SHORT-LIVED FISSION-PRODUCT GAMMA-RAY SPECTRA FROM THE THERMAL FISSION OF U-235

by

WILLIAM EUGENE KENNEDY, JR.

B.S., Kansas State University, 1973

A MASTER'S THESIS

submitted in partial fulfillment of the

requirements for the degree

MASTER OF SCIENCE

Department of Nuclear Engineering

Kansas State University
Manhattan, Kansas

1975

Approved by:

Major Professor

Table of Contents

Table of Figures

Table of Tables

ILLEGIBLE

THE FOLLOWING DOCUMENT (S) IS ILLEGIBLE DUE TO THE PRINTING ON THE ORIGINAL BEING CUT OFF

ILLEGIBLE

ILLEGIBLE DOCUMENT

THE FOLLOWING DOCUMENT(S) IS OF POOR LEGIBILITY IN THE ORIGINAL

THIS IS THE BEST COPY AVAILABLE

I. Introduction

The expansion of the nuclear power industry in the United States and abroad has generated increased world concern about the diversion of nuclear material from peaceful uses to weapons.[1] In an effort to improve nuclear material accountability, nondestructive assay techniques have become important tools for safeguards monitoring. Active and passive assay methods are becoming more widely used with improvements in detection systems and nuclear instrumentation. The large variety of nuclear fissile material forms has led to a diversity of applications of nondestructive assay procedures. Active assay requires penetrating radiation for both the interrogation source and the detected isotopic signatures. Passive assay relies on well known gamma rays or neutron signatures which vary from one nuclide to another.[1]

A requirement for an accurate and rapid active assay method based on the detection of fission-product gamma rays is the availability of absolute, time-dependent, gamma-ray spectra following fission of the major isotopes. In this investigation short-lived fission-product gamma-ray information, which is important to active assay, is collected by transferring an enriched sample of ^{235}U between interrogation neutrons and a Ge(Li) gamma-ray spectrometer system. To enhance the information collected, a cyclic activation technique is used. Attempts to collect gamma-ray signatures using both ^{252}Cf and TRIGA Mk II reactor thermal column neutrons are reported. The entire data collection process, including timed control of the pneumatic transfer system, is coordinated by a small NOVA minicomputer, which is also used in the reduction of spectral data. The resulting fission-product gamma-ray signatures are compared with previous short-lived data, when possible, and possible fission-product contributors are presented.

II. Review of Previous Work

Nondestructive Assay

With increasing demand being placed on nuclear energy systems in meeting the world's energy needs, the accurate accountability of nuclear materials is of paramount importance in the areas of quality and process control, safeguards, criticality safety, and nuclear materials management.[2] Because nondestructive assay techniques for nuclear material control are novel and diverse, a whole new generation of techniques and instrumentation has evolved. The nondestructive assay techniques currently used are classified as being either passive or active. Passive assay uses the detection of naturally ocurring neutron or photon radiations for isotopic concentration determinations. Active assay uses interrogation radiation of either photons or neutrons to induce fissions in a sample which yields detectable radiation for isotopic concentration determination.[1]

Passive Assay Techniques

The general mathematical procedure for gamma assay is given by the equation[3]

$$M = \frac{CR \cdot CF}{K} \qquad \text{II.1}$$

where M = the mass of the isotope of interest

CR = the measured count rate from the signature isotope

CF = the correction factor for sample attenuation

K = the calibration factor

The factor K is determined by a measurement of a standard source located in the same detector geometry as the sample, with a known mass and a known attenuation correction factor. The measurement of the standard source provides an accurate estimation of the detector efficiency, specific activity of the gamma-ray source of interest, and geometric effects.[3] The major limitation of passive

gamma-ray assay is the attenuation of the gamma-ray peaks and the resulting change in the response which occurs depending upon the distribution of the fissile material within a large sample.[1]

To help eliminate the attenuation problem in passive gamma-ray assay, gamma scanners have been developed which incorporate four independent segments: the scanning mechanisms, collimating slits, detector assembly, and the data acquisition unit. The scanning mechanism allows five degrees of freedom: X, Y, Z, tilt, and rotation, which give a precise determination of the examination area of a sample.[4] The data acquisition units of such scanners are often controlled by a small computer, such as a PDP-11, which speeds the calculation of sample assay. The gamma scanner effectively flattens the gamma ray response from various interior positions, and is used to scan samples whose sizes range from quart bottles to 55-gallon barrels. Thus, the key to gamma-ray detection for passive assay is accurate transmission, with existing gamma scanners having accuracies of a few percent for small, well-defined samples to 10-15% for the large 55-gallon samples in a typical analysis time of 10 minutes.[1] Most gamma scanners use NaI crystals in a battery of about eight detectors with vertical and ajustable horizontal collimation. Except for their cost, Ge(Li) detectors are preferred for passive assay since they have superior resolution.[3] Scanning devices have also been used to determine the burnup of power reactor fuels.[5]

Passive neutron detection has been improved with the development of high-efficiency neutron coincidence detectors for assay of spontaneous fission in ^{240}Pu and ^{238}U. Coincidence counting systems employ 4π, polyethylene-moderated assemblies using BF_3 or ^{3}He detectors. This assay system was first developed for a one-gallon sample size, and later it was modified to assay 55-gallon drums. Practical counting systems have been able to detect 50 g or

less of ^{238}U in spite of the low spontaneous fission rate of ^{238}U. Assays have been done to within the 5% precision limit of chemical sampling for 10 kg samples with an analysis time of about 3 minutes.[1]

Active Assay Techniques

Passive assay methods have been proven useful for most plutonium assay cases, but since the only useful passive signature of ^{235}U has an energy of 185 keV, its value is limited because of the highly attenuating nature of most ^{235}U-bearing samples. This low energy gamma ray is also not distinguishable in cases where the sample is highly radioactive such as spent reactor fuel. For these cases, active interrogation provides a means of quantitative assay. Active interrogation uses an external source of neutrons or photons to cause the sample material to fission, so that observations of the emissions after fission may be used for assay.[1]

Neutron interrogation of samples is the most common method of active interrogation. After neutron interrogation, the detection of either neutrons or delayed gamma rays results in quantative assay of unknown samples. The delayed neutron yield method is based on the fact that there are differences in the group yields and energy spectra of the delayed neutrons between fissioning isotopes.[6,7] By varying the energy of the incident neutrons to use the subthreshold and superthreshold characteristics of various fissioning isotopes, isotopic discrimination can be obtained so that separation of the response of fissile isotopes (^{233}U, ^{235}U, and ^{239}Pu) from fertile isotopes (^{238}U and ^{232}Th) can be made.[1,8] By measuring the delayed-neutron activity after neutron interrogation, the concentrations of the fissile isotopes can be deduced.

Incident neutrons produced by irradiating a deuterium target with a bremsstrahlung beam produced by 4 to 6 MeV electrons from an electron linear accelerator (LINAC) system are used in a transportable assay system. In the pulsed source mode, delayed neutrons are measured by proton-recoil detectors with the prompt neutrons used for a normalization of the induced delayed-neutron yield. The high intensity of the LINAC source and the high efficiency of the neutron detectors more than compensate for the fact that the delayed-neutron yield is two orders of magnitude lower than the prompt neutron yield.[8,9]

Spontaneously fissioning radioactive neutron sources provide another source for interrogation neutrons. Such sources are simple to handle and reliable to use so that they are appealing to use for quality control of reactor fuels as well as process stream and scrap material monitoring. Because of their relatively low gamma-ray heat outputs and high neutron yields ($\approx 2.4 \times 10^{12}$ n/g sec), ^{252}Cf sources have received much attention.[1,10] In a current four-rod fuel assay system using ^{252}Cf neutrons, in-plant data shows that the system is uncomplicated to operate and gives immediate assay results with a precision of 1.1 to 1.5% (1σ) and an effective throughput rate of one rod per minute. The system operates using ^{4}He detectors which detect high energy neutrons. The high energy ^{252}Cf source neutrons which reach the detectors give a constant background rate from which a feedback system automatically stabilizes the system response.[11,12]

The fission-product mass yields depend upon the atomic weight of the fissioning nucleus and significant differences have been reported.[13] Figure II.1 shows the mass-yield curves for the thermal-neutron fission of ^{233}U and ^{239}Pu. For each neutron- or photon-induced fission approximately 14 MeV of gamma-ray radiation is emitted with roughly 8 MeV occurring promptly and 6 MeV occurring over a period of up to years.[9] The assay use of

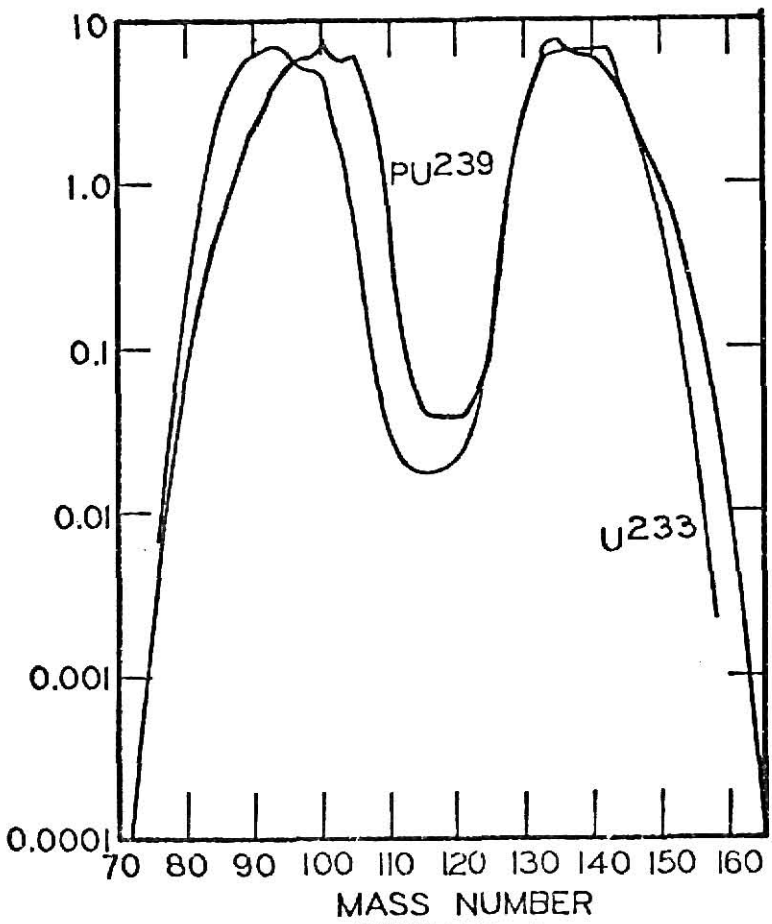


FIGURE II. 1
Mass yield vs mass number
for thermal fission of ^{239}Pu and ^{233}U (13)

fission-product gamma rays has the advantage over neutron assay of removing effects on the data caused by the interrogating neutrons.

A modified ^{252}Cf interrogation system is in operation detecting fission-product gamma rays. The system used both active and passive methods to measure fissile content, as well as pellet-to-pellet composition in the fuel rods.[14] The total fissile content is determined by counting the high-energy delayed gamma rays (>1200 keV) from induced fission with 5" x 5" NaI detectors. The pellet-to-pellet change in enrichment is monitored by detecting the lower energy passive gamma rays from the fuel rods.[15]

Cyclic Activation

The usefulness of short-lived isotope detection for activation analysis depends upon the experimental ability to quickly transfer the sample from a neutron source to a fixed, well-defined geometry at a detection system. By repeating the activation and counting of a sample in a cyclic manner, the sensitivity of assay of short-lived isotopes is enhanced.[16,17,18,19] The total cycle time for one cycle of a cyclic activation (T) is defined as[20]

$$T = t_a + t_d + t_c + t_d' \qquad \text{II.2}$$

where t_a = activation time

t_d = transfer time to detector

t_c = counting time

t_d' = transfer time back to activation site.

The number of counts from one isotope accumulated for one counting period in a cycle is given by the expression[20]

$$R_o = \frac{\varepsilon V \Sigma_a \Phi}{\lambda} \left(1-\exp(-\lambda t_a)\right) \exp(-\lambda t_d)$$

$$\cdot \left(1-\exp(-\lambda t_c)\right) \qquad \text{II.3}$$

where λ = radioactive decay constant

V = sample volume

Σ_a = macroscopic cross section

Φ = neutron flux

ε = detector efficiency

R_o = the total number of detected counts per cycle.

In a cyclic activation scheme, the number of accumulated counts for n cycles is

$$R = R_o G \qquad \text{II.4}$$

where G is the cyclic gain factor defined as

$$G = \left(\frac{1}{1-\exp(-\lambda T)}\right) \cdot \left(n - \frac{\exp(-\lambda T)[1-\exp(-\lambda n T)]}{1-\exp(-\lambda T)}\right) \qquad \text{II.5}$$

and where the total time for n cycles is nT.[20]

For one cycle, the gain factor is equal to 1.0, and for a large number of cycles the gain factor may be approximated by the function

$$G(n,T) \cong \frac{n}{1-\exp(-\lambda T)} - \frac{\exp(-\lambda T)}{[1-\exp(-\lambda T)]^2} \qquad \text{II.6}$$

Thus, once a cycle time T is selected, the gain factor becomes linearly related to the total number of cycles. The advantage of cyclic activation over the single activation and detection method for short-lived isotopes is

that the counts obtained in a gamma-ray peak can be increased without inducing an excessive activity in a sample which causes dead time errors in the detection system. As an example of the importance of this technique, assume n is large and T equals two half-lives, so that for a sample with one short-lived isotope Eq. II.6 becomes[20]

$$G(n) \cong \tfrac{4}{3}\, n - \tfrac{4}{9} \qquad \text{II.7}$$

Figure II.2 shows the increase in detector response using a cyclic scheme versus an activation-counting scheme in which the sample is irradiated for one half the total experiment time and counted for the other.[20]

Short-lived Fission-Product Gamma-ray Spectra

Previous studies of short-lived ^{235}U fission-product gamma rays have been conducted using cyclic activation in attempts to identify isotopic signatures. In 1966 Boggs[21] used a pneumatic rabbit system to transfer a small strip of enriched ^{235}U foil between the thermal column of the Omega West Reactor and a germanium lithium-drifted detector system. For each cycle, eight 200-channel subgroups of spectral information were collected at successively longer times after fission and for successively longer counting periods. The collection of spectra was controlled by scalers and timers which switched the analyzer from subgroup to subgroup. Since only 200 channels were available per spectrum, the energy spread covered was limited. The data showed the gamma-ray activity of energy subgroups for both long and short times after fission. Boggs observed over 200 gamma-ray peaks, about half of which had half-lives less than 20 seconds. Identification of specific isotopes was attempted by computing half-lives for the gamma rays reported and matching these values with tabulated gamma-ray values.

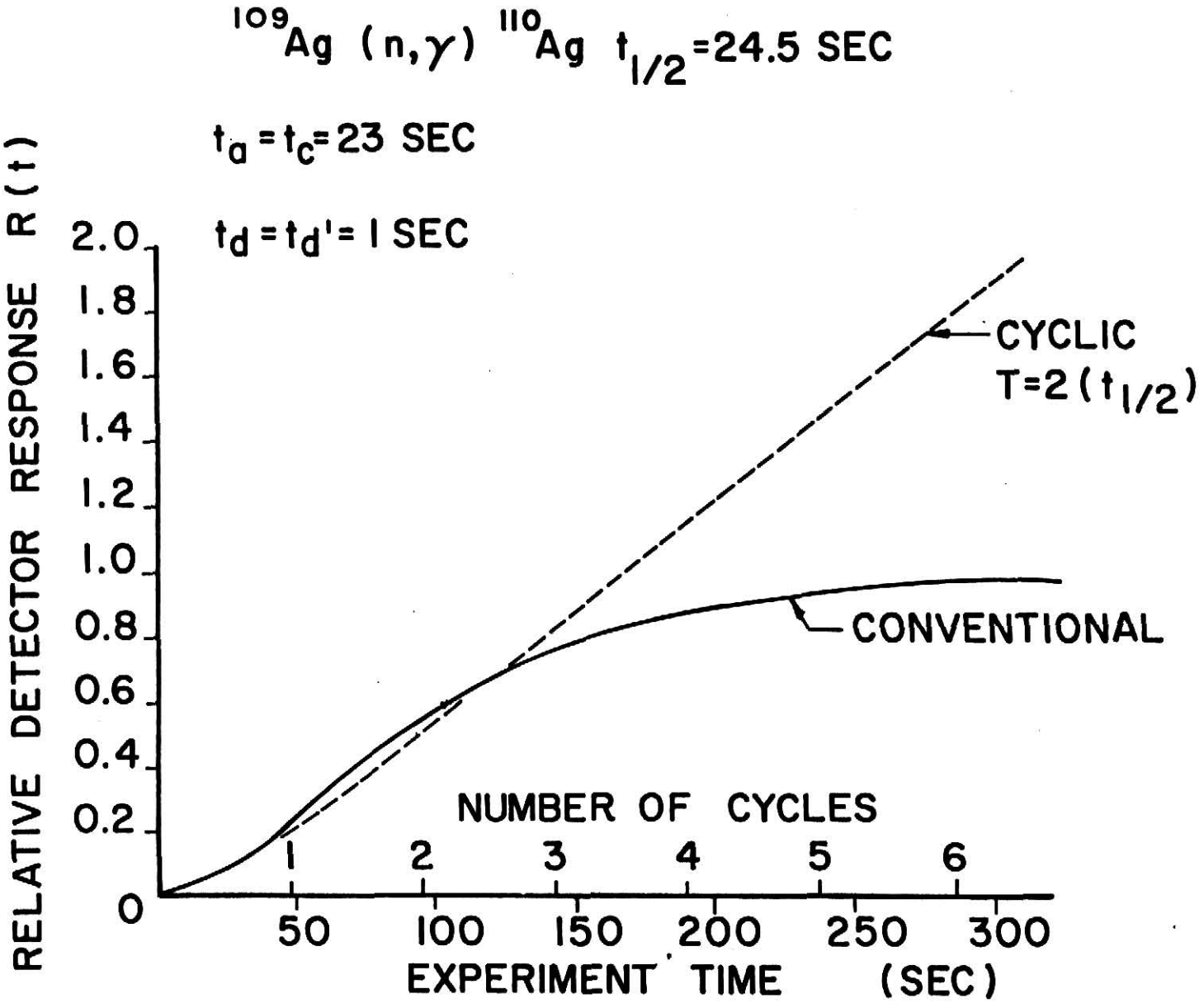


FIGURE II. 2
Detector response from cyclic vs conventional analysis[20]

Twenty-eight fission-product gamma rays with their energies and computed half-lives were assigned to fourteen fission-product isotopes. The results presented did contain more detail than previous spectra, but most of the peaks were unresolved. Boggs recommended that the study be extended to include higher energy peaks with a larger Ge(Li) detector and analyzer system for ^{235}U as well as other fissile isotopes.[21]

In 1971 Atkinson[22] used cyclic activation to extend the detection of short-lived fission-product gamma rays to higher energy and he attempted to use both passive and active assay methods on samples of varying ^{235}U enrichment. Samples of four different enrichments were analyzed by the total-peak-area method for passive assay. Atkinson reported that with his system passive assay was possible provided that standards of similar enrichment to the "unknowns" were available, even though his system was not specifically designed for use in passive assay. He reported a lower limit of detectability of about 130 ppm.[22] Atkinson used a 1 milligram ^{252}Cf source as an interrogation neutron source for active assay and short-lived gamma-ray detection purposes. Cycle control was achieved by a control unit which consisted of a timing circuit of preset scalers, a reference clock, and a valve switching circuit. A total of 17 gamma-ray spectra were collected for active interrogation analysis which exhibited the spectral characteristics of short-lived ^{235}U fission products. As in the Boggs analysis, Atkinson observed many unresolved peaks. While he reported marginal quantitative analysis ability, he was able to detect complex gamma-ray spectra which verified previous work and were composed of gamma-ray peaks unique to short-lived fission products. Atkinson concluded that while cyclic activations using ^{252}Cf appeared promising for assay of fissile materials, the volume of spectral data generated warranted the use of a minicomputer system for cycle control and data reduction.[22]

III. The Experimental Components

Pneumatic Transfer System

The cyclic activation analysis of short-lived ^{235}U fission products required an automated and rapid transfer system. The system needed two terminals: one for neutron activation, and one for gamma-ray spectroscopic analysis. The system needed to be dependable and durable since consistent operation over several hundred cycles was desired. A pneumatic transfer system was chosen as being the most easily constructed, maintained, and operated.

The components of the transfer system included a pair of matched three-way solenoid-operated control valves, transfer tubing, end terminals, gas supply hose, gas pressure regulator, a supply of nitrogen gas, and a particulate filter over the exhaust. The solenoid gas valves used for the transfer system were a matched pair of three-way Skinner Electric Valve Co. ½" orifice, 115 volt, 60 cycle, 10 watt valves, which operated over a pressure range of 5 to 150 psi. These valves were first used with manual switches, and later with electric relays to test their reliability. They gave a consistent switching response up to about fifteen cycles per second. These valves were connected so that when one valve was in the gas inlet position, the other valve was in the exhaust position, and the sample was driven toward the exhaust position valve.

The transfer tubing was seamless polyethylene with a 0.5 inch inside diameter and a 0.05 inch wall thickness. Polyethylene tubing was chosen because of its "firm-wall" nature, low frictional drag with the polyethylene sample capsule, and ease of assembly. The overall length from the neutron source to the detector was about 8 meters. The polyethylene tubing also did not cause excessive activation problems.

The end terminals were made from 0.5 inch inside diameter elbow fittings for common waterline, and were purchased from a hardware store. Inside the elbow fittings, epoxy was inserted, then ground away to form a seat to position

the sample and a gas flow obstruction to channel the gas, ensuring that the sample would transfer smoothly. At the detector, two elbows were fitted together and cemented with epoxy in a groove in a plexiglas detector cover, while at the activation terminal the same style elbows were inserted next to the neutron source. The arrangement of the activation terminal is shown in Figure III.1, and the detector cover plate is shown in Figure III.2. The cover was designed to allow a constant sample geometry over several hundred cycles of operation. The cover plate provided about 0.25 inches of plexiglas shielding for beta radiation and thus reduced the bremsstrahlung interference. Up to about 0.25 inches of lead could be inserted into the cover to absorb some of the lower energy gamma rays emitted from the activated ^{235}U to reduce the analyzer dead time. The lead was not used in all of the data trials, however, and it was not thick enough to stop many of the higher energy gamma rays used in this analysis. A thin (4 mm) foam rubber cushion acted as a shock absorber between the cover and the detector cap to reduce the vibration resulting from the sample transfer to the detector.

The gas supply and exhaust hoses were soft clear-wall plastic. The inside diameter was 0.5 inches and the wall thickness was 5/32 inches.

Nitrogen gas was chosen as being the most readily available and least expensive transfer gas. Helium gas was considered to be the ideal gas since it does not undergo neutron activation, but it is costly and the neutron flux levels were low enough to allow the use of nitrogen without causing excessive activation. The nitrogen was obtained in standard gas bottles pressurized to about 2000 psi. The gas was regulated to a pressure of about 20 psi for the best compromise between gas economy and fast transit time. A block diagram of the complete pneumatic transfer system is shown in Figure III.3.

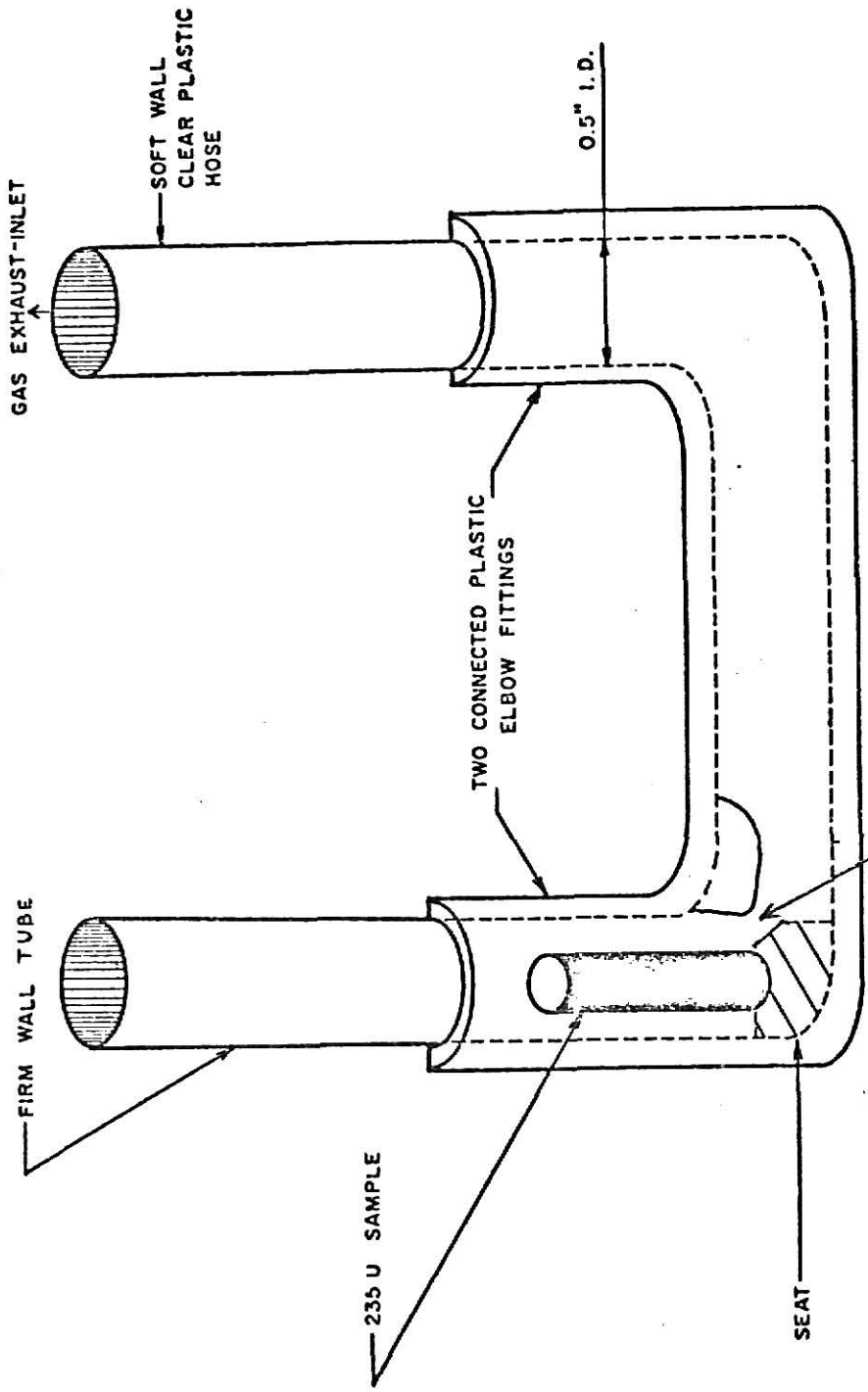


FIGURE III. 1
Activation terminal end fitting

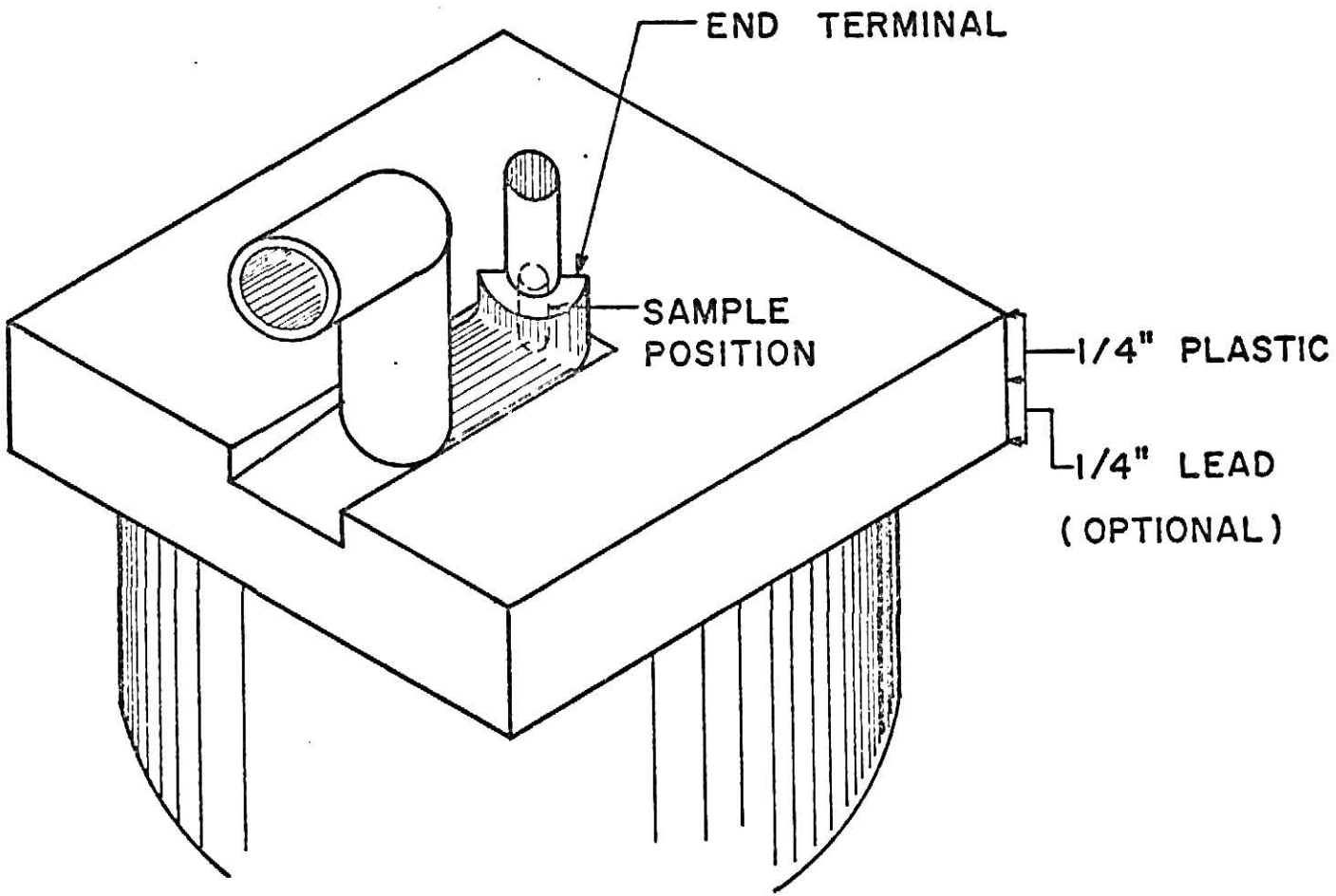


FIGURE III. 2
Detector cover end terminal

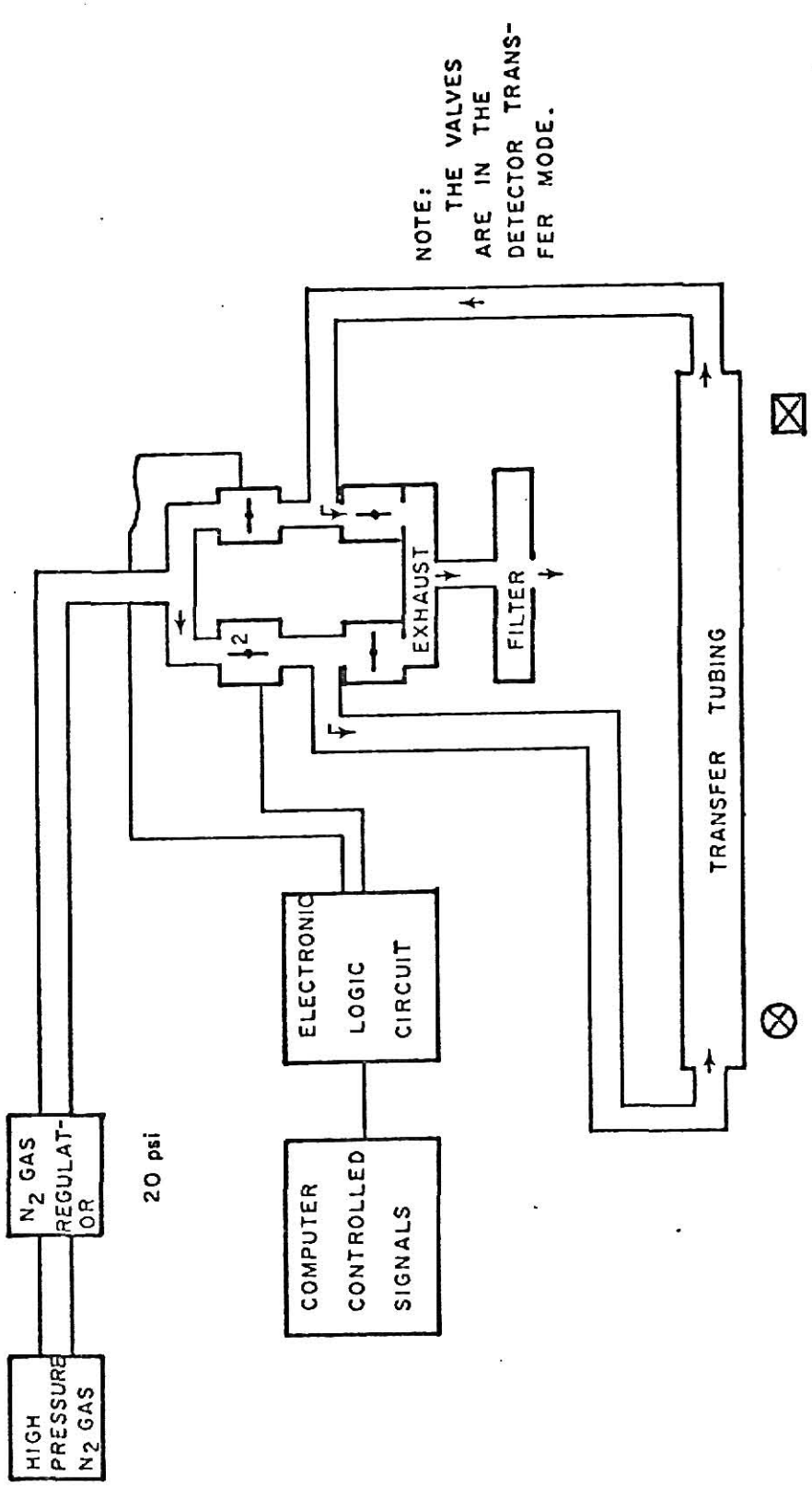


FIGURE III. 3
Block diagram of pneumatic transfer system

To prevent any accidental release of ^{235}U in the event of complete capsule rupture, a high quality two-stage air filter was placed over the exhaust from the transfer system. The filters were obtained from a Kewaunee Mfg. Co. glove box system which was not in use.

Ge(Li) Gamma-Ray Spectrometer

The gamma-ray detector used for this analysis was a Canberra Model 7227 germanium lithium-drifted (Ge(Li)) detector. The crystal had a diameter of 37.5 mm and a length of 22.5 mm and was coaxially drifted. A Canberra Model 970 preamplifier was directly coupled to the Ge(Li) detector. A Fluke Model 415B high voltage power supply provided the detector bias of 1800 volts. An Ortec Model 451 spectroscopy amplifier served as the main amplifier and was coupled to a Northern Scientific Model NS-623 analog-to-digital converter (ADC). The data were transferred to a Northern Scientific Model NS-636 series multichannel analyzer with a total of 4096 channels. Data taken from this unit were then deposited on magnetic tape for later compute analysis. A block diagram of this equipment is shown in Figure III.4.[23]

To reduce the background interference, the detector was housed in a lead shield. The inside walls of the lead shield were no less than four inches from the detector cylinder, and were lined with copper sheet to reduce X-rays produced in the lead by source gamma-rays. A similar shield was constructed in the reactor bay to reduce the background for the analysis using the thermal column neutrons.

Automatic Computer Control

The entire data acquisition and reduction process was automatically controlled by a programmed NOVA computer. The NOVA computer, manufactured by Data General Corporation, was a Model 1220 with an 8K memory and a 16-bit

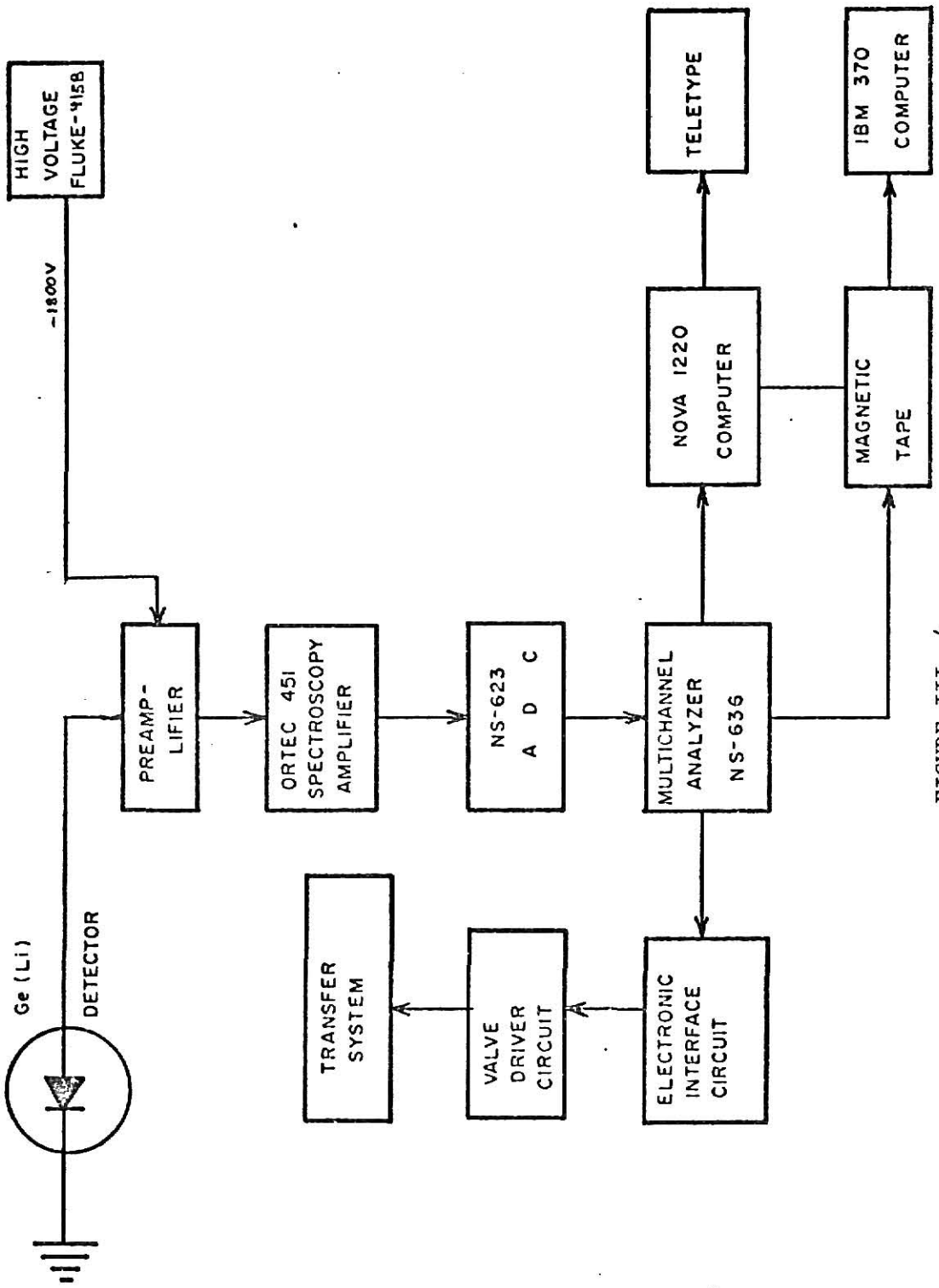


FIGURE III. 4
Block diagram of gamma-ray spectrometer system

word length organized around four accumulators. The available peripheral equipment included teletype, high speed paper tape reader and punch, magnetic tape unit, and an interface to the NS-636 analyzer.[24] By changing the computer program, the programmed NOVA 1220 accurately controlled the activation time, transfer of the sample, spectra collection, data transfer to magnetic tape, and cooling delay of each cycle and was able to identically reproduce the cycle hundreds of times.

After the collection of the spectra on magnetic tape, a NOVA computer program was written to sum the corresponding spectra of the one hundred or more cycles. This was done to produce composite spectra of the corresponding time "windows" of the cycles. The composite spectra were then analyzed for fission-product peaks, and decay half-lives.

Interface Circuit

Control signals were needed for the pneumatic transfer system from the NOVA computer. Since the computer did not have a general-purpose interface for this control signal, it was convenient to use the computer control of the NS-636 analyzer to generate the signal which was sent to an unused output pin on the rear of the analyzer.[25] The plotter enable pin on the rear of the analyzer was selected to furnish signals of either one or two pulses within a few hundred microseconds. An interface circuit was needed between these signals and the three-way solenoid transfer-system valves. The interface had to logically decide the nature of the input signals, and give output signals activating the proper pneumatic valve corresponding to the input. The circuit also had to control the amount of time each pneumatic valve was open.

The plotter enable command from the computer to the NS-636 analyzer gave an output pulse from the NS-636 that rose from 0 to +2 volts and had a duration of about 50 microseconds before falling back to 0 volts. The generated signal was

either a single pulse or a double pulse with a controlled delay between the pulses of the double pulse signal. The electronic interface operated by determining if one or two pulses came in as the input signal.(26)

A simple block diagram of the circuit is shown in Figure III.5. The first input pulse passed through a buffer section which eliminated noise and reshaped the pulse. Next, the pulse was divided to change the output of a flip-flop from the initial $\bar{Q} = 1$ to $Q = 1$ and to fire a long-duration one-shot multivibrator. If a second pulse was received during the output of this one-shot, the state of the flip-flop would be changed back to the initial $\bar{Q} = 1$ state. The falling edge of the pulse from the first one-shot was used to fire a second one-shot of much shorter duration. The logical 1 output from the second one-shot was sent to two NAND gates where it was compared to the $\bar{Q}$ and Q flip-flop outputs. If only one pulse was received as the input signal, then the Q output would have the same logic state as the second one-shot output. This gave the response used to fire a driver circuit connected to the "one pulse" solenoid valve and reset the flip-flop through one-shot number 3. If two pulses were received as the input signal, then the $\bar{Q}$ output would have the same logic state as the second one-shot output. This gave the response which drove the "two pulse" solenoid valve and reset the flip-flop through one-shot number 3. The operation of this circuit had to ensure that the proper output would be given for any sequence of input signals, and that both valves would never be open at the same time.

Since the electronic components for this logic circuit operated from a +5 volts power supply and the solenoid valves operated on 115 volts ac, a +24V dc solenoid relay driven by a transistor switch was used to couple the ac and dc operation.(27) This was done to ensure noise-free operation between the dc and ac components. The design of the solenoid relay driver circuit is shown in Figure III.6.

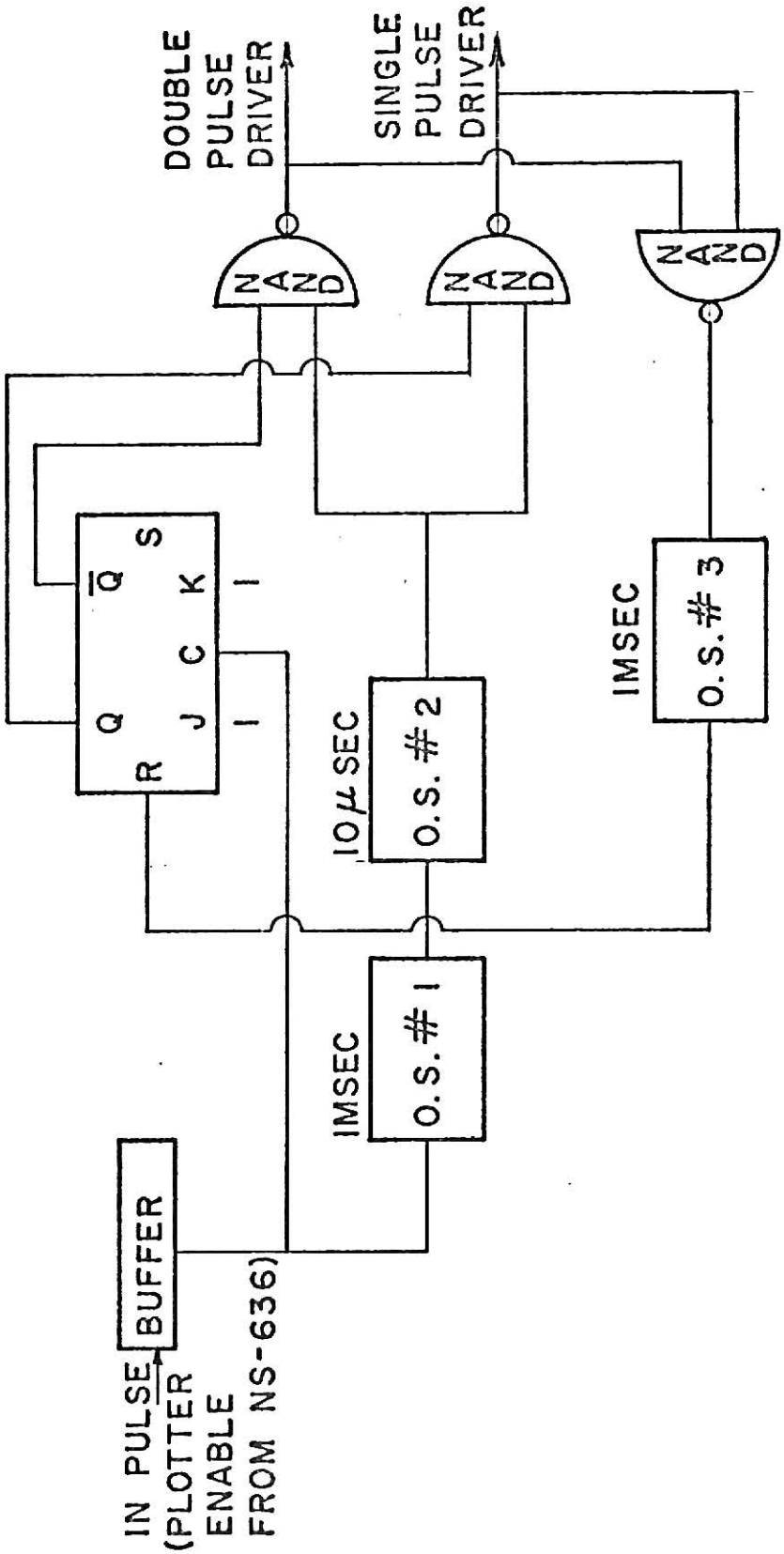


FIGURE III. 5
Block diagram of pneumatic control circuit

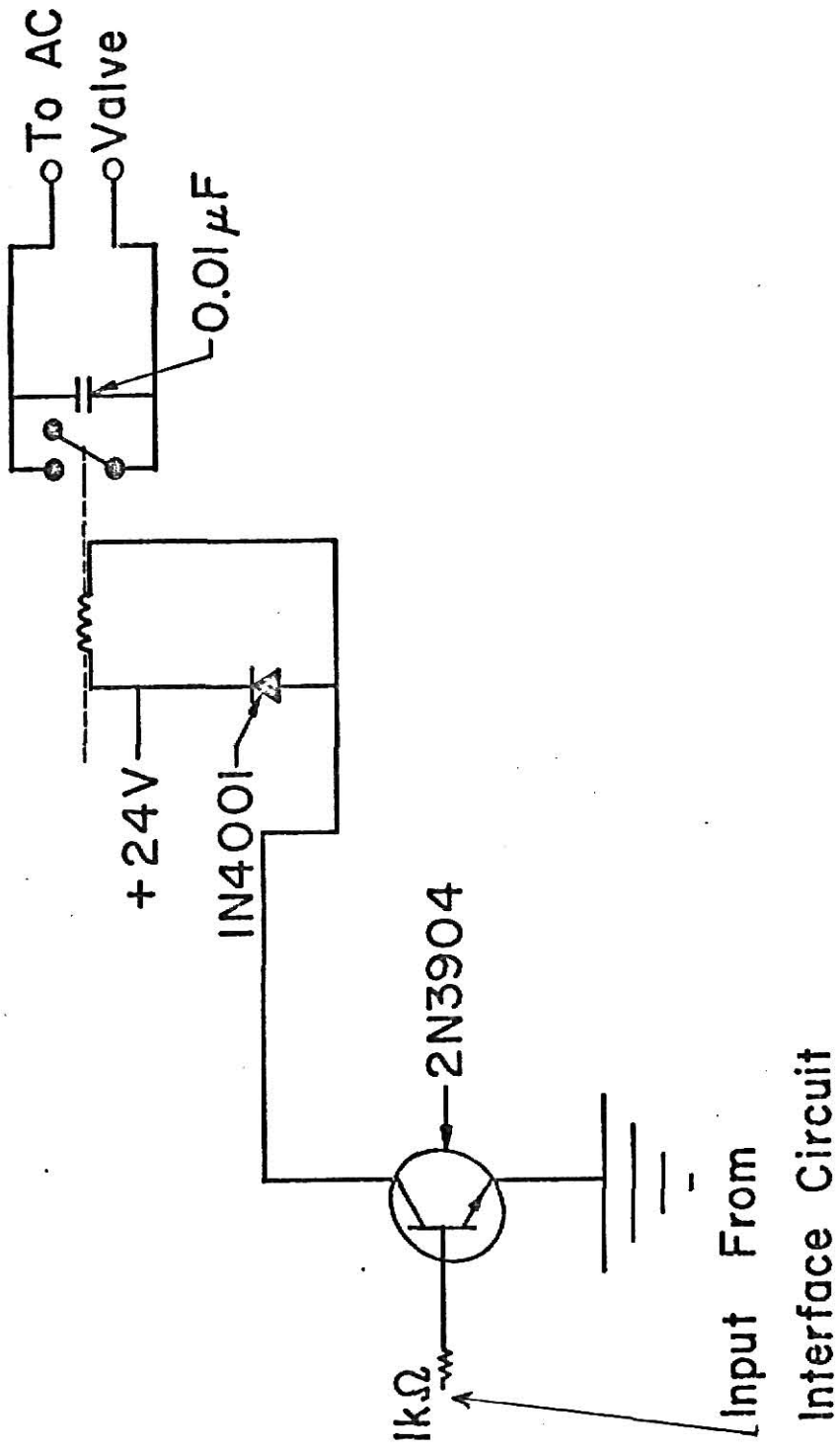


FIGURE III. 6
Solenoid relay driver circuit

The final overall circuit diagram is shown in Figure III.7. Because the pulses from the NS-636 were not very sharp, they were first sent to a pulse generator which gave constant +5 volt, 50 microsecond wide pulses for each input pulse of any amplitude or duration. This extra stage of pulse shaping helped to ensure that properly-shaped, noise-free pulses were received by the interface circuit.

The entire interface circuit was tested extensively with the ac valves connected. Problems with feedback noise from the magnetic fields surrounding the solenoid valves and relays were encountered. Most of these noise problems were cured by housing the circuits in grounded boxes, using shielded signal cables and placing the valves and relays on independent base boards located a few feet away from the interface circuit. With these modifications, the entire transfer unit operated successfully over the hundreds of trials required to acquire the data for this analysis.

Uranium-235 Sample

The analysis of short-lived fission-product gamma rays from the fission of ^{235}U was conducted on a sample containing 2.1047 ± 0.0001 g of 93.11% enriched ^{235}U. The enriched ^{235}U was obtained from Aerojet Nuclear Corporation at Idaho Falls, Idaho, through a grant from the Atomic Energy Commission, Department of Nuclear Education and Training. The sample was contained in a tertiary capsule; the outer layer was a sealed polyethylene vial, the second layer was a machined plexiglas inner vial with a sealed lid, and the third layer was a region of paraffin molded around the metallic ^{235}U strips. The total capsule weight was 4.2735 ± .0001 g . The sample assembly is shown in Figure III.8.

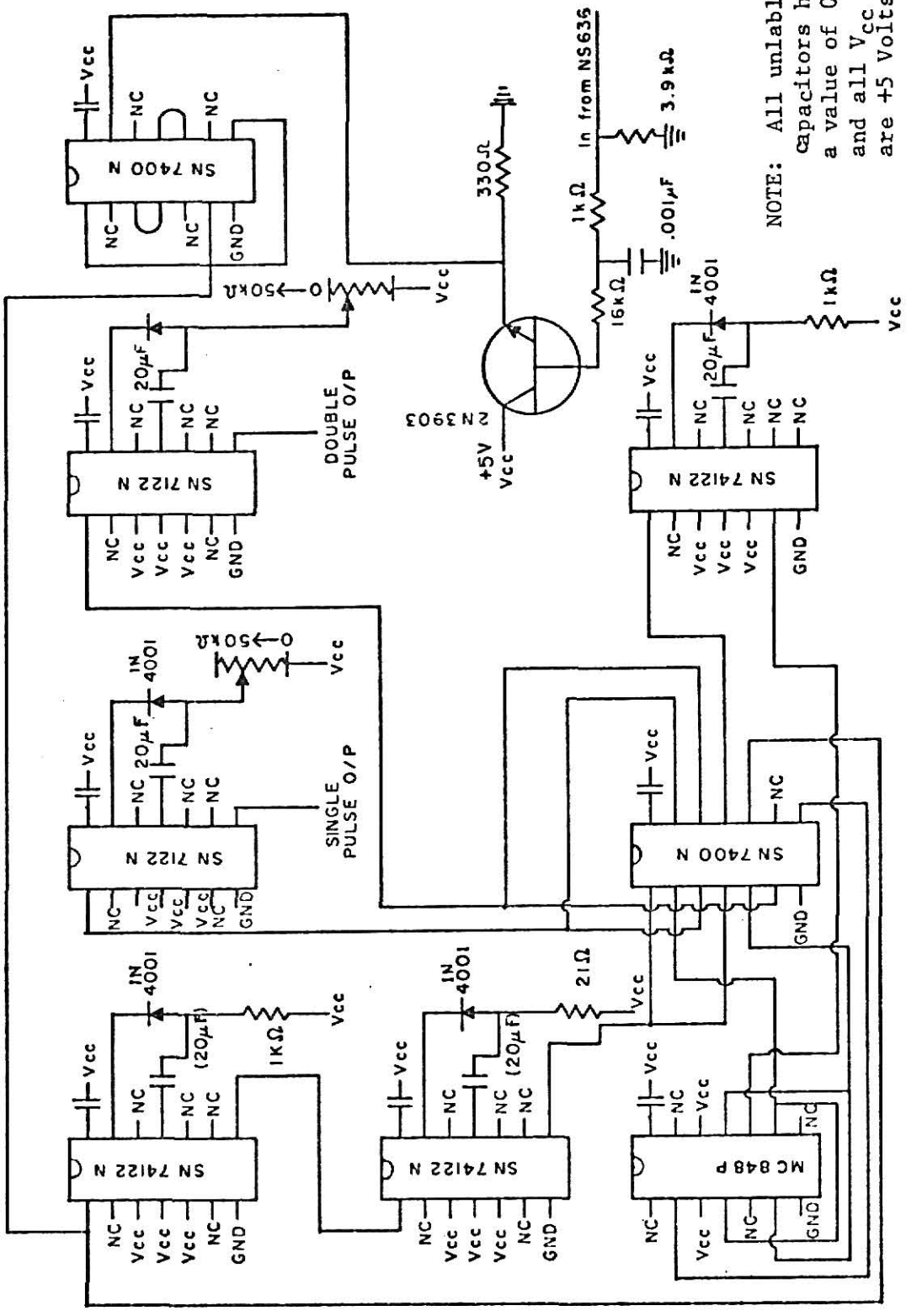


FIGURE III. 7
Interface circuit block diagram

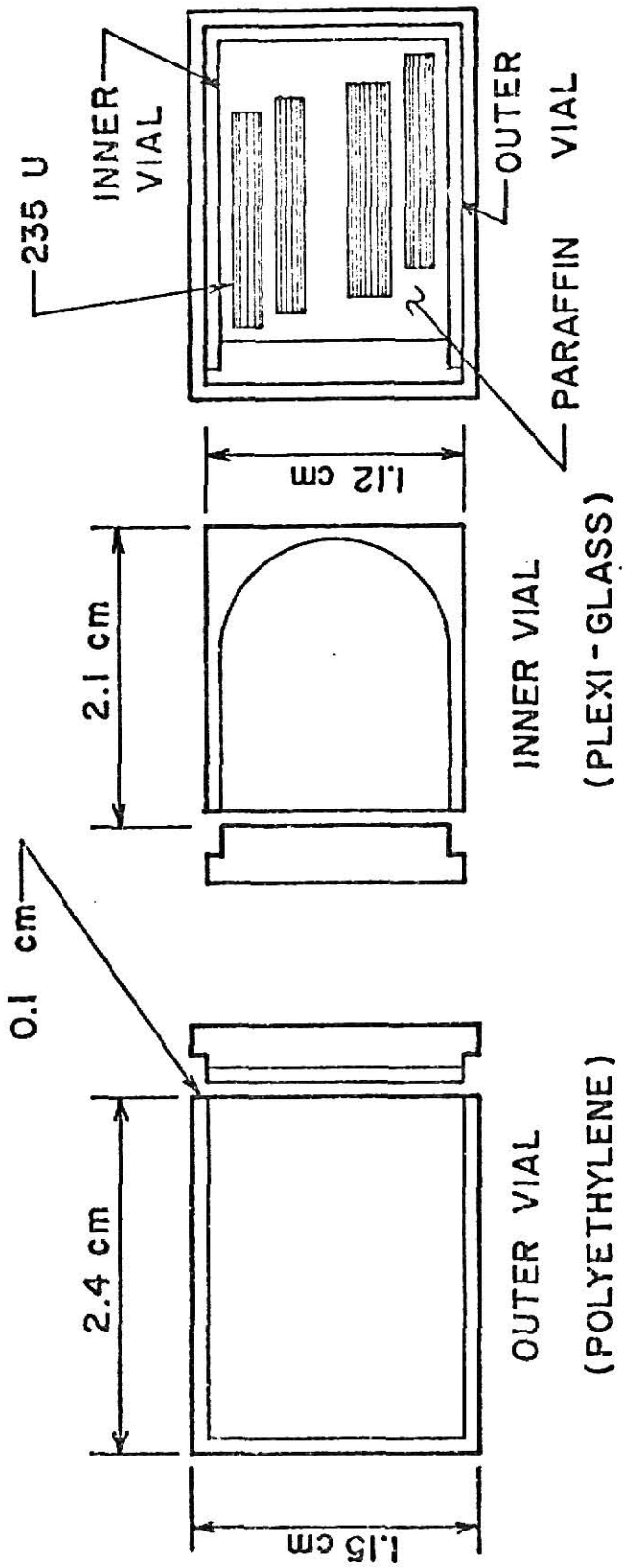


FIGURE III. 8
^{235}U sample assembly

The polyethylene and plexiglas vials were washed with soapy water and rinsed several times; then they were rinsed in dilute nitric acid followed by distilled water. After the vials were air dried, the ^{235}U strips were placed in the inner vial and their mass was determined on a Mettler balance. Next, molten paraffin was poured into the vial to fill any air spaces that would permit the strips to wear against each other during the experiment. The top of the plexiglas vial was sealed with methylethylketone after the paraffin had solidified. Next, the plexiglas vial was inserted into the polyethylene vial so that its lid end fit against the base end of the polyethylene vial. The top of the polyethylene vial was sealed and the hinge tabs were removed.

^{252}Cf Source

The ^{252}Cf source used in the first part of this analysis was a 5.295 microgram sample and was obtained from the USAEC Savannah River Nuclear Engineering and Materials Section.[28] The source strength on March 16, 1973 was specified by the Savannah River Laboratory as 12.237×10^6 neutrons per second with a standard deviation of 3.0%. Since ^{252}Cf has a half-life of 2.65 years, the approximate source strength at the time of this analysis was 7.10×10^6 neutrons per second.

The ^{252}Cf source had a tertiary capsule to protect against any leakage during normal handling.[28] The source was fabricated by the uniform electrodepositing of californium hydroxide on a 90% platinum-10% iridium cathode, and was of the short afterloading cell type (SALC). The cathode was heated so that Cf_2O_3 formed to increase the adherence of the deposit. The primary capsule was sealed in two 90% platinum-10% iridium cells with a "Braze 560" process. The inner capsule had a wall thickness of 0.20 ± 0.012 mm while the outer capsule had a wall thickness of 0.5 ± 0.012 mm. The tertiary container was made of stainless steel with an external diameter of 4.76 ± 0.15 mm, an external length

of 48.50 ± 0.50 mm, and an active length of 15.00 ± 0.50 mm. The tertiary capsule was sealed by a "TIG Fusion Weld". At the top end of the source capsule there was an "eye" fitting (or hole) through which a stranded steel wire about 0.5 meters long was inserted. This wire permitted ease of handling between the shipping container and the moderation-shielding assembly used in this analysis.[28]

The ^{252}Cf source has an effective half-life of 2.646 ± 0.004 years, and an alpha decay half-life of 2.731 ± 0.007 years. The average neutron energy was 2.348 MeV, and the number of neutrons per spontaneous fission is 3.76. The neutron energy spectrum of ^{252}Cf is shown in Figure III.9.[10]

When not in use in the moderation-shielding assembly, the ^{252}Cf source was stored in the KSU TRIGA Mk II reactor bay area in its locked shipping container, which was a steel drum filled with water-extended polyethylene to absorb most of the emitted neutrons.

^{252}Cf Moderation-Shielding Assembly

To ensure as small as possible neutron and gamma-ray doses to both the KSU neutron activation analysis (NAA) laboratory and operating personnel, a moderation-shielding assembly was constructed.[29] The assembly consisted of borated paraffin blocks molded in wooden boxes which measured 16 x 16 x 31.5 cm^3, and smaller boxes of Boraxo which measured 4 x 19 x 12.5 $cm.^3$ The blocks were arranged around a central block of non-borated paraffin cut as shown in Figure III.10, which acted as the activation terminal for the ^{235}U sample. The design of this block was such that the neutron flux at the sample position would be maximum. The approximate location of the shielding assembly with respect to the other components in the laboratory is shown in Figure III.11.

The dose rates of the ^{252}Cf during its use were of interest since it was desired to safely use the source at all times. For this reason, the neutron

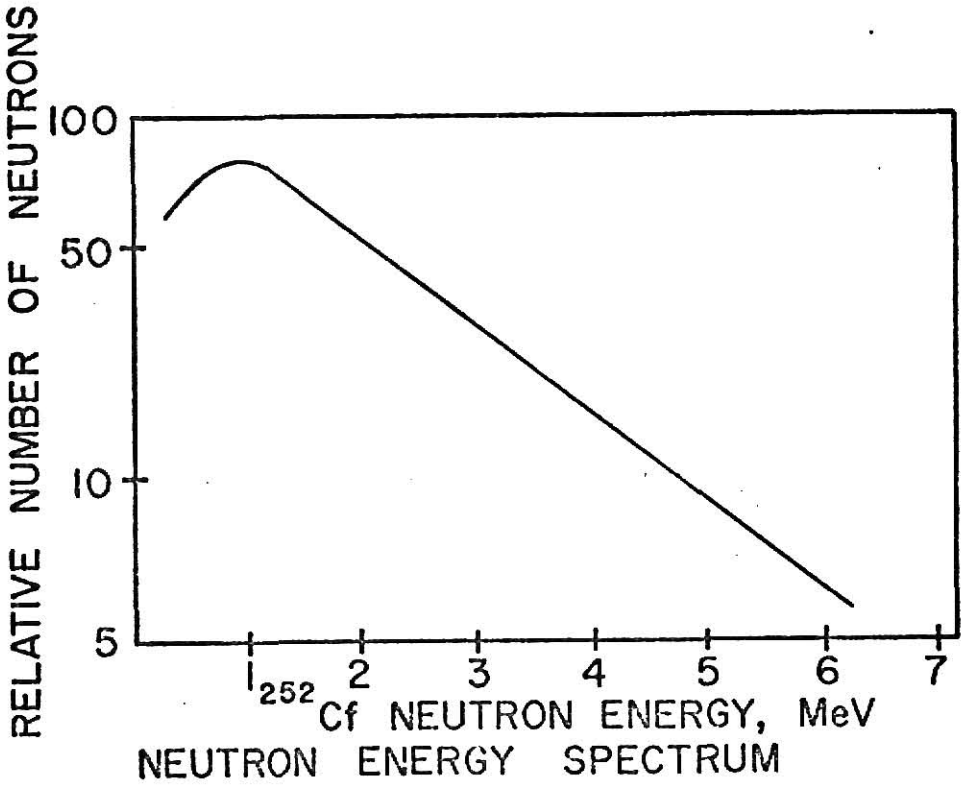


FIGURE III. 9
^{252}Cf neutron energy spectrum (10)

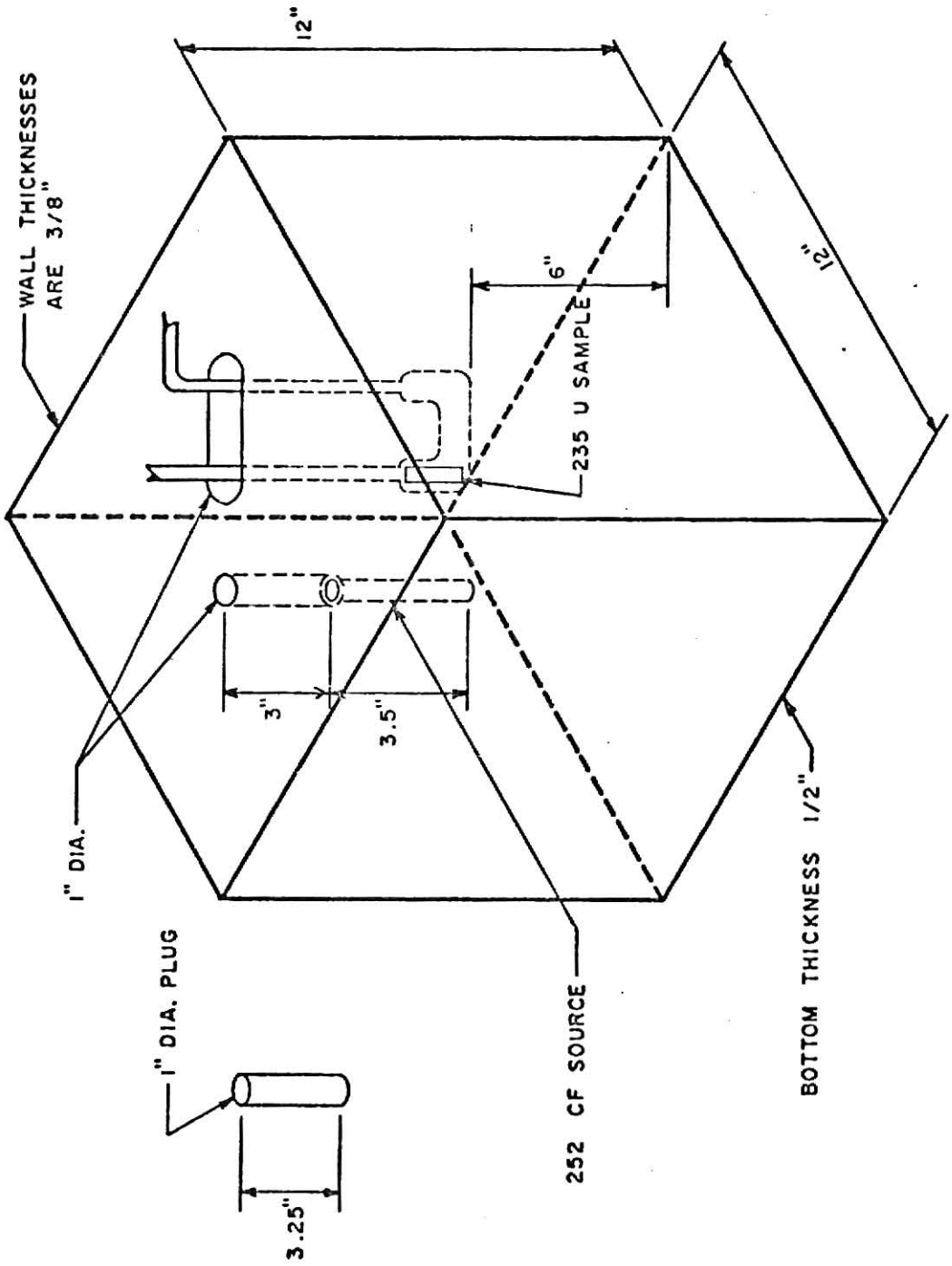


FIGURE III. 10
^{252}Cf activation terminal paraffin block

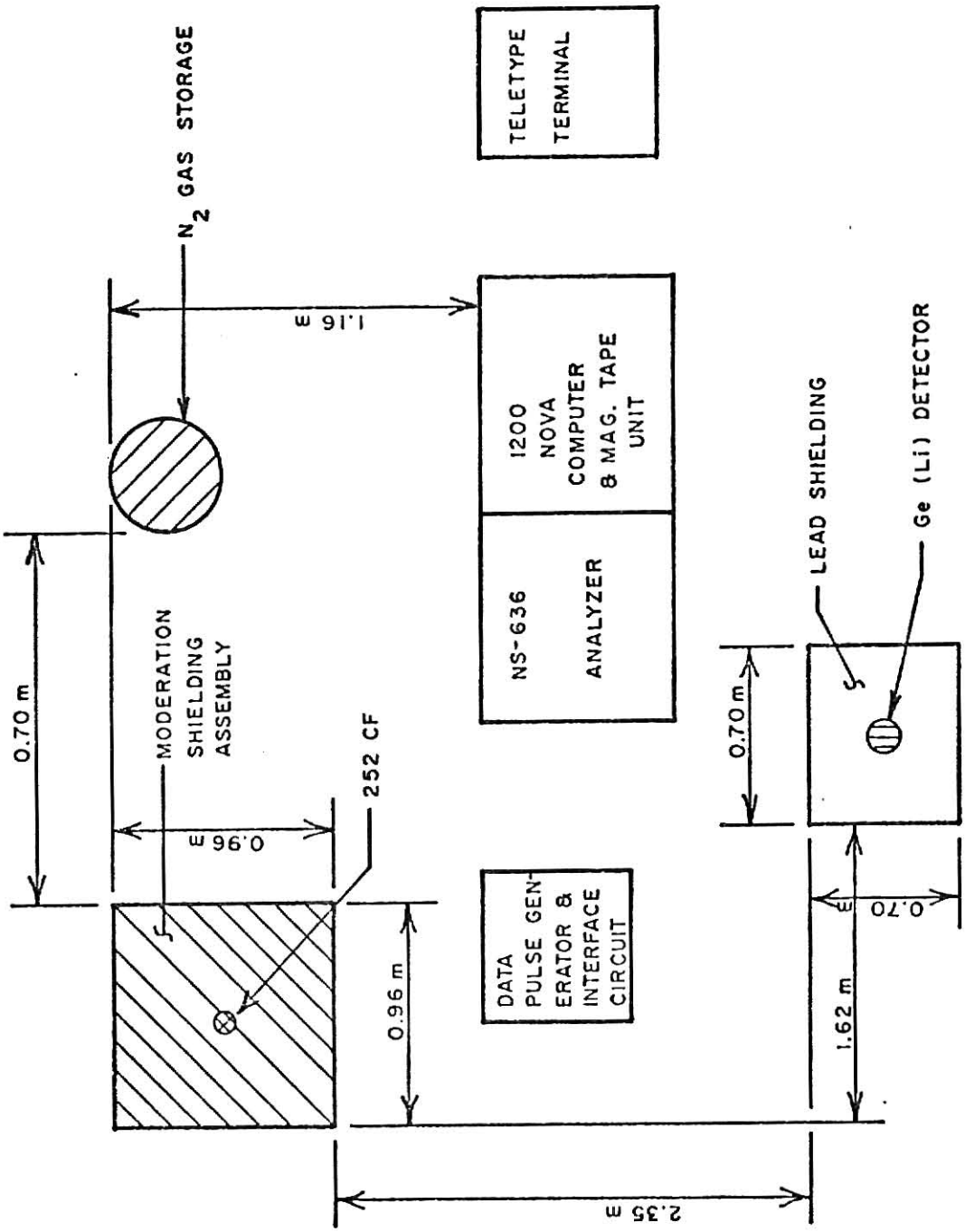


FIGURE III. 11
KSU Neutron Activation Analysis Laboratory; general component locations

and gamma-ray doses were monitored for both the ^{252}Cf in its shipping container, and for the ^{252}Cf in the moderation-shielding assembly. The results are shown in Table III.1.

Table III.1 Dose Rates for the ^{252}Cf source

		Dose Rate Side Surface (mrem/h)	Dose Rate Top Surface (mrem/h)
Shipping Container	Neutron	8	6
	Gamma	10	3
Shielding Assembly	Neutron	0.35	0.70
	Gamma	0	1.0

The neutron dose rates were measured with a Texas Nuclear Model No. 9146 monitor, and the gamma-ray dose rates were measured with an Eberline Instrument Corporation Model E120G survey meter.

The neutron and gamma-ray dose rates in the room were also of importance and were monitored. At a distance of about 4 meters from the shielding assembly near the Ge(Li) detector, the neutron dose was less than about 0.01 mrem/h, and the gamma-ray dose was not measurable. This result showed that there was no significant increase in the background in the room which might affect the analysis.

KSU TRIGA Mk II Research Reactor

To obtain a higher neutron flux to give more intensity to the fission-product gamma rays, neutrons from the thermal column of the KSU TRIGA Mk II research reactor were used. The detector, analyzer, computer, and pneumatic transfer system were moved into the reactor bay. The activation terminal of the transfer system was fully inserted into the central stringer position of the thermal column where the neutron flux was reported as 10.4×10^4 n/cm^2 sec watt by

G. D. Bouchey.[30] By changing the reactor power and activation time, various activation schemes were achieved. The location of the activation terminal is shown in a vertical cross section of the reactor in Figure III.12,[31] and the support equipment locations are shown in a horizontal cross section of the reactor in Figure III.13.[31] An experimental description was written for and approved by the KSU Reactor Safeguards Committee and is included in Appendix A.

The activation terminal and transfer tube were fully inserted into the central stringer of the thermal column. Access was provided to this location by removing the stringer access plug which was made from a heavy-aggregate-concrete-filled steel tube. Shielding around the open access was provided to ensure that radiation levels in the reactor bay were below the limitations specified in 10CFR20.[32]

The transfer tube was bent upward and to the left to prevent any streaming of radiation and about 10 cm of lead and 0.5 meters of concrete were stacked in front of the access hole. A radiation survey in the bay area was taken at low reactor power with the access plug removed and the shielding in place as it was during the collection of data. No excessive leakage of neutrons or gamma rays was observed. Background spectra at power levels of 0.1, 1.0, 10, and 100 watts were recorded to determine how much change in background to expect during the analysis. At these low levels, the background change with power was insignificant at the Ge(Li) detector location. The change in background as recorded with the Ge(Li) spectrometer between the neutron activation analysis laboratory and the reactor bay was next determined by comparing the summation of counts per channel in the low energy region of the spectra between the two locations. The ratio of these summed counts was 4.39; i.e., the background was 4.39 times higher in the bay than in the laboratory.

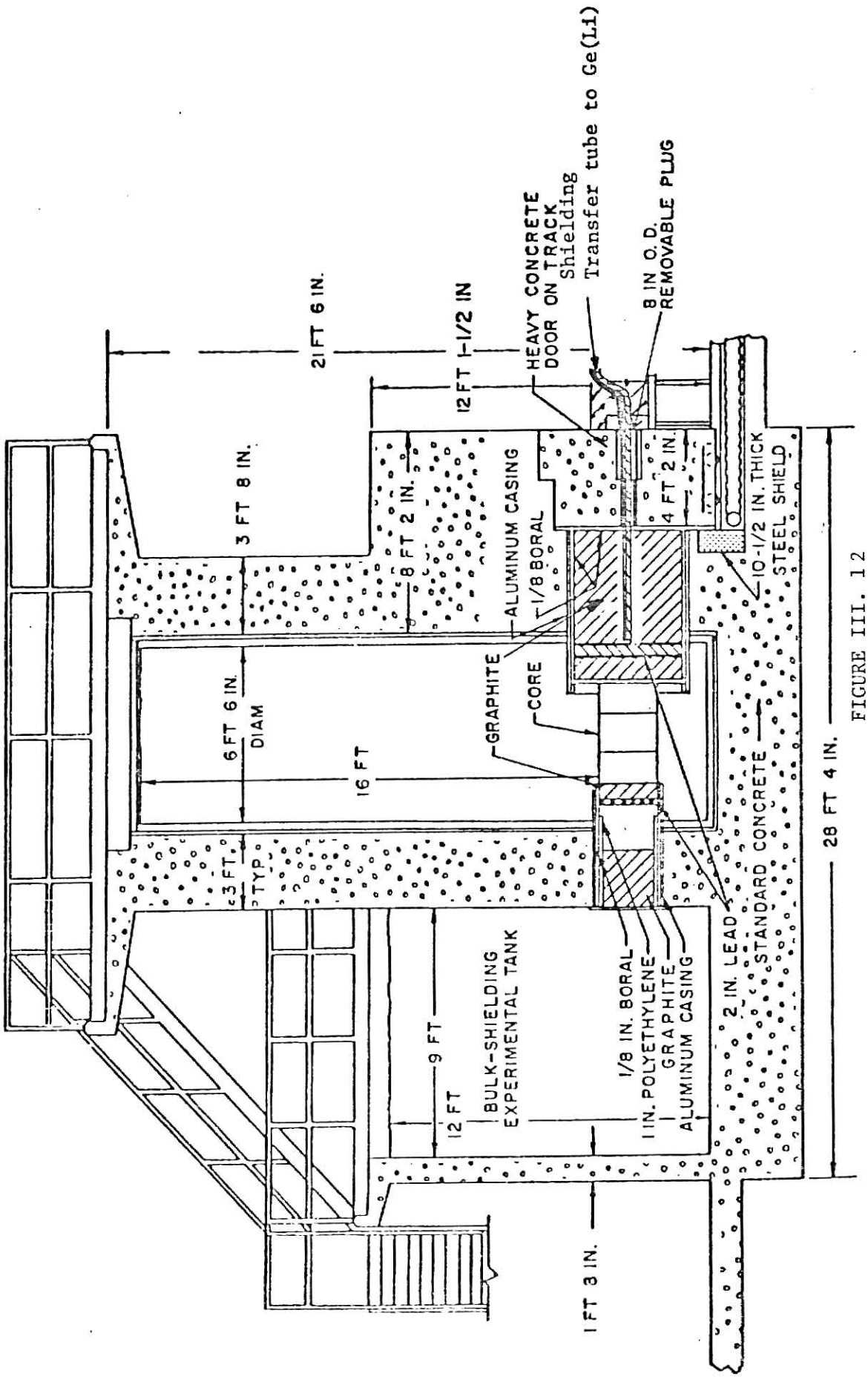


FIGURE III. 12
Vertical cross section of reactor

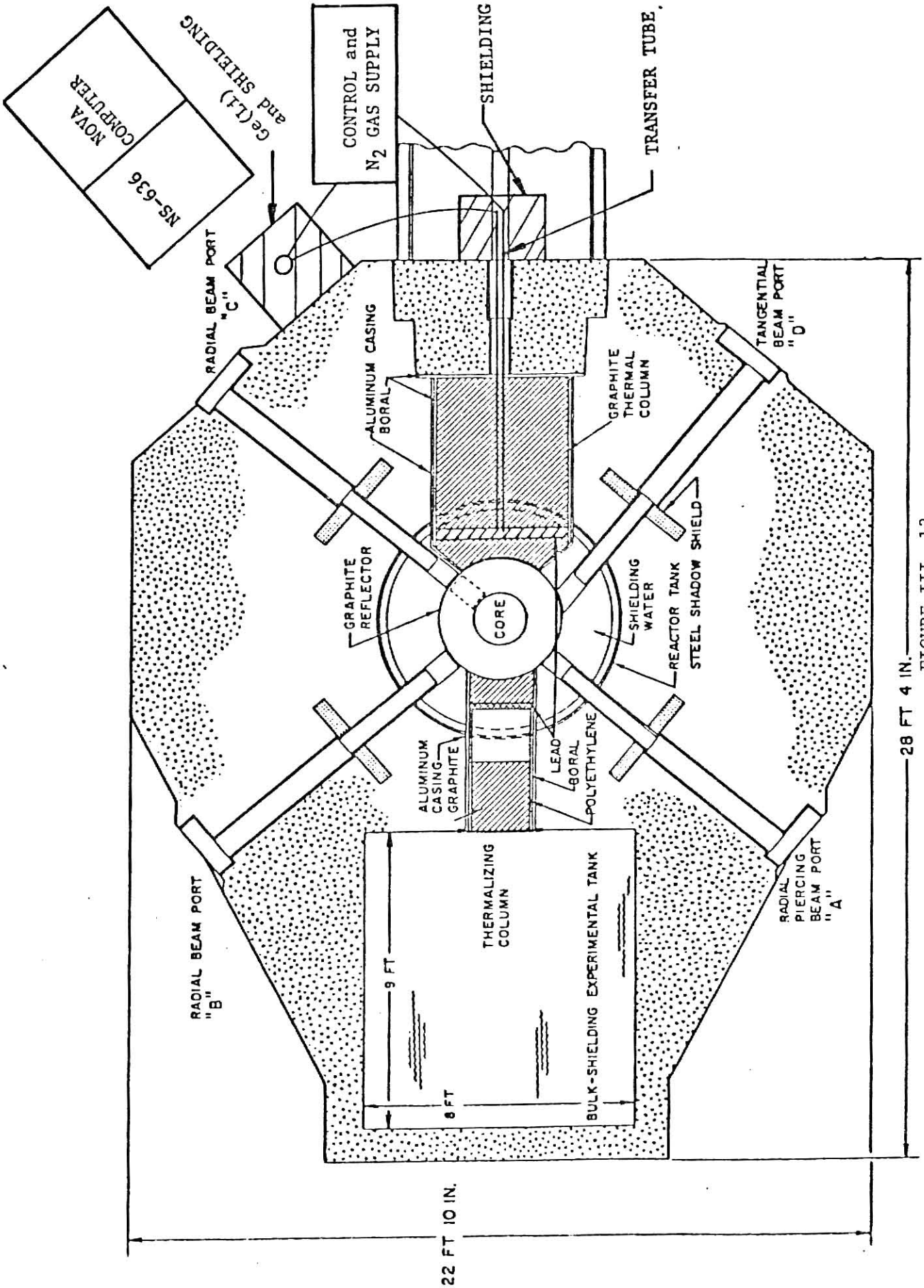


FIGURE III. 13
Horizontal cross section of reactor

IV. Experimental Procedure

Cf-252 Measurements

The analysis of short-lived fission-product gamma rays from ^{235}U was first attempted using neutrons from the 5 microgram ^{252}Cf source. The equipment settings used for the ^{252}Cf analysis are shown in Appendix B.

Each time data was taken, the ^{252}Cf source was moved from its storage location in the KSU TRIGA reactor facility bay into the neutron activation analysis (NAA) Laboratory. A safety procedure for the movement and use of the ^{252}Cf source was written for and approved by the KSU Radiation Safety Officer, and is included as Appendix A. It was removed from its shipping container and carried about two meters to the paraffin activation terminal block in the moderation-shielding assembly. Borated paraffin blocks and Boraxo boxes were stacked around the source in the assembly, and the neutron and gamma-ray surface doses were determined to ensure that there would be no streaming of radiation into the room.

Before the collection of data began, a calibration source spectrum, a spectrum of the ^{235}U sample with the calibration source, and a spectrum of the ^{235}U sample alone were taken so the energy calibration and background level could later be determined. The activation, data transfer, and cooling times were inserted into the computer program, and the computer was programmed to control the collection of data. For the data considered here, the sample was activated for 10 seconds, and three spectra were collected for each cycle at times of 0.3 to 3.3, 6.3 to 9.3, and 19.3 to 23.3 seconds after fission. The transit time for the sample in the NAA Laboratory was 0.3 seconds, and 200 cycles of data were collected. After the data were collected, the ^{252}Cf source was removed from the room in its locked shipping container, and was taken back to the reactor bay.

The data were next compiled by the NOVA program which summed the corresponding spectra of each cycle. Composite spectra of 200 cycles were then deposited on magnetic tape. Table IV.1 shows the spectra generated and their locations on magnetic tape.

Table IV.1 Tagword Assignments for ^{252}Cf Activations

Tagword	Spectrum
0002	120 sec Ccalibration source
0003	500 sec ^{235}U and calibration source
0004	500 sec ^{235}U sample before analysis
0005	0.3 to 3.3 sec summed over 200 cycles
0006	6.3 to 9.3 sec summed over 200 cycles
0007	19.3 to 22.3 sec summed over 200 cycles

KSU TRIGA Mk II Reactor Measurements

The analysis of the short-lived fission products of ^{235}U using neutrons from the KSU TRIGA Mk II reactor required the movement of the detector, analyzer, NOVA computer, and other experimental equipment into the reactor bay. After the equipment was moved and adjusted, a preliminary data trial was run to determine the intensity of the gamma-ray peaks at a reactor power of 10 watts. The equipment settings were the same as for the ^{252}Cf data trial. Two spectra were collected for one hundred cycles at 0.5 to 3.5 and 26.5 to 29.5 seconds after transfer. The activation time was for two seconds. The sample transit time for the system arrangement in the reactor bay was 0.5 seconds.

After the initial test run, three data acquisition trials were run. The first trial was at a reactor power of 10 watts with an activation time of 10 seconds. Spectra at various two second time windows were collected. The sample

had a total analysis and cooling time of 55 seconds per cycle. Since the magnetic tape would only hold about 600 spectra, a total of six spectra per cycle could be taken for 100 cycles to fill one tape. To gain further information at this power level and activation, a second 100 cycle series was run. A spectrum was collected common to both so that a normalizing factor could be found to correct for any intensity changes between the two runs. Table IV.2 shows a list of the spectra collected for this trial, and their locations on magnetic tape.

Table IV.2 Tagword Assignments for Trial 1

Tagword	Spectrum
0001	200 sec ^{235}U before analysis
0002	200 sec ^{235}U and calibration source
0003	2 to 4 sec summed over 100 cycles
0004	7 to 9 sec summed over 100 cycles
0005	12 to 14 sec summed over 100 cycles
0006	17 to 19 sec summed over 100 cycles
0007	50 to 52 sec summed over 100 cycles
Second Data:	
0004	0.5 to 2.5 sec summed over 100 cycles
0005	5.5 to 7.5 sec summed over 100 cycles
0006	12 to 14 sec summed over 100 cycles
0007	50 to 52 sec summed over 100 cycles

The second trial used a higher power and a longer activation time to saturate longer-lived fission products than the first trial. The activation time was 30 seconds at a power level of 20 watts. Spectra were collected at about 10 second intervals, so that the total data aquisition and cooling time per cycle was about 4 minutes and 15 seconds. Including the activation time,

the total cycle time was about 4 minutes and 45 seconds. Because of this long cycle time, only 75 cycles were run. Table IV.3 shows the spectral information for this trial.

Table IV.3 Tagword Assignments For Trial 2

Tagword	Spectrum
0001	200 sec ^{235}U and calibration source before activation
0002	12 to 14.5 sec summed over 75 cycles
0003	23 to 27 sec summed over 75 cycles
0004	33.5 to 37.5 sec summed over 75 cycles
0005	43.5 to 48.5 sec summed over 75 cycles
0006	54.0 to 60.0 sec summed over 75 cycles
0007	230 to 255 sec summed over 75 cycles

The third trial was run to investigate higher energy gamma-ray peaks. To accomplish this, the analyzer settings were changed so that the higher energy information was included on the spectrum. The equipment settings for the third reactor trial are listed in Appendix B.

For the third trial, a reactor power of 100 watts with an activation time of two seconds was used. Two spectra were collected, and no cooling delay was allowed, so that each cycle was only about 19 seconds long. A total of 600 cycles of this nature were collected on two magnetic tapes and later summed to give composite spectra. Table IV.4 shows the spectral information for trial 3.

Table IV.4 Tagword Assignments for Trial 3

Tagword	Spectrum
0001	200 sec ^{235}U and calibration source before analysis
0002	0.5 to 5.5 sec summed over 600 cycles
0003	8.5 to 13.5 sec summed over 600 cycles

V. Computer Operations

NOVA Computer Control

For the cyclic activation analysis of short-lived fission product gamma rays, several unique problems were encountered. The ^{235}U sample had to be transferred between the activation site and the Ge(Li) detector and back again for hundreds of cycles with identical transit times and positions of the sample. Data had to be collected in the same time windows for these cycles and deposited on magnetic tape for later analysis. The sample activation and cooling times had to remain identical for each cycle. The system control had to allow variations which were made between experimental trials. Figure V.1 shows the desired time sequence and control for an example cycle.

This control was achieved by writing assembly-language subroutines for the pulse height analysis program NSI-73-43-01, which was written by Tracor Northern[33] for the NOVA 1220. These subroutines were accessible through keyboard commands from the teletype. The pulse height analysis program was written to allow the completion of a sequence (or chain) of commands. The commands in such a chain could be repeated any selected number of times. The generation of control signals to the pneumatic system, a delay for neutron activation, a delay for data transfer, and a cooling delay were controlled by subroutines in the main program. By changing a timing constant in each of the programs, the cycle timing sequence could be altered from trial to trial.

Teletype commands "U" and "D" were written to control the activation and cooling delays, respectively. To insure that the time allowed for data transfer to magnetic tape would remain constant, a delay was written which started with key "R", allowed the data-transfer commands to occur, and finished with key "F". The timing subroutines are listed in Appendix C.

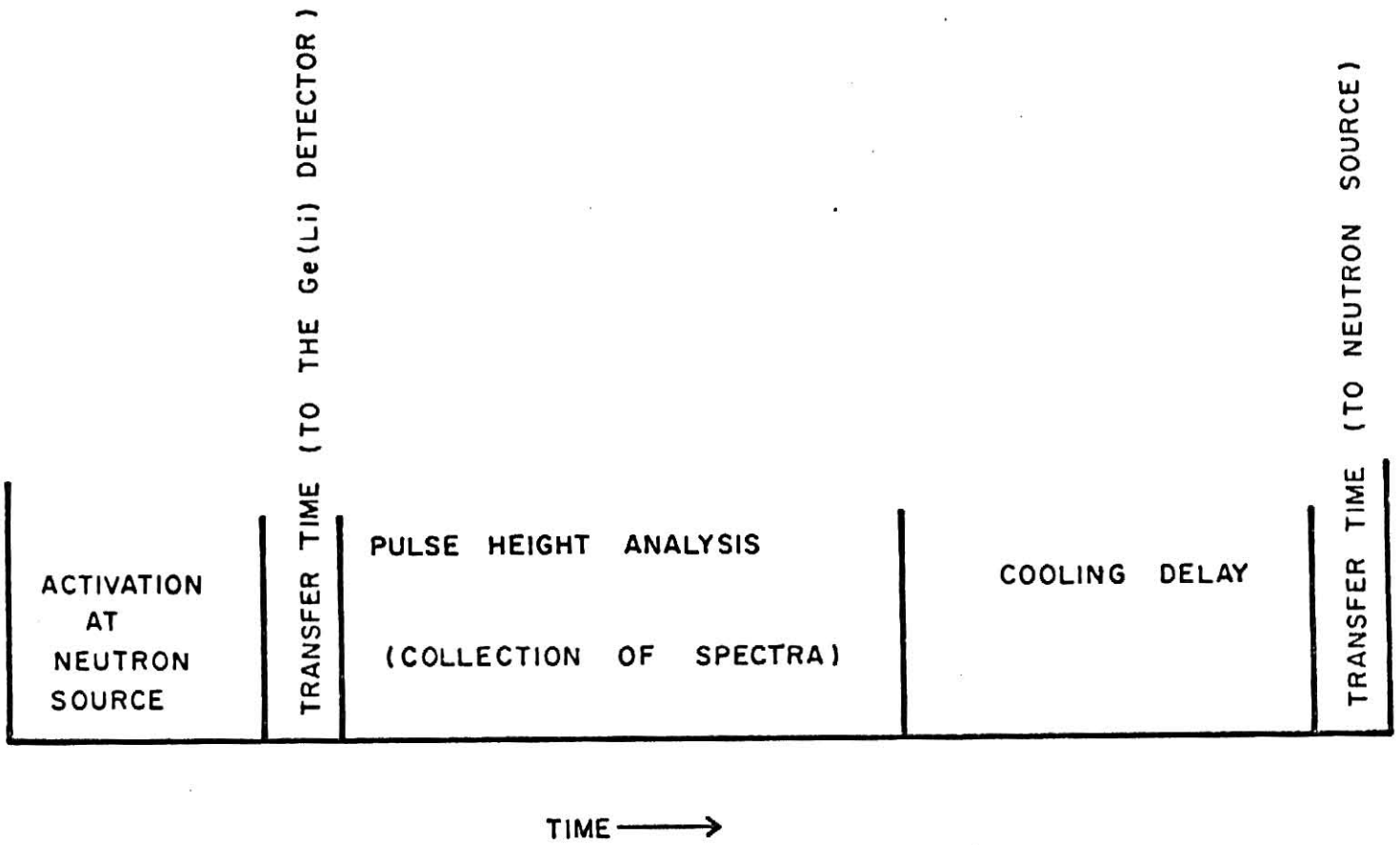


FIGURE V. 1
Time sequence for an example cycle

Since a general-purpose interface for the NOVA 1220 was not available, it was convenient to use computer-generated signals from an unused output pin on the NS-636 analyzer.[25] The plotter-enable signal was a computer-controlled pulse which was used as an input signal to an electronic logic circuit. The circuit activated the pneumatic driver system. Teletype command "Q" was written to generate one plotter enable pulse which was used to send the sample to the detector. Teletype command "J" was written to generate two pulses within a few hundred microseconds which sent the sample to the neutron source. The programs that control these functions are listed in Appendix C.

After the data for each trial were collected on magnetic tape, the spectra for corresponding time windows after transfer were summed and redeposited on the tape. The NOVA system had only 8K of memory, so it was impossible to sum the spectra in the memory while the data was collected. For this reason, a special assembly-language program was written for the NOVA computer. The initial and final tagwords as well as the number of spectra generated per cycle were inputs to the program. The program operated by first initializing the contents of 4096 memory locations with a zero value, and searching the magnetic tape for the initial tagword. When the initial tagword was found, the spectrum was read into the NS-636 memory, and then summed with the zeros in the NOVA memory. The result, which was the initial spectrum, was stored back into the NOVA memory. The program raised the initial tagword by the number of spectra generated per cycle and searched for this, the next corresponding spectrum. In this manner, all of the proper spectra were summed until the last tagword was reached. The program then rewound the tape and deposited the composite spectrum at the initial tagword location. It then incremented the initial tagword, zeroed a portion of the NOVA memory, and continued the summing until all of the composite spectra were generated. A copy of this summation program is included in Appendix C.[34]

IBM 370/158 Computer Analysis

When the composite spectra were generated and deposited on magnetic tape by the NOVA computer, the spectra had to be removed and modified so that further analysis could be done to determine the effective half-lives and energies of the peaks recorded. Using the KSU Department of Nuclear Engineering's TRAPL computer code[35] (Tape Removal And PLot code), the data was removed from magnetic tape. The program was used to punch the spectral information on computer cards, print the number of counts per channel for each 4096 spectrum, and give a semi-log plot of the information. A special loop was written into the program to allow some curve smoothing by summing the information over three channels before the plotting was done. This summing helped visually in deciding which peaks were significant enough for later analysis. A copy of the TRAPL code and an example of the printout is included in Appendix D.

Each spectrum was broken into five segments and the energy axis was expanded to further aid in visual analysis by reading overlapping groups of 1024 punched computer data cards into the Gamma Spectrum Plotting Program Code, or GASP-4.[36] This code gave a linear plot of each spectral segment. Also, the size of the plot was controlled by a data input card. A copy of the GASP-4 computer code is included in Appendix D.

After the data had been plotted, punched, and printed, it was investigated for peak and energy information. Using the calibration-spectrum peaks of known energy and channel number, an energy calibration curve was generated for each trial by fitting the known data to either a linear or quadratic least squares fitting program. The program used for this analysis was from P. R. Bevington[37] modified by N. D. Eckhoff[35]. The program generated errors on the calculated constants, so errors on calculated peak energies could be determined. A copy of this code and the resulting least squares output for the trials are included in Appendix D.

Peak Identification and Analysis

To determine the energy of valid peaks, Lagrangian interpolation was used. In this method, a quadratic polynomial was passed through the centermost three points of a peak, and the maximum location was used in the calibration polynomial expression to determine the peak energy. If n equals the center channel of the three most intense channels of a peak, then the maximum location n_{max} is given by

$$n_{max} = \tfrac{1}{2}\left[\frac{(2n+1)C_{n-1} - 4nC_n + (2n-1)C_{n+1}}{C_{n-1} - 2C_n + C_{n+1}}\right], \qquad \text{V.1}$$

where C_n = counts in the center channel

C_{n-1} = counts in the channel left of center

C_{n+1} = counts in the channel right of center

This procedure is equivalent to finding the centroid of a Gaussian function.[22]

To analyze the exact peak intensity, the total-peak-area (TPA) method was used.[22] In this method, the net peak area N was defined as

$$N = G - B, \qquad \text{V.2}$$

where G was the gross peak area defined as

$$G = \sum_{i=\ell+1}^{r-1} C_i, \qquad \text{V.3}$$

where the base area B was defined as

$$B = \tfrac{1}{2}(C_\ell + C_r)(r-\ell-1), \qquad \text{V.4}$$

and where

C_i = counts per channel i

ℓ = channel number of left minimum

r = channel number of right minimum.

This method is based on the definition of a peak as being that part of a spectrum which is bounded on the left and right by parts which are concave up.

The variance of the net peak area is given by the expression

$$\sigma^2(N) = G + B\ (r-\ell-1)/2 \quad . \qquad \text{V.5}$$

A validity check for each peak considered was made. A peak was considered for analysis if $N \geq 1.96\ \sigma(N)$. The standard deviation associated with each peak area was $\sigma(N)$ of Equation V.5.

The peak energies were determined by substituting the maximum location (n_{max}) found from the previous method into the least squares formula generated by the computer program.

After both the peak intensity and energy were found using a programmable calculator, the data were normalized to counts per second by correcting for long-lived background interference, for the number of cycles and counting time used, and for the detector efficiency. A detector efficiency versus energy curve for a source at the experimental sample position was generated by using a National Bureau of Standards mixed radionuclide gamma-ray point source of known intensity.[38] From a spectrum of this source, the peak areas for ^{137}Cs at 0.66164 MeV, ^{60}Co at 1.17323 MeV, and ^{60}Co at 1.33251 MeV were calculated. The gamma-ray intensity (gammas/min) was corrected for radioactive decay from the calibration data for each of the peaks.[39] The detector efficiency for these three points was found by dividing their detection rates by their intensities:

$$\varepsilon = A/N \qquad \text{V.6}$$

where N = intensity of calibration gamma ray

A = count rate in peak

ε = detector efficiency

The log of the efficiency versus log energy for these points is shown in Figure V.2.

This efficiency measurement was extended to higher energies by activating isotopes with well-known gamma-ray intensities in the KSU TRIGA reactor. The isotopes produced were ^{24}Na and ^{49}Ca. Sodium was selected because it has two gamma rays at 1.36853 and 2.7539 MeV with equal intensities.[40] By extending the detector efficiency curve through the 1.36853 MeV energy position, the efficiency at 2.7539 MeV was found using the equation

$$A_1/\varepsilon_1 = A_2/\varepsilon_2 \qquad \text{or}$$

$$\varepsilon_2 = \frac{A_2}{A_1}\,\varepsilon_1 \qquad \text{V.7}$$

where 1,2 = 1st and 2nd gamma rays.

Calcium was activated because ^{49}Ca has three gamma-ray energies with known relative intensities extending to high energy. The energies used were 3.08, 4.07, and 4.74 MeV. At these energies, the relative intensities were 0.921, 0.070, and 0.0021, respectively.[41] By extrapolating the efficiency curve from 2.75 to 3.10 MeV, the efficiency at 3.10 MeV was estimated. Using the relative intensities of the gamma rays of ^{49}Ca, the detector efficiency up to 4.7 MeV was found. Figure V.2 shows the resulting efficiency versus energy curve with a visually-fit curve through the data points.

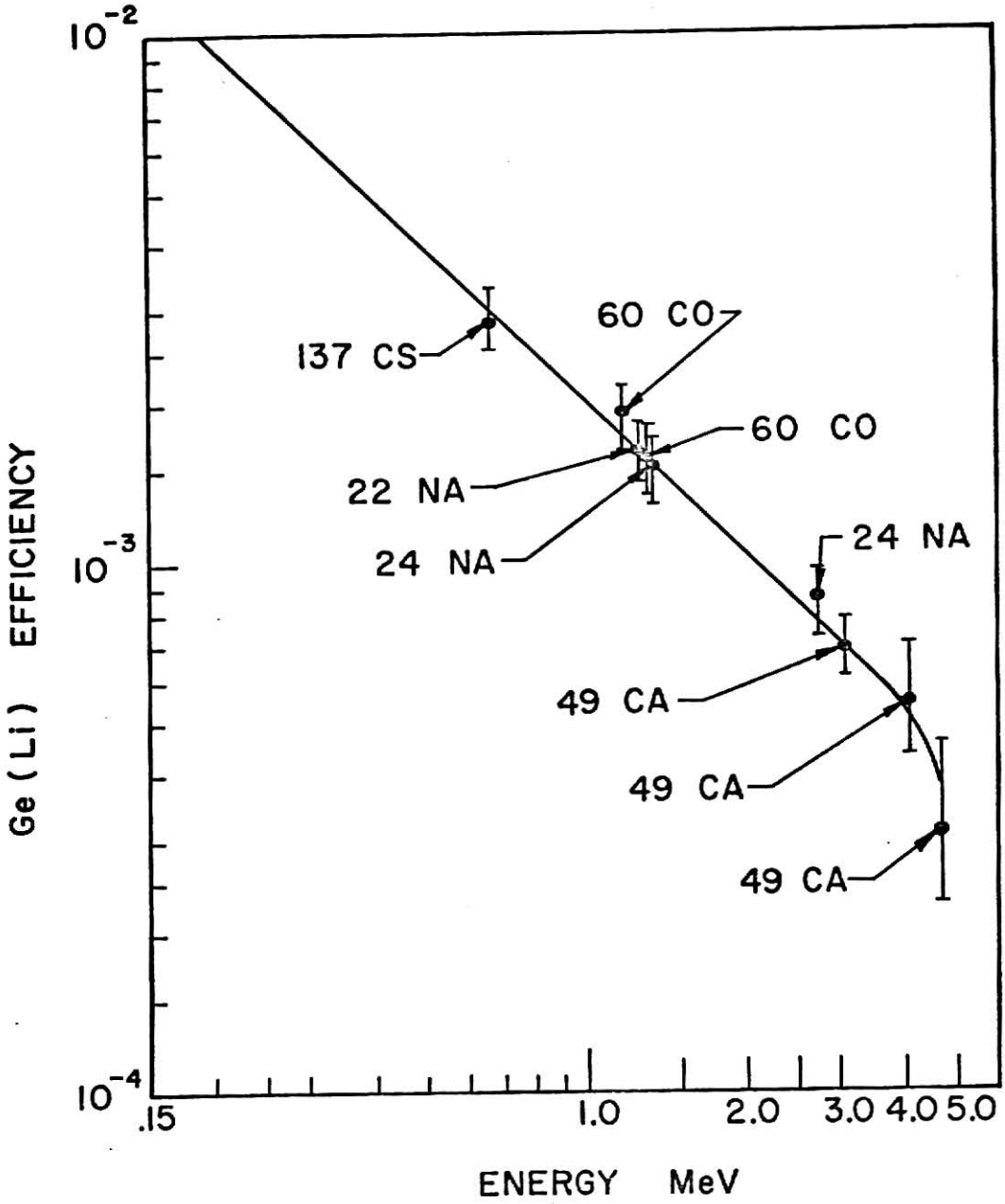


FIGURE V. 2
Ge(Li) detector efficiency curve

VI. Results

^{252}Cf Analysis

The first part of this analysis is devoted to the investigation of the short-lived fission-product gamma rays from ^{235}U after activation with a small (5 microgram) ^{252}Cf source. Several attempts to collect useful data were made, and while each attempt led directly to improvements in the equipment and procedures, none yielded useful information. The collected spectra, located in Appendix E, do not show enough gamma-peak information to allow detailed analysis. The problem encountered was that the neutron flux was not intense enough from the ^{252}Cf source to achieve the gamma-ray intensity required for the short-lived analysis. The decision to extend the scope of the project to include the use of the thermal column of the KSU TRIGA Mk II reactor was made to achieve a greater neutron flux. All data trials, for both the ^{252}Cf analysis and reactor analysis, were made with the National Bureau of Standards mixed gamma-ray source and ^{22}Na calibration source near the detector. The peaks from these sources were used to calibrate the pulse height analyzer and to check the stability of the detection system. The peaks would appear broadened if any gain shifts occurred during the collection of data. No evidence of instability was seen.

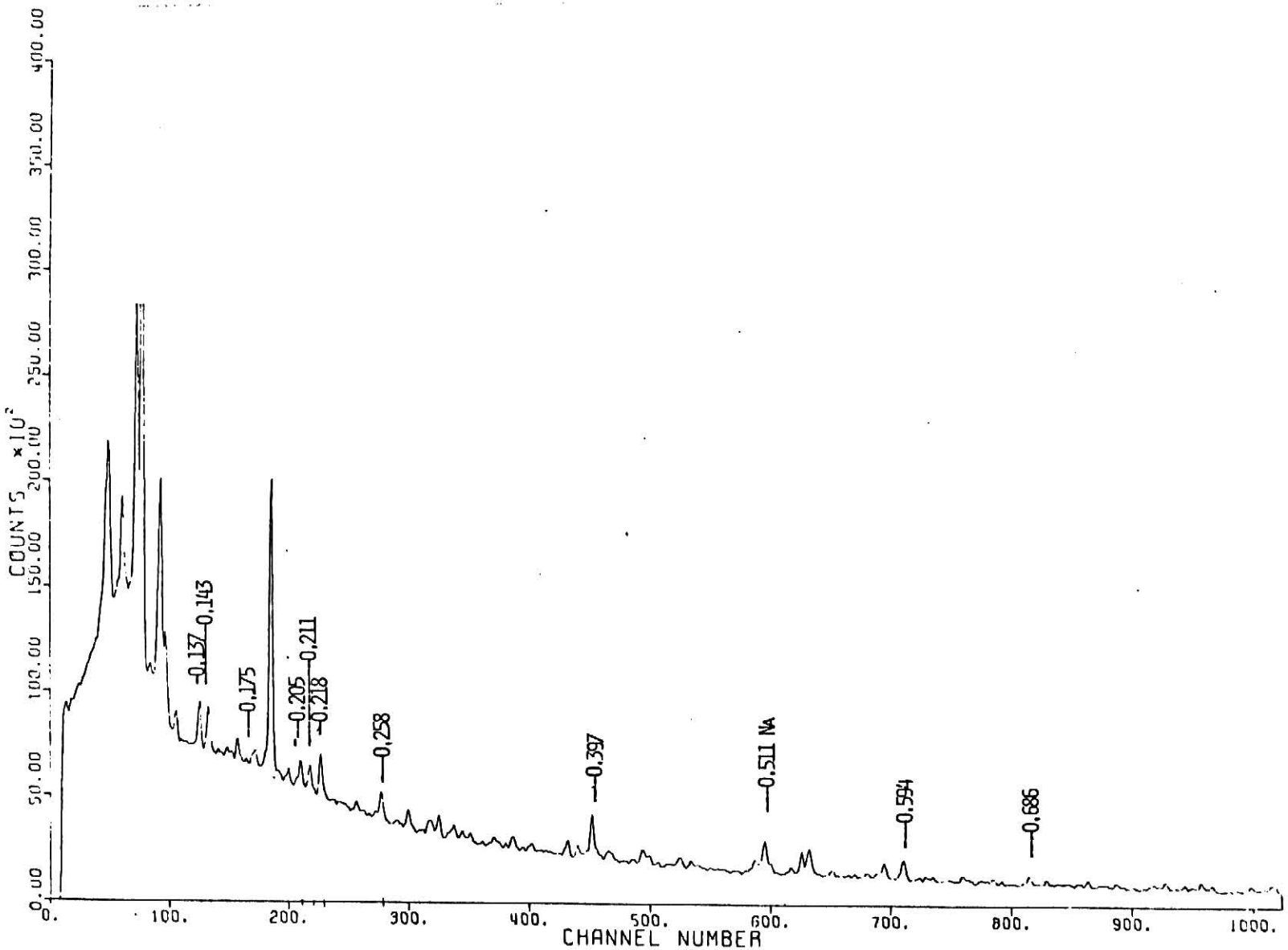
TRIGA Mk II Analysis

Three data collection trials were made transferring the sample between the reactor and the Ge(Li) detector. The activation time, data collection times, reactor power, and number of cycles per trial were shown in Table IV.1.

Table VI.1 Data Trial Information

Trial	Activation Time(sec)	Reactor Power(watts)	Data Time Windows(sec)	Cycles
1.	10	10	0.5-2.5 2 - 4 5.5-7 7 - 9 12 - 14 17 - 19 50 - 52	100
2.	30	20	12 - 14.5 23 - 27 33.5-37.5 43.5-48.5 54 - 60 230 - 255	75
3.	2	100	0.5-5.5 8.5-13.5	600

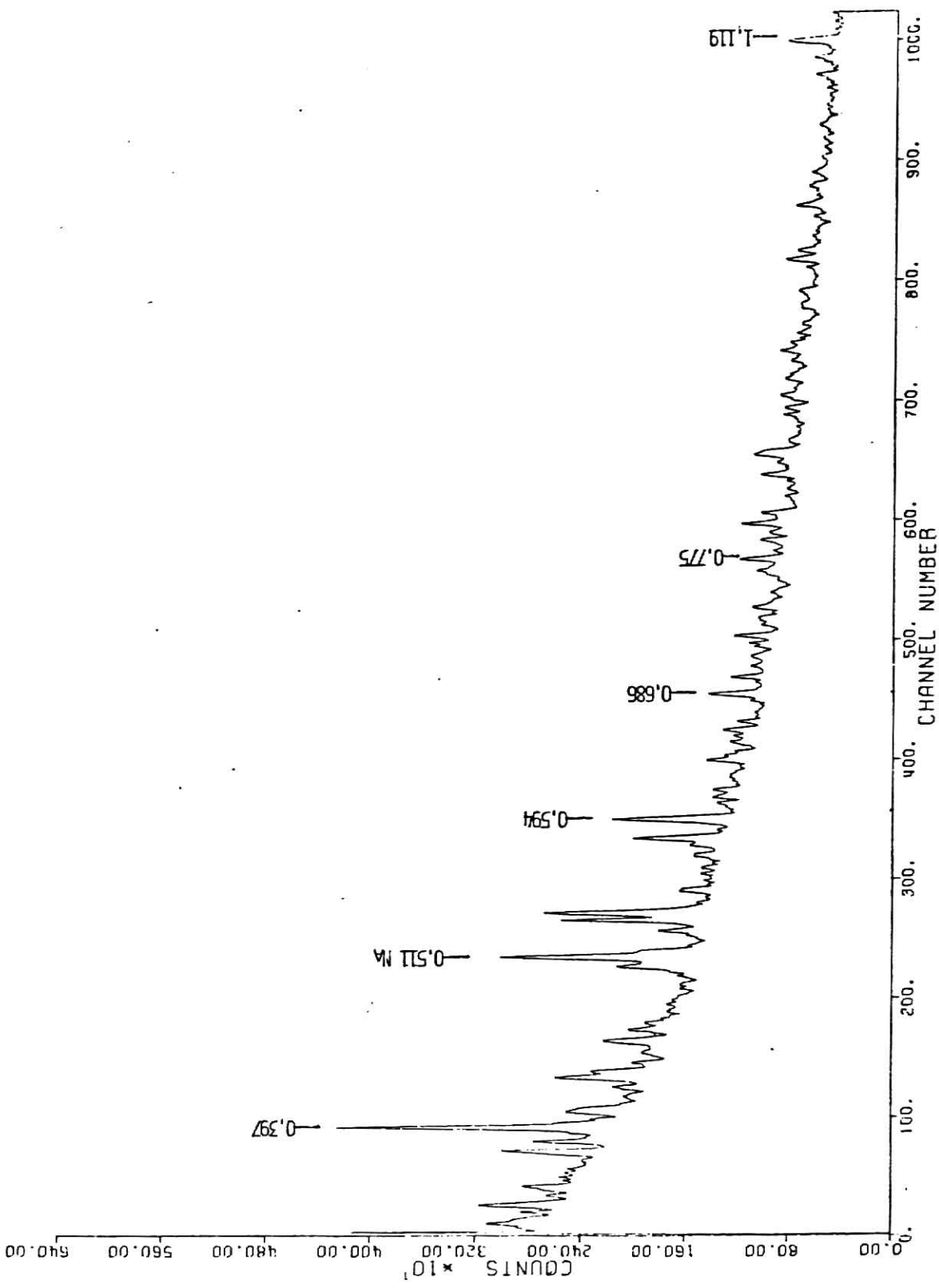
The data collected are plotted on overlapping segments of 1024 channels for ease in the visual determination of peaks. Using the total-peak-area method,[22] the locations of the peaks were found and their areas were calculated. A calculation was made to determine the standard deviation of the peak areas and to determine if the peaks were statistically significant ($N \geq 1.96\ \sigma(N)$). The net peak areas and their standard deviations were corrected for counting time, number of cycles, long-lived background, and detector efficiency. The calculated values for the peaks considered for Trial 3 are included in raw data tables in Appendix F. Representative plots of low and high energy spectra with the energies of the peaks considered are shown in Figures VI.1, VI.2, and VI.3. For a visual comparison of the change in peak area with time after transfer, spectra collected at both short and long times after transfer for Trials 1 and 2 and a complete set of spectra for Trial 3 showing the higher energy peaks considered are shown in Appendix G.



12 TO 14.5 SEC. AFTER FISSION 75 CYCLES 5/30/75 0-1024

FIGURE VI. 1

Low energy spectral segment (0 to 0.71 MeV)



12 TO 14.5 SEC. AFTER FISSION 75 CYCLES 5/30/75 361-1385
FIGURE VI. 2
Low energy spectral segment (0.35 to 1.22 MeV)

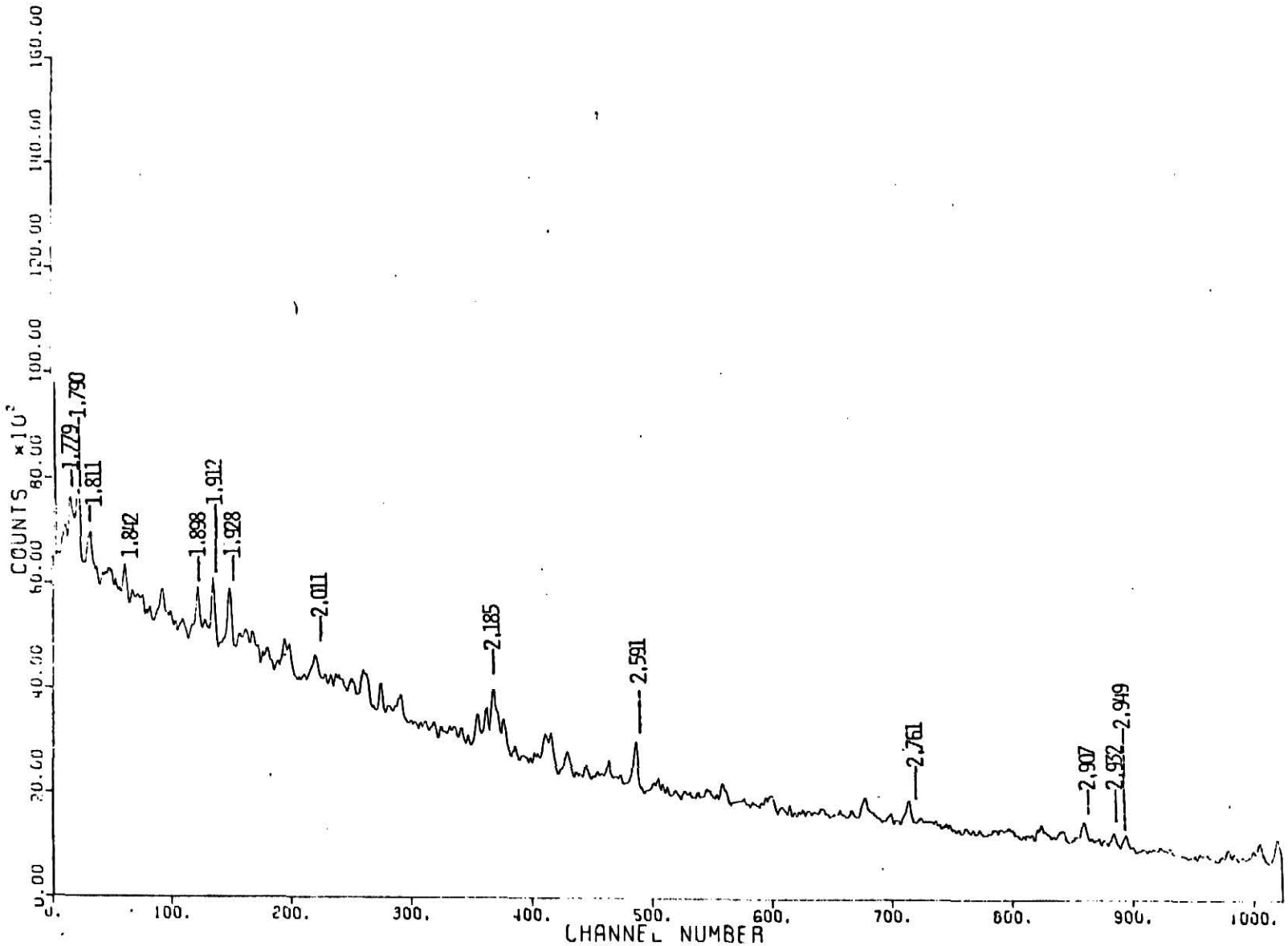


FIGURE VI. 3

High energy spectral segment (1.75 to 3.10 MeV)

Most of the peak-intensity information was complex, that is, there were possibilities that peaks contain several unresolved gamma rays and that some decays represent short-lived fission products with longer-lived precursors.[22] Figure VI.4 shows four examples of the peak intensities, with their standard deviations, plotted versus time. Further time-dependent information for other energies is located in Appendix H. The peaks for Trials 1 and 2 show complex decay behavior. The curves shown were drawn by a visual fit through the data points. In a few cases straight lines were fit through all of the error bars and so an effective half-life from the slope of the line is reported.

Since only two spectra were collected for Trial 3, and the half-lives for the higher energies are short compared to lower energies, the log of both intensity points were plotted and connected with a straight line and effective half-lives are reported for these energies. First and second escape peak contributions from peaks above about 1.5 MeV may have contributed to the complex nature of the data reported here.[39]

The peaks correspond, in most cases, to peaks observed in earlier cyclic activation experiments, except for the higher energy peaks where information from earlier work was incomplete. Table VI.2 shows the peaks which correspond between previous work and this analysis.[21,22] Table VI.3 shows a list of peaks observed which don't correspond with previous cyclic activation analysis information. The peak energies observed and their relative intensities per trial were next compared with literature lists of gamma-ray energies in Table VI.4.[40,42] Few of the gamma-rays observed in this analysis correspond with gamma table values. This is attributed to the fact that the gamma-rays observed have half-lives which are less than one minute, and the fission product data is still incomplete in this time range.

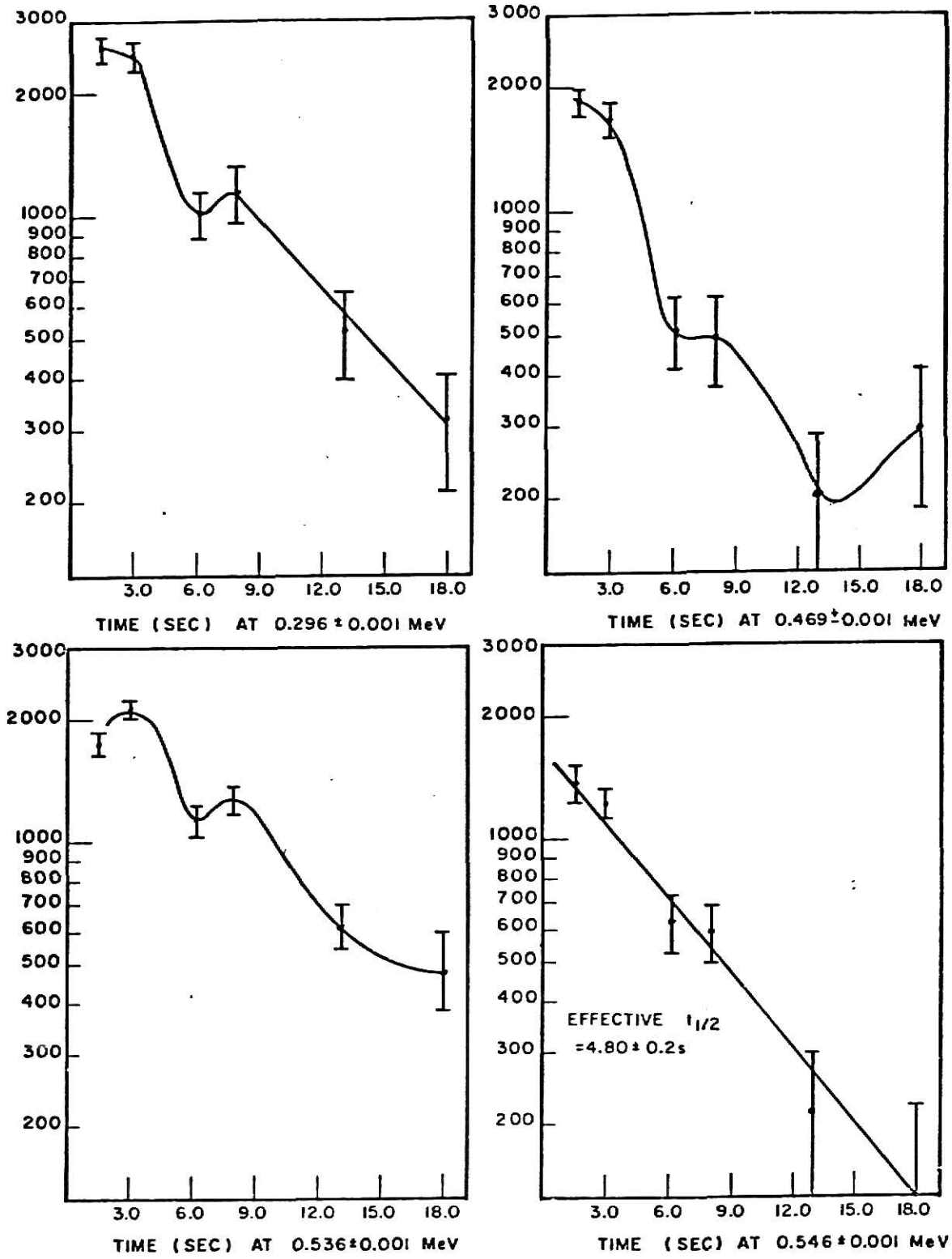


FIGURE VI. 4
Example of log peak area vs time plots

Table VI.2 Comparison of Measured Gamma-ray Energies With Literature Values

Energy Measured (MeV)	$T_{1/2}$ (eff)	Literature Energy	$T_{1/2}$ [a,b]
Trial 1			
0.296 ± 0.001	-	0.29583±0.00018[a] 0.29595±0.00041[a]	- -
0.469 ± 0.001	-	0.46946±0.00049[a]	22 ± 4.5 sec
0.536 ± 0.001	-	0.5368[b]	-
0.546 ± 0.001	4.80 ± .25 sec	0.54464±0.00025[a] 0.54556±0.00034[a] 0.5458[b]	2. ± .3 sec 66 ± 34 min -
1.103 ± 0.002	-	1.10279±0.00063[a]	1.1 sec
1.501 ± 0.002	-	1.5003[b]	-
Trial 2			
0.137 ± 0.001	-	0.137±0.0005[a] 0.136±0.00047[a]	- -
0.143 ± 0.001		0.14199±0.00075[a]	1.8 sec
0.175 ± 0.001	-	0.1741±0.0005[a] 0.1751±0.00061[a]	20. ± 3 sec 1 min
0.211 ± 0.001	-	0.21225±0.00058[a] 0.21146±0.00066[a]	10 ± 4 sec -
0.218 ± 0.001	-	0.21953±0.00063[a] 0.21984±0.0006[a]	24 ± 10 sec -
0.258 ± 0.001	-	0.2596±0.00067[a] 0.25901±0.0007[a]	9 ± 2 sec 13 ± 4 min
0.397 ± 0.001	-	0.39667±0.00042[a]	45 ± 3 sec
0.511 ± 0.001		0.50988±0.00069[a] 0.51043±0.00047[a]	13 ± 7 sec annihilation
0.594 ± 0.001		0.59206±0.00059[a] 0.5964±0.00089[a]	1.7 ± 2 sec -
0.686 ± 0.001		0.68376±0.00081[a]	
0.775 ± 0.001	31.8±2.5 sec	0.77634±0.00034[a] 0.7751[b]	13 ± 3 sec -

Table VI.2 (Continued)

1.119 ± 0.002	-	1.11833±0.000102[a] 1.1189[b] 1.1198[b] 1.1206[b]	47 ± 7 sec
1.250 ± 0.002	30.0±2 sec	1.2484[b]	-
1.313 ± 0.002	-	1.3124[b]	-
1.559 ± 0.002	-	1.565[b]	-
Trial 3			
1.842 ± 0.004	-	1.8415[b]	-
4.273 ± 0.010	11.5 ± 0.3 sec	4.2765[b]	-

a : J. F. Boggs 1966 (reference 21)
b : G. Atkinson, Jr., 1971 (reference 22)

Table VI.3 Unmatched Observed Peaks

	Peak Energy (MeV)	$T_{\frac{1}{2}}$ (eff)(sec)
Trial 1.	0.205±0.001	-
	1.219±0.002	-
Trial 2.	0.205±0.001	-
Trial 3.	1.779±0.003	-
	1.790±0.003	-
	1.811±0.004	-
	1.842±0.004	-
	1.898±0.004	-
	1.912±0.004	-
	1.928±0.004	-
	2.011±0.005	-
	2.185±0.005	-
	2.519±0.005	-
	2.761±0.006	-
	2.907±0.006	10.6±0.3
	2.932±0.006	6.4±0.2
	2.949±0.006	17.0±0.3
	2.968±0.006	13.0±0.3
	3.137±0.007	-
	3.304±0.007	10.4±0.2
	3.344±0.007	35 ± 1.5
	3.5.5±0.008	22.2±0.5
	3.530±0.008	16 ± 0.5
	4.273±0.010	11.5±0.3

Table VI.4. Observed Gamma-ray Energies and Relative Intensities

Observed Values			Literature Values[41]		
Energy(MeV)	$T_{1/2}$(eff.)(sec)	Relative Intensity	Possible Contributor(MeV)	Isotope	$T_{1/2}$(sec)
Trial 1		At 1.5 sec			
0.296±0.001	-	82±6	0.2970	^{134}Sb	10.3
0.469±0.001	-	92±7	0.46798	^{142}Xe	1.22
0.536±0.001	-	100±6	0.5350	^{92}Kr	1.84
0.546±0.001	4.8±0.2	81±8	0.5474	^{142}Xe	1.22
1.103±0.002	-	40±6	1.1006	^{142}Cs	1.67
1.219±0.002	-	62±5	1.2186	^{92}Kr	1.84
			1.2192	^{142}Xe	1.22
1.501±0.002	-	41±6	-	-	-
Trial 2		At 13.25 sec			
0.137±0.001	-	38±2	-	-	-
0.143±0.001	-	23±4	-	-	-
0.175±0.001	-	25±3	0.1749	^{139}Xe	39.7
0.205±0.001	-	29±4	-	-	-
0.211±0.001	-	52±5	-	-	-
0.258±0.001	-	100±4	-	-	-
0.397±0.001	-	71±6	0.397	^{144}La	42.4
0.511±0.001	-	41±7	-	-	-
0.594±0.001	-	86±7	-	-	-
0.686±0.001	-	34±5	-	-	-
0.775±0.001	-	33±5	0.77412	^{140}Xe	13.6
1.119±0.002	-	52±8	-	-	-
1.250±0.002	-	14±4	-	-	-
1.313±0.002	-	33±7	-	-	-
1.559±0.002	-	12±2	-	-	-

Table VI.4 (Continued)

Observed Values			Literature Values[41]		
Energy(MeV)	$T_{1/2}$(eff.)(sec)	Relative Intensity	Possible Contributor(MeV)	Isotope	$T_{1/2}$(sec)
Trial 3		At 3 sec			
1.779±0.003	-	-	-	-	-
1.790±0.003	-	-	-	-	-
1.811±0.004		-	-	-	-
1.842±0.004	-	-	-	-	-
1.898±0.004	-	-	1.8968	$9\bar{2}_{Kr}$	1.84
1.912±0.004	-	-	-	-	-
1.928±0.004	-	-	-	-	-
2.011±0.005	-	-	2.0065	$9\bar{2}_{Rb}$	4.53
2.185±0.005	-	-	-	-	-
2.591±0.005	-	-	2.5875	$9\bar{2}_{Kr}$	1.84
2.761±0.006	-	-	2.759	^{92}Kr	1.84
2.907±0.006	10.6±0.3	54±7	2.913	^{92}Rb	4.53
2.932±0.006	6.4±0.2	100±9	2.924	^{142}Cs	-
2.949±0.006	17.0±0.3	90±14	-	-	-
2.968±0.006	13±0.3	17±8	-	-	-
3.137±0.007	-	65±10	-	-	-
3.304±0.007	10.4±0.2	43±10	-	-	-
3.344±0.007	35±1.5	35±11	-	-	-
3.515±0.008	22±0.5	33±7	-	-	-
3.530±0.008	16±0.3	25±7	-	-	-
4.273±0.010	11.5±0.3	14±5	-	-	-

VII. Conclusions and Recommendations

The short-lived fission-product spectra generated by cyclic activation in this analysis show general agreement with previous attempts at recording the unique gamms-ray signatures of ^{235}U fission-products.[21,22] The data presented in Chapter VI give additional gamma-ray energies not previously reported by the cyclic activation technique. Table VII.1 summarizes information from Tables VI.3 and VI.4 on the energy, effective half-life, and relative intensity of the new peaks observed.

The source strength of the ^{252}Cf used in the first part of this analysis was not enough to generate useful gamma-ray data, and so the experiment was done using TRIGA Mk II reactor neutrons, where the neutron flux was directly controlled by changing the reactor power level. A similar analysis could have been done using a large ^{252}Cf source. To calculate the required ^{252}Cf source size to duplicate the reactor results, the source is assumed to be located at the center of a polyethylene sphere 20 cm in diameter, and the activation location is assumed to be 1 cm away from the source. The polyethylene serves as both a reflector and a moderator for the ^{252}Cf neutrons. For this configuration, the design data is well known;[29] one centimeter from the center of the polyethylene sphere, the thermal flux is 0.0205 neutrons/cm^2 sec for each source neutron emitted, so that the source size is calculated directly from the desired neutron flux knowing that ^{252}Cf emits 2.34×10^{12} n/sec g .[28] Table VII.2 lists the calculated ^{252}Cf source size required to produce the neutron flux of the reactor power levels at the fully inserted position in the thermal column.[30]

Table VII.1 Observed Gamma-rays and Relative Intensities Previously not Reported by Cyclic Activation Analysis

Energy (MeV)	Relative Intensity	$T_{\frac{1}{2}}$(eff.)(sec)
Trial 1		
1.219±0.002	62±5	-
Trial 2		
0.205±0.001	29±4	-
Trial 3		
1.779±0.003	-	-
1.790±0.003		
1.811±0.004		
1.842±0.004		
1.898±0.004		
1.912±0.004		
1.928±0.004		
2.011±0.005		
2.185±0.005		
2.519±0.005		
2.761±0.006		
2.907±0.006	54±7	10.6±0.3
2.932±0.006	100±9	6.4±0.2
2.949±0.006	90±14	17.0±0.3
2.968±0.006	17±8	13.0±0.3
3.137±0.007	65±10	-
3.304±0.007	43±10	10.4±0.2
3.344±0.007	35±11	35±1.5
3.515±0.008	33±7	22.2±0.5
3.530±0.008	25±7	16±0.5
4.273±0.10	14±5	11.5±0.3

Table VII.2 Required ^{252}Cf Size to Duplicate the Neutron Flux for the Reactor Power Levels Used.

Reactor Power (W)	Thermal Column Neutron Flux[30]	^{252}Cf size (g)
10	1.04×10^{6}	2.168×10^{-5}
20	2.08×10^{6}	4.336×10^{-5}
100	1.04×10^{7}	2.168×10^{-5}

With recent developments in the quality of Ge(Li) gamma-ray spectrometers and the ability of minicomputers, the use of sophisticated techniques for non-destructive assay by active interrogation has great potential. An active interrogation assay system using information from high energy gamma-ray fission-product spectra can probably be designed to determine the concentrations in mixed ^{235}U and ^{239}Pu samples. The relative fission-product gamma-ray intensities are dependent on the fission yields of the fissioning isotopes and the energy spectrum of the interrogating neutrons. Figure VII.1 shows the ratio of the thermal fission yields of ^{239}Pu to ^{235}U versus mass number.[43] The variation in this ratio indicates that a computer-controlled assay system could be used to unfold complex spectra of the penetrating gamma-rays to determine the isotopic concentrations of ^{235}U and ^{239}Pu in a sample. The use of ^{252}Cf neutrons would allow changes in the energy of the source neutrons to optimize the fission yield differences.

Extensions of fission-product gamma-ray analysis can be done in several major areas. Further identification of the short-lived fission-product spectra should be done for ^{239}Pu with emphasis on the isotopic discrimination ratios relative to ^{235}U. The differences in the spectra observed for interrogating neutrons of different energies could be explored using a tailored ^{252}Cf source spectrum. A more sophisticated data analysis could be done by extending the

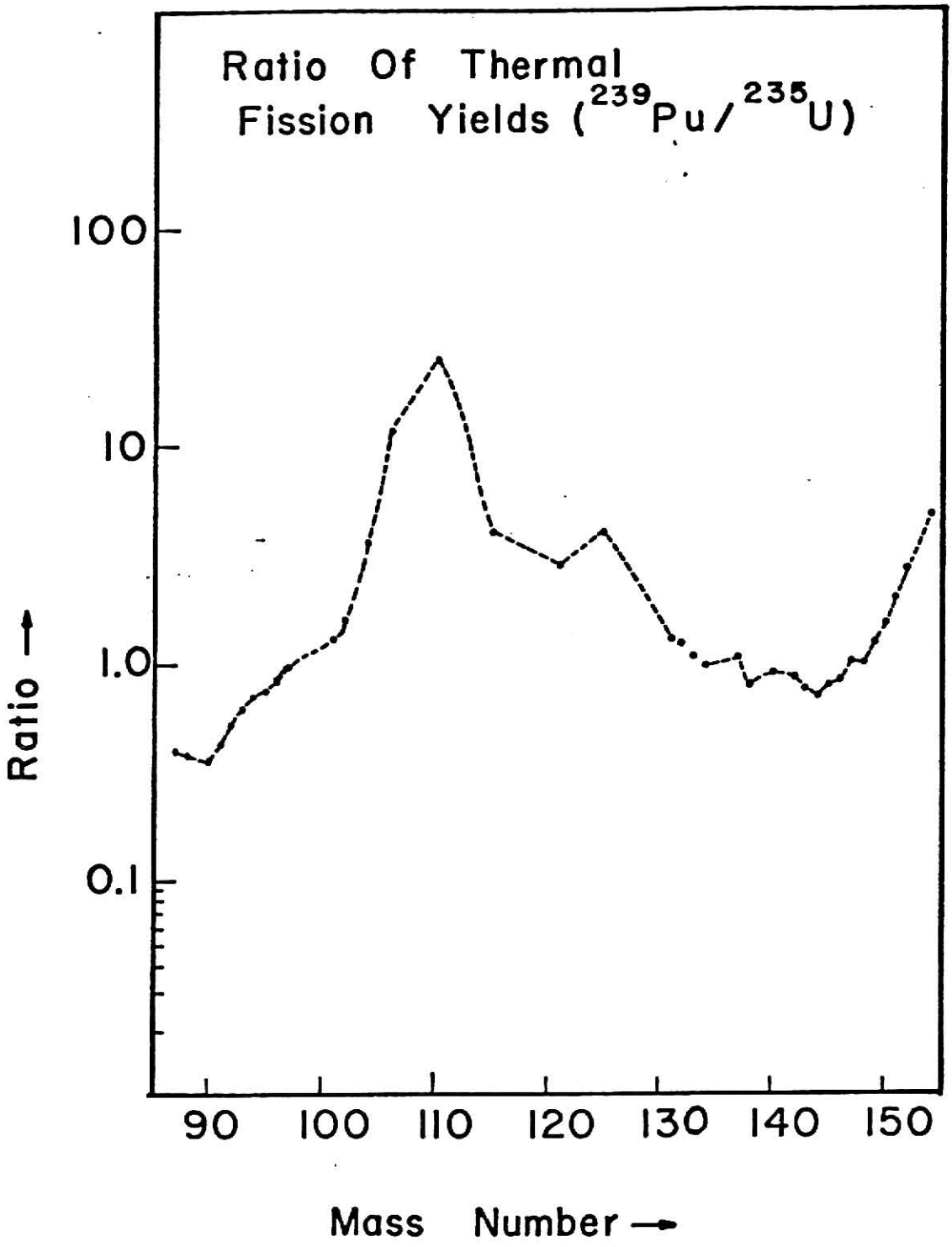


FIGURE VII. 1
Ratio of thermal fission yields ($^{239}Pu/^{235}U$)

memory of the computer. With continued research using modern gamma-ray spectrometers and minicomputer systems, active nondestructive assay of materials containing ^{235}U, ^{239}Pu, or mixtures of these isotopes can be improved.

Acknowledgments

The author wished to express his sincere gratitude to the many individuals from whom assistance and encouragement have been obtained in the preparation of this theses. In particular, appreciation is extended to Dr. M. S. Krick for his outstanding supervision and support during this study.

The assistance and interest of many individuals in the Department of Nuclear Engineering at Kansas State University is appreciated. Most sincere thanks go to Merna Brisbin for the time and effort she spent in typing, and to Oscar Mulhern for his assistance in the electronics shop.

The author also wished to acknowledge his financial support from the Department of Nuclear Engineering at Kansas State University as a graduate research assistant. Appreciation for assistance with the computer programs used is extended to Dr. J. K. Shultis, DR, N. D, Eckhoff, and Mr. H. Ocampo.

W. E. K. Jr.

References Cited

1. Keepin, G. R., Bramblett, R. L., and Higinbottham, W. A., "Development of Techniques and Instrumentation for Non-destructive Assay of Fissionable Materials", Peaceful Uses of Atomic Energy, Proceedings of the Fourth International Conference, Geneva, Volume 9, (September 1971).

2. Thorpe, M. M., "Role of Nuclear Data in the Practical Application of Nondestructive Nuclear Material Assay Methods", Los Alamos Scientific Laboratory, LA-UR-73-226, (1973).

3. Reilly, T. D., Parker, J. L., "Guide to Gamma-ray Assay for Nuclear Material Accountability", Los Alamos Scientific Laboratory, LA-5794-MS, (December 1974).

4. Phillips, J. R., et. al., "New System for Gamma Scanning Fuel Elements", Los Alamos Scientific Laboratory, LA-DC-72-525, (1972).

5. Nodvik, R. J., "Evaluation of Gamma Scanning as a Tool for Determinating Fuel Burnup Distributions in Large Power Reactor Cores", Westinghouse Electric Corp., Pittsburgh, Pa., Atomic Power Div., WCAP-5490, (1964).

6. Keepin, G. R., "Nondestructive Detection, Identification, and Analysis of Fissionable Materials", Los Alamos Scientific Laboratory, LA-3741, (January 1967).

7. Eidens, J., "Identification and Nuclear Spectroscopy of Several Short-lived Fission Products With an On-line Mass Separator", Kernforschungsanlage, Juelich (West Germany) for Los Alamos Scientific Laboratory, LA-4174-TR, (1970).

8. Evans, A. E., Jr., "Nondestructive Assay of Fissile Material Samples in Support of Nuclear Safeguards", Los Alamos Scientific Laboratory, LA-UR-73-1595, (1973).

9. Bramblett, R. L., et. al., "Applications of Photonuclide Reactions to Nuclear Materials Safeguards Problems", Gulf Radiation Technology, GA-10272, (July 1970).

10. Reining, W. C., "Advantages and Applications of ^{252}Cf as a Neutron Source", Nuclear Applications, Vol. 5, (July 1968).

11. Forster, R. A., Smith, D. B., Menlove, H. O., "Cf-252 Fuel Rod Assay System: In Plant Performance", Los Alamos Scientific Laboratory, LA-DC-72-597, (1971).

12. Forster, R. A., Smith, D. B., Menlove, H. O., "Error Analysis of a Cf-252 Fuel-Rod-Assay System", Los Alamos Scientific Laboratory, LA-5317, (April 1974).

13. Keepin, G. R., Physics of Nuclear Kinetics, Addison-Wesley Publishing Company, Inc., Reading, Massachusetts, (1965).

14. Menlove, H. O., et. al., "Cf-252 Assay System for FBR Fuel Pins: Description and Operating Procedure Manual", Los Alamos Scientific Laboratory, LA-5071, (October 1972).

15. Menlove, H. O., "Cf-252 Neutron Source Systems for the Assay of Uranium and Plutonium in Fuel Rods", Los Alamos Scientific Laboratory, LA-DC-72-977, (1972).

16. Adámek, A., Severa, F., "A Dual Pneumatic Tube Transfer System for the Analysis of Fast Neutron Activated Short-lived Isotopes", Journal of Radioanalytical Chemistry, 7, 119, (1971).

17. Gage, S. J., Bouchey, G. D., Attkinson, G. D. Jr., "Cyclic Neutron Activation Analysis with a Cf-252 Neutron Source", Transactions of the American Nuclear Society, 14, 513, (1971).

18. Wiernik, M., Amiel, S., "Use of Very Short-lived Nuclides in Nondestructive Activation Analysis With a Fast Shuttle Rabbit", Journal of Radioanalytical Chemistry, 3, 393, (1969).

19. Jamieson, J. M., and Graham, W. III., "A Cyclic Activation Technique for Non-destructive Fuel Analysis", Institute of Nuclear Materials Management 12th Annual Meeting, June 29-July 1, Vol. II., (1971).

20. Gage, S. J., Atkinson, G. D. Jr., Bouchey, G. D., "Cyclic Activation with a Californium-252 Source", Nuclear Technology, Vol. 17, (March 1973).

21. Boggs, J. F., "Gamma-ray Spectra of Fission Products from ^{235}U as a Function of Time After Fission", The University of Oklahoma, Norman, Oklahoma, Ph.D. Dissertation, (1966).

22. Atkinson, G. Jr., "Nondestructive Uranium Assay by Delayed Gamma-ray Analysis Following Californium-252 Neutron Interrogation", The University of Texas at Austin, Ph.D. Dissertation, (1971).

23. Tomberlin, T. A., "A Cadmium Filter Technique in Variable-Energy Neutron Activation Analysis", a Masters Thesis, Kansas State University, Manhattan, Kansas, (1974).

24. English, W., How to Use the Nova Computers, Data General Corporation, Southboro, Massachusetts, (1972).

25. NS-440N Data General Computer Interface Instruction Manual, Northern Scientific, Inc., Middleton, Wisconsin.

26. Hayt, W. H. Jr., and Kemmerly, J. E., Engineering Circuit Analysis, McGraw-Hill Book Company, New York, New York, (1971).

27. Kowalski, E., Nuclear Electronics, Springer-Verlag, New York, Berlin, (1970).

28. Cornman, W. R., Personal Communications from Californium Information Center of E. I. DuPont deNemours and Company, Atomic Energy Division, Savannah River Laboratory, Aiken, South Carolina, (1972).

29. Nichols, J. P., "Design Data for ^{252}Cf Neutron Source Experiments", Nuclear Applications, Vol. 4, (June 1968).

30. Bouchey, G. D., "Experimental Neutron Flux Measurements and Power Calibration in the Kansas State University TRIGA Mk II Nuclear Reactor", a Masters Thesis, Kansas State University, Manhattan, Kansas, (1967).

31. 250-KW TRIGA Mark II Pulsing Reactor Mechanical Maintenance and Operating Manual, General Atomic Division of General Dynamics, San Diego, California, (August 1962).

32. Title 10, part 20, Code of Federal Regulations of the Federal Register.

33. Tracor Northern, 2551 West Beltline Highway, Middleton, Wisconsin.

34. Ocampo, H., Personal communications, Department of Nuclear Engineering, Kansas State University, Manhattan, Kansas, (1975).

35. Eckhoff, N. D., Personal communications, Department of Nuclear Engineering, Kansas State University, Manhattan, Kansas, (1975).

36. Shultis, J. K., Personal communications, Department of Nuclear Engineering, Kansas State University, Manhattan, Kansas, (1975).

37. Bevington, P. R., Data Reduction and Error Analysis for the Physical Sciences, McGraw-Hill Book Company, St. Louis, Missouri, (1969).

38. Communications from the U. S. Department of Commerce National Bureau of Standards about the Standard Reference Material 4216 Mixed Radionuclide Gamma-ray Emission-rate Point-source Standard.

39. Price, W. J., Nuclear Radiation Detection, Second Edition, McGraw-Hill Book Company, New York, New York, (1964).

40. Martin, M. J., Blichert-Taft, P. H., "Catalogue of Gamma-rays Emitted by Radionuclides", Nuclear Data Tables, Numbers 1-2, Volume 8, Academic Press, New York, New York, (October 1970).

41. Bowman, W. W., Mac Murdo, K. W., "Radioactive-Decay Gammas and Nuclides", Atomic and Nuclear Data Tables, Numbers 2-3, Volume 13, Academic Press, New York, New York, (February 1974).

42. Wakat, M. A., "Catalogue of Gamma-Rays Emitted by Radionuclides", Nuclear Data Tables, Numbers 5-6, Volume 8, Academic Press, New York, New York, (January 1971).

43. Krick, M. S., Personal communications, Department of Nuclear Engineering, Kansas State University, Manhattan, Kansas, (1975).

Appendix A

Correspondence to the KSU Radiation Safety Officer and Reactor Experiment Number 38, "Detection of Short-Lived Fission Products of ^{235}U by Cyclic Activation".

May 20, 1975

Mr. John P. Lambert
Radiation Safety Officer
Ward Hall
Kansas State University

Dear Sir:

In agreement with our discussion on May 18, 1975 about the use of the ^{252}Cf source in the Neutron Activation Analysis Lab (Ward Hall room 11) for my master's thesis, I am submitting the following neutron and gamma-ray activity analysis.

Using the Texas Nuclear Model No 9146 Count Rate Meter, the neutron dose rate from the ^{252}Cf source was monitored at the top and side surfaces of both the ^{252}Cf shipping container and the wax-boron shielding assembly used in my experiment. Table I shows the recorded dose rates.

TABLE I

	Dose Rate Side Surface (mr/hr)	Dose Rate Top Surface (mr/hr)
Shipping Container	8	6
Shielding Assembly	0.35	0.7

At a distance of about ten feet from the shielded assembly near the analyzer and Ge(Li) detector, the activity was recorded as less than 0.01 mr/hr.

Using the Eberline Instrument Corporation model E-120G Geiger Counter, The gamma-ray dose rate at the top and side of both the ^{252}Cf shipping container and the wax-boron shielding assembly was next recorded. The results are shown in Table II.

TABLE II

	Dose Rate Side Surface (mr/hr)	Dose Rate Top Surface (mr/hr)
Shipping Container	10	3
Shielding Assembly	0	1

At the ten foot distance near the detector, the gamma-ray dose rate was not measurable with this instrument.

Next, the effect of the ^{252}Cf on the Ge(Li) system was investigated by comparing two minute spectra taken with the source in and out of room 11. The summed activity of 25 channel groups are shown in Table III.

Table III

Channels	Total Counts ^{252}Cf in Room 11	Total Counts ^{252}Cf out of Room 11
1-25	48	36
26-50	44	36
51-75	60	43
100-125	60	50
200-225	36	32
300-325	29	21
400-425	12	9
500-525	11	11

These results show no significant increase in background because of the presence of the ^{252}Cf source in the room.

To insure adequate safety during my analysis, the following procedures are followed for the handling and use of the ^{252}Cf source.

1.) The analyzer room (Ward Hall Room 11) is kept locked at all times when the source is in the room.

2.) A radiation warning sign and written explaination are located outside the locked door to explain the reason for the door being locked.

3.) The wax-boron shielding assembly and analyzer areas are roped off from the rest of the room.

4.) The Texas Nuclear Model No 9146 Count Rate Meter is located within the roped off area to monitor for neutron activity.

I hope that this letter will answer any safety questions concerning my analysis, and I encourage you to make any further comments.

Thank you.

Sincerely,

William E. Kennedy Jr.

William E. Kennedy Jr.
Graduate Student
Department of Nuclear Engineering
Kansas State University

wek

cc Dr. M. Krick
Dr. N. D. Eckhoff

MEMO

TO: Reactor Safeguards Committee

FROM: M. S. Krick MSK

DATE: 27 May 1975

SUBJECT: Proposed reactor experiment #38: Detection of Short-Lived Fission Products of ^{235}U by Cyclic Activation

A copy of proposed experiment #38, revised according to the recommendations produced in the reactor safeguards committee meeting #77, is attached. Please indicate your approval or disapproval below and return this polling memo.

I approve_____ disapprove_____ proposed reactor experiment #38.

Signed_____

12.39 EXPERIMENT 38 - DETECTION OF SHORT-LIVED FISSION PRODUCTS OF ^{235}U BY CYCLIC ACTIVATION

OBJECT. To determine gamma energies and half-lives of short lived fission products of ^{235}U by a cyclic activation technique using thermal column neutrons.

BACKGROUND. An encapsulated sample of ^{235}U containing 2.1017 gms of 93% enriched ^{235}U with a total capsule weight of 4.2671 gms. will be transferred between the thermal column of the reactor and a Ge(Li) detector automatically by a NOVA computer system. Spectra corresponding to different times after transfer will be generated for each cycle, and corresponding spectra will be summed over hundreds of cycles giving composite spectra which will be analyzed for fission product half-lives and gamma energies. To insure that the sample will not rupture during the analysis, it has been encapsulated with three distinct layers; the outer capsule is a small sealed polyethylene vial, the second capsule is a machined plexiglass container with a sealed lid, and the third is a region of paraffin molded around the ^{235}U metal strips. To prevent saturation of the detector, and for safe operation of the reactor, the power levels of the reactor will be kept as low as possible.

PROCEDURE. After the beam port is opened by the reactor operator and radiation surveys are run to insure as low an exposure to personnel and equipment as possible, the pneumatic transfer system activation terminal will be placed into the thermal column of the reactor, and the detector terminal will be placed over the top of the Ge(Li) detector. For each run, the NOVA computer will control the specific activation delay, data collection procedure, and cooling delay. After the pneumatic transfer system is tested for correct operating pressure, the reactor will be brought to the desired low power level, and the NOVA computer will be programmed to exactly repeat the cycle hundreds of times. When all of the data are collected, the

corresponding composite spectra will be generated by a separate NOVA program. These spectra will be taken to the KSU IBM computer for spectral plots and counts per channel information for determination of gamma energies and half-lives. For an analysis of the amount of radioactive iodine fission product isotopes generated, a conservative over estimate of the power will be assumed of 1 kw. At the innermost position of the column, the flux has been determined to be 10.4×10^4 n/cm^2-sec-watt, or 1.04×10^8 n/cm^2-sec. at a power of 1 kw. The thermal fission cross section of ^{235}U is 577 b, and the sample weighs 2.1017 gms, so the total expected fission rate is calculated as:

$$(5.77 \times 10^{-22}\text{cm}^2)(1.04 \times 10^8 \text{ n/cm}^2\text{-sec})\left(\frac{2.1017 \text{ gms}}{235 \text{ g/gmole}}\right)(6.022 \times 10^{23}\text{at./gmole})(0.9311)$$

$$= 3.006 \times 10^8 \text{ fissions/sec.}$$

The saturated activity of a fission product can be expressed as:

$A_{sat} = \sum_i Y_i^j$ * Fission Rate, where Y_i^j is the fractional yield of the isotope i of the element j. For the isotopes of iodine, the calculation for this conservative case yield:

TABLE 1

$A_{SAT}^{I^{131}} = 2.36 \times 10^{-1}$ mCi

$A_{SAT}^{I^{132}} = 3.49 \times 10^{-1}$ mCi

$A_{SAT}^{I^{133}} = 5.28 \times 10^{-1}$ mCi

$A_{SAT}^{I^{134}} = 6.50 \times 10^{-1}$ mCi

$A_{SAT}^{I^{135}} = 5.20 \times 10^{-1}$ mCi

$\Sigma A_{SAT}^{I} = 2.283$ mCi

The summed activity produced at saturation for these isotopes is 2.283 mCi which is below the 5mCi limit specified in the technical specifications of the R-88 license. It should be noted that this case is for total saturation of the sample and this is an extremely conservative estimation. For the calculation of the maximum concentration in air of these isotopes in the case of

a complete capsule rupture, the saturation case is too conservative since the sample will only be activated less than 10% of the time of each cycle. Since the cycle time will be short compared to the half life of the shortest-lived iodine isotope, the maximum activity reached equals the ratio of the activation time to the cycle time times the saturation activity. The volume of the reactor bay from Appendix A of the TRIGA R-88 license is 144,000 ft^3 or 40.78 X $10^8 cm^3$, so that the resulting maximum concentrations at a factor of 10 below saturation are:

TABLE 2

ISOTOPE	CONCENTRATION IN AIR μCi/ml	RESTRICTED AREA 10CFR20 LIMIT μCi/ml
I^{131}	5.80 X 10^{-9}	9 X 10^{-9}
I^{132}	8.55 X 10^{-9}	2 X 10^{-7}
I^{133}	1.29 X 10^{-8}	3 X 10^{-8}
I^{134}	1.59 X 10^{-8}	5 X 10^{-7}
I^{135}	1.27 X 10^{-8}	1 X 10^{-7}

These calculations show that for this conservative case, the limits of 10CFR20 are not exceeded. If other sample masses or cyclemodes are desired, the experimenter will provide calculations showing that the 10CFR20 limits for iodine will not be exceeded.

REFERENCES.

1. Technical Specifications of the TRIGA R-88 reactor license.
2. "Experimental Neutron Flux Measurements and Power Calibration in the Kansas State University Triga Mark II Nuclear Reactor," A Master's Thesis by Gary Don Bouchey, Kansas State Universiyt, Manhattan, Kansas, 1967.
3. Nuclear Reactor Theory, John R. Lamarsh, New York University, (Addison-Wesley Publishing Company Inc., Reading, Massachusetts), 1966.
4. Title 10, part 20, Code of Federal Regulations of the Federal Register.

5. Personal Communications with Dr. M. S. Krick, 5/75, at Kansas State University.

William E. Kennedy Jr.

Appendix B.

The equipment settings for the data collection trials using both ^{252}Cf and reactor neutrons.

Equipment Settings for both the ^{252}Cf and Reactor Trials 1 and 2.

Device	Settings
Ortec-451	Fine Gain = 13.25
	Course Gain = 20.00
NS-623	Group Size 4096
	Conversion Gain = 4096
	Coupling = DC
	Lower Level = 1.2
	Upper Level = 10.0
	Zero Level = 0.15
	Anti Coincidence = analyze
NS-636	Memory Group = 1/1
	Sub Group = Off
	Mode = Computer
N_2 Gas Supply	Operating pressure = 20 psig

Equipment Settings for Trial 3

Device	Settings
Ortec-451	Fine Gain = 13.25
	Course Gain = 10.00
NS-623	Group Size = 4096
	Conversion Gain = 4096
	Coupling = DC
	Lower Level = 1.2
	Upper Level = 10.0
	Zero Level = 0.74
	Anti Coincidence = analyze
NS-636	Memory Group = 1/1
	Sub Group = Off
	Mode = Computer
N_2 Gas Supply	Operating Pressure = 20 psig

Appendix C

NOVA computer programs written in assembler language for process control and data reduction. The NSI-73-43-13-01 program was originally written as a pulse height analysis program controlling the NS-636 from teletype commands to the NOVA. Control of this experiment was achieved by writing interface subroutines which were activated by teletype commands. The teletype keys and the program descriptions are:

Key U; activation delay (timing constant at location 15555)

Key D; cooling delay (timing constant at location 15561)

Key R; starts a delay which allows data transfer to tape (timing constant at location 15562)

Key F; waits for the finish of the delay started by Key R

Key J; Generates the two-pulse transfer signal

Key Q; generates the one-pulse transfer signal.

The Summation of Spectra program generated the composite spectra for each trial by summing the corresponding spectra of each cycle.

The following program controls the delays activated by teletype keys U, D, and R to F.

```
      000425 .LOC 425
00425 015443 15443          key U
      015443 .LOC 15443     jump to 15443
15443 020512 LDA 0,K2
15444 040577 STA 0,K1       load AC's with
15445 024512 LDA 1,CT1      constant info.
15446 044512 STA 1,CT2
15447 024511 AD:  LDA 1,CT2 ---+
15450 020573      LDA 0,K1     |
15451 040505      STA 0,K3     |
15452 020504      LDA 0,K3     |  waiting loop for all delays
15453 100404      SUB 0,1,SZR  |  this loop tests the difference
15454 000773      JMP AD       |  between two constants and when
15455 000354      JMP 354      |  zero is reached it jumps out to
      010762 .LOC 10762        |  preform the next teletype command
10762 002353 JMP @353 --------+
      000353 .LOC 353 --------+
00353 015644 15644            |
      015643 .LOC 15643       |
15643 077777 K1: 77777        |  these commands enable the clock
      015644 .LOC 15644       |  service routine and decrement the
15644 014777 DSZ K1           |  delay once for each clock pulse
15645 060077 NIO CPU          |  which is compared above for a
15646 002352 JMP @352         |  zero value
      000352 .LOC 352         |
00352 011035 11035 ---------- +
      015555 .LOC 15555
15555 000144 K2: 144
15556 000000 K3: 0
15557 000001 CT1: 1
15560 000000 CT2: 0
15561 000144 K4: 144
15562 000062 K5: 62
      000404 .LOC 404 --------+
00404 015500 15500            |  these commands are from key D and
      015500 .LOC 15500       |  decrement the proper constant once
15500 020461 LDA 0,K4         |  for each clock pulse
15501 040542 STA 0,K1         |
15502 024455 LDA 1,CT1        |
15503 044455 STA 1,CT2        |
15504 000743 JMP AD ----------+
      000422 .LOC 422 --------+
00422 015570 15570            |
      015570 .LOC 15570       |  these commands are for key R which
15570 020772 LDA 0,K5         |  enables the decrementing routine
15571 040452 STA 0,K1         |  without waiting for the finish allowing
15572 000354 JMP 354 ---------+  other commands to operate
      000406 .LOC 406 --------+
00406 015600 15600            |
      015600 .LOC 15600       |  These commands enable key F which
15600 024757 LDA 1,CT1        |  waits for the finish of the decrementing
15601 044757 STA 1,CT2        |  started by key R
15602 000645 JMP AD ----------+
            .END
```

```
AD      015447
CT1     015557
CT2     015560
K1      015643
K2      015555
K3      015556
K4      015561
K5      015562
```

This page shows the constant locations for the previous program.

```
      000421 .LOC 421
000421 015411      15411
      015407 .LOC 15407
15407 000002 M1: 2
15410 000000 M2: 00
      000031 CNTRL = 31
15411 060231 PT:    NIOC CNTRL
15412 060077        NIO CPU
15413 060077        NIO CPU
15414 060077        NIO CPU
15415 024772        LDA 1,M1
15416 066031        DOB 1,CNTRL
15417 060077        NIO CPU
15420 060077        NIO CPU
15421 060077        NIO CPU
15422 060131        NIOS CNTRL
15423 060077        NIO CPU
15424 060077        NIO CPU
15425 060077        NIO CPU
15426 060231        NIOC CNTRL
15427 060077        NIO CPU
15430 060077        NIO CPU
15431 060077        NIO CPU
15432 024756        LDA 1,M2
15433 066031        DOB 1,CNTRL
15434 060077        NIO CPU
15435 060077        NIO CPU
15436 060077        NIO CPU
15437 060077        NIO CPU
15440 060077        NIO CPU
15441 000354        JMP 354
                .END
```

The following program generates one plotter enable pulse used for control of the pneumatic system.

---

when the pulse is generated, the device is disabled and the program returns to service the next teletype command

```
CNTRL   000031
M1      015407
M2      015410
PT      015411
```


This page contains the addresses of the constants used to generate one plotter command.

```
       000412 .LOC 412
00412  005473 5473
       005466 .LOC 5466
05466  000002 K1: 2
05467  000000 K2: 00
       000031 CNTRL = 31
05470  006100 K5: 100
05471  000000 K7: 0
05472  060231 PT2:  NIOC CNTRL
05473  060077       NIO CPU
05474  060077       NIO CPU
05475  060077       NIO CPU
05476  024770       LDA 1,K1
05477  060031       DOB 1,CNTRL
05500  060077       NIO CPU
05501  060077       NIO CPU
05502  060077       NIO CPU
05503  060131       NIOS CNTRL
05504  060077       NIO CPU
05505  060077       NIO CPU
05506  060077       NIO CPU
05507  060231       NIOC CNTRL
05510  060077       NIO CPU
05511  060077       NIO CPU
05512  060077       NIO CPU
05513  024754       LDA 1,K2
05514  060031       DOB 1,CNTRL
05515  024753       LDA 1,K5
05516  044753       STA 1,K7
05517  014752       DSZ K7
05520  000777       JMP .-1
05521  060231       NIOC CNTRL
05522  060077       NIO CPU
05523  060077       NIO CPU
05524  060077       NIO CPU
05525  024741       LDA 1,K1
05526  060031       DOB 1,CNTRL
05527  060077       NIO CPU
05530  060077       NIO CPU
05531  060077       NIO CPU
05532  060131       NIOS CNTRL
05533  060077       NIO CPU
05534  060077       NIO CPU
05535  060077       NIO CPU
05536  060231       NIOC CNTRL
05537  060077       NIO CPU
05540  060077       NIO CPU
05541  060077       NIO CPU
05542  024725       LDA 1,K2
05543  060031       DOB 1,CNTRL
05544  060077       NIO CPU
05545  060077       NIO CPU
05546  060077       NIO CPU
05547  060077       NIO CPU
05550  000354       JMP 354
                .END
```

The following program generates two plotter enable pulses used in control of the pneumatic transfer system

these commands enable the NS-636 in the plotter mode and generate one pulse, then the plotter is disabled and a loop waits about 200 microseconds before re-enabling the plotter for the second pulse.

when finished, the program jumps back to service the next teletype command

```
CNTRL   000031
K1      005466
K2      005467
K5      005470
K7      005471
PT2     005472
```

This page contains the addresses of the constants used to enable the Plotter command.

```
        000100 .LOC 100
        000031 CNTRL=31
        000030 DAC=30
        000022 NSMC=22
        000032 MISS=32
        000033 MPER=33
00100   060277 INTDS
00101   024143 LIMP:  LDA 1,ONE
00102   060031        DOB 1,CNTRL
00103   020144        LDA 0,ZERO
00104   024154         LDA 1,TML
00105   044020        STA 1,20
00106   042020 LOOP:  STA 0,@20,0
00107   014145        DSZ MEMOR
00110   000106        JMP LOOP
00111   020146        LDA 0,MEM
00112   040145        STA 0,MEMOR
00113   004156        JSR ENROL
00114   020147        LDA 0,INITL
00115   040150         STA 0,REFER
00116   004166 OK:    JSR KEY
00117   004226        JSR READT
00120   020151        LDA 0,SIX
00121   024150        LDA 1,REFER
00122   123000        ADD 1,0
00123   040150        STA 0,REFER
00124   030155        LDA 2,LAST
00125   112436        SUBZ# 0,2,SEZ
00126   000116        JMP OK
00127   112415        SUB# 0,2,SNR
00130   000116        JMP OK
00131   004156        JSR ENROL
00132   060077        NIO CPU
00133   000305        JMP DSALE
00134   014152 HERE:  DSZ SOX
00135   000137        JMP BK1
00136   002153        JMP @SALE1
00137   010147 BK1:   ISZ INITL
00140   000101        JMP LIMP
00141   060077        NIO CPU
00142   060077        NIO CPU
00143   000001 ONE:   1
00144   000000 ZERO:   0
00145   010000 MEMOR: 10000
00146   010000 MEM:   10000
00147   000002 INITL: 2
00150   000000 REFER: 0
00151   000006 SIX:   6
00152   000006 SOX:   6
00153   010463 SALE1: FINAL
00154   000462 TML:   TOTAL
00155   001131 LAST:  1131
00156   060231 ENROL:NIOC CNTRL
00157   063632        SKPDN MISS
00160   000157        JMP .-1
00161   060133        NIOS MPER
00162   063632        SKPDN MISS
00163   000162        JMP .-1
00164   060222        NIOC NSMC
```

This is the summation of spectra program for the NOVA 1220 computer written by Hector Ocampo and William Kennedy.

this loop zeros 4096 memory locations

MAIN PROGRAM
subroutine key hunts on the tape the proper tagword
READT reads the proper tagword
the tagword is incremented by SIX

If the tagword is the last one, increment the initial tagword and rewind the tape to deposit the composit data. If not, continue.

DSALE deposits the data at the proper tagword.
If all of the spectra for the trial are complete, the program stops. If not, it continues by zeroing the mempry again.

These are the constants used in setting the program information for a given trial.

ENROL rewinds the magnetic tape unit.

```
00165 001400          JMP 0,3
00166 054225 KEY:   STA 3,SALE2
00167 060231 NIOC CNTRL
00170 060230 NIOC DAC
00171 060222 NIOC NSMG
00172 030220 LDA 2,MGIN
00173 072031 DOB 2,CNTRL
00174 063730 TAGWD:SKPDZ DAC
00175 000174 JMP .-1
00176 063431 SKPBN CNTRL
00177 000176 JMP .-1
00200 074430 DIA 3,DAC
00201 071430 DIB 2,DAC
00202 175200 MOVR 3,3
00203 014221 DSZ T12
00204 000202 JMP .-2
00205 151100 MOVL 2,2
00206 014223 DSZ T4
00207 000205 JMP .-2
00210 157000 ADD 2,3
00211 024222 LDA 1,T12C
00212 044221 STA 1,T12
00213 024224 LDA 1,T4C
00214 044223 STA 1,T4
00215 116414 SUB# 0,3,SZR
00216 000174 JMP TAGWD
00217 002225 JMP @SALE2
00220 000015 MGIN: 15
00221 000014 T12: 14
00222 000014 T12C: 14
00223 000004 T4: 4
00224 000004 T4C: 4
00225 000000 SALE2: 0
00226 054304 READT: STA 3,SALE4
00227 060222         NIOC NSMG
00230 060231         NIOC CNTRL
00231 024277         LDA 1,UNO
00232 067031         DOC 1,CNTRL
00233 060230         NIOC DAC
00234 030300         LDA 2,MIN
00235 072131         DOBS 2,CNTRL
00236 063531         SKPBZ CNTRL
00237 000236         JMP .-1
00240 060131         NIOS CNTRL
00241 063631         SKPBN CNTRL
00242 000240         JMP .-2
00243 063731         SKPDZ CNTRL
00244 000243         JMP .-1
00245 060222         NIOC NSMG
00246 060131         NIOS CNTRL
00247 063631         SKPBN CNTRL
00250 000246         JMP .-2
00251 066031         DOB 1,CNTRL
00252 030154         LDA 2,TML
00253 054021         STA 2,21
00254 054022         STA 2,22
00255 024303 LOOP1:LDA 1,CAN
00256 067030         DOC 1,DAC
00257 060330         NIOP DAC
00260 063530         SKPBZ DAC
```

Subroutine KEY searches the tape for the given tagword and jumps back to the main program when it finds it.

Here are the constants for KEY.

Subroutine READT reads the correct spectrum into the NS-636 memory.

```
00261 000260          JMP .-1
00262 064430          DIA 1,DAC
00263 036021          LDA 3,.21,0
00264 167000          ADD 3,1
00265 046022          STA 1,.22,0
00266 010363          ISZ CAN
00267 014362          DSZ EFFEC
00270 000255          JMP LOOP1
00271 060231          NIOC CNTRL
00272 020361          LDA 0,CHANL
00273 040362          STA 0,EFFEC
00274 024144          LDA 1,ZERO
00275 044363          STA 1,CAN
00276 002364          JMP .SALE4
00277 000001 UNO: 1
00300 000015 MIN: 15
00301 010000 CHANL: 10000
00302 010000 EFFEC: 10000
00303 000000 CAN: 0
00304 000000 SALE4: 0
00305 060231 DSALE:NIOC CNTRL
00306 022550          LDA 0,.INIT
00307 024541          LDA 1,UNO1
00310 122400          SUB 1,0
00311 006543          JSR .KEE
00312 024547          LDA 1,MAG1
00313 066131          DOBS 1,CNTRL
00314 063531          SKPBZ CNTRL
00315 000314          JMP .-1
00316 060131          NIOS CNTRL
00317 063631          SKPDN CNTRL
00320 000316          JMP .-2
00321 063731          SKPDZ CNTRL
00322 000321          JMP .-1
00323 060231          NIOC CNTRL
00324 063632          SKPDN MISS
00325 000324          JMP .-1
00326 060333          NIOP MPER
00327 063632          SKPDN MISS
00330 000327          JMP .-1
00331 060222          NIOC NSMG
00332 020516          LDA 0,UNO1
00333 062131          DOBS 0,CNTRL
00334 063631          SKPDN CNTRL
00335 000334          JMP .-1
00336 060131          NIOS CNTRL
00337 063631          SKPDN CNTRL
00340 000336          JMP .-2
00341 000465          JMP CLEAR
00342 024154 THERE: LDA 1,TML
00343 044023          STA 1,23
00344 020506 LOOP2: LDA 0,FIRST
00345 063630          DOC 0,DAC
00346 060330          NIOP DAC
00347 063530          SKPBZ DAC
00350 000347          JMP .-1
00351 032023          LDA 2,.23,0
00352 071630          DOA 2,DAC
00353 060130          NIOS DAC
00354 063530          SKPBZ DAC
```

These constants are for the data handleing subroutine READT.

Subroutine DSALE searches the tape for the proper tagword and deposits the contents of 4096 memory locations on the magnetic tape.

```
00356 010474          ISZ FIRST
00357 014467          DSZ LOCAL
00360 000344          JMP LOOP2
00361 032475          LDA 2,.INIT
00362 151100          MOVL 2,2
00363 014475          DSZ TT12
00364 000362          JMP .-2
00365 060231          NIOC CNTRL
00366 060077          NIO CPU
00367 060077          NIO CPU
00370 060077          NIO CPU
00371 024456          LDA 1,CERO
00372 020456          LDA 0,UNO1
00373 062031          DOB 0,CNTRL
00374 063531          SKPBZ CNTRL
00375 000374          JMP .-1
00376 071030          DOA 2,DAC
00377 066030          DOB 1,DAC
00400 060077          NIO CPU
00401 060231          NIOC CNTRL
00402 060077          NIO CPU
00403 020446          LDA 0,MAGOF
00404 062131          DOBS 0,CNTRL
00405 063531          SKPBZ CNTRL
00406 000777          JMP .-1
00407 060131          NIOS CNTRL
00410 063631          SKPDN CNTRL
00411 000776          JMP  .-2
00412 063731          SKPDZ CNTRL
00413 000777          JMP .-1
00414 060231          NIOC CNTRL
00415 060077          NIO CPU
00416 060222          NIOC NSMG
00417 020426          LDA 0,CANNL
00420 040426          STA 0,LOCAL
00421 020426          LDA 0,CERO
00422 040430          STA 0,FIRST
00423 020434          LDA 0,TT2
00424 040434          STA 0,TT12
00425 000134          JMP HERE
00426 024421 CLEAR:LDA 1,CERO
00427 030420          LDA 2,CERO
00430 020420          LDA 0,UNO1
00431 107000 NEXT:  ADD 0,1
00432 067030          DOC 1,DAC
00433 071030          DOA 2,DAC
00434 060130          NIOS DAC
00435 063530          SKPBZ DAC
00436 000777          JMP .-1
00437 014405          DSZ CANAL
00440 000771          JMP NEXT
00441 024404          LDA 1,CANNL
00442 044402          STA 1,CANAL
00443 000342          JMP THERE
00444 010000 CANAL:  10000
00445 010000 CANNL:  10000
00446 010000 LOCAL:  10000
00447 000000 CERO:  0
00450 000001 UNO1:  1
```

After the data is deposited on the magnetic tape, the computer memory is cleared and the control constants are incremented to start the summing process again

```
00451 000005 MAGOT: 5
00452 000000 FIRST: 0
00453 000000 SALE6: 0
00454 000166 KEE: KEY
00455 000156 ENRO: ENROL
00456 000147 INIT: INITL
00457 000014 TT2: 14
00460 000014 TT12: 14
00461 000015 MAGI: 15
      010001 TOTAL: .BLK 10001
10463 000401 FINAL: JMP .+1
            .END
```

This page contains the constants used in the previous program and their addresses.

ERT	000137
CAN	000303
CANAL	000444
CANNL	000445
CERO	000447
CHANL	000301
CLEAR	000426
CNTRL	000031
DAC	000030
DSALE	000305
EFFEC	000302
ENRO	000455
ENROL	000156
FINAL	010463
FIRST	000452
HERE	000134
INIT	000456
INITL	000147
KEE	000454
KEY	000166
LAST	000155
LIMP	000101
LOCAL	000446
LOOP	000106
LOOP1	000255
LOOP2	000344
MAG1	000461
MAGOF	000451
MEM	000146
MEMOR	000145
MGIN	000220
MIN	000300
MISS	000032
MPER	000033
NEXT	000431
NSMG	000022
OK	000116
ONE	000143
READT	000226
REFER	000150
SALE1	000153
SALE2	000225
SALE4	000304
SALE6	000453
SIX	000151
SOX	000152
T12	000221
T12C	000222
T4	000223
T4C	000224
TAGWD	000174
THERE	000342
TML	000154
TOTAL	000462
TT12	000460
TT2	000457
UNO	000277
UNO1	000450
ZERO	000144

Appendix D

The IBM 370/158 computer programs. Printouts from TRAPL which was used to print, plot, and punch computer cards of the composite spectral data, GASP-4 which was used to plot overlapping 1024 pieces of each spectra from the punched data cards, and the least squares program which gave quadratic or linear fit for energy calibration to the gamma-ray data, all follow in this appendix.

```
$JOB            NDE,TIME=100,PAGES=100                                  94
      DIMENSION X(100),Y(100),SIGMAY(100),M(10), YFIT(100),A(10),SIGMAA(
     1  10),R(10),TITLE(20)
   10 FORMAT( 8F10.2 )
   11 FORMAT( 20I4 )
   12 FORMAT( 2X,'A(',I4,') =',E15.6,'+OR-',E15.6,'  R(',I4,') =',E15.6)
   13 FORMAT( 2X,' CHISQR = ',E15.6/2X,' RMUL = ',E15.6/2X,'FTEST = ',
     1 E15.6)
   14 FORMAT( 1H1)
   15 FORMAT( 2X,' AO = ',E15.6,'+OR-',E15.6)
   16 FORMAT ( 9X,'X',16X,'Y',14X,'YFIT')
   17 FORMAT( 2X,E15.6,2X,E15.6,2X,E15.6)
   18 FORMAT(20A4)
   99 READ 11,NPTS,MODE,NTERMS
      IF(NPTS.EQ.0)CALL EXIT
      READ 18,(TITLE(I),I=1,20)
      READ 10,(X(I),Y(I),SIGMAY(I),I=1,NPTS)
      READ 11 ,(M(I),I=1,NTERMS)
      CALL REGRES(X,Y,SIGMAY,NPTS,NTERMS,M,MODE,YFIT,AO,A,SIGMAO,SIGMAA,
     1 R,RMUL,CHISQR,FTEST)
      PRINT 14
      PRINT 16,(TITLE(I),I=1,20)
      PRINT 15,AO,SIGMAO
      PRINT 12,(I,A(I),SIGMAA(I),I,R(I),I=1,NTERMS)
      PRINT 13,CHISQR,RMUL,FTEST
      PRINT 16
      PRINT 17,(X(I),Y(I),YFIT(I),I=1,NPTS)
      GO TO 99
      END
C        SUBROUTINE REGRES
C
C        PURPOSE
C          MAKE A MULTIPLE LINEAR REGRESSION FIT TO DATA WITH A SPECIFIE
C             FUNCTION WHICH IS LINEAR IN COEFFICIENTS
C
C        USAGE
C          CALL REGRES(X,Y,SIGMAY,NPTS,NTERMS,M,MODE,YFIT,
C             AO,A,SIGMAO,SIGMAA,R,RMUL,CHISQR,FTEST)
C
C        DESCRIPTION OF PARAMETERS
C          X       - ARRAY OF DATA POINTS FOR INDEPENDENT VARIABLE
C          Y       - ARRAY OF DATA POINTS FOR DEPENDENT VARIABLE
C          SIGMAY  - ARRAY OF STANDARD DEVIATIONS FOR Y DATA POINTS
C          NPTS    - NUMBER OF SETS OF DATA POINTS
C          NTERMS  - NUMBER OF COEFFICIENTS
C          M       - ARRAY OF INCLUSION/REJECTION CRITERIA FOR FOLL
C          MODE    - DETERMINES METHOD OF WEIGHTING, LEAST-SQUARES FIT
C                    +1 (INSTRUMENTAL) WEIGHT(I) = 1./SIGMAY(I)**2
C                    0 (NO WEIGHTING)  WEIGHT(I) = 1.
C                    -1 (STATISTICAL)  WEIGHT(I) = 1./Y(I)
C          YFIT    - ARRAY OF CALCULATED VALUES OF Y
C          AO      - CONSTANT TERM
C          A       - ARRAY OF COEFFICIENTS
C          SIGMAO  - STANDARD DEVIATION OF AO
C          SIGMAA  - ARRAY OF STANDARD DEVIATIONS FOR COEFFICIENTS
C          R       - ARRAY OF LINEAR CORRELATION COEFFICIENTS
C          RMUL    - MULTIPLE LINEAR CORRELATION COEFFICIENT
C          CHISQR  - REDUCED CHI SQUARE FOR FIT
C          FTEST   - VALUE OF F FOR TEST OF FIT
C
```

```
C          SUBROUTINES AND FUNCTION SUBPROGRAMS REQUIRED                  95
C            FCTN(X,I,J,M)
C               EVALUATES THE FUNCTION FOR THE JTH TERM AND ITH DATA POINT
C               USING THE ARRAY M TO SPECIFY TERMS IN THE FUNCTION
C            MATINV(ARRAY,NTERMS,DET)
C               INVERTS A SYMMETRIC TWO-DIMENSIONAL MATRIX OF DEGREE
C               NTERMS AND CALCULATES ITS DETERMINANT
C
C          COMMENTS
C            DIMENSION STATEMENT VALID FOR NPTS UP TO 100 AND NTERMS UP TO
C               10
C
C          FROM P.R. BEVINGTON,DATA REDUCTION AND ERROR ANALYSIS FOR THE
C               PHYSICAL SCIENCES,MCGRAW-HILL,1969,P.172-MODIFIED BY
C               N.D. ECKHOFF,10/27/73
C
      26      SUBROUTINE REGRES(X,Y,SIGMAY,NPTS,NTERMS,M,MODE,YFIT,
             1 AO,A,SIGMAO,SIGMAA,R,RMUL,CHISQR,FTEST)
      27      DOUBLE PRECISION ARRAY,SUM,YMEAN,SIGMA,CHISQ,XMEAN,SIGMAX,DSQRT
      28      DOUBLE PRECISION FTEST
      29      DIMENSION X(100),Y(100),SIGMAY(100),M(10),YFIT(100),A(10),SIGMAA(
             1 10),WEIGHT(100),XMEAN(10),SIGMAX(10),ARRAY(10,10),R(10)
C
C            INITIALIZE SUMS AND ARRAYS
C
      30    11 SUM=0
      31       YMEAN=0
      32       SIGMA=0
      33       CHISQ=0
      34       RMUL=0
      35       DO 17 I=1,NPTS
      36    17 YFIT(I)=0
      37    21 DO 28 J=1,NTERMS
      38       XMEAN(J)=0
      39       SIGMAX(J)=0
      40       R(J)=0
      41       A(J)=0
      42       SIGMAA(J)=0
      43       DO 28 K=1,NTERMS
      44    28 ARRAY(J,K)=0
C
C            ACCUMULATE SUMS
C
      45    30 DO 50 I=1,NPTS
      46    31 IF(MODE) 32,37,39
      47    32 IF(Y(I)) 35,37,33
      48    33 WEIGHT(I)=1./Y(I)
      49       GO TO 41
      50    35 WEIGHT(I)=1./(-Y(I))
      51       GO TO 41
      52    37 WEIGHT(I)=1.
      53       GO TO 41
      54    39 WEIGHT(I)=1./SIGMAY(I)**2
      55    41 SUM=SUM+WEIGHT(I)
      56       YMEAN=YMEAN+WEIGHT(I)*Y(I)
      57       DO 44 J=1,NTERMS
      58    44 XMEAN(J)=XMEAN(J)+WEIGHT(I)*FCTN(X,I,J,M)
      59    50 CONTINUE
      60    51 YMEAN=YMEAN/SUM
```

```
      DO 53 J=1,NTERMS
   53 XMEAN(J)=XMEAN(J)/SUM
      FNPTS=NPTS
      WMEAN=SUM/FNPTS
      DO 57 I=1,NPTS
   57 WEIGHT(I)=WEIGHT(I)/WMEAN
C
C        ACCUMULATE MATRICES R AND ARRAY
C
   61 DO 67 I=1,NPTS
      SIGMA=SIGMA+WEIGHT(I)*(Y(I)-YMEAN)**2
      DO 67 J=1,NTERMS
      SIGMAX(J)=SIGMAX(J)+WEIGHT(I)*(FCTN(X,I,J,M)-XMEAN(J))**2
      R(J)=R(J)+WEIGHT(I)*(FCTN(X,I,J,M)-XMEAN(J))*(Y(I)-YMEAN)
      DO 67 K=1,J
   67 ARRAY(J,K)=ARRAY(J,K)+WEIGHT(I)*(FCTN(X,I,J,M)-XMEAN(J))*
     1 (FCTN(X,I,K,M)-XMEAN(K))
   71 FREE1=NPTS-1
   72 SIGMA=DSQRT(SIGMA/FREE1)
      DO 78 J=1,NTERMS
   74 SIGMAX(J)=DSQRT(SIGMAX(J)/FREE1)
      R(J)=R(J)/(FREE1*SIGMAX(J)*SIGMA)
      DO 78 K=1,J
      ARRAY(J,K)=ARRAY(J,K)/(FREE1*SIGMAX(J)*SIGMAX(K))
   78 ARRAY(K,J)=ARRAY(J,K)
C
C        INVERT SYMMETRIC MATRIX
C
   81 CALL MATINV(ARRAY,NTERMS,DET)
      IF(DET) 101,91,101
   91 A0=0
      SIGMAC=0
      RMUL=0
      CHISQR=0
      FTEST=0
      GO TO 150
C
C        CALCULATE COEFFICIENTS,FIT,AND CHI SQUARE
C
  101 A0=YMEAN
  102 DO 108 J=1,NTERMS
      DO 104 K=1,NTERMS
  104 A(J)=A(J)+R(K)*ARRAY(J,K)
  105 A(J)=A(J)*SIGMA/SIGMAX(J)
  106 A0=A0-A(J)*XMEAN(J)
  107 DO 108 I=1,NPTS
  108 YFIT(I)=YFIT(I)+A(J)*FCTN(X,I,J,M)
  111 DO 113 I=1,NPTS
      YFIT(I)=YFIT(I)+A0
  113 CHISQ=CHISQ+WEIGHT(I)*(Y(I)-YFIT(I))**2
      FREEN=NPTS-NTERMS-1
  115 CHISQR=CHISQ*WMEAN/FREEN
C
C        CALCULATE UNCERTAINTIES
C
  121 IF(MODE) 122,124,122
  122 VARNCE=1./WMEAN
      GO TO 131
  124 VARNCE=CHISQR
  131 DO 133 J=1,NTERMS
```

```
      132 SIGMAA(J)=ARRAY(J,J)*VARNCE/(FREE1*SIGMAX(J)**2)
          SIGMAA(J)=SQRT(SIGMAA(J))
      133 RMUL=RMUL+A(J)*R(J)*SIGMAX(J)/SIGMA
          FREEJ=NTERMS
      135 FTEST = (RMUL/FREEJ)/((1.-RMUL)/FREEN)
      136 RMUL = SQRT(RMUL)
      141 SIGMAO = VARNCE/FNPTS
          DO 145 J=1,NTERMS
          SIGMAO = SIGMAO+XMEAN(J)**2*SIGMAA(J)**2
          DO 145 K=1,NTERMS
      145 SIGMAO=SIGMAO+VARNCE*XMEAN(J)*XMEAN(K)*ARRAY(J,K)/(FREE1*SIGMAX(J)
         1 *SIGMAX(K))
      146 SIGMAO=SQRT(SIGMAO)
      150 RETURN
          END
C         FUNCTION FCTN (POWER SERIES)
C
C         PURPOSE
C           EVALUATE TERMS OF POLYNOMIAL FUNCTION FOR REGRES SUBROUTINE
C              FCTN(X,I,J,JTERMS) = X(I)**JTERMS(J)
C
C         USAGE
C           RESULT = FCTN(X,I,J,JTERMS)
C
C         DESCRIPTION OF PARAMETERS
C           X       - ARRAY OF DATA POINTS FOR INDEPENDENT VARIABLE
C           I       - INDEX OF DATA POINTS
C           J       - INDEX OF TERM IN POLYNOMIAL FUNCTION
C           JTERMS  - ARRAY OF POWERS
C
C         SUBROUTINES AND FUNCTION SUBPROGRAMS REQUIRED
C           NONE
C
C         FROM P.R. BEVINGTON,DATA REDUCTION AND ERROR ANALYSIS FOR THE
C           PHYSICAL SCIENCES,MCGRAW-HILL,1969,P.177
C

      FUNCTION FCTN(X,I,J,JTERMS)
      DIMENSION X(100),JTERMS(10)
    1 JEXP=JTERMS(J)
      FCTN = X(I)**JEXP
      RETURN
      END
C         SUBROUTINE MATINV
C
C         PURPOSE
C           INVERT A SYMMETRIC MATRIX AND CALCULATE ITS DETERMINANT
C
C         USAGE
C           CALL MATINV(ARRAY,NORDER,DET)
C
C         DESCRIPTION OF PARAMETERS
C           ARRAY  - INPUT MATRIX WHICH IS REPLACED BY ITS INVERSE
C           NORDER - DEGREE OF MATRIX(ORDER OF DETERMINANT)
C           DET    - DETERMINANT OF INPUT MATRIX
C
C         SUBROUTINES AND FUNCTION SUBPROGRAMS REQUIRED
C           NONE
C
C         COMMENTS
```

```
      C          DIMENSION STATEMENT VALID FOR NORDER UP TO 10
      C
      C        FROM P.R. BEVINGTON,DATA REDUCTION AND ERROR ANALYSIS FOR THE
      C          PHYSICAL SCIENCES,MCGRAW-HILL,1969,P.302

  128       SUBROUTINE MATINV(ARRAY,NORDER,DET)
  129       DOUBLE PRECISION ARRAY,AMAX,SAVE,DABS
  130       DIMENSION ARRAY(10,10),IK(10),JK(10)
  131    10 DET=1.
  132    11 DO 100 K=1,NORDER
      C
      C        FIND LARGEST ELEMENT OF ARRAY(I,J) IN REST OF MATRIX
      C
  133       AMAX=0.
  134    21 DO 30 I=K,NORDER
  135       DO 30 J=K,NORDER
  136    23 IF(DABS(AMAX)-DABS(ARRAY(I,J))) 24,24,30
  137    24 AMAX=ARRAY(I,J)
  138       IK(K)=I
  139       JK(K)=J
  140    30 CONTINUE
      C
      C        INTERCHANGE ROWS AND COLUMNS TO PUT AMAX IN ARRAY(K,K)
      C
  141    31 IF(AMAX) 41,32,41
  142    32 DET=0.
  143       GO TO 140
  144    41 I=IK(K)
  145       IF(I-K) 21,51,43
  146    43 DO 50 J=1,NORDER
  147       SAVE = ARRAY(K,J)
  148       ARRAY(K,J) = ARRAY(I,J)
  149    50 ARRAY(I,J) = -SAVE
  150    51 J = JK(K)
  151       IF(J-K) 21,61,53
  152    53 DO 60 I=1,NORDER
  153       SAVE = ARRAY(I,K)
  154       ARRAY(I,K) = ARRAY(I,J)
  155    60 ARRAY(I,J) = -SAVE
      C
      C        ACCUMULATE ELEMENTS OF INVERSE MATRIX
      C
  156    61 DO 70 I=1,NORDER
  157       IF(I-K) 63,70,63
  158    63 ARRAY(I,K) = -ARRAY(I,K)/AMAX
  159    70 CONTINUE
  160    71 DO 80 I=1,NORDER
  161       DO 80 J=1,NORDER
  162       IF(I-K) 74,80,74
  163    74 IF(J-K) 75,80,75
  164    75 ARRAY(I,J) =ARRAY(I,J)+ARRAY(I,K)*ARRAY(K,J)
  165    80 CONTINUE
  166    81 DO 90 J=1,NORDER
  167       IF(J-K) 83,90,83
  168    83 ARRAY(K,J) = ARRAY(K,J)/AMAX
  169    90 CONTINUE
  170       ARRAY(K,K) = 1./AMAX
  171   100 DET = DET*AMAX
      C
      C        RESTORE ORDERING OF MATRIX
```

```
      C
  172   101 DO 130 L=1,NORDER                                          99
  173       K = NORDER-L+1
  174       J = IK(K)
  175       IF(J-K) 111,111,105
  176   105 DO 110 I=1,NORDER
  177       SAVE = ARRAY(I,K)
  178       ARRAY(I,K) = -ARRAY(I,J)
  179   110 ARRAY(I,J) = SAVE
  180   111 I = JK(K)
  181       IF(I-K) 130,130,113
  182   113 DO 120 J=1,NORDER
  183       SAVE = ARRAY(K,J)
  184       ARRAY(K,J) = -ARRAY(I,J)
  185   120 ARRAY(I,J) = SAVE
  186   130 CONTINUE
  187   140 RETURN
  188       END
```

```
      $ENTRY
```

0.5 TO 2.5 SEC. AFTER FISSION 5/29/75 100 CYCLES

```
 A0 =     0.387005E-01+0R-    0.184456E-01
A(    1) =    0.794909E-03+0R-    0.113715E-04   R(    1) =    0.999998E 00
A(    2) =    0.155723E-09+0R-    0.523943E-08   R(    2) =    0.997443E 00
 CHISQR =     0.453345E-06
 RMUL =     0.999999E 00
FTEST =     0.239674E 06
```

X	Y	YFIT
0.594080E 03	0.511000E 00	0.510995E 00
0.142724E 04	0.117332E 01	0.117354E 01
0.155411E 04	0.127500E 01	0.127445E 01
0.162751E 04	0.133251E 01	0.133284E 01

```
CORE USAGE        OBJECT CODE=    9672 BYTES,ARRAY AREA=    3284 BYTES,TOTAL AR

DIAGNOSTICS         NUMBER OF ERRORS=        0, NUMBER OF WARNINGS=        0, NUM

COMPILE TIME=     0.92 SEC,EXECUTION TIME=     0.00 SEC,  WATFIV - JUL 1973 VE
```

```
    12 TC 14 SEC. AFTER FISSION CALIB. POINTS
 AO =    0.372915E-01+OR-    0.496542E-02
A(   1) =   0.797259E-03+OR-    0.335590E-05  R(   1) =   0.999999E 00
A(   2) =  -0.946520E-09+OR-    0.150336E-08  R(   2) =   0.996983E 00
 CHISQR =    0.607633E-07
 RMUL =    0.999997E 00
FTEST =    0.151145E 06
        X                   Y                   YFIT
    0.594474E 03        0.511000E 00        0.510906E 00
    0.784039E 03        0.661640E 00        0.661791E 00
    0.142726E 04        0.117332E 01        0.117326E 01
    0.155507E 04        0.127500E 01        0.127479E 01
    0.162800E 04        0.133251E 01        0.133272E 01

CORE USAGE       OBJECT CODE=   10888 BYTES,ARRAY AREA=    3284 BYTES,TOTAL AR

DIAGNOSTICS        NUMBER OF ERRORS=       0, NUMBER OF WARNINGS=       0, NUM

COMPILE TIME=     0.99 SEC,EXECUTION TIME=     0.09 SEC,  WATFIV - JUL 1973 V1
```

```
     0.5 TO 5.5 SEC. AFTER FISSION 5/31/75 600 CYCLES
  AO =     0.343166E 00+OR-    0.316262E-01
 A(   1) =    0.117438E-02+OR-    0.332427E-04  R(   1) =    0.999200E 00
  CHISQR =     0.225412E-03
  RMUL =     0.999199E 00
 FTEST =     0.124751E 04
        X                    Y                   YFIT
    0.275143E 03        0.661640E 00        0.666287E 00
    0.692786E 03        0.117332E 01        0.115676E 01
    0.793032E 03        0.127500E 01        0.127448E 01
    0.853029E 03        0.133251E 01        0.134494E 01

 CORE USAGE        OBJECT CODE=     9672 BYTES,ARRAY AREA=     3284 BYTES,TOTAL AR

 DIAGNOSTICS         NUMBER OF ERRORS=        0, NUMBER OF WARNINGS=        0, NUM

 COMPILE TIME=      0.94 SEC,EXECUTION TIME=      0.04 SEC,  WATFIV - JUL 1973 V1
```

A0 = 0.752961E-01+0R- 0.140342E-02

A(1) = 0.160169E-02+0R- 0.158758E-05 R(1) = 0.999999E 00

CHISQR = 0.241429E-06

RMJL = 0.999999E 00

FTEST = 0.671086E 06

X	Y	YFIT
0.366134E 03	0.661640E 00	0.661731E 00
0.685300E 03	0.117332E 01	0.117293E 01
0.785110E 03	0.133251E 01	0.133280E 01

CORE USAGE OBJECT CODE= 9672 BYTES,ARRAY AREA= 3284 BYTES,TOTAL A

DIAGNOSTICS NUMBER OF ERRORS= 0, NUMBER OF WARNINGS= 0, NU

COMPILE TIME= 0.90 SEC,EXECUTION TIME= 0.01 SEC, WATFIV - JUL 1973 V

```
LEVEL  21                    MAIN                CATE = 75177          11/52/01

C ********************************** TRAPL ***************************************
C *                                                                              *
C *      TRAPL CCCE - TAPE REMCVAL AND PLCT CCDE                                 *
C *      BASED ON THE 'CARDS' CCDE BY M.S. KRICK AND J. DEVORE.                  *
C *      MCDIFIED BY N.D. ECKHCFF AND J.K. SHULTIS (5/15/73).                    *
C *      MCDIFIED BY F.H. PRATT  JANUARY 1974                                    *
C *                                                                              *
C *      THIS PRCGRAM IS USED TC REMOVE GAMMA SPECTRA FROM A MAGNETIC TAPE       *
C *      GENERATED BY THE KSUNES MULTIPARAMETER SYSTEM.  THIS CODE WILL PRINT    *
C *      PUNCH, OR PLOT THE SPECTRA CONTAINED IN SELECTED RECCRDS CN THE
C *      MAGNETIC TAPE.                                                          *
C *                                                                              *
C *      THE SPECTRA MAY BE 1024,2048,3072 CR 4096 CHANNELS IN SIZE. MIXING      *
C *      OF DIFFERENT SPECTRAL SIZES IS PERMITTED.                               *
C *                                                                              *
C *      EACH SPECTRUM REQUIRES CNE CATA CARD.                                   *
C *                                                                              *
C ********************************************************************************
      INTEGER IARRAY(4096), HEADER, WCRDS, HEAC(4), IARR(4096), IHELP(11
     1)
      LCGICAL*1 JARRAY(16384)
      EQUIVALENCE (IARRAY(1),JARRAY(1))
      DIMENSION Y(4098),BUF(1COC),TITLE(17)
C  $$$$$ INITIALIZE THE PLCTTING SUBROUTINE
      CALL PLCTS(BUF,1000)
      CALL PLCT(0.0,-14.0,3)
      CALL PLCT(C.C,-13.0,-3)
      ISTART=1
      READ(5,99) NCH
C  NCH = THE NUMBER CF CHANNELS AVERAGED CVER
      WRITE(6,98) NCH
   98 FORMAT(1HO,'RESULT IS AVERAGED OVER',I3,'   CHANNELS')
C
C  $$$$$ READ IN INPUT DATA FCR EACH CATA SET
   11 READ(5,99,END=24,ERR=23) WORDS,NCARCS,NPLCT,NRECRD,ITAG,(TITLE(I),
     1I=1,17)
C ***************************** INPUT CATA ***************************************
C *                                                                              *
C *    EACH SPECTRUM REQUIRES CNE CATA CARC WITH A FORMAT (I4,I2,2I1,I4,17A      *
C *    FCR THE FOLLOWING QUANTITIES:                                             *
C *         WORDS = NC CF CHANNELS IN SPECTRA                                    *
C *         NCARDS= 0 *** WILL PRODUCE PUNCHED CARDS                             *
C *               = 1 *** WILL NCT PRODUCE PUNCHED CARDS                         *
C *         NPLCT = 0 *** WILL NCT PLCT SPECTRUM                                 *
C *               = 1 *** WILL GIVE CNLY A LINEAR PLCT CF SPECTRUM               *
C *               = 2 *** WILL GIVE CNLY A SEMILOG PLOT CF SPECTRUM              *
C *               = 3 *** WILL GIVE BOTH A LINEAR AND SEMILCG PLOT CF SPECTRUM   *
C *         NRECRD= 0 IF ENTIRE SPECTRUM IS CONTAINED CN ONE TAPE RECORD         *
C *                 1 IF WCRDS > 1024 BUT TAPE HAS SPECTRUM IN 1024 CH RECORDS   *
C *         ITAG  = TAG NO. OF TAPE RECCRD                                       *
C *         TITLE = CAPTICN THAT IS TO APPEAR CN ANY CATA PLOT (68 CHARACTERS)   *
C *                                                                              *
C ********************************************************************************
   99 FORMAT(I4,I2,2I1,I4,17A4)
      IF(WORDS.EQ.0) GC TC 22
      LAST=4*WORDS
   16 CONTINUE
      DO 10 II=1,4096
```

```
      10   IARRAY(II)=0
C  $$$$$   READ A SPECTRUM
       9 READ(8,88)(JARRAY(L),L=2,4)
      88   FORMAT(3A1)
           HEADER=IARRAY(1)/4096
           IF(HEADER.EQ.4095)GOTO12
           HEAD(4)=MOD(HEADER,8)
           DO 300 II=2,4
           HEADER=HEADER/8
      300  HEAD(5-II)=MOD(HEADER,8)
           IF(ISTART.NE.1) GO TO 211
           JTAG=0
           DO 209 I=1,4
           IJ=5-I
           I1=I-1
           JTAG=HEAD(IJ)*10**I1+JTAG
      209  CONTINUE
           IF(JTAG.NE.ITAG) GO TO 9
           ISTART=ISTART+1
           IF (NRECRD) 211,211,212
      211  READ(8,87,END=210,ERR=210)((JARRAY(I+J),J=1,3),I=1,LAST,4)
           GO TO 210
      212  READ(8,87) ((JARRAY(I+J),J=1,3),I=1,4096,4)
           IF (WORDS.LE.1024) GO TO 210
           READ(8,88) (JARRAY(L),L=16382,16384)
           READ(8,87) ((JARRAY(I+J),J=1,3),I=4097,8192,4)
           IF (WORDS.LE.2048) GO TO 210
           READ(8,88) (JARRAY(L),L=16382,16384)
           READ(8,87) ((JARRAY(I+J),J=1,3),I=8193,12288,4)
           IF (WORDS.LE.3072) GO TO 210
           READ(8,88,END=210) (JARRAY(L),L=16382,16384)
           READ(8,87,END=210) ((JARRAY(I+J),J=1,3),I=12289,16384,4)
       87  FORMAT(128(128A1))
      111  FORMAT(2X,4I1,12I6,I2)
      112  FORMAT(2X,I3,I1,2X,10I7)
      113  FORMAT( /2X,'LENGTH OF SPECTRUM = ',I4,' CHANNELS',9X,'SPECTRUM ID
          1 (OCTAL NOTATION)=',4I1,/2X,'COUNT TIME (SEC) IS FOUND IN CHANNEL
          21',/)
      115  FORMAT( 2X,'CHNO',8X,'1',6X,'2',6X,'3',6X,'4',6X,'5',6X,'6',6X,'7'
          1,6X,'8',6X,'9',5X,'10',1X,'1 **CUT**' )
      116  FORMAT(1H1)
      210  CONTINUE
C  $$$$$  CHECK ON TAG NO.
           JTAG=0
           DO 225 I=1,4
           IJ=5-I
           I1=I-1
           JTAG=HEAD(IJ)*10**I1 + JTAG
      225  CONTINUE
           IF(JTAG.NE.ITAG) GO TO 16
           DO 226 M=2,WORDS
           IF(IARRAY(M).GT.838850) IARRAY(M)=0
      226  CONTINUE
C  AVERAGE OVER NCH CHANNELS
           IFIR=(NCH+1)/2
           ILAS=4096-NCH/2
           DO 232 I=IFIR,ILAS
           ISUM=0
```

```
      JFIR=I-NCH/2
      JLAS=I+NCH/2
      DO 230 J=JFIR,JLAS
  230 ISUM=ISUM+IARRAY(J)
  232 IARB(I)=ISUM/1
      DO 234 I=1,4096
  234 IARRAY(I)=IARB(I)
C  $$$$$   PRINT A SPECTRUM
      WRITE(6,116)
      WRITE(6,113) WORDS,HEAD
      WRITE(6,115)
      LLS=0
      II=0
      IL=WORDS/10.0
      LL10=0
      DO 201 LL=1,IL
      LL1=LL10+1
      LL10=LL1+9
      WRITE(6,112)LLS,II,(IARRAY(L),L=LL1,LL10)
  201 LLS=LL
      LL1=LL10+1
      IF(WORDS.GT.1024)GO TO 203
      LL10=1024
      GO TO 206
  203 IF(WORDS.GT.2048)GO TO 204
      LL10=2048
      GO TO 206
  204 IF(WORDS.GT.3072)GO TO 205
      LL10=3072
      GO TO 206
  205 LL10=4096
  206 WRITE(6,112)LLS,II,(IARRAY(L),L=LL1,LL10)
      WRITE(6,113) WORDS,HEAD
      IF (NCARDS.GT.0) GO TO 202
C  $$$$$   PUNCH A SPECTRUM
      LO=1
      IHI=12
      WRITE(7,111)HEAD
      NHI=WORDS/12
      DO 101 I=1,NHI
      WRITE(7,111)  HEAD,(IARRAY(L),L=LO,IHI),I
      LO=LO+12
  101 IHI=IHI+12
      ILO=12*NHI+1
      IL=NHI+1
      WRITE(7,117)  HEAD,(IARRAY(L),L=ILO,WORDS),IL
  117 FORMAT(2X,4I1,4I6,48X,I2)
  202 IF(NPLOT.EQ.0)GO TO 11
C  $$$$$ PLOT THE SPECTRUM
      DO 15 I=1,WORDS
   15 Y(I)=IARRAY(I)
      CALL GASP(Y,WORDS,NPLOT,TITLE)
      GO TO 11
C  $$$$$  END OF PROGRAM OPTIONS
   22 WRITE(6,501)
  501 FORMAT('1  NORMAL END - BLANK DATA CARD ENCOUNTERED')
      GO TO 199
   23 WRITE(6,502)
```

```
      502 FORMAT('1  DATA CARD ERROR')
          GO TO 199
       24 WRITE(6,503)
      503 FORMAT('1  UNEXPECTEDLY RAN OUT OF DATA CARDS')
          GO TO 199
       12 WRITE(6,200)
      200 FORMAT('1  NORMAL END - TAPE NUMBER 7777 ENCOUNTERED')
      199 CALL PLOT(20.0,0.0,999)
          REWIND 8
          STOP
              END
```

```
TRAN IV G LEVEL  21                GASP                DATE = 75177          11/52/01

21              SUBROUTINE GASP(Y,NPTS,LOGLIN,TITLE)
22              DIMENSION Y(4098),TITLE(17),X(4098),DX(4)
23              COMMON/AXICOM/MDEL
24              IF (LOGLIN.EQ.0) RETURN
25              MDEL=4
26              MULT=NPTS/1024
27              DO 4 I=1,NPTS
28            4 X(I)=I
29              IF (MULT-1) 10,10,11
30           11 XLENTH=MULT*6.00
31              YLENTH=8.00
32              GO TO 12
33           10 XLENTH=8.00
34              YLENTH=6.00
35           12 ADV=XLENTH + 6.00
36              X(NPTS+1)=0.0
37              X(NPTS+2)=NPTS/XLENTH
38              DX(1)=0.0
39              DX(2)=NPTS
40              CALL SCALE(DX,XLENTH,2,1)
41              TINC=DX(4)/X(NPTS+2)
          C*** LINEAR PLOT SECTION
42              IF (LOGLIN.EQ.2) GO TO 50
43              CALL SCALE(Y,YLENTH,NPTS,1)
44              CALL DAXIS(0.0,0.0,XLENTH,0.0,TINC,-1)
45              CALL SAXIS(X(NPTS+1),DX(4),0,1,-1,14HCHANNEL NUMBER,-14)
46              CALL AXIS(0.0,0.0,6HCOUNTS,6,YLENTH,90.0,Y(NPTS+1),Y(NPTS+2))
47              CALL LINE(X,Y,NPTS,+1,0,0)
48              CALL SYMBOL(0.0,-0.7,0.14,TITLE,0.0,68)
49              CALL PLOT(ADV,0.0,-3)
          C*** LOG PLOT SECTION
50           50 IF(LOGLIN.EQ.1) RETURN
51              DO 51 I=1,NPTS
52              IF (Y(I).LE.0.0) Y(I)=1.0
53           51 CONTINUE
54              CALL DAXIS(0.0,0.0,XLENTH,0.0,TINC,-1)
55              CALL SAXIS(X(NPTS+1),DX(4),0,1,-1,14HCHANNEL NUMBER,-14)
56              CALL SCALOG(Y,YLENTH,NPTS,1)
57              CALL LGAXIS(0.0,0.0,6HCOUNTS,6,YLENTH,90.0,Y(NPTS+1),Y(NPTS+2))
58              CALL LGLINE(X,Y,NPTS,1,0,0,+1)
59              CALL SYMBOL(7.4,-0.7,0.14,TITLE,0.0,68)
60              CALL DAXIS(0.0,9.0,XLENTH,0.0,TINC,+1)
61              CALL PLOT(ADV,0.0,-3)
62              RETURN
63              END
```

LENGTH OF SPECTRUM = 4096 CHANNELS SPECTRUM ID (OCTAL NOTATION)=0004
COUNT TIME (SEC) IS FOUND IN CHANNEL 1

CHNO	1	2	3	4	5	6	7	8	9	10
00	2322	75	0	0	0	0	0	0	0	0
10	0	0	0	0	0	0	0	0	0	0
20	0	0	0	0	0	0	0	0	0	0
30	0	0	0	0	0	0	0	0	0	0
40	0	0	0	0	0	0	0	0	0	0
50	0	0	0	0	0	0	0	0	0	0
60	0	0	0	0	0	0	0	0	0	0
70	0	0	0	0	0	0	0	0	0	0
80	0	0	0	0	0	0	0	0	0	0
90	0	0	0	0	0	0	0	0	0	0
100	0	0	0	0	0	0	0	0	0	0
110	0	0	0	0	0	0	0	0	0	0
120	0	0	0	0	0	0	0	0	0	0
130	0	0	0	0	0	0	0	0	0	0
140	0	0	0	0	0	0	0	0	0	0
150	0	0	0	0	0	0	0	0	0	0
160	0	0	0	0	0	0	0	0	0	0
170	0	0	0	0	0	0	0	0	0	0
180	0	0	0	0	0	0	0	0	0	0
190	0	0	0	0	0	0	0	0	0	0
200	0	0	0	0	0	0	0	0	0	0
210	0	0	0	0	0	0	0	0	0	0
220	0	0	0	0	0	0	0	0	0	0
230	0	0	0	1	0	0	0	0	0	0
240	22	848	29628	57425	59352	57104	58272	59736	58904	55925
250	55121	56376	56294	55300	55631	56052	55375	54347	52386	51942
260	52055	53038	54189	52137	51055	52513	53433	52478	52565	57543
270	61833	60563	60380	6603	17318	6147	59249	54420	51120	49093
280	48487	48514	48487	49290	52015	54313	53408	51379	53019	54314
290	62114	57259	1854	2936	55696	51011	52313	50165	47158	47326
300	47117	48336	50124	46883	44934	44643	45823	45416	46851	47545
310	45009	44436	43901	43343	42959	45043	47412	46423	43974	43656
320	47266	50485	50839	60025	2904	55369	46477	44911	43706	43272
330	47578	58178	61159	49267	42803	40733	40623	41111	42512	41961
340	40602	41850	42017	42286	44231	41177	40048	40747	39856	39937
350	39992	39644	39078	39332	38817	41493	46537	45144	41861	40761
360	40936	38724	37251	37615	38209	38184	39191	39772	38500	39228
370	36795	35316	36165	35769	34880	35107	35131	34573	34798	34996
380	34648	33763	34026	36280	39855	37169	34071	33707	34457	34661
390	37954	43839	39553	34248	33561	34042	33944	34025	33804	33663
400	37605	34082	32249	31919	31856	33405	33574	35145	41258	39651
410	32734	30633	30706	31026	31526	32650	32269	31597	31542	32125
420	35228	33537	30195	29365	29474	29392	28929	28942	28261	27916
430	27864	28048	27991	28204	29794	33533	34109	30276	29814	29320
440	31710	31858	29811	28699	29074	27895	27564	29930	30775	29911
450	27842	29750	31446	29315	28243	30421	29565	27804	27364	29297
460	28249	26071	25193	25350	25552	26272	26757	27228	25377	24646
470	24265	25054	25160	25428	26005	23850	35341	30303	27093	27319
480	28521	30014	23552	27620	30333	35177	35519	29124	24131	23385
490	23650	23608	23655	23760	23160	22941	23562	23776	23489	23571
500	23874	23980	24174	25569	27936	26601	23822	24206	25375	30236
510	28271	26061	26326	24230	22532	23162	25257	24190	23017	22239
520	21638	21777	22545	22952	22971	22754	22831	25440	25893	24644
530	25993	29420	25671	21403	21160	21505	21272	21060	20989	21579

540	22394	21687	20251	19601	19728	20516	22247	22899	21908	22355
550	23432	21848	21938	23430	21462	19591	19003	18725	13815	18432
560	18468	18698	19414	20103	20265	23084	26666	23563	20071	20379
570	20246	19172	18442	18335	18327	19089	19098	19631	18765	17993
580	17949	18258	18679	19378	18368	17562	17947	19145	22133	22609
590	20914	13837	17711	18150	17944	18200	19057	19821	18941	17413
600	18112	19204	22563	22381	18107	16554	16975	17315	17557	17269
610	16602	16111	15977	15853	15963	16038	16090	15860	15697	14261
620	16719	17176	16812	16470	15853	15794	15610	15863	16881	16329
630	15942	16766	17187	16349	15566	15364	15414	15044	15454	15330
640	15748	16785	17117	17407	16268	15237	15218	15569	15741	15955
650	16350	15870	15113	15065	15348	16444	18952	21044	18359	15244
660	15109	14538	14264	14620	14753	14581	14467	14691	15204	15518
670	15038	14636	14089	14162	14151	15130	14914	13988	13419	13642
680	13482	13403	13500	14025	14021	13582	13453	13769	13543	14047
690	15464	18755	18501	15225	13412	12936	12952	13116	12939	12865
700	12949	12870	12640	12606	13067	13956	14452	13840	13533	13926
710	14495	14479	13583	12672	12478	12046	12136	12173	12640	12866
720	12931	12500	12188	12103	12272	12016	11458	11295	11187	10829
730	11020	10993	11147	10854	10769	10843	11319	11892	13558	14737
740	12536	11368	10675	10651	10577	10421	10984	10614	10340	10409
750	10739	10795	10950	10827	10712	11157	11142	11093	10919	10672
760	10591	10775	10425	10097	9637	9672	10051	10117	10350	9901
770	9649	9485	10039	10208	10153	10207	10393	11008	12327	15017
780	15474	12330	10095	9515	10419	10733	10556	9856	9372	9528
790	10200	12533	14236	12099	9636	8794	8447	8604	8541	8558
800	8583	8481	8262	8354	8344	8235	8439	8506	8455	8605
810	8388	8522	8284	8262	8326	8378	8346	8291	8526	8745
820	8883	8855	9252	10368	11698	11063	9428	8699	8437	8379
830	8117	7924	7724	7793	7690	7811	7718	7576	7755	7742
840	7594	7686	8204	8483	8294	8206	8805	9231	9261	9771
850	11652	15713	18751	13882	10047	10114	11906	13669	11122	8545
860	7586	7431	7133	7216	7000	7337	7128	7064	7215	7460
870	7477	7395	7114	6982	7110	7132	7206	6995	7054	7138
880	7562	7503	7173	6936	6756	6829	6364	6592	6664	6690
890	6706	6777	6804	6589	6705	6710	6733	7183	7432	7275
900	6942	6696	6452	6486	6607	6673	6626	6750	6540	6630
910	6546	6917	7088	7179	7038	6777	7075	7270	7559	7447
920	7643	7901	7053	6808	6597	6445	6408	6554	6499	6591
930	6558	6305	6119	6222	6013	6099	6282	6986	7571	6991
940	6118	5988	5758	5872	5984	5834	6165	6068	5860	5568
950	5691	5554	5406	5567	5640	5613	5637	5605	5654	5605
960	5622	5594	5508	5589	5474	5594	5698	5811	6154	6075
970	6188	6031	5754	5553	5625	5609	5448	5535	5466	5570
980	5782	5934	5882	5647	5429	5432	5460	5542	5539	5414
990	5356	5143	5257	5117	5321	5258	5339	5403	5264	5291
1000	5099	5328	5317	5386	5197	5298	5385	5404	5291	5149
1010	5176	5239	5299	5369	5320	5054	4980	5148	5185	5345
1020	5454	5411	5208	5152	5110	5087	5024	5206	5091	4924
1030	4895	4929	4939	4940	4834	4986	4912	4992	4938	4921
1040	4878	5179	5192	5106	4920	4834	4923	4903	5004	4876
1050	4948	4759	4688	4982	5348	5008	5524	4924	4776	4849
1060	4936	4745	4660	4815	4885	5419	5797	5744	4993	4710
1070	4614	4740	4747	5071	4912	4711	4466	4404	4485	4548
1080	4466	4564	4530	4450	4364	4220	4317	4271	4376	4373
1090	4619	4608	4590	4397	4291	4205	4220	4113	4281	4225
1100	4256	4209	4115	4199	4097	4202	4111	4142	4296	4334
1110	4225	4167	4202	4153	4192	4131	4280	4299	4339	4183
1120	4208	4062	4100	3891	3963	3865	3855	3876	3862	3917
1130	3891	3846	3916	4018	3820	3919	3917	3828	3839	3915

1140	3807	3766	3752	3799	3842	3695	3887	3959	3992	3946
1150	3791	3844	3755	3805	3718	3779	3777	3730	3719	3596
1160	3528	3676	3646	3535	3596	3525	3799	3702	3664	3680
1170	3624	3562	3597	3670	3718	3694	3796	3737	3632	3593
1180	3517	3385	3496	3588	3623	3516	3452	3485	3668	3572
1190	3602	3711	3652	3676	3933	3518	3380	3450	3392	3341
1200	3411	3290	3251	3248	3255	3330	3242	3301	3292	3495
1210	3541	3563	3523	3337	3697	3888	3741	3683	3639	3592
1220	3711	3886	3854	3947	3419	3209	3192	3204	3186	3221
1230	3392	3456	3509	3474	3250	3204	3149	3110	3194	2998
1240	3000	2966	3114	3097	3067	3130	3073	3104	3193	3097
1250	3134	3122	2847	3096	3011	2940	2925	2994	2876	2950
1260	3046	3181	3204	2991	2889	2790	2820	2919	2997	2868
1270	2865	2874	2910	2872	2856	2887	2961	2929	2634	2765
1280	2747	2674	2810	2749	2659	2642	2536	2689	2722	2765
1290	2762	2810	2968	2944	2894	2776	2687	2754	2656	2715
1300	2764	2659	2628	2560	2722	2539	2546	2623	2602	2638
1310	2681	2619	2572	2570	2491	2442	2541	2565	2635	2537
1320	2637	2650	2923	3016	2952	2628	2572	2571	2652	2659
1330	2608	2563	2579	2549	2711	3044	3058	2812	2559	2361
1340	2410	2443	2424	2427	2498	2436	2534	2557	2800	2986
1350	2952	2714	2398	2412	2376	2406	2482	2541	2512	2472
1360	2500	2495	2584	2539	2536	2472	2387	2510	2571	2516
1370	2455	2348	2438	2392	2303	2174	2411	2298	2301	2365
1380	2309	2392	2298	2240	2298	2166	2170	2237	2241	2296
1390	2195	2242	2315	2314	2486	2448	2397	2320	2460	2386
1400	2272	2218	2150	2104	2126	2050	2123	2107	2080	2118
1410	2124	2158	2079	2089	2129	2163	2195	2208	2241	2244
1420	2334	2299	2217	2179	2073	2138	2069	2117	2163	2104
1430	2008	2093	2097	2180	2023	2021	2134	2192	2042	2156
1440	2112	2070	2131	2052	1999	2024	1948	2035	2030	2111
1450	2087	2058	2067	1982	1924	1991	1903	2059	2087	2171
1460	2204	2103	2191	2075	2084	1893	1822	1840	1844	1739
1470	1810	1808	1838	2013	2087	2032	1844	1803	1760	1815
1480	1874	1811	1852	1772	1917	1774	1887	1783	1911	1907
1490	1919	1971	1886	1746	1733	1695	1694	1668	1691	1726
1500	1669	1667	1714	1665	1686	1615	1628	1739	1633	1658
1510	1661	1731	1657	1635	1591	1643	1631	1662	1710	1679
1520	1604	1569	1494	1616	1653	1630	1577	1621	1566	1590
1530	1585	1660	1640	1587	1649	1635	1655	1586	1530	1549
1540	1589	1665	1596	1537	1482	1495	1536	1595	1436	1509
1550	1460	1533	1548	1669	1758	1784	1702	1531	1637	1556
1560	1682	1725	1900	1761	1583	1637	1826	1934	2045	1928
1570	1762	1823	1734	1665	1514	1646	1777	1662	1616	1495
1580	1418	1409	1322	1428	1399	1449	1484	1372	1382	1278
1590	1367	1363	1394	1366	1347	1375	1298	1312	1381	1323
1600	1283	1372	1428	1314	1403	1391	1325	1372	1414	1488
1610	1588	1570	1559	1494	1499	1599	1591	1425	1390	1281
1620	1269	1191	1210	1210	1275	1231	1239	1388	1352	1466
1630	1320	1355	1241	1267	1173	1175	1185	1244	1176	1191
1640	1209	1191	1170	1178	1313	1253	1239	1224	1167	1139
1650	1166	1186	1168	1163	1242	1213	1174	1216	1185	1209
1660	1162	1246	1204	1335	1319	1219	1183	1165	1147	1205
1670	1150	1183	1133	1205	1181	1144	1069	1138	1094	1122
1680	1124	1127	1197	1177	1331	1378	1545	1479	1273	1152
1690	1022	1102	1049	1008	1014	1062	1028	1057	1047	1071
1700	1049	1151	1078	1112	1089	1191	1131	1008	1086	1119
1710	1048	996	1052	1112	991	996	1005	988	1017	1064
1720	1042	1010	989	992	931	1002	1024	1045	1032	1007
1730	1000	1032	1029	956	989	985	966	967	1073	998

1740	975	991	967	999	994	1071	1047	1039	1061	1044
1750	960	1006	942	985	1015	978	940	1034	1116	1111
1760	1072	1019	1068	964	959	881	926	945	927	946
1770	949	932	962	932	973	939	981	974	919	931
1780	908	922	862	947	936	913	933	879	926	868
1790	890	964	913	901	946	1019	949	920	1052	954
1800	1030	983	898	893	812	837	871	883	892	890
1810	843	852	839	800	881	935	792	806	833	842
1820	817	773	882	822	816	852	881	800	819	843
1830	861	841	858	787	866	828	847	821	821	820
1840	877	879	864	875	842	809	835	806	777	823
1850	780	838	808	809	835	807	865	864	795	824
1860	821	798	790	806	781	839	878	856	787	818
1870	780	802	791	818	881	911	948	970	1000	910
1880	837	882	816	864	823	820	811	788	792	799
1890	793	737	753	780	727	800	775	827	795	864
1900	770	708	738	796	719	796	739	723	756	806
1910	800	855	857	939	975	898	820	763	729	768
1920	703	784	736	793	782	778	729	777	735	755
1930	755	753	763	740	739	727	781	724	710	733
1940	730	652	723	698	685	753	661	711	728	681
1950	639	718	652	651	665	677	672	630	643	637
1960	681	693	681	657	645	629	652	683	647	623
1970	647	617	692	656	643	622	612	647	603	623
1980	646	637	619	665	674	657	698	645	636	672
1990	682	671	655	667	675	671	679	707	654	674
2000	677	659	587	636	628	606	593	588	569	657
2010	574	591	647	609	595	620	637	597	555	592
2020	710	649	672	693	751	643	660	656	636	630
2030	627	611	576	592	633	574	598	602	665	633
2040	678	655	663	653	582	555	553	553	552	589
2050	563	604	604	600	610	579	629	695	731	780
2060	696	669	615	567	575	599	589	552	562	587
2070	543	621	575	509	536	599	560	570	543	514
2080	568	570	584	654	644	595	558	521	519	497
2090	521	592	562	638	626	581	511	516	473	516
2100	521	469	445	461	496	448	479	483	490	491
2110	481	441	457	503	437	482	488	465	489	479
2120	457	490	518	478	489	466	493	465	454	494
2130	500	472	435	439	437	446	418	450	416	437
2140	437	432	402	437	405	397	428	409	426	401
2150	420	440	373	405	436	448	426	419	441	414
2160	449	434	406	411	403	397	404	382	379	364
2170	381	370	439	411	430	400	400	473	484	492
2180	451	395	422	456	436	411	379	383	403	393
2190	440	402	397	394	403	411	434	388	447	505
2200	435	453	453	511	542	558	496	425	429	388
2210	374	358	359	375	369	417	438	460	551	617
2220	557	508	453	424	341	395	401	387	412	430
2230	406	373	400	426	460	499	456	435	376	350
2240	374	351	355	346	406	381	378	370	348	331
2250	393	368	347	364	349	390	342	382	327	320
2260	331	325	346	353	392	335	298	310	356	311
2270	296	333	334	315	309	323	295	336	306	302
2280	273	352	319	337	324	299	322	320	306	289
2290	296	285	304	324	326	314	297	331	296	324
2300	263	280	352	287	327	324	311	298	342	353
2310	309	312	293	281	316	301	302	280	316	290
2320	256	301	328	309	299	266	267	274	276	279
2330	273	273	267	254	297	309	278	265	317	263

2340	278	271	293	311	257	264	286	284	291	267
2350	279	256	293	285	289	282	272	267	248	236
2360	288	274	293	272	266	266	280	257	240	250
2370	262	295	298	286	299	293	263	281	303	351
2380	315	290	256	265	270	265	276	289	257	256
2390	268	254	222	265	245	265	284	210	249	272
2400	230	239	249	245	242	217	212	252	237	243
2410	228	233	251	252	252	233	227	244	214	254
2420	207	239	250	226	248	248	255	234	257	243
2430	290	233	231	251	236	241	250	250	212	229
2440	229	234	209	213	198	226	208	218	212	228
2450	208	254	230	215	178	207	186	204	206	228
2460	188	199	199	216	219	201	189	187	205	193
2470	226	217	210	208	190	211	206	169	186	213
2480	205	174	196	203	199	192	206	211	187	177
2490	190	197	187	166	196	177	188	202	214	215
2500	193	185	179	177	208	181	162	175	204	194
2510	172	182	175	191	200	210	205	207	221	244
2520	238	268	208	218	176	207	187	188	169	194
2530	168	164	171	189	194	169	148	155	157	173
2540	147	158	174	172	168	169	182	162	174	166
2550	181	155	180	190	190	236	246	254	195	191
2560	147	160	158	177	163	175	163	165	141	160
2570	161	166	178	152	161	178	191	174	175	171
2580	180	151	147	169	150	174	178	150	162	139
2590	138	120	162	160	149	151	135	120	155	117
2600	124	135	138	140	143	148	120	123	121	138
2610	148	127	154	143	121	121	122	158	155	143
2620	128	134	127	120	121	139	140	141	126	135
2630	119	131	142	124	131	130	143	151	146	146
2640	128	126	135	121	100	138	153	128	121	126
2650	126	104	123	123	123	107	109	123	120	97
2660	124	114	121	112	110	115	126	123	136	134
2670	110	108	114	101	103	121	108	113	116	122
2680	123	111	88	100	115	106	130	93	101	94
2690	99	102	127	111	116	124	126	116	134	151
2700	192	203	191	177	145	105	112	116	110	125
2710	116	122	147	170	171	149	118	104	112	103
2720	108	111	108	108	100	94	120	101	130	106
2730	104	91	81	99	98	88	93	97	111	83
2740	95	91	86	85	91	106	110	94	80	98
2750	104	93	110	91	105	108	110	116	103	112
2760	96	98	96	87	99	88	97	90	90	96
2770	95	88	107	101	101	108	97	90	94	67
2780	109	114	104	114	131	103	124	120	105	81
2790	106	89	87	85	99	79	77	66	67	73
2800	68	65	80	70	72	75	91	80	67	83
2810	81	76	71	75	70	81	77	64	89	69
2820	93	89	77	86	84	87	94	101	78	63
2830	64	82	77	74	90	71	66	65	95	65
2840	82	79	80	81	89	79	67	85	70	68
2850	65	91	89	78	69	80	69	67	81	64
2860	73	61	61	67	82	82	78	70	79	55
2870	74	59	79	60	76	53	72	80	87	82
2880	100	79	50	56	64	64	69	69	83	68
2890	77	68	70	75	76	80	67	74	84	60
2900	56	77	64	61	66	61	62	67	69	57
2910	54	48	64	71	81	79	60	56	72	71
2920	57	68	70	76	65	63	75	64	73	57
2930	49	46	53	66	65	64	71	63	60	58

2940	65	58	75	58	66	53	61	57	43	52
2950	63	42	46	51	60	55	55	45	57	56
2960	48	54	64	60	70	60	65	72	53	47
2970	42	64	41	37	41	57	51	48	51	45
2980	48	52	63	50	46	44	55	46	44	39
2990	44	72	48	46	41	53	46	44	50	50
3000	39	39	47	46	62	46	54	51	49	49
3010	52	41	51	47	47	39	43	43	41	42
3020	51	46	47	53	37	48	41	47	54	41
3030	62	48	51	58	75	48	53	53	53	50
3040	38	41	39	38	50	42	44	42	37	47
3050	48	46	40	46	48	41	35	39	43	41
3060	28	32	37	41	41	43	38	37	36	57
3070	41	38	43	40	42	56	43	52	48	47
3080	46	36	42	38	44	45	41	19	43	43
3090	32	37	44	47	44	33	38	38	37	34
3100	36	41	42	41	45	51	51	42	28	34
3110	40	30	41	43	38	35	37	45	41	29
3120	34	35	37	24	32	29	28	22	39	25
3130	30	28	37	36	29	37	25	39	43	38
3140	29	36	37	47	25	36	36	27	32	29
3150	43	23	35	36	42	29	35	36	20	34
3160	36	22	36	30	33	31	27	33	29	18
3170	27	25	30	33	36	28	28	27	32	40
3180	25	21	34	35	30	24	28	30	29	23
3190	23	26	33	27	25	37	22	28	26	21
3200	41	22	37	32	33	27	23	22	28	29
3210	33	26	22	24	29	29	24	32	43	38
3220	43	37	23	28	33	26	25	26	33	20
3230	25	19	27	39	26	23	28	16	16	32
3240	28	22	23	30	23	28	21	29	23	18
3250	19	20	21	15	12	24	24	30	13	25
3260	26	26	15	19	24	17	17	23	21	32
3270	20	29	9	15	19	16	21	16	23	20
3280	17	16	22	16	22	15	21	12	16	13
3290	25	18	18	12	16	17	14	16	14	15
3300	14	17	14	16	15	20	19	19	22	9
3310	20	21	18	17	16	11	10	17	19	16
3320	18	10	12	18	11	11	13	15	13	16
3330	15	19	18	16	11	14	11	18	13	8
3340	16	13	16	12	11	15	15	14	21	24
3350	18	22	30	31	29	38	31	21	27	20
3360	17	14	12	10	9	17	8	15	13	12
3370	16	16	9	14	14	12	13	7	8	8
3380	18	20	17	15	11	11	14	11	19	15
3390	16	9	12	12	7	12	11	20	7	12
3400	8	20	9	10	15	13	16	11	7	5
3410	7	12	11	12	15	12	9	18	11	12
3420	13	13	10	16	20	16	22	16	15	18
3430	17	11	8	5	9	12	11	7	14	13
3440	8	8	8	11	10	15	13	7	12	6
3450	15	10	10	9	10	10	8	9	9	14
3460	11	8	9	11	9	13	14	12	10	7
3470	11	8	7	8	10	14	8	6	12	8
3480	9	12	13	11	13	13	11	13	8	8
3490	7	7	7	6	6	9	12	7	9	4
3500	12	11	11	4	11	7	8	4	8	2
3510	5	12	7	11	4	9	5	6	7	9
3520	10	13	5	10	7	9	3	9	6	5
3530	6	6	12	10	7	5	5	8	11	11

540	12	12	8	9	8	9	7	4	12	8
550	10	5	5	10	5	10	7	5	3	3
560	6	4	6	5	4	8	8	7	3	5
570	6	11	11	11	7	5	6	5	5	6
580	1	6	5	13	3	5	8	1	10	5
590	8	5	3	6	7	7	6	7	8	6
600	9	4	4	6	8	9	7	9	6	2
610	6	7	5	3	6	8	5	8	6	5
620	8	8	4	7	3	3	7	8	3	8
630	5	7	6	7	5	6	3	4	5	9
640	3	6	10	6	4	6	6	6	6	4
650	4	3	5	5	2	6	8	2	7	8
660	5	6	8	4	3	3	7	7	3	4
670	6	4	4	4	6	5	6	8	6	8
680	5	2	3	2	3	4	2	2	5	2
690	4	5	4	2	4	1	4	4	1	3
700	4	4	1	5	4	5	6	5	6	6
710	2	3	3	5	5	5	7	6	5	5
720	8	6	3	4	6	2	2	7	5	4
730	1	5	3	5	3	4	2	1	2	5
740	5	6	3	5	2	4	3	3	3	3
750	2	4	1	3	2	2	1	2	3	3
760	3	1	5	3	2	4	2	5	1	2
770	7	2	0	5	2	2	5	4	3	3
780	1	4	1	3	2	3	3	0	6	0
790	4	3	1	3	2	0	5	5	0	2
800	3	3	2	2	2	3	3	7	7	3
810	2	1	3	4	3	4	3	6	2	0
820	3	1	1	5	2	1	6	3	1	6
830	5	2	3	1	4	2	3	3	8	1
840	0	7	4	1	2	0	1	1	1	1
850	3	2	0	1	1	4	2	3	3	1
860	2	0	3	1	4	2	3	2	2	3
870	1	3	1	1	2	6	4	2	4	1
880	4	2	1	3	0	4	2	2	3	4
890	1	2	3	2	3	2	3	2	2	1
900	3	1	2	2	0	4	4	2	2	1
910	7	2	3	3	2	3	2	0	1	3
920	0	1	4	2	1	1	1	3	2	3
930	2	0	1	2	0	0	0	0	0	1
940	1	0	1	0	1	1	2	3	1	0
950	1	3	0	1	3	4	3	0	0	4
960	1	1	1	2	1	1	0	0	1	1
970	2	0	2	2	2	3	0	1	3	1
980	1	3	4	2	0	1	1	2	0	2
990	2	1	1	0	0	1	4	1	2	2
000	1	1	0	1	2	0	0	1	0	0
010	1	2	3	0	3	1	0	2	1	0
020	0	1	2	0	1	4	0	2	1	0
030	1	2	2	1	3	0	0	2	1	1
040	2	1	0	0	0	2	0	1	3	2
050	0	0	2	1	3	1	0	3	1	1
060	2	0	0	1	3	2	1	0	2	0
070	1	0	2	1	1	0	0	0	3	2
080	1	0	1	0	0	1	0	0	0	0
090	0	0	1	0	0	57345				

LENGTH OF SPECTRUM = 4096 CHANNELS SPECTRUM ID (OCTAL NOTATION)=0004
COUNT TIME (SEC) IS FOUND IN CHANNEL 1

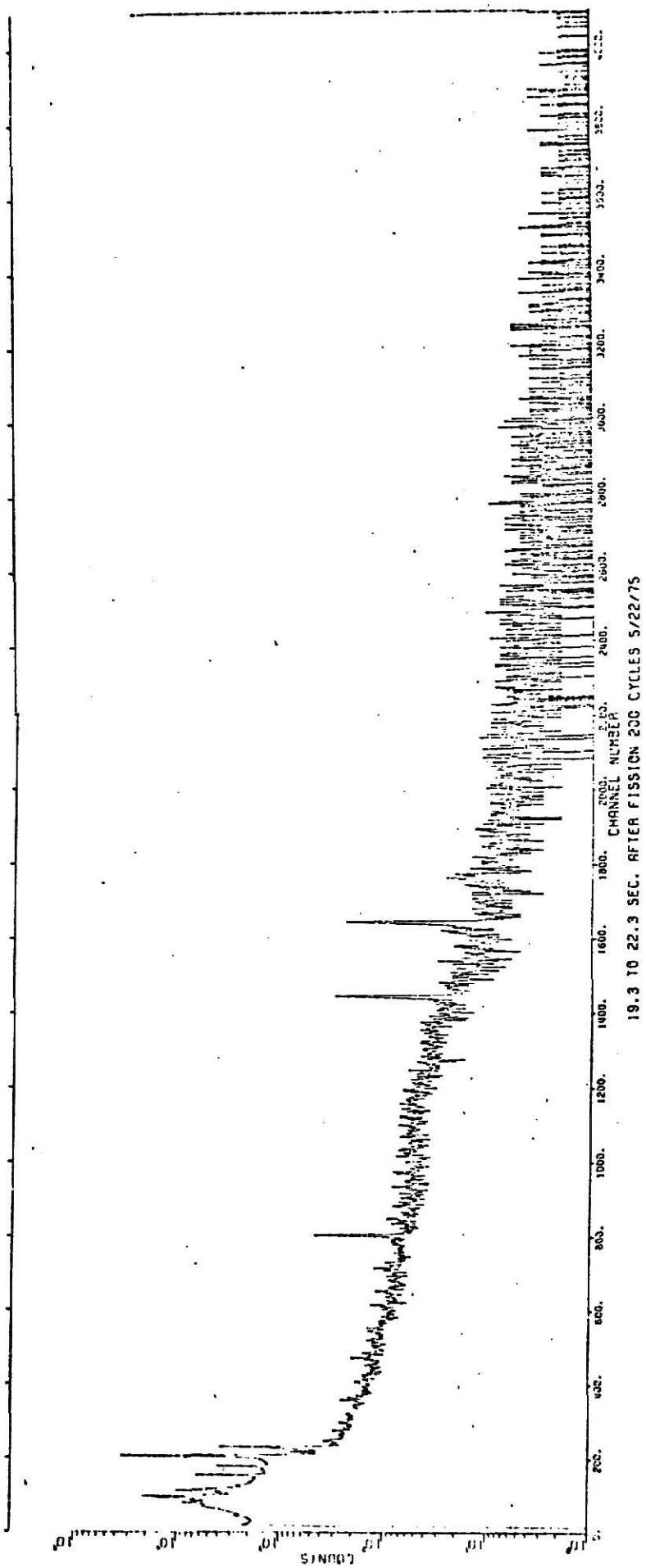
```
G LEVEL  21                    MAIN                    DATE = 75160          12/34/40

C   ************** GAMMA SPECTRUM PLOTTING PROGRAM - GASP-4 ********************
C   *    J. KENNETH SHULTIS, NUCLEAR ENGINEERING,  5/8/73                       *
C   *    THIS PROGRAM PLOTS THE GAMMA SPECTRUM AS A CARTESIAN OR SEMILOG PLOT   *
C   *    PROGRAM IS DIMENSIONED FOR A MAXIMUM OF 4096 DATA POINTS               *
C   *    VARIABLE SIZED PLOTS ARE PERMITTED                                     *
C   ***************************************************************************
      DIMENSION BUF(1000),TITLE(19),X(4098),Y(4098),DX(4),IHELP(11),IARR
     1(4096)
      COMMON/AXICOM/MDEL
      CALL PLOTS(BUF,1000)
      MDEL=4
C*** ORIGIN PEN AT +1.0 INCHES
      CALL PLOT(0.0,-14.0,3)
      CALL PLOT(0.0,-13.0,-3)
C   ************************** INPUT DATA ***********************************
C   *  CARD 1:  (2I,2X,19A4)                                                  *
C   *           LOGLIN = TYPE OF PLOT DESIRED, =1 FOR LINEAR PLOT, =2 FOR SEMI-*
C   *                    LOG PLOT, =3 FOR BOTH LINEAR AND SEMILOG PLOTS       *
C   *           MULT   = NUMBER OF MEMORY SLICES TO BE PLOTTED                 *
C   *                    I.E. NUMBER OF CHANNELS = MULT*1024                   *
C   *           ISIZE  = PARAMETER INDICATING SIZE OF PLOT, =0 FOR STANDARD    *
C   *                    SIZED PLOT (MULT*6.0 X 8.0), =1 IF SIZE IS READ IN    *
C   *           TITLE  = TITLE OF PLOT (76 CHARACTERS)                         *
C   *                                                                          *
C   *  CARD 2:  (2F10.2)    -    NOT INCLUDED IF ISIZE = 0                     *
C   *           XLENTH = LENGTH OF X AXIS IN INCHES                            *
C   *           YLENTH = LENGTH OF Y AXIS IN INCHES                            *
C   *                                                                          *
C   *  SPECTRUM DATA: (6X,12F6.0,2X)                                           *
C   *           Y(I)   = GAMMA SPECTRUM                                        *
C   *                                                                          *
C   *  EACH DATA SET CONSISTS OF (MULT*86+1) CARDS IF THE STANDARD SIZE PLOT   *
C   *  IS DESIRED.  IF THE PLOT SIZE IS SPECIFIED THEN EACH DATA SET CONSISTS  *
C   *  OF (MULT*86+2) CARDS.  PROGRAM IS TERMINATED BY ADDING A BLANK CARD     *
C   *  AFTER ALL THE DATA SETS.                                                *
C   ***************************************************************************
   99 READ 1, LOGLIN,MULT,ISIZE,TITLE
    1 FORMAT(3I1,1X,19A4)
      IF (LOGLIN.NE.0) GO TO 2
      CALL PLOT(2.*XLENTH,0.0,999)
      STOP
    2 NPTS=MULT*1024
      DO 4 I=1,NPTS
    4 X(I)=I
      IF (ISIZE) 5,5,6
    5 XLENTH=MULT*6.00
      YLENTH=8.0
      GO TO 8
    6 READ 9, XLENTH,YLENTH
    9 FORMAT(2F10.2)
      READ(5,97) NCH
   97 FORMAT(I4)
    8 READ 3, (Y(I),I=1,NPTS)
    3 FORMAT(6X,12F6.0,2X)
      Y(1)=1.
C ***SUMMED OVER NCH CHANNELS***
      WRITE(6,98) NCH
   98 FORMAT(1H0,'RESULT IS SUMMED OVER ',I3,' CHANNELS')
```

```
      IFIR=(NCH+1)/2
      ILAS=1024-NCH/2
      DO 232 I=IFIR,ILAS
      ISUM=0
      JFIR=I-NCH/I
      JLAS=I+NCH/2
      DO 230 J=JFIR,JLAS
  230 ISUM=ISUM+Y(J)
  232 IARR(I)=ISUM/1
      DO 234 I=1,1024
  234 Y(I)=IARR(I)
      ADV=XLENTH + 6.0
      X(NPTS+1)=0.0
      X(NPTS+2)=NPTS/XLENTH
      DX(1)=0.0
      DX(2)=NPTS
      CALL SCALE(DX,XLENTH,2,1)
      TINC=DX(4)/X(NPTS+2)
C*** LINEAR PLOT SECTION
      IF (LOGLIN.EQ.2) GO TO 50
      CALL SCALE(Y,YLENTH,NPTS,1)
      CALL DAXIS(0.0,0.0,XLENTH,0.0,TINC,-1)
      CALL SAXIS(X(NPTS+1),DX(4),0,1,-1,14HCHANNEL NUMBER,-14)
      CALL AXIS(0.0,0.0,6HCOUNTS,6,YLENTH,90.0,Y(NPTS+1),Y(NPTS+2))
      CALL LINE(X,Y,NPTS,+1,0,0)
      CALL SYMBOL(0.0,-0.7,0.14,TITLE,0.0,76)
      CALL PLOT(ADV,0.0,-3)
      PRINT 7, (TITLE(I),I=1,19)
    7 FORMAT('0A LINEAR PLOT WAS COMPLETED - - ',19A4)
C*** LOG PLOT SECTION
   50 IF (LOGLIN.EQ.1) GO TO 99
      DO 51 I=1,NPTS
      IF (Y(I).LE.0.0) Y(I)=1.0
   51 CONTINUE
      CALL DAXIS(0.0,0.0,XLENTH,0.0,TINC,-1)
      CALL SAXIS(X(NPTS+1),DX(4),0,1,-1,14HCHANNEL NUMBER,-14)
      CALL SCALOG(Y,YLENTH,NPTS,1)
      CALL LGAXIS(0.0,0.0,6HCOUNTS,6,YLENTH,90.0,Y(NPTS+1),Y(NPTS+2))
      CALL LGLINE(X,Y,NPTS,1,0,0,+1)
      CALL SYMBOL(0.0,-0.7,0.14,TITLE,0.0,76)
      CALL PLOT(ADV,0.0,-3)
      PRINT 52, (TITLE(I),I=1,19)
   52 FORMAT('0A LOG PLOT WAS COMPLETED - - ',19A4)
      GO TO 99
      END
```

Appendix E

Typical spectrum collected from the ^{252}Cf trial.

NOTE: The pronounced peaks on the spectrum are due to National Bureau of Standards calibration sources which were used to identify the energy calibration and monitor for energy spectral shifts. The low energy gamma-ray peaks are due to the ^{235}U.


COUNTS
CHANNEL NUMBER
19.3 TO 22.3 SEC. AFTER FISSION 200 CYCLES 5/22/75

Appendix F

Raw data for the calculations of reactor trial 3.

Peak Area Determination

Trial 3

Peak (Pr)	Energy (MeV)	N(600 cycles)	N(ε corrected)/sec-cycle
2183.13 (1)	2.90699	369.60±50.39	843.84±115.05
(2)		218.41±35.50	498.65±81.05
2204.70 (1)	2.93233	678.13±62.44	1569.75±144.54
(2)		279.32±38.43	646.57±88.96
2219.02 (1)	2.94914	609.73±93.10	1411.41±215.51
(2)		440.91±66.59	1020.62±154.14
2234.9756 (1)	2.96788	436.27±65.93	1016.95±153.68
(2)		282.16±62.11	657.72±144.78
2379.07 (1)	3.13710	105.40±53.02	262.19±131.89
(2)		177.27±40.21	440.97±100.02
2520.83 (1)	3.30358	259.73±58.88	687.12±155.55
(2)		148.07±38.88	391.72±102.86
2555.62 (1)	3.344	206.53±62.42	555.19±167.80
(2)		177.05±39.37	475.94±451.07
2700.98 (1)	3.51514	187.73±43.02	515.46±118.12
(2)		145.57±30.60	399.70±84.02
2713.54 (1)	3.52990	142.13±43.02	390.25±118.12
(2)		99.32±26.44	272.71±72.60
3555.06 (1)	4.28328	53.07±21.38	215.73±86.91
(2)		32.95±8.39	133.94±34.11

Standard Deviation Analysis

Trial 3

Peak Energy (MeV)	$\sigma^2(N)$	$\sigma(N)$	196 $\sigma(N)$
2.907	2534.40	50.39	98.67
	1260.23	35.50	69.58
2.932	3898.47	62.44	122.38
	1476.59	38.43	75.32
2.949	8668.33	93.10	182.48
	4434.66	66.59	130.52
2.968	4346.67	65.93	129.22
	3857.73	62.11	121.74
3.137	2810.90	53.12	103.91
	1617.27	40.21	78.82
3.304	3466.4	58.88	115.39
	1511.87	38.88	76.21
3.515	1850.93	43.02	84.32
	936.65	30.60	59.98
3.530	1319.73	36.33	71.20
	699.32	26.44	51.83
4.28	456.93	21.38	41.90
	70.45	8.39	16.45

Appendix G

Selected fission-product spectra. The spectra included are:

1) 0.5 to 2.5 sec Trial 1.

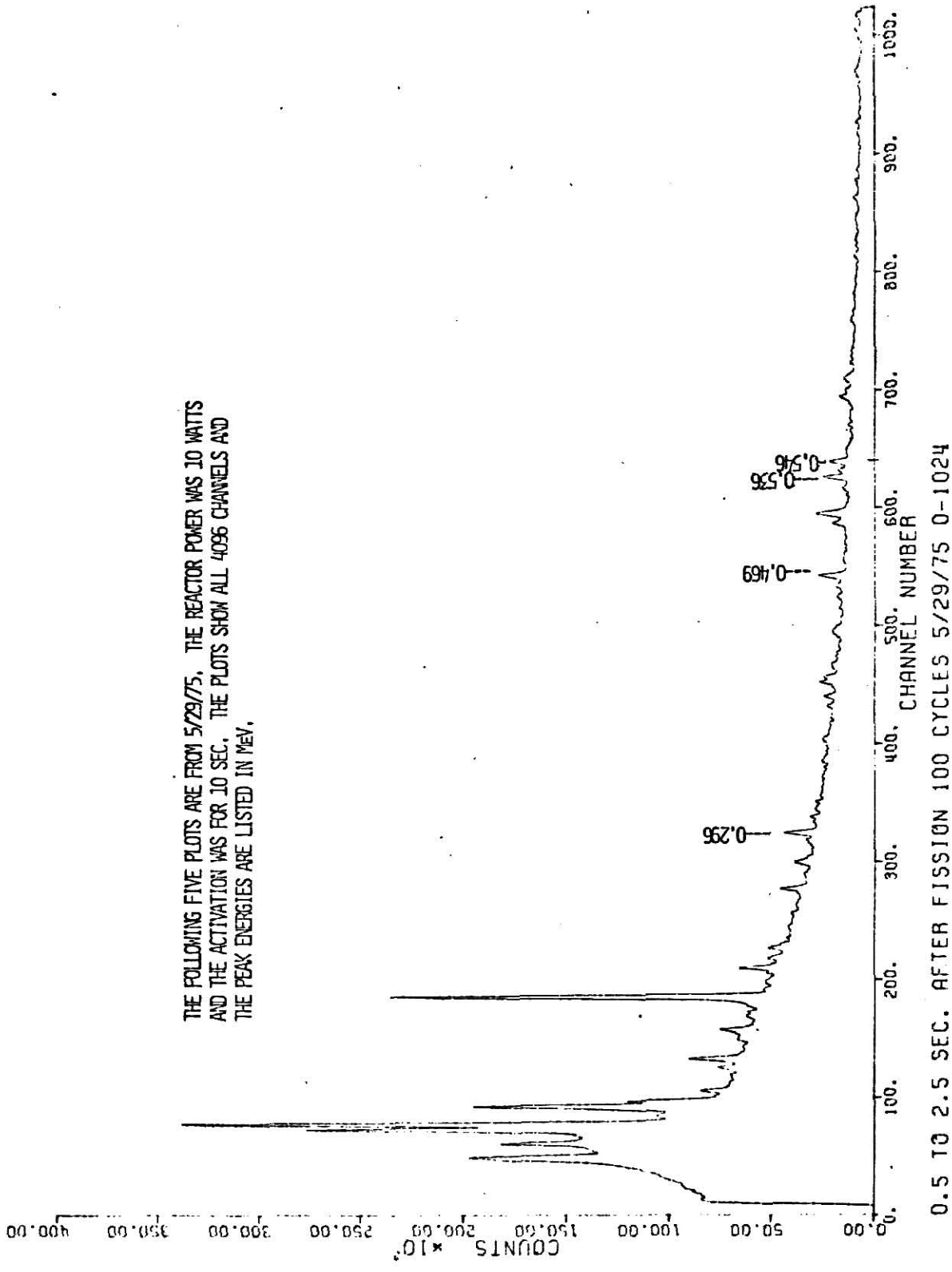
2) 50 to 52 sec Trial 1.

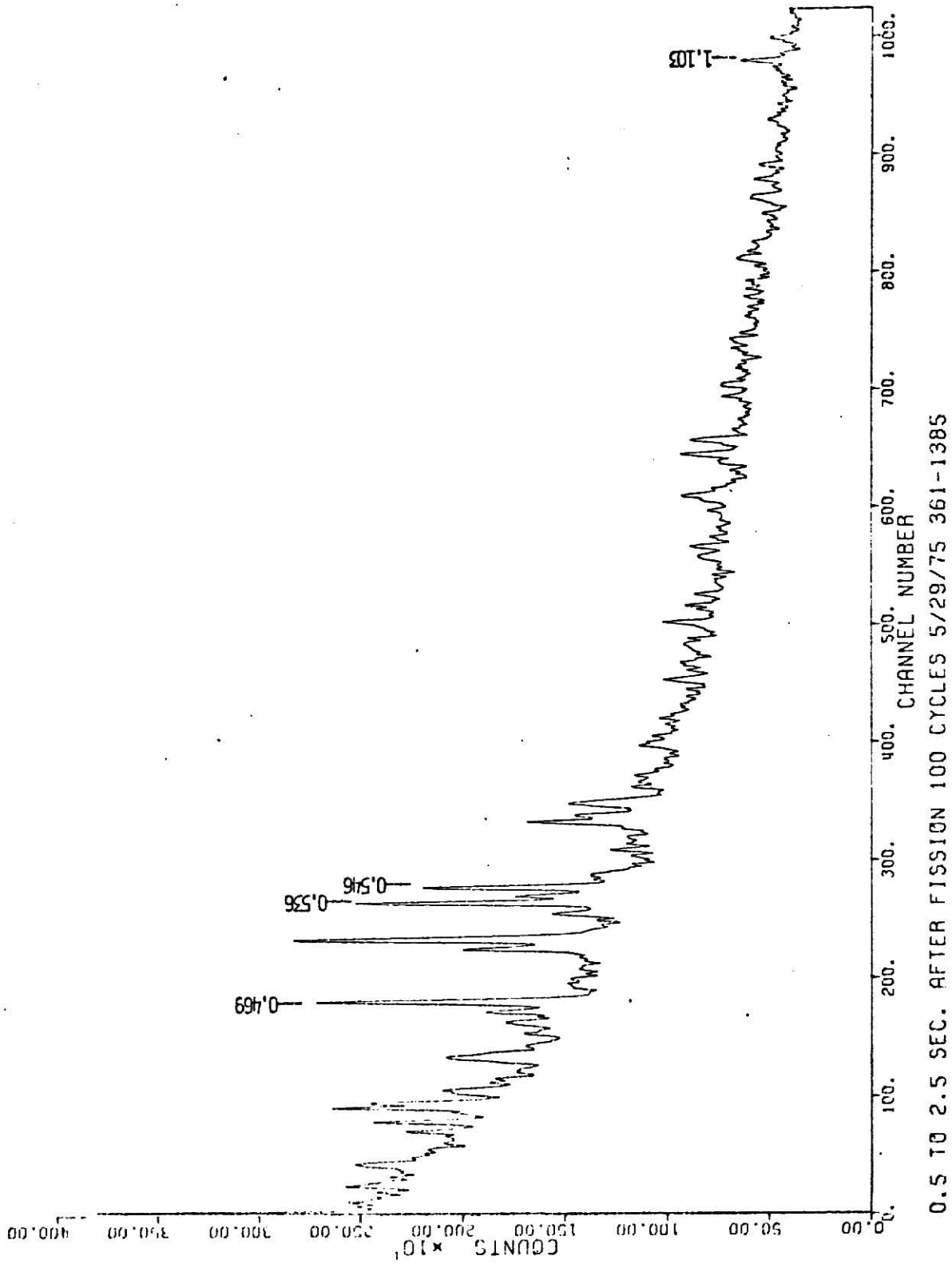
3) 12 to 14.5 sec Trial 2.

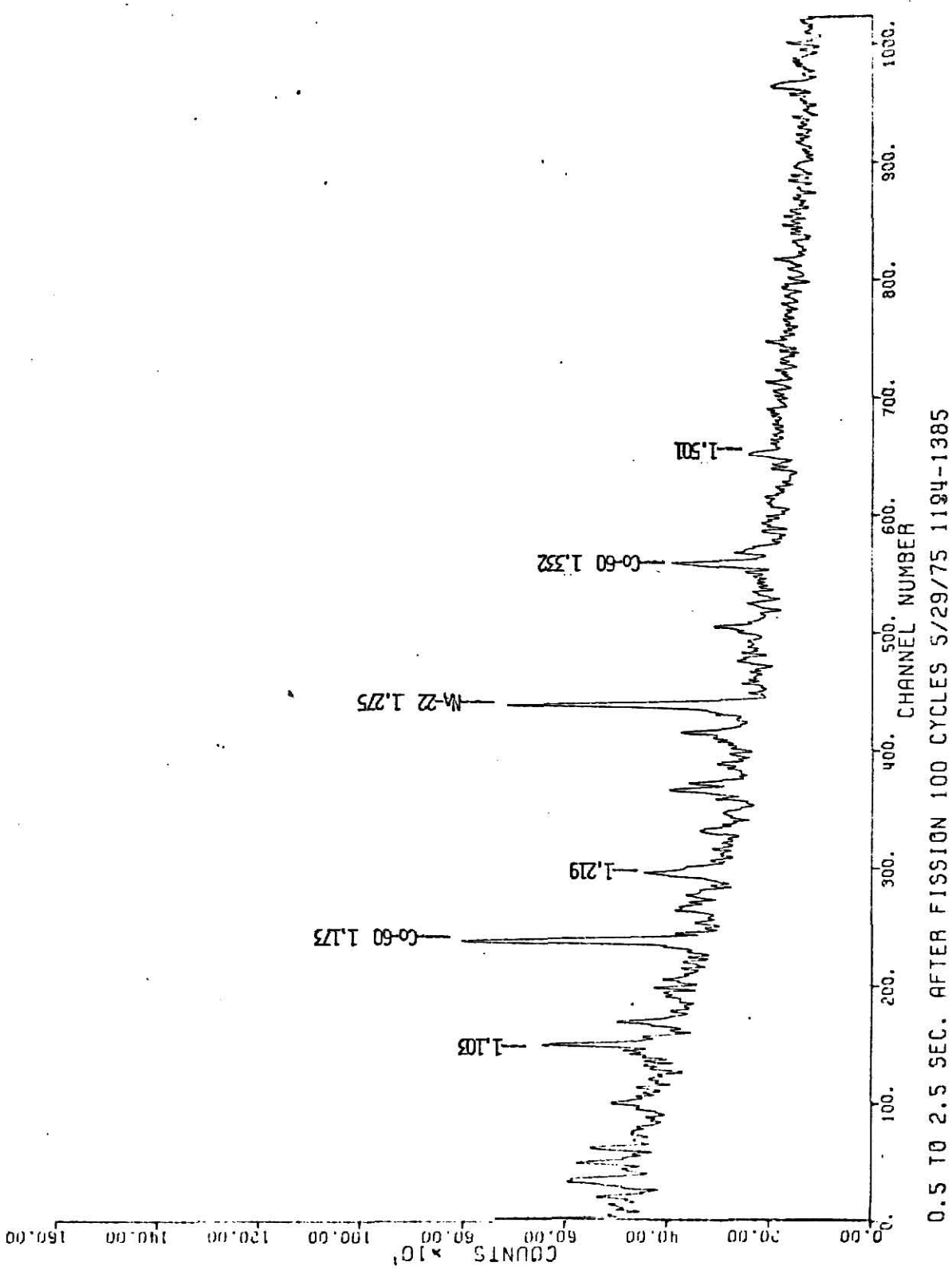
4) 230 to 255 sec Trial 2.

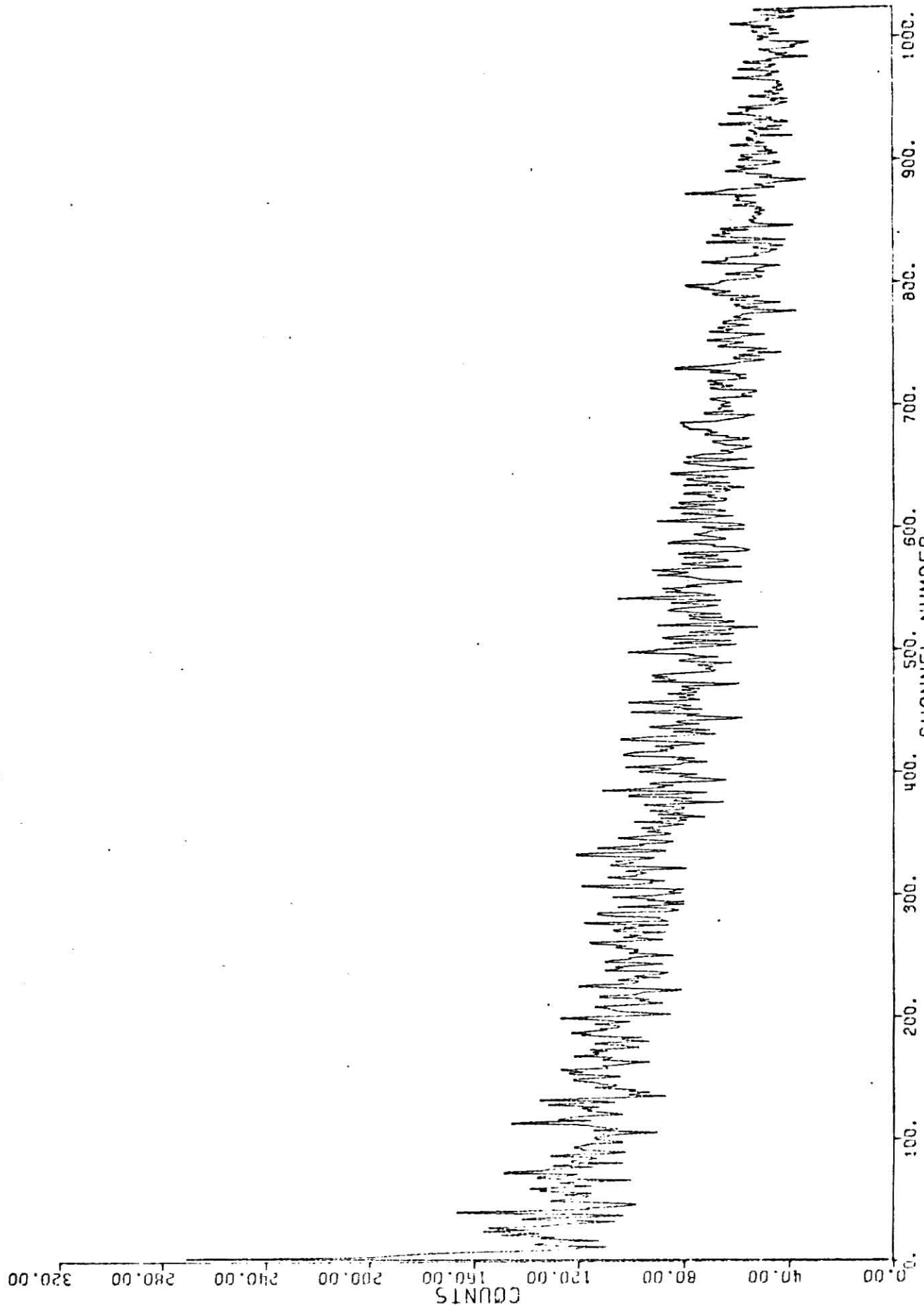
5) 0.5 to 5.5 sec Trial 3.

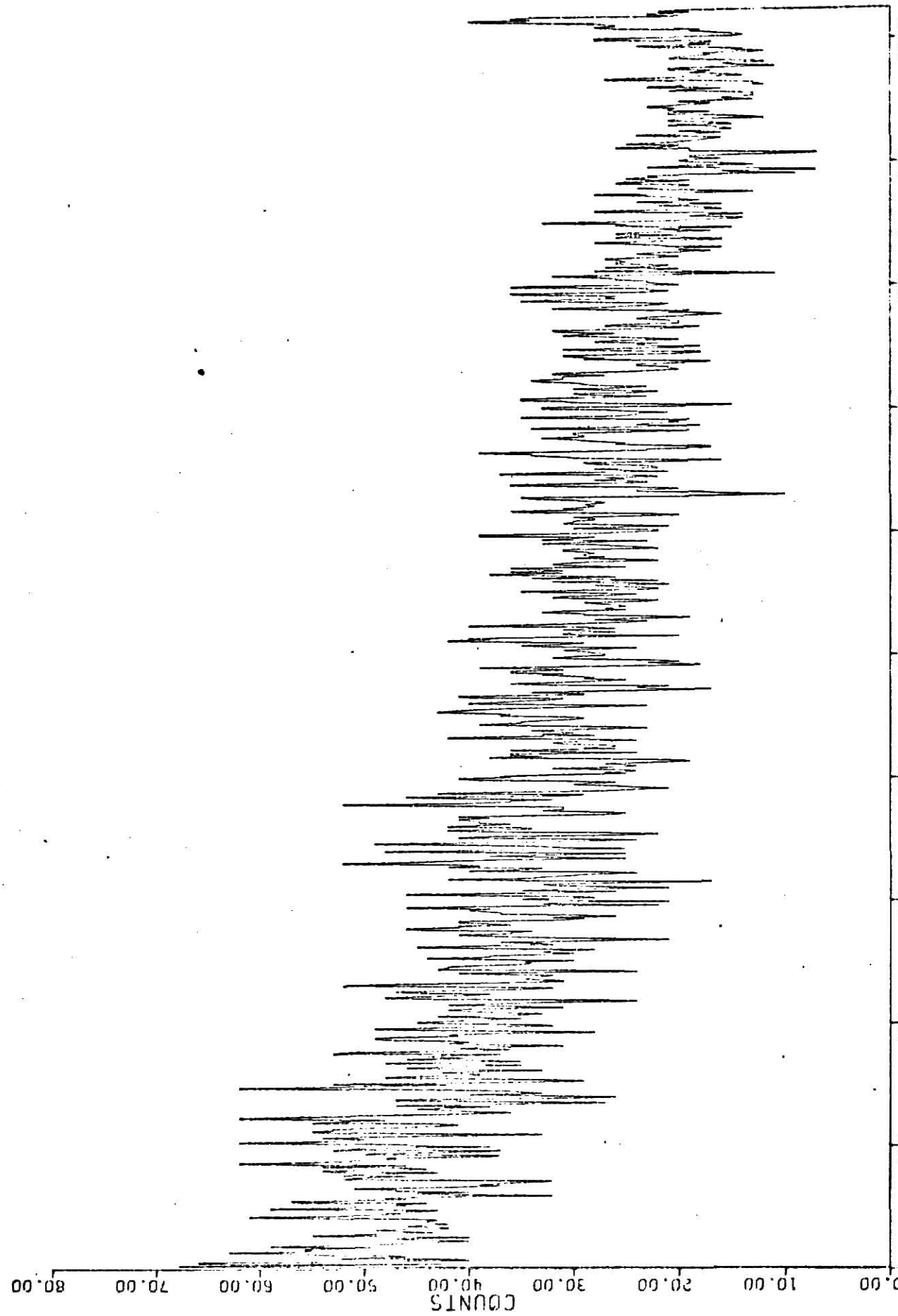
The channels included in each plot are listed in the title of each (below the scale).

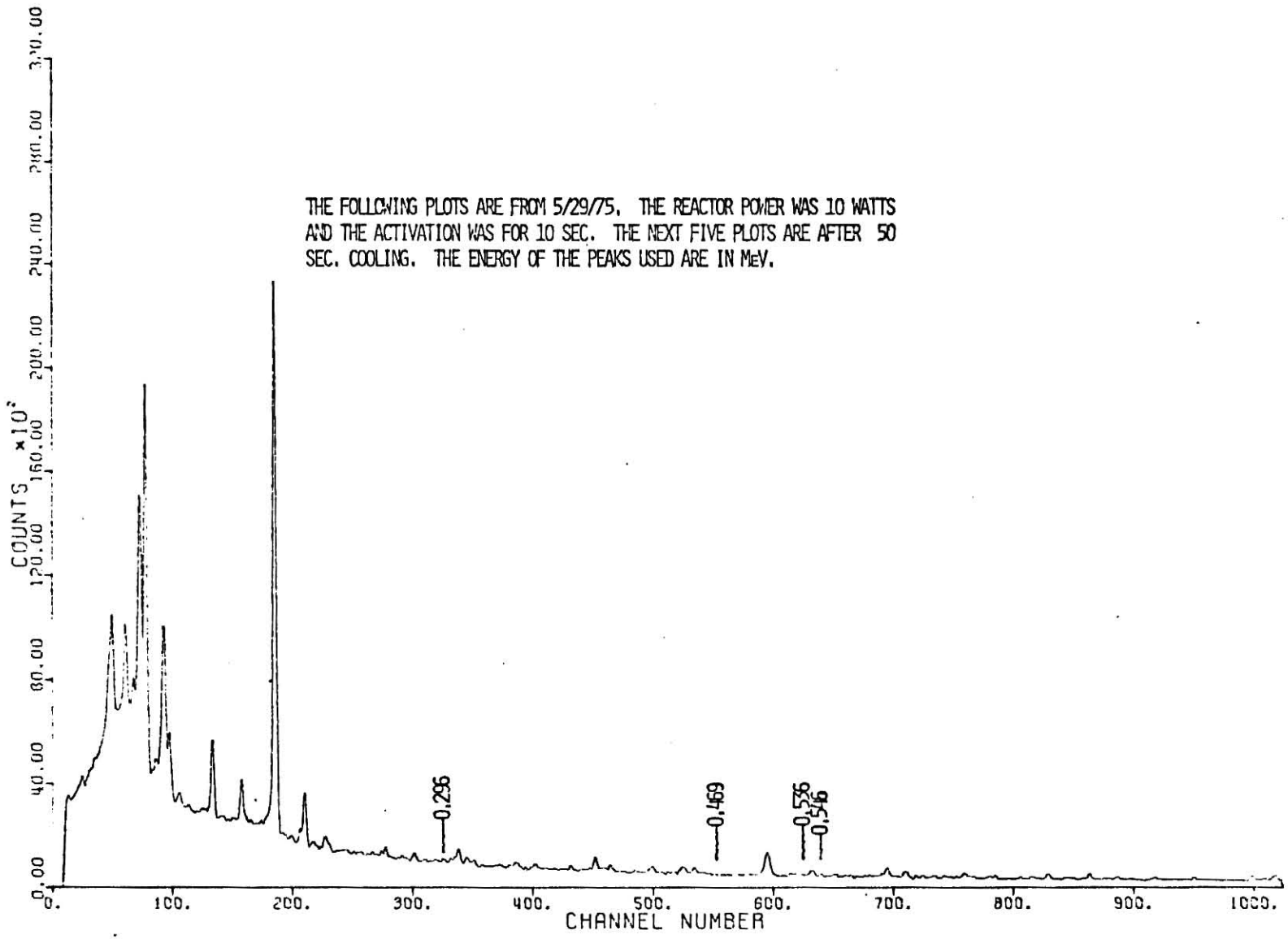

THE FOLLOWING FIVE PLOTS ARE FROM 5/29/75, THE REACTOR POWER WAS 10 WATTS
AND THE ACTIVATION WAS FOR 10 SEC. THE PLOTS SHOW ALL 4096 CHANNELS AND
THE PEAK ENERGIES ARE LISTED IN MeV,
0.256
0.469
0.536
0.546
0.5 TO 2.5 SEC. AFTER FISSION 100 CYCLES 5/29/75 0-1024
CHANNEL NUMBER
COUNTS x10

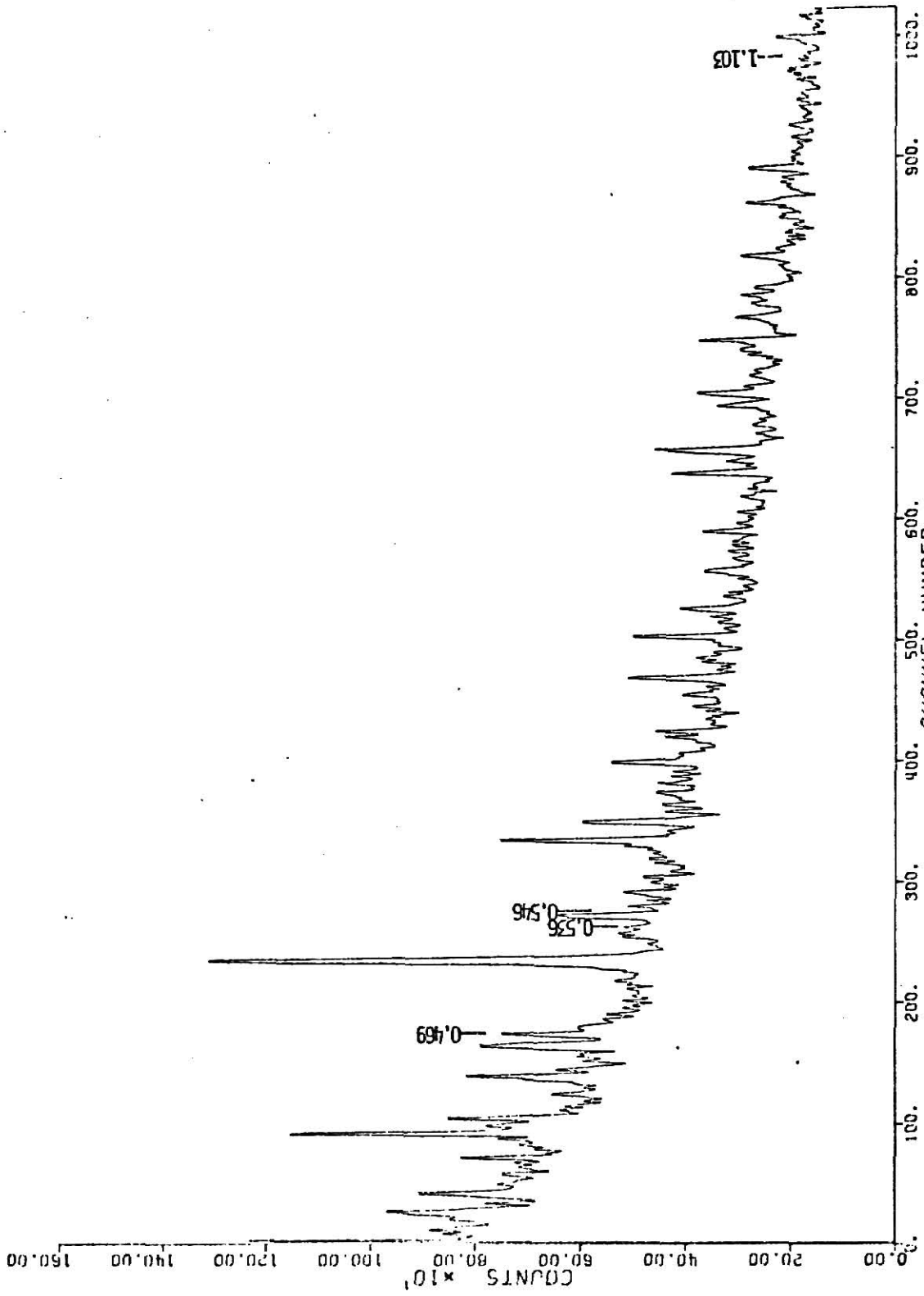

0.469
0.536
0.546
1.108
COUNTS x 10'
CHANNEL NUMBER
0.5 TO 2.5 SEC. AFTER FISSION 100 CYCLES 5/29/75 361-1385

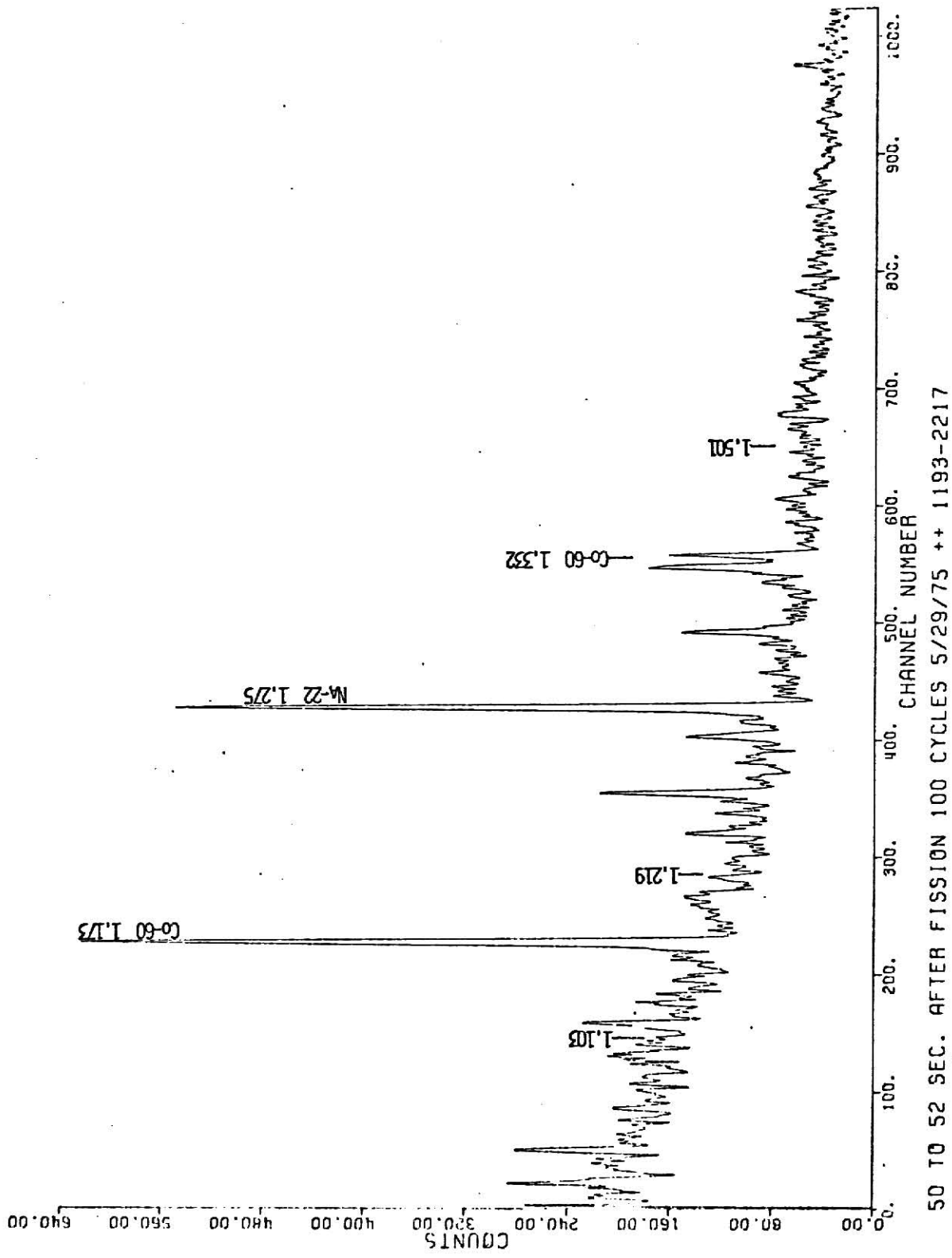

0.5 TO 2.5 SEC. AFTER FISSION 100 CYCLES 5/29/75 1194-1385
CHANNEL NUMBER
COUNTS x 10¹
1,501
Co-60 1,332
Na-22 1,275
1,219
Co-60 1,173
1,103

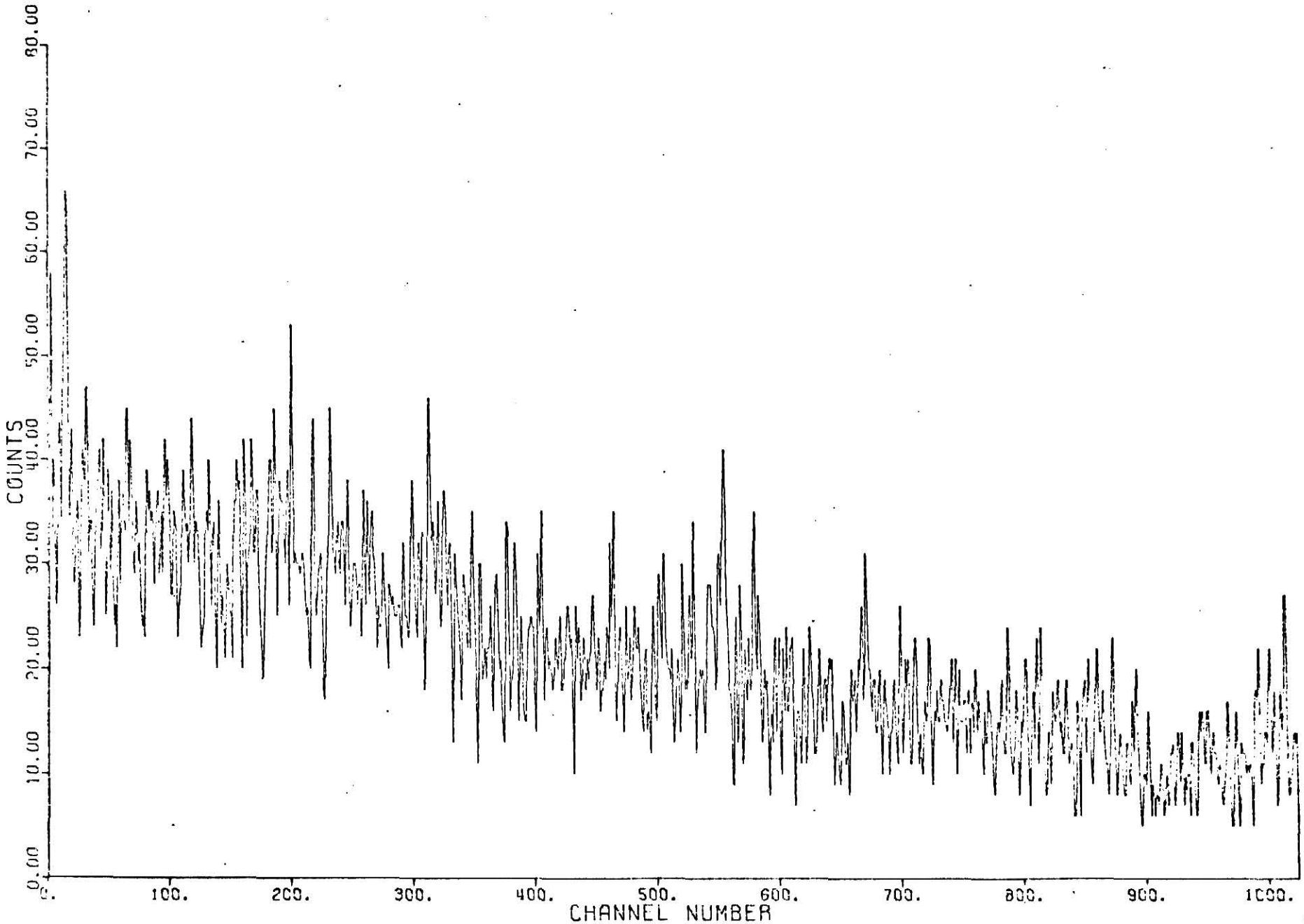

0.5 TO 2.5 SEC. AFTER FISSION 100 CYCLES 5/29/75 2145-3169
CHANNEL NUMBER
0.
100.
200.
300.
400.
500.
600.
700.
800.
900.
1000.
COUNTS
0.00
40.00
80.00
120.00
160.00
200.00
240.00
280.00
320.00


0.5 TO 2.5 SEC. AFTER FISSION 100 CYCLES 5/29/75 3072-4096
CHANNEL NUMBER
0. 100. 200. 300. 400. 500. 600. 700. 800. 900. 1000.
COUNTS
0.00 10.00 20.00 30.00 40.00 50.00 60.00 70.00 80.00

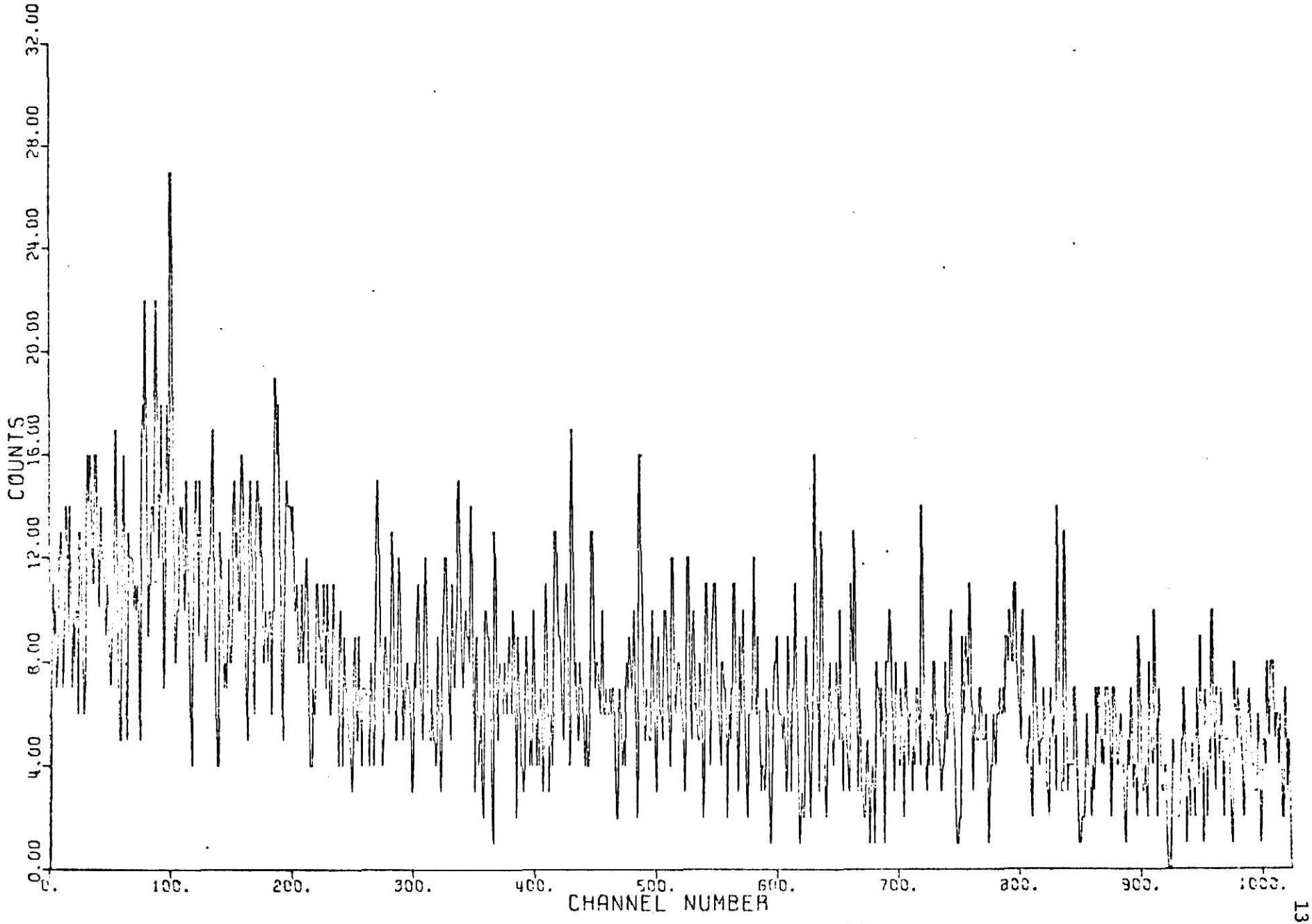

THE FOLLOWING PLOTS ARE FROM 5/29/75. THE REACTOR POWER WAS 10 WATTS AND THE ACTIVATION WAS FOR 10 SEC. THE NEXT FIVE PLOTS ARE AFTER 50 SEC. COOLING. THE ENERGY OF THE PEAKS USED ARE IN MeV.
COUNTS ×10²
0.00
40.00
80.00
120.00
160.00
200.00
240.00
0.
100.
200.
300.
400.
500.
600.
700.
800.
900.
1000.
CHANNEL NUMBER
0.296
0.469
0.575
0.595
50 TO 52 SEC. AFTER FISSION 100 CYCLES 5/29/75 ++ 0-1024


50 TO 52 SEC. AFTER FISSION 100 CYCLES 5/29/75 ++ 361-1385
CHANNEL NUMBER
COUNTS ×10¹
0.469
0.536
0.596
1.103

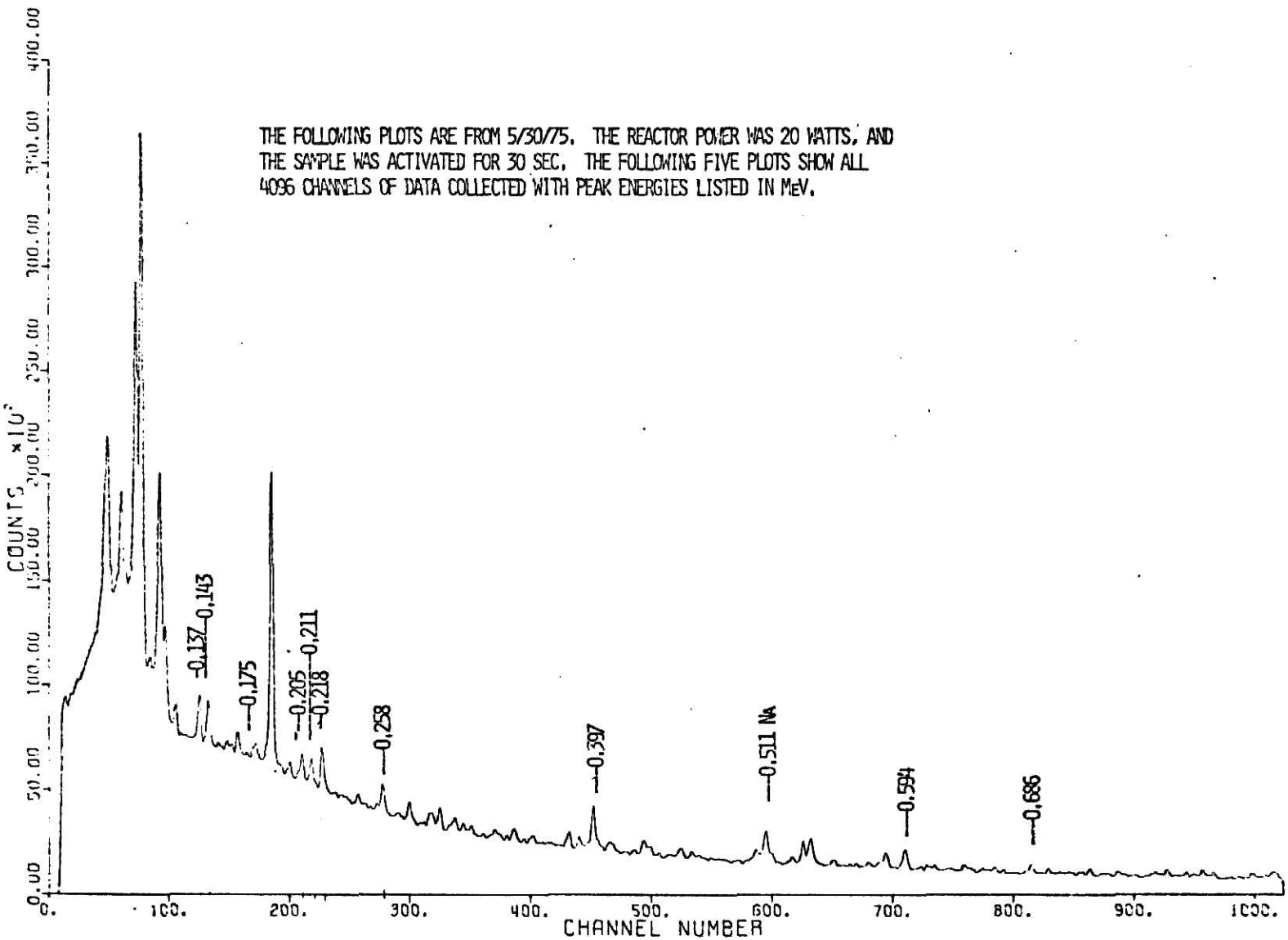

50 TO 52 SEC. AFTER FISSION 100 CYCLES 5/29/75 ++ 1193-2217
CHANNEL NUMBER
COUNTS
0.
100.
200.
300.
400.
500.
600.
700.
800.
900.
1000.
0.00
80.00
160.00
240.00
320.00
400.00
480.00
560.00
640.00
1.103
Co-60 1.173
1.219
Na-22 1.275
Co-60 1.332
1.501



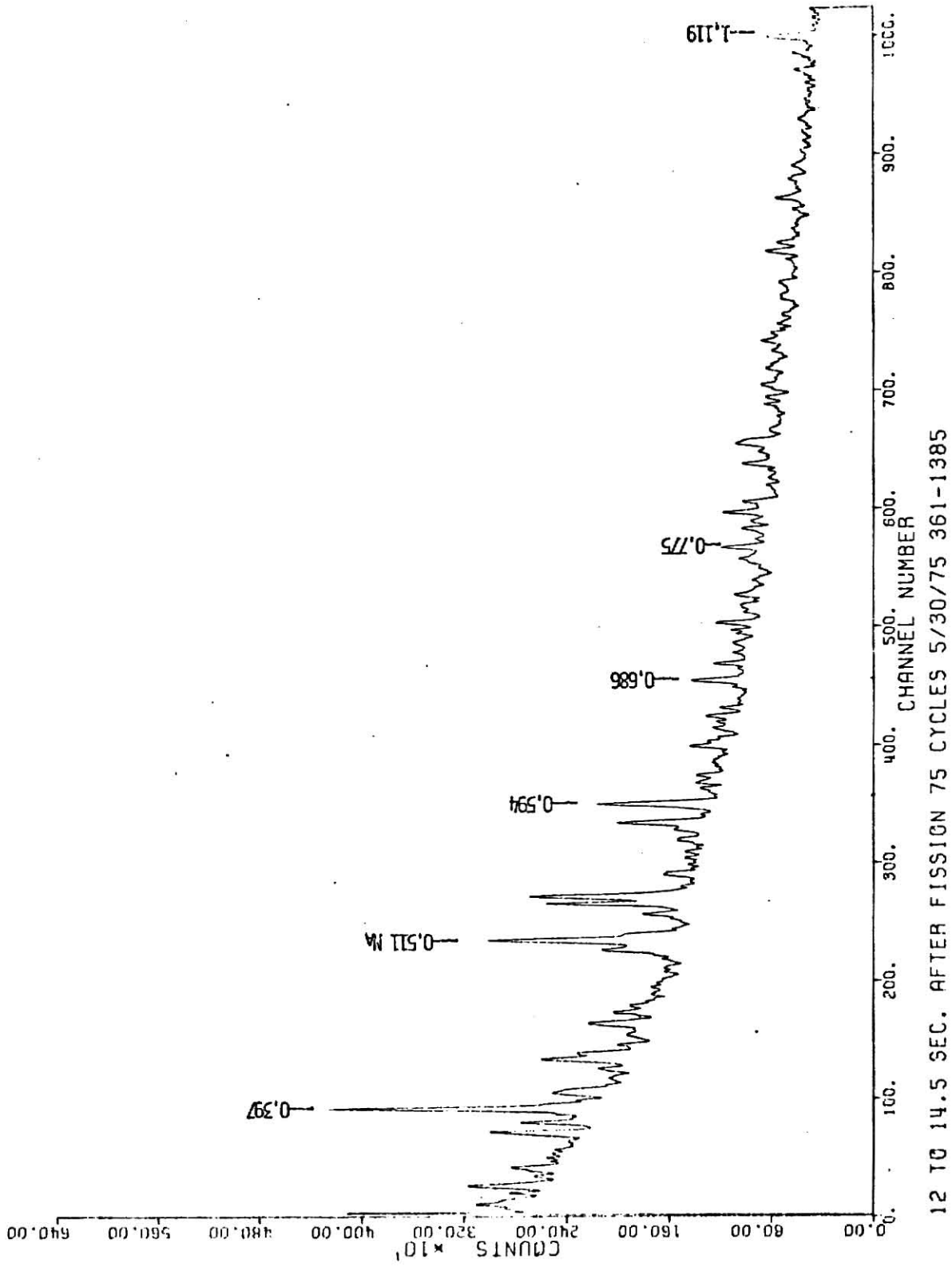
50 TO 52 SEC. AFTER FISSION 100 CYCLES 5/29/75 ++ 2145-3169

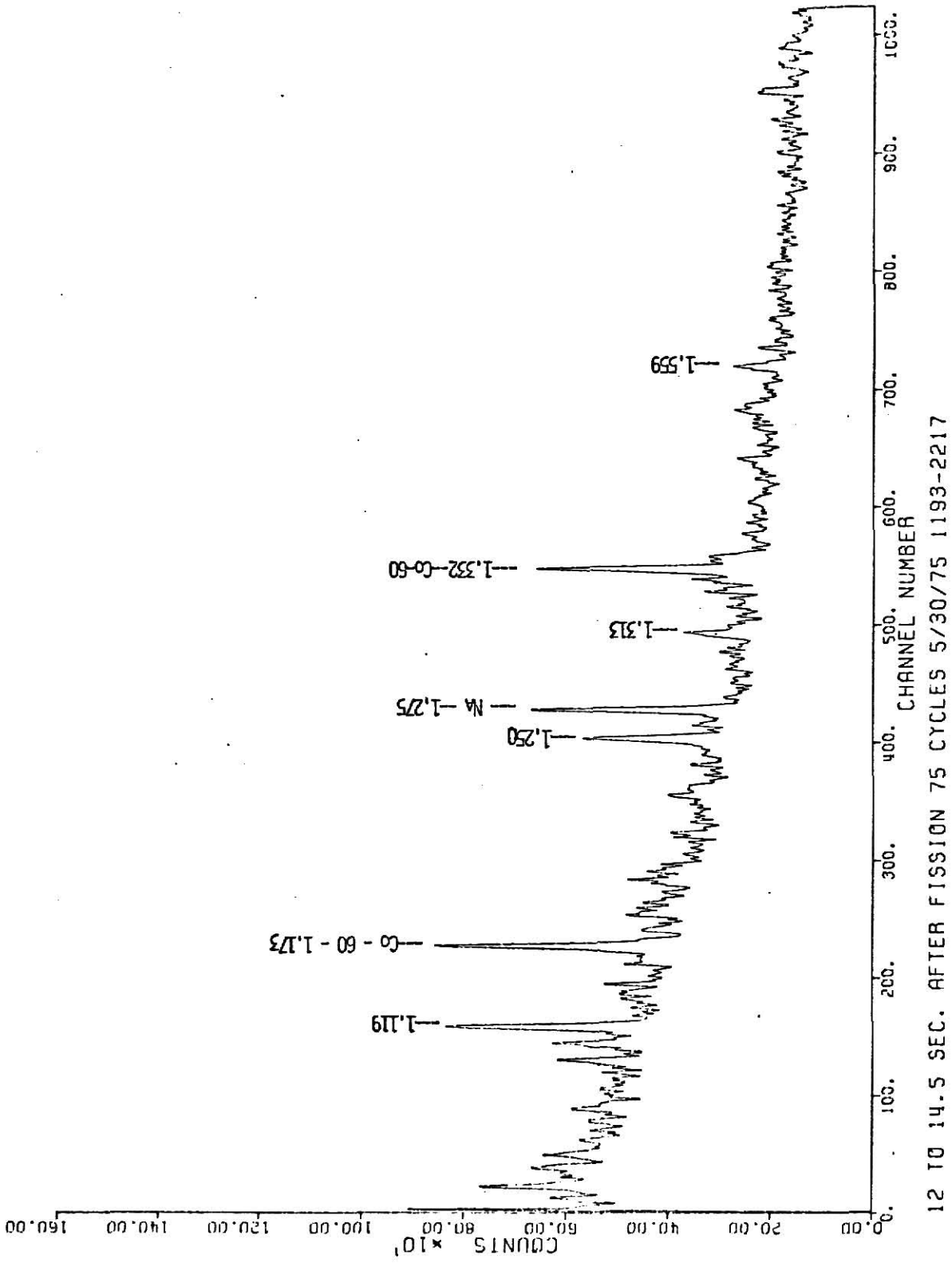


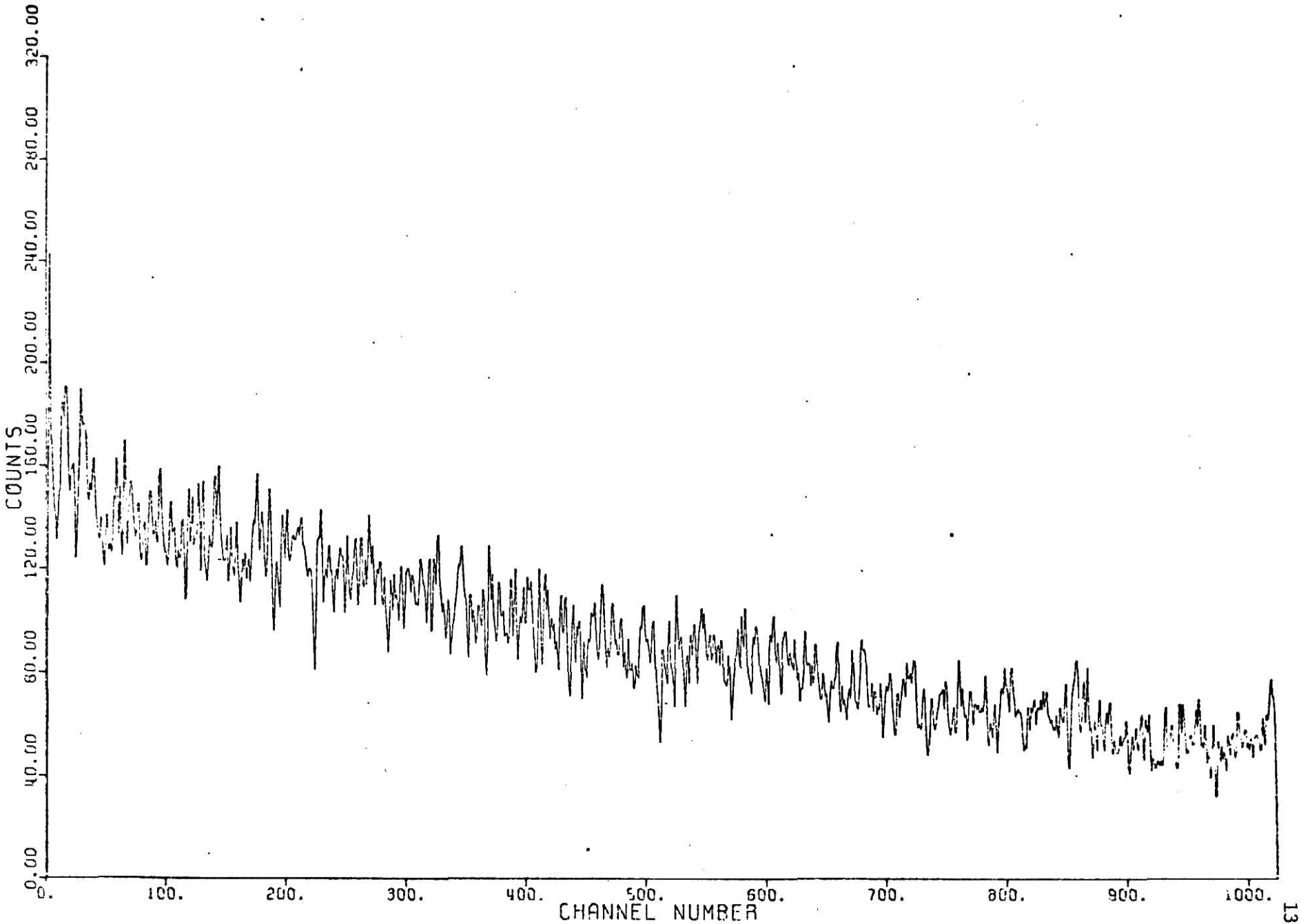
50 TO 52 SEC. AFTER FISSION 100 CYCLES 5/29/75 ++ 3072-4096



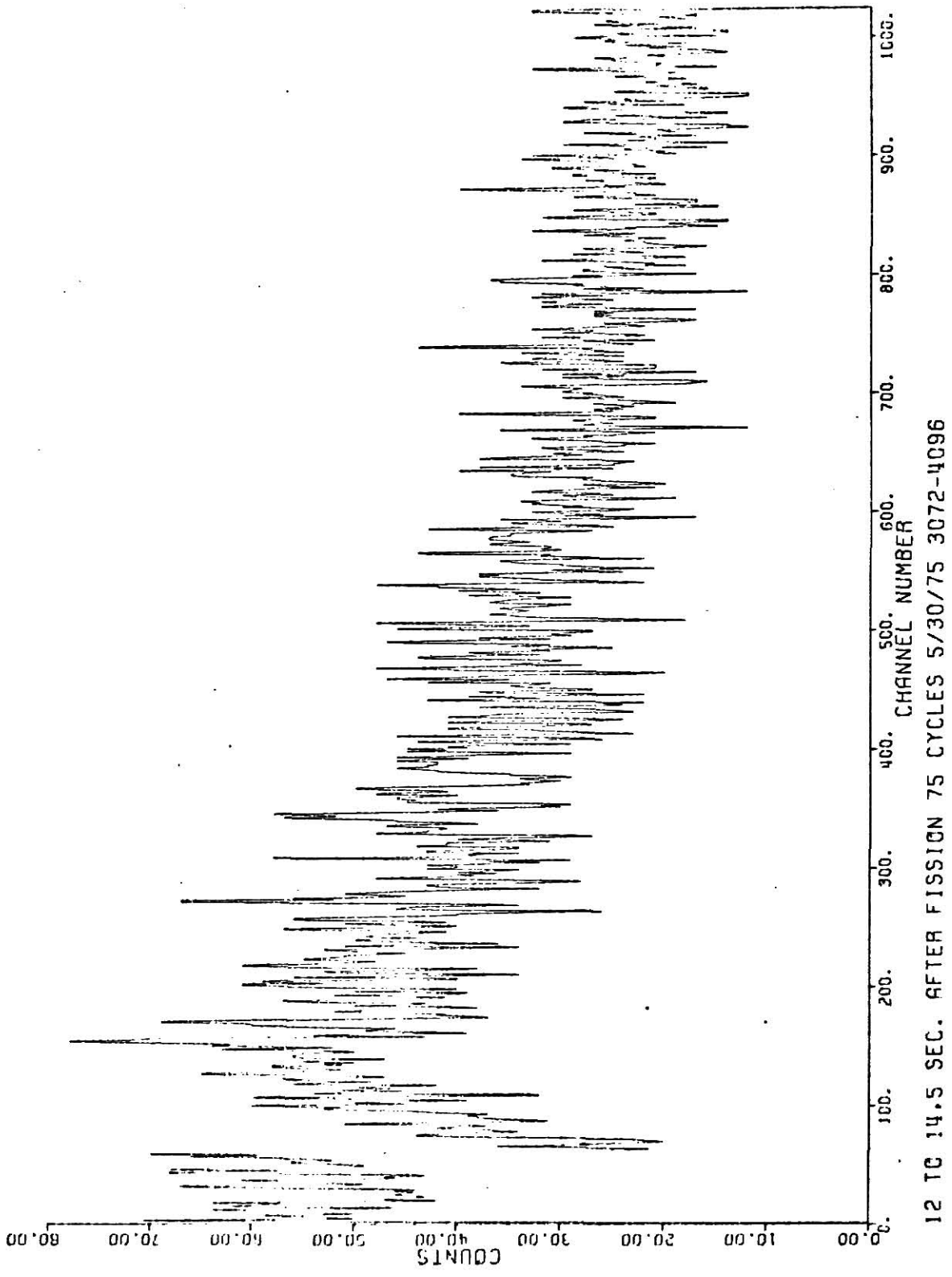
12 TO 14.5 SEC. AFTER FISSION 75 CYCLES 5/30/75 0-1024

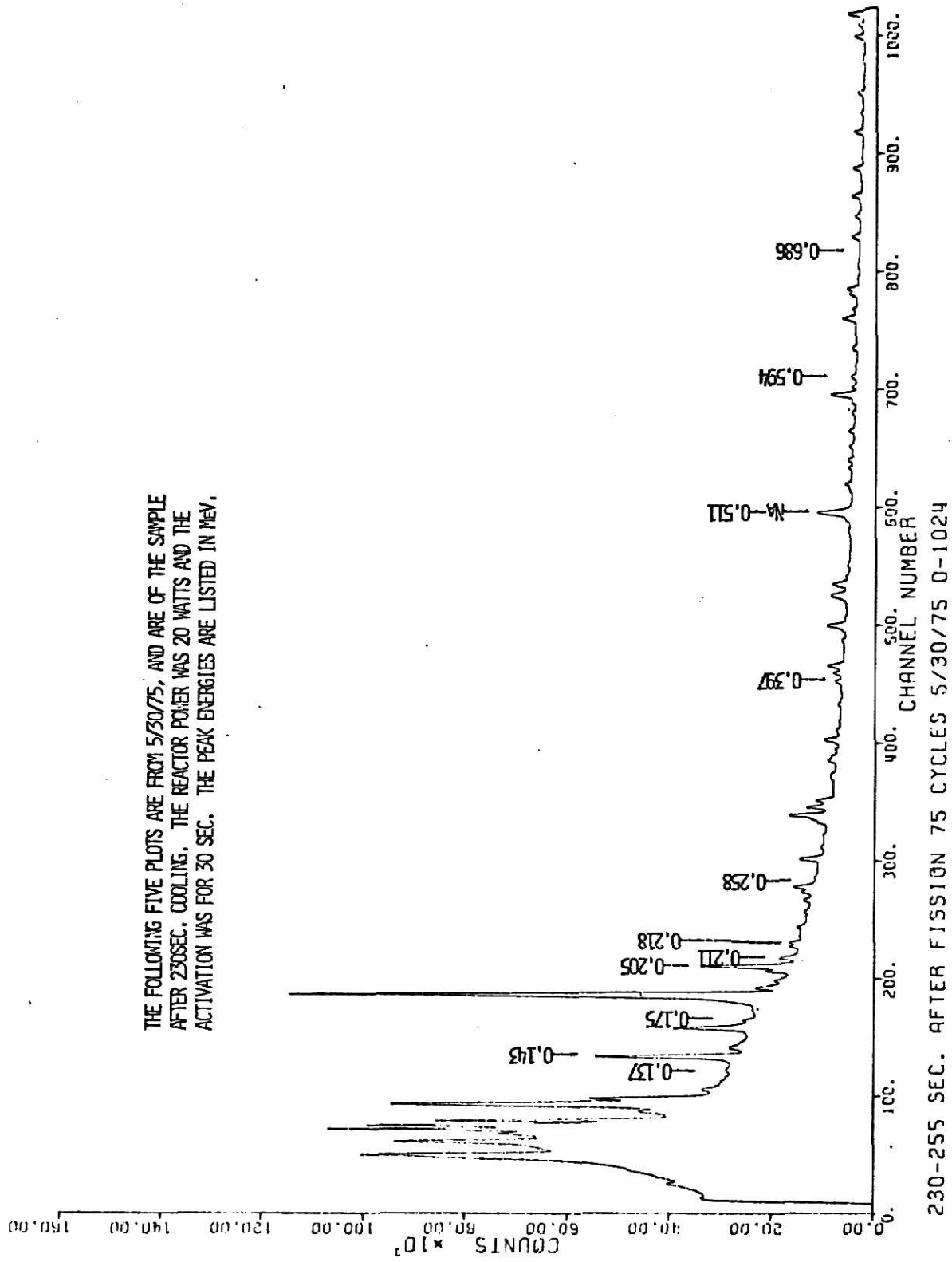

0.397
0.511 MA
0.594
0.686
0.775
1.119
COUNTS x10¹
0.00
80.00
160.00
240.00
320.00
400.00
480.00
560.00
640.00
0.
100.
200.
300.
400.
500.
600.
700.
800.
900.
1000.
CHANNEL NUMBER
12 TO 14.5 SEC. AFTER FISSION 75 CYCLES 5/30/75 361-1385

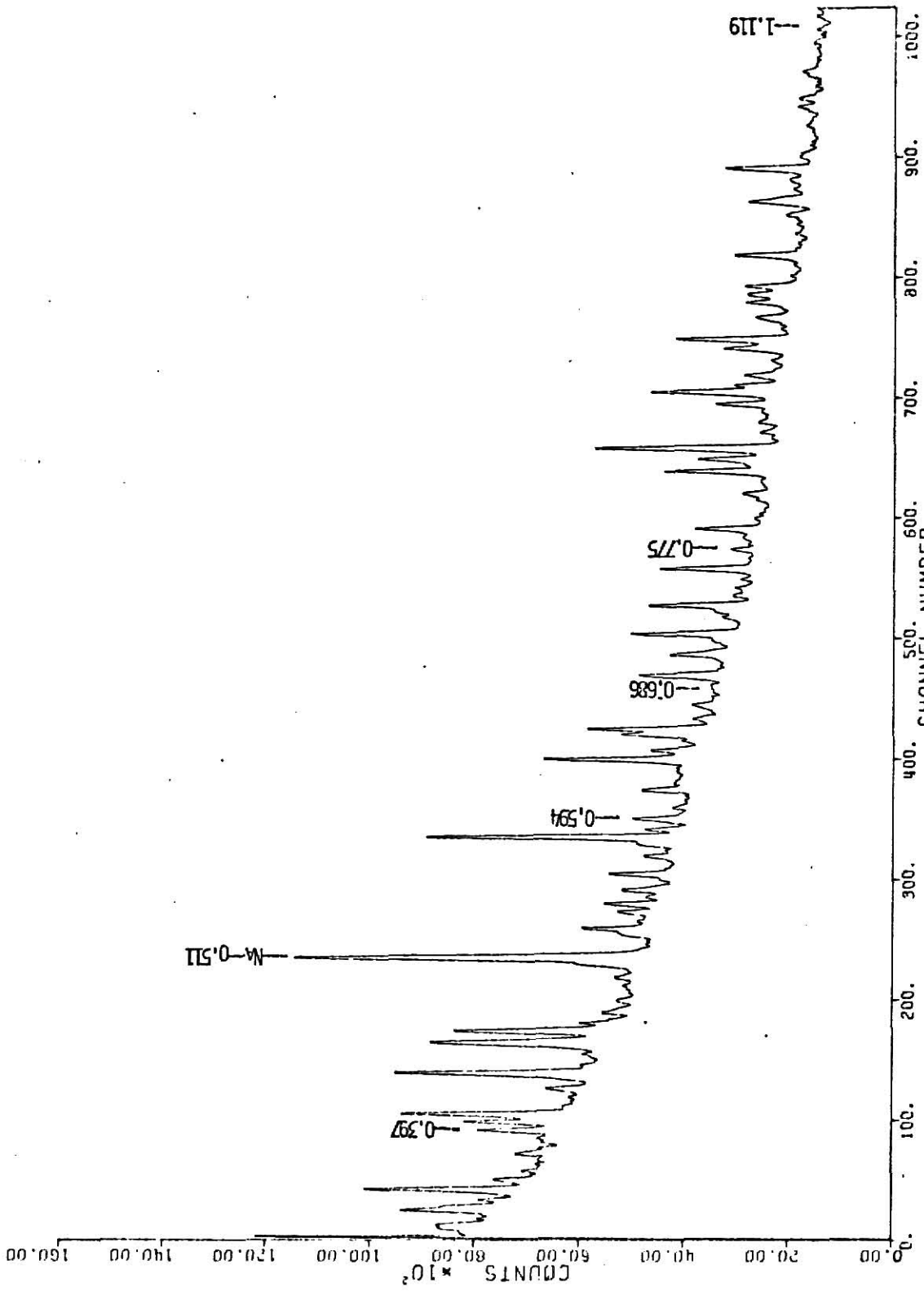

12 TO 14.5 SEC. AFTER FISSION 75 CYCLES 5/30/75 1193-2217
COUNTS x10¹
CHANNEL NUMBER
0.00
20.00
40.00
60.00
80.00
100.00
120.00
140.00
160.00
0.
100.
200.
300.
400.
500.
600.
700.
800.
900.
1000.
1,119
Co - 60 - 1,173
1,250
Na -1,275
1,313
1,332-Co-60
1,559

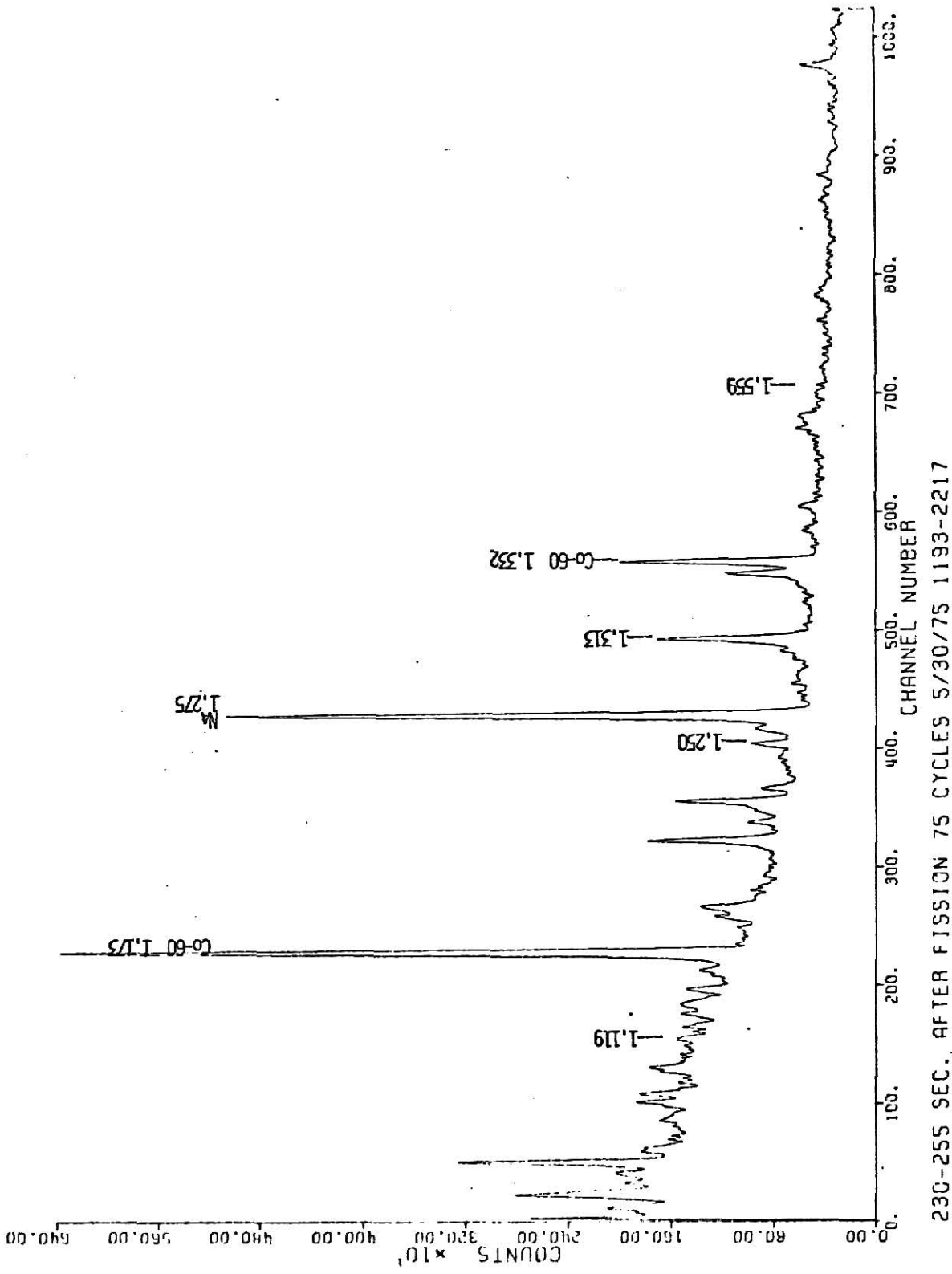


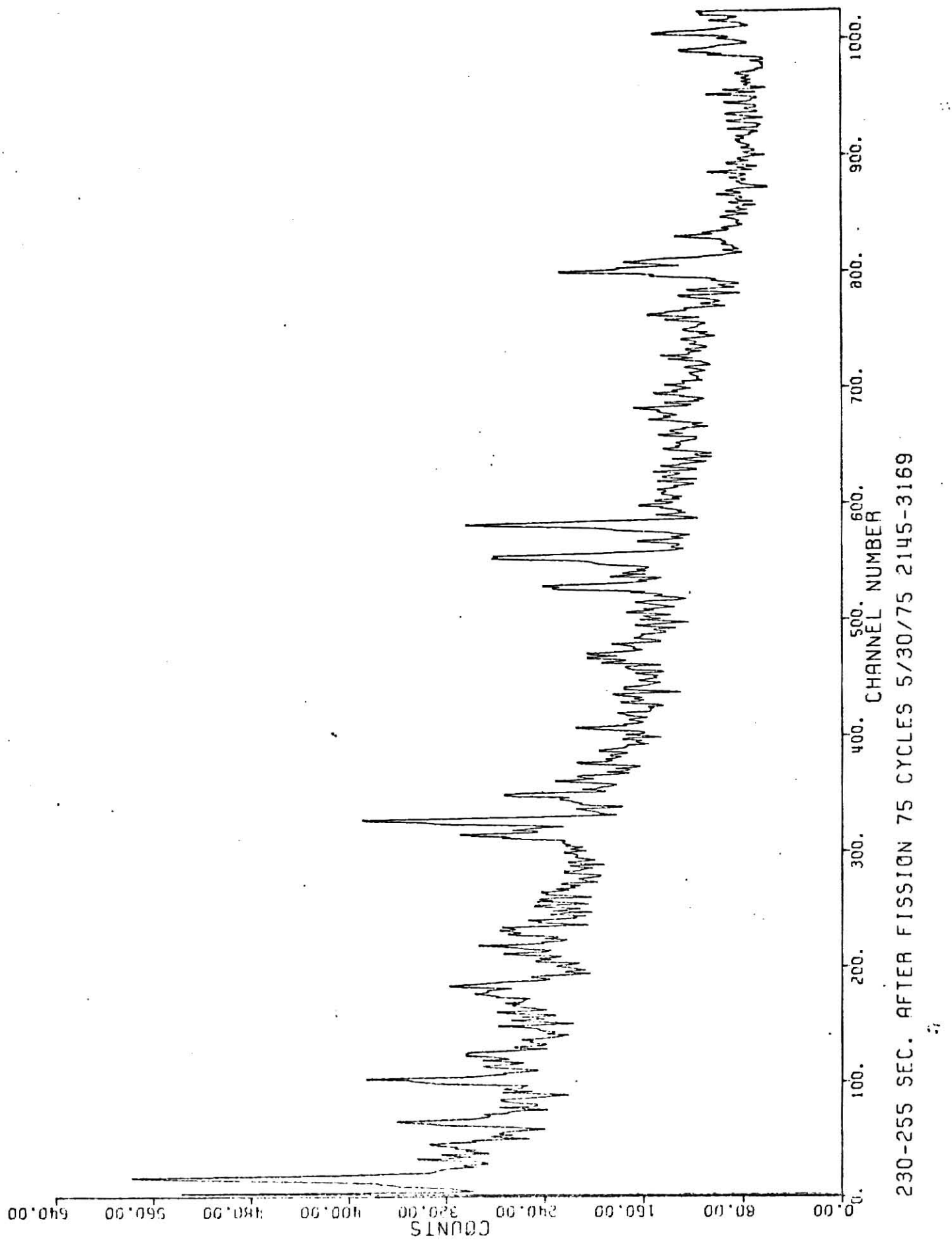
12 TO 14.5 SEC. AFTER FISSION 75 CYCLES 5/30/75 2145-3169

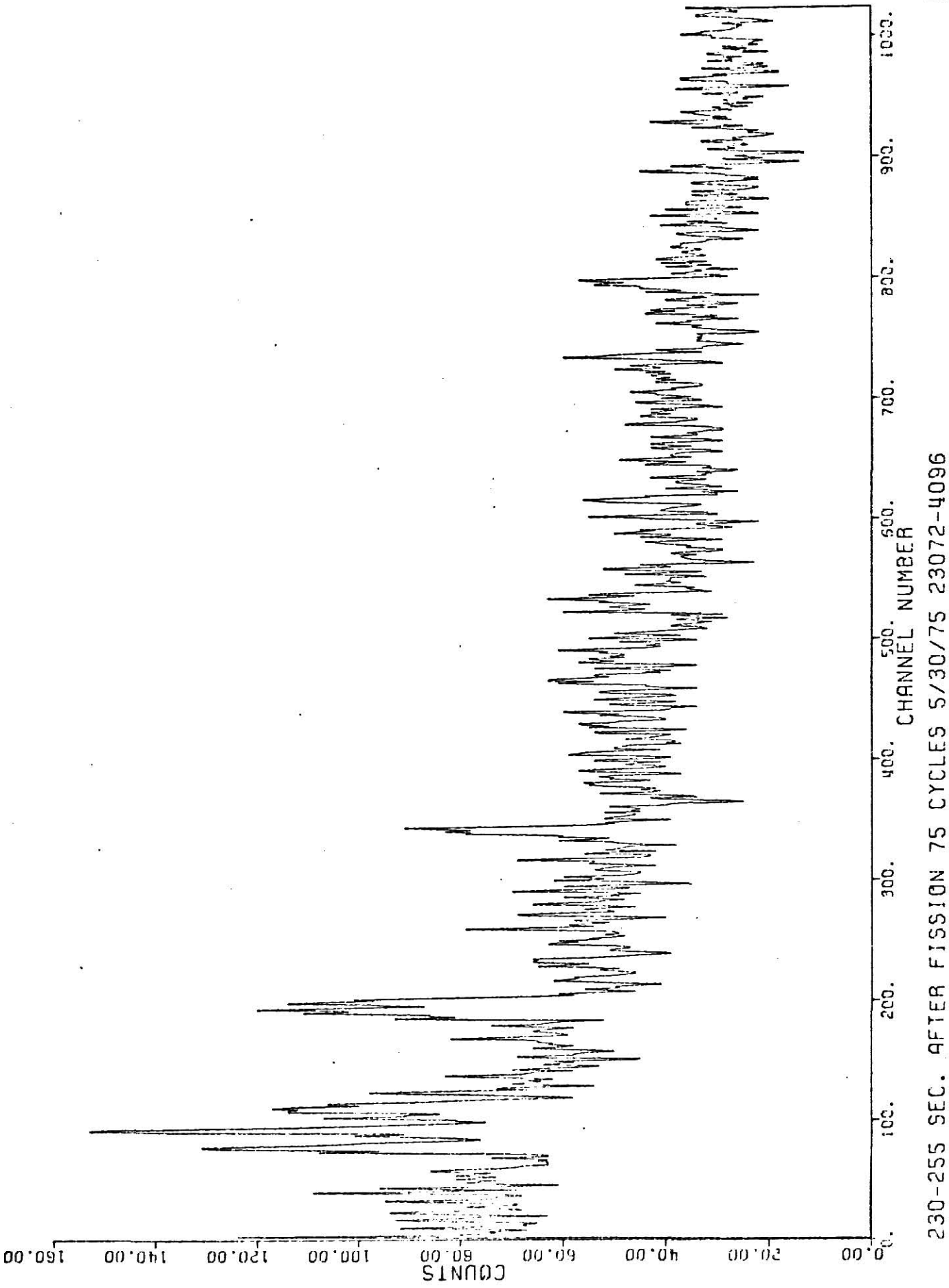

COUNTS
0.00
10.00
20.00
30.00
40.00
50.00
60.00
70.00
80.00
0.
100.
200.
300.
400.
500.
600.
700.
800.
900.
1000.
CHANNEL NUMBER
12 TO 14.5 SEC. AFTER FISSION 75 CYCLES 5/30/75 3072-4096

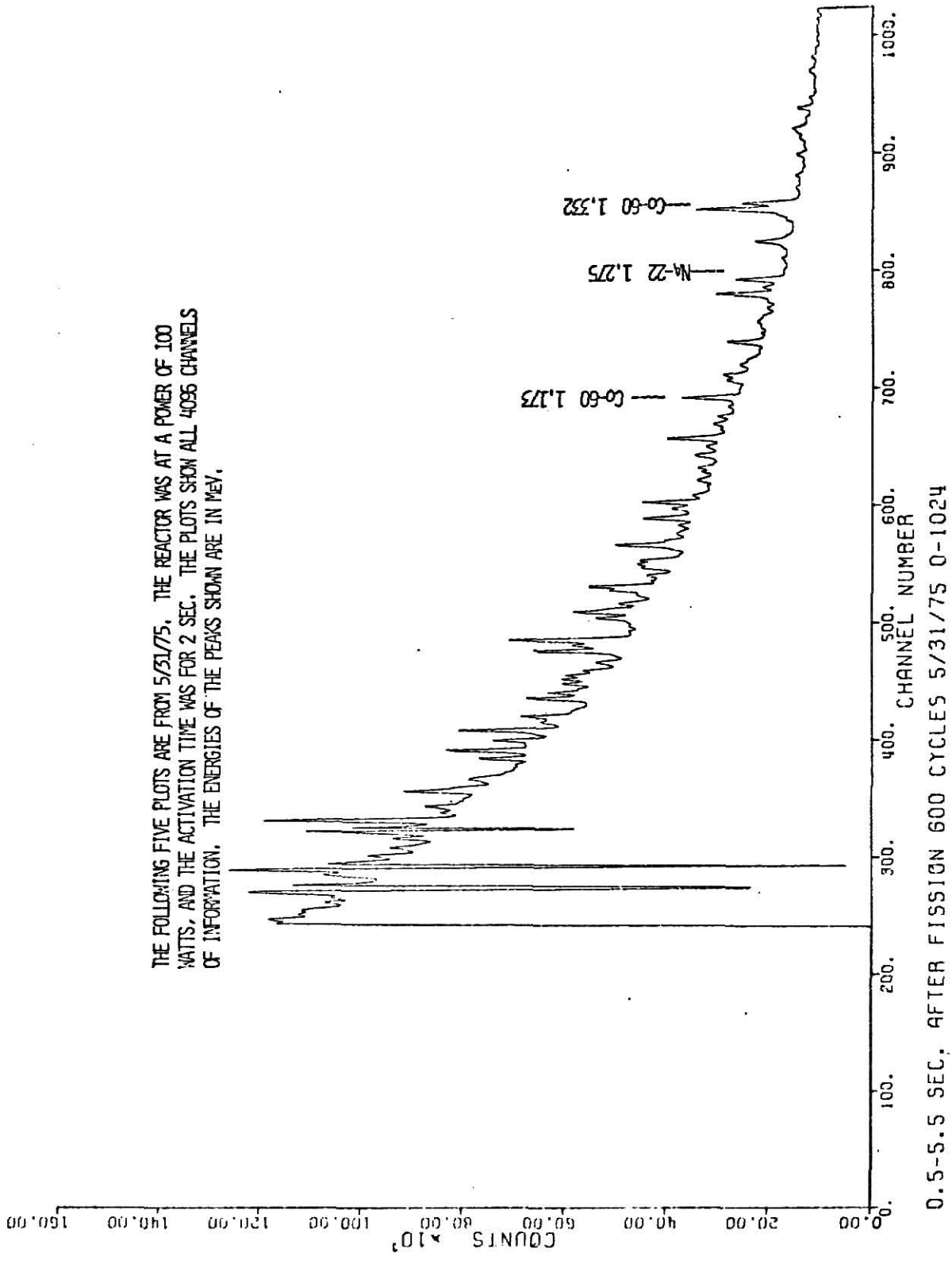

THE FOLLOWING FIVE PLOTS ARE FROM 5/30/75, AND ARE OF THE SAMPLE AFTER 230SEC. COOLING. THE REACTOR POWER WAS 20 WATTS AND THE ACTIVATION WAS FOR 30 SEC. THE PEAK ENERGIES ARE LISTED IN MeV.
0.686
0.594
0.511
0.397
0.258
0.218
0.211
0.205
0.175
0.143
0.137
COUNTS x10³
CHANNEL NUMBER
230-255 SEC. AFTER FISSION 75 CYCLES 5/30/75 0-1024

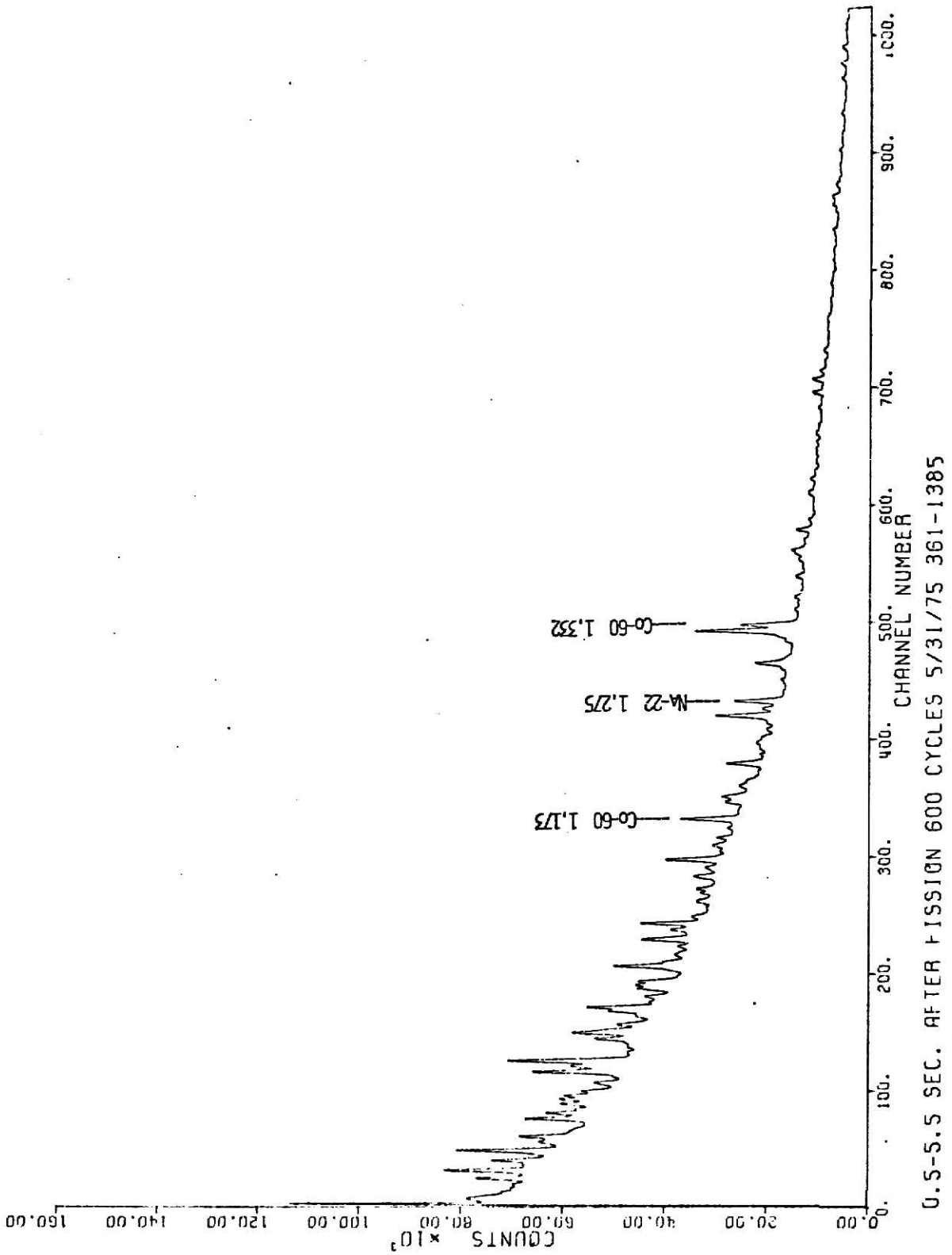

230-255 SEC. AFTER FISSION 75 CYCLES 5/30/75 361-1385
CHANNEL NUMBER
COUNTS ×10²
0.
100.
200.
300.
400.
500.
600.
700.
800.
900.
1000.
0.00
20.00
40.00
60.00
80.00
100.00
120.00
140.00
160.00
1.119
0.775
0.686
0.594
M-0.511
0.397

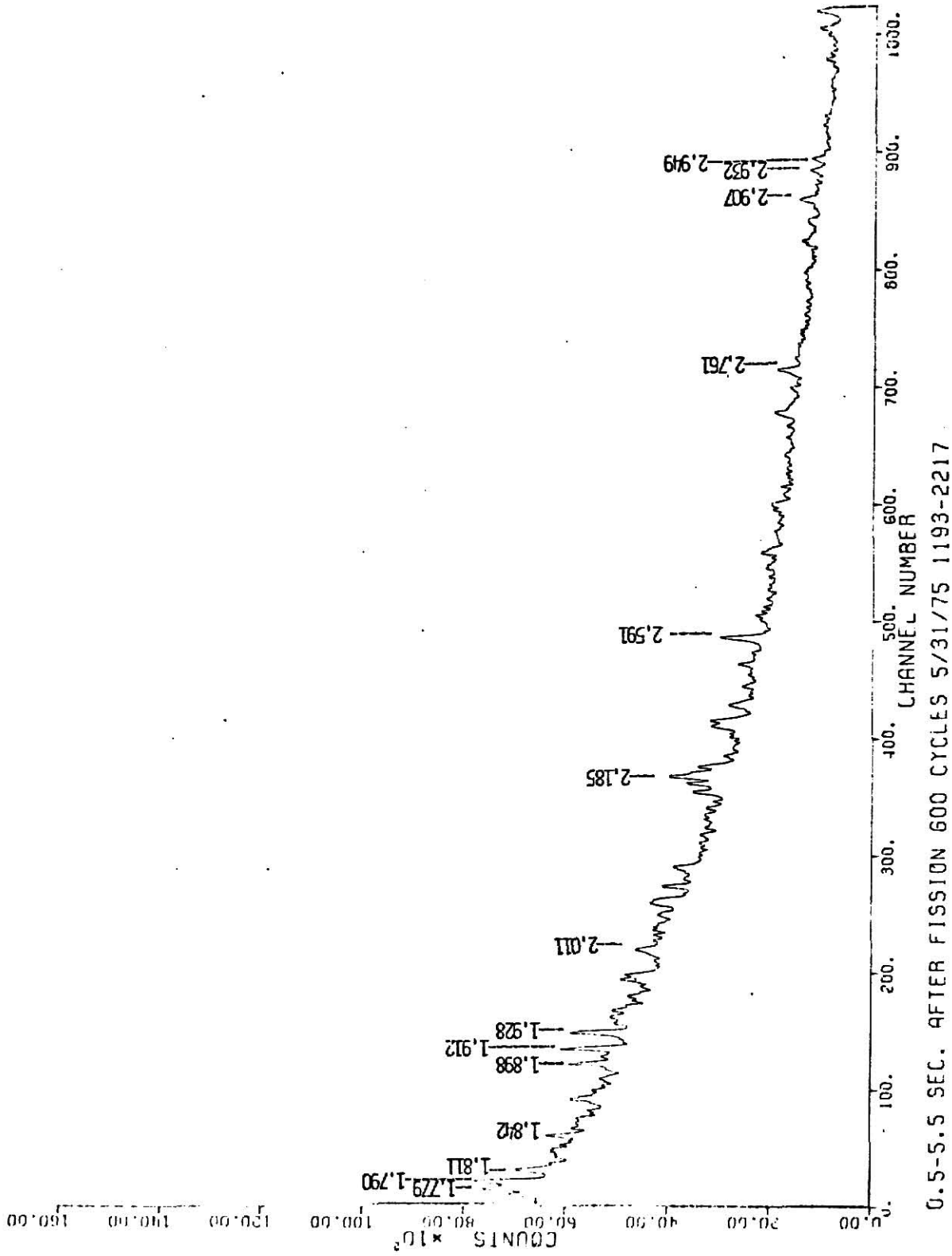

230-255 SEC. AFTER FISSION 75 CYCLES 5/30/75 1193-2217
CHANNEL NUMBER
COUNTS ×10¹
0.
100.
200.
300.
400.
500.
600.
700.
800.
900.
1000.
0.00
80.00
160.00
240.00
320.00
400.00
480.00
560.00
640.00
1,119
Co-60 1,173
1,250
Na 1,275
1,313
Co-60 1,332
1,599

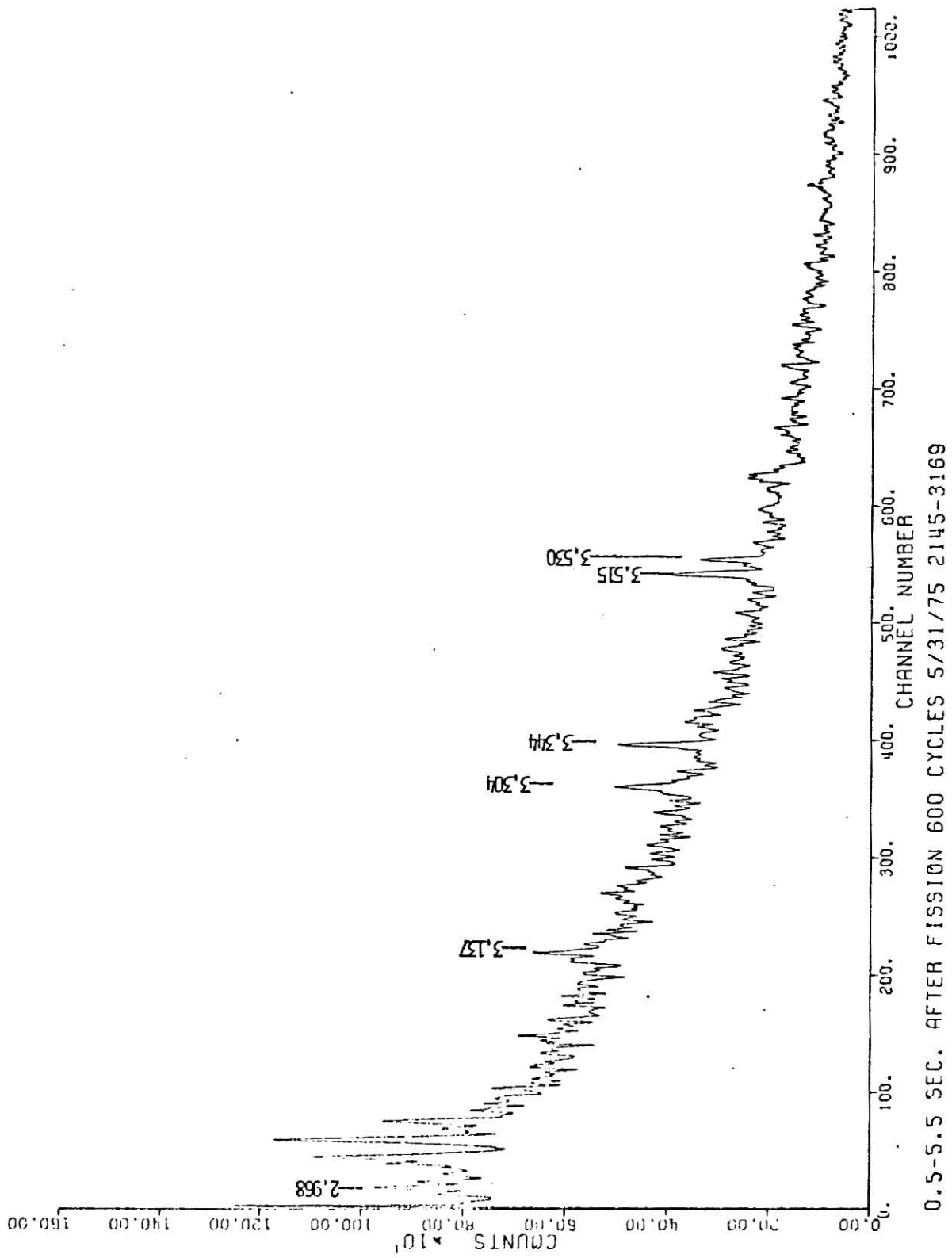

COUNTS
0.00
80.00
160.00
240.00
320.00
400.00
480.00
560.00
640.00
0.
100.
200.
300.
400.
500.
600.
700.
800.
900.
1000.
CHANNEL NUMBER
230-255 SEC. AFTER FISSION 75 CYCLES 5/30/75 2145-3169

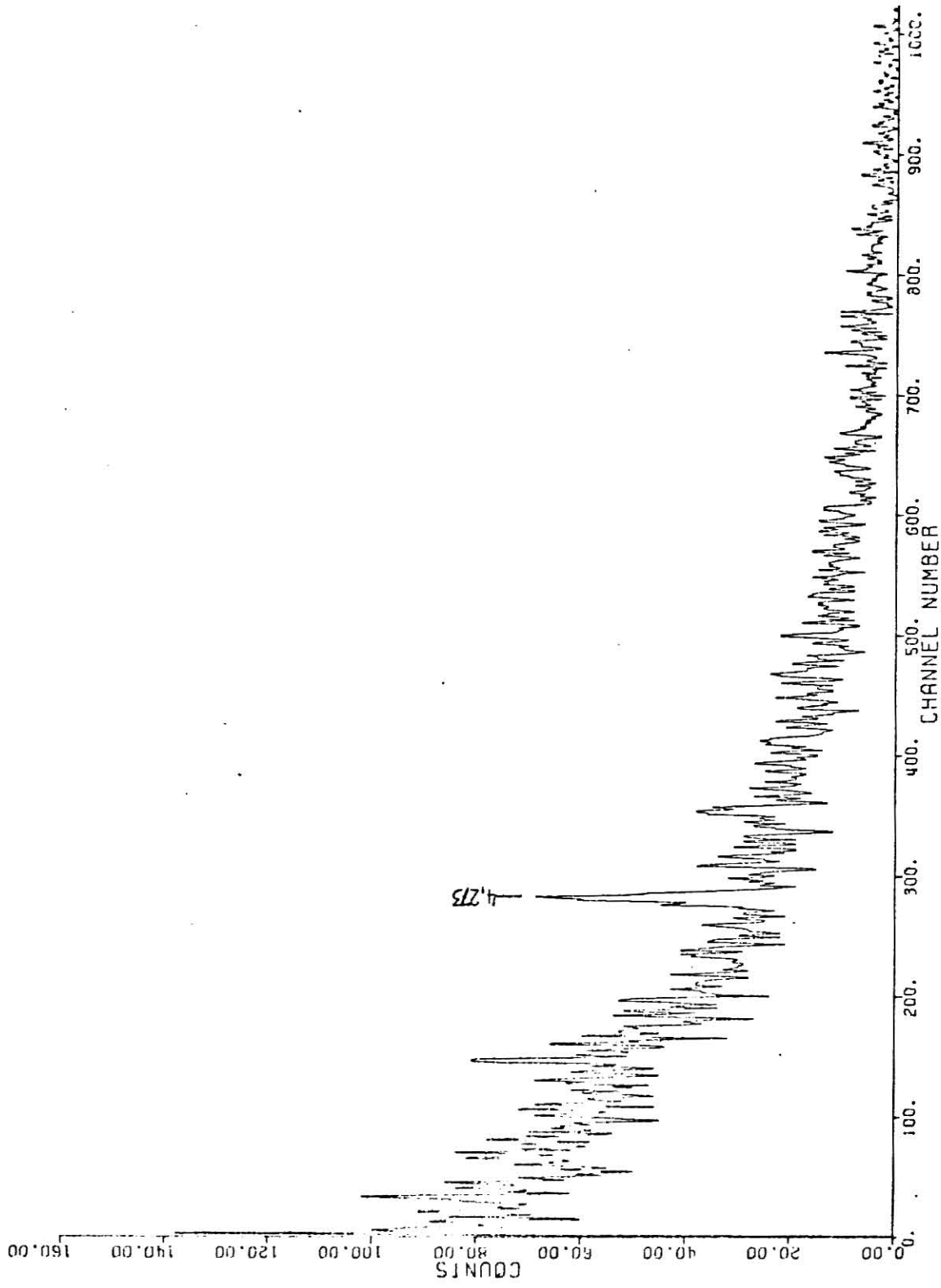

230-255 SEC. AFTER FISSION 75 CYCLES 5/30/75 23072-4096
CHANNEL NUMBER
0.
100.
200.
300.
400.
500.
600.
700.
800.
900.
1000.
COUNTS
0.00
20.00
40.00
60.00
80.00
100.00
120.00
140.00
160.00


THE FOLLOWING FIVE PLOTS ARE FROM 5/31/75. THE REACTOR WAS AT A POWER OF 100 WATTS, AND THE ACTIVATION TIME WAS FOR 2 SEC. THE PLOTS SHOW ALL 4096 CHANNELS OF INFORMATION. THE ENERGIES OF THE PEAKS SHOWN ARE IN MEV.
Co-60 1.332
Na-22 1.275
Co-60 1.173
0.5-5.5 SEC. AFTER FISSION 600 CYCLES 5/31/75 0-1024
COUNTS x10³
CHANNEL NUMBER
0.00
20.00
40.00
60.00
80.00
100.00
120.00
140.00
160.00
0.
100.
200.
300.
400.
500.
600.
700.
800.
900.
1000.


0.5-5.5 SEC. AFTER FISSION 600 CYCLES 5/31/75 361-1385
CHANNEL NUMBER
0.
100.
200.
300.
400.
500.
600.
700.
800.
900.
1000.
COUNTS ×10³
0.00
20.00
40.00
60.00
80.00
100.00
120.00
140.00
160.00
Co-60 1.173
Na-22 1.275
Co-60 1.332

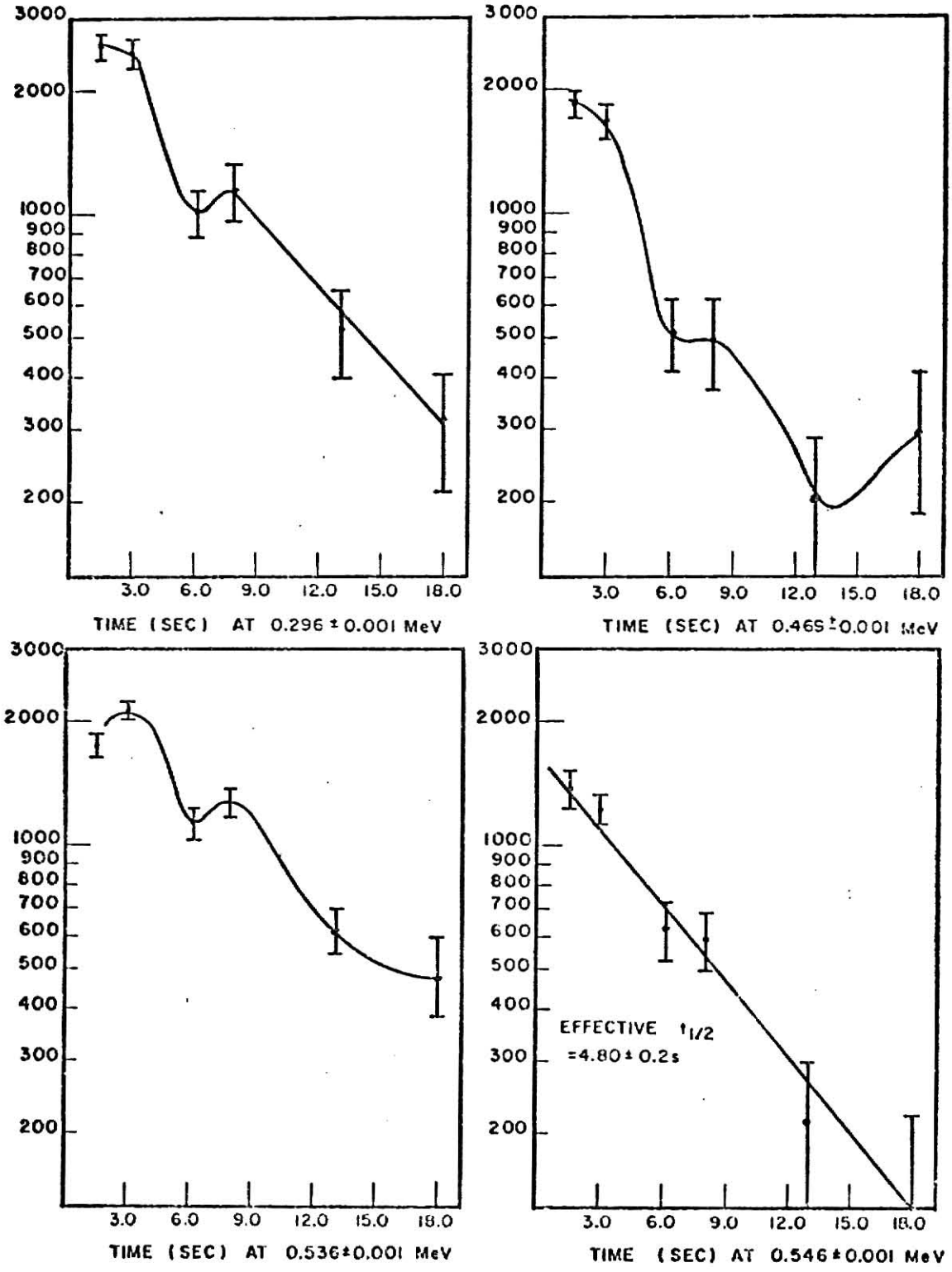

0.5-5.5 SEC. AFTER FISSION 600 CYCLES 5/31/75 1193-2217
CHANNEL NUMBER
COUNTS x10²
1,729
1,790
1,811
1,842
1,898
1,912
1,928
2,011
2,185
2,591
2,761
2,907
2,922
2,949

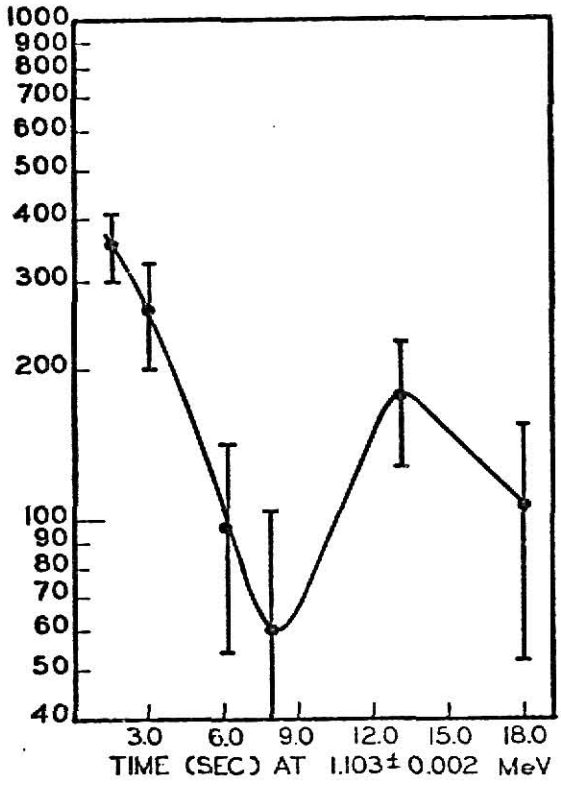

0.5-5.5 SEC. AFTER FISSION 600 CYCLES 5/31/75 2145-3169
CHANNEL NUMBER
COUNTS x10¹
2,968
3,137
3,304
3,344
3,515
3,530

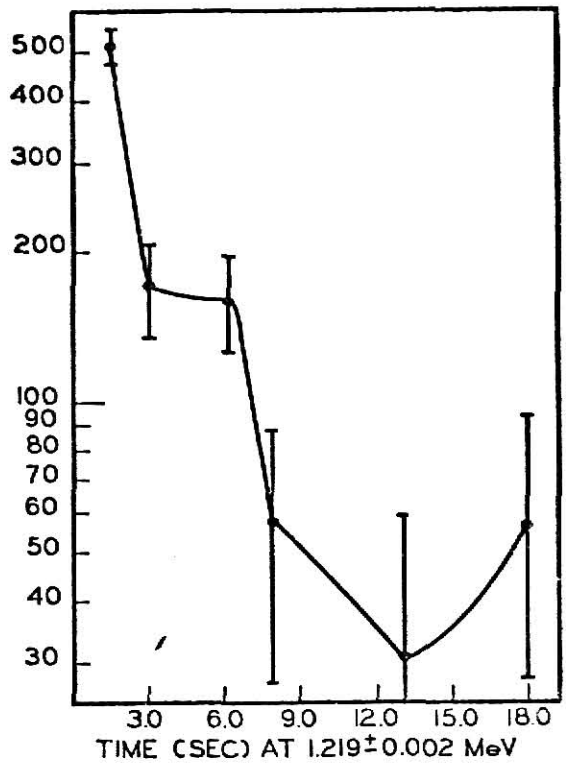

0.5-5.5 SEC. AFTER FISSION 600 CYCLES 5/31/75 3072-4096
CHANNEL NUMBER
0.
100.
200.
300.
400.
500.
600.
700.
800.
900.
1000.
COUNTS
0.00
20.00
40.00
60.00
80.00
100.00
120.00
140.00
160.00
4,273

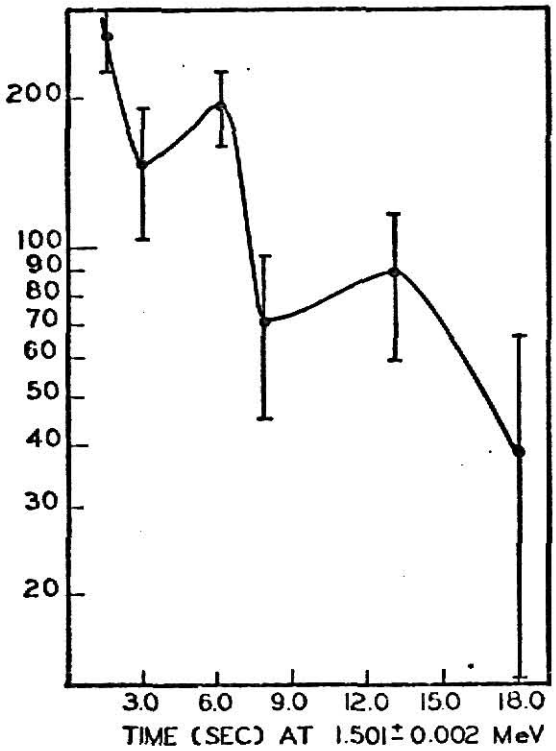
Appendix H

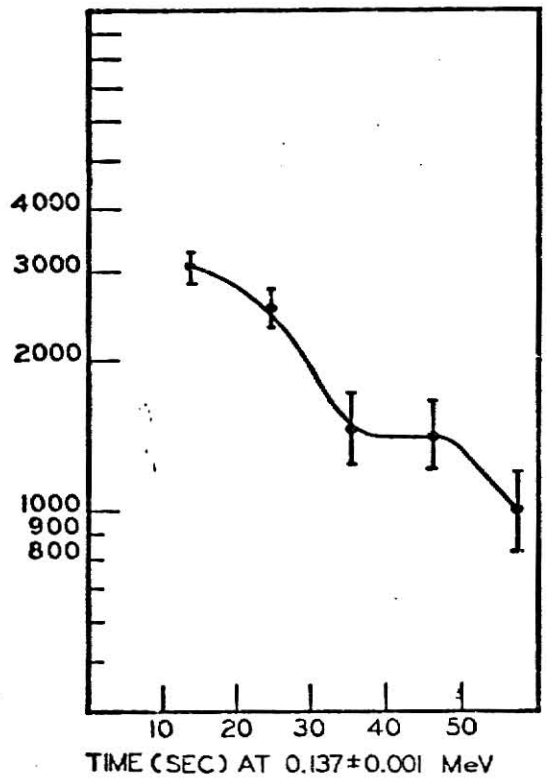
Plots of the log of the observed intensity versus time for the major gamma-ray energies observed.

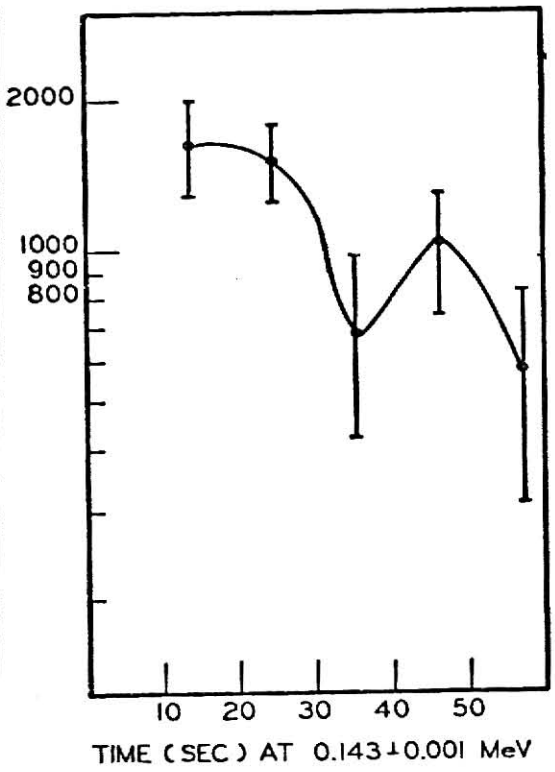

3000
2000
1000
900
800
700
600
500
400
300
200
3.0
6.0
9.0
12.0
15.0
18.0
TIME (SEC) AT 0.296 ± 0.001 MeV
TIME (SEC) AT 0.465 ± 0.001 MeV
TIME (SEC) AT 0.536 ± 0.001 MeV
EFFECTIVE $t_{1/2}$
=4.80 ± 0.2 s
TIME (SEC) AT 0.546 ± 0.001 MeV

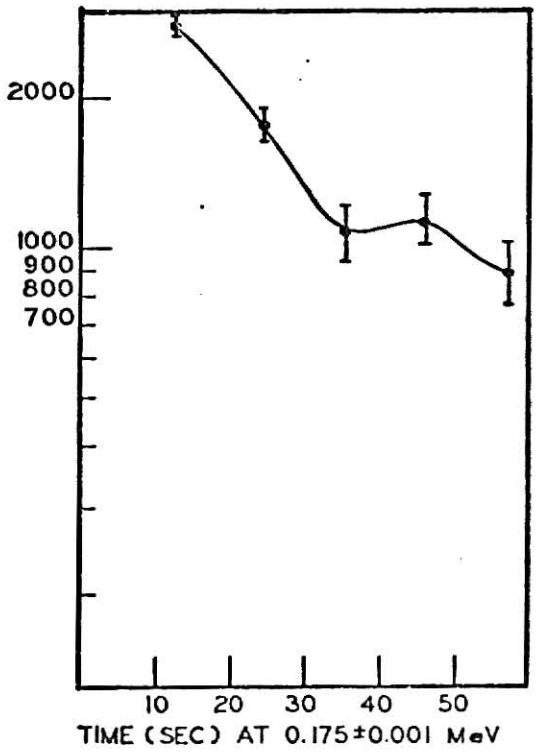

1000
900
800
700
600
500
400
300
200
100
90
80
70
60
50
40
3.0
6.0
9.0
12.0
15.0
18.0
TIME (SEC) AT 1.103± 0.002 MeV

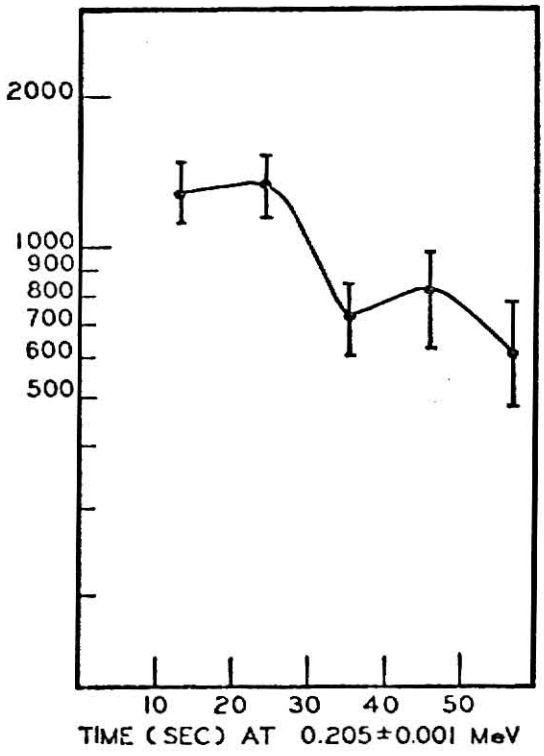

500
400
300
200
100
90
80
70
60
50
40
30
3.0
6.0
9.0
12.0
15.0
18.0
TIME (SEC) AT 1.219± 0.002 MeV

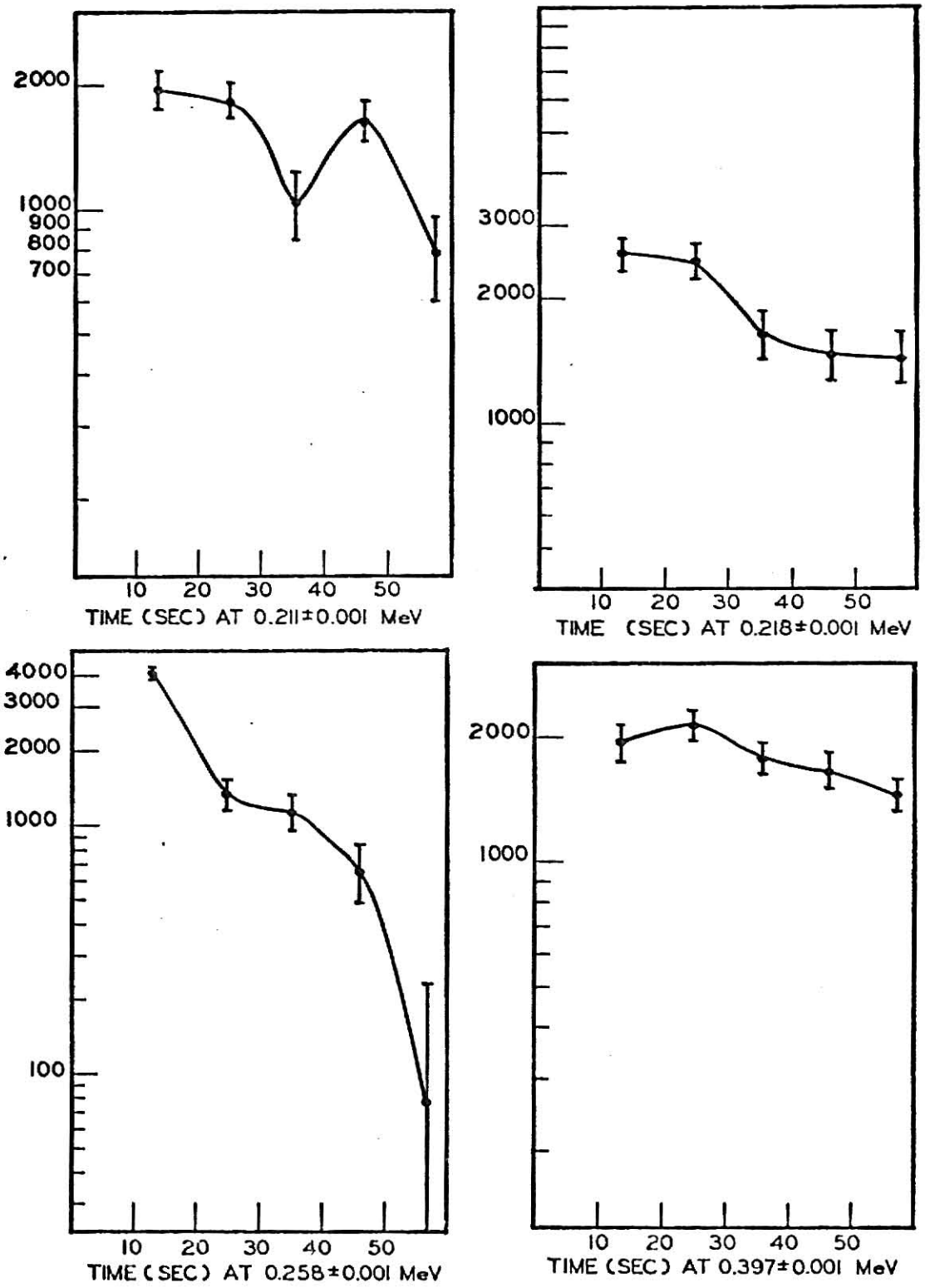

200
100
90
80
70
60
50
40
30
20
3.0
6.0
9.0
12.0
15.0
18.0
TIME (SEC) AT 1.501± 0.002 MeV

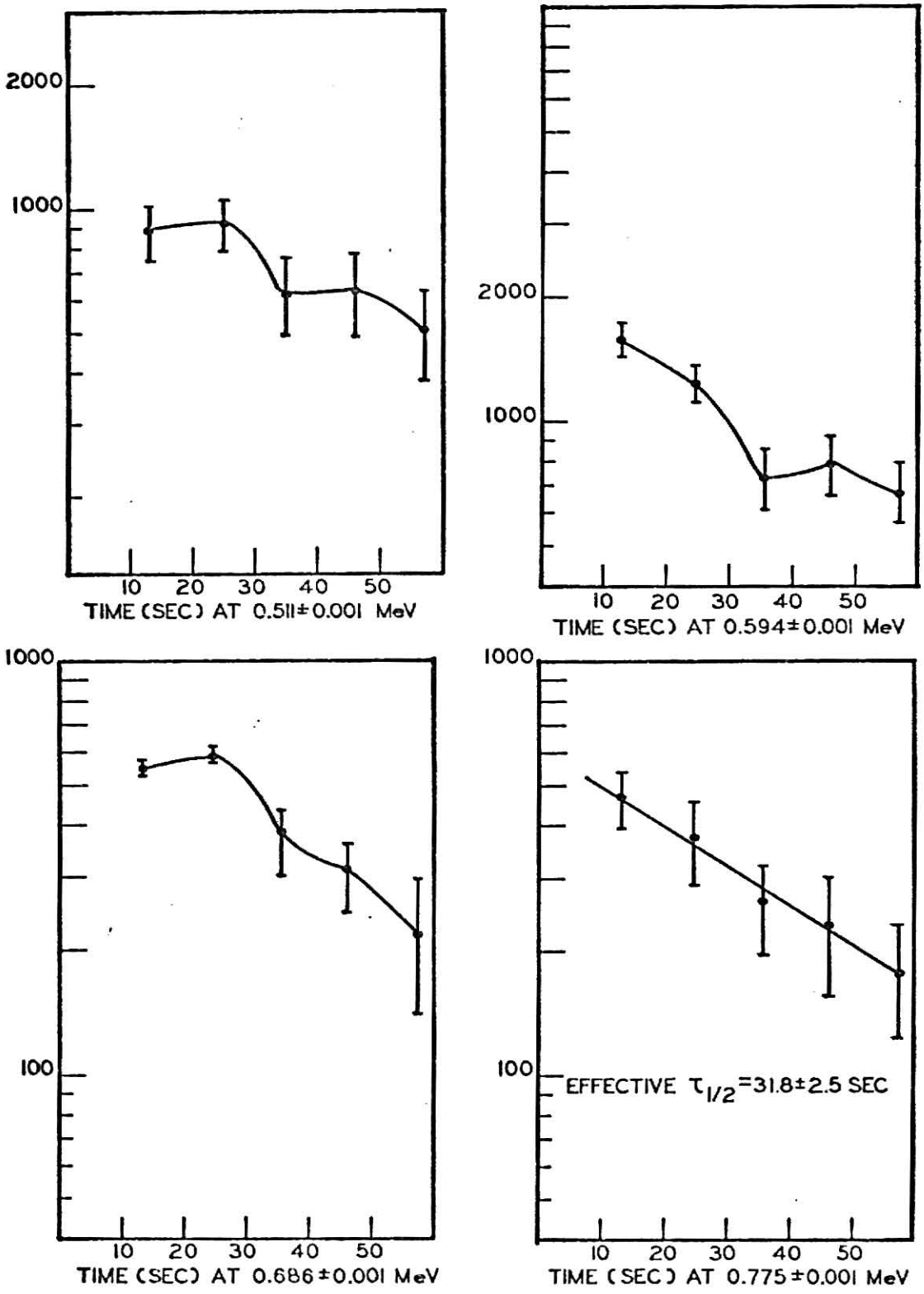

4000
3000
2000
1000
900
800
10 20 30 40 50
TIME (SEC) AT 0.137±0.001 MeV

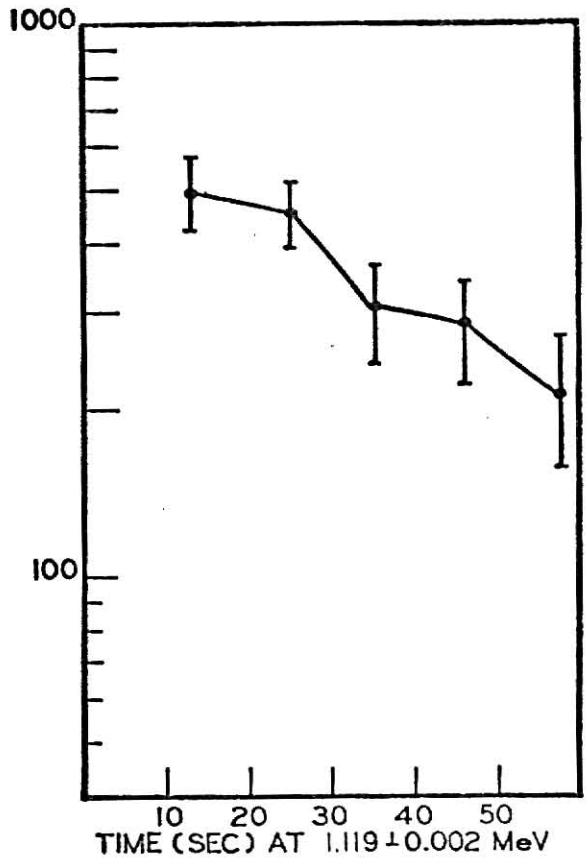

2000
1000
900
800
10 20 30 40 50
TIME (SEC) AT 0.143±0.001 MeV

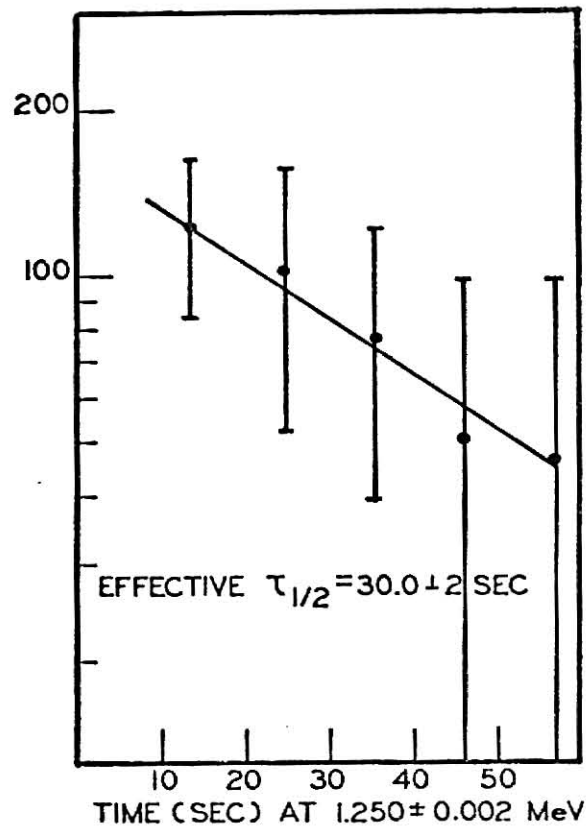

2000
1000
900
800
700
10 20 30 40 50
TIME (SEC) AT 0.175±0.001 MeV

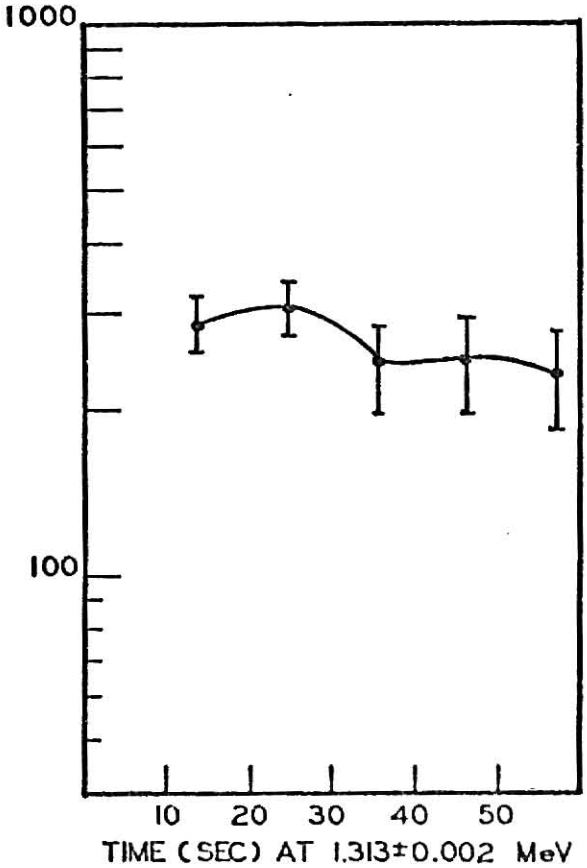

2000
1000
900
800
700
600
500
10 20 30 40 50
TIME (SEC) AT 0.205±0.001 MeV

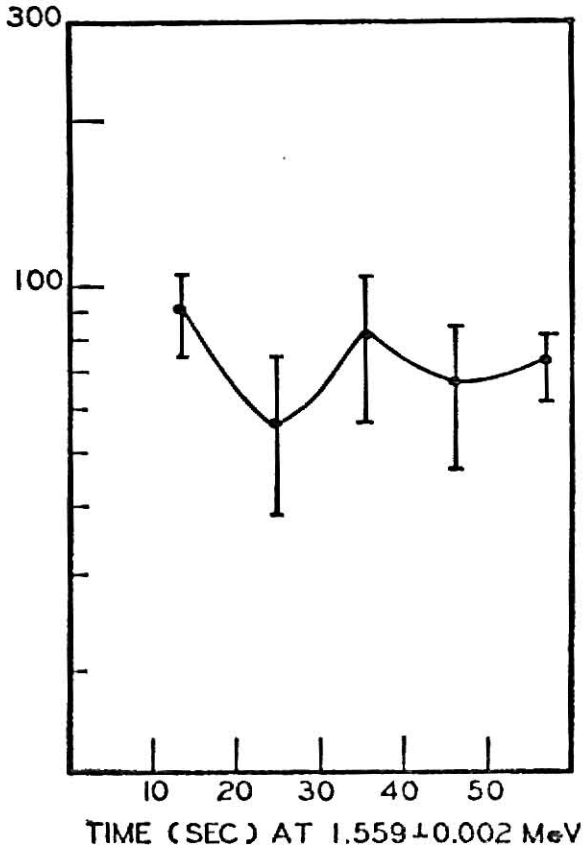

2000
1000
900
800
700
10 20 30 40 50
TIME (SEC) AT 0.211±0.001 MeV
3000
2000
1000
10 20 30 40 50
TIME (SEC) AT 0.218±0.001 MeV
4000
3000
2000
1000
100
10 20 30 40 50
TIME (SEC) AT 0.258±0.001 MeV
2000
1000
10 20 30 40 50
TIME (SEC) AT 0.397±0.001 MeV

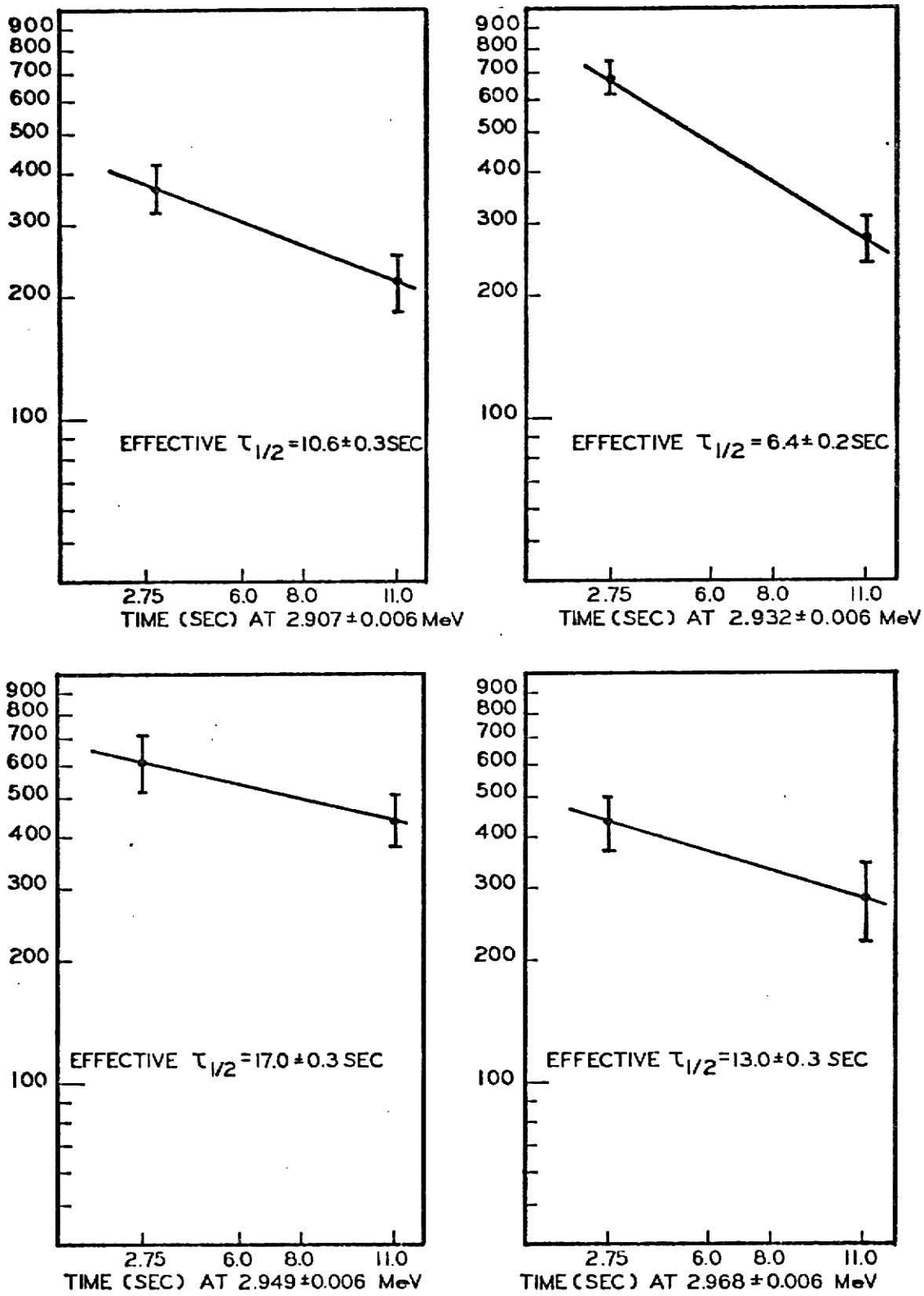

2000
1000
10 20 30 40 50
TIME (SEC) AT 0.511±0.001 MeV
2000
1000
10 20 30 40 50
TIME (SEC) AT 0.594±0.001 MeV
1000
100
10 20 30 40 50
TIME (SEC) AT 0.686±0.001 MeV
1000
100
EFFECTIVE $\tau_{1/2}$=31.8±2.5 SEC
10 20 30 40 50
TIME (SEC) AT 0.775±0.001 MeV

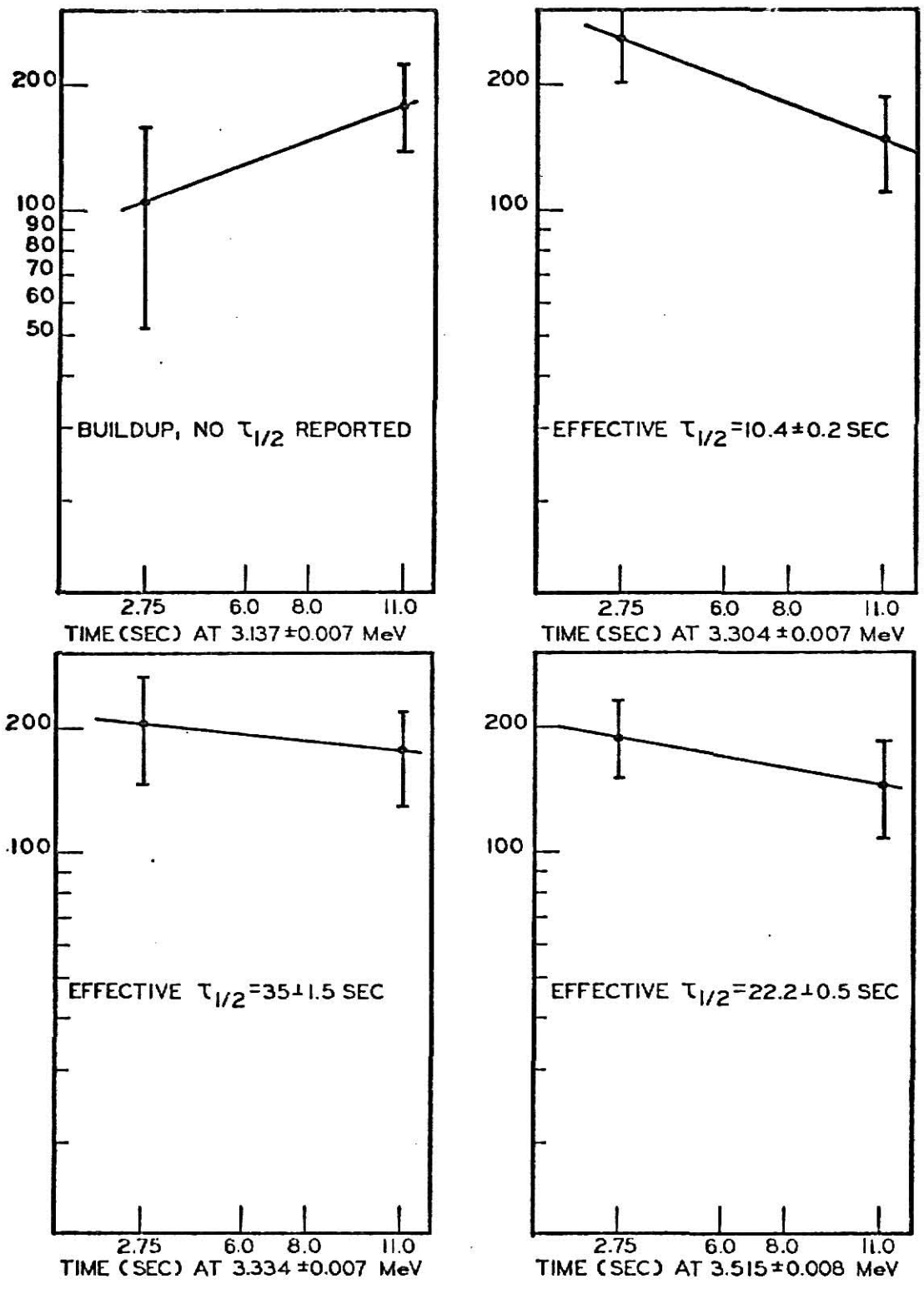

1000
100
10 20 30 40 50
TIME (SEC) AT 1.119±0.002 MeV

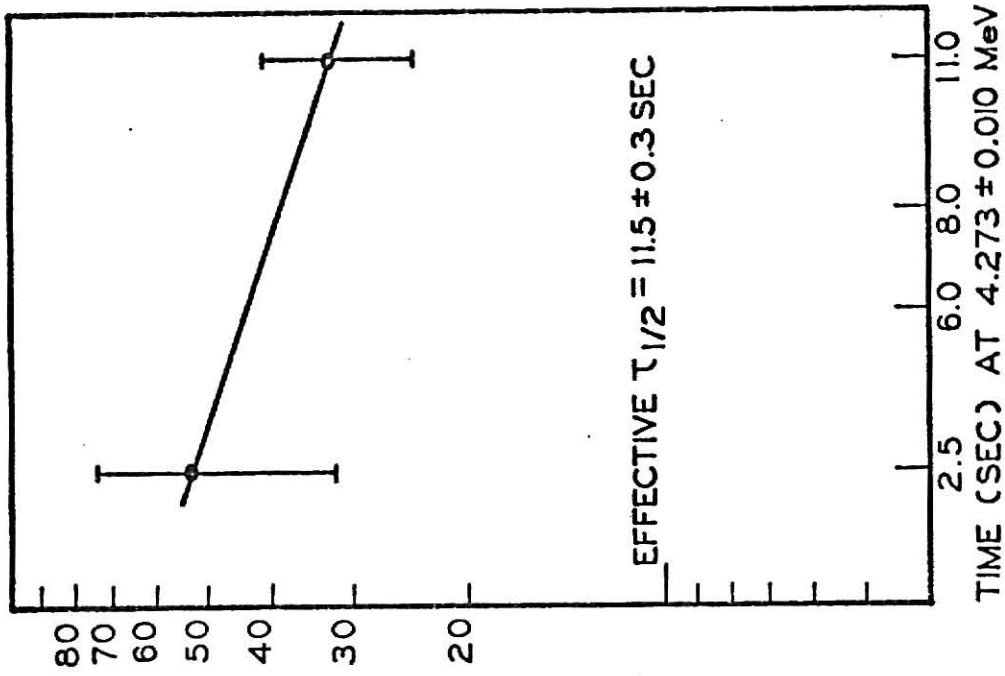

200
100
EFFECTIVE $\tau_{1/2}$=30.0±2 SEC
10 20 30 40 50
TIME (SEC) AT 1.250± 0.002 MeV

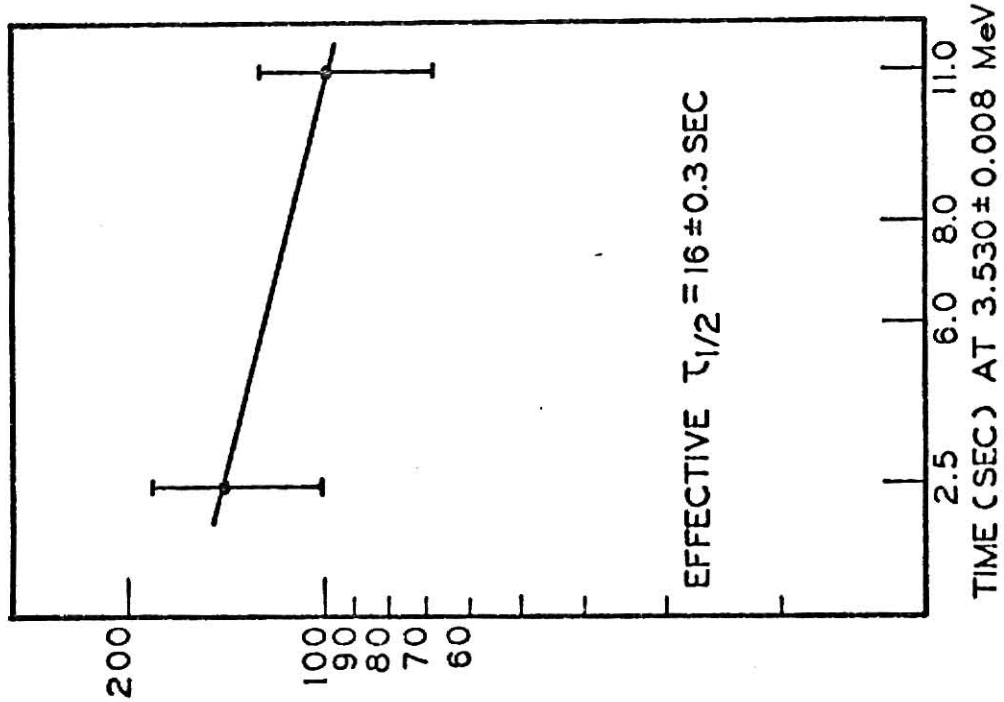

1000
100
10 20 30 40 50
TIME (SEC) AT 1.313±0.002 MeV


300
100
10 20 30 40 50
TIME (SEC) AT 1.559±0.002 MeV


900
800
700
600
500
400
300
200
100
EFFECTIVE $\tau_{1/2}$ = 10.6 ± 0.3 SEC
2.75
6.0
8.0
11.0
TIME (SEC) AT 2.907 ± 0.006 MeV
EFFECTIVE $\tau_{1/2}$ = 6.4 ± 0.2 SEC
TIME (SEC) AT 2.932 ± 0.006 MeV
EFFECTIVE $\tau_{1/2}$ = 17.0 ± 0.3 SEC
TIME (SEC) AT 2.949 ± 0.006 MeV
EFFECTIVE $\tau_{1/2}$ = 13.0 ± 0.3 SEC
TIME (SEC) AT 2.968 ± 0.006 MeV


200
100
90
80
70
60
50
BUILDUP, NO $\tau_{1/2}$ REPORTED
2.75
6.0
8.0
11.0
TIME (SEC) AT 3.137 ±0.007 MeV
200
100
EFFECTIVE $\tau_{1/2}$=10.4±0.2 SEC
2.75
6.0
8.0
11.0
TIME (SEC) AT 3.304 ±0.007 MeV
200
100
EFFECTIVE $\tau_{1/2}$=35±1.5 SEC
2.75
6.0
8.0
11.0
TIME (SEC) AT 3.334 ±0.007 MeV
200
100
EFFECTIVE $\tau_{1/2}$=22.2±0.5 SEC
2.75
6.0
8.0
11.0
TIME (SEC) AT 3.515 ±0.008 MeV


80
70
60
50
40
30
20
EFFECTIVE $\tau_{1/2}$ = 11.5 ± 0.3 SEC
2.5
6.0
8.0
11.0
TIME (SEC) AT 4.273 ± 0.010 MeV


200
100
90
80
70
60
EFFECTIVE $\tau_{1/2}$ = 16 ± 0.3 SEC
2.5
6.0
8.0
11.0
TIME (SEC) AT 3.530 ± 0.008 MeV

SHORT-LIVED FISSION-PRODUCT GAMMA-RAY SPECTRA FROM THE THERMAL FISSION OF U-235

by

WILLIAM EUGENE KENNEDY, JR.

B.S., Kansas State University, 1973

AN ABSTRACT OF A MASTER'S THESIS

submitted in partial fulfillment of the

requirements for the degree

MASTER OF SCIENCE

Department of Nuclear Engineering

Kansas State University
Manhattan, Kansas

1975

Abstract

Time-dependent, fission-product, gamma-ray spectra, useful in the active nondestructive assay of nuclear materials, were collected following the thermal-neutron induced fission of ^{235}U. Fission-product gamma rays were investigated by cyclic activation analysis using interrogating neutrons from both a small ^{252}Cf source and the thermal column of a TRIGA Mk II reactor. Fission-product gamma rays with half-lives shorter than one minute were investigated using several data acquisition trials consisting of up to 600 cycles per trial. Each data acquisition trial had a different activation and cooling time. In each cycle spectra were accumulated in as many as six time intervals after activation. The entire data collection process, including control of the pneumatic transfer system and reduction of the spectral information, was controlled by a NOVA minicomputer. The major peaks identified were compared with previous cyclic measurements and known fission-product gamma-ray energies. High energy gamma-ray peaks (above 2 MeV) which had not previously been analyzed by this method are reported. Peaks which appear to be particularly useful for active nuclear material assay are tabulated.