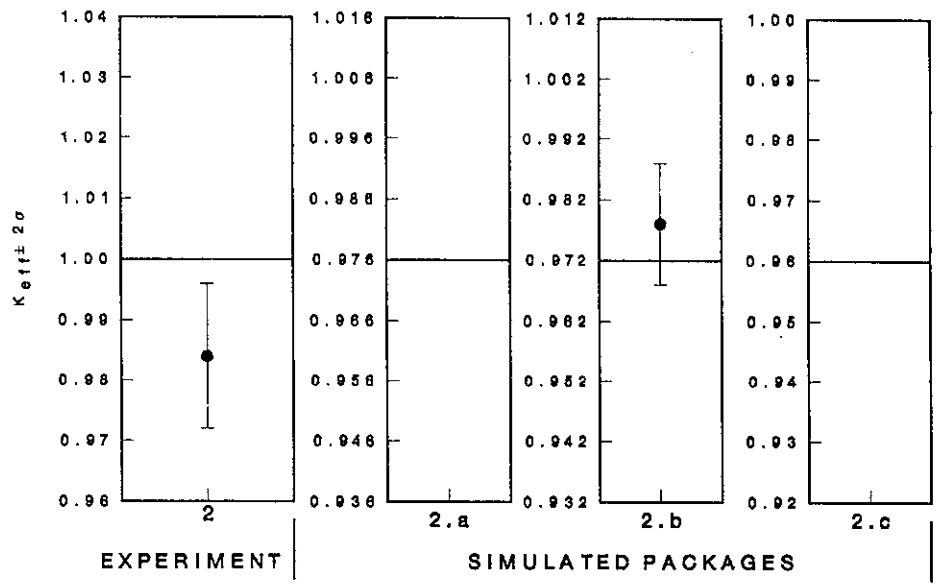


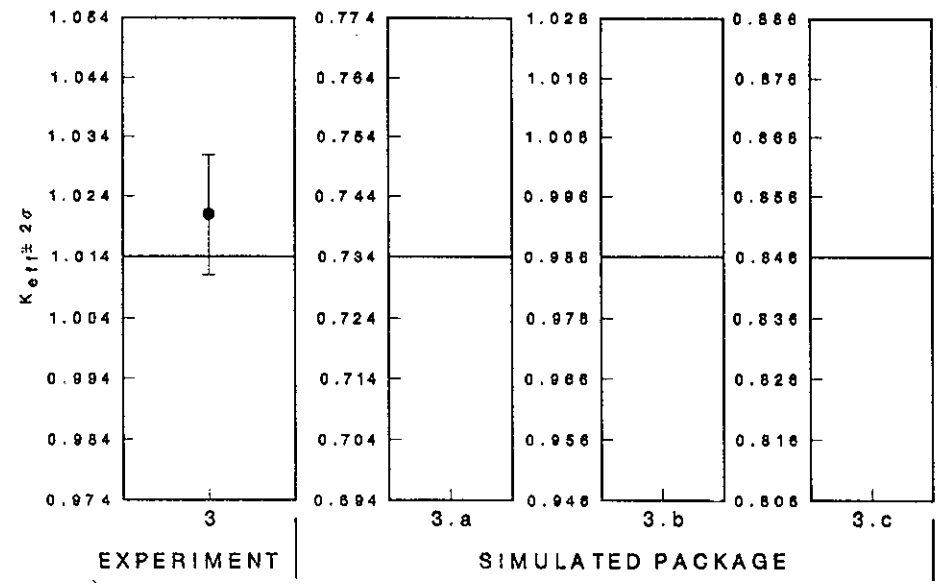
94 [illegible]69

ORNL-DWG 84C-12738

ENEA Results Program KENO GAM-GATHER Cross Sections







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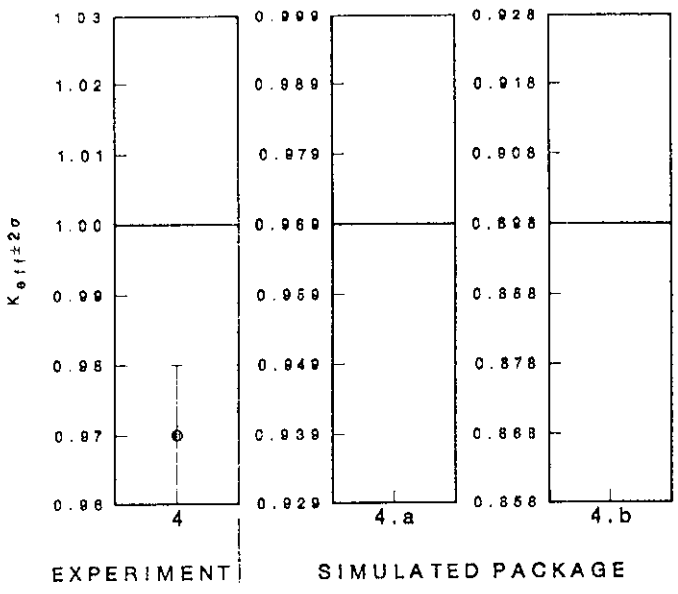
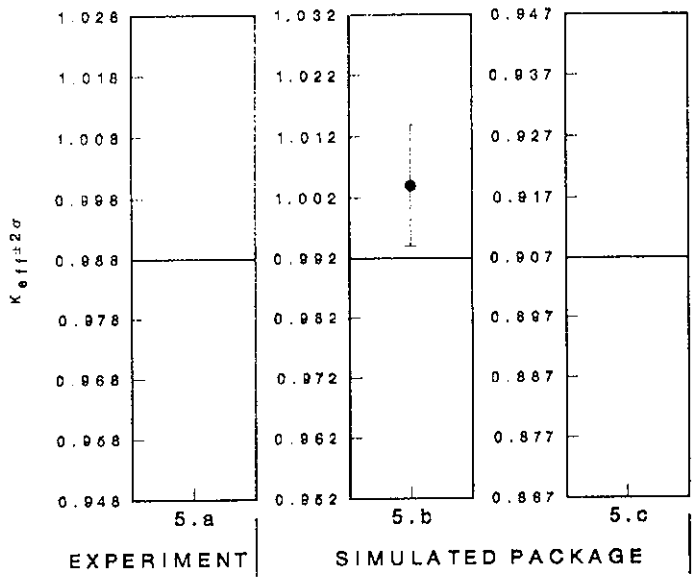
ENEA Results Program KENO GAM-GATHER Cross Sections (continued)

ORNL-DWG 84C-8762





9[illegible]007

Appendix C
Members of the Working Group

94030072

Austria	Dr. G. KAMELANDER Osterreichische Studiengesellschaft fur Atomenergie Vienna
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	Mr. L. MAUBERT Commissariat a l'Energie Atomique Fontenay-aux-Roses
	Mr. G. POULLOT Commissariat a l'Energie Atomique Fontenay-aux-Roses
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	Mr. W. WEBER Gesellschaft fur Reaktorsicherheit Garching
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9423U073

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Switzerland	Mr. J. M. PARATTE Federal Institute for Reactor Research Wurenlingen
United Kingdom	Mr. G. WALKER United Kingdom Atomic Energy Authority Culcheth
United States	Mr. G. E. WHITESIDES (Chairman) Oak Ridge National Laboratory Oak Ridge, Tennessee
OECD Nuclear Energy Agency	Dr. M. E. STEPHENS (Secretary) Nuclear Safety Division

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