Cs^{133} (n,2n) CROSS-SECTION

AT 15.6 AND 16.1 MeV

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THESIS

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TABLE OF CONTENTS

LIST OF TABLES

LIST OF ILLUSTRATIONS

CHAPTER I

INTRODUCTION

There has been some interest in recent years in the investigation of the (n,n'γ) reaction in Cs^{133} (3, 5, 6). Since the (n,2n) reaction is the principal competing reaction with this process at neutron energies between 12 and 19 MeV (1), a knowledge of the (n,2n) reaction cross-section in this energy range should prove quite useful. There have been a few published values for this reaction around 14 MeV (2, 7, 8, 9, 10) and one excitation function (2). These values are listed in Table I and shown graphically in Figure 1. It can be seen that the experimentally obtained cross-sections have relatively large errors and that the data must certainly contain systematics errors, since values of the same cross-section measured by different authors differ more from each other than can be expected from the given error limits.

In all cases the cross-section has been obtained by the activation method. This method makes use of the fact that the residual nucleus, Cs^{132}, decays to excited states of Xe^{132}, which decay to the Xe^{132} ground state by gamma emission. In this method decay schemes must be well known and absolute calibrations, or at least comparison with calibrated standards, must be possible. The equation through which the cross-section

is determined will be developed thoroughly in a later chapter, but stated simply it is

$$Y = N\phi\sigma(1-e^{-\lambda t}) \qquad (1.1)$$

where

Y is the absolute disintegration rate of the product nucleus at the end of the irradiation,

N is the number of target nuclei,

ϕ is the neutron flux density,

σ is the cross-section,

λ is the decay constant of the product nucleus, and

t is the duration of the irradiation, the irradiation time.

This simplified equation neglects the variation of the neutron output with time, but this will be taken up in the appropriate chapter.

In all previous cases listed in Table I the activity of the sample, Y, has been based on the detection of the 670 keV gamma ray, given off in the decay of Xe^{132}, by NaI scintillation detectors. In the present work a 2 cc Ge(Li) detector is used, and its superior resolution has shown that there exists a gamma ray of 630 keV which is unresolved from the 666 keV gamma ray when a NaI detector is used. A typical spectrum obtained by the Ge(Li) detector is shown in Figure 2. This spectrum is the same as that obtained by Johnson (4), who also used a Ge(Li) detector. The decay scheme which he has proposed (Figure 3) has been adopted for the purposes of this analysis.

The intent of this investigation is, then, the determination of the values of the $Cs^{133}(n,2n)Cs^{132}$ cross-section at neutron energies of 15.6 and 16.1 MeV. Neutrons of this energy are produced with comparative ease by means of the D-T reaction, in which deuterons of energy 500 and 750 keV, respectively, are impingent upon a tritium target. The accelerating device used in the experiment was the 2-Mv Van de Graaff accelerator of the Regional Nuclear Physics Laboratory, which is operated jointly by North Texas State University and Southern Methodist University. Other instrumentation is listed in Appendix D.

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CHAPTER II

DETERMINATION OF NEUTRON ENERGIES

The energy of the neutrons which participate in the Cs reaction and the spread in this energy are determined by the following factors:

1. deuteron energy loss in the tritium target,
2. the angular distribution of neutrons due to the finite size of the beam spot and the irradiated sample,
3. variation of the deuteron energy via changes in the accelerating voltage,
4. scattering by materials between the target and sample, and
5. self-shadowing of the sample.

These factors are listed in the order of their importance and will be discussed in reverse order.

The self-shadowing of the sample is typically negligible. For a sample thickness of 50 mils, which was the case in this experiment, and a total inelastic scattering cross-section of Cs and Cl of approximately 7 barns, the fraction of impinging neutrons which undergo some interaction is considerably less than one percent.

As the illustration in Figure 6 shows, the only material between the tritium target and the sample is the copper backing on the target and the nylon cap used to hold the CsCl sample. The number of neutrons scattered from both these materials is readily calculated to be less than one percent of the total number of neutrons available for interaction.

As mentioned earlier in Chapter I, the accelerator used in this work was the 2-Mv Van de Graaff accelerator of the Regional Nuclear Physics Laboratory. For the two energies reported in this work D_2^+ ions were subject to accelerating voltages of 1.0 and 1.5 MeV. The RF ion source of the accelerator system typically produces three masses of deuterium ions; therefore, the beam was passed through a bending magnet in order to separate the atomic ions, D_2^+, from the rest of the beam. The beam was then caused to pass through a slit assembly which was insulated from ground. Whenever the beam hits either slit a signal is sent through an amplifier to the corona points, causing the corona current to either increase or decrease thus, maintaining a constant beam energy. Thus, the energy spread due to voltage irregularity is kept extremely small. A diagram of the accelerating apparatus is shown in Figure 4.

The size of the beam spot was well defined by means of a circular aperture of 7/16 inch diameter placed in front of the target, as shown in Figure 4. The diameter of the sample was 1/2 inch, and the perpendicular distance from target to sample was 1 28/64 inches. The geometrical relationship

between the sample and target is shown in Figure 6. A computer program was developed to obtain the mean energy of the neutrons striking the sample, given the dimensions of the target and sample, their separation distance, and the energy of the deuteron when it undergoes reaction with the tritium. The program also calculates the mean deviation from the mean and the percent abundance, in the sample, of neutrons emitted at various angles and, thus, at various energies. The program is reproduced in Appendix B along with a sample of the output to be expected.

The target used in this experiment was a Texas Nuclear-produced 6 curie titanium tritide foil of approximately 1 mg/cm^2 thickness (2). From works by Gunnersen et al. (1) and extensions of this work by Nagel (3), a mean energy loss of 50 keV has been assigned with a spread in energy of $\pm$ 50 keV. This corresponds to a neutron energy of 15.60 $\pm$ .12 MeV for a deuteron energy of 500 keV and 16.08 $\pm$ .10 MeV for a deuteron energy of 750 keV. A combination of the energy spread due to energy loss in the target with the spread in energy due to the finite size of the beam and sample yields a neutron energy of 15.60 $\pm$ .20 MeV at 500 keV deuteron energy and 16.10 $\pm$ .20 MeV at 750 keV.

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CHAPTER III

FLUX DENSITY MEASUREMENT

Examination of equation (1.1) shows the cross-section to be directly dependent on the neutron flux density. It is necessary, therefore, to have as accurate a measurement as possible for this quantity. There are several methods commonly used to determine the flux: the associated particle method, the proton recoil method, the long counter, calculations involving the beam current, and through a reaction of well-known cross-section.

Since there is a one-to-one correspondence between the number of neutrons and the number of alpha particles emitted, the flux density may be measured by mounting an alpha counter under some fixed angle and, after determining the solid angle intercepted by using an absolutely calibrated alpha source, counting the alphas given off in the reaction. This method has certain drawbacks, however, when a thick target is used. The deuterons lose energy on entering the target and the alpha particles in leaving the target. This can cause a change in the neutron emission angle relative to the fixed alpha-emission angle. This factor can be minimized, however, by proper placement of the alpha detector. In this experiment, use was made of a silicon surface barrier detector (alpha

detector) placed at 175° to the incident deuteron beam direction. No attempt was made to use the emitted alphas as an absolute flux measuring device due to the difficulties involved in calibrating the detector for alpha particle energies between 2 and 3 MeV. However, the alpha count was used in checking the constancy of the flux throughout each run. An alpha spectrum obtained from a run using 1.50 MeV D_2^+ ions is shown in Figure 7. With some further work the alpha spectrum could be used to obtain information about the energy of the deuterons when they react with the tritium and the spread in this energy.

In the proton recoil method a good knowledge of the n+p scattering cross-section is required. This is the primary reaction examined by any of the following devices: the telescope counter, ionization chamber, stillbene crystal, and nuclear emulsions. Use of one of these methods should prove a useful check on the method used in this work.

A long counter counts thermalized neutrons. Its efficiency for the primary energy of the incident neutrons depends on the size and shape of the paraffin moderator surrounding it. For this reason this device is usually used only as a monitor of the neutron flux. It has one serious deficiency in this respect, however, in that it does not distinguish between different neutron energy groups. Thus, in the case of long runs there may be extensive deuteron buildup in the target, with the resulting emission of D-D neutrons, and the count

rate in the long counter may increase, whereas the flux density of the D-T neutrons may in fact be constant. A spherical neutron dosimeter with a polyethylene sphere of one foot diameter was used in this work as a monitor. It works on the same principle as the long counter and is subject to the same faults. It does provide, in conjunction with the alpha count, a measure of the constancy of the flux during a run.

Calculations involving the beam current require specific knowledge about the beam and target composition. Also, corrections must be made for deuteron and neutron absorption in the target. In general the method is too laborious to be of use. At best it can serve as a rough estimate of the flux density.

By far the most preferred method is that of using a reaction of well established cross-section to determine the flux. In this work two such reactions have been used, the $Fe^{56}(n,p)Mn^{56}$ reaction and the $Cu^{65}(n,2n)Cu^{64}$ reaction. The Q-values for these reactions, as well as for the $Cs^{133}(n,2n)Cs^{132}$ reaction, are such as to preclude any interaction with neutrons produced by the D-D reaction. Thin foils of the desired monitor are placed in front of and behind the sample of CsCl being irradiated. Any difference in the flux measured by the two foils has been completely accounted for by the fact that they intercept different solid angles, as seen by the target, due to the fact that they are different distances from the target (5). Figure 6 shows the relative positions of target, sample, and

monitor foils. The fact that the difference in flux is wholly accounted for on a geometrical basis is further validation of the negligibility of the neutron absorption by the nylon cap and the CsCl sample. The cross-section for the $Fe^{56}(n,p)Mn^{56}$ reaction was taken to be 85 millibarns at 15.6 MeV neutron energy (1,3). The cross-section used for the $Cu^{65}(n,2n)Cu^{64}$ reaction was 1050 millibarns at 15.6 MeV and 1070 millibarns at 16.1 MeV (2,4).

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CHAPTER IV

DETECTOR CALIBRATION AND ABSOLUTE ACTIVITY MEASUREMENT

In order to determine the absolute activity of the sample or monitor in equation (1.1), the efficiency of the Ge(Li) detector must be determined for the energy of the gamma ray under consideration. This was accomplished in this experiment through comparison with calibrated standards which produced essentially the same energy gamma as the sample or monitor. In the case of Cs^{132}, whose decay results in 666 keV gamma rays, a calibrated standard of Cs^{137} (gamma energy 662 keV) was used. For Cu^{64} (511 keV annilation radiation) a calibrated standard of Na^{22} was used; and for Mn^{56} (845 keV), a standard of Mn^{54} (840 keV). For the detector and analyzer system used in this work a difference in energy in the sample and standard of 5 keV should result in negligible difference in efficiency. The 2 cc Ge(Li) detector, in conjunction with the Nuclear Data 1024 channel analyzer, gives a resolution for the 662 keV gamma from Cs^{137} of 6.5 keV at full-width-half-maximum intensity. All calibrated standards were furnished by New England Nuclear Corporation, who list a reliability of 5 percent in their calibration of the standards used. In all

cases considered, twice the high energy side of the photopeak (less background) was taken to be the photopeak area. It was necessary to consider factors such as branching ratios, internal conversion coefficients, geometrical efficiency, and attenuation effects in both the sample and the calibrated standard. From these considerations the following equation derives:

$$\Delta a = P \frac{Y_S}{P_S/t_S} \frac{b_S}{b} \frac{(1+\alpha)}{(1+\alpha_S)} a' g \qquad (4.1)$$

where:

P is the photopeak area of the sample or monitor,

Y_S the activity of the standard in dis/sec,

P_S the photopeak area of the standard obtained in a time

t_S if the half-life is large compared to t_S,

b is the branching ratio,

α the internal conversion coefficient,

a' the ratio of geometrical efficiencies for sample and standard, and

Δa the absolute number of disintegrations during some particular time interval.

It was determined experimentally that the ratio of attenuation factors, a', was 1; and, from a paper by Ruffle (1), the factor g was determined to be 1.1 for a ratio of point-to finite-geometry efficiencies. Figure 5 shows the geometrical relationship between the sample and detector. If "e" stands

for the ratio $Y_s t_s/P_s$ and "B" for the ratio $b_s(1+\alpha)/b(1+\alpha_s)$, then the values for e and B are given in Table II for Cs^{132}/Cs^{137}, Cu^{64}/Na^{22}, and Mn^{56}/Mn^{54}.

From this, then, the absolute activity of the sample can be determined. From the well known radioactive decay equation,

$$a = a_o e^{-\lambda t}, \qquad (4.2)$$

where a is the number of radioactive atoms in the sample at any time t and a_o is the number present at $t = 0$, the number of atoms which decay from some time t_1 to some other time t_2 is

$$\Delta a = a_o(e^{-\lambda t_1} - e^{-\lambda t_2}). \qquad (4.3)$$

Then use of the fact that the absolute activity at the end of the irradiation time, Y_o, is just equal to λa_o yields

$$Y_o = \frac{\lambda \Delta a}{(e^{-\lambda t_1} - e^{-\lambda t_2})}. \qquad (4.4)$$

Combining equations (4.4) and (4.2) gives

$$Y_o = \frac{\lambda P e B}{(e^{-\lambda t_1} - e^{-\lambda t_2})}, \qquad (4.5)$$

where the attenuation and geometrical effects have been neglected.

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CHAPTER V

CORRECTION FOR VARIABLE NEUTRON OUTPUT AND CALCULATION OF CROSS-SECTION

To evaluate the variation of neutron output with time during a run, consider equation (1.1) to apply to each of n time intervals of length δ. Then for the i^{th} time interval

$$Y_i = CM_i\sigma N(1-e^{-\lambda\delta})e^{-\lambda(T-i\delta)}, \qquad (5.1)$$

where $\phi_i = CM_i$, T is the total irradiation time, M_i the total alpha count during the i^{th} time interval, and Y_i the activity due to the i^{th} increment of irradiation time. The constant C is determined by means of the monitor reaction. Thus the total activity at the end of the irradiation is

$$Y_o = CN\sigma(1-e^{-\lambda\delta})e^{-\lambda T}\sum_{i=1}^{i=n} M_i e^{i\lambda\delta}. \qquad (5.2)$$

Combination of equation (4.5) with (5.2) and use of the subscript c to represent quantities associated with the cesium sample and f for those associated with the flux monitor lead to the following expression for calculation of the cross-section.

$$\sigma_c=\sigma_f \frac{\lambda_c}{\lambda_f}\frac{P_c}{P_f}\frac{e_c}{e_f}\frac{B_c}{B_f}\frac{A_c}{A_f}\frac{(e^{-\lambda t_1}-e^{-\lambda t_2})_f}{(e^{-\lambda t_1}-e^{-\lambda t_2})_c}\frac{M_f}{M_c}\frac{(1-e^{-\lambda f\delta})}{(1-e^{-\lambda c\delta})}$$
$$\times \frac{e^{-\lambda_f T}}{e^{-\lambda_c T}}\frac{M_i e^{i\lambda_f\delta}}{M_i e^{i\lambda_c\delta}} \qquad (5.3)$$

Needless to say, evaluation of this equation could consume vast quantities of time if all the exponentials had to be looked up in tables. For this reason a computer program was developed which would handle the chore nicely. The program (Appendix C) is written in FORTRAN II (AFIT) and does not require input FORMAT statements. Thus the data can be key-punches in the most convenient manner, subject to the order of the READ statements. The half-life used in this calculation was determined to be 6.57 days.

It might be noted that although equation (5.3) was used in all calculations, the deviation from values obtained using equation (1.1) was never greater than 5 percent. The cross-section values obtained at 15.6 MeV neutron energy were 1223 $\pm$ 50 mb; and at 16.1 MeV, 1090 $\pm$ 75 mb. The spread in the cross-section values is due to the mean deviation of several runs. Two runs were made at 15.6 MeV, using iron foil monitors and four runs using copper foil monitors. Two runs were made at 16.1 MeV using only copper foil monitors.

CHAPTER VI

CONCLUSION

The results of this work tend to confirm the work done by Bormann (1) and to indicate that the values obtained by Nagel (2) and Vonach (4) are too high. The theoretical values due to Pearlstein (3) seem to indicate a continuing increase in the cross-section versus energy curve, but this work indicates this is definitely not the case.

Another point brought out by this investigation is that more work needs to be done in the area of deuteron energy loss in the target. If thinner targets were used, a lower energy loss should be expected; however, information on the tritium distribution in the target is virtually nonexistent. The scarcity of experimental data on this subject is unfortunate.

It might prove useful to calibrate the detector by an independent method just as a check on the method used in this work. Another check would be to use calibrated standards from another supplier and check the reproducibility of the values.

As concerns the reproducibility of the values in this work, the six values obtained at 15.6 MeV show very good reproducibility as demonstrated in the mean deviation quoted.

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APPENDIX A

TABLE I

CROSS-SECTION VALUES

Reference	Energy (MeV)							
	13.1	14.1	14.6	15.1	15.6	16.1	18.0	19.6
Bormann	1200 ±110	1290 ±110	....	....	....	1140 ±90	1100 ±90	870 ±70
Nagel	....	1682 ±140	1612 ±140	....	....	....	....	....
Pearlstein (Theoretical)	1535	1700	....	1859	....	....	....	....
Vonach	....	1550 ±250	....	....	....	....	....	....
This Work	....	....	....	....	1223 ±50	1090 ±75	....	....

TABLE II

COEFFICIENTS USED IN CROSS-SECTION CALCULATION

Coefficients	Cs^{132}/Cs^{137}	Cu^{64}/Na^{22}	Mn^{56}/Mn^{54}
e	2270	920	3512
B	.879	5.051	1.01

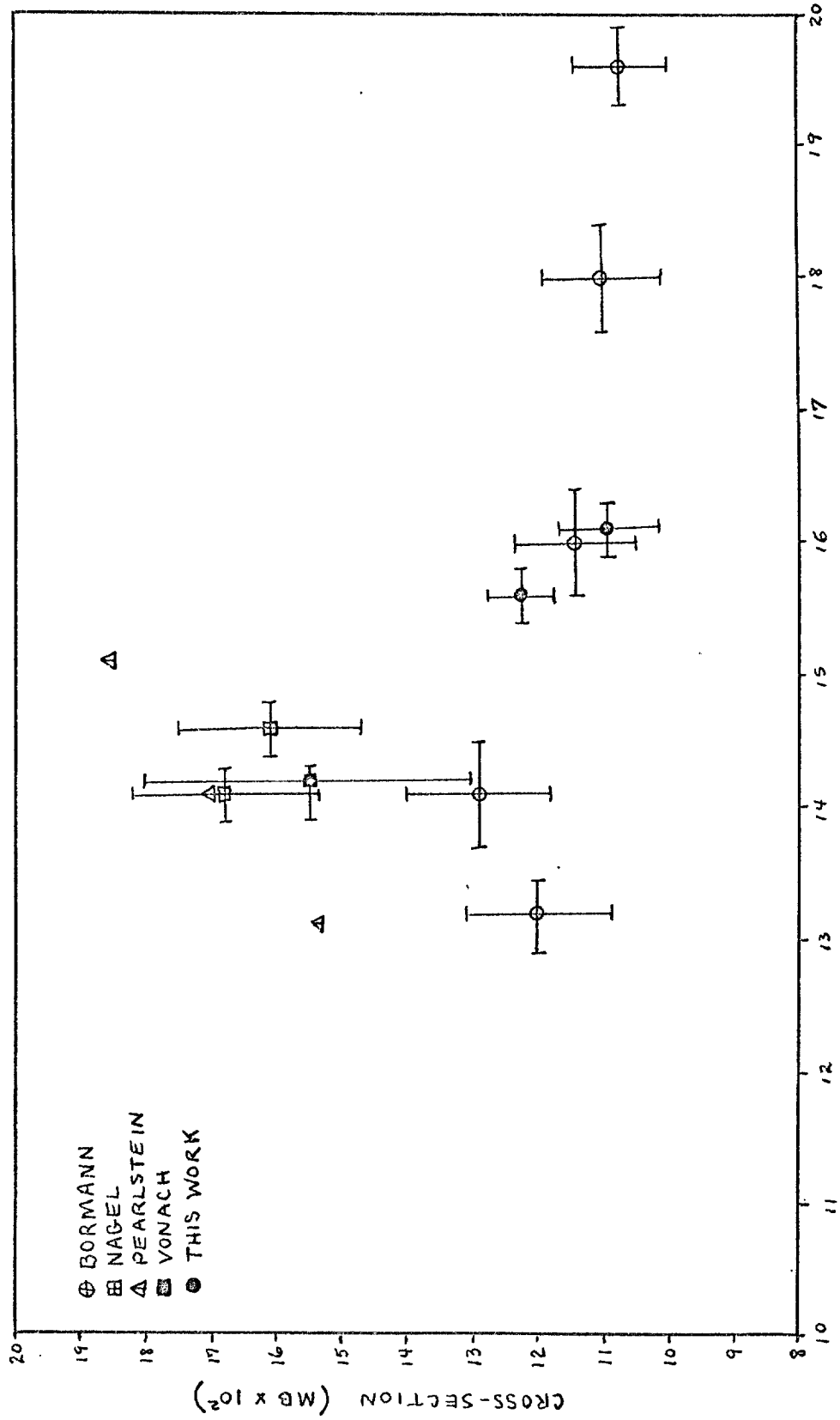

Fig. 1--Graph of cross-section values.

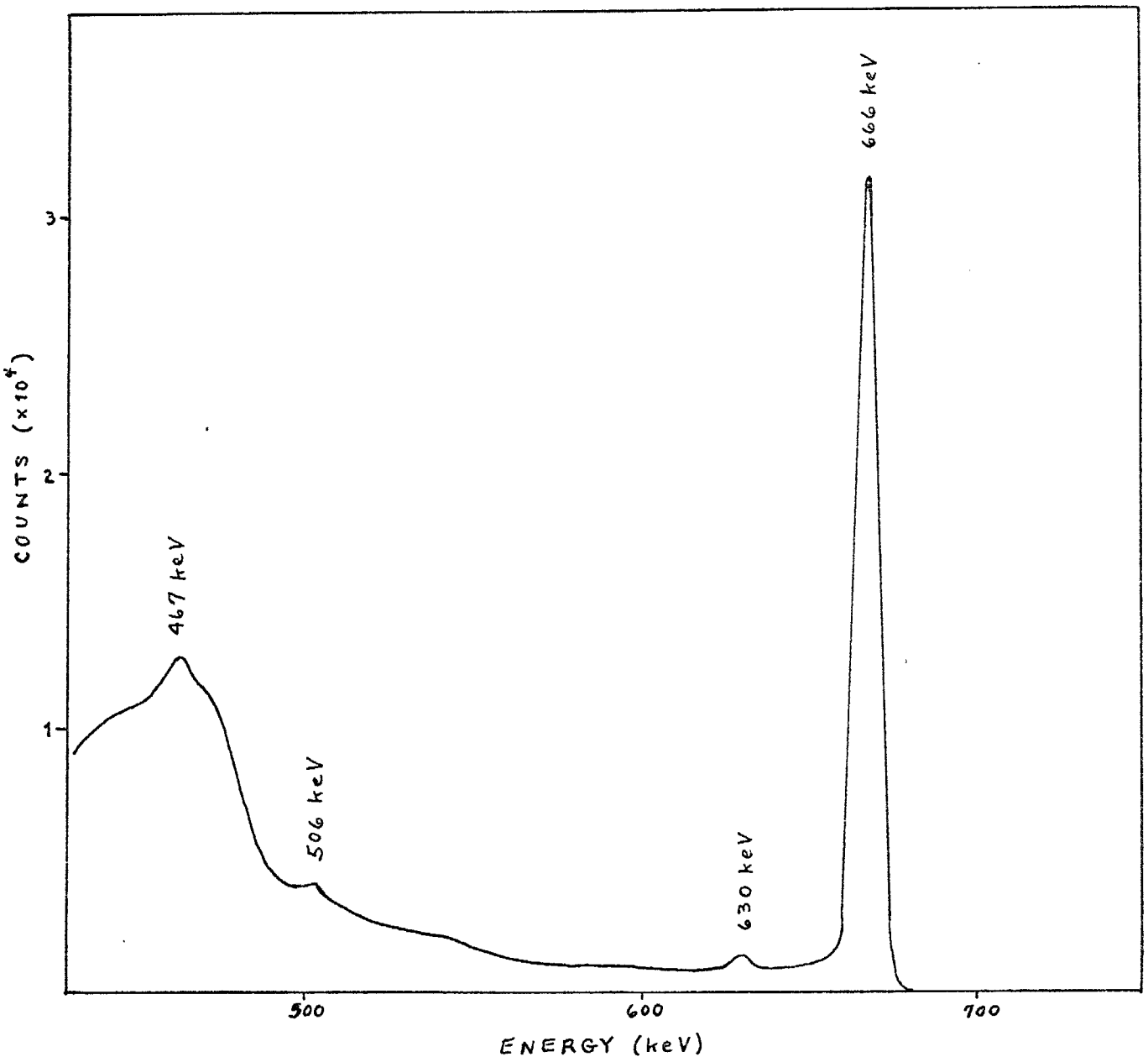

Fig. 2--Low-energy portion of Cs^{132} gamma-ray spectrum taken with lithium-drifted germanium detector.

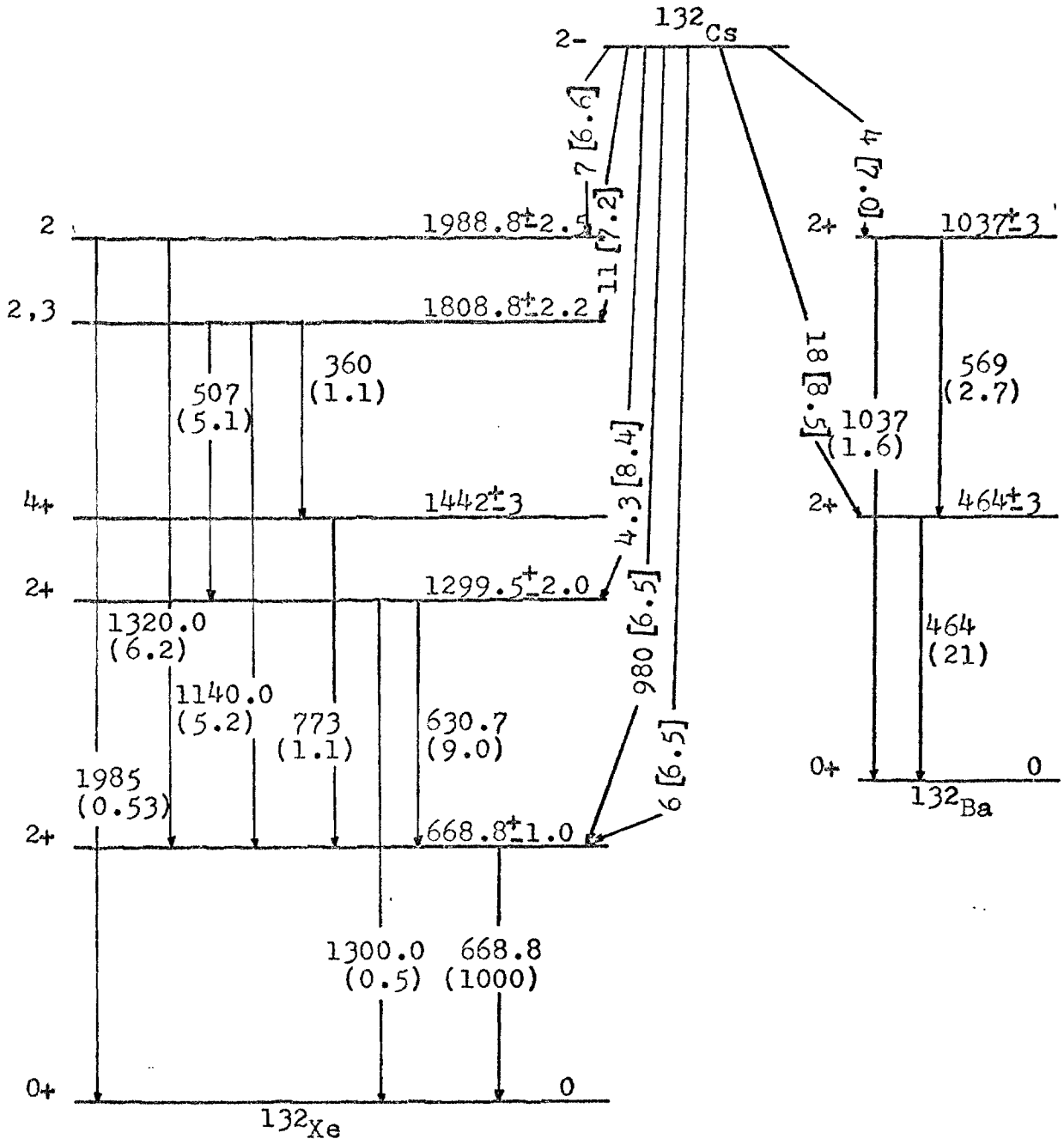

Fig. 3--Decay scheme for Cs^{132}

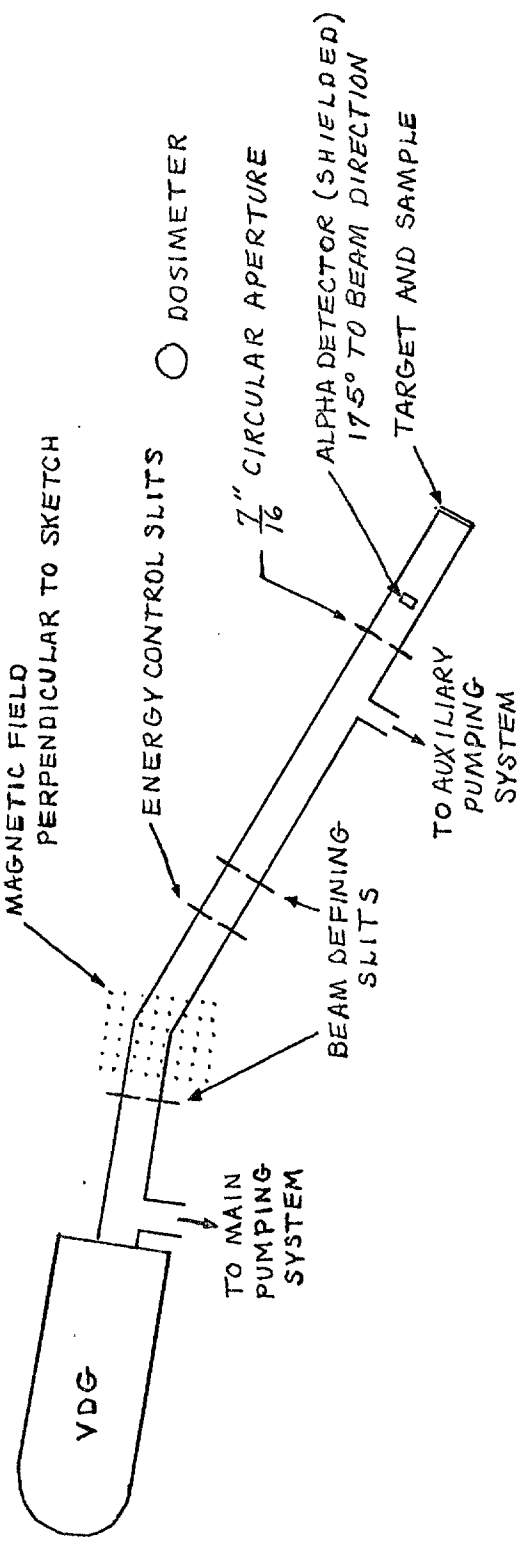

Fig. 4--Sketch of accelerator apparatus.

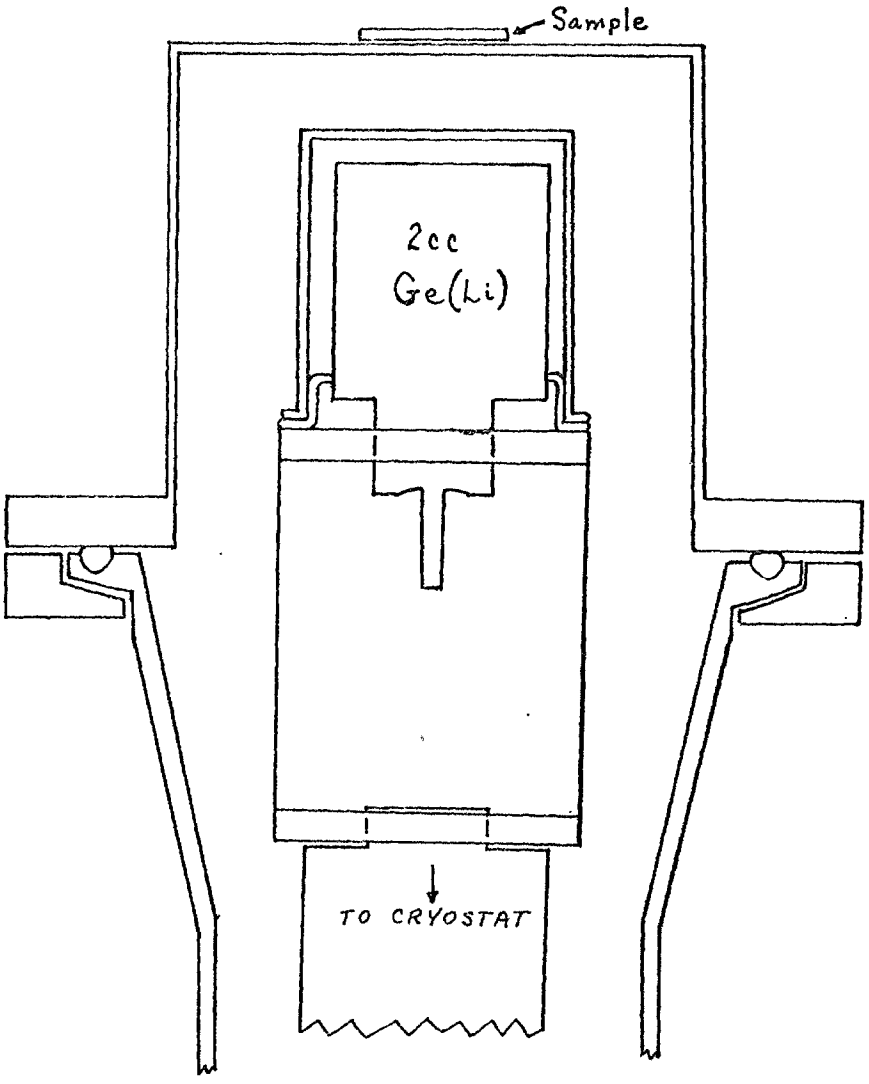

Fig. 5-- Sample-detector geometry.

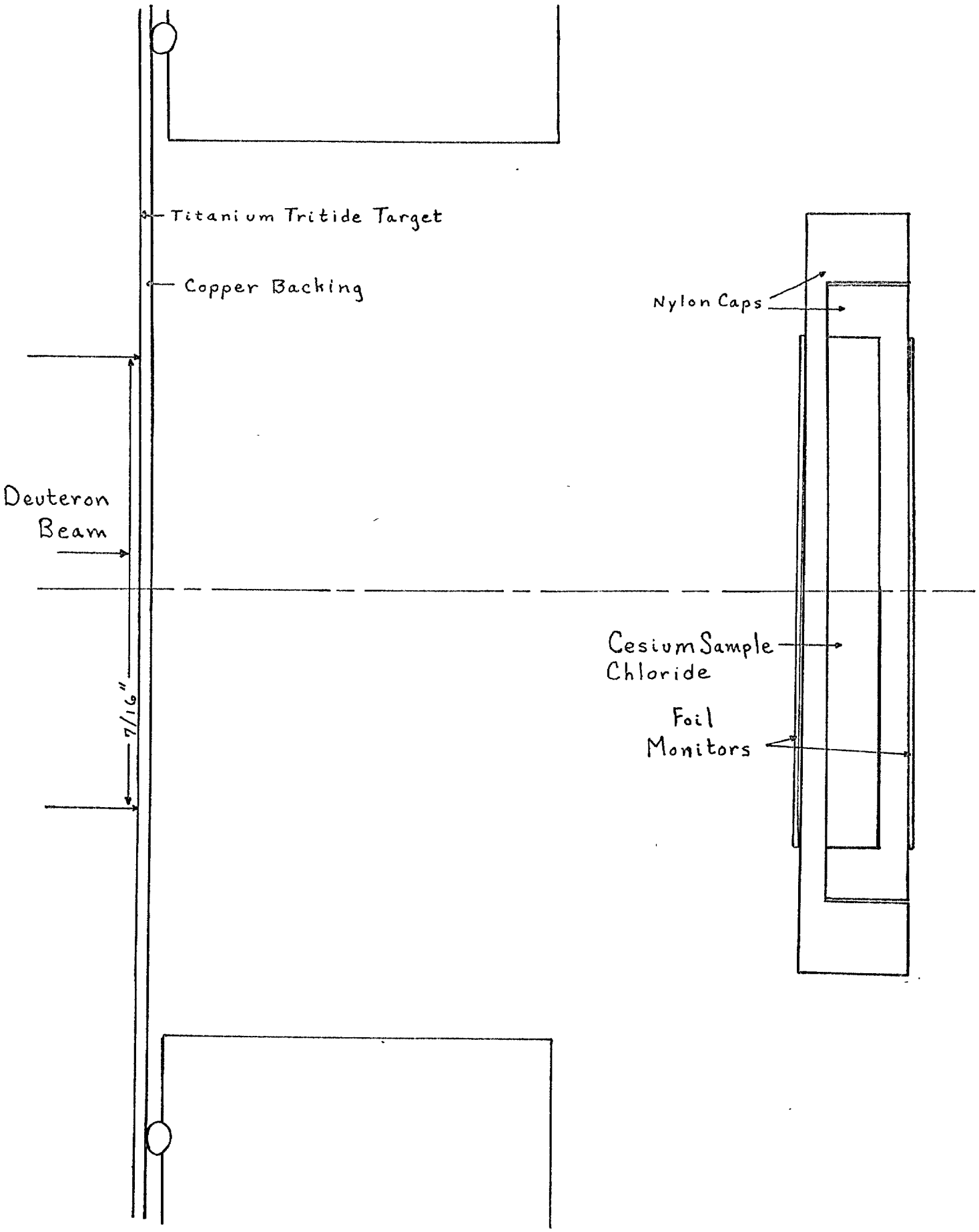

Fig. 6-- Target-sample geometry.

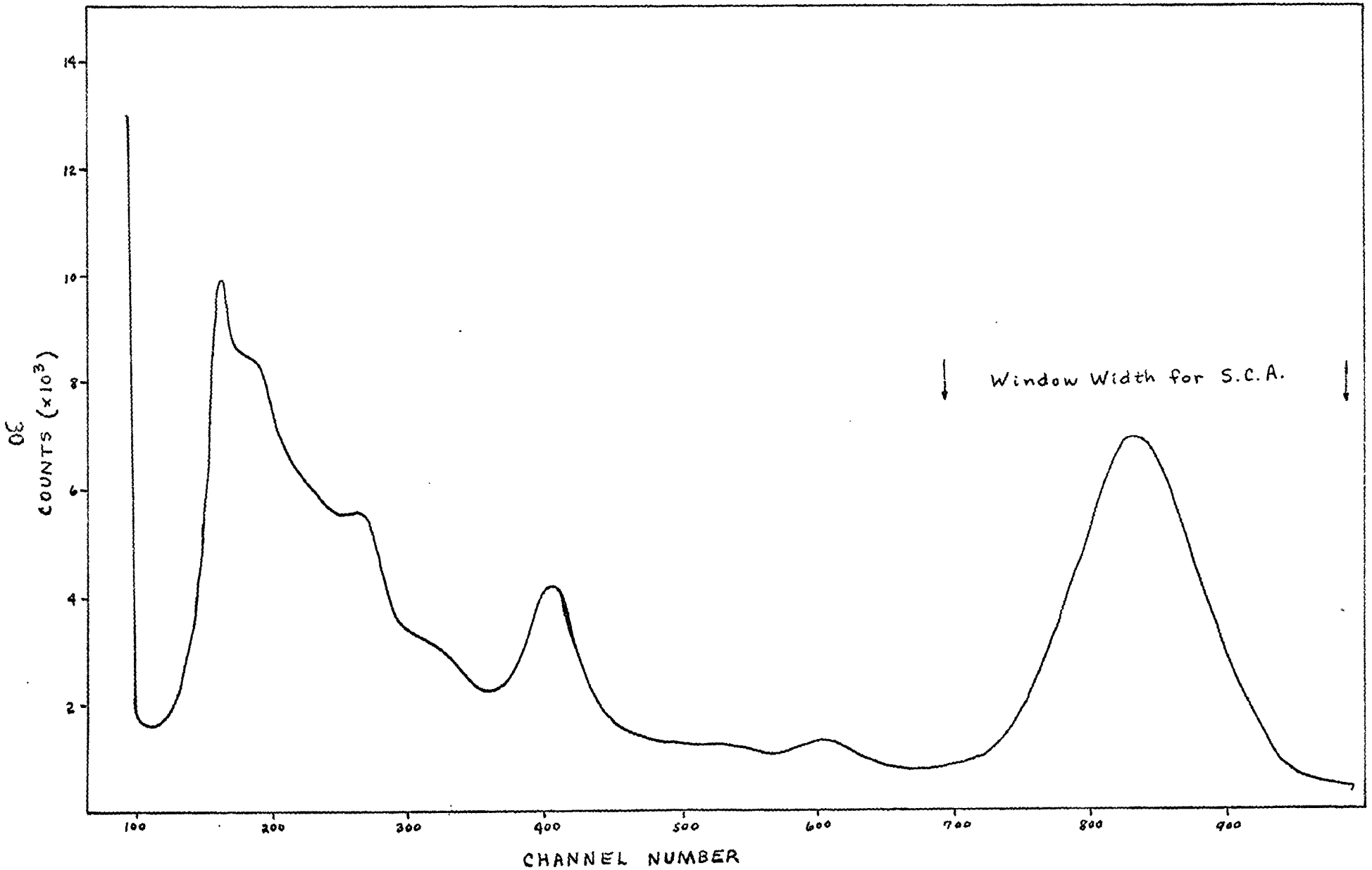

Fig. 7--Alpha spectrum from D-T reaction with 750 keV deuterons

APPENDIX B

```
C    GEORGE H. PEPPER
C    PROGRAM TO CALCULATE MEAN ENERGY OF NEUTRONS IN SAMPLE
C    FORTRAN IV
C    INPUT - FORMAT 30
C      1.  INCREMENT IN RADIUS OF TARGET
C      2.  INCREMENT IN ANGLE OF NEUTRON EMISSION
C      3.  RADIUS OF SAMPLE
C      4.  RADIUS OF TARGET
C      5.  SEPARATION DISTANCE
C      6.  ENERGY OF DEUTERON
C    OUTPUT
C      1.  ANGLE OF EMISSION
C      2.  NEUTRON ENERGY AT THAT ANGLE
C      3.  PERCENT ABUNDANCE OF THAT ENERGY IN THE SAMPLE
C      4.  MEAN ENERGY AND MEAN DEVIATION
      DIMENSION AA(40), E(40), P(40)
      REAL*8 XM1/1876.04467/,XM3/939.525697/,XM4/3728.25439/,
     2Q/17.577/
      REAL*8 E1, A,B,D,SQT,E3,COSD,THETD
   30 FORMAT(6F10.0)
   40 FORMAT(1H1,42X,7HPERCENT)
   50 FORMAT(11X,5HANGLE,10X,6HENERGY,10X,9HABUNDANCE/)
   60 FORMAT(8X,F8.3,9X,F7.3,10X,F7.3)
   70 FORMAT(1X,18HNEUTRON ENERGY  # ,F7.3,2H *,F7.3//)
   80 FORMAT(1X,18HDEUTERON ENERGY # ,F7.3)
    4 READ 30,DR,DTHT,RS,RT,DIST,E1
      PRINT 40
      PRINT 50
      SUMEJ=0
      SUMAJ=0
      DEV=0.
      A=Q+E1*(1.-XM1/XM4)
      B=1.+XM3/XM4
      C=XM1*XM3*E1/XM4/XM4
      DTHET=3.1416*DTHT/180.
      IFIN=RT/DR
      J=1
    6 AJ=0.
      XJ=J
      THETA=XJ*DTHET
      THETD=DBLE(THETA)
      COSD=DCOS(THETD)
      SINT=SIN(THETA)
      COST=COS(THETA)
      RJ=DIST*SINT/COST
      DO 10 I=1,IFIN
      XI=I-1
      RI=DR*(.5+XI)
      IF(RS+RI-RJ)10,8,8
    8 IF(RI+RJ-RS) 7,7,9
    7 PHIJ=2.*3.1416
```

```
      GO TO 11
    9 PHIIJ=2.*ARCOS(((RS*RS)-(RI*RI)-(RJ*RJ))/(2.*RI*RJ))
      IF (RJ-RI)13,11,11
   13 PHIIJ=2.*3.1416-PHIIJ
   11 AJ=2.*3.1416*RI*DR*PHIIJ*SINT*DTHET+AJ
   10 CONTINUE
      SUMAJ=SUMAJ+AJ
      IF(AJ)24,18,14
   14 D=2.*A*B+4.*C*COSD*COSD
      SQT=DSQRT(D*D-4.*B*B*A*A)
      E3=(D+SQT)/(2.*B*B)
      AA(J)=AJ
      E(J)=SNGL(E3)
      J=J+1
      GO TO 6
   18 K=J-1
      DO 20 I=1,K
      XI=I
      THETA=XI*DTHT
      PJ=100.*AA(I)/SUMAJ
      P(I)=PJ
      SUMEJ=E(I)*PJ/100.+SUMEJ
   20 PRINT 60,THETA,E(I),PJ
      DO 22 I=1,K
      DEVT=SUMEJ-E(I)
   22 DEV=DEV+DEVT*DEVT*P(I)/100.
      DEV=SQRT(DEV)
      PRINT 80, E1
      PRINT 70, SUMEJ,DEV
   24 GO TO 4
      END
```

TYPICAL OUTPUT

ANGLE	ENERGY	PERCENT ABUNDANCE
1.000	15.838	0.333
2.000	15.837	0.666
3.000	15.836	0.991
4.000	15.834	1.284
5.000	15.832	1.553
6.000	15.829	1.794
7.000	15.826	2.011
8.000	15.822	2.201
9.000	15.818	2.367
10.000	15.813	2.507
11.000	15.807	2.383
12.000	15.802	2.218
13.000	15.795	2.232
14.000	15.788	2.223
15.000	15.781	2.317
16.000	15.773	2.553
17.000	15.765	2.874
18.000	15.756	3.233
19.000	15.757	3.564
20.000	15.737	3.962
21.000	15.727	4.341
22.000	15.716	4.668
23.000	15.705	5.003
24.000	15.693	5.218
25.000	15.681	5.353
26.000	15.669	5.397
27.000	15.656	5.338
28.000	15.642	5.168
29.000	15.628	4.610
30.000	15.614	4.135
31.000	15.599	3.507
32.000	15.584	2.713
33.000	15.569	1.284

DEUTERON ENERGY = 0.500
NEUTRON ENERGY = 15.718 * 0.075

APPENDIX C

```
C   GEORGE H. PEPPER
C   PROGRAM TO CALCULATE CROSS-SECTION
C   FORTRAN II (AFIT)
  110 FORMAT(///10X28H****************************)
  120 FORMAT(38X28H****************************///)
  130 FORMAT(26XI3,30XI3)
   10 FORMAT(10X23HSAMPLE          CU      10X23HSAMPLE
                                                  CS    /)
   20 FORMAT(10X23HDATE           / /      10X23HDATE
                                                  / /  /)
   30 FORMAT(10X23HTIME             *      10X23HTIME
                                                  *    /)
   40 FORMAT(10X23HACTIVATION            UC10X23HACTIVATION
                                                  UC)
   80 FORMAT(23XF8.3,25XF8.3///)
   90 FORMAT(22X,3I3,24X,3I3)
  100 FORMAT(24X,2I3,27X,2I3)
   50 FORMAT(22X14HCU65(N,2N)CU64,7X16HCS133(N,2N)CS132/)
   60 FORMAT(10X6HENERGY,5X17HCROSS SECTION(MB),5X17HCROSS SE
                                                  CTION(MB)//)
   70 FORMAT(10XF6.2,10XI5,17XI5/)
      READ,E1,B1,ATTN1,XG1,T1,A1,E2,B2,ATTN2,XG2,T2,A2
    2 READ,P1,P2,XM1,XM2,XSEC2,ENERG,I,DELTA,TA,T11,T12,T21,
                                                  T22,K
      VI1=0
      VI2=0
      XL1=0.693/T1
      XL2=0.693/T2
      DO 1 J=1,I
      XJ=J
      READ,VI
      VI1=VI1+VI*EXP(XJ*DELTA*XL1)
    1 VI2=VI2+VI*EXP(XJ*DELTA*XL2)
      VI21=VI2/VI1
      Q1=EXP(-1.0*XL1*T11)-EXP(-1.0*XL1*T12)
      Q2=EXP(-1.0*XL2*T21)-EXP(-1.0*XL2*T22)
      Q21=Q2/Q1
      R1=P1*E1*B1*ATTN1*XG1
      R2=P2*E2*B2*ATTN2*XG2
      R12=R1/R2
      ACTV1=R1*XL1/Q1/3.7/10000.
      ACTV2=R2*XL2/Q2/3.7/10000.
      S21=(1.0-EXP(-1.0*XL2*DELTA))/(1.0-EXP(-1.0*XL1*DELTA))
      T21=T2/T1
      XM21=XM2/XM1
      A12=A1/A2
      TA21=EXP(-1.0*(XL2-XL1)*TA)
      XSEC1=XSEC2*R12*T21*Q21*XM21*A12*S21*TA21*VI21
      MSEC1=XSEC1
      MSEC2=XSEC2
      IF(K)4,3,4
```

```
3 READ,MDA,MON,MYR,MHR,MIN,NCS,NCU
  PRINT 110
  PRINT 120
  PRINT 130,NCU,NCS
  PRINT 10
  PRINT 90,MDA,MON,MYR,MDA,MON,MYR
  PRINT 20
  PRINT 100,MHR,MIN,MHR,MIN
  PRINT 30
  PRINT 40
  PRINT 80,ACTV2,ACTV1
  PRINT 50
  PRINT 60
4 PRINT 70,ENERG,MSEC2,MSEC1
  GO TO 2
  END
```

TYPICAL OUTPUT

SAMPLE	CU2	SAMPLE	CS1
DATE	4/16/69	DATE	4/16/69
TIME	3*39	TIME	3*39
ACTIVATION	174.443UC	ACTIVATION	52.818UC

ENERGY	CU65(N,2n)CU64 CROSS SECTION(MB)	CS133(N,2n)CS132 CROSS SECTION(MB)
16.30	1070	1152

APPENDIX D

INSTRUMENTATION

Accelerator	HIGH VOLTAGE ENGINEERING CORPORATION Model AN2000
Alpha Preamp Power Supply	ORTEC Model 106
Alpha Preamp	ORTEC Model 105
Amplifier	TENNELEC Model TC200
Single Channel Analyzer	ORTEC Model 406A
Delay Amplifier	ORTEC Model 427
Alpha Scaler	COMPUTER MEASUREMENTS COMPANY Model 201C
Gamma Detector	NUCLEAR DIODE Type LG-4.0-5
Gamma Preamp	TENNELEC Model TC130
Multichannel Analyzer	NUCLEAR DATA Model ND150M 1024 Channel Monitor Analyzer
Dosimeter	TEXAS NUCLEAR Model 9120

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