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THE APPLICATION OF SHIELDING BENCHMARK EXPERIMENTS TO THE TESTING OF NUCLEAR DATA, CALCULATIONAL METHODS, AND PROCEDURES*

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ABSTRACT

A number of experiments have been conducted in the United States which have served as benchmarks for the testing of nuclear data important in shielding applications and in the validation of shielding calculational methods and procedures. Over the past several years, a cooperative program among U. S. governmental and industrial laboratories has been developed for the purpose of providing a data base for general use in nuclear calculations. This work has been coordinated through the Cross Section Evaluation Working Group (CSEWG) administered by Brookhaven National Laboratory and has resulted in the publication of several versions of an Evaluated Nuclear Data File (ENDF/B). The number of nuclides included in the file, the amount and accuracy of data given for a particular nuclide, and the types of data given for a particular nuclide have been greatly expanded from version to version of ENDF/B. To give an example of particular interest to the shielding community, the most recent version will contain 37 nuclides for which complete gamma-ray production data will be provided. The testin and validation of the ENDF/B data is done within the CSEWG by calculating a selected group of benchmark experiments, each of which provides insight to a particular aspect of the data. Although these benchmarks are specifically selected for data testing, they also serve the useful purpose of validating the data processing codes and the calculational design codes. Existing shielding experiments are continually being identified as being suitable for benchmarks, and new experiments are being planned specifically for data and calculational code testing. In this regard, the participating laboratories have worked in close cooperation with the United States Atomic Energy Commission, the Defense Nuclear Agency, and other governmental agencies.

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*Work performed under the auspices of the United States Atomic Energy Commission

I. INTRODUCTION

Effective design of shields for fission and fusion reactors requires a reliable data base of neutron and photon cross sections and proven methods of calculational modeling and computation. Independent validation of these two requirements can seldom be achieved, and the normal procedure is to calculate a model of an actual experiment with a particular data base and code. Experiments which have been selected for this sort of application have come to be termed "benchmarks."

E. A. Straker[1] has identified two general categories of problems that have been accepted as benchmarks; namely, integral experiments and reference calculations. Integral experiments are oriented toward nuclear data testing, such as gamma ray production cross sections, or checking a specific feature of a calculational code, such as a geometry routine in a transport code. Reference calculations, on the other hand, are used to verify that the physics of a calculational code is properly coded. Rather comprehensive lists of both types of benchmarks are given in Ref. 1, but in this report emphasis will be placed on benchmarks that can be used for the testing of nuclear data important in shielding applications.

The data base which is being developed as a cooperative effort among governmental and industrial laboratories in the United States is the Evaluated Nuclear Data File (ENDF/B). This work is coordinated through the Cross Section Evaluation Working Group (CSEWG) which is administered by the National Neutron Cross Section Center (NNCSC) at Brookhaven National Laboratory. A parallel data file in the ENDF/B format, but containing data for nuclides of particular interest to the U. S. Department of Defense, is administered by the Radiation Shielding Information Center (RSIC) at Oak Ridge National Laboratory. The latest version of ENDF/B now in preparation will be particularly applicable to shielding problems in that it will contain a) neutron-induced gamma-ray production for 37 nuclides; b) decay data for 825 fission products; and c) complete reevaluations of the photon interaction cross sections for 87 elements. A list of the nuclides containing gamma-ray production data is given in Table I.

II. SHIELDING DATA TESTING BENCHMARKS

ENDF/B data are constantly being checked through a comprehensive data testing program carried out by CSEWG. In Phase I data testing, the evaluations are checked for accuracy, completeness, and use of all experimental and theoretical information. In Phase II testing, the evaluated data are processed and used in calculating certain integral experiments which have been specifically selected as data testing benchmarks. Two types of experiments currently being used in the shielding data testing program are shown in Table II. In the first type, experiments are chosen on a single material basis and calculations are expected to reveal specific data deficiencies. The second type of experiments used in Phase II shielding data testing consists of multi-material shield mock-ups. Calculations of these experiments not only indicate deficiencies in the evaluated nuclear data, but also provide tests of the entire shielding code system.

Of the benchmarks included in the "single material" type, the "broomstick" experiments performed at Oak Ridge National Laboratory are the simplest to calculate. Basically, these transmission experiments were designed to test total cross section data for several given samples over the MeV region. In each experiment (see Fig. 1) the neutron source was the Oak Ridge Tower Shielding Reactor II (TSR). The detector was a nominal 5-cm by 5 cm NE-213 scintillator and samples were placed midway between source and detector. Care was taken to collimate the neutron beam and reduce air scattering by proper shielding of all elements of the experiment.

The calculations of the various broomstick experiments* consist of folding the detector resolution function with a transmitted uncollided spectrum derived from the total cross section of the nuclide under investigation. As an example of the application of the broomstick benchmarks, results of calculations using the ENDF/B Version III Fe (MAT 1180) are shown in Fig. 2 for two sample thicknesses. Discrepancies between calculation and experiment as well as new measurements on the Oak Ridge Electron Linear Accelerator have led to rather significant changes in the iron evaluation for subsequent versions of ENDF/B.

Another set of "single material" benchmarks, namely the Lawrence Livermore Laboratory "pulsed sphere" experiments listed as SDT10 in Table II, has been calculated by several laboratories and has served as a cross-check on codes and methods in addition to data testing. These experiments have been calculated with continuous energy Monte Carlo, multigroup Monte Carlo, and S_n methods. The experimental set up is shown in Fig. 3. Neutrons emitted from pulsed spheres were measured as a function of time, usually at two angles. Measurements were made of the energy spectra leaking through the surface of the sphere with a "nominal" 14-MeV pulsed neutron source at the center. Sphere thicknesses of 1/2 to 3 mfp were employed. The measured neutron time spectra are sensitive, not only to the magnitude of the elastic and inelastic neutron cross sections, but also to the shapes of the secondary energy and angular distributions for incident energeries near 14 MeV. Materials studied include ^{6}Li, ^{7}Li, Be, C, N, O, Mg, Al, Ti, Fe, H_2O, D_2O, CH_2, and CF_2. Examples of the application of the SDT10 benchmark are shown in Figs 4 and 5. In Fig. 4, a calculation performed by LLL[27] using the TART code[28] is compared with the pulsed sphere experimental results for nitrogen, and Fig. 5 compares these same results with calculations[29] performed using the Los Alamos code ANDY[30] and the Oak Ridge National Laboratory code O6R (unpublished). All calculations show good agreement for this ENDF/B nitrogen evaluation[31] (MAT 1133). For other materials, particularly heavy nuclides, these experiments have revealed a deficiency in the production of high energy secondary neutrons.

Two benchmarks in Table II, namely SDT6 and SDT7, were specifically selected for checking gamma-ray production cross sections for single materials. The experimental arrangement for the two experiments was essentially identical and is shown in Fig. 6. The only difference was that 15-cm of lithium hydride, not shown in the figure, was placed in the detector beam between the slab and the detector collimator in SDT7. The thermal and fast incident neutron spectra were measured for each experiment, and the resulting photons were detected with a 12-cm x 12-cm NaI(Tl) detector. Calculations of these experiments can be achieved with any gamma-ray production multigrouping code such as AMPX,[32] MINX,[33] or LAPHANO.[34] Table III illustrates calculations vs experiment for

*A code "BROMSTK" for calculating the broomstick experiments with ENDF/B data is available upon request from Oak Ridge Radiation Shielding Information Center.

SDT6 and Table IV shows similar results for SDT7. Note that for all materials given as examples in these tables, only the calculations for iron (MAT 1180) lie outside experimental error.

The calculations for multi-materials benchmarks are different from the single material type in that they can be of a "coupled" type; that is, the gamma heating calculations are dependent upon prior neutron flux calculations. Also, the cross section processing is much more extensive as such effects of resonance interference, broadening, and energy and spatial self-shielding must be taken into account. Finally, multi-dimensional models are often needed to properly calculate leakage from region to region. There are advantages, however, in that much of the nuclear data for the various nuclides in the experiments is exercised, including the elemental photon interaction cross sections needed for the calculation of gamma-ray transport. Also, these benchmarks provide a stringent test of calculational techniques.

SDT8 is the ZPPR/FTR-2 shield experiment performed by Argonne National Laboratory[15] and Hanford Engineering Development Laboratory.[14] In that experiment, the Zero Power Plutonium Reactor (ZPPR) at the National Reactor Testing Station, Idaho Falls, Idaho, was loaded to approximate the basic features of the Fast Test Reactor (FTR): a 1000-liter, two-zone core surrounded by a one-foot thick nickel reflector and a sodium-steel shield. A cross-sectional view is shown in Fig. 7. Various neutron reaction rates were measured, including $^{10}B(n,\alpha)$, $^{238}U(n,f)$, $^{239}Pu(n,f)$, and $Na(n,\gamma)$. Gamma-ray heating was studied using LiF thermoluminescent dosimeters (TLD), which were read and analyzed at Hanford Engineering Development Laboratory.

A seven-zone, two-dimensional (r-z, cylindrical) model, that has been used for the calculation of SDT8[12] and in a previous calculational analysis,[13,14] is shown in Fig. 8. If a two-dimensional multigroup diffusion code such as TWODS,[35] or S_n code such as DOT[36] or TWOTRAN,[37] or a multigroup Monte Carlo code such as ANDY is used for the calculation, multigroup data processing can be achieved with such codes as MC-2,[38] ETOX-IDX,[39] ENDRUN-TDOWN,[40] ETOG-GAM,[41] MINX, and AMPX for the neutron cross sections, LAPHANO, MINX, and AMPX for the gamma-ray production cross sections, and GAMLEG,[42] MINX, and AMPX for the photon interaction cross sections. If a continuous energy Monte Carlo code such as MCN-MCG[43] is used in the analysis, the basic ENDF/B data can be processed by a code such as ETOPL.[44] Sample calculations of the neutron part of the experiment[45] using multigroup diffusion and S_n theories are shown in Fig. 9 and those for the photon part of the experiment[46] using S_n theory shown in Fig. 10. Note in Fig. 9 that the agreement of the calculation with experiment is good except in the deep shield region. Improvement here might require more detailed treatment of the cross section data, including higher order scattering, and/or a more complex geometrical model of the experiment. As can be seen in Fig. 10, the poor calculation of the neutron flux is also evident in the gamma-ray production calculation in the shield. The discrepancy shown in the depleted zone in Fig. 10, however, is due to an incorrect estimate of the resonance self-shielding of the gamma-ray production cross sections in the region and has been completely accounted for in a subsequent calculation.

III. EXTENSION OF THE SHIELDING BENCHMARK PROGRAM

Although great progress has been made in the CSEWG data testing benchmark program in the last few years, the list of experiments shown in Table II is by no means exhaustive, and plans are to increase the scope of the program to provide better testing of both the more extensive nuclear data to be included in later versions of ENDF/B and the more sophisticated calculational approaches being developed in the United States and in the rest of the world. This objective is being achieved by the specification of new benchmarks from among the many existing shielding experiments and by the design of new experiments for particular use as shielding benchmarks. In the first category are additional "broomstick" experiments for stainless steel, calcium, nickel, chromium, carbon, uranium-238, oxygen, and neon that need to be activated as single material benchmarks. Also, there are the sphere experiments for nitrogen, carbon, aluminum, and steel performed at Intelcom Rad Tech[47] in which secondary neutron and gamma-ray spectra were measured at several angles. The experimental geometry for this experiment is shown in Fig. 11. A high-power Ta/Al/Be Linac target was used to produce intense 50-nsec (FWHM) photoneutron pulses. The small test samples with diameters on the order of a neutron mean-free-path were located at the detector end of the neutron flight path. The 5-cm x 5-cm NE-213 detector used to detect secondary gamma-rays and scattered neutrons as a function of time (hence incident neutron energy) at several angles is shown in the figure. A sample of the data from one of these experiments is shown in Fig. 12. As can be noted from Table II, there are no single material benchmarks for testing photon interaction data. There are several United States experiments[49,50] that are currently being examined for suitability of application in this area.

Several multi-material experiments are being considered for selection as CSEWG benchmarks. One example is an experiment[48] conducted on the engineering mock-up of the FFTF. This experiment was performed at the ZPPR (ZPPR-3) and is similar to the ZPPR/FTR-2 experiment but much more extensive in scope. A three-dimensional model is needed to adequately describe the experiment so calculations would either have to be done with three-dimensional diffusion theory or Monte Carlo or "model reduction" correction factors would need to be provided for calculations in less than three dimensions. Also, some experiments recently carried out at the TSR may qualify as benchmarks. These experiments include the measurement of gamma-ray production in an inconel slab, the FTR first-fission measurement in stored fuel positions, and the control rod channel streaming experiment.

We now turn to the subject of new integral experiments. Several new experiments are planned in the next few years, primarily to test shielding methods. The distinction between data testing and methods testing benchmarks is rarely clearcut. However, in planning new data testing experiments, one attempts to use only one, or at most a few, materials, and to minimize the calculational approximations. In methods testing benchmarks, one tries to use materials with well-known cross sections in relatively simple configurations. In both types of experiment, one seeks to use simple, well-defined sources in order to minimize the effects of source uncertainty. Planned experiments which address important problem areas in FBR shielding design include measurements of neutron deep penetration, secondary gamma-ray production, gamma-ray heating, pin streaming, direct streaming, and cavity streaming.

IV. SUMMARY

Considerable progress has been made in the United States in the last few years in identifying specific experiments for use as shielding benchmarks. These benchmarks have been successfully applied not only in the testing of nuclear data used in shielding calculations, but also in validation of data processing, modeling, and calculational techniques. This program is being expanded not only to include more of the existing shielding experiments, but also to include new experiments specifically designed for application as shielding benchmarks.

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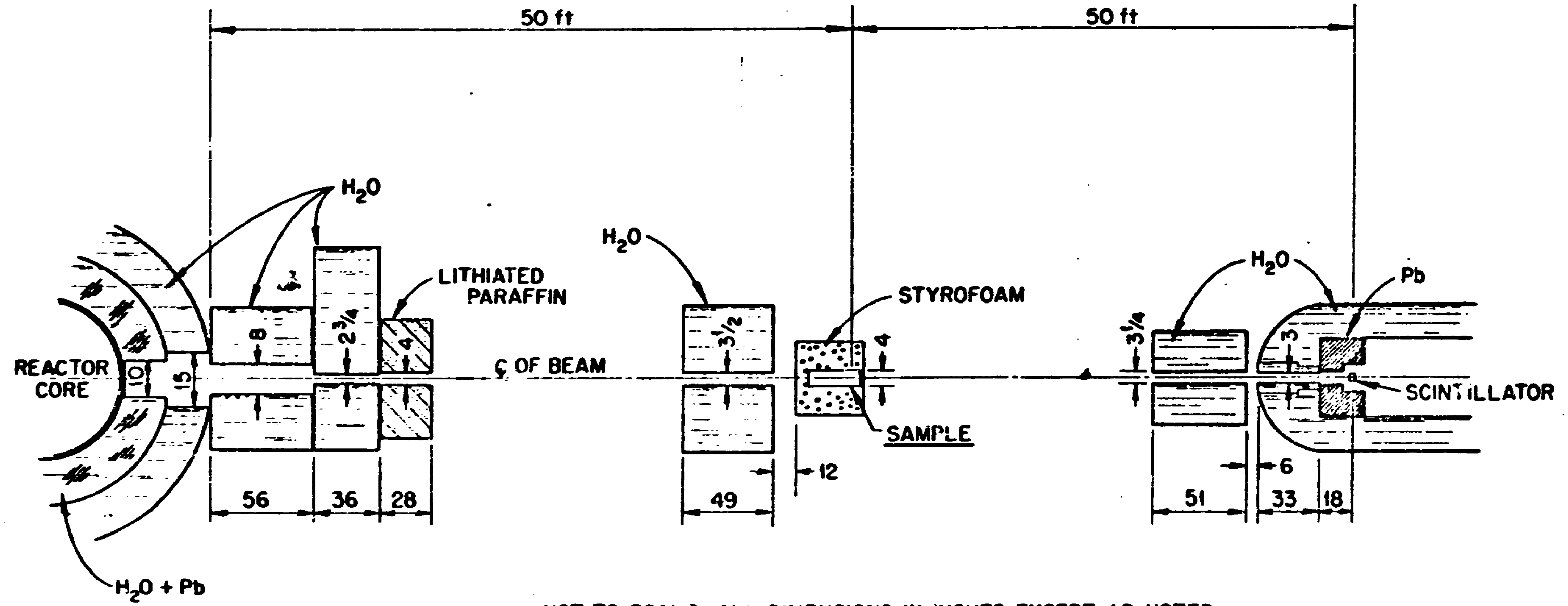

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Fig. 1. Schematic of Experimental Arrangement at the Tower Shielding Reactor.

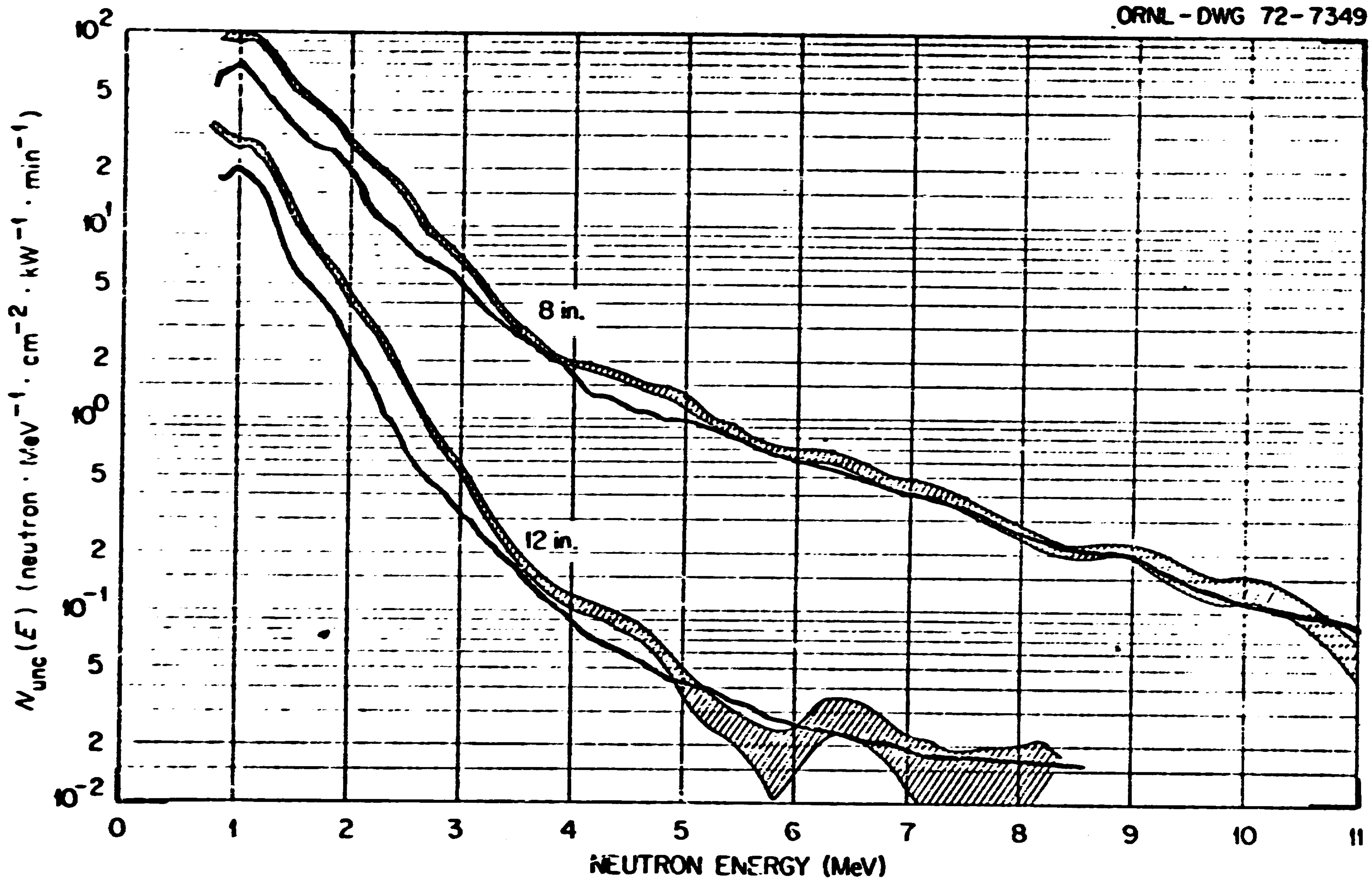

Fig. 2. Transmitted Spectrum Through Iron.

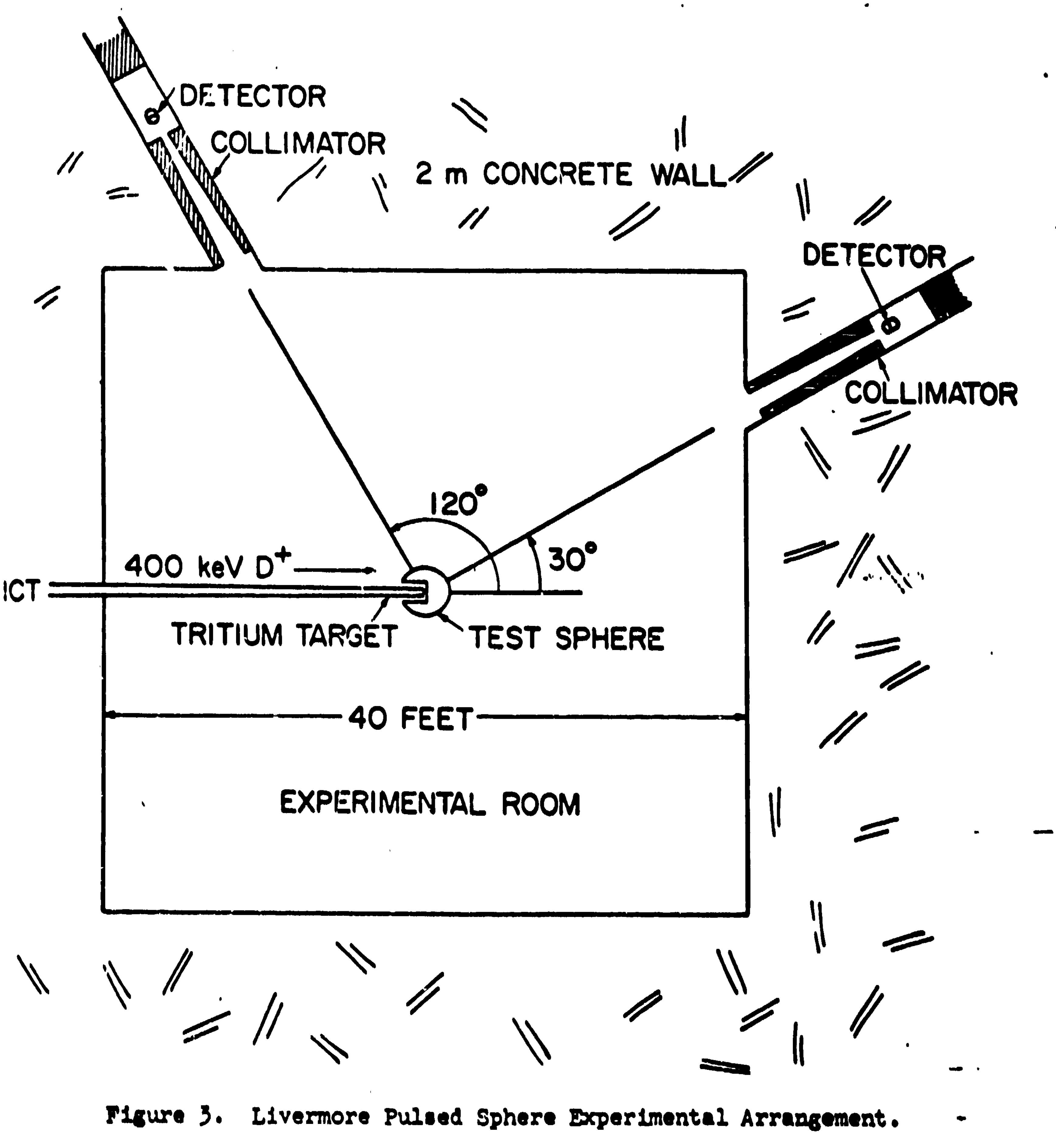

Figure 3. Livermore Pulsed Sphere Experimental Arrangement.

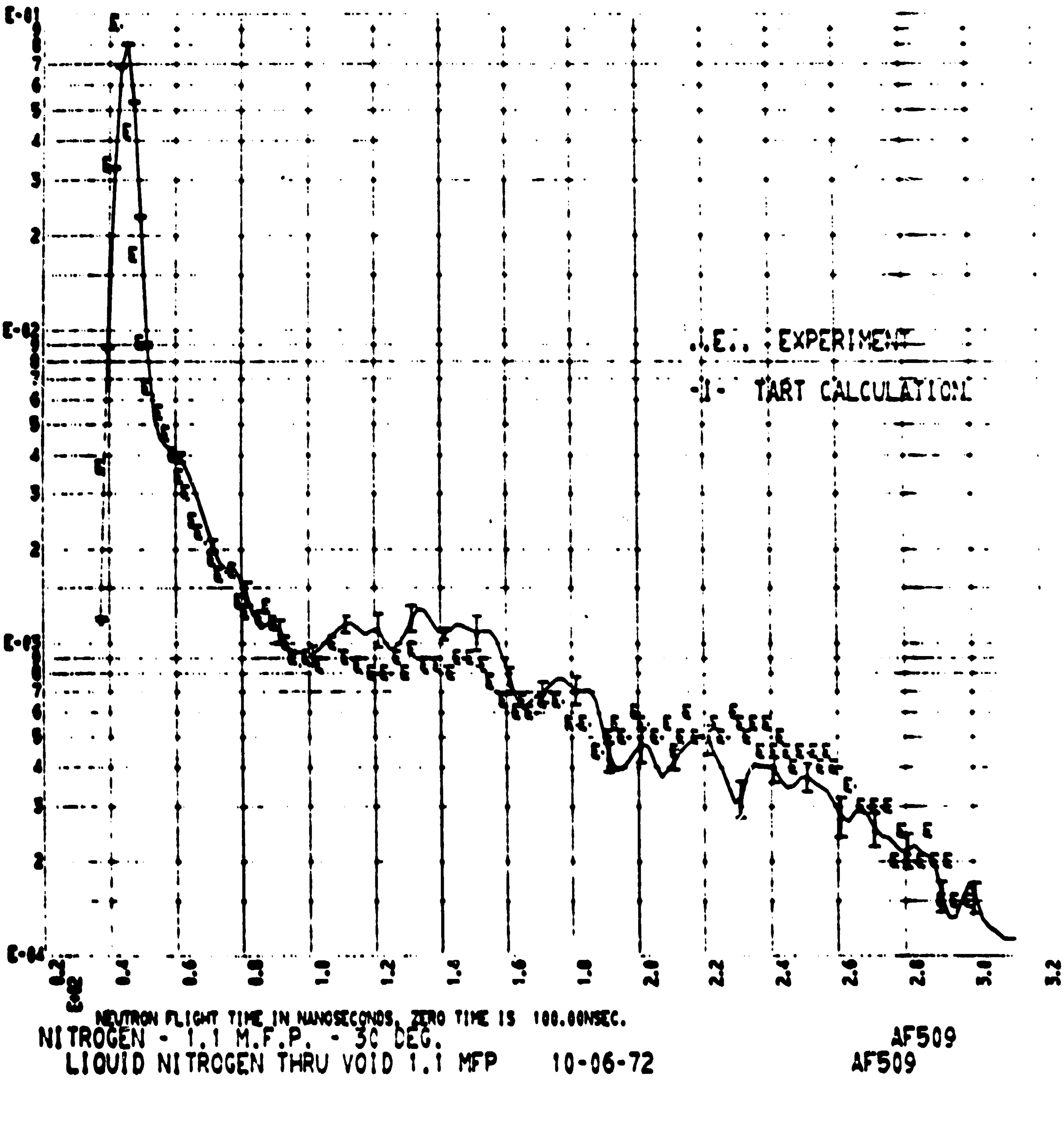

Fig. 4. Nitrogen pulsed sphere results (1.1 M.F.P.)

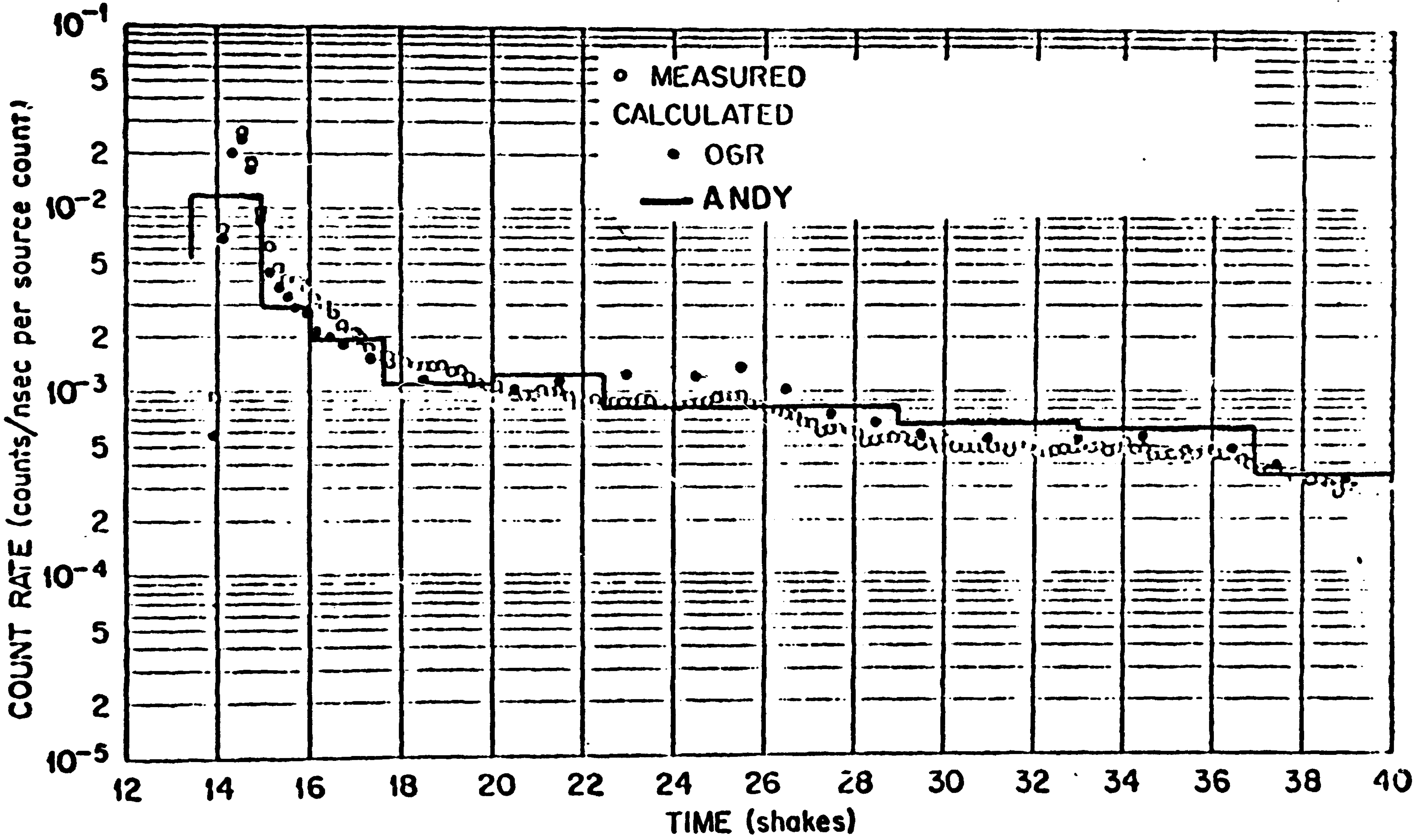

Figure 5. Comparison of LLL Pulsed Sphere (2.40 cm ID, 111.8 cm OD) Experiment with O6R Continuous Energy Monte Carlo and ANDY 30 Group (10 Groups Above 1 1/2 MeV) P_0 Transport Corrected Calculations, Both Based on Young-Foster Cross Section Evaluations.

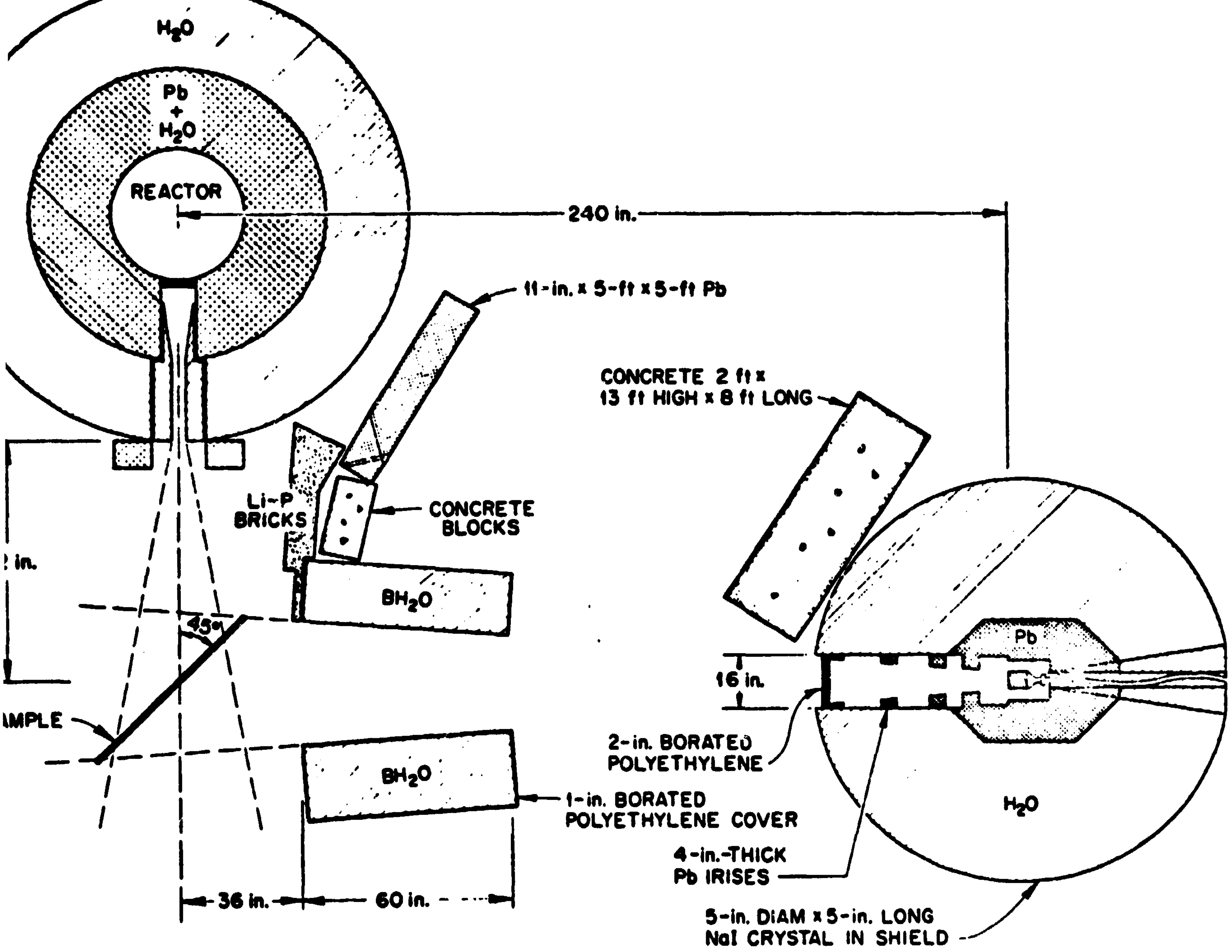

Fig. 6. Schematic Diagram of Geometry for Thermal-Neutron Capture Gamma-Ray Experiments.

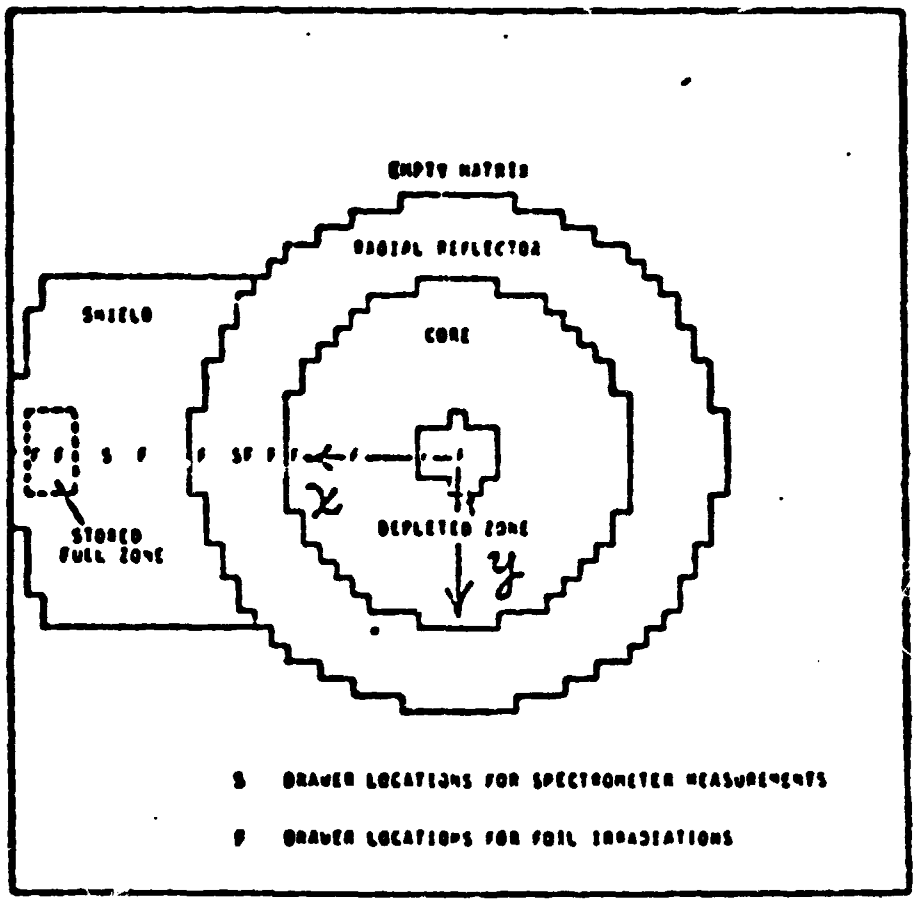

Fig. [illegible]. x-y Cross Sections of ZPPR/FTR-2 (Ref. 2)

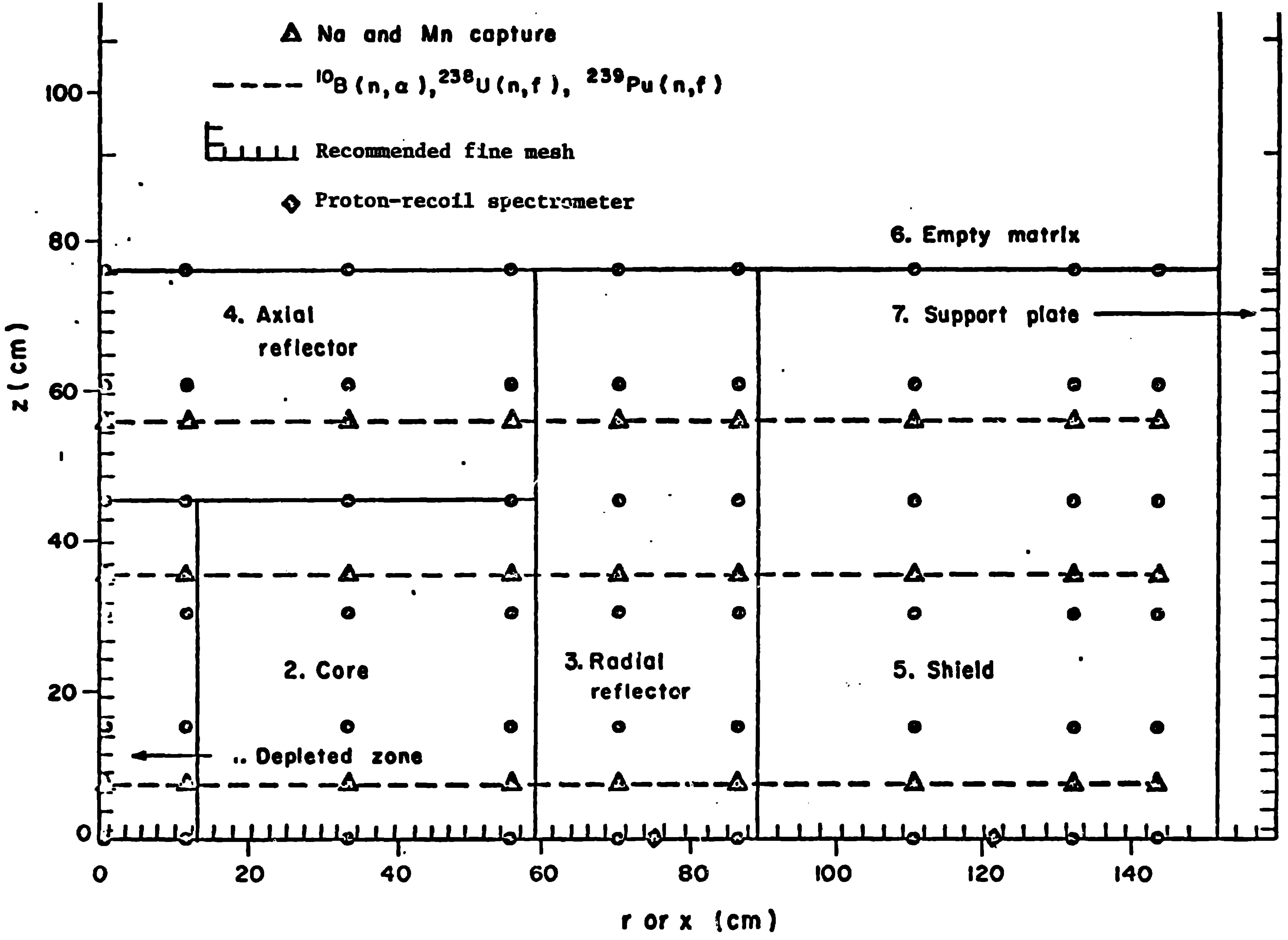

Fig. 8. Calculational Model Showing Recommended Fine Mesh and the Location of the Detectors.

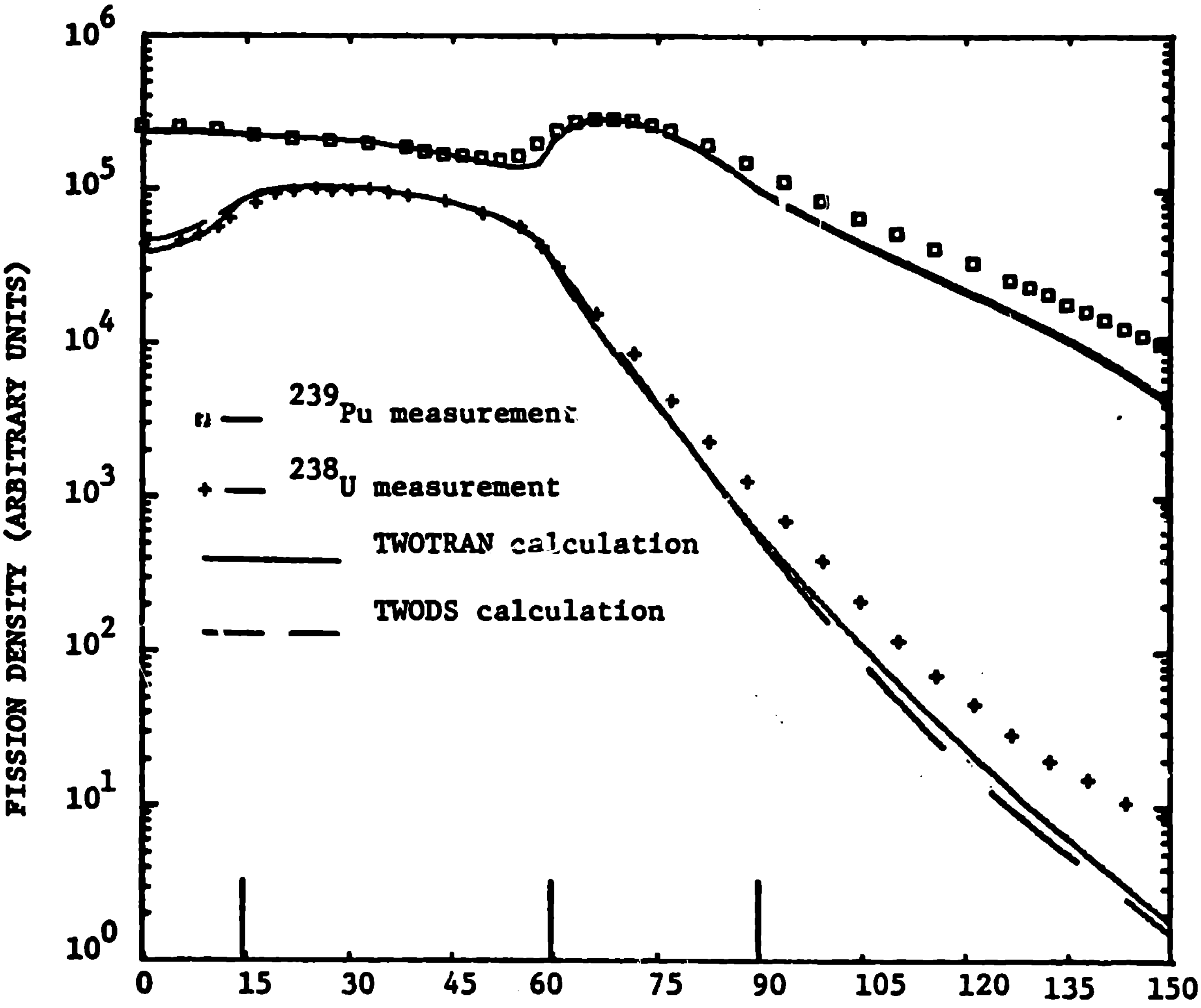

Fig. 9. Measured and calculated ^{239}Pu and ^{238}U fission traverses at z = 35.56 cm. The calculations are normalized to the (arbitrarily scaled) experimental points at r = 33 cm.

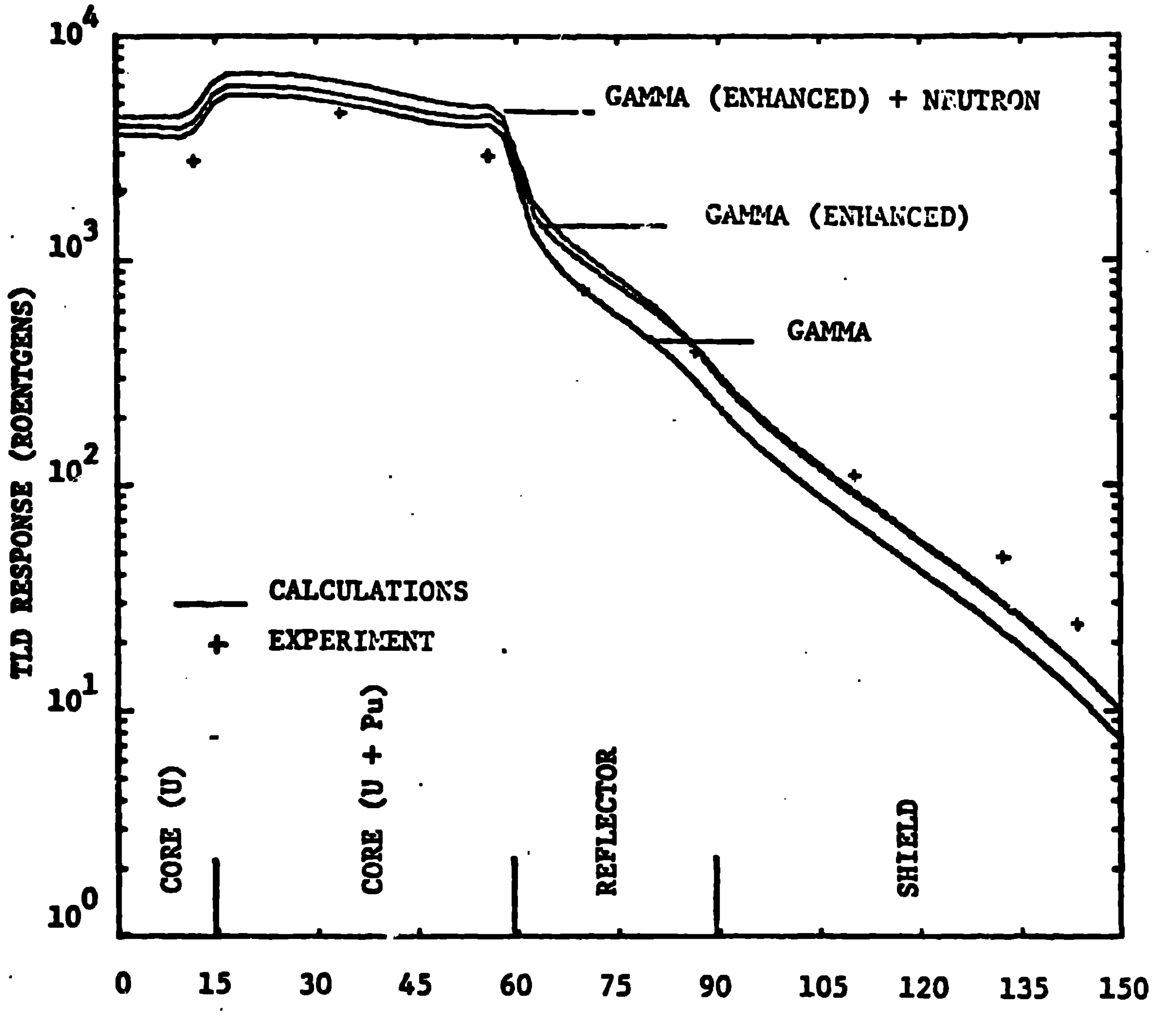

Fig. 10. Measured and calculated TLD response in ZPPR/FTR-2 for a central traverse.

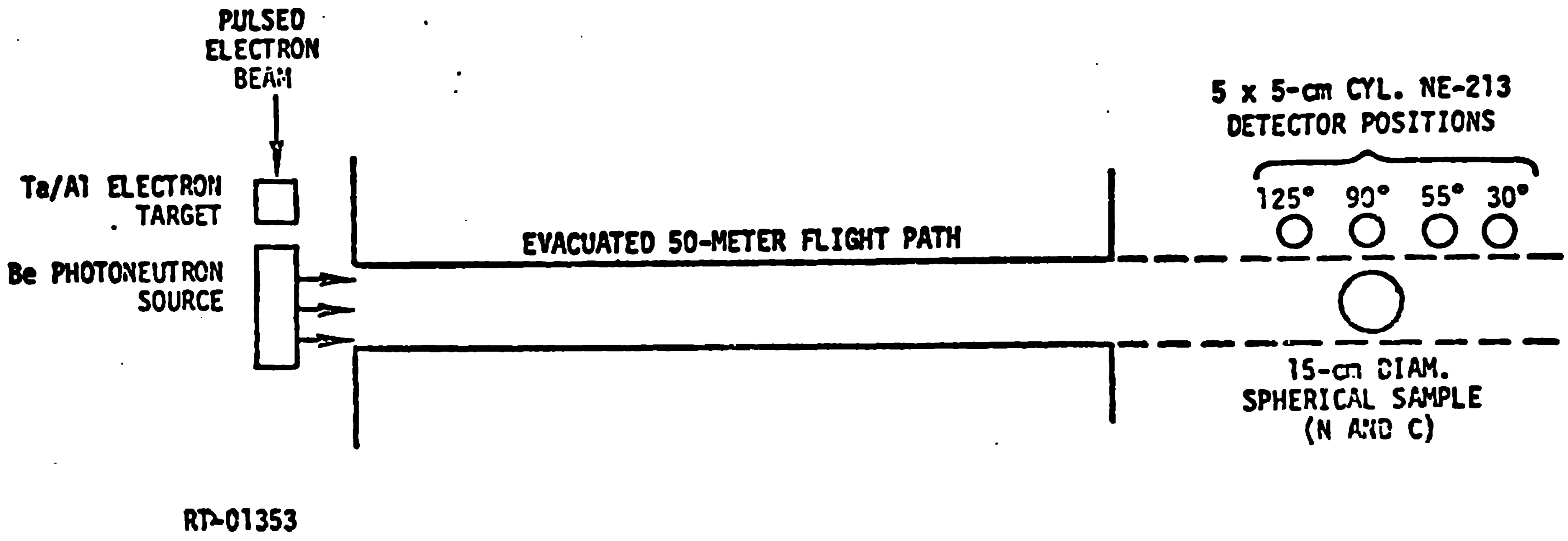

Fig. 11. Experimental geometry for integral experiment to test gamma-ray production and neutron scattering cross sections.

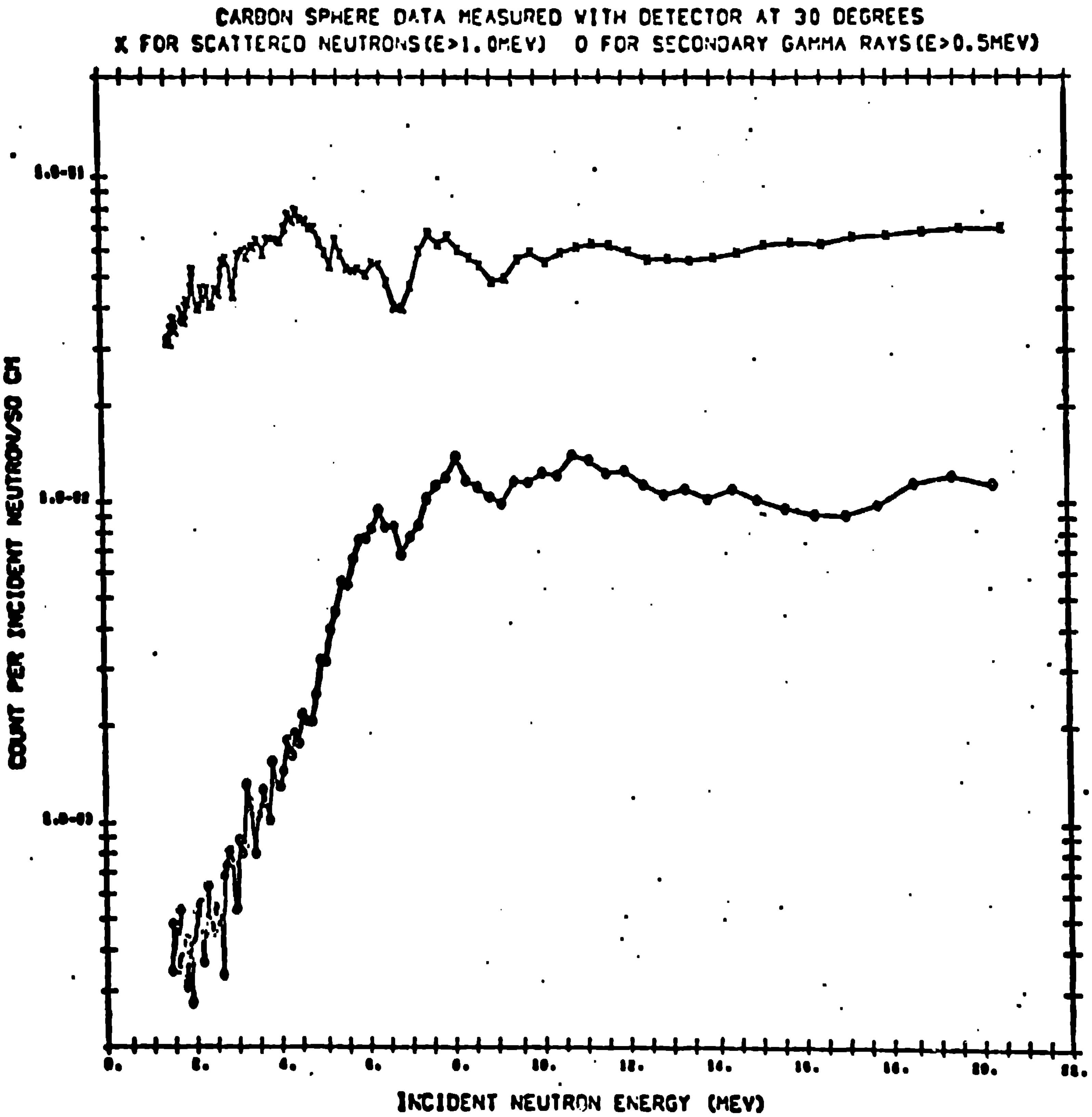

Fig. 12. Carbon sphere data measured with detector at 90 degrees.

TABLE I

NUCLIDES WITH GAMMA-RAY PRODUCTION DATA IN ENDF/B.

NUCLIDE	MAT-NO.	RESPONSIBLE LABORATORY
1-H-1	1269	LOS ALAMOS SCIENTIFIC LAB.
1-H-2	1120	LOS ALAMOS SCIENTIFIC LAB.
2-HE-4	1270	LOS ALAMOS SCIENTIFIC LAB.
3-LI-6	1271	LOS ALAMOS SCIENTIFIC LAB.
3-LI-7	1272	LOS ALAMOS SCIENTIFIC LAB.
4-BE-9	1289	LAWRENCE LIVERMORE LAB.
5-B-10	1273	LOS ALAMOS SCIENTIFIC LAB.
6-C-12	1274	OAK RIDGE NATIONAL LAB.
7-N-14	1275	LOS ALAMOS SCIENTIFIC LAB.
8-O-16	1276	LOS ALAMOS SCIENTIFIC LAB.
11-NA-23	1156	OAK RIDGE NATIONAL LAB.
12-GM	1280	SCIENCE APPLICATIONS, INC.
13-AL-27	1294	LOS ALAMOS SCIENTIFIC LAB.
14-SI		OAK RIDGE NATIONAL LAB.
20-CA	1195	OAK RIDGE NATIONAL LAB.
22-TI	1286	ARGONNE NATIONAL LAB.
23-V	1196	OAK RIDGE NATIONAL LAB.
24-CR	1191	BROOKHAVEN NATIONAL LAB.
25-MN-55	1197	BROOKHAVEN NATIONAL LAB.
26-FE-55	1192	OAK RIDGE NATIONAL LAB.
27-CO-59	1198	ARGONNE NATIONAL AND BROOKHAVEN NATIONAL LAB.
28-NI	1190	BROOKHAVEN NATIONAL LAB.
29-CU		SCIENCE APPLICATIONS.INC.
41-NB-93	1189	ARGONNE NATIONAL AND LAWRENCE LIVERMORE NATIONAL LAB.
42-MO	1287	ARGONNE NATIONAL AND LAWRENCE LIVERMORE LAB.
63-EU-151	1290	BROOKHAVEN NATIONAL LAB.
63-EU-153	1291	BROOKHAVEN NATIONAL LAB.
73-TA-181	1285	LAWRENCE LIVERMORE LAB.
74-W-182	1128	ATOMICS INTERNATIONAL AND LOS ALAMOS SCIENTIFIC LAB.
74-W-183	1129	ATOMICS INTERNATIONAL AND LOS ALAMOS SCIENTIFIC LAB.
74-W-184	1130	ATOMICS INTERNATIONAL AND LOS ALAMOS SCIENTIFIC LAB.
74-W-186	1131	ATOMICS INTERNATIONAL AND LOS ALAMOS SCIENTIFIC LAB.
82-PB	1288	OAK RIDGE NATIONAL LAB.
92-U-235	N/A	LOS ALAMOS SCIENTIFIC LAB.
92-U-238	N/A	WESTINGHOUSE ADVANCED REACTORS DIV. AND LAWRENCE LIVERMORE LAB.
94-PU-239	N/A	GENERAL ELECTRIC AND LOS ALAMOS SCIENTIFIC LAB.
94-PU-240	1265	ARGONNE NATIONAL AND LOS ALAMOS SCIENTIFIC LAB.

TABLE II
CSEWG SHIELDING BENCHMARK EXPERIMENTS

ENCHMARK NUMBER	TYPE	EXPERIMENT	MATERIALS TESTED	NUCLEAR DATA TESTED	REFERE
SDT1	SINGLE-MATERIAL	BROOMSTICK	FE	TOTAL NEUTRON CROSS SECTION	2,3
SDT2	SINGLE-MATERIAL	BROOMSTICK	O	TOTAL NEUTRON CROSS SECTION	4,3
SDT3	SINGLE-MATERIAL	BROOMSTICK	N	TOTAL NEUTRON CROSS SECTION	5,3
SDT4	SINGLE-MATERIAL	BROOMSTICK	NA	TOTAL NEUTRON CROSS SECTION	6,3
SDT5	SINGLE-MATERIAL	BROOMSTICK	STAINLESS STEEL	TOTAL NEUTRON CROSS SECTION	7,3
SDT6	SINGLE-MATERIAL	GAMMA-RAY PRODUCTION FROM CAPTURE	FE,STEEL, N,AND NA	CAPTURE GAMMA-RAY SPECTRA	8,9
SDT7	SINGLE-MATERIAL	GAMMA-RAY PRODUCTION IN FAST SPECTRUM	FE,STEEL, O,AND NA	GAMMA-RAY PRODUCTION IN FISSION SPECTRUM	10,11
STD8	MULTI-MATERIAL	ZPPR/FTR-2 SHIELD	PU AND U ISOTOPES, O,NA,C,FE, NI,CR	ALL NEUTRON, GAMMA-RAY PRODUCTION, AND PHOTON INTEACTION CROSS SECT.	12,13 15
SDT9	MULTI-MATERIAL	FFTF RADIAL SHIELD	B,C,NA,O, N,AL,FE, NI,CR,CU, MO,SI,MN, U-235,U-238	NEUTRON CROSS SECT.	16,17 19,20
SDT10	SINGLE-MATERIAL	LLL PULSED SPHERES	LI-6,LI-7, BE,C,N,O, MG,TI,FE, PB,U-235, U-238,PU-239	HI-ENERGY NEUTRON CROSS SECT.	21,22
SDT11	SINGLE-MATERIAL	IRON BENCH-MARK	STAINLESS STEEL,FE	NEUTRON CROSS SECT.	23,24
SDT12	SINGLE-MATERIAL	SODIUM BENCH-MARK	NA	NEUTRON CROSS SECT.	25,2(

TABLE II
CSEWG SHIELDING BENCHMARK EXPERIMENTS

TYPE	EXPERIMENT	MATERIALS TESTED	NUCLEAR DATA TESTED	REFERENCES
LE-MATERIAL	BROOMSTICK	FE	TOTAL NEUTRON CROSS SECTION	2,3
LE-MATERIAL	BROOMSTICK	O	TOTAL NEUTRON CROSS SECTION	4,3
LE-MATERIAL	BROOMSTICK	N	TOTAL NEUTRON CROSS SECTION	5,3
LE-MATERIAL	BROOMSTICK	NA	TOTAL NEUTRON CROSS SECTION	6,3
LE-MATERIAL	BROOMSTICK	STAINLESS STEEL	TOTAL NEUTRON CROSS SECTION	7,3
LE-MATERIAL	GAMMA-RAY PRODUCTION FROM CAPTURE	FE,STEEL, N,AND NA	CAPTURE GAMMA-RAY SPECTRA	8,9
LE-MATERIAL	GAMMA-RAY PRODUCTION IN FAST SPECTRUM	FE,STEEL, O,AND NA	GAMMA-RAY PRODUCTION IN FISSION SPECTRUM	10,11
I-MATERIAL	ZPPR/FTR-2 SHIELD	PU AND U ISOTOPES, O,NA,C,FE, NI,CR	ALL NEUTRON, GAMMA-RAY PRODUCTION, AND PHOTON INTEACTION CROSS SECT.	12,13,14, 15
I-MATERIAL	FFTF RADIAL SHIELD	B,C,NA,O, N,AL,FE, NI,CR,CU, MO,SI,MN, U-235,U-238	NEUTRON CROSS SECT.	16,17,18, 19,20
LE-MATERIAL	LLL PULSED SPHERES	LI-6,LI-7, BE,C,N,O, MG,TI,FE, PB,U-235, U-238,PU-239	HI-ENERGY NEUTRON CROSS SECT.	21,22
LE-MATERIAL	IRON BENCH-MARK	STAINLESS STEEL,FE	NEUTRON CROSS SECT.	23,24
LE-MATERIAL	SODIUM BENCH-MARK	NA	NEUTRON CROSS SECT.	25,26

TABLE III

PRELIMINARY RESULTS

GAMMA-RAY PRODUCTION ARISING FROM THERMAL CAPTURE

GAMMA ENERGY INTERVAL (MEV)	MAT=1180 IRON EXP	CALC	PCT DIFF	MAT=1133 NITROGEN EXP	CALC	PCT DIFF	MAT=1156 SODIUM EXP	CALC	PCT DIFF
1.0 - 1.5	126	152	- 21	0.0	0.2	--	79	90	- 14
1.5 - 2.0	455	336	+ 26	30.0	25.9	14.7	76	48	+ 37
2.0 - 2.5	99	163	- 65	0.0	0.1	--	143	132	+ 8
2.5 - 3.0	152	181	+ 19	6.4	5.9	7.8	254	232	+ 9
3.0 - 3.5	233	302	- 30	0.0	0.7	--	76	92	- 21
3.5 - 4.0	88	194	-120	22.8	20.3	10.9	182	199	- 9
4.0 - 4.5	182	271	- 49	0.0	0.0	--	34	15	+ 56
4.5 - 5.0	83	60	+ 28	13.4	12.4	7.5	14	3	+ 79
5.0 - 5.5	25	35	- 40	43.3	38.7	10.6	12	6	+ 50
5.5 - 6.0	250	151	+ 40	25.2	21.7	14.9	31	33	- 6
6.0 - 6.5	255	236	+ 7	15.2	14.3	5.9	115	118	- 3
6.5 - 7.0	13	7	+ 46	0.0	0.0	--	0	0	--
7.0 - 7.5	139	125	+ 10	7.8	7.4	5.1			
7.5 - 8.0	1280	1268	+ 1	0.0	0.0	--			
8.0 - 8.5				3.5	3.4	2.9			
8.5 - 9.0	20	25	- 25	0.17	0.1	--			
9.0 - 9.5	83	102	- 23	1.6	1.1	31.3			
9.5 - 10.0				0.0	0.1	--			
10.0 - 10.5				0.0	0.0	--			
10.5 - 11.0				11.0	10.0	9.1			
	WT AV DIFF		31	WT AV DIFF		10.9	WT AV DIFF		14

TABLE IV

PRELIMINARY RESULTS

GAMMA-RAY PRODUCTION FROM A FAST NEUTRON SPECTRUM

GAMMA ENERGY INTERVAL (MEV)	MAT=1180 IRON EXP	CALC	PCT DIFF	MAT=1134 OXYGEN EXP	CALC	PCT DIFF	MAT=1156 SODIUM EXP	CALC	PCT DIFF
1.0 - 1.5	278	252	+ 9				59	2	+ 97
1.5 - 2.0	132	100	+ 24				131	99	+ 24
2.0 - 2.5	101	94	+ 7				47	37	+ 21
2.5 - 3.0	74	84	- 14				50	60	- 20
3.0 - 3.5	44	31	+ 30				22	8	+ 64
3.5 - 4.0	33	24	+ 27				15	12	+ 20
4.0 - 4.5	8.7	2.9	+ 67				7.3	9.4	- 29
4.5 - 5.0	5.6	3.1	+ 45				4.6	1.3	+ 72
5.0 - 5.5	3.8	2.8	+ 26				2.9	1.0	+ 65
5.5 - 6.0	2.4	2.4	0				3.7	4.1	- 11
6.0 - 6.5				90	80	+ 11	4.2	1.5	+ 64
6.5 - 7.0									
7.0 - 7.5				55	47	+ 15	2.4	0.3	+ 88
7.5 - 10.0									
	WT AV DIFF		15	WT AV DIFF		13	WT AV DIFF		40