Fission-Product Energy Release for Times Following Thermal-Neutron Fission of ^{235}U Between 2 and 14000 Seconds

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FISSION-PRODUCT ENERGY RELEASE FOR TIMES FOLLOWING
THERMAL-NEUTRON FISSION OF ^{235}U BETWEEN 2 AND 14000 SECONDS

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ABSTRACT

Fission-product decay energy-release rates have been measured for thermal-neutron fission of ^{235}U . Samples of mass 1 to 10 μgm were irradiated for 1 to 100 sec using the fast pneumatic-tube facility at the Oak Ridge Research Reactor. The resulting beta- and gamma-ray emissions were counted for times-after-fission between 2 and 14,000 secs. The data were obtained for beta and gamma rays separately as spectral distributions, $N(E_\gamma)$ vs E_γ and $N(E_\beta)$ vs E_β . For the gamma-ray data the spectra were obtained using a NaI detector, while for the beta-ray data the spectra were obtained using an NE-110 detector with an anticoincidence mantle. The raw data were unfolded to provide spectral distributions of modest resolution. These were integrated over E_γ and E_β to provide total yield and energy integrals as a function of time after fission. Results are low compared to the present 1973 ANS Decay-heat standard.

A complete description of the experimental apparatus and data-reduction techniques is presented. The final integral data are given in tabular and graphical form and are compared with published data.

1. INTRODUCTION

In the event of a hypothetical loss-of-coolant-accident (LOCA) in a light-water reactor the fissioning process would cease, and a few seconds later the major source of heat in the fuel rod would be due to the beta and gamma rays from the decay of the accumulated fission products. In 1971 the American Nuclear Society Standards Committee No. 5 presented a standard curve¹ (revised in 1973) for the fission-product decay heat due to thermal-neutron fission of ^{235}U . This curve is shown in Fig. 1. The assigned uncertainty is also shown in this figure; the dashed line labelled "ANS + 20%" is the curve presently used in safety analyses to determine maximum fuel rod temperature. In 1973 an evaluation of then existing data was reported by Perry, Maienschein and Vondy;² they obtained an evaluated energy-release rate for infinite thermal-neutron irradiation of ^{235}U which varied between 0.98 and 1.08 times the 1973 ANS standard. They concluded that discrepancies among (and uncertainties assigned to) various data sets were such that a conservative (10) uncertainty assigned to a value of the rate of energy release rate after a time t following shutdown would be about 15%. It seemed to us that the overall uncertainty could be reduced experimentally by at least a factor of three. So, in 1974 we initiated measurements designed to obtain a short-time decay-heat curve for thermal fission of ^{235}U ; values for fission-product decay power derived from our final results are shown in Fig. 1 for the time interval 2.2 to 14,000 sec following fission. Our overall one-standard-deviation uncertainty is $\sim 4\%$ and so the comparisons shown in Fig. 1 indicate that "ANS + 20%" is a conservative curve for after-shutdown safety analysis.

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Fig. 1. Comparison of Fission-Product Decay Power as derived from the present experimental results with the Current (1973) ANS Standard for Fission-Product Decay Heat due to Thermal-Neutron Fission of ^{235}U (see Ref. 1).

The decay heat energy-release rate from thermal-neutron fission of ^{235}U can be obtained by two complementary methods. The first is based on measurements of the total beta and gamma power released; the second consists of cumulating by calculation the individual energies released by each one of the fission products. The second method, a large calculational problem, has the merit that once it is proven to reproduce measurements satisfactorily, it could be applied to more complex situations in a reactor, taking into account for example reactor power variations or the evolution of fissionable isotopes as a function of reactor operation. There are, at present, at least five active calculational efforts.³⁻⁸ A data base is required for all fission products which includes rates-of-production and energies and branching ratios of the decay products. Compilations of such data exist;⁹⁻¹³ however, especially for short-lived fission products, the data are incomplete or include substantial components from theoretical estimation.

Since 1973 four new measurements of fission-product decay power were initiated in the United States: (1) a calorimetric measurement at Los Alamos Scientific Laboratory,¹⁴ (2) a total-absorption measurement at Intelcom Radiation Technology, San Diego,¹⁵ (3) a second calorimetric measurement at the University of California, Berkeley,¹⁶ and (4) the present measurements at ORNL, consisting of separate differential energy measurements of beta and gamma rays for specified counting time intervals following specified irradiation-time intervals. The first three experiments mentioned provide integral energy-release data; the present experiment provides, in addition, spectral distributions for computing e.g. high-energy gamma-ray leakage. Furthermore, the spectral distributions provide a more stringent test of the validity of the "microscopic" calculational approach.

As mentioned above the present experimental results indicate that the present "ANS + 20%" curve is conservative compared to the fission product decay power and uncertainties for an infinite irradiation derived from our measured data and assigned uncertainties. The primary purpose of this report is to present the experiment in detail and to tabulate the final results in the format natural for our experiment. Also included is the analysis of our data to put them into the format displayed in Fig. 1.

2. EXPERIMENTAL METHOD, PREVIEW

The purpose of this experiment is to measure the total beta and gamma energy release rate from fission product decay following thermal neutron fission of the fuel element ^{235}U . The total energy release rate was obtained by measuring separately the energy release rates for each component. That is, one set of data was obtained for gamma energy release using a gamma-ray detector, and another set was obtained for beta decay energy release using a beta-ray detector. The Oak Ridge Research Reactor (ORR) was used to provide the thermal neutrons. Small samples of ^{235}U were irradiated for short periods of time and were rapidly recovered following irradiation. Data were obtained by detecting and measuring individual events for several irradiation times (t_{irrad}), waiting times following the end of the irradiation (t_{wait}), counting times starting at the end of the waiting time (t_{count}), and particle energies (E_{β} or E_{γ}). The resulting energy spectra were integrated over particle energy to obtain the energy release for each component for every combination of t_{irrad} , t_{wait} , and t_{count} , and the data are reported in this form in Section 7.

Measurements were originally proposed¹⁷ to study the time interval t from 2 to 2000 sec after a pulse of fissions, where $t \approx t_{\text{wait}} + 0.5 \times (t_{\text{irrad}} + t_{\text{count}})$. It was anticipated that the upper limit of 2000 sec would be sufficiently long that the data would mesh with calculations using compilations of radiochemical data. Preliminary gamma-ray measurements provided a surprise; the data agreed with calculations for t between 50 and 800 sec, but diverged from calculations for t between 800 and 10000 sec. So, the upper time limit was extended to 14,000 sec.

The experimental method calls for short irradiations (compared to decay time prior to beginning the observation), nominally 1, 10, and 100 sec; small samples, $\sim 1, 5$, and 10 μgm of U_3O_8 enriched to 93.5% in the isotope ^{235}U ; and spectral measurements for specified, and short, counting intervals. Except for the shortest time intervals, the experiment closely approximates differential energy release rates for time t after a pulse of fissions.

There were several important reasons for choosing the experimental method presently described to obtain decay-heat data. One of these was the fact that several major, and expensive, capital items - the fast pneumatic rabbit transport and the Digital Equipment Corporation PDP-15 computer - were already in existence and could be scheduled for use on this program. Also, techniques and computer routines to "unfold" experimental data had been developed and completely tested (requiring ~ 5 years) and had been used routinely for several years for neutron and gamma-ray data reduction. In addition, personnel were available having wide experience at nuclear spectroscopy measurements using scintillation and Ge(Li) detectors, in developing and using computers for data acquisition, and in radiochemical sample preparation and assay.

There was one important question during the planning stage that required an answer, and that was the relationship between data obtained for short irradiation periods and the ANS standard for an "infinite" irradiation. This relationship is understood by considering a simpler situation, viz. production of a single radionuclide by activation, e.g. ^{20}F by $^{19}\text{F}(n,\gamma)$. The amount of the produced nuclide in an irradiation period, t_i , is

$$N = \frac{R}{\lambda}(1 - e^{-\lambda t_i}) \quad (2.1)$$

where $\lambda = \ln 2/t_{1/2}$ is the decay constant, and R is the rate of production, assumed to be time independent. After cooling the sample for t_w sec (w for waiting), and then counting the emissions from the produced isotope for a time t_c , the yield is given by

$$Y(t_i, t_w, t_c) = \frac{R}{\lambda}(1 - e^{-\lambda t_i}) e^{-\lambda t_w} (1 - e^{-\lambda t_c}) \quad (2.2)$$

(We make use of this equation for ^{20}F in section 3.A)

Observe the symmetry in Eq. (2.2) for t_i and t_c ; the same yield will be observed if one exchanges irradiation and counting times:

$$Y(t_c, t_w, t_i) = Y(t_i, t_w, t_c) \quad (2.3)$$

This relationship is valid for every fission-produced nuclide. One can prove that this relationship is valid for every daughter in the chain neglecting interactions other than decay. Thus, if $E(t_i, t_w, t_c)$ is the integral energy release measured for a given irradiation time, waiting time, and counting time, the same reciprocity relationship, Eq. (2.3), is valid. This reciprocity is shown schematically in Fig. 2. The upper

TWO EQUIVALENT SITUATIONS

'A' IS LIKE REACTOR

'B' IS LIKE EXPERIMENT

Fig. 2. Two Equivalent Situations. The yield and energy-release obtained in a short counting period following a given waiting time after a long irradiation are the same as those obtained for the situation where the irradiation and counting times are reversed.

part of this figure represents reactor operation; an "infinite" irradiation corresponds to extending the X's to $-\infty$. The lower part represents our experimental system, and equivalence is obtained by extending the counting interval dashes to $+\infty$. However, this last step is not necessary since for some upper limit t_w' , $E(t_i, t_w', t_c + \cdot)$ is calculated by summation methods accurately enough. We need extend our counting interval to this upper limit, and then use calculations to provide the unmeasured energy release which will have occurred after our chosen counting interval. For this experiment the counting time interval extended almost to 4 hr, providing > 80% of the total equivalent energy release for an "infinite" irradiation $E(t_i \rightarrow \infty, t_w < 10 \text{ sec}, t_c = 1 \text{ sec})$ for short waiting periods.

The importance of the approximations in the equivalence

$$E(t_i, t_w, t_c) = E(t_c, t_w, t_i) \quad (2.4)$$

was investigated. The first assumption, that R is a constant independent of time implies (a) a constant flux of neutrons, and (b) no depletion of the fuel. The second approximation, "...neglecting interactions other than decay," means e.g. neglecting capture by fission products. For our thermal neutron flux of $\approx 3 \times 10^{17} \text{ n/m}^2/\text{sec}$, the "effective half life" for depletion of ^{235}U is ~ 1 year. One may verify mathematically that a short irradiation followed by a long count will result in a larger yield than a long irradiation followed by a short count; there will be less correction for sample depletion. The large ^{135}Xe capture cross section will also affect the results since now $\lambda_{\text{decay}} \neq \lambda_{\text{irradiation}}$ because the latter includes the effective capture

$$\lambda_{\text{capture}} = \sigma_{\text{capture}} N_n \sim 1 \times 10^{-4} \text{ sec}^{-1} \quad (2.5)$$

compared to $\lambda_{\text{decay}} \sim 2 \times 10^{-5}$. That is, for our thermal flux, N_n , capture to stable ^{136}Xe during the irradiation is ~ 5 times more probable than decay. The independent yield for ^{135}Xe is 0.11% compared to a cumulative yield of 6.7% according to the Meek and Rider compilation.¹⁸ For this effect there is essentially no correction for a short irradiation followed by a long count; none was applied to our data. All other fission products have thermal neutron capture cross sections $\ll$ that for ^{135}Xe (as far as is known at this writing), hence extended irradiations are required to study contributions from neutron capture. Calculations¹⁹ show that for irradiations of ~ 2.5 yrs and for normal the. reactor conditions, the correction to the fission-product decay power is $\sim 1\%$ for $t_w < 10^3$ secs.

To bring this experiment to fruition required state-of-the art developments, as well as using available equipment at maximum capability. The next sections present discussion basic to the understanding of these developments, with the recognition of the difficulty in presenting a complete report on them. For example, electronic units were developed to interface the PDP-15 computer to the CAMAC system so that the computer could "talk" to the sample transport control, to the display, and to the data-acquisition electronics. We do not provide schematic drawings nor a detailed report on this development in this report. Nor do we provide a detailed discussion of the code written for the PDP-15 for data acquisition, leaving this task to a separate report.²⁰ We do, however, include listings of most of the FORTRAN-IV computer routines used for data reduction. The one important exception is the unfolding routine, FERD, which is being documented.²¹

There were a number of "preliminary" data-taking periods designed to prove the working of equipment or else to uncover important problem areas. Some data which were obtained during those periods were made available; the report²² on those data stressed:

"These interim results are reported at this time primarily to provide a lead time for preparation by other related programs which will use the final results. It should be stressed that the present (interim) results will be superseded in the near future, and their use should be confined to unpublished work."

These early periods are referred to in this text as preliminary and served their primary purpose.

During the course of the final two data-taking periods (about two weeks for beta-ray data, about 10 days for gamma-ray data) ~ 100 fissile samples were run in addition to about an equal number of nonfissile samples. Between 15000 and 17000 channels of data were acquired for each fissile sample. Many checks and redundancies were designed as part of the experiment. These included, for example, a method to measure any gain change of photomultiplier tube output regardless of the count rate incident on the detector. Checks were performed at regular intervals during a data-taking period to ensure that pulse outputs from the various electronic units were correct. Equipment malfunction was usually spotted prior to the initiation of the next irradiation.

Although checks on certain integral quantities were made at the end of each irradiation and counting intervals, automatic handling of many irradiation details was mandatory and considered reliable. Some portions of the data handling could be and were checked manually, but for the most

part the automatic systems were checked by obtaining and reducing data for irradiation of nonfissile samples, which checks gave results in agreement with prior measurements by other techniques.

In Section 7 we compare our results with other published data,²³⁻²⁸ data which for the most part have been available for several years and have been the bases for the evaluation and theoretical study. We do not compare our data with results of the three most recent experiments¹⁴⁻¹⁶ in this report since these are in the same stage of evaluation as our own. In our opinion the most important use of our data will be to guide the calculational efforts to a point where they may be relied upon to compute correctly the decay heat for any history of operation.

3. EXPERIMENTAL DETAILS, HARDWARE

An overview of the experimental system is shown schematically in Fig. 3. At the Sample Preparation Area in Hood No. 1 the samples are prepared by dissolving uranium oxide (U_3O_8) in nitric acid and depositing a known volume of the solution onto a sample holder and then evaporating the liquid under a heat lamp. These were then capped and sealed and marked for later identification. The samples were nominally 1, 5, or 10 μgm of U enriched to 93.5% in the ^{235}U isotope. For convenience in spotting experimental difficulties, the samples were prepared to have their nominal weights to within $\pm 1\%$.

A data-accumulation run consisted of the following steps; (a) a sample was mounted in a sample carrier (rabbit) and placed in the sample loader in Hood No. 1; (b) the parameters of the experiment, e.g. nominal irradiation time, nominal counting time intervals (up to 17), and other

Fig. 3. Schematic Representation of Experimental Arrangement for Fission-product Decay-heat Measurements. Samples of ^{235}U are placed in a rabbit and put in the Sample Sender. The Central Processor controls the movement of the rabbit to and from the Irradiation Position, then to the Sample Holder. After a specified cooling time, either photons or beta rays are counted, and the data are stored in the computer. At the completion of data accumulation the data are stored on magnetic tape for off-line reduction.

pertinent information, were entered into the PDP-15 Data Accumulation Computer using the teletype; and (c) the "Start" command was entered using the teletype. From this point on all rabbit movements and data accumulation were controlled by the computer. These steps consisted of: (a) the necessary computer memory locations needed to store the data were cleared, and a signal from the computer actuated the pneumatic system sending the rabbit to the Irradiation Position; (b) at the expiration of the irradiation time, the computer signaled the pneumatic system sending the rabbit to the Sample Holder in Hood No. 2; this signal also started the "waiting" time measurement in the computer; (c) when the rabbit reached the Sample Holder, an electronic-eye system signaled the computer; (d) data accumulation commenced at the expiration of the first input waiting time, unless the signal from the last step was not received, in which case the run was aborted; (e) at the beginning of each succeeding input counting time interval, the computer memory location corresponding to the first channel was incremented by twice the number of channels per spectrum; (f) at the end of the last input counting-time interval data accumulation ceased, and the computer awaited further commands. Normally, the next command was to dump the data on magnetic tape for off-line processing. Finally, the rabbit was removed from the Sample Holder in Hood No. 2.

The experiment calls for measurements of both beta and gamma radiation, and Fig. 3 shows the positions of both detectors. However, only one can be used at a time. A different Sample Holder is needed for each detector; in addition, although the same set of electronics was used, the analog settings are different for each detector.

For the gamma-ray data, background was measured by running a blank sample. Some of the background was time dependent, corresponding to activity (e.g. ^{24}Na) picked up in the movement of the sample to and from the reactor. Some of the background was time independent, consisting primarily of ^{41}Ar decay plus a low-energy contribution.

For the beta-ray detector, two samples had to be run, one with no magnetic deflection, measuring $(\beta + \gamma)$ (although only a portion of the gamma rays are detected). The second sample was measured with magnetic deflection, yielding (γ) only. Background from activity on the rabbit was nearly eliminated by tight collimation. From these data the pre-FERD data handling (discussed in Section 6.A) included subtraction of "magnet-up" data from "magnet-down" data. This operation is equivalent to

$$(\beta) = (\beta + \gamma) - (\gamma) . \quad (3.1)$$

Following the irradiation and measurements the samples were cooled for varying periods and then counted for characteristic gamma rays associated with decay of ^{97}Zr , ^{99}Mo , and ^{132}Te . These measurements were performed in a separate low-background counting area.

Prior to, and several times during each data-taking period, the thermal-neutron flux and the ratio of thermal-to-resonance were measured using well-known foils of $\sim 20 \mu\text{gm Au}$ and $\sim 20 \mu\text{gm Mn}$ sandwiched together. The thermal flux varied between 2 and $3 \times 10^{17} \text{ n/m}^2/\text{sec}$ and the ratio of thermal to resonance varied between 36 and 40. Although this means that the number of fissions in the ^{235}U sample due to epithermal neutrons is less than 1.5% of the total number of fissions, we were concerned in the planning stages about the possibility of a different decay-heat power

from epithermal fission of ^{235}U compared to decay-heat power from thermal fission -- sufficiently different to introduce a systematic error in our results. Tests were performed to see if gamma-ray spectra could be obtained for samples surrounded by Cd. A simple technique was not found because we needed a larger amount of Cd than our sample containers could hold. Before considering revision of the sample containers, calculations were performed using ORIGEN⁶ for the decay power for times following a pulse of fissions for both thermal- and fast-neutron fission of ^{235}U . These calculations indicated differences between the values for thermal-neutron fission and those for fast-neutron fission of $\sim 1\%$ for times between 2 and 10^4 sec following fission.

Since the calculations suggested we were not likely to observe any difference, this experimental task was relegated to a low priority and has not been completed.

In the following sections each major piece of equipment is discussed in detail. It is important to get a full description of the method in order to assess the validity of assigned uncertainties.

3.A. Pneumatic Tube System at the ORR

The pneumatic tube system is a device for transporting small plastic capsules (rabbits) from the counting room to a position near the face of the reactor lattice and, after a predetermined irradiation period, rapidly recovering them for analysis. The rabbits are transported by air pressure from a hooded loading station in the basement of the ORR building to the lattice and, after irradiation, are quickly returned to a hooded station. Transit time for the rabbits was reported as ≈ 1.5 sec.²⁹ The system is automatic in operation, but can be manually overridden. In automatic mode, a preselected irradiation time is set on a timer. A rabbit is inserted in the loading chamber and the system is energized. The rabbit is blown down the tube line to the lattice face, a distance of ~ 26 m. At the end of the selected time the rabbit is returned either to the loading station or to another station.

The present experiment required two modifications. The first was to return the rabbit to Hood No. 2 and to design a sample-counting station in Hood No. 2, and the second was to interface the pneumatic tube controller system to the computer. A switching mechanism was installed in main portion of the pneumatic tube which shunts the returning rabbit to Hood No. 2. A relay box was designed and built to provide commands to the pneumatic tube equipment which are in parallel with the existing control system. The relays in the relay box are actuated by computer commands. Status signals (e.g. "Is the loading station in Hood No. 1 closed?") are made available to the computer program.

Design of the sample counting stations was more of a challenge. The requirements were (a) to stop the returning rabbit quickly (~ 0.1 sec) and

know its position, and (b) to provide the least amount of scattering material. Following analysis of preliminary experiments a third requirement was added for the beta-ray measurements, namely to position the rabbit just as close to the beta-ray detector system as possible to reduce scattering and energy loss in air. There was sufficient orthogonality in the gamma-ray and beta-ray measurements to require designing a separate counting station for each type of measurement.

The gamma-ray measurements utilize a simple sample holder (shown schematically in Fig. 4) made out of low-density polyethylene block with a hole through the center large enough to allow the rabbit to move horizontally from the back to the front, where it comes to a stop against a polyethylene door. At this point the sample is 1.000 ± 0.005 m from the front face of the NaI detector. The arrival of the sample is detected by an electronic eye system, which transmits the information to the computer. About 0.25 sec prior to the first counting-time interval, the computer sends a signal to open the door pneumatically; the door swings clear of the direct line between sample and detector prior to initialization of counting, coming to rest at a position below the direct line-of-sight of the gamma-ray detector, as shown in Fig. 4.

The beta-ray measurements require a somewhat more complicated sample holder because of the need to minimize the amount of air between the sample and detector, as well as the need to ensure precise positioning of the sample vis-a-vis the collimators between the sample and detector. The configuration adopted is shown schematically in Fig. 5. Shown in this figure is the front view. The sample enters from the rear into the sample holder being stopped by the horizontal metal strip. After the

MATERIAL **ALUMINUM** **POLYETHYLENE****ALL FITTINGS ARE BRASS****GAMMA-RAY RABBIT COUNTING STATION**

Fig. 4. Gamma-ray Counting Station in Hood No. 2. The door is closed prior to initiating the experiment. The rabbit returns from the reactor (from the right side of the drawing), and is detected by the light emitting diode as it stops against the closed door. The door is opened to the position shown prior to gamma-ray measurements.

Fig. 5. Beta-ray Counting Station in Hood No. 2. The metal strip is in the closed position with the hole off center as shown. After the presence of the rabbit has been sensed by the Rabbit Position Sensor, the Pneumatic Cylinder is actuated causing the metal strip to move to the right, centering the hole in the strip on the rabbit.

pneumatic tube overpressure has bled off, the metal door strip is slid horizontally to the right, causing the hole (shown in the figure just below the Rabbit Position Sensor) to center on the sample. Then a small puff of air is introduced behind the rabbit causing it to move forward, i.e. toward the detector, through the hole in the metal door strip.

Another view of the beta-ray sample holder is shown in Fig. 6. The horizontal tube in the center of the photograph is part of the beta-ray detector and will be discussed in Section 3.C. Just to the left of the left end of this tube is a white block; this is the sample holder which is made out of polyethylene. The rabbit enters from the left, is stopped by the metal strip, and its presence is detected by the electronic eye system (wires entering near the top and bottom are part of this system). Prior to the beta-ray measurement the sample holder is positioned next to the end of the evacuated tube, as shown in Fig. 7. In this position, when the rabbit is moved to the right by the puff of air, it moves about 10 mm, and is centered on the end cap of the evacuated tube about 8 mm from the entrance foil. Commands for opening the metal door and subsequently moving the rabbit are sent by the computer during an actual run.

One important problem had to be resolved, and that was to relate the actual irradiation period to that entered into the computer via the teletype. Prior to our use of the pneumatic tube the nominal irradiation time commenced upon a signal from a pressure-sensing switch which actuated on a change in the pressure in the tube caused by the stopping of the sample at the irradiation position. The addition of the pneumatic tube to Hood No. 2 reduced the sensitivity of this switch, causing occasional incorrect signals from the pressure sensitive switch. Therefore, for the computer-controlled rabbit movements, "irradiation" time was measured from the

Fig. 6. Beta-ray Counting Station and Detector System in Hood No. 2. The counting station (white polyethylene block in left center of the photograph) has been moved away from the end of the evacuated tube. At the other end of the evacuated tube, behind the lead shielding, is the beta-ray detector. The tube base is observed on the other side of the lead shielding (right center of photograph). Vacuum in the tube is maintained at ~ 600 Pa ($\sim .006$ atm). The track under the rubber tubing (which goes to the vacuum pump) supports a magnet.

Fig. 7. Close-up of Beta-ray Counting Station. The station is in place against the end of the evacuated tube. This photograph also shows the magnet in place.

time the signal was sent to initiate rabbit movement from the loading station. Clearly the actual irradiation time is shorter, and the problem was to determine the relationship between nominal (typed in) and actual irradiation times. For this purpose a ~ 2 mg sample of CF_2 was used, utilizing the 11-sec decay of ^{20}F . The total number of ^{20}F decays observed for a given irradiation time t_i , waiting time t_w , and counting time t_c is

$$Y(t_i, t_w, t_c) = \frac{R}{\lambda} (1 - e^{-\lambda t_i}) e^{-\lambda t_w} (1 - e^{-\lambda t_c}) \quad (2.2 \text{ repeat})$$

where λ is the decay constant for ^{20}F and R is the instantaneous rate of production of ^{20}F in the reaction $^{19}\text{F}(n, \gamma)^{20}\text{F}$. The measurement consisted of five runs for constant $t_w = 3$ sec and $t_c = 30$ sec but varying the nominal t_i between 2 and 20 sec. A 0.5 sec variation for $t_i = 20$ sec results in $\sim 1\%$ change in $Y(20, 3, 30)$, but a very large change for $t_i = 2$ sec. The time variation was readily determined from these measurements. These measurements were made prior to every session of data taking; the measured $\Delta t (= T_{\text{typed-in}} - T_{\text{actual}})$ varied between + 0.6 and + 0.7 sec.

This was a rather unexpected result, since we anticipated Δt to be $\sim$ the time for the rabbit to travel to the irradiation position, i.e. ~ 1.5 sec. As we used the system the nominal (typed in) irradiation commenced with the signal to send the rabbit from the loading station in Hood No. 1, and terminated with the signal to return the rabbit from the irradiation position. The transit time of the rabbit was then carefully measured, assuming it to be the time after the signal was transmitted by the computer to send the rabbit from the loader until the signal from the pressure-sensing switch was received by the computer. This measurement gave 1.3 ± 0.05 sec as the transit time. It must be, therefore, that

irradiation begins prior to receipt of the signal from the pressure-sensing switch, and continues after the signal is sent to return the rabbit. It was assumed that the difference between Δt measured by ^{20}F decay and the measured transit time to the irradiation position was divided evenly before the pressure-sensing switch signal and after the "return" signal was sent to the pneumatic tube control unit. Experiments were considered and tried which were designed to ascertain any asymmetry in the division of the "missing" ~ 0.6 sec; none were successful in proving or disproving the assumed division of time. The most probable division of time is an equal division; a division of 0.4 sec before - 0.2 sec after or of 0.2 sec before - 0.4 sec after is also reasonable, but any further asymmetric division appears unlikely to us. Consequently, in the final data reduction (discussed in Section 7), all of the nominal cooling times were reduced by 0.3 ± 0.1 sec.

3.B. Sample Containers and Carrier (Rabbit)

Two styles of sample containers were used to hold the ^{235}U or other (nonfissile) samples. These are illustrated in Fig. 8. The style no. 1 utilized existing materials, namely a small polyethylene cup and a tight-fitting lid. An aliquot of the ^{235}U -loaded solution was placed on the bottom of the inside of the cup and dried with a heat lamp. Then the lid covered the cup. Some of the sample containers were then heat-sealed using a specially-made soldering gun tip. For some of the sample containers an epoxy glue was placed on the rim of the cup before the lid was snapped on. About half of the containers were not either heat-sealed or glue-sealed. During the preliminary experiments measurements of the accumulated data (including scaler readings) were studied to ascertain if any differences could be ascribed to heat-, glue-, or no-sealed containers, and there was no correlation observed which favored any of the three types of containers.

The preliminary and final gamma-ray energy release data were obtained with samples in the style no. 1 containers, as were the preliminary beta-ray data. For the latter, the 500 g/m^2 thickness corresponds to the approximate range of a 230-keV beta particle.³⁰ Higher energy beta rays suffered energy losses or were scattered in traversing this thickness. Measured response to monoenergetic conversion-electron sources were corrected for these effects. However, as discussed later in Section 6.C, extrapolation of the response beyond the measured region was required, and for this thickness the reliability of the extrapolation was modest at best. So after completing the preliminary beta-ray energy-release measurements, a new design for a sample container was initiated. Several designs were tried, resulting in the final design illustrated in Fig. 8 labelled style no. 2.

NOT TO SCALE

ALL CONTAINER MATERIAL IS
POLYETHYLENE
GLUE IS EPOXY
SAMPLE IS DRIED URANIUM OXIDE

Sample Containers for Decay-Heat Measurements.

Fig. 8. Two Styles of Sample Containers. Style no. 1 is used for gamma-ray measurements, and style no. 2 is used for beta-ray measurements.

This style utilized an existing high-walled polyethylene cup, very similar to the cup used in style no. 1, but with sides extending the full length of the inside of a sample-transport container (discussed in the next section). In which would normally be the bottom of the cup, a small circular depression was cut 5 mm diameter by 0.25 mm deep, leaving a rim of ~ 2 mm. This rim was etched using a solution of $K_2Cr_2O_7$ and H_2SO_4 heated to $70^\circ C$; then into this depression the sample was deposited in liquid solution. The liquid was evaporated, leaving the uranium oxide sample inside the depression, but somewhat more concentrated along the outer circumference of the depression. Then a $\sim 50 \text{ g/m}^2$ polyethylene foil was glued to the outer etched surface of the disk using Biggs R-313 Epoxy. This glue required one day to cure. When the glue was dried the sample was inspected under a magnifying glass to ensure that a seal had been formed completely around the glued circumference and to be sure that none of the glue covered any portion of the sample.

The thickness of polyethylene behind the sample was $\sim 0.25 \text{ kg/m}^2$. There is some backscattering from this backing, the magnitude of which can be estimated from an empirical relation given by Tabata³¹ for the backscattering coefficient from a semi-infinite scatterer and then adjusting the result for the 0.25 kg/m^2 backing thickness from Koral and Cohen.³² These estimates indicate that the backscattering yield from the backing will be $< 0.2\%$ for $E_\beta > 1 \text{ MeV}$, $< 1.0\%$ for $E_\beta > 0.4 \text{ MeV}$, $< 3\%$ for $E_\beta < 0.2 \text{ MeV}$. These are upper limits of the number of back-scattered beta's at a given E_β since the angular distributions will reduce the backscatter yield in the solid angle subtended by the detector.

All of the final beta-ray data were obtained with samples in the style no. 2 container.

The sample in the sample container was transported to and from the irradiation position in a capsule referred to as a "rabbit." Standard rabbits designed for transporting samples contained in the style no. 1 sample containers were modified for our measurements. As fabricated for the major portion of the work performed at the Fast Pneumatic Tube Laboratory, a rabbit completely encloses a sample container. For our purpose the end of the rabbit nearest the sample has too much mass, and so a hole was machined through the end of the standard rabbit. The resulting configuration is illustrated in Fig. 9. The illustration shows the mounting of the style no. 1 container; the orientation in this figure is such that the fission-product gamma radiation would emanate toward the right edge. The length of the hollow sleeve is such that when the rabbit screw cap is tight, the sample container is securely clamped between the sleeve and the lip at the end of the rabbit. The length of the style no. 2 container is such that when the screw cap is tight, the sample container (particularly the glued-on cover) is securely clamped between the container and the lip. For both styles of container the diameter of the sample is smaller than the diameter of the hole made in the end of the rabbit.

Samples in the style no. 2 container irradiated for 10 sec or less ($n_{th} \approx 3 \times 10^{17}/m^2/sec$) did not exhibit evidence of deterioration, and were transported in the open-ended rabbits. Samples irradiated for 100 sec exhibited a bulging of the $50 \text{ g}/m^2$ covering due to thermal heating; however, the glue remained fast, and there was no indication of loss of sample material. Because of this bulging, however, samples subjected to long irradiations were transported in rabbits with solid ends. Upon return from the irradiation position, the container was removed from the

ALL MATERIAL HIGH-DENSITY POLYETHYLENE
NOT TO SCALE

Sample Transport System for ORR Fast Pneumatic Tube.

Fig. 9. Sample Transport System. These capsules, also known as rabbits, are modified from ORNL standard rabbits by drilling a hole in the end opposite the screw cap to reduce attenuations. This figure shows a style no. 1 sample container and spacer.

transporting rabbit and inserted into another rabbit with a hole in the end, which was then inserted into position at the end of the beta-ray detector (as described in Section 3.D). This was no burden to results, since the experimental method doesn't require counting samples immediately after the longer exposures, and so the sample container could be removed from the rabbit during the first waiting time interval.

3.C. Gamma-Ray Scintillation System

The gamma-ray detector is shown schematically in Fig. 10. This is a 127 mm diameter by 127 mm deep NaI crystal covered on the front face by 1.3 mm Al, mounted on a photomultiplier tube (RCA 4522) and positioned inside a massive lead shield. The inset shows the position of the alpha source used as a light pulser to monitor possible gain shifts. A permanent magnet, > 2200 Gauss between pole tips 64 mm apart, was placed just in front of the "44.5 mm DIAM COLLIMATOR" of Fig. 10.

This size detector was chosen over smaller and less efficient detectors for two reasons. The first was our current familiarity with the general characteristics of its response because of experience with this size in other experiments at ORNL.³³ The second reason was that the accuracy would be improved by the high percentage of all pulses from a monoenergetic source detected within the full-energy peak (often known as the "photo peak"). This characteristic was enhanced by the choice of collimation since a lower percentage of gamma rays first interacting on the outside edge of the detector will be detected within the full-energy peak than for those gamma rays first interacting near the center of the detector. The FERD unscrambling system produces a weighting vector for any particular output quantity desired, and the estimated magnitude of that quantity is then given by the inner product of the pulse-height spectrum with the computed weighting vector. To compute the emitted energy, the more closely the weighting vector approaches the asymptotic relation corresponding to constant efficiency and unique detector response (e.g. a pure Gaussian distribution) the less possibility for error exists.

Fig. 10. Gamma-ray Detector Arrangement. The detector is enclosed in a lead cave, 0.1 m thick on the top, sides, and bottom. The inset shows the position of the alpha source used as a light pulser to monitor possible gain shift. For beta-ray deflection a permanent magnet was positioned between the 44.5 mm diam collimator and the sample.

One drawback in the use of a large volume detector is sensitivity to room background, and the present counting room is situated only 10 to 20 m from an ion-exchange unit which is sufficiently active to warrant signs discouraging close human proximity. In addition, there are background gamma rays from neutron interactions with all structural material between the reactor and the detector. Shielding was required, and a lead cave was built on a sturdy table, as shown in Fig. 11. Also shown in the lower right-hand corner is the track on which the permanent magnet is placed. There is an iron plate on the front face of the lead collimator system which was used to divert magnetic lines of force away from the photocathode of the photomultiplier tube. The handles along the top of the lead bricks in the collimator system were installed to facilitate shifting the lead collimators during the initial phase of the experiment. (Part of the electronics required in the counting area is shown on top of the main lead shield.) Room background rates were reduced to $\sim 50 \text{ sec}^{-1}$ for $E_\gamma \geq 0.06 \text{ MeV}$. Much of the remaining room background was due to decay of ^{41}Ar made by neutron interactions with argon in the air. About one-third of this ^{41}Ar contribution ($E_\gamma = 1.3 \text{ MeV}$) was removed by blowing fresh air into the region surrounding the detector. Room background was monitored throughout the gamma-ray data-taking experimental run and very little variation was observed. The background contribution was negligible for nearly all gamma-ray data obtained, except in data for cooling times $> 4000 \text{ sec}$.

Since it was anticipated that use of absorbers to eliminate beta rays from the gamma-ray spectra would bias against low-energy gamma rays and possibly require larger corrections for bremsstrahlung, a magnetic field was employed to deflect the beta rays emanating from the sample.

Fig. 11. Gamma-ray Detector Shielding. Gamma rays emanating from the sample enter (from the right of the photograph) the first of two collimators shown at the end of the table. A magnet similar to that in Fig. 7 is placed on the track shown in the lower right corner. The NaI crystal is behind the front part of the main part of the lead shielding; if the detector were visible, it would be in the center of the photograph. A portion of the electronics is also shown.

Bremsstrahlung production in the region near the source was minimized by use of light construction and liners of low- z materials (refer to Fig. 5). Bremsstrahlung production in the area surrounding the source was estimated to be less than 1% of the total gamma-ray yield and confined to gamma-ray energies < 0.1 MeV. Bremsstrahlung production by beta rays deflected by the magnetic field into the collimation system was more difficult to estimate. Only high-energy beta rays ($E_\beta > 3$ MeV) have enough rigidity to pass through the first collimator, and of these only beta rays having an original direction into the weaker fringe field may be "inscattered" since even 10-MeV beta rays having an original direction toward the detector will be deflected sufficiently to strike the front face of the first collimator. A measurement was made to test the degree of suppression of source-emitted beta rays without additional bremsstrahlung. A useful activity for this purpose is ^{42}K , which has a 12-hr half-life and decays primarily by 3.5-MeV beta emission to the ^{42}Ca ground state, but has an 18% branching for a 2.0-MeV beta emission followed by a 1.52-MeV gamma ray. A weak 0.32-MeV gamma ray is also observed. The overall observed spectrum (i.e. without magnet) is shown in the upper half of Fig. 12; the beta-ray contribution is clearly observed for energies greater than 1.6 MeV (these beta rays were partially attenuated by the aluminum cover on the NaI crystal). Then the permanent magnet was placed about 0.1 m in front of the front collimator, and the spectrum shown in the lower half of Fig. 12 was obtained. The contribution to the spectrum for gamma-ray energies between 1.6 and 2.5 MeV has been substantially reduced ($> 97\%$). The magnetic field has eliminated all of the ground-state decay beta rays, as the remaining portion of the spectrum for $E_\gamma > 1.6$ MeV not only has the wrong shape to

ORNL-DWG 77-5457

Fig. 12. Spectra of the 1.52-MeV Gamma Ray from the Decay of ^{42}K . The top spectrum was obtained without magnetic deflection of emitted beta rays; the bottom spectrum with magnetic deflection.

be caused by unsuppressed beta rays but can be qualitatively identified with other gamma rays emanating from ^{42}K or known background. In addition, if there were bremsstrahlung associated with suppressed beta rays interacting with the magnet or first collimator, the low energy (< 0.3 MeV) portion of the spectrum should increase. Such increase is not evident in the lower half of Fig. 12.

Previous work²³ indicated that the beta-ray spectra observed for thermal-neutron fission of ^{235}U were softer than that for ^{42}K , since the average beta-ray energy due to ^{235}U fission-product decay is ~ 1 MeV, whereas the average beta-ray energy for ^{42}K decay is ~ 1.4 MeV. Thus, prior to measurements of energy release from ^{235}U , it was felt that the ^{42}K measurement was sufficient to verify the absence of beta-ray induced bremsstrahlung despite the fact that a good quantitative upper limit was not determined from these results. However, our final beta-ray energy-release measurements (presented later in Section 7) indicated an average beta-ray energy due to ^{235}U fission-product decay of ~ 2 MeV for waiting times < 10 sec. We then became concerned again about the possible contribution of bremsstrahlung to our gamma ray spectra. Another experiment was performed using a small sample (~ 2 mg) of Li_2CO_3 , making the 0.84-sec capture product ^8Li , which decays by beta decay only (no gamma rays), with a beta-ray end point of 13 MeV and an average beta-ray energy of ~ 6 MeV. For decay from this isotope (^8Li) a low-energy photon spectrum was observed, peaked at the lowest observed photon energy (0.05 MeV), and having an integral yield of $0.15 \times N_\beta$ and an average photon energy of 0.9 MeV. This spectrum was very similar in shape to one calculated for us by T. Nakamura³⁴ for 4.0 MeV electrons incident upon a 3 mm thick polyethylene slab. Other

calculations provided us by Nakamura^{34,35} indicate that the integrated bremsstrahlung yield increases approximately as E_β^2 and that the average photon energy increases approximately as E_β . Hence, for our shortest waiting times for which the average beta-ray energy is ~ 2 MeV, the estimated bremsstrahlung is $\sim (2 \text{ MeV}/6 \text{ MeV})^3$ less than the 13.5% energy release rate measurement for ${}^8\text{Li}$, that is, the contribution to the ${}^{235}\text{U}$ fission-product decay energy-release rate is $< 1\%$ of the measured gamma-ray energy release rate. This measurement, using ${}^8\text{Li}$, also verified that the magnetic deflection of high-energy beta rays was very satisfactory since the observed number of counts in the magnet-up spectrum for $E_Y > 3$ MeV was $< 0.05\%$ of the number of counts observed in the same pulse-height region with the magnet removed.

There is the possibility of significant phototube gain drift during experiments of this type in which the source strength seen by the phototube varies strongly. This drift can be nearly eliminated by phototube selection and minimization of anode current; proof of stability can be given by measuring tagged gamma rays or light pulses. We used an alpha source, ${}^{241}\text{Am}$ grown in NaI, shown in the inset of Fig. 10, to monitor possible gain shifts in the spectrometer. The alpha pulses in NaI are readily distinguished from gamma-ray pulses in NaI using the electronics system discussed in Section 3.E. For each gamma-ray spectrum obtained there was a separate alpha spectrum obtained at the rate of ~ 100 counts/sec. For the final gamma-ray data-taking run these alpha spectra were monitored routinely. No gain shifting or zero channel displacement sufficient to indicate a change in energy calibration by as much as 1% was observed.

3.D. Beta-Ray Scintillation System

As shown in Fig. 13, the beta-ray detector is made of two scintillators, a 34 mm diameter by 34 mm deep NE-110 plastic scintillator mounted inside a CaF_2 cup having overall dimensions of 44 mm diameter by 40 mm deep. This two-crystal system is designed to identify a portion of gamma radiation detected, since pulses due to interactions in the CaF_2 scintillator are readily distinguished from those due to events in the NE-110 by using pulse shape discrimination. An alpha source is positioned on the periphery of the CaF_2 as shown in Fig. 13, and it provides an alpha spectrum which is used to identify and quantify system gain shifts in a manner similar to that discussed in the last section for the gamma-ray detector.

After optical coupling the scintillators to each other and to the photomultiplier tube (RCA 8850), the detector is mounted at one end of a 0.5 m long evacuated tube. A thin window, 10 g/m² aluminized mylar, covers the entrance at the opposite end of the tube, and collimators made of 15 mm thick Al are spaced along the inside of the evacuated tube. The separate components prior to assembly are shown in Fig. 14. The system in position in Hood No. 2 is shown in Fig. 6 in Section 3.A. Additional features shown in this photograph not discussed above include (a) the lead shielding surrounding the detector; (b) other lead shielding behind the Sample Holder; and (c) the "track" below the left end of the evacuated tube. The lead shielding is required to reduce room background contributions. The "track" holds the permanent magnet (~ 2700 Gauss between pole tips 64 mm apart) which is installed when the experiment calls for measuring γ only. Without the magnet the system measures $\beta + \gamma$. (Figure 7 shows the magnet in place.) Figure 15 shows a close-up view of the end of the

Fig. 13. Beta-ray Scintillation Spectrometer. The beta-ray source position is about 8 mm from the entrance foil. For some measurements a strong magnetic field (~ 2700 gauss) is placed between the entrance foil and the first collimator perpendicular to the path between source and detector. The "Light Pulser, ^{241}Am " is used to monitor possible gain shifts during data taking.

Fig. 14. Exploded View of Beta-ray Spectrometer. The aluminum collimators and spacing rods are shown in the left and center foreground. To the right of the collimators is the detector. The CaF₂ cup encloses the NE-110, the face of which can be seen in the center of the left-hand side of the detector. A rabbit is shown to the right of the detector for size comparison. Above the rabbit is the photomultiplier tube and its housing. To the right of the tube is the tube base. To the left of the tube, above the collimators, is the pipe which is evacuated during measurements. There is a vacuum seal around the end of the photomultiplier tube where the screws are shown. Not shown is the end piece which goes on the left end of the pipe.

Fig. 15. Close-up of the End of the Evacuated Pipe. A mirror has been installed between the end piece of the evacuated pipe and the beta-ray counting station to show the details of the end piece. A thin polyethylene insert (not shown) is placed in the depression to cushion the impact of the rabbit on its final movement. The four grooves allow air escape ahead of the rabbit during this movement.

evacuated pipe. A mirror was positioned between the Sample Holder and the evacuated tube to get this photograph. In the center of the mirror image is the aluminized mylar entrance foil; then shown is the concentric depression which allows the centering of the rabbit after it passes through the hole in the door strip. A thin polyethylene lining is inserted in this depression to cushion the impact of the rabbit. The four grooves cut almost radially from the center are required to allow the escape of air ahead of the rabbit on its final movement.

Not shown clearly in any of these figures are the pieces of annealed steel used in addition to standard mu-metal shielding to absorb the magnetic flux from the strong magnet when it is in position. One piece is cylindrical and surrounds the photomultiplier tube and crystals, and two other pieces are flat and are mounted perpendicular to the axis of the evacuated tube on the upright to the left of the lead shielding surrounding the detector. With these in place, the gain change due to placing the magnet on its track amounts to $< 1/2\%$.

The collimation system consists of four collimators made out of 15 mm thick Al. Behind each collimator is a 1.6 mm thick piece of lead. As shown in Fig. 14 the collimators are spaced using rods with the first collimator being just behind the entrance foil and the last collimator being centered at 0.1 m in front of the front face of the detector. The collimator holes are beveled at an angle of 2.67° to reduce beta-ray scattering from the exposed surfaces. The diameters of the collimator holes measured on the sides closer to the source are 8.19 mm, 12.51 mm, 13.59 mm, and 15.75 mm, respectively starting with the collimator closest to the entrance foil; the diameters of the holes measured on the sides

closer to the detector are slightly larger. This collimator effectively shields the detector from most of the radiation emanating from the outer surface of the rabbit. The two middle collimators are antiscattering baffles, and the last collimator is the solid-angle defining collimator. As fabricated (verified by counting known conversion electron sources as discussed in Section 6.C) the solid angle is ~ 0.0001 sr, allowing the central 20 mm diameter portion of the 34 mm diameter NE-110 detector to be illuminated by radiation from the source. As in the case of the gamma-ray detector, the reduced solid angle (compared with that subtended by the face of the NE-110 scintillator) has the advantage of improving the detector response to beta rays of energies > 0.5 MeV.

The depth of the NE-110 scintillator, 34 mm, was chosen because 34 mm is the most probable range of 8-MeV beta rays, and based upon earlier work²³ we anticipated observing very few beta rays having energies $E_\beta > 6$ MeV. The NE-110 plastic will also record pulses due to incident gamma radiation; however, for gamma rays having $E_\gamma > 2$ MeV, $> 80\%$ will not interact in the plastic. For gamma rays having $E_\gamma < 0.04$ MeV the probability of interaction is nearly 100% with most of the interactions due to photoelectric capture. For gamma rays having energies E_γ between 0.04 MeV and 2 MeV, the probability varies inversely with E_γ . Nearly all of the interactions for $E_\gamma > 0.1$ MeV are due to Compton scattering, and many of these Compton scattered gamma rays will interact in the surrounding CaF_2 scintillator. Such an event will be identified by the electronics system and will be stored in the alpha-spectrum region. As a consequence of this design, except for $E_\gamma < 0.04$ MeV, only 10 to 15% of the incident gamma radiation will result in events recorded solely in the NE-110 scintillator detector. From earlier studies² the number of beta rays

emanating from fission products produced in thermal-neutron fission of ^{235}U is approximately the same as the number of gamma rays emanating from the same fission products for times following fission of interest to this experiment.

The design of the equipment thus presented a favorable ratio of "magnet down" -- recording $(\beta + \gamma)$ -- to "magnet up" -- recording (γ) only -- reducing the uncertainties associated with the subtraction $(\beta) = (\beta + \gamma) - (\gamma)$, provided (a) that the deflected beta rays in the "magnet up" did not produce additional bremsstrahlung and (b) that beta rays in the "magnet down" did not also interact in the CaF_2 and thus be rejected by the electronics. The placing of the magnet just behind the entrance collimator meant that most of the deflected beta rays would strike the second collimator or the tube, the more energetic striking the third collimator. By making the tubing and collimators out of aluminum rather than higher-Z material the bremsstrahlung production was reduced in intensity and average bremsstrahlung energy. Thin pieces of lead were mounted behind each collimator to further attenuate any low-energy bremsstrahlung produced in the collimator. A careful study was made using a $\sim 20 \mu\text{Ci } ^{90}\text{Sr}$ source ($E_{\beta\text{max}} = 2.3 \text{ MeV}$, no gamma rays) and there was no evidence of detected bremsstrahlung having $E_\gamma > 0.08 \text{ MeV}$ (the electronic low-energy cutoff was $\sim 0.06 \text{ MeV}$ detected energy for all beta-ray data accumulation). Higher energy beta emitters were studied, e.g. ^{56}Mn (2.9 MeV), ^{38}Cl (4.8 MeV), and ^{20}F (5.4 MeV), where the numbers in parentheses are the maximum beta-ray end points. All of these radionuclides also emit gamma rays. In none of the studies was there clear-cut evidence for bremsstrahlung detection in the "magnet-up" configuration. It is somewhat more difficult to place a quantitative value on the amount of

bremsstrahlung which could have been present because of the competition from the real gamma radiation emanating from each of these three radio-nuclides. Such a quantitative analysis would have required obtaining an accurate gamma ray response matrix for this detector. (Our original experimental plan called for obtaining the gamma ray response, but the amount of time required to do so was more than available, and the priority for this task was low.) The isotope ^{20}F emits one gamma ray, $E_\gamma = 1.6$ MeV, and the shape of the detected response for $E_\beta > 0.2$ MeV is about that expected. There is no substantial rise in shape for $E_{\beta+\gamma} < 0.2$ MeV in the "magnet-up" spectrum which would signify a large bremsstrahlung contribution, but an accurate quantitative value cannot be ascertained. We estimate that for 5.4-MeV beta rays the maximum contribution to the integrated energy due to bremsstrahlung having $E_\gamma > 0.15$ MeV is $< 1\%$ of the total beta-ray integrated energy. In fact, a more serious problem for 5.4-MeV beta rays appears to be slit penetration observed in the "magnet down." This effect is discussed later (in Section 6.C) with regard to the beta-ray response matrix. The contribution to the integrated energy from slit-penetration has been measured as $\sim 0.4\%$ of the total beta-ray integrated energy.

The possibility of beta rays in the "magnet down" also interacting in the CaF_2 , and thus being rejected, was studied using the same sources listed above (^{90}Sr , ^{56}Mn , ^{38}Cl , and ^{20}F) and, in addition, the 0.8-sec isotope ^8Li produced in the $^7\text{Li}(n,\gamma)$ reaction. ^8Li is a beta emitter with end point ~ 13 MeV. Its very short lifetime precludes obtaining desirable statistical accuracy, but qualitatively the results are very useful. The design of the collimation system is such that beta rays having $E_\beta < 1.5$ MeV cannot escape into the CaF_2 and the probability of

escape for $E_\beta < 3$ MeV is almost negligible. No effect of this type is observed in the ^{56}Mn spectrum. A small effect is observed for the highest energy beta rays from ^{20}F , being at most a few percent for $E_\beta \approx 5$ MeV. Past work²³ indicated that the important region of E_β for beta-ray energy-release measurements was between 0.5 and 2.5 MeV, and that a 10% uncertainty in yield for $E_\beta > 4$ MeV corresponded to < 1% uncertainty in energy-release values. The "loss" of beta rays into the CaF_2 was estimated from the very qualitative data obtained from the ^6Li measurements, and is included in the beta ray response matrix, as discussed in Section 6.C.

3.E. Electronic System

The output of the photomultiplier tube (of either detector) contains all of the information related to the event which occurred in the detector. The purpose of the electronic system is to process this output to obtain the desired information from the photomultiplier output and transform it into the proper format to be sent to the computer. The important information consists of (a) the amplitude of the original light pulse caused by the event in the scintillator, and (b) the type of pulse, that is, for the beta-ray detector whether or not some of the interaction occurred in the CaF_2 , or for the gamma-ray detector if the pulse was due to alpha emission from the ^{241}Am source. In addition, the electronic system is designed to have a method for verifying its own working order which does not interfere with the measurement.

A block diagram of the electronics is shown in Fig. 16. The linear signal to the preamplifier is obtained from a dynode; the fast signal is obtained from the anode. Short cables are used to transfer these signals to their respective inputs. The time pickoff is of the constant-fraction type, and provides a standard fast negative output. Long cables connect the outputs from the time pickoff and the preamplifier to the electronics in the counting room.

Referring to Fig. 16 for the electronics in the counting room, the upper branch contains the fast-logic signal processing and the lower branch contains the time-to-amplitude for pulse shape discrimination conversion. In the middle is the pulse generator, which is used to monitor the stability of the amplification of the linear signal. If there is a gain shift in the preamplifier or amplifier, the peak in the spectrum corresponding

Fig. 16. Electronics Block Diagram. The electronics in the Detection Area (left of the divider) are shown in Fig. 11. This circuitry is set up for pulse amplification and pulse-shape discrimination.

to the pulser will shift; but if there is a shift due to the detector, the pulser peak will not shift. The main requirement is to ensure that the pulse from the pulser is treated as a nontagged event.

A properly "tagged" event is one which (a) in the beta-ray detector occurred only in the NE110 scintillator, or (b) in the NaI detector was a gamma ray and not an alpha from the ^{241}Am source. Conversely, an "untagged" event is everything else; (a) in the beta-ray detector all or some of the event occurred in the CaF_2 scintillator or else was an overload (much too big) event in the detector, or (b) in the NaI detector was an alpha from the ^{241}Am or else an overload event in this detector.

It is the differences in the characteristics of the scintillation processes for various (charged) particles that allow us to decide which events should be "tagged." The method used here is to measure the time from the initiation of the pulse (using the Time Pickoff) to the zero, or base-line crossing of a prompt bipolar pulse from the Main Amplifier. For ideally shaped pulses there would be two distinct times, different by ≥ 50 nsec. It is this difference, rather than the absolute times, that the electronic system exploits.

Figure 17 shows oscilloscope displays of four important parts of the system as set up for the gamma-ray detector. The first display is the input to the fast Coincidence unit. The pulse from the Time Pickoff is delayed until it is definitely bracketed by the pulse from the Zero Crossing. Then the lower-level discriminator of the Zero Crossing is raised until it is certain that there is a fast pulse (upper trace) for every Zero Crossing Pulse. In this fashion, we ensure that the low pulse-height bias is determined by the Zero Crossing Discriminator, which is a

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INPUTS TO COINCIDENCE CIRCUIT
 UPPER TRACE IS THE DELAYED OUTPUT
 OF THE TIME PICKOFF.
 LOWER TRACE IS THE OUTPUT OF THE
 ZERO CROSSING DISCRIMINATOR.

**INPUTS TO THE TIME-TO-AMPLITUDE
 CONVERTER**
 UPPER TRACE IS THE OUTPUT OF THE
 OR CIRCUIT.
 LOWER TRACE IS THE DELAYED OUT
 PUT OF THE ZERO CROSSING DIS
 CRIMINATOR.

**OUTPUT FROM THE TIME-TO-
 AMPLITUDE CONVERTER**
 LEFT LOWER TRACE IS THE UNGATED
 OUTPUT.
 RIGHT UPPER TRACE IS THE OUTPUT
 CORRESPONDING TO SINGLE CHANNEL
 ANALYZER SETTINGS FOR "TAGGED"
 PULSES.

**INPUTS TO ANALOG-TO-DIGITAL
 CONVERTER**
 UPPER TRACE IS THE DELAYED OUT
 PUT OF THE MAIN AMPLIFIER.
 LOWER TRACE IS THE EXTERNAL
 TRIGGER.

Fig. 17. Four Oscilloscope Displays. These displays show pulse shapes at various points in the electronics system when the system is working properly.

sharp cutoff, and not the Time Pickoff discriminator, which is a rounded cutoff. The output of this Coincidence is in time synchronization with the delayed signal from the Time Pickoff. This output does two things: (a) it provides a start pulse for the Time-to-amplitude Converter, and (b) after some delay provides an external trigger for the Analog-to-digital Converter. The OR Circuit is used to allow a pulse from the Pulse generator to do the same in parallel.

The second display (upper right) of Fig. 17 illustrates the inputs to the Time-to-amplitude Converter. The upper trace is the output of the OR Circuit. The lower trace is the delayed output of the Zero Crossing and shows three distinct groups, most readily discerned in the illustration by the bottoms of the pulses. The brightest of these corresponds to gamma-ray identification; to the left is that corresponding to alpha identification; to the left of the alphas is a faint group corresponding to the pulser. The time scale for this display is 100 nsec/division.

The lower left-hand display in Fig. 17 shows the output from the Time-to-amplitude Converter. Clearly these are two dominant groups, the smaller in amplitude corresponding to alpha identification and the larger to gamma-ray identification. A single channel analyzer is set about the upper group; the other trace in this display (displaced upward and to the right) shows the Time-to-amplitude Converter output corresponding to Single Channel Analyzer settings for "tagged" pulses.

The lower right-hand display in Fig. 17 shows the two inputs to the Amplitude-to-digital Converter, in particular their time relationship. The time scale for this display is 1 μ sec/division. Not shown is the "TAG" pulse which is a positive pulse starting $\sim 0.5 \mu$ sec before the

External Trigger, and lasting about 3 μ sec. The output of the Analog-to-digital Converter is in the form of 9 bits (corresponding to 512 channels) to the CAMAC Interface. The "TAC" is routed to the most significant bit of a 10-bit word. The CAMAC Interface then transmits a 10-bit word to the computer.

Ten scalers are used to monitor various points in the electronic system to ensure that it is working properly. For example, the contents of S3 should be the same (or very nearly the same) as the contents of S2 or else something is wrong with the Pulse Shape and Delay following the Time Pickoff. The contents of S5 are used during data reduction to estimate the efficiency of the Analog-to-digital Converter. The contents of S9 indicate the number of events with amplitudes greater than the lower-level discriminator of the Zero Crossing, and should be less than the contents of S3. The contents of S10 indicate events with amplitudes greater than the upper-level discriminator of the Zero Crossing; these are usually over-load pulses from the Main Amplifier.

A recording and reading of the scalers at the end of each run was standard practice for these measurements. These readings usually provided the first indications of trouble. Scanning the displayed spectrum during accumulation also provided indications of trouble. We were able in every instance of known component failure to stop data-taking with a minimum of lost time.

The Main Amplifier was used with two gain settings, a Low setting corresponding to ~ 8 MeV full scale (and ~ 0.2 MeV lower-level cutoff), and a High setting corresponding to ~ 2 MeV full scale (and ~ 0.05 MeV lower-level cutoff). Some preliminary gamma-ray data were obtained with

an even higher gain setting to study the gamma-ray region ≈ 30 keV corresponding to x-rays from the heavier fission products.

The electronics system as shown was used for either detector, and for the most part, the settings were very similar. The major difference was in the settings of the lower- and upper-level discriminators on the Single Channel Analyzer providing the "TAG" pulse. These settings were accomplished readily by replacing the input to the Analog-to-digital Converter with a second output from the Time-to-amplitude Converter. The spectrum observed for the ungated output of the Time-to-amplitude Converter for the gamma-ray detector is shown in the upper illustration of Fig. 18; that for the beta-ray detector is shown in the lower illustration. Identification of the peaks and indications of the Single Channel Analyzer settings are indicated on Fig. 18. Note that in both cases the pulser is recorded in the untagged spectrum.

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GAMMA-RAY DETECTOR

BETA-RAY DETECTOR

Fig. 18. Two Pulse-height Spectra of the Output of the Time-to-Amplitude Converter. The upper spectrum is for gamma-ray measurements and the lower for beta-ray measurements. The positions of the Single-channel Analyzer settings (lower level and upper level) are indicated. Both of these spectra exhibit the excellent separation between "tagged" and "not-tagged" pulses.

3.F. Counting Equipment

The equipment in the counting room is shown in Fig. 19 (refer to Fig. 3 for a schematic representation). In the foreground of Fig. 19 is the teletype used as both input to and output from the computer. Just above the teletype is the console to the computer, a Digital Equipment Corporation PDP-15 with 24K memory. To the left of the computer is a fast printer which is not used in this experiment. To the right of the computer is a large oscilloscope which is used to display data accumulated in the Nuclear Data pulse-height analyzer. This equipment was used during the set-up phase of the experiment. To the right of the large oscilloscope are the two Analog-to-Digital Converters (ADC) for this analyzer as well as two DECTAPE units. Then to the right of this equipment is the rack of electronics specifically for (and belonging to) this experiment.

At the top of the rack is a control box for the display of the data, which is discussed in more detail in section 4. The display appears on the oscilloscope mounted just below the control box. Below the display are two NIM bins containing standard and nonstandard electronics including the ADC used in this experiment to process the signals from the detectors. Then at the bottom is the CAMAC³⁶ crate containing several types of electronics. The left-most units in the CAMAC crate contain 12 scalars. In the middle are units which interface the CAMAC system to either the ADC or the display. At the right are the units which control and monitor the rabbit movement system. The right-most unit in the crate is the interface to the computer, and is the only unit requiring modification should the system be transferred to a different computer.

Fig. 19. Counting Equipment. This photograph shows the equipment in the counting room (see Fig. 3 for details).

Originally the Nuclear Data Analyzer shown in Fig. 19 was also to be used to count gamma rays from ^{99}Mo 50 to 80 hours after the irradiation to determine the number of fissions, n_f . A routine is included in the data-taking code (discussed in section 4) to dump data from the analyzer onto DECTAPE in a format required for further processing using an existing³⁷ code written for this PDP-15 system. This code cannot be used during our data-taking, nor can analyzer data be dumped as long as valid beta- or gamma-ray data are stored in memory. Preliminary runs indicated that using this analyzer for n_f determinations would pose serious limitations on the experiment and so the delayed-gamma-ray counting was shifted to another detector in a low-background counting room. This move allowed more freedom in scheduling measurements for n_f determination; so two measurements were made on each sample, the first between 16 and 24 hours after irradiation to obtain the yield of the 0.658-MeV gamma ray emanating from ^{97}Nb , a short-lived daughter of 17.0-hr ^{97}Zr , and the second between 50 and 80 hours to obtain the yield of the 0.140-MeV gamma ray emanating from 66.0-hr ^{99}Mo and the yield of the 0.228-MeV gamma ray emanating from 78.0-hr ^{132}Te . The primary detector used for these measurements is a 90 cm³ Ge(Li) detector having a full-energy efficiency of 15% with respect to a 76 mm diameter by 76 mm deep NaI crystal for $E_\gamma = 1.33$ MeV at a source-detector distance of 25 cm. Electronics used with this detector include a high-voltage supply, a high-resolution amplifier including pulse-shaping and pole-zero adjustments, and an existing pulse-height analyzer having a capacity of 1024 channels.

Additional measurements were made on the samples at various times following irradiation using an existing x-ray detector. This detector

is made of intrinsic germanium, 5 mm deep and 200 mm² surface area, and has a resolution of ~ 0.0007 MeV to 0.1- MeV photons. Measurements made with this detector were used to (a) estimate the importance of low-energy x-rays to the total energy integral and (b) ensure that peaks observed in spectra obtained with the 90-cm³ detector for $E_{\gamma} < 0.3$ MeV were due only to single gamma rays. In addition, during the course of the experiment, a method of using this detector to determine source strengths for certain sources was developed³⁸ and used as an adjunct to the efficiency calibration of the 90-cm³ detector. This detector was also used in a careful reexamination³⁹ of the decay of ⁹⁹Mo when it became obvious during preliminary measurements that such study was required.

4. EXPERIMENTAL DETAILS, SOFTWARE

A brief description of the main features of the computer program written for the PDP-15 computer to control and monitor the experiment is presented in this section. A complete description and listing of the code has been published.²⁰

Once the sample has been prepared, mounted in a rabbit, and the rabbit placed in position in Hood No. 1 (see Fig. 3), the computer controls the subsequent rabbit movements and data-acquisition. To do this the desired parameters are entered into computer memory through the use of the command BKK_ [note: to have a command take effect, the three characters must be followed by a space (_) or a carriage return (+)]. Fig. 20 illustrates an example of parameter input. The underlined portions are the typed-in entries by the user; the portions not underlined are typed out by the computer. In this routine a comma (,) or a carriage return (+) is used as a number delimiter.

The experiment is initiated by entering the command STR_. After transfer of the rabbit to and from the irradiation position, the first counting interval is between 70 and 110 sec after the nominal end of irradiation (the 0.3-sec reduction discussed in Section 3.A is not included in these times); the second counting interval starts at the end of the first; the third counting interval starts at the end of the second; etc. At the end of the last counting interval the computer reports END OF RUN.

Each count registered in the detector is processed by the electronics (see Fig. 16) and the output of the ADC, a pattern of 9 bits, and the Tag pulse, if present, are transmitted to the computer by the CAMAC interface.

```

BKK+
INPUT DATA LOG
235-U B-21 1UGM G=8 22:25 9-JULY+
RUN NUMBER = 2821,JKD
IRRADIATION TIME = 100.7,
NO OF TIME INTERVALS = 15,
70, 110, 170, 250, 350, 550, 750, 1150, 1550, 1950+
2450, 2950, 3950, 5950, 9950, 13950,
S(SOURCE) OR R(RABBIT)? R

```

Fig. 20. Example of initiation of data taking; user entries are underlined. BKK+ initiates the bookkeeping routine. The next user entry is an identification title. The next entry is a run number, followed by the irradiation time in sec. In this example the sample cooling time is 70 sec, and the first of 15 counting intervals is 40 sec. The last entry, R, causes the rabbit transport system to be initiated. The irradiation time entry is 0.7 sec greater than the actual irradiation time, as discussed in Section 3.A.

During the course of the experiment many such data are transferred to the computer. Data transmission rates varied between $\sim 150/\text{sec}$ for the longest time intervals to $\sim 12000/\text{sec}$ for the shortest time intervals following end of irradiation. The program was proved capable of handling data rates in excess of $20000/\text{sec}$ without loss of information. The data transmission takes place without interrupting the program counter, on a cycle-stealing basis. When a 256-word data buffer is full, a program interrupt occurs causing a new buffer region to be defined. The computer then processes each of the 256 words to determine the pulse height in one of two 512-channel spectra separated according to the tag bit and adds one count to previously analyzed data which are being stored in memory. At the end of the first counting period a program interrupt occurs causing a new buffer region to be defined. The computer processes the partially-filled old buffer adding to results already in core. Then the first-word-address of the spectrum storage region is incremented by 1024 locations.

In this manner spectra are obtained for up to 17 consecutive time intervals; most of the 24K-word storage is used to store accumulated data.

When the computer is not otherwise engaged, a portion of the data-accumulation region is displayed. The display and other electronics is shown in Fig. 21. At the top is the display controller unit and just below it is the display. The six push buttons on the left side of the controller determine horizontal scaling, overlap display, and choice of data to be displayed. The four push buttons in the center control movements of two displayed vertical lines. The left-most multiple switch controls the vertical scale of the display and the right-most switch

Fig. 21. Shown in this photograph are, from top to bottom, (a) the display control box, (b) the display oscilloscope, (c) two NIM bins containing modular electronics, and (d) the CAMAC bin containing: 12 scalers at the left of the bin, in the center are the control units for the ADC, the display, and the telephone transmission, and to the right is the interface module to the PDP-15.

partially controls the choice of region(s) to be displayed. The CAMAC interface is used to transmit trace coordinates to an oscilloscope, and to strobe the status of ten push buttons and two multiple position switches and transmit this status to the computer upon command. This status determines which of the 34 512-channel regions to display, the horizontal and vertical scales, and display and movement of vertically displayed lines which are used to define regions of interest. The computer operates in the display loop essentially all of the time; count rates $> 20000/\text{sec}$ will affect the display continuity sufficiently to be observable. All, or a portion, of a second data-accumulation region may be simultaneously displayed (overlap display).

Primary data storage is on DECTAPE; however, the computer may act as a remote terminal to a PDP-10 (through other peripheral computers) and transmit data to specified disk storage.

Data on DECTAPE may be read into the computer. All or some of the data dumped with a given filename may be retrieved. A partial readin may be routed into a desired part of the data storage area to facilitate overlap display with data from a different run.

The various subprograms are initiated by entering a 3-character command via the teletype. These are summarized in Table 1. Some operations render other commands inoperable until completion of the desired operation. For example, writing on a DECTAPE prohibits any other DECTAPE command.

During the course of this experiment many hundreds of irradiations were required (almost 200 for the final data-taking runs). The software was designed to provide an efficient and reproducible method of controlling and monitoring each irradiation and subsequent beta- or gamma-ray counting, and also to be simple to use.

TABLE 1. TYPE-IN INSTRUCTIONS

Command	Subprogram performed.
BKK	Receive from teletype and store information and parameters for the next data-taking period.
BUN	Abort status of rabbit transport system.
DDT	Jump to DDT for debugging. Not used in full version of code.
DEC	Set for decimal input and output.
DIR	Type out DECTAPE directory.
DMP	Dump data on DECTAPE.
ETA	Set rabbit transport system for beta-ray experiment.
GET	Get information from DECTAPE.
HIL	Set vertical display line at channel typed in.
HI=	Type channel no. and contents of right-most vertical display line.
HLT	Halt data accumulation.
INT	Integrate data displayed between vertical lines. (Centroid is obtained also.)
JFE	Dump contents of Nuclear Data 5050 Analyzer onto DECTAPE along with typed-in information.
KLN	Kill vertical lines on the display.
LNS	Type channel numbers and contents for both displayed lines.
LOL	Set a second vertical display line at typed-in channel.
LO=	Type channel number and contents of left vertical line.
MMA	(Last 3 letters of GAMMA.) Set rabbit transport system for gamma-ray experiment.
OCT	Set for octal input.

TABLE 1. TYPE-IN INSTRUCTIONS (continued)

OVR	Type out overflow list.
RUN	Type out current run number.
SCA	Type out scaler information.
SOU	Start a one-spectrum run without BKK. Used for sources during set-up procedures.
STR	Start data run.
TEN	Send data to PDP-10.
TOL	Type out data displayed between vertical lines.
TS=	Type out current time (since end of irradiation).
TYI	Enter N integers into memory starting at absolute location M. Not often used.
XTN	Change the current filename extension.

5. DETERMINATION OF THE NUMBER OF FISSIONS

Before the analysis of the data can be completed, the number of fissions, n_f , for each sample must be determined. This information is required to properly combine data obtained at different gains and to normalize the beta-decay magnet-up data to the magnet-down data for proper subtraction. Originally¹⁷ there were to be three methods of determining n_f ; the first by measuring the thermal flux, and knowing the sample size and fission cross section to obtain the number of fissions, the second by monitoring the 140.5-keV gamma ray from ^{99}Mo sixty or more hours following fission using a high resolution detector, and the third was to use an existing delayed-neutron detector (shown in Fig. 3). The third method was never used because (a) it would limit the beta- or gamma-ray decay power measurements to times-after-irradiation > 200 sec, and (b) an absolute calibration of the delayed-neutron detector was required. The first two of these methods were used for preliminary data taking.¹⁷ For the ^{99}Mo measurement the Ge(Li) detector was calibrated by fabricating a source of ^{99}Mo and determining its source strength by beta-ray counting using a calibrated ion chamber. This calibration resulted in knowing the source strength to $\sim 3\%$. The thermal-neutron flux was determined to $\sim 1\%$ but the agreement between the two methods of n_f determination was poorer than expected from the assigned uncertainties. Further study of this discrepancy indicated that the sample position and/or incident flux during irradiation was not sufficiently reproducible to warrant assigning an uncertainty $< 5\%$ to n_f determined using this method. It was decided that only the second method would provide data to determine n_f to better than 3% , and it was decided to add other delayed gamma

rays to the Ge(Li) measurements and not rely solely on the 140-keV gamma ray from ^{99}Mo . After some trial and error, the additional gamma rays chosen were the 49.7- and 228-keV gamma rays in ^{132}Te decay and the 658-keV gamma ray in the ^{97}Nb (from ^{97}Zr) decay. These were chosen because, like the 140-keV gamma ray from ^{99}Mo decay, they are of sufficient energy separation from other fission product gamma rays to be clearly separated by the resolving power of a standard Ge(Li) detector, the branching ratios are well known, and the cumulative fission yields for the parent nuclides are well known. In addition to these parameters, a precision calibration of the Ge(Li) detector's efficiency as a function of gamma-ray energy was required. This was done by first determining the efficiency calibration for $D = 200$ mm (where D = distance of the source from the top of the housing containing the detector), and then determining the variation of efficiency for different distances D . A check was made on this procedure by thermal-neutron irradiation of a specially designed fission chamber, and then counting the delayed fission-product gamma rays using the Ge(Li) detector. The next four sections discuss pertinent aspects of these calibrations.

5.A. Determination of Detector Efficiency as a Function of Gamma-ray Energy

For $D = 200$ mm, real summing in the detector of gamma rays in cascade may be neglected. Accuracy of calibration is limited only by current knowledge of standard source strengths and nuclide branching-ratio data, both of which are $\sim 2\%$ for sources we used for a combined uncertainty of $\sim 3\%$ for a given gamma ray.

Particular emphasis was placed on the gamma-ray energy range of 100 to 1000 keV, since most of the gamma rays from fission products being used to measure n_f lie in this energy band. We did not study the detector's characteristics for $E_\gamma > 1620$ keV. Our well-calibrated sources have source strengths between 1 and 10 μCi , which provided satisfactory data for $D = 200$ mm. The most careful efficiency calibration was done at this distance; Table 2 gives a detailed work sheet for this calibration. Having obtained the experimental calibration, a computer subroutine was written which combines an analytic expression and table look-up procedures to provide efficiencies for data-reduction calculations. The efficiencies computed by the routine are included in Table 2, and variances between experimental results and calculation are included. These latter results appear to be quite satisfactory from a random statistical viewpoint, and suggest that the overall average uncertainty (1 σ) in the efficiency calibration is $\leq 2\%$ for any given photon energy for $D = 200$ mm.

TABLE 2. Summary of Calibration of Ge(Li) Detector at D = 200 mm

E _γ (keV)	Source Information ^a	Efficiency Values ^c				Del ^f
		Branching (%)	Reference ^b	Experiment ^d	Code ^e	Difference (%)
59.6	²⁴¹ Am; A-S; 11.70; 1.7%; 1-4-74	36.3 ± 0.4	R1	4.648 ± 0.098	4.760	-2.40
80.58	¹³⁷ Ba; A-S; 10.58; 1.0%; 1-4-74	35.6 ± 0.6	5114	10.37 ± 0.24	9.992	3.57
88.0	¹⁰⁹ Cd; A-S; 11.10; 3.3%; 1-5-76	3.72 ± 0.11	5114	11.19 ± 0.55	11.41	-1.97
96.7	⁷⁵ Se; ORN; 1.75; 3.0%; 31-12-75	3.21 ± 0.06	AD	12.68 ± 0.76	12.38	2.37
121.1	⁷⁵ Se; (96.7)	16.15 ± 0.50	AD	15.40 ± 0.66	15.16	1.56
121.8	¹³² Eu; BNM; 11.84; 0.5%; 11-2-76	28.20 ± 0.31	Fab	15.30 ± 0.17	15.19	0.72
122.0	⁵⁷ Co; A-S; 10.16; 1.5%; 1-2-76	85.59 ± 0.19	5114	15.13 ± 0.24	15.20	-0.46
136.0	⁷⁵ Se; (96.7)	56.28 ± 0.70	AD	15.51 ± 0.51	15.59	-0.52
136.5	⁵⁷ Co; (122.0)	10.61 ± 0.18	5114	15.80 ± 0.37	15.59	1.33
165.9	¹³⁹ Ce; A-S; 11.60; 3.3%; 1-5-76	80.06 ± 0.13	5114	14.50 ± 0.48	15.05	-3.79
198.6	⁷⁵ Se; (96.7)	1.39 ± 0.05	AD	13.70 ± 0.95	13.79	-0.66
238.68	ThC"; ORN; 3.12; 5.0%; 1-1-76	47.4 ± 1.4	H	12.26 ± 0.73	12.15	0.90
244.78	¹⁵² Eu; (121.8)	7.42 ± 0.09	Fab	12.38 ± 0.18	11.93	3.63
255.1	¹¹³ Sn; A-S; 12.60; 2.6%; 1-5-76	2.12 ± 0.07	5114	11.55 ± 0.50	11.57	-0.17
264.7	⁷⁵ Se; (96.7)	57.28 ± 1.55	AD	11.33 ± 0.46	11.23	0.88
276.4	¹³³ Ba; (80.5)	7.30 ± 0.13	5114	10.83 ± 0.26	10.86	-0.28
277.4	ThC"; (238.6)	2.21 ± 0.11	H	10.66 ± 0.76	10.83	-1.59
279.2	²⁰³ Hg; A-S; 22.94; 1.0%; 1-2-76	81.5 ± 0.8	5114	10.91 ± 0.15	10.78	1.19
279.5	⁷⁵ Se; (96.7)	24.76 ± 0.55	AD	10.45 ± 0.40	10.78	-3.16
295.9	¹⁵² Eu; (121.8)	0.423 ± 0.006	Fab	10.27 ± 0.53	10.35	-0.78
300.0	ThC"; (238.6)	2.95 ± 0.11	H	10.87 ± 0.66	10.25	5.70
302.9	¹³³ Ba; (80.5)	18.66 ± 0.21	5114	9.93 ± 0.20	10.16	-2.32
303.9	⁷⁵ Se; (96.7)	1.32 ± 0.05	AD	9.93 ± 0.61	10.13	-2.01
344.3	¹⁵² Eu; (121.8)	26.40 ± 0.26	Fab	9.203 ± 0.098	9.033	1.85
356.0	¹³³ Ba; (80.5)	62.4 ± 0.3	5114	8.725 ± 0.146	8.760	-0.40
367.8	¹⁵² Eu; (121.8)	0.840 ± 0.014	Fab	8.427 ± 0.335	8.502	-0.89
383.9	¹³³ Ba; (80.5)	8.86 ± 0.14	5114	8.210 ± 0.159	8.174	0.44
391.7	¹¹³ Sn; (255.1)	64.9 ± 0.2	5114	8.002 ± 0.210	8.025	-0.29
400.6	⁷⁵ Se; (96.7)	11.20 ± 0.50	AD	7.82 ± 0.46	7.861	-0.50
411.38	¹⁵² Eu; (121.8)	2.39 ± 0.03	Fab	8.248 ± 0.186	7.673	6.97
444.08	¹⁵² Eu; (121.8)	3.08 ± 0.03	Fab	7.194 ± 0.115	7.153	0.57
452.9	ThC"; (238.6)	0.364 ± 0.030	H	7.11 ± 0.67	7.024	1.21
497.1	¹⁰³ Ru; ORN; 13.20; 3.3%; 19-12-75	86.4 ± 2.4	5114	6.49 ± 0.28	6.449	0.62
511.0	²² Na; A-S; 10.93; 1.3%; 1-4-74	180.0	5114	6.154 ± 0.085	6.288	-2.18
564.58	¹⁵² Eu; (121.8)	0.614 ± 0.014	Fab	5.59 ± 0.38	5.739	-2.68

569.7	²⁰⁷ Bi; A-S; 13.30; 3.0%; 1-5-76	97.8 ± 0.5	5114	5.53 ± 0.17	5.690	-2.89	0.94
583.2	ThC"; (238.6)	28.70 ± 0.10	H	5.74 ± 0.35	5.57 ^a	2.96	0.49
610.3	¹⁰³ Ru; (497.1)	5.3 ± 0.3	5114	5.46 ± 0.41	5.342	1.47	0.20
661.6	¹³⁷ Cs; A-S; 10.55; 1.3%; 1-4-74	86.0 ± 0.9	R-2	4.966 ± 0.081	4.961	0.10	0.06
688.68	¹⁵² Eu; (121.8)	0.849 ± 0.012	Fab	5.13 ± 0.25	4.782	6.82	1.40
727.3	ThC"; (238.6)	6.36 ± 0.19	H	4.56 ± 0.28	4.548	0.25	0.04
778.9	¹⁵² Eu; (121.8)	13.00 ± 0.14	Fab	4.237 ± 0.054	4.271	-0.80	0.63
834.8	⁵⁴ Mn; A-S; 10.70; 1.3%; 1-2-76	100.0	5114	4.045 ± 0.060	4.018	0.66	0.45
860.5	ThC"; (238.6)	4.31 ± 0.27	H	3.96 ± 0.32	3.915	1.14	0.14
867.4	¹⁵² Eu; (121.8)	4.16 ± 0.04	Fab	3.976 ± 0.072	3.889	2.19	1.21
964.08	¹⁵² Eu; (121.8)	14.48 ± 0.15	Fab	3.483 ± 0.037	3.553	-2.01	1.89
1063.6	²⁰⁷ Bi; (569.7)	74.3	Fab	3.246 ± 0.098	3.266	-0.62	0.20
1086.48	¹⁵² Eu; (121.8)	11.84 ± 0.14	Fab	3.127 ± 0.039	3.208	-2.59	2.08
1112.08	¹⁵² Eu; (121.8)	13.55 ± 0.14	Fab	3.110 ± 0.035	3.144	-1.09	0.97
1115.5	⁶⁵ Zn; ORN; 11.63; 1.5%; 16-1-76	50.75 ± 0.10	5114	3.133 ± 0.047	3.136	-0.10	0.13
1173.2	⁶⁰ Co; A-S; 10.66; 0.7%; 1-4-74	99.90 ± 0.02	5114	3.008 ± 0.023	3.004	0.13	0.17
1212.9	¹⁵² Eu; (121.8)	1.390 ± 0.017	Fab	2.860 ± 0.055	2.919	-2.06	1.07
1274.5	²² Na; (511.0)	99.94 ± 0.02	5114	2.760 ± 0.038	2.798	-1.38	1.00
1298.78	¹⁵² Eu; (121.8)	1.743 ± 0.026	Fab	2.613 ± 0.054	2.754	2.26	1.09
1332.5	⁶⁰ Co; (1173.2)	100.0	5114	2.751 ± 0.021	2.694	2.07	2.71
1408.0	¹⁵² Eu; (121.8)	20.70 ± 0.20	Fab	2.530 ± 0.026	2.570	-1.58	1.54
1457.6	¹⁵² Eu; (121.8)	0.47 ± 0.03	H	2.57 ± 0.16	2.495	2.92	0.47
1528.0	¹⁵² Eu; (121.8)	0.27 ± 0.03	H	2.42 ± 0.24	2.396	1.00	0.10
1620.7	ThC"; (238.6)	1.41 ± 0.09	H	2.30 ± 0.19	2.277	1.00	0.12

a In this order: Isotope; Fabricator; Original Source Strength (S) in μ Ci; estimated standard deviation; date of S given as day-month-year. Fabricator keys are (a) A-S, Amersham-Searle; (b) BNM = Commissariat A L'Energie Atomique (France), Bureau National de Metrologie; and (c) ORN = ORNL. This information is given for the lowest energy gamma ray of the source; repeated references to the same source refer to the energy of that gamma ray in parentheses.

b Key to references: (a) R1, J. Legrand, J. P. Perolat, C. Bac, and J. Gorry, Int. J. App. Rad. Isotopes 26, 179 (1975); (b) 5114, "Nuclear Decay Data for Selected Radionuclides," ed. M. J. Martin, ORNL-5114 (March 1976); (c) AD, W. W. Bowman and K. W. MacMurdo, Atomic Data and Nucl. Data Tab. 13, 89 (1974); (d) Fab, data given by fabricator; (e) H, adapted from R. L. Heath, ANCR-1000-2, Vol II (undated); and (f) R-2, I. W. Goodier, J. L. Makepeace, L. E. Stuart, Int. J. App. Rad. Isotopes 26, 490 (1975).

c Units are peak-counts/10000 source gamma rays.

d Assigned uncertainty, σ , is one standard deviation.

e Values obtained from computer subroutine which is used for number-of-fission analysis.

f Del = Absolute value of Difference/ σ , where σ = sum of all uncertainties in quadrature.

g Data given for sum of closely spaced doublets.

5.B. Determination of Efficiency as a Function of Distance

For $D \leq 100$ mm real summing in the detector cannot be neglected,⁴⁰ so to determine the efficiency as a function of distance a few selected gamma rays were used for which real summing is either absent or not observed (because of the detector's low efficiency for x-rays). These included 88 keV (^{109}Cd), 159 keV ($^{123}\text{Te}^*$), 365 keV (^{113}Sn), 412 keV (^{198}Au), 662 keV (^{137}Cs), and 834 keV (^{54}Mn). From these data, a computer algorithm for obtaining the change in efficiency as a function of source-detector distance for the 90 cm³ detector and for constant E_γ was developed. The method first computes the total efficiency for energy E_γ , at some source-detector distance D , by calculating the probability of an interaction along straight path length L in the detector on a radial from the source entering the front face of the detector and leaving it either at the back face or else at the side;

$$d\varepsilon(E_\gamma, L) = \left[1 - e^{-\mu(E_\gamma)L} \right] d\omega, \quad (5.B.1)$$

where ε = efficiency for any interaction, $\mu(E_\gamma)$ = attenuation coefficient in Ge, and $d\omega$ is the solid angle of an incremental area on the detector front face, chosen small enough such that the path length L along the radial from the source is essentially constant. Numerical integration of the function in Eq. (5.B.1) yields the total efficiency $\varepsilon(E_\gamma, D)$. Empirically we observed that the peak-to-total ratio, which includes absorptions not explicit in Eq. (5.B.1), was a function of D , but was almost completely independent of E_γ . Therefore, the peak efficiency is determined from

$$\varepsilon_{pk}(E_\gamma, D) = \varepsilon(E_\gamma, D) R(D) \quad (5.B.2)$$

where $R(D)$ is interpolated from tabulated values determined from experiment using the sources mentioned above.

5.C. Efficiency Calibration for an Intrinsic-Ge X-ray Detector

This x-ray detector was borrowed for the purpose of looking at delayed x-rays and low-energy gamma rays ($E_\gamma < 0.3 \text{ MeV}$) with better resolution than the primary $90 \text{ cm}^3 \text{ Ge(Li)}$ detector discussed above. The x-ray detector was 5 mm deep by 200 mm^2 area behind two $254 \text{ }\mu\text{m}$ thick Be windows. The efficiency calibration was obtained for $D = 28 \text{ mm}$ using calibrated sources of ^{241}Am , ^{243}Am , ^{109}Cd , ^{57}Co , ^{139}Ce , and ^{203}Hg . This detector's efficiency curve is shown in Fig. 22. The overall average uncertainty (1σ) is estimated to be 3% for this curve.

Data were obtained using this detector for n_f determinations using the 49.7-keV gamma ray from ^{132}Te decay; in addition, data were obtained for gamma rays due to decay of ^{235}U which were used to establish the number of ^{235}U atoms in fission chamber foils, as discussed later in Section 5.E.

5.D. Number of Fissions Determined from Cumulative Fission Yields,
Branching Ratios and Efficiencies for ^{99}Mo , ^{132}Te , and ^{97}Zr

After irradiation and beta- or gamma-ray decay-heat power measurement, each ^{235}U sample was removed to the low background counting room for determination of n_f , the number of fissions in the sample. Each sample was counted using the large-volume Ge(Li) detector at between 16 and 24 hours after irradiation. Each sample was counted a second time at between 60 and 80 hours. Because of the length of time needed for each count (from 1 to 4 hours), the second count sometimes utilized the x-ray detector in the last section. About one-quarter of the samples were counted a third time at about 7 days following irradiation using the large-volume Ge(Li) detector.

The number of fissions was obtained from these data by

$$n_f = \frac{Y e^{\lambda T_{\text{wait}}}}{(1 - e^{-\lambda T_{\text{count}}})} \cdot \frac{1}{\epsilon(E_Y, D) B C_Y}, \quad (5.D.1)$$

where Y = measured peak yield, λ = appropriate decay constant, T_{wait} = sample cooling time to beginning of counting time (T_{count}), $\epsilon(E_Y, D)$ = efficiency obtained as discussed above, C_Y = cumulative fission yield for the isotope, and B = branching ratio for the desired gamma ray corrected for daughter lifetimes.

Table 3 summarizes the results for the 1-sec exposures for the gamma-ray decay power measurements, a statistically difficult case because of the relatively small number of fissions per sample. For this case the 49.7-keV gamma ray was used to obtain information on ^{132}Te because the resulting n_f were more consistent than those obtained for $E_Y = 228$ keV. Analysis of the variation in these data shows that the

TABLE 3. Determination of Number of Fissions
for Gamma-Energy Release Data, $T_{\text{irrad}} = 1 \text{ sec}$

Isotope	⁹⁹ Mo	¹³² Te	⁹⁷ Zr(⁹⁷ Nb)	
Fission Product Yield (%)	6.15±0.04	4.24±0.04	5.87±0.06	
Gamma-Ray Energy (keV)	140.5	49.7	658	
Gamma-Ray Branching (%)	90.7±0.6	14.5±0.2	105.7±0.5 ^a	
Sample	Mass			
B52	5 ^b	1.25 ^{c,d}	1.23	1.26
B61	10	2.61	2.54	2.44
B62	10	2.66	2.50	2.55
B53	5	1.31	1.32	1.28
B54	5	1.30	1.27	1.26
B63	10	2.71	2.66	2.57
B64	10	2.51	2.69	2.67
B65	10	2.43	2.42	2.69
B55	5	1.41	1.28	1.48
B66	10	2.97	3.01	2.81
B67	10	2.66	2.65	2.74
Total		23.82	23.57	23.75

^aCorrected for ^{97}Nb half life at equilibrium with ^{97}Zr decay.

^bNominal mass of sample is in micrograms.

^cMultiply by 10^8 to get the number of fissions.

^dStatistical uncertainties on individual measurements vary between 0.7 and 4%.

observed values scatter with a standard error ranging between 2% for ^{99}Mo to 4% for ^{97}Zr ; the estimates of number of fissions given by the three nuclides scatter with a standard error of 1.2%. This σ is valid for the combined decay-heat energy release data having $0.5 \leq E_{\gamma} \leq 1.6$ MeV, since this table has information for both NaI-spectrometer gain settings. Standard errors for the separate gain settings are $\sim 50\%$ larger. The total uncertainty must also include uncertainties in fission-product yields,^{18,41} gamma-ray branching ratios, and detector efficiencies (discussed above). We assign an uncertainty of 2.5% to the total number of fissions determined by this method for all samples used; most of this uncertainty is related to the uncertainty in detector efficiency calibration.

5.E. External-beam Normalization Check

Of some concern was the possibility of an undetected error in values of one, or more, of the variables mentioned in the discussion of Eq. (5.D.1). As a check on this question a separate experiment was initiated; for this experiment a suitably designed fission-chamber system containing ^{235}U was irradiated in an external beam at the ORR. Prior to irradiation optical realignment of the external beam hole was completed, and the monitoring chambers mounted adjacent to the beam hole were calibrated. As shown in Fig. 23, there are two chambers. In each chamber is a ^{235}U foil of $\sim 1 \text{ g/m}^2$ thickness. Each chamber was filled to a pressure of 10^5 Pa (1 atm) with pure methane. The first step in the experiment was to obtain a pulse-height spectrum of the fission-fragment pulses at very low incident-neutron intensity, and then to set a low-energy discriminator using the measured pulse-height spectrum. Then the incident-neutron intensity was increased so that ~ 20000 fissions/sec occurred in each chamber. Examples of pulse-height spectra and low-energy discriminator setting are shown in Fig. 24. The chambers were irradiated for $\sim 15 \text{ hr}$ oriented such that neutrons were impinging from the top in Fig. 23, and then they were reversed so that neutrons were impinging from the bottom for another $\sim 20 \text{ hr}$.

This irradiation was preceded by $\sim 30 \text{ hr}$ of irradiation of a second set of fission chambers. One of this second set was similar to the bottom chamber in Fig. 23; the only difference was that the 15° bevel was opposite that shown. The other chamber was identical in construction (shape) to the top chamber in Fig. 23 but had a foil $\sim 4 \text{ g/m}^2$. The purpose of studying data obtained for this second set of chambers was

Fig. 24. Portion of Two Spectra of Fission Fragments Obtained from a Thin-Foil Chamber. The spectra continue to channel 200. The counting rate was $\sim 22000/\text{sec}$. V_P indicates the channel containing the maximum number of counts. $V_U = 0.54 V_P$ and $V_L = 0.5 V_U$. The electronic bias of the present system is vic. channel 44. Also shown is a (no-beam) alpha spectrum. There were 1.876×10^6 counts in the ungated (internal trigger) spectrum and 1.687×10^6 counts in the gated (external trigger) spectrum obtained in 80 sec live-time runs.

to determine corrections to the measured n_f due to (a) loss of fragments in the sample (i.e. fission fragment paths parallel to and effectively remaining in the sample), and (b) estimation of the fraction of low-energy events resulting in pulse magnitudes $<$ electronic bias,⁴² in order to determine the absolute efficiency of each chamber for counting fission reactions.

Following the irradiations the chambers were taken to the low-background counting room. The fission product gamma rays were counted for each chamber separately and for the pair together, oriented as in Fig. 23 for the Ge(Li) detector placed at the bottom of this figure. Data were obtained for cooling times ranging from 12 hours to 9 days and for nominal chamber-to-detector distances of 20, 30, and 200 mm. For each distance, and for each gamma ray of interest (140.5 keV from ^{99}Mo , 228 keV from ^{132}Te , and 658 keV from ^{97}Zr - ^{97}Nb) a K-factor was determined. This K-factor contains the detector efficiency, B, and C_Y from Eq. (5.D.1), that is

$$K = \frac{1}{\epsilon(E_Y, D) B C_Y} \quad (5.E.1)$$

For the two close distances, $D = 20$ and 30 mm, the finite extension of the source (because fission products are embedded in the ceramic and Al foils) required determining a correction to the observed gamma-ray yield because of the variation in detector efficiency for a source not on the center line. This effect was measured for point sources of ^{139}Ce (165 keV), ^{203}Hg (279 keV) and ^{54}Mn (834 keV) and within the accuracy of the measurement the efficiency variation with distance $\underline{d}$ from the center line was independent of E_Y . For $D = 30$ mm, the efficiency for $\underline{d} = 1.2$ cm

was 0.8% less than that for $d = 0.0$, and for $d = 2.4$ cm was $3.5 \pm 0.8\%$ less than that for $d = 0.0$. Assuming an uniform distribution of half of the fission products in the ^{235}U foil and a distribution of the other half of the fission products in the Al foils and ceramic to be $\sim$ inversely proportional to d , the correction factor for the observed gamma-ray yield is $0.62 \pm 0.12\%$.

Gamma-ray attenuations by the Al foils were also computed using tabulated attenuation factors.⁴³ These ranged between 0.5% for $E_\gamma = 140$ keV to $< 0.1\%$ for $E_\gamma = 658$ keV.

As mentioned above, the fission chambers were exposed for about 30 hours to a thermal-neutron beam having a flux of $\sim 10^8$ n/cm²/sec. The chambers were exposed in pairs, the first pair being the thick-foil, thin-foil combination and the second pair being the two thin-foil chambers shown in Fig. 23 back to back. A negative high voltage of 300 volts was applied to the foils having the ^{235}U deposits. Separate signal leads were obtained from the (virtual) ground side of each chamber (i.e. the foils opposite the ^{235}U deposits).

The counting channel for each fission chamber consisted of a fast current preamplifier, a constant fraction timing discriminator and two independent scalers. The discriminator thresholds were set as low as the system noise level permitted. The thresholds were checked approximately every 8 hours by analyzing the charge output of the preamplifier using the constant fraction discriminator to externally trigger the pulse-height analyzer. During the course of the experiment dead time checks were made alternately on the two counting channels using a pulser and measuring the number of pulser events lost due to dead time of the counting channels.

The pulser output was inserted into the preamplifier input and also counted by a scaler. Two methods were used to measure the losses due to dead time. A coincidence circuit having approximately a 25 nsec resolving time provided a count of number of events at the discriminator output in coincidence with the pulser. The output of the coincidence circuit was counted in two independent scalers. The second method consisted of starting a time-to-height converter with the pulser and stopping with the discriminator output. The output of the time-to-height converter was analyzed providing a time distribution of events preceding the pulser by ~ 50 nsec and the pulser events.

In this manner the dead time τ was determined to be about 75 nsec from the equation⁴⁴

$$N_{\text{obs}} = N_{\text{true}} \exp(-\tau N_{\text{true}}), \quad (5.E.2)$$

an appropriate form for the constant-fraction discriminator used.

The scaler system consisted of two CAMAC 100 Mhz scaler modules, a crate controller, an auxiliary controller, and an ASR 33 Teletype. Starting, stopping, and clearing of the scalers was via commands from the teletype. Periodically the scaler data were output to the teletype. These data consisted of the date and time of day, the total accumulated counts in each scaler, the increment to each scaler since the last output, and the average count rate since the last output. The beam port flux monitor was recorded on a strip chart recorder.

At periodic intervals, fission-fragment pulse-height spectra were obtained to ensure that the electronic discriminator level had not changed. The spectrum shown in Fig. 24 was obtained for a thin-foil chamber at a count rate of 22000/sec.

The next step was to determine the number of fissions that occurred in the fission chambers, and then to determine the "effective" number of fissions for each of the three fission products of interest, that is, to determine the equivalent to the number of fissions which would have occurred in a "pulse" at the clock time corresponding to the end of the irradiation which would have created the same number of the fission-product nuclide of interest as was obtained at the end of the actual irradiation. Because some small variations in fission rates were observed during the exposures, the number of fissions was determined for each time interval of record (about every five minutes) assuming a constant fission rate during that time interval (i.e., a histogram representation). The number of fissions for a given time interval "decayed" to the end of the irradiation according to the decay constant of the fission product for which the calculation was being made. The result was the "effective" number of fissions for that particular fission product at the end of the irradiation for some particular time period during the irradiation. These were then summed to give the total "effective" number of fissions for the particular fission product. The lifetimes of the precursors (^{93}Nb , ^{132}Sb , and ^{97}Y) are all short and did not affect the computation.

Determining the efficiency of the thin-foil chamber was more of a challenge. We followed methods outlined in a recent paper by Grundl, Gilliam, Dudey, and Popek³³ ("Grundl") on the measurement of absolute fission rates. As mentioned above, the two related sources for correction are (a) loss of fragments remaining in the foil, C_a , and (b) estimation of low-pulse-height events resulting in pulse magnitudes less than

the electronic discriminator setting, C_b . The total correction C_o is given by

$$C_o = C_a + C_b. \quad (5.E.3)$$

Grundl gives for C_a and its uncertainty

$$C_a(\%) = t/2R \approx 0.65 t \quad (5.E.4)$$

$$\Delta C_a = 0.25 C_a, \text{ or } 0.35\% \text{ whichever is larger,} \quad (5.E.5)$$

where t is the thickness of the deposit (U_3O_8) in g/m^2 and R is the average fission-fragment range in the deposit material. For C_b and its uncertainty Grundl determines a value called "etz" which is the number of counts between two discriminator settings V_L and V_U divided by the total number of recorded counts above the V_L setting. For ^{235}U

$$V_U = 0.54 V_p \quad (5.E.6)$$

$$V_L = 0.5 V_U \quad (5.E.7)$$

where V_p is the peak channel of the fission-fragment spectrum. Grundl assumes C_b , the number of counts due to fission fragments between 0 and V_L , to be equal to "etz". V_p , V_U , and V_L are indicated for our spectrum in Fig. 24. Since our discriminator setting is between V_L and V_U our correction C_b will be different from Grundl's C_b by some multiplying constant. Grundl measured "etz" for a number of different thickness foils. A linear least-square fit to these data can be represented by

$$\text{etz}(\%) = 0.075 + 0.385 t \quad (5.E.8)$$

where t is the deposit thickness in g/m^2 . Grundl gives the uncertainty on estimated C_b as

$$\Delta C_b = 0.5 C_b . \quad (5.E.9)$$

Except for the small constant in Eq. (5.E.8) the total correction C_o is a multiple of the foil thickness t . From the data shown in Fig. 24 "etz" = 0.957%, and from a similar spectrum for the thick-foil chamber "etz" = 2.33%. The ratio of these values is assumed to be the ratio of the total correction C_o , that is,

$$R_c = C_o(\text{thick})/C_o(\text{thin}) = 2.33/0.957 = 2.43. \quad (5.E.10)$$

The goal of the thick-foil, thin-foil fission-chamber combination was to determine C_o for the thin-foil chamber. Then $K(E_Y, D)$ satisfies

$$n_f (1.0 + C_o) = K(E_Y, D) n_Y \quad (5.E.11)$$

where n_f is the measured number of fissions giving pulses above threshold and n_Y is the gamma-ray yield corrected for decay. For the thick-foil chamber

$$n_f(\text{thick}) (1.0 + R_c C_o) = K(E_Y, D) n_Y(\text{thick}) . \quad (5.E.12)$$

Taking the ratios of the last two equation results in

$$R_f \frac{(1.0 + R_c C_o)}{(1.0 + C_o)} = R_Y \quad (5.E.13)$$

where

$$R_f = n_f(\text{thick})/n_f(\text{thin}) \quad (5.E.14)$$

and

$$R_Y = n_Y(\text{thick})/n_Y(\text{thin}) \quad (5.E.15)$$

are experimentally determined ratios of n_f and n_Y for the two chambers.

Solving for C_o

$$C_o = \frac{R_Y - R_f}{R_f R_c - R_Y} \quad (5.E.16)$$

Clearly C_o is most dependent upon the difference $(R_Y - R_f)$. R_Y was determined by measuring the three fission-product gamma rays of interest at various distances D for each of the two detectors discussed above, first measuring one chamber and then immediately the other. The resulting yields were corrected for fission-product decay but not for a possible variation in the average position of the sources inside the chamber. Five pairs of measurements were made, yielding 15 ratios for the fission-product gamma rays. For the fission-product gamma rays

$$R_Y = 3.93 \pm 0.08 . \quad (5.E.17)$$

For comparison the 185-keV gamma ray of the decay of ^{235}U was also studied in the same manner, and the ratio determined for this gamma ray was

$$R_Y = 3.90 \pm 0.10 . \quad (5.E.18)$$

R_f was determined from the scaler information recorded during the irradiation. Although there was some variation during the exposure, the mean value for most of the irradiation was

$$R_f = 3.65 \pm 0.05 . \quad (5.E.19)$$

Using $R_Y = 3.93$ and $R_f = 3.65$ yield

$$C_o = 5.6 \pm 2.3\% \quad (5.E.20)$$

where the assigned uncertainty is based upon Grundl's estimates (Eqs.

5.E.5 and 5.E.9) and estimating $C_a \approx C_b \approx 0.5 C_o$.

This value is somewhat larger than expected from Grundl's formulas and the measured "etz" from the data shown in Fig. 24. For this "etz" and using Eq. (5.E.8) the thickness for our thin foil would be estimated as $\sim 2.3 \text{ g/m}^2$, which in turn should give C_a of $\sim 1.5\%$. With our discriminator setting $C_b \sim 1.5\%$ for a total C_o of $\sim 3.0\%$, smaller than that determined above. It is likely that the thin-foil ^{235}U deposits are neither uniform nor on a sufficiently smooth foil.⁴⁵ In this case the "etz" will be an underestimate of the correction for pulse heights less than V_L , and this possibility is suggested by the data shown in Fig. 24. The estimated foil thickness of 2.3 g/m^2 is larger than expected from the measured number of 185-keV gamma rays and foil size. These measurements indicate a foil thickness of $\sim 1.5 \text{ g/m}^2$, suggesting a pronounced lack of uniformity.

Although it would be of great interest to resolve these questions, the primary purpose of this experiment was accomplished, though not with the hoped-for accuracy. This was to determine the K-factors so as to get the number of fissions for the in-pile irradiations without having to rely upon cumulative fission yields, branching-ratio information, or detector efficiency calibration. K-factors were determined for three gamma rays at three distances; these are summarized in Table 4. Each assigned ΔK includes $\sim 1\%$ uncertainty associated with extraction of each gamma-ray yield from the raw data.

TABLE 4. $K(E_Y, D)$ for $E_Y = 140.5, 228.0$ and 657.8 keV
and for $D = 20, 30$, and 200 mm

E_Y (keV) =	140.5	228.0	657.8
Nuclide	^{99}Mo	^{132}Te	$^{97}\text{Zr}-^{97}\text{Nb}$
$D = 20$ mm	351.4	807	1446
$D = 30$ mm	547.7	1172	20100
$D = 200$ mm	1446	2062	32720

$\Delta K(E_Y, D) = 2.25\%$ for each of the 9 values, including 2% common to all values from the uncertainty in n_f , and $\sim 1\%$ (on the average) data extraction uncertainty for each gamma ray.

5.F. Intralaboratory Comparison-Sample Measurements

This experiment was initiated at Los Alamos Scientific Laboratory (LASL) on 8 November 1976 with the irradiation of a 63 mg sample enriched in ^{235}U for 29 sec, producing $\sim 10^{13}$ fissions in the sample. The sample was dissolved into one liter of solution and a bottle containing 1.00% of the original amount was shipped to ORNL, and an identical amount was shipped to Idaho National Engineering Laboratory (INEL). The 10 ml sample shipped to INEL was fabricated into a single sample and studied using Ge(Li) γ -ray assay.⁴⁶ Upon local receipt of our 10 ml sample two samples were prepared, one containing 1.00% and the other 5.00% of the amount received. Both of these samples were mounted on 20 g/m² polyethylene foils and dried and then covered with similar foils. These samples were counted using the Ge detectors as used for n_f determination. Analysis was carried out using both methods discussed above, viz. the "absolute" method discussed in Section 5.D and the "K-factor" method discussed in Section 5.E.

Gamma-ray counting was initiated 28 hr 58 min following irradiation, with an estimated uncertainty of 2 min. The last measurement was made 244 hr 50 min following irradiation, and then the 5% sample was shipped to INEL for further comparison. We quote the INEL report⁴⁶ on this comparison:

After ORNL had completed their counting, the source containing 5.00% of their sample was sent to the INEL. If the sample preparations were all correct, the relative activity of the original INEL sample to that of this sample should be 20.00. This ratio, as measured at the INEL, was 19.97 ± 0.14 . This indicates that the preparations were satisfactory.

A side experiment was carried out on the remaining solution to ascertain the concentration of ^{235}U . This was done by comparing delayed-neutron yields from small samples from the LASL solution with delayed-

neutron yields from known amounts of NBS-standard ^{235}U solutions. We obtained a value of 61.0 $\mu\text{g/ml}$ of ^{235}U (uncertainty < 1%) compared to the expected 63 $\mu\text{g/ml}$. This difference does not necessarily affect the comparison of n_f .

The data obtained for our n_f results and those received from LASL⁴⁷ and from INEL⁴⁶ are presented in Table 5. The results are in good agreement within assigned uncertainties.

TABLE 5. Results of Analysis of LASL ^{235}U
Sample of 8 November 1976

A. LASL Analysis; β rays; K-factor; assigned systematic uncertainty is 0.73%.

^{99}Mo	$1.180 \pm 0.002 \times 10^{10}$ fiss./ml
^{140}Ba	$1.171 \pm 0.006 \times 10^{10}$
Average	$1.179 \pm 0.009 \times 10^{10}$

B. INEL Analysis; γ rays; absolute; uncertainties for individual gamma rays include counting uncertainties, detector efficiency uncertainties and half-life and branching ratio uncertainties but not fission yield uncertainties; this last uncertainty is included only in the "final" average.

Isotope	Number of Fissions ($10^{10}/\text{ml}$)	
	ILRR yields	Meek and Rider yields
$^{95}\text{Zr-724}$	1.176 ± 0.020	1.186 ± 0.023
-756	1.165 ± 0.020	1.175 ± 0.023
^{103}Ru	1.189 ± 0.019	1.141 ± 0.022
$^{140}\text{Ba-}^{140}\text{La}$	1.171 ± 0.020	1.158 ± 0.020
Average	1.176 ± 0.010	1.164 ± 0.011
(Final Average)	1.176 ± 0.026	1.164 ± 0.022

C. ORNL Analysis; γ rays; absolute and K-factor; all assigned total uncertainties are $\pm 2.5\%$.

E_{γ}	140.5	228.0	657.9
Nuclide	^{99}Mo	^{132}Te	$^{97}\text{Zr-}^{97}\text{Nb}$
K-factor D = 20 mm	1.173×10^{10}	1.194×10^{10}	1.197×10^{10}
K-factor D = 30 mm	1.228×10^{10}	1.178×10^{10}	1.197×10^{10}
K-factor D = 200 mm	1.172×10^{10}	1.178×10^{10}	1.176×10^{10}
Absolute	1.193×10^{10}	1.197×10^{10}	1.190×10^{10}

TABLE 5 -continued-

K-factor uncertainties include 2% common to all nine values and ~ 1% unrelated data-reduction uncertainty for each gamma ray to obtain the K-factor (as presented in Table 4), and ~ 1% unrelated data-reduction uncertainty for each gamma ray to obtain the data in this table. Absolute uncertainties include uncertainties associated with fission yields (Table 3), branching ratios (Table 3), and detector efficiencies (Section 5.A and 5.B), plus the ~ 1% unrelated data-reduction uncertainty for each gamma ray to obtain the data in this table.

5.G. Final Determination of the Number of Fissions

The absolute method, as outlined in Section 5.D, was applied to all samples irradiated in-pile for beta- and gamma-ray spectral decay-power measurements, and these data were reported in a recent quarterly report⁴⁸ on this project. Chronologically, the fission chamber external-beam measurements were made after the release of this quarterly report. The primary objective of the fission chamber experiment was as a check on the absolute method. The most careful and complete comparison is with the LASL sample as recorded in Table 5. We did not apply the K-factors of Table 4 to all of the in-pile sample delayed-gamma data; selected calculations showed differences from the absolute method of precisely the amount expected.

There are several ways to analyze the K-factor data in Table 5. For example, the unweighted average of the 9 entries is 1.188×10^{10} with a 1 σ standard deviation of the 9 entries of 1.8×10^8 ($\sim 1.6\%$), both values influenced strongly by the $D = 30$ mm result for $E_\gamma = 140.5$ keV. Excepting that value leads to an average of 1.183×10^{10} with a 1 σ standard deviation of 1.1×10^8 ($\sim 1\%$). However, the results for $D = 200$ mm are more consistent among themselves, which may reflect the fact that an error in determination of D has a much smaller error propagation in the K-factor for $D = 200$ mm than for the closer distances. Averaging the 3 values for $D = 200$ mm gives 1.175×10^{10} , with an absolute uncertainty of $\sim 2.2\%$, made up primarily of 2% common to all K-factors, with all remaining uncertainties in K-factor data of Table 5 being associated with data reduction of individual gamma-ray datum.

The average of the three absolute results is 1.193×10^{10} fiss./ml. The uncertainty associated with this value must be computed from uncertainties in cumulative fission yields, detector efficiencies, branching ratios, and data reduction. Of these, the last two are independent, but there is some correlation for the detector-efficiency calibration, and some correlation in the fission yields. If these were completely uncorrelated, the uncertainty on the average of 1.193×10^{10} would be $\sim 1.5\%$; assuming complete correlation in detector efficiency results in an uncertainty of 2.2% . We assume the latter value, since any reduction in it due to less than complete correlation in detector efficiency may be compensated by some correlation in the cumulative fission yields.

The two methods of determining n_f for the LASL sample are comparable in uncertainty, so neither one is preferred over the other. Averaging the two values gives 1.184×10^{10} fiss./ml. Since each method used the same gamma-ray data obtained from measurements of the LASL sample, the overall uncertainty on the number 1.184×10^{10} is $\sim 1.7\%$. Of this uncertainty, we assign $\sim 1.4\%$ as the overall normalization uncertainty excluding the uncertainty associated with reduction of the gamma-ray data of the LASL sample.

To get the overall average uncertainty associated with the n_f determination for our samples we need to determine the overall data-reduction uncertainty associated with the Ge(Li) measurements discussed in Section 5.D. The data presented in Table 3 are a good representation of all of these data. As noted in footnote d the individual measurements have varying uncertainties. However, the totals are quite similar. The average is 23.71×10^8 with a standard deviation of 0.54% . This deviation

is larger than the overall statistical uncertainty of $< 0.4\%$ for each of the individual totals, indicating that the 0.54% deviation includes uncertainties other than statistical. It is not easy to separate all of the components of the uncertainty on the overall values of n_f for our six sets of data, but it is probable that the overall Δn_f lies between 1.4 and 1.7% , based upon the analysis of the LASL sample. The overall uncertainty of 1.5% on n_f was determined by combining the minimum 1.4% and the 0.5% associated with the average 1σ standard deviation for the data in Table 3; it is a reasonable assignment and is not overly conservative.

Thus, the normalization of the reported decay-power data in the quarterly report⁴⁸ mentioned above requires a change, by reducing all n_f determined by the absolute method, e.g. the data presented in Table 3, by 0.76% . To this must also be subtracted 0.33% due to an improvement in the value of the branching ratio of the 140.5 -keV gamma ray in ^{99}Mo . The decay-power data as reported in Ref. 48 were obtained using a branching ratio of 89.8% for this gamma ray. Subsequent careful analysis³⁹ resulted in $90.7 \pm 0.6\%$ for the branching ratio. This 1% change results in reducing the previous determinations of n_f by 0.33% . Therefore, all of the yield and energy-release data reported in Ref. 48 have been increased by 1.09% . The uncertainties given in Ref. 48 are also incorrect since these assumed a Δn_f of 2.5% .

6. DATA ANALYSIS

Having determined n_f for each sample, the next step was to prepare the accumulated beta- or gamma-ray data for unfolding. The data reduction for the beta-ray data differed from that for the gamma-ray data in two aspects: (a) for the beta-ray data there was subtraction of the magnet-up data from the magnet-down data which required knowledge of the relative values of n_f for each data set, and (b) the energy-binning of the beta-ray data was different from the energy-binning of the gamma-ray data. Otherwise the steps in data reduction were essentially the same for both beta- and gamma-ray measurements.

6.A. Data Manipulation to Prepare Data for Unfolding

Computer routines which were written for the pre-unfolding data reduction are listed in Appendices A to D; the uses of these routines are discussed in this section.

Most of the data for the final runs were transmitted directly from the PDP-15 Data Acquisition Computer to storage on disk at the PDP-10 with the PDP-15 acting as a remote terminal. For those data which were not directly transmitted, but instead were saved on DECTAPE, the routine FILEX was used. This routine is a general purpose routine written by Digital Equipment Corporation, and has the options of reading PDP-15 binary format and writing PDP-10 binary format on disk. Data transfer was effected as soon as practical so that the DECTAPES could be reused.

The first step was the transfer of data from binary to ASCII. The program GET2X (Appendix 6A) prepared an ASCII file from the PDP-10 binary file created by the direct transmission data dump. (A very similar routine GETT22 did the same for binary files created by FILEX.) These long ASCII files containing all stored information concerning a run were printed out, and the printed information was scanned for anything suspicious. Information was also obtained which was used to determine dead-time corrections to the data. This correction was determined for each spectrum by comparing the total number of counts, tagged plus not-tagged, with the number of trigger pulses (given by scaler no. 5 in Fig 16). The dead time was also estimated by multiplying the average count rate by the average analyzing time ($\sim 12.5 \mu\text{sec}$ for the ADC we used). The two methods gave nearly equal results for all spectra. A second binary file was also created which differed from the original binary file in that the second

file could be randomly accessed (rather than sequentially), and this feature greatly speeded up the process next described.

The addition of data from several equivalent runs was accomplished using the program URANM2 (Appendix B). For each time interval, all of the data--foreground and background--were corrected for dead time and relative numbers of fission, and then summed to obtain a single 512-channel pulse-height spectrum and associated statistical uncertainties. Gain shifts of $< 1\%$ were not corrected for. If some spectra had to be gain shifted, the code URANM2 performed this option at a sacrifice in running time.

Energy calibration of spectral distributions was determined during the measurement periods by obtaining data from known sources or from irradiation of nonfissile samples. For gamma rays these included gamma rays having energies of 0.20 MeV (from decay of ^{205}Hg), 0.51 MeV (^{22}Na), 0.84 MeV (^{54}Mn), 0.90 MeV (^{88}Y), 1.115 MeV (^{65}Zn), 1.62 MeV (^{20}F), 1.84 MeV (^{88}Y), 2.75 MeV (^{24}Na), and 6.13 MeV (^{16}N made by irradiation of a sample of oxalic acid). For beta rays, energy calibration for $E_\beta < 1$ MeV was determined using monoenergetic conversion electron sources having energies of 0.36 MeV (^{113}Sn), 0.62 MeV (^{137}Cs) and 0.98 MeV (^{207}Bi). For $E_\beta > 1$ MeV, end points of beta-ray distributions were used; these included $E_{\beta\text{max}} = 2.27$ MeV (^{90}Sr), 2.85 MeV (^{56}Mn), 4.92 MeV (^{38}Cl) and 5.42 MeV (^{20}F). For $E_\beta > 5.5$ MeV and for $E_\gamma > 6.2$ MeV, the energy calibrations were extrapolated linearly from the highest-energy measured points with an average slope determined from the lower-energy measured data. For $E_\beta < 0.36$ MeV and $E_\gamma < 0.20$ MeV, the energy calibrations were extrapolated linearly from the lowest-energy measured points to the measured zero-energy

channel. For both the beta-ray and the gamma-ray calibrations it was difficult to obtain sufficient data (because of the very short lifetimes of ^{20}F and ^{16}N) to get an accurate determination of end-point channel for $E_{\beta} = 5.42$ MeV or centroid for $E_{\gamma} = 6.13$ MeV, and these difficulties are reflected in uncertainty assignments in Section 6.G. For the beta-ray calibration the assigned uncertainties also included uncertainties in end-point beta-ray energies and difficulties in ascertaining end points. Uncertainty assignments to the gamma-ray calibration are not as large as for the beta-ray calibration, and were determined by variations in peak centroid determinations (e.g. the 1.84-MeV gamma ray was measured at least once a day) and by deviations of the positions of measured peaks corresponding to source-emitted gamma rays from the expected positions.

The next step was the binning of the pulse-height data using the program ANLYZB (Appendix C). For each time interval and gain setting, this routine accepted Energy-vs-Channel calibration points, and for the energy bins (specified for either the gamma-ray or the beta-ray unfolding) prepared a file suitable for unfolding. Nearly always, however, the data set for high-gain setting (range 0.05 to 2.0 MeV) were combined with the data set for the low-gain setting (range 0.2 to 8.0 MeV) using the program DATMIX (Appendix D). This very short routine combined the two binned-pulse-height data sets for each time interval by using the data from the high-gain setting for E_{β} (or E_{γ}) < 0.5 MeV, combining data for E_{β} (or E_{γ}) between 0.5 and 1.6 MeV (thus reducing statistical uncertainties) and using the data for E_{β} (or E_{γ}) > 1.6 MeV from the low-gain settings. These "combined" files were ready for unfolding at this point in the data preparation.

6.B. Gamma-Ray Response Matrix

Besides the appropriately binned data file, the unfolding routine requires two other files. One is called the window-function array, which for the present data were Gaussian distributions corresponding to an "ideal" response with a width, σ , determined from experimental responses. The other is the response matrix which contains the detector response for as many beta- or gamma-ray energies as desired. In this section the gamma-ray response matrix is discussed.

Basically, the method employed to obtain the gamma-ray response matrix was to measure⁴⁹ the response for as many gamma rays as could be obtained from one- or two-transition sources, then to obtain smoothed curves as a function of gamma-ray energy for the resolution and peak-to-total ratios. The total efficiency was computed using the measured solid angle (defined by the 76.2 mm collimator nearest the detector), attenuations in 1 m air and the Al cover, and total absorption in 127 mm NaI. We were unable to parametrize satisfactorily the shapes of the Compton distributions; however, the iodine x-ray escape⁵⁰ and the backscatter peak position and yield were estimated as smooth functions of E_γ .

An existing program⁵¹ was modified to compute the gamma-ray detector's response. The present program is listed in Appendix E, along with the input data used to compute the gamma-ray response matrix. For this detector and collimation a good representation of the resolution (or width σ) was given by

$$\sigma = 0.01 E_\gamma (1.3522 + 5.0636/\sqrt{E_\gamma})/2.35482 \quad (6.B.1)$$

where σ and E_γ are in MeV.

Table 6 has a listing of the isotopes used to get experimental gamma ray responses. Information on peak-to-total ratios for the full-energy peaks as well as the escape peaks were also determined from these spectra. A comparison of calculated full-energy peak efficiencies with experimental values is shown in Fig. 25. What remained were the shapes of the Compton distributions, and these were input as tables of numbers. The program first determines the full-energy and escape-peak responses and then adds the Compton shape interpolated from the tabular input data to obtain the total response. This method of determining the response matrix, rather than complete reliance on total interpolation between measured responses, has an advantage of reducing any uncertainties in unfolding due to inconsistencies of the response matrix.

The next step in determining the response matrix was to decide on the response bin structure. After some trial-and-error, the basic response matrix was chosen to be 176 comparison energies for (the same) 176 response energies, between 0.05 and 8 MeV. The energy intervals were chosen to give 3 to 4 bins per resolution width (as given by Eq. 6.B.1).

Ten of the computed 176 response arrays are shown in Fig. 26. The full-energy peak is the primary response for $E_\gamma < 3$ MeV; for $E_\gamma < 6$ MeV the full-energy peak response retains a plurality of the total response; and only for $E_\gamma > 6.5$ MeV is the full-energy peak weaker than the single-escape plus Compton-edge peak. This feature of the response matrix (the dominance of the full-energy peak) arose from choosing the collimation system to enhance the peak-to-total ratio at the expense of total efficiency.

TABLE 6. Gamma-ray Sources Used in Determining
Responses of Gamma-ray Detector

E_{γ} (MeV)	Isotope	Source ^a
0.060	²⁴¹ Am	A-S
0.088	¹⁰⁹ Cd	BNM
0.122	⁵⁷ Co	A-S
0.166	¹³⁹ Ce	A-S
0.279	²⁰³ Hg	A-S
0.511	²² Na	A-S
0.662	¹³⁷ Cs	A-S
0.834	⁵⁴ Mn	A-S
0.898	⁸⁸ Y	A-S
1.115	⁶⁵ Zn	BNM; ORNL
1.274	²² Na	A-S
1.332	⁶⁰ Co	A-S
1.369	²⁴ Na	²³ Na(n,γ)
1.524	⁴² K	⁴¹ K(n,γ)
1.779	²⁸ Al	²⁷ Al(n,γ)
1.835	⁸⁸ Y	A-S
2.614	²⁰⁸ Tl	ThC'' ORNL
2.754	²⁴ Na	²³ Na(n,γ)
3.103	³⁷ S	³⁶ S(n,γ)
4.434	¹² C*	Am-Be ORNL
6.130	¹⁶ O*	Cm- ¹³ C ORNL; ^b ¹⁶ O(n,p) ¹⁶ N ^c
6.9-7.1	¹⁶ O*	¹⁶ O(n,p) ¹⁶ N ^c

^a A-S for Amersham-Searle; BNM for Commissariat A L'Energie Atomique (France), Bureau National de Metrologie; ORNL for local fabrication; X(n,γ) for activation of non-fissile samples using same equipment and technique as used for ²³⁵U irradiation and count.

^b Ref. 52.

^c Sample in the form of oxalic acid, and using the fast neutrons present in a "thermal" spectrum.

Fig. 25. Gamma-ray Detector Full-energy Peak Efficiency. The solid curve indicates the values calculated by the response-matrix generating code. Solid points represent experimental values obtained from several of the sources listed in Table 6. The two open circles ($E_\gamma = 0.51$ and 3.1 MeV) represent experimental peak-to-total ratios multiplied by calculated total efficiency.

Fig. 26. Examples of Responses of the Gamma-ray Detector to Monoenergetic Gamma Rays.

6.C. Beta-Ray Detector Response Matrix

Obtaining a satisfactory beta-ray detector response matrix was considerably more difficult than obtaining the gamma-ray response matrix. This was because of the lack of monoenergetic electron sources to span the region of E_β up to 8 MeV. The only readily available sources of conversion electrons were ^{207}Bi (0.98 MeV), ^{137}Cs (0.62 MeV), ^{113}Sn (0.37 MeV), and ^{139}Ce (0.14 MeV). Spectra taken with these sources showed that most of the response was in the full-energy peak, and that this peak was nearly Gaussian for the central 80% of its area. The low-energy portion of the response, however, could not be obtained for the ^{207}Bi source because of weaker 0.57-MeV conversion electron emission, nor from the ^{137}Cs source because of the beta rays from the decay of ^{137}Cs . The spectrum obtained from the ^{113}Sn indicated a low-energy response of $\sim 10\%$ of the total yield observed. The spectrum obtained from the ^{139}Ce showed considerable attenuation and energy loss and indicated that for $E_\beta < 0.15$ MeV the data would be less reliable than for $E_\beta > 0.15$ MeV. The data from these sources provided information such as total geometric efficiency, Gaussian width, and peak-to-total ratios for $E_\beta < 1$ MeV.

To obtain information on the detector response for $E_\beta > 1$ MeV, we had to rely on beta decay distributions. One definitive measurement was made of the $E_{\beta\text{max}} \approx 2.99$ MeV decay of ^{144}Pr which is $\sim 97\%$ of the total beta emission in the ^{144}Pr decay. The raw data for $E_\beta > 1$ MeV were nearly correctly reproduced by a calculated spectrum (see Appendix H for details of the computer routine used for this calculation), except that the calculated spectrum was $\sim 10\%$ larger in absolute yield. The calculated absolute yield was determined by measuring the parent ^{144}Ce source

strength via the intensity of the 0.13-MeV gamma ray. The uncertainty in the gamma-ray measurement, including uncertainty in the branching ratio, meant that the 10% extra yield in the calculation was not definitive; what seemed valid was the expectation that the dominant response, at least for $E_\beta < 2.8$ MeV, was a full-energy Gaussian distribution, and that the total efficiency (in detected events per emitted beta) should be that determined for $E_\beta < 1$ MeV, or else demonstrated to the contrary.

We started with a Gaussian response, but then included effects which altered the response from the Gaussian. The average energy loss through the style no. 2 covers, ~ 8 mm of air, and the entrance window of the beta-ray detector (~ 80 g/m² including $\sim 10\%$ allowance for nonuniformity of foil thicknesses) was estimated from range-energy relations.³⁰ Attenuation of beta rays at the entrance foil was estimated by determining the maximum entrance angle for which there is no inscattering to compensate for out-scattering and assuming that the scattering is due to nuclear scattering of electrons. Contributions to the response assumed to be due to slit-scattering (primarily from the collimator nearest the detector) were estimated by assuming a 10% effect at 0.35 MeV decreasing to a 2% effect at 3.5 MeV. The 10% at 0.35 MeV was, as mentioned above, determined from measurements using ¹¹³Sn; the 2% at 3.5 MeV was determined by trial to produce a satisfactory comparison in shape to the ¹⁴⁶Pr spectrum, and to give a good yield result to a spectrum obtained from ⁹⁰Sr. This isotope has two distinct beta-ray groups: the decay of ⁹⁰Sr having $E_{\beta\text{max}} = 0.55$ MeV, and the decay of the daughter ⁹⁰Y having $E_{\beta\text{max}} = 2.27$ MeV. These transitions are highly forbidden, however, and difficult to calculate, so comparison with experiment is not conclusive. Another effect is that of

electrons scattered backwards out of the detector, which was calculated using a semiempirical formula given by Tabata;³¹ and a related effect is due to electrons scattering out of the NE-110 into the CaF_2 . The back-scattering alters responses primarily for $E_\beta < 1$ MeV, while the latter effect can occur only for $E_\beta > 2$ MeV because of the size of the solid-angle defining collimator.

To test the latter effect on high-energy beta transmission, a sample of Li_2CO_3 was irradiated for a few seconds and counting of the 13-MeV betas from decay of the 0.8-sec ^8Li was initiated as soon as possible. Data in the "untagged" spectrum were compared with those in the "tagged" region. There was a complete loss of "tagged" data for $E_\beta > 10$ MeV and no loss for $E_\beta < 4$ MeV. So an attenuation due to this effect was included in the response calculation based upon these ^8Li measurements.

All of these effects were included in a computer routine used to calculate the total response matrix. (This routine is given in Appendix F.) The width of the Gaussian distribution was calculated from

$$\sigma = 0.01 E_\beta \sqrt{25.0 + 90.0/E_\beta} / 2.35482 \quad (6.C.1)$$

where σ and E_β are in MeV which is a good representation to the peaks observed at 0.35, 0.62, and 0.98 MeV, but there is no experimental verification for $E_\beta > 1$ MeV. Some of the calculated responses are illustrated in Fig. 27. A total of 98 such responses were calculated and binned into 98 comparison-energy groups to obtain the beta-ray detector's response matrix.

These response functions may be compared with those obtained by Wohn et al.⁵³ for a 65 mm diam by 58 mm deep cylindrical Pilot B plastic detector

Fig. 27. Examples of Responses of the Beta-ray Detector to Monoenergetic Beta Rays.

having a 23 mm deep well in the front face of the detector for a source-to-detector distance of ~ 25 mm. Despite the geometrical differences between the two systems the responses are quite similar, being primarily a Gaussian peak and a low-energy tail. The peak-to-tail ratio as a function of E_β for the Wohn system is very similar to the present system; however, the tail of the Wohn system remains finite even for zero pulse height.

Several beta-decaying sources were made by activating nonfissile samples. Shown in Fig. 28 is the results from unfolding of data obtained for ^{56}Mn . An absolute intensity determination was made for the amount of ^{56}Mn made by counting the 0.84-MeV gamma ray¹³ to $\pm 2\frac{1}{2}\%$. The calculated spectra utilized the code discussed in Appendix H. The comparison is very favorable. Similar data were obtained and unfolded for ^{20}F beta decay, $E_{\beta\text{max}} = 5.42$ MeV and are shown in Fig. 29. The calculated spectrum compares very favorably for $E_\beta > 0.4$ MeV; for $E_\beta < 0.6$ MeV the unfolded experimental results do not decrease with decreasing energy but stay about constant and then rise at the lowest E_β . The excess low-energy yield is $\sim 6\%$ of the total; the excess low-energy energy release is $\sim 0.4\%$ of the total. A verifiable explanation of this behavior has not yet been determined; these low-energy data may be due to bremsstrahlung from high-energy beta rays striking the final collimator. If this explanation is correct, then this contribution should have very little effect ($< 0.5\%$) on beta-ray energy-release data for the ^{235}U sample, especially since the most important beta-ray energy region is $0.5 \leq E_\beta \leq 2.0$ MeV for the data we obtained and present in a later section in this report. Several other sources were studied, e.g. ^{137}Cs , ^{198}Au , ^{51}V , ^{38}Cl , and ^{28}Al . Our calculated spectra for $Z \leq 30$ compared reasonably well with the unfolded experimental results. For fission-product nuclides we compared with recent

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Fig. 28. Measured and Calculated ^{56}Mn Beta-ray Spectrum. The beta-ray branching ratios were obtained from Ref. 13. The raw data were unfolded using the detector response matrix shown in Fig. 27.

Fig. 29. Measured and Calculated ^{20}F Beta-ray Spectrum. The raw data were unfolded using the detector response matrix shown in Fig. 27. The calculated spectrum was normalized by calculating the number of ^{20}F created during the irradiation from the mass of the CF_2 sample, the thermal capture cross section, and the beam flux. The estimated uncertainty is $\sim 8\%$.

calculated results of England and Stamatelatos;⁵⁴ an example for ^{137}Cs is shown in Fig. 30. Comparisons of the unfolded beta-ray data with calculations for the other samples studied (^{51}V , ^{38}Cl , ^{28}Al , and ^{198}Au) were similar in quality to those shown in Figs. 28 and 29 for $E_\beta > 0.5 \times E_{\beta\text{max}}$ of the isotope studied. However, for $E_\beta < 0.5 \times E_{\beta\text{max}}$, the calculated $N(E_\beta)$ tended to be smaller than the experimental unfolded data; for some E_β the calculated $N(E_\beta)$ was as much as 30% smaller than experimental $N(E_\beta)$. The last satisfactory comparison was for the beta spectrum from ^{28}Al , with the largest differences being for $E_\beta < 0.8$ MeV. It was recognized that these differences could be due to sources external to the beta-ray response matrix. For example, in the case of the ^{38}Cl spectrum imprecise beta-ray branching ratios could be an explanation for the observed low-energy differences. It is clear that a better understanding, hence reduction of these discrepancies, would lessen the probability of presently unknown sources of error in the beta-ray response matrix being used. However, our present understanding of the detector response is sufficient to provide the necessary response matrix to unfold the beta-ray energy-release raw data.

The strength of this method of determining the response matrix lies in the fact that the detector response is primarily a Gaussian distribution centered at nearly the incident energy even for high energy beta rays. Hence, relatively large variations in the low-energy portion of the response will affect the unfolding calculation very little, particularly as long as the beta spectrum to be unfolded decreases with increasing beta-ray energy as is the situation for ^{235}U beta-ray energy release. The error on the integral number of betas detected should be quite small, approaching

Fig. 30. Measured and Calculated ^{137}Cs Beta-ray Spectrum. The raw data were unfolded using the detector response matrix shown in Fig. 27. The calculations are taken from England and Stamatelatos (Ref. 54) for beta-decay only. Also shown, however, is the peak due to conversion electrons in ^{137}Ba .

statistical counting uncertainties, since the integral efficiency for a given E_β is very nearly geometrical except for the smallest- and largest-energy beta rays. A somewhat larger error may occur for the integral energy-release result because of incorrect low-energy responses. An estimate of this error was made by assuming a change in the ratio of response yield in the full-energy peak to that in the low-energy portion of the present response, but keeping the shapes the same. Then reducing the full energy peak from the present response by 10% results in $\sim 4\%$ reduction in energy-release. The results shown in Fig. 28 support the belief that the ratio of peak response to low-energy response is not in error by as much as 5%, at least for $E_\beta < 2.5$ MeV. For $E_\beta > 3$ MeV the experimental data from the nonfissile samples cannot support assigning as much precision to the response as can be assigned to the response for $E_\beta < 2.5$ MeV.

The weakness of this method is in the lack of experimental verification for $E_\beta > 1$ MeV, except by inference from beta-decay spectral distributions and by comparison with the beta-ray responses of Wohn *et al.*⁵³ Consideration was given to utilizing a system to provide higher energy nearly monoenergetic electrons by using magnetic field selection of energy E_β from e.g. a ^{90}Sr source. The difficulty was that obtaining ΔE_β (using slits) comparable to σ (of Eq. 6.C.1) resulted in a very low counting rate, comparable to random background. A much more intense source of ^{90}Sr would have been an improvement but still the time required to make the system work properly and then to obtain sufficient data would have been prohibitive. Consideration was also given to performing some type of Monte Carlo transport calculations, and for this

purpose we obtained the code CYLTRAN.⁵⁵ A considerable amount of effort was required to adapt the code to our IE 360/91 computer and to get the test case to reproduce the correct results. From this experience we estimated the amount of effort required to obtain sufficient results for our system, and concluded that there was insufficient time to do this for this report.

Balancing the strength and the weakness of the method of determining the beta-ray detector's response, we have determined estimates of 1σ uncertainties for the unfolded ^{235}U beta-ray spectra. These are discussed more fully in Section 6.G.

6.D. The FERD Unfolding Routine

The FERD unfolding routine was developed about 15 years ago by R. Burrus⁵⁶ to unfold neutron-scattering data obtained with liquid scintillators. In the intervening years the computer code has been refined and expanded by various workers. Although a version known as COOLC is available for distribution by the Radiation Shielding Information Center,⁵⁷ the complete FERD code has never been documented. We have contracted with the original author to prepare a complete report describing the FERD code, and the first draft of the report is presently undergoing revision. It is, however, unlikely that the report on FERD will be generally available prior to publication of the present report. A good description of the mathematical foundation has been given by Rust.⁵⁸ We do not follow this description, however, as it is too detailed for this report. Instead we discuss in this section the basic aspects of the FERD code, recognizing that justification of its use and reliance on the results stems from many man years of experience with the code.³³

Let $X(E)$ be the unknown spectrum, either a continuous function as in the case of our beta-ray data, or else made up of very many (say, $\underline{n}$) nearly indistinguishable components as in the case of our gamma-ray data. These are the cases for which FERD is an optimal analysis code. We measure counts within a finite number of pulse height bins b_i when the spectrum is detected by the spectrometer, where $\underline{i}$ is generally much smaller than $\underline{n}$. For each $\underline{i}$ an instrument response function $A_i(E)$ is determined. The relationship of these quantities is given by $\underline{i}$ equations:

$$b_i = \int A_i(E) X(E) dE \quad (6.D.1)$$

The problem is to determine $X(E)$ and its uncertainty from the measured b_i , but this problem does not have a unique solution, since there are n (or more) unknowns but only i equations, and because instrument resolution makes almost meaningless the definition of X values spaced more closely than the resolution width. In addition, there is a measuring error associated with the spectrum b_i that is not explicitly shown in Eq. (6.D.1), but which must be considered in solving the problem.

Consider an arbitrary function $W_k(E)$ which might be defined as an "ideal" instrument response but can be any function chosen by the experimenter. (The index k is not restricted to i , for example.) Then we consider the following relationship:

$$P_k = \int W_k(E) X(E) dE. \quad (6.D.2)$$

P_k are a relatively few solutions which may satisfactorily represent $X(E)$, and it is the $P_k \pm \Delta P_k$ which we seek. If the $W_k(E)$ functions could be exactly expressed as linear combinations of the $A_i(E)$, that is if one could solve for U_{ki} and that

$$W_k(E) = \sum U_{ki} A_i(E), \quad (6.D.3)$$

then the solution for P_k could be obtained directly from the b_i as

$$P_k = \sum U_{ki} b_i \quad (6.D.4)$$

with readily propagated uncertainties. However, it is generally not possible to find such a U_{ki} matrix in Eq. (6.D.3). Instead one searches for bounds on $W_k(E)$ such that

$$W_k^{\text{low}}(E) \leq W_k(E) \leq W_k^{\text{high}}(E) \quad (6.D.5)$$

such that Eq. (6.D.3) holds exactly for $w_k^{high} \rightarrow u_{ki}^{high}$ and $w_k^{low} \rightarrow u_{ki}^{low}$. One may then compute two sets of permissible solutions p_k^{low} and p_k^{high} from Eq. (6.D.2) which are known to bracket the unknown $X(E)$. The "confidence interval" on $X(E)$ will be represented by the differences between p_k^{low} and p_k^{high} when the propagation of statistical uncertainties and response-matrix errors has been included. If a tight match is sought to minimize the differences between $w_k^{low}(E)$ and $w_k^{high}(E)$, large coefficients u_{ki} result such that some terms in Eq. (6.D.3) [hence (6.D.4)] have large positive values while others have large negative values. On the other hand, if too loose a match is sought in order to reduce the u_{ki} , the result is a smaller statistical contribution to the confidence interval at the expense of a larger confidence interval.

The key to the FERD technique is to minimize the total width of the confidence interval due to both sources of uncertainty, balancing one type against the other. Uncertainties in the b_i are included. The FERD code uses a complex inequality technique including the uncertainties Δb_i , but (usually) assuming that the response functions $A_i(E)$ are known quite accurately compared to b_i . This latter requirement is quite stringent and demands a careful determination of the $A_i(E)$. (Upper and lower response matrices can be entered.)

As will be observed in the discussion of the spectral data, the confidence interval for a given channel b_i may be quite large, and indeed the smallest individual confidence interval may be larger than the desired uncertainty on the integrated energy for that spectrum. However, this is because the "window functions" $w_k(E)$ were chosen to emphasize the raw data in the energy interval near the i^{th} channel; hence, the confidence interval may be very largely due to the statistical uncertainties associated with

the raw data in this energy interval. The statistical uncertainties associated with the total raw spectrum, however, are nearly negligible, being $< 1\%$ for all data presented in this report. We include two extra "window functions" $W_k(E)$, one ($W = 1$) to give a $P = \int X(E) dE$, and the other ($W = E$) to give a $P = \int E X(E) dE$. Each of these has a confidence interval which includes statistical uncertainties, but not uncertainties in energy calibration. A contribution to the total uncertainty in total energy due to uncertainty in energy calibration is later added quadratically to the FERD output results. (There is some propagation of uncertainty in energy calibration through the response matrix due to changes in response with energy but this contribution is small compared to the overall uncertainty in energy calibration.)

The unique capability of FERD is the inclusion of a "hard" component of the overall uncertainty to represent the difficulty in matching each window function with linear combinations of channel response functions. If one would ask $\int X(E) W(E) dE$ for a $W(E)$ much narrower than the inherent resolution, a very large output uncertainty would result. Conversely, if one utilizes very smooth window functions such as 1 and E, there is no fitting problem and the "unfolding" component of the corresponding uncertainty is also very small. For the results presented here the "unfolding" uncertainties, exclusive of any uncertainties propagated from the response matrices, amounted to a small fraction of a percent.

6.E. Final Steps in Differential and Integral Data Reduction

The output of the FERD unfolding is a spectrum of beta or gamma rays representing sufficiently the absolute total number $N(E)$ vs E that was measured for the particular t_{irrad} , t_{wait} , and t_{count} . The next step is to normalize these results to the number of fissions determined by methods discussed in Section 5. The PDP-10 computer routine HEAT4 is used for gamma-ray data (see Appendix G) and HEAT5 for beta-ray data. These codes prepare a graphic display of each spectrum either for hard-copy plots or oscilloscope display, and the important integrals are obtained. These include total integral yield and integral energy release, each integral obtained by two methods: (a) the two window functions $W = 1$ and $W = E$ discussed in the last section, and (b) direct integration of the spectral data. Values determined by the two methods were nearly identical (within $\pm 0.5\%$) and were averaged together. Uncertainties determined by method (a) were used since they were smaller than those determined by method (b) [they were smaller because method (b) assumed completely correlated errors for all data points in the integral].

In addition to the total integrals, the code calculated partial yield for $E_{\beta(\gamma)} < 0.28 \text{ MeV}$, partial energy for $E_{\beta(\gamma)} < 0.28 \text{ MeV}$, and approximate energy-release rates for an equivalent pulse of fissions at $t = T_{\text{wait}} + 0.5 \times (T_{\text{irrad}} + T_{\text{count}})$. Since it is the total energy that is desired, the other data represent a bonus provided by the chosen experimental method. In particular the spectral data will provide a more stringent test of the summation calculations than do the integral data.

No further analysis was performed on the spectra, but the integral data were corrected for several contributions which were not included in

the unfolding process. For both beta- and gamma-ray data yields for very low energies (i.e. for $E_\gamma < 0.05$ MeV and $E_\beta < 0.15$ MeV) were estimated and added to the integral yields already obtained. For the gamma-ray data we relied on preliminary measurements²² of data between 0.025 and 0.05 MeV, representing x-ray data from the heavy-mass fission products. The yields for these data varied between 4 and 7% of the yields for $E_\gamma > 0.05$ MeV, hence are not important in the total energy integral. For the beta-ray data the low-energy contribution was determined by plotting the spectrum $E_\beta \times N(E_\beta)$ vs E_β and then extrapolating the resulting data for $E_\beta < 0.16$ MeV. An example is shown in Fig. 31; these data are for the last counting-time interval for $t_{\text{irrad}} = 100$ sec, hence for the case having the largest low-energy extrapolation. The low-energy contribution thus obtained varied between 1 and 14% of the yield for $E_\beta > 0.16$ MeV. Our confidence in this procedure may be assessed by our assignment of 30% of the fraction of the yield added and for the total energy release 3% of the fraction of the yield added as the probable uncertainty for this contribution. For the gamma-ray data 40% of the fraction added is assigned as the probable uncertainty for the gamma-ray yield and 1.4% of the fraction added is assigned as the energy-release uncertainty.

The other important correction to the integral data is that due to loss of fission-gas products from the sample prior to measurement. This contribution is described in detail in the next section.

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Fig. 31. Present Beta-ray Data plotted as Yield x Energy vs. Energy to Show the Extrapolation at Low E_{β} in Order to Obtain the Integral Heat. For shorter waiting times the estimated heat for $E_{\beta} < 0.16$ MeV is a smaller fraction of the total integral than shown here.

6.F. Loss of Fission-Product Gases from Sample Containers

Early in the program an experiment was performed to test the use of the 140.5-keV gamma ray observed in the decay of ^{99}Mo as one of the planned methods of determining the number of fissions, n_f , created in the ^{235}U sample. A 1 μg sample of ^{235}U in the style no. 1 container was irradiated for 2 sec, allowed to cool for ~ 2 days, and then the fission-product gamma rays were counted overnight using a good-resolution Ge(Li) detector. For the same cooling and counting times, all expected gamma-ray energies and intensities were calculated using ORIGEN;⁶ these results were then modified by the detector's known peak resolution, efficiency, and energy calibration. A portion of these experimental and calculated data are shown in Fig. 32. An important result was the lack of agreement between calculation and experiment for the strong gamma rays due to decay of xenon isotopes,[†] as shown in Fig. 32 for the 81-keV gamma ray due to decay of ^{133}Xe . The experimental yield is $\sim 10\%$ of the calculated yield.

An estimate of the fission-product gas loss-rate from these data was made using the assumptions (a) that the loss-rate was proportional to the number of ^{133}Xe nuclei remaining in the sample, (b) that there was no loss of the parent ^{133}I other than radioactive decay to ^{133}Xe , (c) that the ^{133}Xe concentration was initially zero, and (d) that the calculated yield for the 81-keV gamma ray would have given essentially the correct result for no loss-rate of ^{133}Xe other than by radioactive

[†] Most of the observed disagreement was for gamma rays in the decay of ^{135}Xe , and was traced to incorrect branching ratios in the ENDF/B data file then used for ^{135}Xe .

Fig. 32. Portion of Ge(Li) Spectrum of Fission Product Decay ~ 60 Hours After Irradiation. The calculation was obtained using ORIGEN (Ref. 6) and smearing the results with the detector resolution function. However, the calculations do not have a "Compton continuum" which would have to be added to get a better comparison for the weaker gamma rays.

decay. Assumption (a) is mathematically equivalent to radioactive decay, that is the loss-rate from leakage is given by

$$\frac{dN_l}{dt} = -\lambda_l N \quad (6.F.1)$$

where λ_l may be taken as a loss-rate constant mathematically equivalent to the radioactive decay constant

$$\lambda = \frac{\ln 2}{t_{1/2}} \quad (6.F.2)$$

Assumption (b) seemed valid from comparison of ^{133}I data observed at higher energies; later it was checked very carefully as described later. Assumption (c) is the extreme situation and will result in the fastest loss-rate. The actual fraction of ^{133}Xe initially in the mass = 133 chain is $\sim 0.05\%$.¹⁸ Assumption (d) remains to be verified experimentally.

The total loss rate for the daughter isotope (^{133}Xe) is given by

$$\frac{dN_2}{dt} = \lambda_1 N_1(0) e^{-\lambda_1 t} - \lambda_2 N_2 \quad (6.F.3)$$

where the subscripts for the present case are: 1 = ^{133}I ($t_{1/2} = 20.8$ hr) and 2 = ^{133}Xe ($t_{1/2} = 5.29$ d) and $N_1(0)$ = number of ^{133}I at $t = 0$. Using assumption (c), $N_2(0) = 0$, so at some later time t ,

$$N_2 = \frac{\lambda_1}{\lambda_2 - \lambda_1} N_1(0) \left(e^{-\lambda_1 t} - e^{-\lambda_2 t} \right) \quad (6.F.4)$$

Using assumption (b)

$$\lambda_1 = \lambda(^{133}\text{I}) \quad (6.F.5)$$

For the ^{133}Xe , however,

$$\lambda_2 = \lambda(^{133}\text{Xe}) + \lambda_g \quad (6.F.6)$$

The change from t_1 to t_2 in the number of ^{133}Xe nuclei (i.e. the difference between the number created by decay of ^{133}I and the number lost plus decayed) is given by

$$\Delta N_2 = N_2(t_2) - N_2(t_1) = \frac{\lambda_1}{\lambda_2 - \lambda_1} N_1(0) \left(e^{-\lambda_1 t_1} - e^{-\lambda_2 t_1} - e^{-\lambda_1 t_2} + e^{-\lambda_2 t_2} \right) \quad (6.F.7)$$

The total number of ^{133}Xe nuclei lost plus decayed is

$$\Delta N_{\text{total}} = \Delta N_1 - \Delta N_2 \quad (6.F.8)$$

where ΔN_1 is the change in the parent ^{133}I concentration, and is always a positive number. The number of ^{133}Xe which decay is given by

$$\Delta N_{\text{observed}} = \frac{\lambda(^{133}\text{Xe})}{\lambda_2} \Delta N_{\text{total}} \quad (6.F.9)$$

In the absence of nonradioactive loss (i.e., $\lambda_g = 0$) the number of ^{133}Xe decays should have been

$$\Delta N \sim 0.070 N_1(0) . \quad (6.F.10)$$

(Experimental parameters, e.g. detection efficiency, branching ratios, etc., have been ignored for simplicity.) By assumption (d) actual number of decays observed was

$$\Delta N' \sim 0.0070 N_1(0) . \quad (6.F.11)$$

By trial-and-error, λ_g is determined from Eqs. (6.F.6), (6.F.7), (6.F.8), and (6.F.9), to be $\sim 0.091/\text{hr}$, corresponding to a loss-rate of 50% of ^{133}Xe in ~ 7.6 hr.

Another experiment was performed to try to verify this loss-rate. This experiment involved attaching a plastic cup very similar to the style no. 1 container to a short copper tubing. A standard pressure gauge was also attached to the middle, and a valve was attached to opposite end of the tubing. After the tube was satisfactorily leak checked it was filled with xenon gas to + 2 atmospheres pressure, and the valve shut off. The recorded pressure loss was $\leq 10\%$ in 4 days. Assuming that all of the loss was diffusion through the 500 g/cm² polyethylene container, the calculated loss-rate expected for fission-product xenon in the style no. 1 container was 50% in ≥ 20 hr. This measurement suggested that the gas loss observed in the results shown in Fig. 31 was due to several mechanisms, and that more careful measurements were required. Therefore, several experiments were performed to obtain more reliable information on fission-gas loss-rates. Study of possible loss rate in ⁸⁸Kr is reported in the next section, and for ¹³³I in the following section.

6.F.1. Determination of ^{88}Kr Loss-rate

A 1- μg sample of ^{235}U contained in the thin-window (or style 2) sample container was irradiated for 100 sec and then removed to the low-background Ge(Li) detector. After ~ 135 min the sample had cooled sufficiently so that the decay-product gamma emission could be measured at a source-to-detector distance of 50 mm with $< 5\%$ count-rate losses. Five partial spectra ($t_{\text{count}} = 4000$ sec) were obtained emphasizing the 2400-keV gamma-ray energy region. The detector resolution was sufficient to resolve the 2392-keV gamma ray due to decay of ^{88}Kr from the 2398-keV gamma ray due to decay of ^{142}La . The detector response was determined for the 2614-keV gamma ray due to decay of ^{208}Tl (the source was ^{228}Th). This response was nongaussian for the measurements performed. For each of the ^{235}U decay-product spectra, the doublet peak $E_{\gamma} \approx 2395$ keV was analyzed to obtain approximate yields for the two components. Figure 33(a) indicates the peak fitting estimates for the first spectrum. For each component the yield was plotted as a function of decay time, as shown in Fig. 33(b). The decay of the 2398-keV gamma ray is observed in this figure to be that expected for ^{142}La ; however, the 2392-keV gamma ray is observed to decay faster than expected for ^{88}Kr . The difference is assumed to be due to loss through the 50 g/m² covering of the style no. 2 containers which is mathematically equivalent to diffusion. That is, the loss-rate is proportional to the difference in concentration of ^{88}Kr molecules from inside the 50 g/m² covering to outside the covering.

A further assumption is that once the ^{88}Kr molecules escape the ^{235}U container, they dissipate quickly, so that no ^{88}Kr molecules remain outside of the container. Thus, if N is the number of ^{88}Kr molecules inside the container, the loss dN_{ℓ} is given by Eq. (6.F.12)

Krypton Fission Gas Loss Experiment

Fig. 33. ^{88}Kr Fission Gas Loss Experimental Results. (a) Ge(Li) gamma-ray pulse-height spectrum ~ 3 hours after irradiation showing the resolution of the ^{88}Kr data from the ^{142}La data; (b) yields vs. decay time for these two peaks.

$$dN_{\ell} = -\lambda_{\ell} N dt \quad (6.F.12)$$

where λ_{ℓ} is a function of krypton permeability and foil thickness and composition. It is mathematically equivalent to the decay constant $\lambda = \ln(2)/t_{1/2}$. Therefore, the total loss dN is given by

$$dN = -(\lambda_{\ell} + \lambda)N dt \quad (6.F.13)$$

since ^{88}Kr is monoatomic.

The problem is to determine λ_{ℓ} from the measurements shown in Fig. 33-b. Let ΔN be the total number of ^{88}Kr atoms lost through decay and "diffusion."

$$\Delta N = N \left[1 - e^{-(\lambda_{\ell} + \lambda)t} \right] \quad (6.F.14)$$

Note that not all of ΔN was observed as 2392-keV gamma rays, because of the assumed dispersal of the escaping ^{88}Kr to distances much greater than the 50-mm source-to-detector distance. Possibly some of the radiation from decay of escaped ^{88}Kr atoms was recorded by the detector. The assumption was made that the contribution to the measured gamma-ray spectra was a constant fraction of the total loss of ^{88}Kr . The measured yield Y_1 for the first spectrum is

$$Y_1 = k_1 \Delta N_1 = k_2 N_0 \left(1 - e^{-\lambda_{\text{tot}} t} \right) \quad (6.F.15)$$

where k_1 is the fraction of ΔN , for which gamma rays were observed, N_0 = number of ^{88}Kr in the sample 135 min after the irradiation, and

$$\lambda_{\text{tot}} = \lambda_{\ell} + \lambda \quad (6.F.16)$$

Then for the second spectrum

$$Y_2 = k_2 \Delta N_2 = k_2 N_1 \left(1 - e^{-\lambda_{\text{tot}} t} \right) \quad (6.F.17a)$$

$$= k_2 N_0 e^{-\lambda_{\text{tot}} t_d} \left(1 - e^{-\lambda_{\text{tot}} t} \right) \quad (6.F.17b)$$

$$= Y_1 e^{-\lambda_{\text{tot}} t_d} \quad (6.F.17c)$$

where t_d is the time between the beginning of the first spectrum and the beginning of the second spectrum, and $k_2 = k_1$ because of the assumption stated above. Thus, a value for λ_{tot} can be determined from the ratio of Y_1 to Y_2 ; similarly λ_{tot} can be determined from data of the last three spectra. These results are summarized in Table 7.

TABLE 7. Loss-rate of ^{86}Kr	
Ratio	$\lambda_t (\text{sec}^{-1})$
Y_1/Y_2	1.41 E-4
Y_2/Y_3	1.51 E-4
Y_3/Y_4	0.88 E-4
Y_4/Y_5	<u>1.45 E-4</u>
Average $1.31 \pm 0.29 \text{ E-4}$	

The loss-rate is determined from Eq. (6.F.15), where $\lambda = 0.690$ E-4/sec for ^{86}Kr . Hence

$$\lambda_{\ell} \approx 0.62 \pm 0.14 \text{ E-4/sec} \quad (6.F.18)$$

corresponding to a loss-rate of 50% in ≈ 3.1 hr. For calculations presented in Section 6.F.4, $\lambda_{\ell} = 0.65 \text{ E-4}$ (corresponding to a loss-rate of 50% in 3.0 hr) was used as being slightly more convenient and well within the uncertainties of this analysis.

6.F.2. Determination of ^{133}I Loss-Rate

A 1 μg sample of ^{235}U in the style no. 1 sample container was irradiated for 100 sec and then the fission-product gamma rays were counted using a 90 cm^3 Ge(Li) detector. The total counting time was 220,000 sec broken up into 33 separate intervals. The 530-keV gamma ray emanating from the 20.83-hr isotope of ^{133}I was studied; the results are shown in Fig. 33 where the experimental yield has been divided by the calculated yield. The measurements for $t > 10^5$ sec indicate a negligible loss-rate for the thick source holder. A similar experiment was carried out for a 1 μg ^{235}U sample in the style no. 2 sample containers; however, the counting time interval extended only to 150,000 sec. These results are also shown in Fig. 34. Both data sets curve upwards for short times, presumably due to contributions from other fission products. The two sets of data appear to be $\sim 3\%$ different in normalization, probably due to the fact that the measurements were separated by about 2 weeks, and required separate measurements of n_f . The dashed line is the same "shape" deduced from the solid triangle data modified for an assumed steady loss-rate of 50% in 1 days. The actual loss-rate appears to be $\leq 50\%$ and could be due to diffusion or leakage through a pinhole. It is clear that for purposes of beta- and/or gamma-ray energy-release measurements for $t < 10^5$ sec the loss-rate can be neglected. (Measurement of the yield of this gamma ray became part of the data-reduction procedure discussed in Section 5.D to ensure that there was no appreciable ^{133}I loss. None was observed.)

Fig. 34. ^{133}I Fission Gas Loss Experimental Results. These data suggest no measurable loss rate from the style number 1 holders, and a loss rate of 50% in ~ 90 days for the style number 2 holder.

6.F.3. Determination of Br Loss-Rate

Although the possible loss of fission-product Br isotopes should not affect the energy-release measurements very much (because the important Br isotopes have $t_{1/2} \leq 3$ min) an experiment was performed to ensure that the polyethylene material used as covers for the style no. 2 sample containers would contain radioactive Br isotopes in compound form. For this experiment a small sample of ammonium bromide was irradiated at the sample irradiation position, producing the 35.4-hr ^{82}Br . The sample was dissolved and transferred onto a piece of the 50 g/m² thick polyethylene. After drying, a second piece of polyethylene covered the sample. The decay of ^{82}Br was studied for three days with no measurable loss of ^{82}Br (i.e. < 2% in three days).

This result is necessary but not sufficient to assert that there is no loss of fission-product Br isotopes, since most of the ^{82}Br ions remain bound by an ammonium radical whereas fission-product Br isotopes are more likely to be free ions. It seems reasonable, however, to assume bromine ions will behave like iodine ions insofar as loss-rate affects energy-release measurements.

6.F.4. Estimation of Contributions to the Energy-Release Rates from the Loss of Fission-Product Kr and Xe Isotopes

The loss-rate determined for ^{86}Kr was assumed for loss-rates of the Xe isotopes, although systematic examination of Xe and Kr permeabilities through other types of materials suggests that the Xe should diffuse more slowly.⁵⁹ Prior to the analysis just presented, fractional losses of beta- and gamma-ray energy-release had already been computed⁶⁰ for a loss-rate of 2 hr and ^{235}U fission-product decay times between 750 and 14,400 sec. Similar calculations⁶⁰ were performed for an assumed Kr and Xe loss-rate of 20 hr. These data are summarized in Table 8.

The two loss-rates (2 hr and 20 hr) for calculation of the effect due to losses were chosen to be the probable bounds of the actual fission gas loss-rates which had not been determined at the time of the calculation. It is evident that for assumed loss-rate $T_{1/2}(\text{loss}) \geq 20$ hr, the total correction required is small and within the overall uncertainty of the data. For the rapid loss-rate, $T_{1/2}(\text{loss}) = 2$ hr, the correction required becomes appreciable and important only for cooling times ≥ 2000 sec; for most of the energy-release data reported herein, the correction is small and easily managed. Only for the last four time bins of the 100-sec irradiation to measure beta energy release is the correction large enough to warrant careful use of the data shown in Table 8.

The largest correction is for the last time interval for the beta-ray data. An example of the method used to obtain an estimate for this correction is now presented. The measured beta-ray energy release is given approximately as

$$E(t_1, t_2) = \sum_i N_i E_i \left(e^{-\lambda_i t_1} - e^{-\lambda_i t_2} \right) + \sum_j N_j E_j \frac{\lambda_j}{\lambda_j + \lambda_\ell} \left[e^{-(\lambda_j + \lambda_\ell) t_1} - e^{-(\lambda_j + \lambda_\ell) t_2} \right] \quad (6.F.18)$$

TABLE 8. Calculated^a Fractional Contribution to Total Energy Release Due to Loss of Fission Product Noble Gases. Tabulated values represent the fractional loss of the total energy-release rate due to loss of Kr and Xe isotopes.

T_{wait} (sec)	Assumed $T_{1/2}(\text{loss}) = 2 \text{ hr}$		Assumed $T_{1/2}(\text{loss}) = 20 \text{ hr}$	
	Beta	Gamma	Beta	Gamma
750	0.011	0.012	0.0013	0.0014
930	0.013	0.014		
1340	0.015	0.018		
1740	0.019	0.024	0.0022	0.0029
2190	0.023	0.028		
2690	0.031	0.034		
3410	0.040	0.041		
3950	0.048 ^b	0.046 ^b	0.0053	0.0042
6000	0.079	0.063	0.0093	0.0073
8000	0.105	0.077	0.0128	0.0098
10000	0.129	0.094	0.0169	0.0129
12000	0.148	0.111	0.0204	0.0158
14400	0.168 ^c	0.133 ^d	0.0242	0.020

^a These calculations are due to R. Schenter (Ref. 60)

^b Dominated by ^{138}Cs decay

^c Dominated by ^{88}Rb decay

^d Dominated by ^{86}Kr decay

where the subscript $\underline{j}$ represents all of the fission products Kr, Xe, and their daughters, and the subscript $\underline{i}$ represents all the other fission products. $\lambda_{\underline{j}}$ is the loss-rate constant, assumed to be the same value for all Kr and Xe isotopes. $N_{\underline{i}}$ (or $N_{\underline{j}}$) = no. of atoms at time $t = 0$ assuming very short lifetimes for the decay of the isotopes preceding $N_{\underline{i}}$ (or $N_{\underline{j}}$) in the mass chain. This assumption is not valid for $^{133}, ^{135}\text{Xe}$, but is valid for ^{86}Kr , and ^{86}Rb from ^{86}Kr contributes $\sim 50\%$ to the calculated loss in Table 8 for a 4-hr cooling time. $E_{\underline{i}}$ (or $E_{\underline{j}}$) = beta-ray energy release for the $\underline{i}$ (or $\underline{j}$) isotope.

Since the ^{86}Kr - ^{86}Rb decay chain contributes $\sim 50\%$ of the total loss at the longest cooling time, an approximation is made that the second term in Eq. (6.F.18) is

$$\approx 2 N_{\text{Kr}} E_{\text{Kr}} \frac{\lambda_{\text{Kr}}}{\lambda_{\text{Kr}} + \lambda_{\underline{j}}} \left[e^{-(\lambda_{\text{Kr}} + \lambda_{\underline{j}})t_1} - e^{-(\lambda_{\text{Kr}} - \lambda_{\underline{j}})t_2} \right] \quad (6.F.19)$$

where λ_{Kr} = decay constant for ^{86}Kr and E_{Kr} is the beta-ray energy release due mostly to decay of the daughter ^{86}Rb .

The following table is obtained using $\lambda_{\underline{j}} = 0.65 \text{ E-4/sec}$, $t_1 = 10000$ sec and $t_2 = 14000$ sec. Comparing the second column in Table 9 at

TABLE 9. Estimated Effect Due to Loss of Fission Gases for ^{86}Kr - ^{86}Rb Beta-Ray Energy Release for $t_{\text{wait}} = 10000$ sec, $t_{\text{count}} = 4000$ sec. T is fission-gas loss-rate time for 50% loss. "Fraction" is the ratio of that amount of energy release defined in Eq. (6.F.19) which would have been observed for $\lambda_{\underline{j}} = 0$ (i.e. no gas loss).

T (hr)	Fraction
∞	1.00
20.0	0.892
3.0	0.469
2.0	0.322

$T = 20$ hr and $T = 2$ hr with the entry for beta-ray energy release in Table 8 for $T_{\text{wait}} = 12000$ sec indicates that the $\bar{\epsilon}_j$ in Eq. (6.F.18) represents $\approx 20\%$ of $E(t_1, t_2)$. The estimated correction for fission gas loss-rate $\lambda_g = 0.65 \text{ E-4/sec}$ is ~ 0.106 for beta-ray energy release, or ~ 5 times the correction for λ_g corresponding to 20 hr loss-rate.

Similar calculations were used to obtain corrections to the measured beta-ray energy-release data for the three preceding time intervals. An uncertainty was assigned to each estimate corresponding to $\lambda_g = 0.1 \text{ E-4/sec}$, which corresponds to ~ 0.3 hr uncertainty (1 σ) to the loss-rate estimate in section 6.F.1.

For the gamma-ray energy-release data, estimations of the fractional corrections to the data were obtained in the same manner and based upon a fission-gas loss-rate of 50% in 10 ± 5 hr. This loss-rate is faster than expected for diffusion. The measurements for ^{86}Kr (Section 6.F.1) in the style no. 2 holder yield 50% loss in ~ 3 hrs; therefore the expected diffusion loss-rate for the style no. 1 holder ought to be 50% loss in ~ 30 hrs, since the cover is ~ 10 times thicker. The most likely explanation for the more rapid loss rate observed for the style no. 1 (i.e. 50% in 7.6 hrs for the sample discussed in Section 6.F) is leakage between the sample holder and its lid. This leakage could vary from sample to sample. For the gamma-ray data the NaI-measured spectra were a benefit, since the higher-energy gamma rays, $E_\gamma > 2 \text{ MeV}$, are due in large part to decay of the Kr and Xe isotopes and their daughters. The raw NaI-detector gamma-ray spectrum for each sample was scanned to ensure that there were approximately the expected number of large pulse-height counts. Data for two samples were discarded because there appeared to be an unusually small number of counts.

This preliminary scan was insufficient to detect $< 10\%$ loss for a given sample, because of statistical variations in integrated counts. It seems reasonable, however, to assume that the uncertainty in the estimated fractional loss is adequately represented by the 5 hr uncertainty in fission-gas loss-rate time.

6.G. Determination of Uncertainties

The uncertainties assigned to the data in the spectral distributions represent the 1 σ confidence interval determined by the FERD unfolding calculation. There were no corrections applied to the spectral data, and there were no further additions to the output confidence-interval limits. This lack of further correction to the confidence intervals means that the uncertainties assigned to the spectral data do not represent 1 σ limits but some limit $< 1\sigma$. However, it is likely that adding in the remaining uncertainties would result in very small changes to the confidence-interval limits, and would not be worth the considerable effort required to do this task.

The uncertainties (1 σ) assigned to each gamma-ray integral value were derived by combining assigned uncertainties to separate components as follows:

- (a) Fission number, n_f ; 1.5% as discussed in Section 5.G.
- (b) Dead-time corrections; 3% for short cooling times (where the count rate exceeded 10,000 pulses/second and the correction was $\sim 15\%$) to $< 1\%$ for long cooling times.
- (c) Detector response; 1% for primary E_γ between 0.5 and 2 MeV, which is the important region for integral energy-release data, to 5% for $E_\gamma > 4$ MeV.
- (d) Background subtraction; this correction in yield is taken as 20% of the fraction of the total spectrum due to background as measured by blank rabbits. For all of the 1-sec irradiation data and most of the 10- and 100-sec data the background was $< 5\%$ of the total spectrum; hence, the uncertainty is $< 1\%$ for

these data. For longer waiting times for 10- and 100-sec irradiations, the uncertainty increases to a maximum of 7% of net yield for last entry, where the background is $\sim 35\%$ of the total spectrum. The uncertainty for net energy release is 0.5 that for yield because the background is mostly < 0.2 MeV.

- (e) Counting statistics; insignificant for integral data, and as shown in the figures presented in Section 7 for differential spectra. The error bars in the figures are $> 90\%$ due to counting statistics, $< 10\%$ associated with the unfolding computation.
- (f) Energy gain calibration; 1% for E_Y between 0.5 and 2 MeV, 2% for E_Y between 2 and 4 MeV, 3% for E_Y between 0.1 and 0.5 MeV, and 5% to 10% for $E_Y < 0.1$ MeV and for $E_Y > 4$ MeV, contributing primarily to the energy release data.
- (g) Contribution for E_Y between 0.025 and 0.05 MeV: for the total yield 40% of fraction added; for the total energy release data 1.4% of the fraction added.
- (h) Fission-gas loss; 50% of the estimated fraction lost.

These uncertainties were quadratically combined for each irradiating time, cooling time, and counting time interval, arriving at 1σ uncertainties for total yield and total energy release for each integral datum. An example of this analysis is given in Table 10 for $T_{\text{irrad}} = 10$ sec.

The uncertainties (1σ) assigned to each beta-ray integral value were derived in a fashion similar to the gamma-ray uncertainties. The separate components are as follows:

- (a) Fission number, n_f ; 1.5% as discussed in Section 5.G.
- (b) Dead-time corrections; 3% at short cooling times, negligible for longer cooling times.

TABLE 10. Detailed Error Analysis for the Gamma-Ray
Data Set for $T_{\text{irrad}} = 10 \text{ sec}$

t_{wait} (sec)	t_{count} (sec)	Uncertainties ^a (in %) for					Total Yield ^c	Total Energy ^d
		Dead-Time Correction	Background Estimation	Fiss-Prod Gas-Loss Correction	Low-Energy Gamma-Ray Addition ^b			
10.7 ^e	6	2.0	0.3	-	2.88	4.10	3.28	
16.7	8	1.6	0.3	-	2.92	3.95	3.05	
24.7	10	1.1	0.4	-	2.84	3.73	2.83	
34.7	10	0.9	0.6	-	2.76	3.65	2.77	
44.7	10	0.7	0.7	-	2.60	3.50	2.71	
54.7	20	0.6	0.8	-	2.40	3.36	2.69	
75	20	0.4	1.0	-	2.16	3.21	2.68	
95	20	0.3	1.2	-	2.0	3.16	2.69	
115	40	0.2	1.4	-	2.0	3.23	2.70	
155	60	-	1.6	-	2.0	3.33	2.76	
215	80	-	2.1	-	2.0	3.60	2.81	
295	100	-	2.6	0.1	2.0	3.92	2.92	
395	200	-	3.3	0.5	2.0	4.43	3.23	
595	200	-	4.2	1.0	2.0	5.21	3.89	

^aAverage error (1 σ) contribution from uncertainties in:

Response Matrix 1.5%
 E_{γ} Calibration 1.5%
 No. of Fissions 1.5%

^bThis uncertainty is 40% of estimated photon yield for E_{γ} between 0.025 and 0.05 MeV, and applies to Total Yield. For Uncertainty to Total Energy multiply by 0.035.

^cIncludes Uncertainties in Response Matrix, No. of Fissions, Dead-Time, Fiss-Prod Gas-Loss, Low-Energy Addition, and Background Estimation.

^dIncludes Uncertainties in Response Matrix, E_{γ} Calibration, No. of Fissions, Dead-Time, 2 x Fiss-Prod Gas-Loss, 0.035 x Low-Energy Addition and 0.5 x Background Estimation.

^eIncluding 0.3-sec correction discussed in Section 3.A. for $t_{\text{wait}} < 70 \text{ sec}$.

- (c) Detector response; 2% for primary E_β between 0.3 and 1.5 MeV where the response is dominated by experimental responses measured using conversion-electron sources, 3% for $E_\beta < 0.3$ MeV and E_β between 1.5 and 3 MeV (because of the ^{56}Mn analysis shown in Fig. 28), and 4% and larger for $E_\beta > 3$ MeV.
- (d) Background subtraction; negligible for all data, even at long cooling times, since the background was determined to be constant with time unless there was a change in reactor operating power. The statistical errors associated with the subtraction of "magnet-up" data from "magnet-down" data where both data sets had background, were handled in the pre-FERD analysis, becoming part of the confidence interval of the FERD output.
- (e) Counting statistics; unlike the gamma-ray data the counting statistics do contribute some amount to the integral data because of the subtraction of the "magnet-up" data from the "magnet-down" data. These uncertainties vary between 1 and 3% for the present data, and are included in the confidence interval.
- (f) Energy-gain calibration; 1% for E_β between 0.3 and 1 MeV, increasing by about 1% for each 0.75 MeV above 1 MeV due mostly to the lack of well-defined end-points observed for beta-ray spectrum, but also partly due to uncertainties in the maximum beta energies for high-energy beta decay spectra.
- (g) Contribution for $E_\beta < 0.16$ MeV addition; for the total yield 30% of the fraction added, for the total energy release 3% of the fraction added.
- (h) Fission gas loss; 50% of the estimated fraction lost.

6.H. Summary of Data Reduction Procedures

As discussed in Section 6.A, the first step in data reduction is the summing of equivalent pulse-height data and the subtraction of background. The summing includes the dead-time correction, so that at this point two sources of uncertainty given e.g. in Table 10 are accounted for. The next step is to bin these data utilizing the measured energy calibration, and the overall uncertainty is now increased by the error in this calibration. The data are unfolded, and statistical errors and response-matrix consistency errors are added into the overall uncertainty. The unfolded spectra are divided by the number of fissions to yield normalized spectra, samples of which are presented in Section 7.A. Then the spectra are integrated for total yield and total energy, N_Y and E_Y (and similarly N_β and E_β). To these integrals are added the low-energy additions and the fission-product gas loss (if needed) to give the final corrected values reported in Section 7.B.

7. DATA PRESENTATION AND COMPARISON

The final output data are of two types: (a) differential energy spectra, i.e. $N(E_\beta)$ vs E_β and $N(E_\gamma)$ vs E_γ , and (b) the integral values

$$N_{\text{tot}} = \int_{E_{\text{min}}}^{8 \text{ MeV}} N(E) dE \quad (7.1)$$

$$E_{\text{tot}} = \int_{E_{\text{min}}}^{8 \text{ MeV}} E N(E) dE \quad (7.2)$$

for beta- and gamma-ray data separately. As mentioned in the last section, several corrections were applied to the integral data N_{tot} and E_{tot} which were not applied to the differential data. Thus the differential data, in addition to being of modest energy resolution, are not as accurate as the integral data. Both representations of the data should be very useful for comparison with results of summation calculations in aiding in locating those data in the basic data files which need improvement.[†] In addition, the integral data may be used in a direct determination of fission-product decay power, particularly in the time region of interest for a hypothetical Loss-of-Coolant Accident (≤ 15 min after shutdown).

[†] The first report²² of our preliminary gamma-ray data was instrumental in locating an error in the branching ratios of the decay of ^{98}Zr .

7.A. The Differential (Spectral) Data

A total of 86 differential data sets were measured, 43 for beta-ray energy release each containing 98 data, and 43 for gamma-ray energy release each containing 176 data. These data will be published later in a separate report.⁶¹ For the present report we chose to present 3 examples of our beta-ray data and 2 examples of our gamma-ray data.

The examples of our beta-ray data are shown in Figs. 35 to 37; these figures include results of earlier measurements²³ and very recent calculations.⁶² The earlier data were obtained for slightly different values for the time intervals (t_{irrad} , t_{wait} , and t_{count}) and the data were adjusted to be equivalent to those which would have been obtained had the time intervals been equivalent.[†] The calculations, on the other hand, are for our values for the time intervals. The calculated data include estimates for off-stability short-lived fission products for which there are no accurate experimental decay-scheme data. It may be that the comparisons in Figs. 35 and 36 will be improved when data for these short-lived nuclei become available. The comparison shown in Fig. 37 for $E_{\beta} > 1 \text{ MeV}$ is very good, as are other comparisons for $T_{\text{wait}} > 10^3 \text{ sec}$. Although it is tempting to suggest that the very good agreement just discussed validates the methods used to obtain the data, it is incorrect to do so.

The examples of our gamma-ray data are shown in Figs. 38 and 39; these figures include examples of earlier measurements^{24,25} and recent ORNL calculations. We have observed an improvement of the comparison of the calculations with our data with increasing T_{wait} , as expected.

[†] Using the appropriate "pulse" data of Tables I and II of Ref. 23, multiplied by the present t_{count} . Estimated error in our normalization of the data of Ref. 23 is $< 10\%$; note that our normalization is for graphical (comparison) purposes only.

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Fig. 35. Spectrum of Beta Rays Due to Thermal-neutron Fission of ^{235}U . The solid points are the data of Tsoulfanidis *et al.* (Ref. 23) and the calculations are the work of England and Stamatelatos (Ref. 62). The irradiation time, waiting time, and counting time intervals are given in the legend.

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Fig. 36. Spectrum of Beta Rays Due to Thermal-neutron Fission of ^{235}U . The solid points are the data of Tsoulfanidis *et al.* (Ref. 23) and the calculations are the work of England and Stamatelatos (Ref. 62). The irradiation time, waiting time, and counting time intervals are given in the legend.

Fig. 37. Spectrum of Beta Rays Due to Thermal-neutron Fission of ^{235}U . The solid points are the data of Tsoulfanidis *et al.* (Ref. 23) and the calculations are the work of England and Stamatelatos (Ref. 62). The irradiation time, waiting time, and counting time intervals are given in the legend.

Fig. 38. Spectrum of Gamma Rays Due to Thermal-neutron Fission of ^{235}U . The dashed line represents earlier ORNL data (Ref. 24) and the solid line represents data of Fisher and Engle (Ref. 25). The solid line is a calculated spectrum using ORIGEN (Ref. 6).

Fig. 39. Spectrum of Gamma Rays Due to Thermal-neutron Fission of ^{235}U . The histogram represents earlier ORNL data (Ref. 24) binned as shown by the authors of Ref. 2. The solid line is a calculated spectrum using ORIGEN (Ref. 6).

There is a difference, however, between the calculations for the gamma-ray comparison and those for the beta-ray comparison; the gamma-ray calculations contain data only for the ~ 180 fission-product nuclides for which there are experimental decay schemes known. One other aspect that has been observed with previous gamma-ray spectral data when compared to ours is that earlier measurements obtain somewhat larger values for $N(E_\gamma)$ for most E_γ . An attempt was made to determine if this difference was due to the fact that we used a magnet to deflect beta rays from the NaI detector, whereas the other experimenters used absorbers. The result was negative; both methods of beta-ray elimination gave the same results. At the present time there is no explanation for the differences. One may observe in Fig. 38 and 39 that the calculated spectra are closer to the lower yields as measured by the present system.

The gamma-ray spectra are very similar to those observed in our preliminary work,²² and the integrated yields and energies reported in the next section average within $\pm 2\%$ of the preliminary data. The same situation is not true for the final beta-ray measurements vis-a-vis the preliminary beta-ray measurements.²² There were improvements in the beta-ray detector's configuration following the preliminary measurements. The present beta-ray results supersede all prior reports of beta-ray data measured in this program.

7.B. The Integral Data

The differential data were integrated over beta- or gamma-ray energy, and the resulting integral data were corrected for low-energy contributions and, when needed, for fission-gas losses. The resulting 168 integral values and uncertainties are presented in Table 11 for the beta-ray data and in Table 12 for the gamma-ray data.

Summation calculations were performed^{6‡} to obtain comparison data for the energy integrals for our time intervals and these are shown also in Tables 11 and 12. The last columns indicate the ratios of experiment to calculation for the energy integrals.

The present data may be used to obtain the function $f(t)$ which is defined as the rate of energy release t seconds following an instantaneous pulse of fission, and has units of (MeV/sec)/fission. This function is easy to estimate from our integral energy data if the integrals are divided by the counting time, and the time t is taken to be

$$t = t_{\text{wait}} + 0.5 (t_{\text{irrad}} + t_{\text{count}}) \quad (7.B.1)$$

The difference between our estimate and an "exact" value is small if t_{wait} is $\gg (t_{\text{irrad}} + t_{\text{count}})$, becoming larger for short waiting times compared to irradiation and count times. The calculations given in Tables 11 and 12 were used to adjust the estimated $f(t)$ from our data resulting in better estimates of this function for short waiting times.

The function $f(t)$ decreases with t roughly as t^{-1} . Hence for presentation purposes it has become common practice to illustrate the "pulse" function as $t f(t)$. This presentation has the advantage of expressing

[‡] The decay energy release units are in MeV/f, that is, the integrated energy is divided by the total number of fissions. These units are not necessarily a contraction of $\frac{\text{MeV/s}}{\text{f/s}}$.

the $t f(t)$ axis on linear graphics, but the disadvantage of changing presentation because of an error in t . Acknowledging this defect, our beta-ray data are presented in Fig. 40 in this format for comparison with summation calculation⁶³ and with other data^{23,26,27} sets, particularly those for which the data were obtained in an integral format. The present data agree very well with the earlier data for t between 15 and 500 sec, but not so well with the calculation in this time interval. The present data are 5 to 7% high for t between 1000 and 5000 sec, and in reasonable agreement for longer times.

Our gamma-ray data are presented in similar format in Fig. 41. Our data are smaller than the other experimental data,^{24,25,28} although within assigned uncertainties of previous ORNL measurements.²⁴ The comparison of calculation⁶³ with the present data is good for t between 20 and 1000 sec and for $t > 5000$ sec. There is a disagreement for $t < 10$ sec which may be indicative of incomplete information in the basic data files. The disagreement for t between 1000 and 5000 sec is rather unexpected, and suggests that further study is needed in this time interval.

These figures (40 and 41) emphasize the differences between our data and calculated values for a pulse of fissions. The differences are much less obvious for extended irradiations.

TABLE 11. Beta-ray Energy Release and Yields from Fission Created by Thermal-Neutron Fission of ^{235}U

Irradiation Time (sec)	Waiting Time ^c (sec)	Counting Time ^d (sec)	"Average" Time ^e (sec)	Experiment ^a		Calculation ^b	Ratio Experiment/ Calculation
				Energy Release (MeV/fission)	Yield (Betas/fission)	Energy Release (MeV/fission)	
1.0 ± 0.1	1.7 ± 0.1	1	2.7	0.241 ± 0.020	0.138 ± 0.011	0.210	1.15
	2.7	1	3.7	0.194 ± 0.012	0.1020 ± 0.0061	0.168	1.16
	3.7	1	4.7	0.161 ± 0.009	0.0961 ± 0.0045	0.140	1.15
	4.7	2	6.2	0.254 ± 0.012	0.158 ± 0.007	0.226	1.12
	6.7	3	8.7	0.282 ± 0.013	0.181 ± 0.008	0.255	1.11
	9.7	5	12.7	0.305 ± 0.014	0.203 ± 0.008	0.305	1.00
	14.7	5	17.7	0.206 ± 0.009	0.152 ± 0.006	0.219	0.94
	19.7	5	22.7	0.155 ± 0.007	0.123 ± 0.005	0.169	0.92
	24.7	10	30.2	0.223 ± 0.009	0.190 ± 0.007	0.250	0.89
	34.7	10	40.2	0.159 ± 0.007	0.138 ± 0.006	0.183	0.87
	44.7	15	52.7	0.175 ± 0.007	0.163 ± 0.007	0.205	0.85
	59.7	15	67.7	0.131 ± 0.005	0.122 ± 0.005	0.155	0.85
	75	15	83	0.1042 ± 0.0040	0.0999 ± 0.0040	0.123	0.85
	90	20	100	0.1094 ± 0.0042	0.1078 ± 0.0045	0.131	0.84
10.0	10.7	6	18.7	0.246 ± 0.012	0.181 ± 0.009	0.255	0.96
	16.7	8	25.7	0.218 ± 0.011	0.170 ± 0.008	0.241	0.90
	24.7	10	34.7	0.192 ± 0.009	0.159 ± 0.007	0.216	0.89
	34.7	10	44.7	0.142 ± 0.006	0.122 ± 0.005	0.164	0.87
	44.7	10	54.7	0.114 ± 0.005	0.1028 ± 0.0042	0.131	0.87
	54.7	20	69.7	0.174 ± 0.008	0.157 ± 0.007	0.201	0.87
	75	20	90	0.129 ± 0.006	0.120 ± 0.005	0.149	0.87
	95	20	120	0.0971 ± 0.0038	0.0907 ± 0.0036	0.121	0.80
	115	40	140	0.147 ± 0.006	0.144 ± 0.006	0.175	0.84
	155	60	190	0.154 ± 0.006	0.155 ± 0.006	0.179	0.86
	215	80	260	0.135 ± 0.005	0.145 ± 0.006	0.162	0.83
	295	100	350	0.122 ± 0.005	0.134 ± 0.006	0.140	0.87
	395	200	500	0.170 ± 0.006	0.185 ± 0.008	0.187	0.91
	595	200	700	0.126 ± 0.005	0.145 ± 0.007	0.128	0.98

TABLE 11. Beta-ray Energy Release and Yields from Fission Created by Thermal-Neutron Fission of ^{235}U -- continued --

Irradiation Time (sec)	Waiting Time ^c (sec)	Counting Time ^d (sec)	"Average" Time ^e (sec)	Experiment ^a		Calculation ^b	Ratio Experiment/ Calculation
				Energy Release (MeV/fission)	Yield (Betas/fission)	Energy Release (MeV/fission)	
100.0	70	40	140	0.168 ± 0.007	0.164 ± 0.007	0.186	0.90
	110	60	190	0.166 ± 0.007	0.168 ± 0.007	0.184	0.90
	170	80	260	0.148 ± 0.006	0.155 ± 0.006	0.164	0.90
	250	100	350	0.129 ± 0.005	0.143 ± 0.006	0.142	0.91
	350	200	500	0.178 ± 0.007	0.206 ± 0.009	0.188	0.95
	550	200	700	0.128 ± 0.005	0.153 ± 0.007	0.128	1.00
	750	400	1000	0.183 ± 0.006	0.225 ± 0.010	0.176	1.04
	1150	400	1400	0.128 ± 0.005	0.159 ± 0.007	0.121	1.06
	1550	400	1800	0.0961 ± 0.0031	0.126 ± 0.006	0.0908	1.06
	1950	500	2250	0.0930 ± 0.0030	0.121 ± 0.006	0.0865	1.08
	2450	500	2750	0.0703 ± 0.0023	0.0960 ± 0.0046	0.0667	1.05
	2950	1000	3500	0.1020 ± 0.0037	0.139 ± 0.008	0.0967	1.05
	3950	2000	5000	0.123 ± 0.005	0.181 ± 0.010	0.120	1.03
	5950	4000	8000	0.129 ± 0.008	0.192 ± 0.014	0.131	0.99
	9950	4000	12000	0.0792 ± 0.0054	0.121 ± 0.012	0.0796	0.99

^a Measured values for $E_{\beta} \geq 0.16$ MeV plus estimated corrections for $E_{\beta} < 0.16$ MeV and for fission-product noble gas loss. Assigned uncertainties are absolute and represent 1σ .

^b B. I. Spinrad (Ref. 63): calculated results multiplied by counting-time interval.

^c Uncertainty in the initial waiting time for each irradiation period is 0.1 sec. Subsequent waiting times have a relative uncertainty of 0.0167 sec with respect to the initial waiting time. (Values rounded off to nearest second for times greater than 70 sec.)

^d Uncertainty on each counting-time interval is 0.0167 (i.e. 1/60) sec. This is not a random value, however, since time lost to one counting-time interval will be added either to the prior or to the subsequent interval.

^e $T_{\text{average}} = T_{\text{wait}} + 0.5 \times (T_{\text{irrad}} + T_{\text{count}})$.

TABLE 12. Gamma-ray Energy Release and Yields from Fission Created by Thermal-Neutron Fission of ^{235}U

Irradiation Time (sec)	Waiting Time ^c (sec)	Counting Time ^d (sec)	"Average" Time ^e (sec)	Experiment ^a		Calculation ^b	
				Energy Release (MeV/fission)	Yield (Gamma/fission)	Energy Release (MeV/fission)	Ratio Experiment/Calculation
1.0 ± 0.1	1.7 ± 0.1	1	2.7	0.188 ± 0.010	0.206 ± 0.012	0.150	1.25
	2.7	1	3.7	0.146 ± 0.007	0.164 ± 0.008	0.120	1.22
	3.7	1	4.7	0.119 ± 0.005	0.133 ± 0.006	0.0990	1.20
	4.7	2	6.2	0.188 ± 0.008	0.212 ± 0.009	0.160	1.18
	6.7	3	8.7	0.210 ± 0.008	0.237 ± 0.010	0.184	1.14
	9.7	5	12.7	0.248 ± 0.008	0.281 ± 0.011	0.227	1.09
	14.7	5	17.7	0.182 ± 0.006	0.209 ± 0.009	0.173	1.05
	19.7	5	22.7	0.143 ± 0.005	0.166 ± 0.008	0.139	1.03
	24.7	10	30.2	0.223 ± 0.007	0.253 ± 0.010	0.217	1.03
	34.7	10	40.2	0.170 ± 0.005	0.189 ± 0.008	0.166	1.02
	44.7	15	52.7	0.199 ± 0.006	0.216 ± 0.008	0.193	1.03
	59.7	15	67.7	0.154 ± 0.005	0.164 ± 0.006	0.150	1.03
	75	15	83.7	0.124 ± 0.004	0.128 ± 0.005	0.121	1.02
	90	20	100.7	0.131 ± 0.004	0.138 ± 0.005	0.130	1.00
10.0	10.7	6	13.7	0.216 ± 0.007	0.249 ± 0.010	0.202	1.07
	16.7	8	25.7	0.211 ± 0.007	0.241 ± 0.010	0.202	1.04
	24.7	10	34.7	0.178 ± 0.006	0.226 ± 0.009	0.192	1.03
	34.7	10	44.7	0.156 ± 0.005	0.171 ± 0.006	0.151	1.03
	44.7	10	54.7	0.127 ± 0.004	0.138 ± 0.005	0.124	1.02
	54.7	20	69.7	0.197 ± 0.006	0.211 ± 0.007	0.194	1.01
	74.7	20	89.7	0.149 ± 0.004	0.157 ± 0.005	0.147	1.01
	95	20	110	0.117 ± 0.004	0.122 ± 0.004	0.122	0.96
	115	40	140	0.175 ± 0.005	0.185 ± 0.006	0.176	1.00
	155	60	190	0.177 ± 0.005	0.191 ± 0.007	0.180	0.98
	215	80	260	0.158 ± 0.005	0.176 ± 0.007	0.161	0.98
	295	100	350	0.137 ± 0.004	0.161 ± 0.006	0.138	0.99
	395	200	500	0.186 ± 0.006	0.225 ± 0.010	0.186	1.00
	595	200	700	0.132 ± 0.005	0.166 ± 0.010	0.131	1.00

TABLE 12. Gamma-ray Energy Release and Yields from Fission Created by Thermal-Neutron Fission of ^{235}U -- continued --

Irradiation Time (sec)	Waiting Time ^c (sec)	Counting Time ^d (sec)	"Average" Time ^e (sec)	Experiment ^a		Calculation ^b	
				Energy Release (MeV/fission)	Yield (Gamma/fission)	Energy Release (MeV/fission)	Ratio Experiment/ Calculation
100.0	70	40	140	0.182 ± 0.008	0.186 ± 0.009	0.186	0.98
	110	60	190	0.181 ± 0.008	0.189 ± 0.009	0.184	0.98
	170	80	260	0.157 ± 0.007	0.171 ± 0.008	0.164	0.96
	250	100	350	0.134 ± 0.006	0.153 ± 0.007	0.140	0.96
	350	200	500	0.180 ± 0.007	0.216 ± 0.009	0.186	0.97
	550	200	700	0.125 ± 0.005	0.158 ± 0.007	0.131	0.95
	750	400	1000	0.177 ± 0.007	0.229 ± 0.010	0.190	0.93
	1150	400	1400	0.126 ± 0.005	0.167 ± 0.007	0.140	0.90
	1550	400	1800	0.0970 ± 0.0040	0.130 ± 0.005	0.112	0.87
	1950	500	2250	0.0953 ± 0.0041	0.129 ± 0.006	0.113	0.84
	2450	500	2750	0.0772 ± 0.0036	0.1035 ± 0.0049	0.0921	0.84
	2950	1000	3500	0.117 ± 0.006	0.157 ± 0.008	0.142	0.82
	3950	2000	5000	0.157 ± 0.010	0.206 ± 0.013	0.187	0.84
	5950	4000	8000	0.187 ± 0.015	0.236 ± 0.019	0.205	0.91
	9950	4000	12000	0.114 ± 0.013	0.139 ± 0.016	0.112	1.02

^a Measured values for $E_\gamma \geq 0.05$ MeV plus estimated corrections for $E_\gamma < 0.05$ MeV and for fission-product noble gas loss. Assigned uncertainties are absolute and represent 1 σ .

^b B. I. Spinrad (Ref. 63): calculated results multiplied by counting-time interval.

^c Uncertainty in the initial waiting time for each irradiation period is 0.1 sec. Subsequent waiting times have a relative uncertainty of 0.0167 sec with respect to the initial waiting time. (Values rounded off to nearest second for times greater than 70 sec.)

^d Uncertainty on each counting-time interval is 0.0167 (i.e. 1/60) sec. This is not a random value, however, since time lost to one counting-time interval will be added either to the prior or to the subsequent interval.

^e $T_{\text{average}} = T_{\text{wait}} + 0.5 \times (T_{\text{irrad}} + T_{\text{count}})$.

Fig. 40. Beta Energy Emission Rate Following an Instantaneous Pulse of Thermal-neutron Fissions of ^{235}U . The abscissa, t , is the time after a pulse of fissions. The ordinate is a quantity derived by obtaining $f(t)$ and then multiplying it by t . The units are a contraction of $\frac{\text{MeV/sec}}{\text{fission}} \times \text{sec}$. The solid circles represent the present data as described in the text. The open triangles are data of McNair *et al.*, (Ref. 26), the open squares of MacMahon *et al.*, (Ref. 27), and the open circles of Tsoulfanidis *et al.*, (Ref. 23). The calculation was carried out by R. Schenter (Hanford) using the RIBD code (Ref. 4) and the ENDF/B-IV data file (Ref. 12).

Fig. 41. Photon Energy Emission Rate for Thermal-neutron Fission of ^{235}U . The abscissa, t , is the time after a pulse of fissions. The ordinate is a quantity derived by obtaining $f(t)$ and then multiplying it by t . The units are a contraction of $\frac{\text{MeV/sec}}{\text{fission}} \times \text{sec}$. The solid circles represent the present data as described in the text. The open squares are the data of Peelle *et al.*, (Ref. 24), open circles represent data of Fisher and Engle, (Ref. 25), and the open triangles are data of Bunney and Sam (Ref. 28). The calculation was carried out by R. Schenter (Hanford) using the RIBD code (Ref. 4) and the ENDF/B-IV data file (Ref. 12).

8. CONCLUSIONS AND RECOMMENDATIONS

As mentioned in the Introduction an important goal in this experiment was to substantially reduce the uncertainties associated with fission-product decay power for thermal-neutron fission of ^{235}U , and this we have done. For most of the energy integrals presented in Tables 11 and 12 the uncertainties (1 σ) are less than 4%, with the largest uncertainties occurring for the longest t_{wait} periods which were not in the original scope of the program. A second important goal was to determine the adequacy of the present "ANS + 20%" standard, which as shown in Fig. 1, is conservative compared to decay-heat power derived from our results. It may have been noted that the 1973 ANS standard calls for an "infinite" operating history. To obtain decay-heat power in this format from our finite-irradiation integral results of Tables 11 and 12 requires mathematical manipulations discussed in Section 2. The method is presented in detail in the next section.

We have discussed several aspects of this study which were not as thoroughly studied as had been originally planned. Of these, probably the most important concerns the beta-ray detector's response matrix for $E_{\beta} > 1 \text{ MeV}$. It is clear that the essential correctness of the present matrix depends upon the assumptions that most of the response is in the full-energy peak and that the low-energy response is featureless and can be determined by an iterative search on beta-decay data such as those shown in Figs. 28 and 29. These assumptions ought to be checked either by measurement or by calculation for monoenergetic electrons. The measurement would be more definitive but much harder; the calculation isn't as definitive but can be performed using CYLTRAN⁵⁵ and we intend to study our particular configuration using CYLTRAN when time permits.

8.A. Total Energy-Release Results

The energy-release data in Table 11 are added to the equivalent data in Table 12 to obtain the total energy-release results. To compare our data with the current (1973) American Nuclear Society standard¹ requires deriving decay-heat power from our data and presenting it in the same format as the standard, viz. as a ratio of decay power to operating power. This is done in the following manner. $F(t_w, T)$ is referred to as the energy release per fission t_w sec following an operating period T sec, and is obtained from $f(t_w)$ as follows:

$$F(t_w, T) = \int_{t_w}^{t_w+T} f(t) dt \quad (\text{MeV/fission}) \quad (8.A.1)$$

An approximation to $F(t_w, T_1)$ may be obtained from Eq. (8.A.1) as follows:

$$F(t_w, T_1) = \int_{t_w}^{t_w+T_1} f(t) dt \approx T_1 f\left(t_w + \frac{T_1}{2}\right) \quad (8.A.2)$$

This approximation may be shown by expanding $f(t)$ in a Taylor's series about $t = t_w + \frac{T_1}{2}$:

$$\begin{aligned} f(t) &= f\left(t_w + \frac{T_1}{2}\right) + \left(t - t_w - \frac{T_1}{2}\right) f'\left(t_w + \frac{T_1}{2}\right) \\ &+ \frac{1}{2} \left(t - t_w + \frac{T_1}{2}\right)^2 f''\left(t_w + \frac{T_1}{2}\right) + \dots \end{aligned} \quad (8.A.3)$$

and integrating. The second term (also fourth, sixth, etc.) of this expansion yields a zero in the definite integral, and the integral becomes

$$\int_{t_w}^{t_w+T_1} f(t) dt = T_1 f\left(t_w + \frac{T_1}{2}\right) + \frac{1}{12} T_1^3 f''\left(t_w + \frac{T_1}{2}\right) + \dots \quad (8.A.4)$$

For $f(t) \sim t^{-1}$, $f''(t) \sim t^{-3}$, hence the second term of Eq. (8.A.4) is very small compared to the first term for $t_w > T_i$.

To relate our data, which are total energy measurements for given T_i , t_w , t_c , we observe that

$$E(T_i, t_w, t_c) = \frac{1}{T_i} \int_{t_w}^{t_w + t_c} F(t, T_i) dt. \quad (8.A.5)$$

Using the approximation given in Eq. (8.A.2) and then redefining the waiting time as

$$t = t_w + \frac{T_i}{2} \quad (8.A.6)$$

results in

$$E(T_i, t_w, t_c) = \int_t^{t+t_c} f(t') dt' \quad (8.A.7)$$

so that

$$E(T_i, t_w, t_c) = F(t, t_c). \quad (8.A.8)$$

This function $F(t, t_c)$ is related to the energy release per fission following an infinite irradiation:²

$$F(t, t_c) = F(t, \infty) - F(t + t_c, \infty). \quad (8.A.9)$$

Thus, the energy release per fission following an infinite period of operation is related to our measured values by iteratively applying the expression

$$F(t, \infty) = F(t, t_c) + F(t + t_c, \infty). \quad (8.A.10)$$

Here we obtain $F(t, t_c)$ from an experimentally measured value, i.e. from Eq. (8.A.8), and $F(t, \infty)$ from computed values for the energy release per fission for waiting times t in excess of the experimental measurements. Errors assigned to this procedure do not enter importantly into the uncertainties for $t_w < 1000$ sec. For the present experiment the maximum $t = 14000$ sec, and $F(14000, \infty) = 1.73$ MeV, or 0.86% of the operating power. To obtain $F(10000, \infty)$ from our data we add to $F(14000, \infty)$ the last entry in Table 11 and the last entry in Table 12. That is, we add our total energy-release data for the last 4000 sec of our measurement. Similarly to obtain $F(6000, \infty)$ we add our energy-release data for the next-to-last 4000 sec of our measurement to $F(10000, \infty)$. In this fashion our data are related to $F(t, \infty)$ for $2.2 \leq t \leq 14000$ sec. These results are given in Table 13 for the total power as well as for beta- and gamma-ray power separately. Average values were used for energy-release data measured for two different T_{irrad} (e.g. $T_{\text{irrad}} = 10$ sec, $T_{\text{wait}} = 215$ sec and $T_{\text{irrad}} = 100$ sec, $T_{\text{wait}} = 170$ sec). The most conservative uncertainty assigned to a $F(t, \infty)$ is obtained from the sum of uncertainties assigned to all energy-release data used to obtain $F(t, \infty)$, and these conservative uncertainties are given in Table 13. One may note, however, that some of the uncertainties of the measurement are not common to all of the data, and therefore the uncertainties assigned to the derived $F(t, \infty)$ should be 10 to 15% smaller than the most conservative approach. There is no uncertainty assigned to the $F(14000, \infty)$ although we recognize that this number has an associated uncertainty. Whatever this uncertainty is ($\sim 1.2\%$ according to Schmittroth⁶⁵) it must be added to the associated uncertainties shown in Table 13. The evaluation of $\Delta F(14000, \infty)$ is beyond the scope of this report; we note that it is small and may well be absorbed in the conservative uncertainties given in Table 13, at least for $t < 100$ sec.

TABLE 13. Fission Product Decay Power (MeV/fission)
Determined from Present Data

t (sec)	Beta-ray	Gamma-ray	Total
2.2	5.441	5.762	11.203 $\pm$ 0.404
3.2	5.200	5.556	10.756 $\pm$ 0.374
4.2	5.006	5.392	10.398 $\pm$ 0.355
5.2	4.845	5.259	10.104 $\pm$ 0.341
7.2	4.591	5.047	9.638 $\pm$ 0.321
10.2	4.309	4.810	9.119 $\pm$ 0.300
15.2	4.004	4.529	8.533 $\pm$ 0.278
20.2	3.798	4.320	8.118 $\pm$ 0.263
25.2	3.643	4.154	7.797 $\pm$ 0.251
35.2	3.420	3.900	7.320 $\pm$ 0.235
45.2	3.261	3.712	6.973 $\pm$ 0.223
60.2	3.086	3.496	6.582 $\pm$ 0.210
80	2.929	3.306	6.235 $\pm$ 0.197
100	2.800	3.149	5.949 $\pm$ 0.187
120	2.703	3.027	5.730 $\pm$ 0.179
160	2.556	2.842	5.398 $\pm$ 0.168
220	2.402	2.651	5.053 $\pm$ 0.157
300	2.260	2.493	4.753 $\pm$ 0.144
400	2.135	2.358	4.493 $\pm$ 0.133
600	1.961	2.175	4.136 $\pm$ 0.119
800	1.834	2.046	3.880 $\pm$ 0.109
1200	1.651	1.869	3.520 $\pm$ 0.096
1600	1.523	1.743	3.266 $\pm$ 0.086
2000	1.427	1.646	3.073 $\pm$ 0.079
2500	1.334	1.550	2.884 $\pm$ 0.072
3000	1.264	1.473	2.737 $\pm$ 0.066
4000	1.162	1.356	2.518 $\pm$ 0.056
6000	1.039	1.199	2.238 $\pm$ 0.041
10000	0.910	1.012	1.922 $\pm$ 0.018
14000 ^a	0.830	0.898	1.728 $\pm$ 0.000

^a Obtained from summation calculation using the ORIGEN code⁶ for $T_{\text{irrad}} = 10^{13}$ sec, no fuel depletion and no fission-product neutron capture. There is no assigned error to this value.

8.B. Final Remarks

The experimental contributions to the last column of Table 13 are 85% at 2.2 sec, 51% at 1200 sec and 10% at 10^4 sec. Thus, the present data provide a substantial base for analysis of hypothetical LOCA, as discussed in the Introduction. The total amount of fission-product beta-plus gamma-ray energy available ~ 1 msec after fission is ~ 12.7 MeV/fission.⁶⁶ From the results in Table 13, for example, 10 sec following shutdown of an "infinite" operation 72% of the total amount of fission-product energy is still available. This effect can occur only because most of the fission products contributing energy at $t = 10$ sec following shutdown were the products of fissions which occurred less than ~ 30 sec before shutdown. These are the short-lived isotopes for which there is little hard experimental information. Put another way $\sim 28\%$ of the energy release following shutdown of an "infinite" operation was due to fission products created in the last ~ 30 sec and was deposited in the reactor within 10 sec after shutdown. Thus, it is important in studying the consequences of a hypothetical LOCA to have an experimental measure of energy-release data for the first few hundred sec following shutdown due to fission products created in the last few hundred sec of operation, especially since these are the short-lived fission products lacking well-measured decay schemes.

The present experiment has provided data for these time regions, and they are more sensitive for comparison with calculations for time intervals within a few hundred sec before and after shutdown, than are data from experiments with much longer irradiation periods. In addition, the complete separation of beta rays from gamma rays and spectral distribution of each should provide even more information for improving the data bases used in the calculations.

As mentioned at the end of Section 2, the optimal result of study of our experimental results, as well as those of previous experiments²³⁻²⁸ and current experiments¹⁴⁻¹⁶ will be a calculational system capable of computing the decay heat correctly. Our experiment is suited for as much analysis as is desired to meet these objectives. Until such time as these objectives are realized, our data will be used to provide information on decay heat from thermal-neutron fission of ^{235}U . We anticipate that the gamma-ray spectral data will be used in understanding gamma-ray transport into and out of fuel-rod assemblies.

The experiment provided many more data than called for in the original planning stage. But this was because the preliminary results were somewhat unexpected, and additional data were required to ensure agreement with calculations based upon solid experimental radiochemical data. Clearly the resolution of the differences shown in Figs. 40 and 41 should have a high priority. In the meantime, we intend to obtain the fission-product decay-power from thermal-neutron fission by other fissionable nuclides occurring in fuel rods.

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APPENDIX A. DATA TRANSFER FROM SEQUENTIAL PACKED BINARY TO ASCII
(READABLE TEXT) AND TO RANDOM-ACCESS BINARY USING
THE PROGRAM GET2X

The purpose of this code is to transfer and unpack the data stored on disk using the direct (telephone line) dump of the data. Logical 23 is used for the input data, logical 26 is the random-access binary output file, and logical 22 is the ASCII file. Logical 5 is the teletype for both input and output.

A dead-time correction based upon an average Analog-to-Digital Converter conversion time of 12.5 μ sec is included in the teletype output. This value must be compared with estimated dead times based upon scaler readings for dead times $> 10\%$, since the calculated dead time based upon the 12.5 μ sec conversion time tends to underestimate the overall real dead time.

Two small routines, LS6 and BUST, were written in PDP-10 MACRO language.

```

COMMON IA(128),IZ(257),ITXT(9),ITIMS(18,2),ISC(18,12)
COMMON NBVLST(2,32),IDAT(1024),IRPUT(126),DTC('P')
COMMON NTBT,NRUN,N,IRRADT,NOVER,NREJCT,TYS
DOUBLE PRECISION FILNAM,NFL
DATA FILNAM/'RANDOM.ACS'/
DATA JS/126/
DATA A7/9/
DATA ITXT(1)/5H      /
WRITE (5,99)
99  FORMAT(' INPUT FILENAME = 'S)
READ (5,100) NFL
100  FORMAT (A10)
      OPEN(UNIT=23,FILE=NFL,ACCESS='SEQUIN',MODE='IMAGE')
      IV=1
      CALL DEFINE FILE(26,JS,NV,FILNAM,0,0)
      REAC (23) IA
      IZZ=0
C READ FIRST BLOCK OF INFORMATION
      IX = IA(1) .AND. =7777700000(
      IF (IX.EQ. 0) GO TO 333
      DO 334 J=1,128
334  IZ(J+1) = IA(J)
      CALL BUST (IA,IZ(2))
      IZZ=1
333  K=2
C FIRST 16 ARE TEXT STUFF
      DO 3 J=1,16,2
      ITXT(K)=LS6(IA(J))+IA(J+1)
      3  K=K+1
      NTBT=IA(18)
      IF (NTBT.GT.18) NTBT=18
      NRUN=IA(19)
      ITIMS(1,1)=IA(20)
      ITIMS(1,2)=IA(21)
      IRRADT=IA(22)
      TYS=FLRAT(IRRADT)/60.
C NEXT ARE SCALAR INFORMATION FOR SPECTRUM NO 1
      K=1
      DO 4 J=23,46,2
      ISC(1,K)=IA(J)+262144*IA(J+1)
      4  K=K+1
      NREJCT=IA(47)
      NOVER=IA(48)
      WRITE (5,199) NRUN,ITXT,NTBT,TYS,NREJCT,NOVER
C NEXT ARE OVERFLOWS, CHANNEL THEN NUMBER OF, IF NOVER > ZERO
      IF (NOVER.EQ.0) GO TO 2
      N=2*NOVER+48
      K=1
      DO 5 J=49,N,2
      NBVLST(1,K)=IA(J)
      NBVLST(2,K)=IA(J+1)
      5  K=K+1
      2  NLINS=NTBT+140
      WRITE (22,199) NRUN,ITXT,NTBT,TYS,NREJCT,NOVER
      IF (NOVER.EQ.0) GO TO 52
      WRITE (22,196)
      WRITE (22,198) ((NBVLST(J,K),J=1,2),K=1,NOVER)
      52  CONTINUE
      WRITE (5,209)
209  FORMAT(3X2HNS9X2H16X2H125X22HCH 0  CH 512  UNTAGGED7X
X      8H- YAGGED-)
C NEXT COMPUTE NUMBER OF REMAINING BLOCKS OF DATA TO PROCESS
      NREADS=NLINS/16
      K=1
      N=1
C LAST 16 IA=ARRAY ARE FIRST 16 CHANNELS OF DATA OF FIRST SPECTRUM
      IF (IZZ.NE. 0) GO TO 80
      DO 6 J=113,128
      IDAT(K)=IA(J)
      6  K=K+1
      IS=1

```

```

C ISW IS A SWITCH SINCE AT END OF EACH BLOCK OF INPUT DATA THE CODE
C MAY BE PROCESSING SCALERS, DATA, OR OTHER STUFF
      DO 20 KK=1,NREADS
      JMX=128
      IF (KK.EQ.NREADS) JMX=1024-KK+1
      READ (23) (IA(LL),LL=1,JMX)
      J=1
      GO TO (10,11,12,13), ISW
10      IDAT(K)=IA(J)
      J=J+1
      IF (J.GT.128) GO TO 10
      K=K+1
      IF (K.LE.1024) GO TO 10
      IF (N.GT.NTOT) GO TO 20
C IF IT GETS TO HERE A WHOLE SPECTRUM HAS BEEN READ IN
C SO DUMP IT INTO FILES 22 AND 26
      CALL DUMP
11      NZET=IA(J)
      IF (NZET.EQ.N+1) GO TO 9
      WRITE(5,401) NZET
      NTOT=N
      GO TO 7
      9      N=NZET
      IF (N.LE.NTOT) GO TO 21
      WRITE (5,401) N,J
401      FORMAT(' N= ',I7,', J= ',I3,', ERROR IN N= /')
C SO START THE NEXT DATA SET, THE N-TH SPECTRUM
21      J=J+3
      ITIMS(N,1)=IA(J)
      ITIMS(N,2)=IA(J+1)
      ISW=3
      K=1
      J=J+1
14      J=J+2
      IF (J.GT.128) GO TO 20
12      ISC(N,K)=IA(J)+262144*IA(J+1)
      K=K+1
      IF (K.LE.12) GO TO 14
      K=1
      J=J+2
      ISW=4
C NEXT 66 IA ARE REDUNDANT
13      JUNK=IA(J)
      K=K+1
      IF (K.GT.66) GO TO 17
      J=J+1
      IF (J.GT.128) GO TO 20
      GO TO 13
17      J=J+1
      IF (J.GT.128) GO TO 19
      K=1
      ISW=1
      GO TO 10
18      ISW=2
      K=K+1
      IF (K.GT.1024) GO TO 20
19      ISW=1
20      CONTINUE
C HOPEFULLY ALL DATA CORRECTLY READ IN
      7      CONTINUE
199      FORMAT(' RUN NUMBER= ',I6,', '9AS/
      X      ' 5X NUMBER OF SPECTRA= ',I7/
      X      ' 5X, IRRADIATION TIME= ',F10.2,' SECS'/
      X      ' 5X, TIME AT END OF RUN= ',I7 'SECS'/5X, 'NUMBER OF OVERFLOWS= ',I5)
196      FORMAT(' OVERFLOW LIST FOLLOWS. ')
198      FORMAT(2I10)
204      FORMAT(15,2I7,2I8,2F12.0,F8.2)
200      FORMAT(' SPECTRUM NO: ',I3,', DECAY TIME ',I7,' TO ',I7,' SECS'/
      X      ' 1 SCALERS ARE ',I2(I12/))
      END FILE 26

```

```

      WRITE (5,217) (I,DTC(I),I=1,NTOT)
217  FORMAT(16,' D.T.CORRECTION',F8.2,'X')
      DO 27 I=1,NTOT,3
        I1=I+1
        I2=I+2
      WRITE (5,207) I,NRUN,I1,NRUN,I2,NRUN
207  FORMAT(/3(14,18))
      IF (I2.GT.NTOT) I2=NTOT
      DO 27 J=1,12
        IF(J.EQ.5 .OR. J.EQ.9) WRITE(5,219)
219  FORMAT(1H )
        WRITE (5,208) (ISC(K,J),K=1,12)
208  FORMAT(3I12)
      END FILE 22
      WRITE (5,666) FILNAM
666  FORMAT(' FOR22.DAT FINISHED'/2X10,' ALSO')
      STOP
80  K=1
      JX=113
81  IDAT(K) = IA(JX)
      K=K+1
      JX=JX+1
      IF (JX .GT. 256) CALL INBUF(JX)
      IF (K .NE. 1025) GO TO 81
      CALL DUMP
      IF (N .EQ. NTOT) GO TO 7
      NZET= IA(JX)
      IF (NZET .NE. N+1) GO TO 11
82  N=NZET
      JX=JX+3
      ITIMS(N,1)=IA(JX)
      ITIMS(N,2)=IA(JX+1)
      JX=JX+3
      K=1
83  ISC(N,K) = IA(JX)+262144+IA(JX+1)
      JX=JX+2
      IF (JX .GT. 256) CALL INBUF(JX)
      K=K+1
      IF (K .LE. 12) GO TO 83
      K=1
      JX=JX+66
      IF (JX .LT. 256) GO TO 81
      JZ=JX+256
      CALL INBUF(JX)
      JX= JZ
      GO TO 81
      END

```



```

SUBROUTINE DUMP
COMMON I4(120),I2(257),ITXT(9),ITIMS(10,2),ISC(10,12)
COMMON NOVLST(2,32),IDAT(1024),IROUT(126),DTC(10)
COMMON NTBT,NRUN,M,IRRADT,NOVER,NREJCT,TTS
JS=126
IF (NOVER,EQ.0) GO TO 34
DO 33 NN=1,NOVER
KN=NOVLST(1,NN)
JJ=KN/1024 +1
IF (JJ.NE.N) GO TO 33
JAD=262144+NOVLST(2,NN)
J2=KN-1024*(JJ-1) +1
IDAT(J2)=IDAT(J2)+JAD
33 CONTINUE
34 SUM=0.
SUM1=0.0
CC=0.
DO 24 I=2,511
DD=IDAT(I+512)
SUM=SUM+DD
SUM1=SUM1+FLOAT(IDAT(I))
24 CC=CC+DD*FLOAT(I+1)
DTBT=SUM+SUM1
DTBT=DTBT+FLOAT(IDAT(1)+IDAT(513))
C TRY DEAD TIME CORRECTION FOR 12.5 MICROSEC CONV TIME
C IN ADC,... FEB 76
DTC(N)=100.*DTBT*12.5E-6/FLBAT(ITIMS(N,2)-ITIMS(N,1))
IF (SUM.LE.0.0) GO TO 26
CC=CC/SUM
26 WRITE (5,204) N,ITIMS(N,1),ITIMS(N,2),IDAT(1),
X IDAT(513),SUM1,SUM,CC
WRITE (22,200) N,ITIMS(N,1),ITIMS(N,2),(ISC(N,JB),JB=1,12)
DO 25 I=1,1024,8
JM=I+7
K=I-1
WRITE (22,202) K,(IDAT(JV),JV=I,JM)
202 FORMAT(10,2X,0I10)
25 CONTINUE
DO 40 I=1,1024,JS
KB=I
NK=1
KS=I+JS
IF (KS.GT.1024) KS=1024
DO 42 K2=1,JS
IROUT(K2)=0
42 DO 41 K2=KB,KS
IROUT(NK)=IDAT(K2)
41 NK=NK+1
IF (KS.LT.1024) GO TO 47
IROUT(17)=NRUN
IROUT(18)=IRRADT
IROUT(19)=ITIMS(N,1)
IROUT(20)=ITIMS(N,2)
IROUT(40)=NTBT
JQ=21
DO 45 JP=1,12
IROUT(JQ)=ISC(N,JP)
45 JQ=JQ+1
47 WRITE (26,1V) (IROUT(K2),K2=1,JS)
40 IV=IV+1
204 FORMAT(15,2I7,2I8,2F12.0,F8.2)
200 FORMAT(' SPECTRUM NO',I3,',', ' DECAY TIME',I7,',', ' TO',I7,',', ' SECS',/
X ' SCALERS ARE',I2(12/))
RETURN
END

```

```

SUBROUTINE INBUF(JX)
COMMON IA(128),IZ(257),ITXT(9),ITIMS(18,2),ISC(18,12)
COMMON NOVLST(2,32),IDAT(1024),IROUT(126),DTC(18)
COMMON NTST,NRUN,N,IRRADT,NPVEN,NREJCT,TTS
DIMENSION IAA(128)
EQUIVALENCE (IZ(2),IAA)
READ (23,END=1) IAA
1  CALL BUST(IA,IAA)
   JX=1
   RETURN
END

```

	TITLE	LS6
	ENTRY	LS6,BUST
LS6:	MOVE	0,00(16)
	LSH	0,22
	POPJ	17,
BUST:	MOVEI	6,400
	MOVE	4,(PRINT 18,0)
	MOVE	5,(PRINT 36,0)
	MOVE	3,0(16)
	HRRM	3,5
	MOVE	3,1(16)
	HRRM	3,4
BUSTA:	ILDB	3,4
	IDPB	3,5
	SOJN	6,BUSTA
	POPJ	17,
	END	

APPENDIX B. SUMMING OF DATA FROM EQUIVALENT RUNS USING URANM2

This program is used to read selected data from the random-access binary files created by GET2X, to multiply each data set by an input constant (which may be negative) and then summing the data sets and computing the statistical uncertainty for each summed datum. The program can also gain-shift a data set prior to summation; in practice gain shifts of $< 1\%$ were ignored to save computing time.

The subroutine GAINSH (page 180) uses a simple algorithm to perform the task of providing a new spectrum from the input data set according to the new calibration parameters which maintains conservation of total yield. The algorithm is first to obtain a running sum of the input spectrum, then to determine the partial running sum for each new calibration point by interpolation, and finally to subtract the n^{th} partial running sum from the $(n + 1)^{\text{th}}$ partial running sum to get the new datum for the new n^{th} channel.

Subroutine INTERP is found on page 195.

```

C THIS IS FILE URA:M2.F4
C PROGRAM TO SUM DATA FOR DECAY-HEAT EXPERIMENT
  DIMENSION IDAT1(1134),A(512),DA(512),EB(2),FILNM1(25)
  DIMENSION AMULT(25),FA(512),C(25),DC(25)
  COMMON /QS/ FILNAM,XMA,NE
  DOUBLE PRECISION FILNM1,FILNAM
  DATA NE/512/
  DATA ZC/6.0/
C      ZC=C*CHANNEL "ZERO" ACCORDING TO TAL
C
  CALL @FILE(20,'@SUMT')
  DO 1 I=1,NE
    A(I)=0.
    DA(I)=0.
1    FA(I)=0.
  WRITE (5,100)
100  FORMAT(' TYPE SPECTRUM NE & NE.BF FILES (2)')
  READ (5,101) N1,NF
101  FORMAT(2I)
  WRITE (5,102) NF
102  FORMAT(' TYPE '12,' N=FACTORS, C, DELTA-C & FILENAMES '
  X      'IN (3F,A10)')
  READ (5,103) (AMULT(I),C(I),DC(I),FILNM1(I),I=1,NF)
103  FORMAT(3F,A10)
  DO 3 I=1,NF
    FILNAM=FILNM1(I)
    CALL GETDAT(FILNAM,N1,IDAT1)
    IF (I.GT.1) GO TO 4
    EB(1)=FLOAT(IDAT1(1027))
    EB(2)=FLOAT(IDAT1(1028))
    XMA=FLOAT(IDAT1(1026))
    NRUN=IDAT1(1025)
4    IF (ABS(DC(I)).LE.0.0001) GO TO 2
    CALL SHFT(NE,C(I),DC(I),ZC,IDAT1)
2    DO 3 J=1,NE
      D=FLOAT(IDAT1(J))
      A(J)=A(J)+AMULT(I)*D
      DA(J)=DA(J)+D
      FA(J)=FA(J)+ABS(AMULT(I)*D)
      IF (I.LT.NF) GO TO 3
      IF (DA(J).LT.1.0) DA(J)=1.0
      DA(J)=FA(J)/SORT(DA(J))
3    CONTINUE
    CALL @SUMTF(A,DA,EB,NRUN)
    WRITE (5,93)
93  FORMAT(' FINISHED @SUMT.DAT FILE')
  END FILE 20
  STOP
  END

```

```

  SUBROUTINE @SUMTF(A,DA,EB,NE)
  DIMENSION A(1),DA(1),EB(2)
  COMMON /QS/ IDA,XMA,NPHS
  DOUBLE PRECISION IDA
C WRITES ON DISK 20 DATA IN SUMTF OUTPUT STYLE
  WRITE (20,60) IDA,XMA,NPHS,NE
60  FORMAT(A10,E15.5,2I10)
  IF (EB(1).GE.9999.9) EB(1)=9999.9
  IF (EB(2).GE.9999.9) EB(2)=9999.9
  WRITE (20,49) EB
49  FORMAT(2E15.5)
  WRITE (20,50) (A(K),K=1,NPHS)
50  FORMAT(10F8.1)
  WRITE (20,50) (DA(K),K=1,NPHS)
  RETURN
  END

```

```

SUBROUTINE SHFT(NE,C,DC,CZERO,IDAT)
COMMON /AA/ NN,DA(512),SDA(512)
DIMENSION IDAT(1)
NN=NE
E1=0.
E2=C-CZERO
EE=EGAM2(-1,1.,CZERO,E1,C,E2)
DO 1 I=1,NE
1  DA(I)=FLOAT(IDAT(I))
   C1=C-DC
   CALL GAINSH(CZERO,E1,C1,E2)
   IF (DC.GT.0.0) GO TO 2
   NN=FIX(0.5*FLOAT(NE)+C1/C)
   IF (NN.GT.NE) NN=NE
2  P=0.
   DO 3 I=1,NN
   IDAT(I)=FIX(SDA(I)+P)
3  P=SDA(I)+P-FLOAT(IDAT(I))
   IF (NN.GE.NE) RETURN
   NN=NN+1
   DO 4 I=NN,NE
4  IDAT(I)=0
   RETURN
END

SUBROUTINE GAINSH(C1,E1,C2,E2)
C      GAIN-SHIFT ROUTINE FROM CURRENT ENERGY CALIBRATION
C      TO THAT COMPUTED FOR C1,E1,C2,E2
COMMON /AA/ NE,PXSEC(512),DATA(512)
DIMENSION DAT(512),DB(512),EA(512)
DATA NMAX/512/
DAT(1)=PXSEC(1)
DO 1 I=2,NE
1  I=I
   DAT(I)=DAT(I-1)+PXSEC(I)
   IF (NE.GE.NMAX) GO TO 3
   DO 2 J=I,NMAX
2  DAT(J)=DAT(NE)
3  DO 4 I=1,NMAX
3  CI=FLOAT(I-1)
   DB(I)=CI
   DATA(I)=0.
   EA(I)=EGAM2(2,CI,0.,0.,0.,0.)
4  CONTINUE
CHANGE ENERGY CALIBRATION
EE=EGAM2(-1,1.,C1,C2,E2)
A=FLOAT(NE)
DO 5 I=1,NMAX
5  I=I
   CI=FLOAT(I-1)
   E=EGAM2(2,CI,0.,0.,0.,0.)
   CALL INTERP(EA,DB,NMAX,3,E,C)
   IF (C.LT.0.) GO TO 5
   IF (C.GT.A) GO TO 12
   CALL INTERP(DB,DAT,NMAX,5,C,X)
   DATA(I)=X
5  CONTINUE
   GO TO 6
12 DO 8 J=I,NMAX
8  DATA(J)=DATA(I-1)
6  J=NMAX-1
   NN=J
   DO 9 I=2,NN
   D=DATA(J)-DATA(J-1)
   IF (D.LT.0.0) D=0.
   DATA(J)=D
9  J=J-1
   D=DATA(1)
   IF (D.LT.0.0) D=0.0
   DATA(1)=D
   RETURN
END

```

```

      FUNCTION EGAM2(N,CH,X1,Y1,X2,Y2)
CC      USE ENERGY=A+B*CH TO COMPUTE GAMMA RAY ENERGY
CC      FOR N=1 COMPUTE COEFFICIENTS A, B, GIVEN X1,Y1, ETC.
CC      2 COMPUTE ENERGY AT CHANNEL CH
CC      3 COMPUTE CHANNEL FOR ENERGY CH
      DATA A/-6.0/
      DATA B/0.0/
      N1=IABS(N)
      IF (N=2) 1,2,3
CCCC      CALCULATE ENERGY FOR INPUT CHANNEL (=CH ARGUMENT)
      2 EGAM2=A+B*CH
      RETURN
CCCC      CALCULATE COEFFICIENTS
      1 B=(Y2-Y1)/(X2-X1)
      A=Y2-B*X2
      EGAM2=0.
      IF (N.GT.0) WRITE(5,99) A,B,X1,X2,Y1,Y2
99 FORMAT(/16X35#ENERGY CALIBRATION COEFFICIENTS ARE 2E11.3/12X
1 17#USING 2 CHANNELS, 2F8.2/16X13# 2 ENERGIES, 2F8.2/)
      RETURN
CC      CALCULATE CHANNEL LOCATION FOR INPUT ENERGY (=CH ARGUMENT)
      3 EGAM2=(Y2-A)/B
      RETURN
      END

```

C THIS IS FILE GETDAT.F4

```

C      USES 25, 26, & 27 FOR INPUT
SUBROUTINE GETDAT(FILEAP, NSPEC, IDAT)
  DIMENSION IDAT(1), IFIK(126)
  DOUBLE PRECISION FILNAM, FPREV8, FPREV1, FPREV2
  DATA FPREV8/10H      /
  DATA FPREV1/13H      /
  DATA FPREV2/10H      /
  DATA ISO/1/
  NV=1
  IF (FILNAM.EQ.FPREV8) GO TO 18
  IF (FILNAM.EQ.FPREV1) GO TO 1
  IF (FILNAM.EQ.FPREV2) GO TO 2
  GO TO (7,8,9), ISO
7  CALL DEFINE FILE(26,126,NV,FILNAM,8,8)
  FPREV1=FILNAM
  WRITE (5,77) FILNAM,ISO
77  FORMAT (2X,A10,' IS NOW FILE NO. 'I1/)
  ISO=2
1  LL=26
5  READ (LL,9) IRIN
  JSP=IRIN(48)
  IF (NSPEC.GT.JSP) GO TO 58
  X=9+(NSPEC-1)+1
  K9=X+8
  J=1
  DO 4 K=X,K9
  READ (LL,K) IRIN
  DO 3 M=1,126
  IDAT (J)=IRIN(M)
3  J=J+1
4  CONTINUE
  RETURN
8  CALL DEFINE FILE (27,126,NV,FILNAM,8,8)
  FPREV2=FILNAM
  WRITE (5,77) FILNAM,ISO
  ISO=3
2  LL=27
  GO TO 5
9  CALL DEFINE FILE(25,126,NV,FILNAM,8,C)
  FPREV8=FILNAM
  WRITE (5,77) FILNAM,ISO
  ISO=1
18  LL=25
  GO TO 5
58  WRITE (5,99) FILNAM,JSP,NSPEC
99  FORMAT(/' ***'/ ' GETDAT CALLED 'A10,' , HAS 'I3,' SPECTRA'/
X    ' CALLED FOR SPECTRUM NO. 'I4/)
  DO 68 I=1,1134
68  IDAT(I)=3
  RETURN
END

```

APPENDIX C. PREPARATION OF SUMMED DATA FOR FERD
UNFOLDING; THE CODES ANALYZB AND ANALYZC

These two codes differ only in the output bin structures, ANALYZB for beta-ray data and ANALYZC for gamma-ray data. Energy-gain calibration is introduced in this routine, once for the first case, and using the same gain calibration for subsequent cases. The only important difference between ANALYZB and ANALYZC is for the output energy bin structure gotten from variable ELL, EU, and ED used in the DO 32 loop at the beginning of the program.

The algorithm for determining the integral counts for given energy limits for a bin is the same as discussed for the subroutine GAINSH (page 180) in Appendix B. Binned data are read onto Logical 22 along with appropriate control cards required by the FERD unfolding code.

Subroutine INTERP is found on page 195.


```

C THIS IS FILE ANV20.F4
C THIS CODE PREPARES SUMTF OUTPUT IN FORMAT FOR FERD UNFOLDING
C FOR DECAY HEAT DATA
C PARTICULARLY FOR THE BETA RAY DATA
C
  DIMENSION BIN(512),VIN(512),DOUT(110),UOUT(110),FLO(111),
  X EOUT(512),BAT(512),EDBOUND(2),E1(10),C1(10),X1(10),IDENT(2)
  DIMENSION ITITLE(9),ECEN(110),UTOT(512)
  DIMENSION ELL(11),EU(11),ED(11)
  DOUBLE PRECISION NWFIL
  EQUIVALENCE (ELL(2),EU(1))
  DATA ELL(1)/0.06/
  DATA EU/.12,.26,.5,.9,1.4,2.,3.2,3.8,5.,6.4,8./
  DATA EC/.015,.02,.03,.04,.05,.06,.08,.1,.12,.14,.16/
  ELO(1)=ELL(1)
  J=1
  NCASE=1
  DO 32 I=2,101
    ELO(I)=ELO(I-1)+ED(J)
    IF (ELO(I) .GE. EU(J)-0.0001) J=J+1
  32 ECEN(I-1)=0.5*(ELO(I-1)+ELO(I))
C...
  CALL OFILE(21,'ENERGY')
  22 WRITE (5,97)
  97 FORMAT (1H 'TYPE FERD FILENAME'/)
  READ (5,98) NWFIL
  98 FORMAT(A5)
  IF (NWFIL.EQ.5H ) GO TO 77
  CALL OFILE(22,NWFIL)
  GO TO 23
  93 WRITE (5,92) IEEE
  92 FORMAT(' ERROR IIFILE, IEEE='I10/)
  23 WRITE (5,96)
  96 FORMAT(1H 'TYPE -SUMTF-OUTPUT FILENAME'/)
  READ (5,100) NWFIL
  100 FORMAT(A10)
  CALL IIFILE(23,NWFIL,083,IEEE)
  24 WRITE (5,93)
  93 FORMAT (1H 'TYPE IN TITLE (36 MAX) 1'/)
  READ (5,91) ITITLE
  91 FORMAT (9A4)
  IF (NCASE.GT.1) GO TO 57
  25 WRITE (5,95)
  95 FORMAT (1H 'INPUT NUMBER OF ENERGY CALIBRATION PAIRS (<=10)'/
  X ' THEN E(I) IN MEV,C(I),I=1,N IN (2F) FORMAT'/)
  READ (5,99) NEC
  READ (5,100) (E1(I),X1(I),I=1,NEC)
  100 FORMAT (2F)
  WRITE (5,61) (E1(I),X1(I),I=1,NEC)
  WRITE (5,90)
  90 FORMAT (2X'IF ALL INPUT OKAY TYPE 0,ELSE 1 TO 4'/)
  READ (5,99) NOKAY
  99 FORMAT (1)
  IF (NOKAY.NE.0) GO TO (22,23,24,25),NOKAY
  DO 56 I=1,NEC
    C1(I)*X1(I)=1.5
  56 CHANNEL LABELLED "0" HAS INDEX 1, ALSO 0.5 CHANNEL DIFFERENCE
  C BETWEEN "OBSERVATION" & CALCULATION
C...
  57 READ (23,60) IDENT,XMA,NPWS,NE
  WRITE (5,60) IDENT,XMA,NPWS,NE
  DO 1 I=1,NPWS
    A1=I
    CALL INTERP(C1,E1,NEC,2,A1,E2)
    1 EOUT(I)=E2
  WRITE (21,101) (I,EOUT(I),I=1,NPWS)
  101 FORMAT (2X,5(I5,F12,5))
  60 FORMAT (2A5,E15,5,2(I10)
  READ (23,61) EDBOUND(1),EDBOUND(2)
  61 FORMAT (2E15,5)

```

```

C NEXT WRITE FERD CONTROL CARDS PRECEEDING DATA
  WRITE (22,64) ITITLE,EBBUND(1),EBBUND(2)
64 FORMAT (3X6H 90100,23X9A4,F6.1,F6.0)
  WRITE (22,65)
65 FORMAT (12H0FFBH      1.60X/33X3H100.16711H      1 +117X/
X 32H0FORMAT R0H 03 (1A6,14,12,3E15.0)40X)
  READ (23,62) (DIN(J),J=1,NPHS)
62 FORMAT (10F0.1)
  READ (23,62) (UIN(J),J=1,NPHS)
10  DAT(1) = 0.0
  UTOT(1) = 0.0
  DO 2 J=2,NPHS
    UTOT(J) = UTOT(J-1) + UIN(J) * 0.2
  2  DAT(J) = DIN(J) + DAT(J-1)
    CALL INTERP(E1,C1,NEC,2,EL0(1),X)
    XL=X
    IX=X
    IF (IX.LT.1) IX=1
    UL=UTOT(IX)
    DL=DAT(IX)
    D=DL
    J=1
  3  EX=EL0(J+1)
    CALL INTERP(E1,C1,NEC,2,EL0(J+1),X)
    IX=X
    IF (IX.LT.0 .OR. IX.GT.NPHS) GO TO 4
    IF (IX.LT.2) IX=2
    IF (IX.GT.510) IX=510
    CP=FLOAT(IX)
    EP=DAT(IX)
    EQ=DAT(IX-1)
    ER=EQ-EP
    EQ=DAT(IX+1)
    ER=EQ-EP
    ER=ER-EP
    CALL FIT2EC(T,U,V,EP,=1.,0.,0.,EQ,1.,ER,2.,1.0)
    D=EP+T*U*(X-CP)+V*(X-CP)**2
  4  DOUT(J)=D-DL
    DL=D
    UNC=0.
    IF (IX.LT.0 .OR. IX.GT.NPHS) GO TO 6
    UP=UTOT(IX)
    UQ=UTOT(IX-1)
    UR=UR-UP
    UQ=UTOT(IX+1)
    UR=UR-UP
    UR=UTOT(IX+2)
    UR=UR-UP
    CALL FIT2EC(T,U,V,UQ,=1.,0.,0.,UQ,1.,UR,2.,1.)
    UNC=UP+T*U*(X-CP)+V*(X-CP)**2-UL
  6  IF (UNC.LT.1.0) UNC=1.0
    UOUT(J)=SORT(UNC)
    XL=X
    UL=UL+UNC
    J=J+1
    IF (J.LE.100) GO TO 3
    DO 7 J=1,100
  7  WRITE (22,170) J,ECEN(J),DOUT(J),UOUT(J)
170 FORMAT (110,2X,3E15.0,23X)
C WRITE FERD TRAILER CARDS
  WRITE (22,74)
74 FORMAT (7H=END MT)
  8 CONTINUE
  END FILE 22
  CALL RELEASE(23)
  NCASE=NCASE+1
  GO TO 22
77  END FILE 22
  STOP
  END

```

```

      SUBROUTINE FIT2EC(A,B,C,E1,C1,E2,C2,E3,C3,E4,C4,AN)
C      FIND COEFF A,B,C TO FIT  $E(C) = A + B \cdot C + C \cdot C^{1/2}$ 
C      IF AN .NE. ZERO, THE C(I) ARE EQUISPACED BY AN
      SC=C1+C2+C3+C4
      SE=E1+E2+E3+E4
      CA=C1+C1
      CB=C2+C2
      CC=C3+C3
      CD=C4+C4
      SEC=C1+E1+C2+E2+C3+E3+C4+E4
      SEC2=CA+E1+CB+E2+CC+E3+CD+E4
      SC2=CA+CB+CC+CD
      SC3=CA+C1+CB+C2+CC+C3+CD+C4
      SC4,CA=CA+CB+CB+CC+CC+CD+CD
      IF (AN .GE. 0.1) GO TO 2
      D=DET3(4,SC,SC2,SC,SC2,SC3,SC2,SC3,SC4)
      IF (ABS(D)-1E-5) 1,1,3
2      D=80,0AN=0.6
3      A=DET3(SE,SEC,SEC2,SC,SC2,SC3,SC2,SC3,SC4)
      A=A/D
      B=DET3(4,SC,SC2,SE,SEC,SEC2,SC2,SC3,SC4)
      B=B/D
      C=DET3(4,SC,SC2,SC,SC2,SC3,SE,SEC,SEC2)
      C=C/D
      RETURN
1  WRITE (21,100) E1,C1,E2,C2,E3,C3,E4,C4,D
100  FORMAT(2X,10(1H+)/3X13HEXERROR FIT2EC 9E12,4/)
      A=0.
      B=0.
      C=0.
      RETURN
      END
      FUNCTION DET2(A1,A2,B1,B2)
      DET2=A1*B2-A2*B1
      RETURN
      END
      FUNCTION DET3(A1,A2,A3,B1,B2,B3,C1,C2,C3)
      DET3= A1*B2*C3+C1*A2*B3+B1*C2*A3-C1*B2*A3-B1*A2*C3-A1*C2*B3
      RETURN
      END

```

APPENDIX D. COMBINING DATA (IN FERD INPUT FORMAT)
FOR LOW- AND HIGH-GAIN SETTINGS;
THE CODES DATMIX AND DATMXB

These two codes differ only in input and output bin structures. The high-gain (i.e. low energy) data are normalized to the low-gain data using relative values of n_f determined as discussed in Section 5. Output data for E_β (or E_γ) < 0.5 MeV are taken from the renormalized high-gain file, for E_β (or E_γ) > 1.6 MeV from the low-gain file, and between 0.5 and 1.6 MeV by combining the two data sets by a simple averaging procedure. The uncertainties are taken directly from the low-gain file for $E_\beta < 0.5$ MeV, from the high-gain file for $E_\beta > 1.6$ MeV, and a nonstatistical average for the combined data.

A new file, on logical 23, is created with the file name XXXYY.DAT, where XXX is the logical AND of "234" with the first 3 characters of the high-gain file name, and YY is the logical AND of "??" with the last 2 characters of that file name. The bit representation of "?" for the PDP-10 is all ones, hence YY is the last 2 characters of that file name. Care must be taken not to use an input file name which becomes the output file name.

```

C THIS IS FILE DATNIX.F4
  DIMENSION I(50)
  DATA IC/5HCCCCC/,100/5H00000/,IBLANK/5H
  DATA IN/5HNNNNN/,IB/5H88888/,II/5H11111/
  DATA IM/5HNNNNN/,IEE/5HEEEEE/,ID/5HDDDDD/
  DATA MASK/5H23477/
  TYPE 64
64  FORMAT(' LOW ENERGY DATA ')
  TYPE 66
66  FORMAT(' INPUT FILENAME(A5)= 'S)
  ACCEPT 57, KFILE
67  FORMAT(A5)
  CALL IFILE(22, KFILE)
  KFILE=KFILE.AND.MASK
  CALL OFILE(23, KFILE)
  TYPE 68
68  FORMAT(' LOW-ENERGY MULTIPLYING FACTOR= 'S)
  ACCEPT 51, AMUL
61  FORMAT(F)
  WRITE (23,23)
  23  FORMAT('*END DATA ')
  READ (22,68) I
68  FORMAT(30A1)
  I(33)=IC
  I(34)=100
  I(35)=IN
  I(36)=IB
  I(37)=II
  I(38)=IM
  I(39)=IEE
  I(40)=ID
  I(41)=IBLANK
  I(42)=IBLANK
  WRITE(23,68) I
  TYPE 68, I
  READ (22,68) I
  WRITE (23,68) I
  READ (22,68) I
  WRITE (23,69) (I(K),K=1,52)
69  FORMAT(52A1,'-1 -1 1 +1 ')
  READ (22,68) I
  WRITE (23,68) I
  DO 3 J=1,30
  READ (22,49) (I(K),K=1,27),DLO,ELO
  DLO=DLO*AMUL
  ELO=ELO*AMUL
44  FORMAT(57A1)
  WRITE (23,49) (I(K),K=1,27),DLO,ELO
  3  CONTINUE
  TYPE 65
65  FORMAT(/' HIGH ENERGY DATA ')
  TYPE 66
  ACCEPT 57, KFILE
  CALL IFILE(21, KFILE)
  READ (21,68) I
  TYPE 68, I
  DO 4 J=1,33
  READ (21,44) (I(K),K=1,57)
  DO 6 J=31,77
  READ (22,48) DLO,ELO

```

```

48  FORMAT(27X,2E15.8)
    DLO=DLO+AMUL
    ELO=ELO+AMUL
    READ (21,49) (I(K),K=1,27),DNI,ENI
49  FORMAT(27A1,2E15.8)
    DLO=0.5*(DLO+DNI)
    ELO=0.4*(ELO+ENI)
6   WRITE (23,49) (I(K),K=1,27),DLO,ELO
    DO 5 J=78,176
    READ (21,44) (I(K),K=1,57)
    WRITE (23,44) (I(K),K=1,57)
5   CONTINUE
    WRITE (23,46)
46  FORMAT('*END MI  ')
    END FILE 23
    TYPE 77,MFILE
77  FORMAT(' FINISHED 'A5, '.DAT FILE'//)
    STOP
    END

```

APPENDIX E. THE COMPUTER CODE TO GENERATE THE GAMMA-RAY RESPONSE MATRIX

The gamma-ray response matrix code is an adaptation of an earlier version written by G. L. Morgan.⁵¹ Most of the code is the same including the subroutines PLT (page 195), INTERP (page 195), XINT (page 196), INTEG (page 197), AREA (page 197), and MATINV (page 198). Most of the logical development of the main routine is the same. For each "Comparison" energy a total response is calculated for several hundred evenly spaced intervals. Then the total response is integrated for each bin of the desired group structure.

The important differences between the present code and the earlier version is in the handling of the total efficiency and peak-to-total ratios, the inclusion of a subroutine XESCAP (page 194), and the inclusion of a defined backscatter correction. In the earlier version these portions of the response were included with tabulated input response for a given E_γ . In the present code, the total efficiency was determined from attenuation coefficients⁴³ in air, Al, and NaI, and the various peak-to-total ratios are smooth estimations from the experimental responses.

A listing of the input data starts on page 200.

```

//JKDF0002 JOB (10647,...25,4000), 'SAVE104,1116 BL0025', PSGLVE1=1
//*CLASS CPU01,2M,10,2,REGION=270,LIRES=12,CARDS=24
/*ROUTE PRINT LOCAL
/*ROUTE PUNCH REMOTE5
// EXEC FORTNCLG,REGION.G0=256K,PARN.G0='E0=1,EU=1,SP=51'
//FORT.SYSIN DD *
    DIMENSION NAME(20),ELBW(20), ENGW(20), EDEL(20), NPTS(25),
    1 ES(25,200), RS(25,200), E,3600),RM(3000),EC(200),DELEC(200),
    2 X1(500), X2(500), Y1(500), Y2(500), ORW(200), FGAM(25),
    3 XNWR(25), NPLT(20)
    DIMENSION ESAB(30),TOT(30),PK2TOT(30),SE2TOT(30),BKSC(30),
    X COMPT(30)
    REAL*8 IVAR(9)
    DOUBLE PRECISION SUM
CARD READ NO 1 TITLE
    1 READ (5,10) (NAME(I), I = 1, 20)
    10 FORMAT (20A4)
    WRITE (51,0005) (NAME(I), I = 1, 20)
    0005 FORMAT (1M120A4///)
CARD READ NO 2
    READ (5,20) DELP, DE2SE, NGRP, NSPEC, NTERMS
    20 FORMAT (2F10.0,4I5)
    WRITE (51,0004) NTERMS
    0004 FORMAT (1M 15H NTERMS = 11/)
    WRITE (51,0003) DE2SE,DELP
    0003 FORMAT (1M'DE/SE RATIO = 'E13.5/ 4X
    X 'ENERGY CURVE = 0.0 * ' F5.3,'*EC*(I=1)'//)
CARD READ NO 3 WHICH SPECTRA TO PLOT
    READ (5,25) IP, (NPLT(I), I = 1, IP)
    25 FORMAT (14I5)
CARD READ NO 4 NSAB=NUMBER OF EFFICIENCIES
    READ (5,26) NSAB
    26 FORMAT (15)
CARD READ NO 5 IVAR IS A FORMAT FOR THE EFFICIENCY DATA
    READ (5,40) IVAR
    40 FORMAT (9A0)
CARD READ NO 6A TO 6--
    DO 99 I=1,NSAB
    READ (5,IVAR) ESAB(I),TOT(I),PK2TOT(I),SE2TOT(I),BKSC(I)
C ESAB=ENERGY
C TOT=TOTAL EFFICIENCY
C PK2TOT = PEAK-TO-TOTAL RATIO
C SE2TOT = SINGLE-ESCAPE-TO-TOTAL
C BKSC=BACKSCATTER (ALSO RATIO TO TOTAL)
    COMPT(I)=TOT(I)*(1.-PK2TOT(I))-(1.-DE2SE)*SE2TOT(I)
    IF (COMPT(I) .GE. 0.0) GO TO 99
    WRITE (51,90) I,TOT(I),PK2TOT(I),SE2TOT(I),DE2SE
    90 FORMAT(' ERROR== N= '12.4E12.4)
    STOP
    99 CONTINUE
    WRITE (51,7005)
    7005 FORMAT (1M113HEFFICIENCIES //)
    WRITE (51,7006)
    7006 FORMAT(1M 3X6HEENERGY7X5HTOTAL0X'PK/TOT'7X'SE/TOT BACKSCATTER'//)
    DO 2 I=1,NSAB
    2 WRITE (51,7007) ESAB(I),TOT(I),PK2TOT(I),SE2TOT(I),BKSC(I)
    7007 FORMAT(E14.5,4E13.5)
CARD READ NO 7 BIN LIMITS AND WIDTHS
    DO 100 I = 1, NGRP
    100 READ (5,30) ELBW(I), ENGW(I), EDEL(I)
    30 FORMAT (3F10.0)
    WRITE (51,0001)
    0001 FORMAT (1M015HGROUP STRUCTURE/)
    WRITE (51,0006)
    0006 FORMAT (1M 30H LEDGE WEDGE MEV/BIN/)
    WRITE (51,0002) (ELBW(I), ENGW(I), EDEL(I), I = 1, NGRP)
    0002 FORMAT (1M 3F10.3)
    DO 200 I = 1, NSPEC

```



```

CARD READ NO 8
  READ (5,50) NPTS(I), EGAM(I), XNOR(I)
50 FORMAT (I5, 2F10.4)
  WRITE(51,50) NPTS(I), EGAM(I), XNOR(I)
  NPTS = NPTS(I)
CARD READ NO 9 A FORMAT
  READ (5,40) IVAR
CARD READ NO 10 ETC ENERGY-YIELD PAIRS
  READ (5,IVAR) (ES(I,J), RS(I,J), J = 1, NPTS)
200 CONTINUE
  DO 250 J = 1, NSPEC
    KT = NPTS(J)
    DO 250 I = 1, KT
      250 RS(J,I) = XNOR(J)*RS(J,I)
      WRITE (7,1043)
1043 FORMAT('FORMAT COL 80 (1A2,2I3,0E9.3)')
      WRITE (51,7000)
7000 FORMAT (1H115MINPUT RESPONSES///)
      DO 7001 I = 1, NSPEC
        WRITE (51,7000)
7001 FORMAT (1H //)
        WRITE (51,7002) EGAM(I)
7002 FORMAT (1H018MINCIDENT ENERGY = F6.3,' MEV')
        WRITE (51,7003)
7003 FORMAT (1H 3(5X6MEGEGY5X13MCOUNTS/MEV/UI//)
        NPTS = NPTS(I)
        DO 150 K = 1, NPTS
          E(K) = ES(I,K)
150 RM(K) = RS(I,K)
          AUX = E(NPTS)
          BVAS = 0.0
          CALL XINT (E,RM,NPTS,NTEPMS,BVAS,AUX,SUM)
          AUX = SUM
          K = (NPTS-1)/3 + 1
          DO 7022 J1=1,K
            WRITE (51,7004) (ES(I,J), RS(I,J), J = J1, NPTS,K)
7022 CONTINUE
            WRITE (51,7103) AUX
7103 FORMAT (1H01INTEGRATED EFFICIENCY = F12.7,' COUNTS/UI//)
7004 FORMAT (3(F14.6,F16.0))
7001 CONTINUE
CC SETS UP GRPL STRUCTURE
  K = 0
  DO 400 I = 1, NGRP
    NLU = (ENGH(I) - ELOW(I))/EDEL(I) + 0.0001
    DO 400 J = 1, NLU
      K = K + 1
      EC(K) = ELOW(I) + (J - 0.5)*EDEL(I)
      DELEC(K) = EDEL(I)/2.0
400 CONTINUE
    MCP = K
    DO 1000 K = 1, MCP
      DO 500 I = 1, NSPEC
        IF (EC(K).LT.EGAM(I)) GO TO 550
500 CONTINUE
        I = NSPEC
550 KEH = I
        IF (KEH.EQ.1) KEH = KEH + 1
        KEL = KEH - 1
        NPT1 = NPTS(KEL)
        DO 600 I = 1, NPT1
          X1(I) = ES(KEL,I)
600 Y1(I) = RS(KEL,I)
          NPT2 = NPTS(KEH)
          DO 700 I = 1, NPT2
            X2(I) = ES(KEH,I)

```

```

700 Y2(I) = WS(KEL,I)
    DELE = EC(K)*DELP
    NSTD=1.2*EC(K)/DELE + 0.75
    DO 300 J = 1, NSTD
        X1 = I + 1
        E(I)=DELE*X1
300 R1(I) = 0.0
    DO 000 J = 1, NSTD
        AUX = E(J)
        F1 = AUX*EGAM(KEL)/EC(K)
        F2 = AUX*EGAM(KEM)/EC(K)
        CALL INTERP (X1,Y1,NPT1,NTERMS,F1,R1)
        IF (R1,GE, 0.0) GO TO 776
        WRITE (51,7202) R1,EGAM(KEL),E1
7202 FORMAT(' R1 CALCULATED='E12.4,' FOR EGAM='E12.4,
    X ' AT RESPONSE POINT='E12.4/)
        R1=0.0
776 CALL INTERP (X2,Y2,NPT2,NTERMS,E2,P2)
    IF (R2,GE, 0.0) GO TO 777
    WRITE (51,7202) R2,EGAM(KEM),E2
    P2=0.0
777 RM(J) = (EC(K) - EGAM(KEL))*R2 + (EGAM(KEM) - EC(K))*R1
    RM(J) = RM(J)/(EGAM(KEM) - EGAM(KEL))
000 CONTINUE
    DO 050 I = 1, 200
050 RM(I) = 0.0
CCALCULATE ABSOLUTE NORMALIZATION FOR COMPTON EFFICIENCY
    AUX=EC(K)
    CALL INTEHP(FSAB,COMPT,NSAB,3,AUX,TS)
    AUX=EC(K)
    CALL XINTIE,RM,NSTD,NTERMS,0.0,AUX,SUM)
    TS=TS/SUM
    DO 010 J=1,NSTD
010 RP(J)=RM(J)*TS
    SOE=SQRT(EC(K))
C*** THIS IS FOR THE DECAY HEAT PROJECT DETECTOR
    FWHM=1.3522*5.0636/SOE
C*** 12.7 CM X 12.7 CM BUT COLLIMATED TO ABOUT 9 CM ON FACE
C*** GAMMA SOURCE IS AT 100 CM
    FWHM=FWHM*0.01*EC(K)
    SIGMA=FWHM/2.35482
    AUX=EC(K)
    CALL INTERP(FSAB,TOT,NSAB,NTERMS,AUX,AA)
    AUX=EC(K)
    CALL INTERP(FSAB,PK2TOT,NSAB,NTERMS,AUX,AB)
    AS=AA*AB
    XRAY=XESCAP(EC(K))
    AX=AS*XRAY
    AS=AS-AX
    DO 030 J=1,NSTD
    AD=0.0
    X=ABS(E(J)-EC(K))/SIGMA
    IF (X,GT,6.0) GO TO 025
    AD=AS*0.3989423*EXP(-0.5*X*X)/SIGMA
025 EXRAY=EC(K)-0.020
    IF (EXRAY,LT,0.001,OR, AX,LE,0.0) GO TO 030
    X=ABS(E(J)-EXRAY)/SIGMA
    IF (X,GT,6.0) GO TO 030
    AD=AD+AX*0.3989423*EXP(-0.5*X*X)/SIGMA
030 RM(J)=RM(J)+AD
    AUX=EC(K)
    CALL INTERP(FSAB,SE2TOT,NSAB,2,AUX,SE2)
    IF (SE2,LE,1E-5) GO TO 070
    AS=AA*SE2
    AD=DE2SE*AS
    DO 035 J=1,NSTD
    AD=0.0
    X=ABS(E(J)-EC(K))*0.511/SIGMA
    IF (X,GT,6.0) GO TO 045
    AD=AD+AS*0.3989423*EXP(-0.5*X*X)/SIGMA

```

```

845 X=X+0.511/SIGMA
    IF (X.GT.6.0) GO TO 835
    AD=AD+AQ*0.3989423*EXP(-0.5*X*X)/SIGMA
835 RM(J)=RM(J)+AD
870 ALPHA=EC(K)/0.511
    AUX=EC(K)
    CALL INTERP(ESAB,BKSC,NSAB,3,AUX,BKS)
    ENUP=EC(K)/(1.+1.707107*ALPHA)
    AS=AA*BKS
    DO 855 J=1,NSTD
    AD=0.
    X=ABS(E(J)-ENUP)/SIGMA
    IF (X.GT.6.0) GO TO 855
    AD=AD+AS*0.3989423*EXP(-0.5*X*X)/SIGMA
855 RM(J)=RM(J)+AD
    U2 8U2 JPL = 1. IP
    IF (K.EQ.NPLOT(JPL)) GO TO 904
802 CONTINUE
    G4 TO 803
804 WRITE (51,820) EC(K)
820 FORMAT (1H16HRESPONSE AT E = F8.4//)
    JK=1
    CALL PLT(RM,E,JK,NSTD,2)
803 CONTINUE
    DO 900 I = 1, MCP
    AUX = EC(I)
    E1 = AUX - DFLEC(I)
    E2 = AUX + DFLEC(I)
    CALL XINT (E,RM,NSTD,NTERMS,F1,E2,SUM)
    BRM(I) = SUM
    IF (SUM.LT.0.0) BRM(I) = 0.0
    IF (SUM.LT.1.0E-09) GO TO 950
900 CONTINUE
950 NBRM = 1
    IF (NBRM.GT.MCP) NBRM = MCP
    WRITE (51,5000) K, EC(K), AA,AB SE2,BKS
5000 FORMAT (1H16HCOMPARISON PRINT ...', E=F8.4', FFF='1P4F11.4//)
    WRITE (51,5001) (I, EC(I), BRM(I), I = 1, NBRM)
5001 FORMAT (1H 3(I8, 2E15.5))
    THISUM=0.0
    DO 955 I=1,NBRM
955 THISUM=THISUM+BRM(I)
    WRITE (51,7100) THISUM
    NCARD = NBRM/A + 1
    DO 960 I = 1, NCARD
    NIND = (I - 1)*8 + 1
    IF (NIND.GT.NBRM) GO TO 1000
    NAUX = NBRM - NIND
    IF (NAUX.GT.7) NAUX = 7
    NSTOP = NIND + NAUX
    WRITE (7, 5002) NIND, K, (BRM(L), L = NIND, NSTOP)
5002 FORMAT (2X, 2I3, 1P8E9.3)
960 CONTINUE
1000 CONTINUE
    WRITE (7,1044)
1044 FORMAT('END MT ')
2000 WRITE (7, 1600) (I, EC(I), I = 1, MCP)
1600 FORMAT (6X14, 2XE15.8)
    GO TO 1
END

```

```

C
FUNCTION XESCAP(E)
  THIS TO ACCOUNT FOR INDINE X-RAY ESCAPE
  DIMENSION EG(11),RATIO(11)
  DATA EG/.05,.06,.07,.08,.09,.1,.11,.125,.15,.19,.25/
  DATA RATIO/.16,.125,.1,.083,.067,.057,.0475,.037,.025,.014,.005/
  X=0.
  IF (E.LT.EG(1) .OR. E.GT.EG(11)) GO TO :
  CALL INTERP(EG,RATIO,11,2,E,X)
1 XESCAP=X
  RETURN
END

```

```

SUBROUTINE PLT (A,C,NLOW,NHIGH,NCYC)
DIMENSION A(1), C(1)
DIMENSION NYM(120)
DATA NAM/4HXXXX/
DATA NDM/4H /
DATA NCH/4H0000/
IF (NCYC.LE.0) NCYC = 2
DO 10 I = 1, 30
10 NYM(I) = NCM
WRITE (51,20) (NYM(I), I = 1,30)
20 FORMAT (1H 5X30A4)
DO 30 I = NLOW, NHIGH
IF (ABS (A(I)).LT.1.0E-08) GO TO 103
XN = 10.0 + ALOG10 ( ABS(A(I)))
NX = XN
NX = NX/NCYC
NX = NX*NCYC
NX = 100.0*(XN - FLOAT(NX))/FLOAT(NCYC) + 0.5
GO TO 104
103 NX = 0
104 CONTINUE
DO 50 K = 1, 120
50 NYM(K) = NDM
IF (NX.GT.0) NYM(NX) = NAM
IF (C(1).LT.1.0) GO TO 75
WRITE (51,80) C(1), (NYM(K), K = 1, 105), A(1)
GO TO 30
75 WRITE (51,70) C(1), (NYM(K), K=1,105), A(1)
70 FORMAT(1H F7.3,2H *105A1,1PE10,3)
30 CONTINUE
80 FORMAT(1H F7.2,2H *105A1,1PE10,3)
RETURN
END

SUBROUTINE INTERP (X, Y, NPTS, NTERMS, XIN, YOUT)
DOUBLE PRECISION DELTAX, DELTA, A, PROD, SUM
DIMENSION Y(1), Y(1)
DIMENSION DELTA(10), A(10)

C
C SPECIAL TEST FOR NAI MATRIX GENERATION
C
IF (XIN.LE.X(NPTS)) GO TO 1000
YOUT = 0.0
RETURN
1000 CONTINUE

C
C SEARCH FOR APPROPRIATE VALUE OF X(1)
C
11 DO 19 I = 1, NPTS
IF (XIN - X(I)) 13, 17, 19
13 I1 = I + NTERMS/2
IF (I1) 15, 15, 21
15 I1 = I
GO TO 21
17 YOUT = Y(I)
19 GO TO 61
19 CONTINUE
I1 = NPTS - NTERMS + 1
21 I2 = I1 + NTERMS - 1
IF (NPTS - I2) 23, 31, 31
23 I2 = NPTS
I1 = I2 - NTERMS + 1
25 IF (I1) 26, 26, 31
26 I1 = I
27 NTERMS = I2 - I1 + 1

```

```

C
C      EVALUATE DEVIATIONS DELTA
C
31 DENOM = X(I1+1) - X(I1)
   DELTA = (XIN - X(I1)) / DENOM
   DO 35 I = 1, NTERMS
   IX = I1 + I - 1
35 DELTA(I) = (X(IX) - X(I1)) / DENOM

C
C      ACCUMULATE COEFFICIENTS A
C
40 A(1) = Y(I1)
41 DO 50 K = 2, NTERMS
   PROD = 1.
   SUM = 0.
   IMAX = K - 1
   IXMAX = I1 + IMAX
   DO 49 I = 1, IMAX
   PROD = 1.0
   DO 48 IJ = 1, IMAX
48 PROD = PROD * (DELTA(K) - DELTA(IJ))
49 SUM = SUM + A(IJ) / PROD
   PROD = 1.0
   DO 47 I = 1, IMAX
47 PROD = PROD * (DELTA(K) - DELTA(I))
50 A(K) = SUM + Y(IXMAX) / PROD

C
C      ACCUMULATE SUM OF EXPANSION
C
51 SUM = A(1)
   DO 57 J = 2, NTERMS
   PROD = 1.
   IMAX = J - 1
   DO 56 I = 1, IMAX
56 PROD = PROD * (DELTA(J) - DELTA(I))
57 SUM = SUM + A(J) * PROD
60 YOUT = SUM
61 RETURN
   END

SUBROUTINE XINT (E,F,NPTS,NTERMS,E1,E2,SUM)
  DIMENSION E(1), F(1), X(3000), Y(3000)
  DOUBLE PRECISION SUM
C  ADDED TO GLP XINT PROGRAM FOR NAI-RESPONSE GENERATION CODE
  IF (E1.LT.E(NPTS)) GO TO 50
  SUM=0.
  RETURN
50 DO 100 I = 1, NPTS
  IF (E1.LT.E(I)) GO TO 200
100 CONTINUE
200 IL = I
  SUM = 0.0
  DO 300 I = 1, NPTS
  IF (E2.LT.E(I)) GO TO 400
300 CONTINUE
400 IH = I - 1
  IF (IH.LT.IL) GO TO 700
  ILA = IL - NTERMS/2
  IF (ILA.LE.0) ILA = 1
  IF (ILA + NTERMS = 1.GT.NPTS) ILA = NPTS - NTERMS + 1
  IHA = IH - NTERMS/2
  IF (IHA.LT.1) IHA = 1
  IF (IHA + NTERMS = 1.GT.NPTS) IHA = NPTS - NTERMS + 1
  CALL INTEG (E,F,NTERMS,ILA,E1,E(IL),SUM)
  CALL INTEG (E,F,NTERMS,IHA,E(IH),E2,SUM)
  GO TO 800

```

```

700 ILA = IL - NTERMS/2
   IF (ILA.LE.0) ILA = 1
   IF (ILA + NTERMS - 1.GT.NPTS) ILA = NPTS - NTERMS + 1
   CALL INTEG (E,F,NTERMS,ILA,E1,E2,SUM)
800 CONTINUE
   K = 0
   DO 500 I = IL, IM
     K = K + 1
     X(K) = E(I)
500 Y(K) = F(I)
     IF (K.LE.1) GO TO 900
     NPTSA = K
     SUM = SUM + AREA(X,Y,NPTSA,NTERMS)
900 CONTINUE
   RETURN
   END

```

```

SUBROUTINE INTEG (X, Y, NTERMS, I1, X1, X2, SUM)
DOUBLE PRECISION XJK, ARRAY, A, DENOM, DELTAX, SUM
DIMENSION X(1), Y(1)
DIMENSION ARRAY (15,15)

```

```

C
C
C      CONSTRUCT SQUARE MATRIX AND INVERT

```

```

11 DO 17 J = 1, NTERMS
   I = J + 11 = 1
   DELTAX = X(I) - X(I1)
   XJK = 1.
   DO 17 K = 1, NTERMS
     ARRAY(J,K) = XJK
17 XJK = XJK*DELTAX
21 CALL MATINV (ARRAY,NTERMS,DET)
   IF (DET) 31, 23, 31
23 IMID = I1 + NTERMS/2
   SUM = SUM + Y(IMID)*(X2 - X1)
   GO TO 40

```

```

C
C
C      EVALUATE COEFFICIENTS AND INTEGRATE

```

```

31 DX1 = X1 - X(I1)
   DX2 = X2 - X(I1)
33 DO 39 J = 1, NTERMS
   A = 0.
   DO 37 K = 1, NTERMS
     I = K + 11 = 1
37 A = A + Y(I)*ARRAY(J,K)
   DENOM = J
39 SUM = SUM + (A/DENOM)*(DX2**J - DX1**J)
40 RETURN
   END

```

```

FUNCTION AREA (X,Y,NPTS,NTERMS)
DOUBLE PRECISION SUM
DIMENSION X(1), Y(1)
11 SUM = 0.
   IF (NPTS = NTERMS) 21, 21, 13
13 NEVEN = 2*(NTERMS/2)
   IDelta = NTERMS/2 + 1
   IF (NTERMS = NEVEN) 31, 31, 51

```

```

C
C
C      FIT ALL POINTS WITH ONE CURVE

```

```

21 X1 = X(1)
   X2 = X(NPTS)
23 CALL INTEG(X,Y,NPTS,1,X1,X2,SUM)
   GO TO 71

```

C
C
C

EVEN NUMBER OF TERMS

```

31 X1 = X(1)
   J = NTERMS - IDELTA
   X2 = X(J)
   CALL INTEG (X,V,NTERMS,1,X1,X2,SUM)
   I1 = NPTS - NTERMS + 1
   J = I1 + IDELTA
   X1 = X(J)
   X2 = X(NPTS)
39 CALL INTEG (X,V,NTERMS,I1,X1,X2,SUM)
   IF (I1 = 2) 71, 71, 41
41 IMAX = I1 + 1
   DO 46 I = 2, IMAX
   J = I + IDELTA
   X1 = X(J)
   X2 = X(J+1)
46 CALL INTEG (X,V,NTERMS,I,X1,X2,SUM)
   GO TO 71

```

C
C
C

ODD NUMBER OF TERMS

```

51 X1 = X(1)
   J = NTERMS - IDELTA
   X2 = (X(J) + X(J+1))/2.
   CALL INTEG (X,V,NTERMS,1,X1,X2,SUM)
   I1 = NPTS - NTERMS + 1
   J = I1 + IDELTA
   X1 = (X(J) + X(J+1))/2.
   X2 = X(NPTS)
59 CALL INTEG (X,V,NTERMS,I1,X1,X2,SUM)
   IF (I1 = 2) 71, 71, 61
61 IMAX = I1 + 1
   DO 66 I = 2, IMAX
   J = I + IDELTA
   X1 = (X(J+1) + X(J))/2.
   X2 = (X(J+2) + X(J+1))/2.
66 CALL INTEG (X,V,NTERMS,I,X1,X2,SUM)
71 AREA = SUM
   RETURN
   END

```

```

SUBROUTINE MATINV (ARRAY, NORDER, DET)
  DOUBLE PRECISION ARRAY, AMAX, SAVE
  DIMENSION ARRAY(15,15), IK(15), JK(15)
10 DET = 1.
11 DO 100 K = 1, NORDER

```

C
C
C

FIND LARGEST ELEMENT ARRAY(I,J) IN REST OF MATRIX

```

   AMAX = 0.
21 DO 30 I = K, NORDER
   DO 30 J = K, NORDER
23 IF (DABS(AMAX) - DABS(ARRAY(I,J))) 24, 24, 30
24 AMAX = ARRAY (I,J)
   IK(K) = I
   JK(K) = J
30 CONTINUE

```

C
C
C

INTERCHANGE ROWS AND COLUMNS TO PUT AMAX IN ARRAY(K,K)

```

31 IF (AMAX) 41, 32, 41
32 DET = 0,
   GO TO 140
41 I = IK(K)
   IF (I = K) 21, 51, 43

```

```

43 DO 50 J = 1, NORDER
   SAVE = ARRAY(K,J)
   ARRAY(K,J) = ARRAY(I,J)
50 ARRAY(I,J) = -SAVE
51 J = JK(K)
   IF (J = K) 21, 61, 53
53 DO 60 I = 1, NORDER
   SAVE = ARRAY(I,K)
   ARRAY(I,K) = ARRAY(I,J)
60 ARRAY(I,J) = -SAVE

C
C      ACCUMULATE ELEMENTS OF INVERSE MATRIX
C
61 DO 70 I = 1, NORDER
   IF (I = K) 63, 70, 63
63 ARRAY(I,K) = -ARRAY(I,K) / AMAX
70 CONTINUE
71 DO 80 I = 1, NORDER
   DO 80 J = 1, NORDER
   IF (I = K) 74, 80, 74
74 IF (J = K) 75, 80, 75
75 ARRAY(I,J) = ARRAY(I,J) + ARRAY(I,K)*ARRAY(K,J)
80 CONTINUE
81 DO 90 J = 1, NORDER
   IF (J = K) 83, 90, 83
83 ARRAY(K,J) = ARRAY(K,J) / AMAX
90 CONTINUE
   ARRAY(K,K) = 1. / AMAX
100 DET = DET * AMAX

C
C      RESTORE ORDERING OF MATRIX
C
101 DO 130 L = 1, NORDER
   K = NORDER - L + 1
   J = IK(K)
   IF (J = K) 111, 111, 105
105 DO 110 I = 1, NORDER
   SAVE = ARRAY(I,K)
   ARRAY(I,K) = -ARRAY(I,J)
110 ARRAY(I,J) = SAVE
111 I = JK(K)
   IF (I = K) 130, 130, 113
113 DO 120 J = 1, NORDER
   SAVE = ARRAY(K,J)
   ARRAY(K,J) = -ARRAY(I,J)
120 ARRAY(I,J) = SAVE
130 CONTINUE
140 RETURN
   END

/*
//LKED.SYSIN DD *
/*
//GO.FT07F001 DD SYSOUT=B,DCB=(RECFM=FB,LRECL=80,BLKSI2F=3520),
// SPACE=(3520,(30),RLSE)
//GO.FT05F001 DD *

```


[illegible]

14	0.275	0.0005	
(4F10.2)			
0.0	0.21	0.02	0.24
0.05	0.275	0.075	0.33
0.1	0.37	0.125	0.45
0.15	0.5	0.175	0.5
0.2	0.41	0.215	0.32
0.225	0.2	0.235	0.12
0.24	0.08	0.245	0.02
17	0.412	0.001	
(4F10.2)			
0.0	0.1	0.05	0.115
0.1	0.142	0.15	0.19
0.175	0.25	0.2	0.23
0.22	0.43	0.23	0.47
0.24	0.47	0.2499	0.4
0.20	0.23	0.3	0.2
0.32	0.17	0.34	0.13
0.36	0.06	0.37	0.02
0.30	0.005		
21	0.511	0.00005	
(4F0.2)			
0.0	0.12	0.05	0.142
0.1	0.104	0.125	0.21
0.15	0.232	0.175	0.26
0.2	0.268	0.225	0.322
0.25	0.355	0.275	0.39
0.3	0.42	0.312	0.41
0.325	0.35	0.35	0.21
0.375	0.14	0.4	0.115
0.425	0.095	0.45	0.05
0.462	0.025	0.475	0.01
0.405	0.004		
22	0.662	0.001	
(4F7.2)			
0.0	0.12	0.05	0.125
0.1	0.137	0.15	0.15
0.175	0.162	0.2	0.175
0.225	0.2	0.25	0.225
0.275	0.275	0.3	0.207
0.325	0.262	0.35	0.224
0.375	0.215	0.4	0.237
0.425	0.3	0.45	0.425
0.475	0.25	0.5	0.137
0.525	0.06	0.55	0.048
0.575	0.02	0.59	0.004
22	0.835	0.001	
(4F0.2)			
0.0	0.09	0.1	0.1
0.2	0.12	0.3	0.15
0.35	0.17	0.4	0.172
0.5	0.2	0.55	0.24
0.59	0.25	0.62	0.23
0.635	0.17	0.65	0.16
0.67	0.14	0.68	0.11
0.69	0.07	0.7	0.051
0.715	0.048	0.73	0.047
0.75	0.042	0.77	0.033
0.79	0.01	0.81	0.001
21	1.115	0.000436	
(4F10.2)			
0.0	0.25	0.1	0.25
0.2	0.26	0.3	0.20
0.4	0.3	0.5	0.32
0.6	0.36	0.65	0.4
0.7	0.46	0.75	0.53
0.8	0.61	0.825	0.61
0.85	0.56	0.875	0.46
0.9	0.35	0.925	0.22
0.95	0.12	0.975	0.07
1.0	0.065	1.02	0.02
1.03	0.004		

22	1.274	8.001	
(4F0.2)			
0.0	0.11	0.1	0.117
0.25	0.122	0.36	0.13
0.5	0.141	0.6	0.151
0.75	0.16	0.64	0.2
0.9	0.22	0.95	0.24
0.97	0.24	1.0	0.231
1.025	0.2	1.05	0.15
1.08	0.05	1.1	0.064
1.12	0.051	1.14	0.049
1.16	0.042	1.18	0.05
1.2	0.01	1.22	0.002
27	1.524	0.0002	
(4F9.3)			
0.0	0.52	0.04	0.53
0.1	0.36	0.15	0.6
0.3	0.6	0.35	0.65
0.55	0.73	0.7	0.705
0.8	0.825	0.9	0.87
1.0	0.925	1.1	0.985
1.2	1.045	1.25	1.075
1.29	1.06	1.31	1.015
1.325	0.9	1.335	0.797
1.355	0.612	1.375	0.475
1.395	0.368	1.4	0.355
1.41	0.345	1.42	0.3
1.43	0.15	1.42	0.05
1.45	0.005	1.44	
26	1.78	0.001	
(4F0.4)			
0.0	0.1	0.06	0.102
0.12	0.103	0.21	0.106
0.38	0.11	0.5	0.112
0.69	0.12	0.9	0.135
0.99	0.144	1.1	0.161
1.107	0.18	1.27	0.205
1.36	0.23	1.394	0.25
1.444	0.27	1.483	0.268
1.5	0.252	1.53	0.2
1.56	0.15	1.58	0.12
1.6	0.097	1.623	0.074
1.644	0.061	1.665	0.037
1.68	0.008	1.707	0.001
28	2.25	0.00011	
(4F10.2)			
0.0	0.6	0.1	0.6
0.4	0.66	0.61	0.73
0.8	0.78	1.0	0.85
1.2	0.9	1.4	1.0
1.6	1.2	1.72	1.4
1.84	1.6	1.92	1.816
1.96	1.914	2.0	1.68
2.02	1.361	2.04	0.98
2.06	0.623	2.1	0.19
2.12	0.05	2.13	0.01
22	2.754	0.000125	
(4F0.2)			
0.0	0.26	0.15	0.27
0.4	0.3	0.5	0.31
0.75	0.36	1.0	0.41
1.25	0.48	1.5	0.58
1.6	0.7	1.9	1.0
2.0	1.1	2.1	1.21
2.35	1.6	2.4	2.0

2.45	2.0	2.5	1.6
2.55	0.92	2.58	0.57
2.61	0.5	2.63	0.25
2.65	0.09	2.67	0.02

22	3.103	0.00013
----	-------	---------

(4F6.2)

0.0	0.22	0.16	0.24
0.6	0.27	1.0	0.32
1.5	0.4	1.9	0.54
2.0	0.7	2.3	1.04
2.45	1.1	2.5	1.35
2.6	1.65	2.7	2.0
2.75	1.85	2.8	1.7
2.85	1.35	2.9	0.94
2.93	0.78	2.95	0.7
2.98	0.35	3.0	0.15
3.015	0.08	3.03	0.02

25	3.0	0.00014
----	-----	---------

(4F8.4)

0.0	0.3	0.12	0.28
0.4	0.28	0.8	0.29
1.2	0.3	1.6	0.31
2.0	0.35	2.25	0.4
2.6	0.53	2.7	0.6
2.8	0.79	2.9	0.93
3.0	1.04	3.1	1.04
3.2	1.12	3.3	1.27
3.4	1.4	3.44	1.4
3.48	1.32	3.52	1.16
3.56	0.95	3.6	0.77
3.64	0.35	3.68	0.11
3.72	0.012		

24	4.434	0.0001
----	-------	--------

(4F7.3)

0.0	0.31	0.5	0.305
1.0	0.3	2.0	0.3
2.5	0.33	3.0	0.42
3.2	0.6	3.3	0.72
3.4	0.9	3.5	1.1
3.6	1.32	3.7	1.49
3.8	1.79	3.9	2.1
4.0	2.4	4.05	2.4
4.1	2.3	4.15	1.89
4.2	1.45	4.25	1.2
4.3	0.9	4.33	0.4
4.36	0.1	4.37	0.05

24	6.13	0.00017
----	------	---------

(6F6.2)

0.0	0.1	1.0	0.09	2.0	3.08
2.5	0.08	3.0	0.087	3.5	0.098
4.0	0.132	4.5	0.23	4.8	0.35
5.0	0.45	5.1	0.5	5.2	0.56
5.4	0.7	5.5	0.83	5.6	1.0
5.7	1.22	5.75	1.2	5.8	1.0
5.9	0.7	5.95	0.65	5.98	0.5
6.02	0.3	6.04	0.08	6.06	0.011

25	3.0	0.00013
----	-----	---------

(8F10.2)

0.0	0.075	1.0	0.075	2.0	0.075	3.0	0.081
4.0	0.085	5.0	0.09	5.5	0.094	6.0	0.12
6.25	0.16	6.5	0.26	6.75	0.45	6.9	0.64
7.0	0.01	7.1	1.07	7.2	1.35	7.3	1.6
7.4	1.77	7.5	1.84	7.6	1.75	7.7	1.48
7.75	1.31	7.8	0.63	7.825	0.35	7.9	0.15
7.95	0.03						

/%
//

APPENDIX F. THE COMPUTER CODE TO GENERATE THE BETA-RAY RESPONSE MATRIX

This code is very similar to the code listed in the last appendix. Several of the subroutines are the same. These are PLT, INTERP, XINT, INTEG, AREA, and MATINV given on pages 195 to 198. The major difference in the calculation is the use of subroutines to estimate the nongaussian contribution to the response. The backscatter from the detector is approximately given³¹ by the defined function ETA(Z) in the main routine. Energy loss due to foil thicknesses is determined using BREREL (page 209). Attenuation of electrons by the entrance foil is determined using function ENONZ (page 207), and is the estimated difference between outscattering of electrons originally directed toward the detector and inscattering of electrons originally directed away from the detector. Estimation of the contribution due to slit penetration is calculated using SLIT (page 207); the remaining low-energy "tail" is estimated using BRESP (page 208). These two effects are estimated separately because the collimator sizes were changed several times during the course of the experiment.

The input data used to compute the beta-ray response matrix is listed on page 210. The "8 mg/cm²", 80 g/m² in proper units, is made up of the ~ 30 g/m² entrance foil and air plus 50 g/m² foil cover of the beta-ray insert, style no. 2 in Fig. 8.

```

//JNDV JOB (10647),.SAVE104.1116 06025',MSGLEVEL=1
//*CLAS CPU09135S.10=2,REGION=270,LINES=10,CARDS=11
//ROUTE PRINT LOCAL
//ROUTE PUNCH REMOTES
// EXEC FORTMCLG,REGION,CO=256K,PARM,GM=Eg=1,EU=1,Sc=51'
//FORT.SYSIN DD *
    DIMENSION NAME(20),ELON(20), ENGH(20), EDEL(20), NPTS(25),
    1 E(2000),RM(2000),EC(200),DELEC(200),
    2 BRM(200), EGAM(25),CPE(200),
    3 XMR(25), NPL0T(15)
    DOUBLE PRECISION SUM,SMC
    REAL*8 IVAR(5)
    DATA TMENR/0.0016/,DHP4P/1.0001013/
C DHP4P1 CHANGED MAY 10, 1976, FOR SMALLER COLLIMATOR SYSTEM
C
    EYA(2)=0.0439*EXP(-0.742*Z**0.65)
C SEE T.TAOATA PHYS REV 162,336 (1967) FOR APPROX BACKSCATTER EFFECT
C IN THIS CASE FOR BETA SCATTERING FROM CARBON.
C HIS EQN FOR EYA(E0,Z,INF) WAS USED FOR Z=6 AND THEN MULTIPLIED
C BY 1.4 TO GET BETTER NORMALIZATION TO HIS DATA.
C
CARD READ NO 1 TITLE
    1 READ (5,10) (NAME(I), I = 1, 20)
    10 FORMAT (20A4)
    WRITE (51,0005) (NAME(I), I = 1, 20)
    0005 FORMAT (1H120A4///)
CARD READ NO 2
    READ (5,20) THIK,ANGL,DELP,NCPJ,NPMI,NTERMS
C THIK=SAMPLE CONTAINER THICKNESS IN G/CM**2
C ANGL=ACCEPTANCE ANGLES AT SNOUT ENTRANCE IN DEGREES
C NCPJ=NO. COMPARISON POINTS (J)
C NPMI=NO. PULSE HEIGHTS, OR SPECTRAL ENERGY POINTS (I)
C NTERMS=NO IN INTEGRATION, ETC., ROUTINES
    WRITE (51,21) THIK,ANGL,NTERMS,DELP
    21 FORMAT(' SAMPLE-CONTAINER THICKNESS='F10.5,' G/CM**2/'
    X ' ACCEPTANCE ANGLE ='F9.3,' DEGREES/' ' NTERMS ='
    Y 13/' DELP='F10.4//)
    20 FORMAT (3F10.0,4I5)
    25 FORMAT (10I5)
CARD READ NO 3 WHICH SPECTRA TO PLT
    READ (5,20) IP,(NPL0T(I),I=1,IP)
    20 FORMAT(10I5)
CARD READ NO 4 IVAR IS A FORMAT FOR THE EFFICIENCY DATA
    READ (5,40) IVAR
    40 FORMAT (9A8)
CARD READ NO 5 COMPARISON POINT ENERGIES
    READ (5,IVAR) (CPE(I),I=1,NCPJ)
C CPE=COMPARISON POINT ENERGIES
    WRITE (51,41)
    41 FORMAT(' COMPARISON POINT ENERGIES')
    N3=NCPJ/3 +1
    DO 50 I=1,N3
    L=1
    K=L+2*N3
    IF (K.GT,NCPJ) K=K-N3
    WRITE (51,51) (CPE(J),J=L,K,N3)
    51 FORMAT(3F15.4)
    50 CONTINUE
CARDS FOLLOWING CPE-DATA BIN LIMITS AND WIDTHS
    DO 100 I = 1, NPMI
    100 READ (5,30) ELON(I), ENGH(I), EDEL(I)
    30 FORMAT (3F10.0)
    WRITE (51,0001)
    0001 FORMAT (1H01NGROUP STRUCTURE/)
    WRITE (51,0006)
    0006 FORMAT (1H 30W LEDGE WEDGE MEV/BIN/)
    WRITE (51,0002) (ELON(I), ENGH(I), EDEL(I), I = 1, NPMI)
    0002 FORMAT (1H 3F10.3)

```

```

CC SETS UP GROUP STRUCTURE
  K = 0
  DO 400 I = 1, NPHI
    NLU = (ENGM(I) - ELOW(I))/EDEL(I) + 0.0001
    DO 400 J = 1, NLU
      K = K + 1
      EC(K) = ELOW(I) + (J - 0.5)*EDEL(I)
      DELEC(K) = EDEL(I)/2.0
400 CONTINUE
  NRS = K
  WRITE (7,999)
999  FORMAT('FORMAT COL 00 (1A2.2I3.0E9.3) ')
C THIS IS HEADER CARD FOR PUNCHED OUTPUT
  DO 1000 K = 1, NCPJ
    CELE = CPE(K)*DELP
    IF (DELE.LT.0.301) DELE=0.001
    NSTD = IFIX(1.2*CPE(K)/DELE + 2./CPE(K))
    IF (NSTD.GT.2000) NSTD=2000
    DO 300 I = 1, NSTD
      XI = FLOAT(I - 1)
      E(I)=DELE+XI
300 RM(I) = 0.0
    DO 350 I = 1, 200
      BRM(I) = 0.0
350 BRM(I) = 0.0
C
C NOW CALCULATIONS OF RESPONSES FOR GIVEN COMPARISON POINT ENERGY
C
C CHANGED SEPT '76: DIFFERENT RESPONSE FOR SMALLER COLLIMATORS
  AUX=CPE(K)
  CALL BREREL (1,AUX,RNG)
  IF (RNG.LT.THIK) GO TO 500
  TK=RNG-THIK
  CALL BREREL (2,AUX,TK)
  GO TO 550
500 CALL BREREL (2,AX,THIK)
  AUX=CPE(K)-AX
  WRITE(51,510) AX,K,CPE(K)
510 FORMAT(' 000 THICKNESS EQUIVALENT TO 'E13.5, ' MEV, VS CPE('
  X 13,')= ' E13.5, ' MEV'//)
550 RE=AUX
  IF (RE.LT.0.04) RE=0.04
  REAV=0.5*(CPE(K)+RE)
  SIGMA=SQRT(25. + 120./REAV)*0.01*REAV/2.35402
  ATTN=ENGM2(ANGL,THENTR,RE)
C==
  ETAUX=ETA(AUX)
570 AREA=ATTN*DM*4*PI
  IF (CPE(K) .GT.4.0) AREA=AREA*(1.-((CPE(K)-4.0)/6.0)**2)
C  FFF EFF > 4 MEV ASSUME LOSS OF EFFICIENCY
  AREA=AREA
  DO 610 JJ=1,2
    SA=0.
    DO 600 J=1,NSTD
      EE1=E(J)
      CPD=AUX
      AD=ERESP(ARA,CPD,SIGMA,ETAUX,EE1)
      RM(J)=AD
      SA=SA+RM(J)
      SA=SA*DELE
610  AREA=AREA+ARA/SA
      CALL SLIT(RM,NSTD,E,CPE(K),AREA)
      DO 802 JPL=1,JP
        IF (K.EQ.NPLPT(JPL)) GO TO 804
802 CONTINUE
      GO TO 803
804 WRITE (51,820) CPE(K)
820 FORMAT (1M16HRESPONSE AT E = F8.4//)
      JK=1
      CALL PLT(RM,E,JK,NSTD,2)
803 SMC=0.
      DO 900 I = 1, NRS
        NORM=1
        AUX = EC(I)

```

```

E1 = AUX - DELEC(I)
E2 = AUX + DELEC(I)
CALL XINT (E,RH,NSTD,NTERMS,F1,F2,SUM)
IF (SMC.LT.SUM) SMC=SUM
BRN(I) = SUM
IF (SUM.LT.0.0) BRN(I) = 0.0
IF (SUM.LT. SMC*1.0D-09 .AND. E1.GT.CPE(K)) GO TO 950
900 CONTINUE
950 CONTINUE
IF (NRHM.GT.NRS) NRHM=NRS
WRITE (51,5000) K,CPE(K), AREA,CPO,SIGMA,ETAUX,ATTN,ETAUX
5000 FORMAT (1H16HCOMPARISON POINT 14., E='F0.4/
X ' AREA, EZER0, SIGMA, ETAUX, ATTN, ETA(E) ='1P0E11.4/)
WRITE (51,5001) (I, EC(I), BRN(I), I = 1, NRHM)
5001 FORMAT (1H 3(10, 0P2E15.5))
THISUM=0.0
DO 955 I=1,NRHM
955 THISUM=THISUM+BRN(I)
WRITE (51,7100) THISUM
7100 FORMAT('0 INTEGRATED EFFICIENCY ='1PE12.4,' COUNTS/UNIT-INT'/)
NCARD = NRHM/8 + 1
DO 960 I = 1, NCARD
NIND = (I - 1)*8 + 1
IF (NIND.GT.NRHM) GO TO 1000
NAUX = NRHM - NIND
IF (NAUX.GT.7) NAUX = 7
NSTOP = NIND + NAUX
WRITE (7, 5002) NIND, K, (BRN(L), L = NIND, NSTOP)
5002 FORMAT (2X, 2I3, 1P0E9.3)
960 CONTINUE
1000 CONTINUE
WRITE (7,1044)
1044 FORMAT('0END HT ')
2000 STOP
END

```

```

SUBROUTINE SLIT(RH,N,E,EBETA,AREA)
C TO CALCULATE EFFECT ASSUMED FOR LOW ENERGY SLIT SCATTERING FROM
C COLLIMATORS. ASSUME 10% AT 0.35 MEV AND 2% AT 3.5 MEV.
C (MEASURED PERHAPS 10% AT 0.35 MEV -- 113-SN)
C (CALCULATE ABOUT A FACTOR OF 5 FOR A DECADE IN ENERGY)
DIMENSION RH(1),E(1)
A=0.10*AREA/EBETA**3.2
DO 1 J=1,N
IF (E(J).GE.EBETA) RETURN
ED=SQRT(EBETA-E(J))
1 RH(J)=RH(J)+A*E(J)*ED
RETURN
END

```

```

FUNCTION ENONZ(TMIN,TN,E)
C TO GET ATTENUATION OF BETAS IN ENTRANCE FOIL REGION DUE TO NUCLEAR
C SCATTERING OF BETAS. THOFOIL THICKNESS IN G/CM**2, TMIN=MINIMUM THETA
C EB=INCIDENT BETA ENERGY. USE 2 FOR CARBON, THE FACTOR 0.7947E+23
C FOR POLYETHYLENE EQUIVALENT TO CARBON
DATA Z/6./
S=SIGBNT(2,EB,TMIN)
EN=EXP(-S*TH*0.7947E+23)
ENONZ=EN
RETURN
END

```



```

      FUNCTION SIGINT(Z,EBETA,TMIN)
C  INTEGRATE SIGINT FROM TMIN(DEG) TO 100 DEG TO GET INTEGRATED CROSS
C  SECTION FOR DECAY HEAT BETA SPECTROMETER, IN PARTICULAR,
C  FOR ELECTRON-NUCLEUS SCATTERING
C  NOTE EBETA IN MEV
      DATA THDP1/0.283105/, RAD/97.29578/
      TOT=0.
      DT=1./RAD
      L=FIX(TMIN+0.001)
      TS=DT*(FLOAT(L)+0.5)
      L=100-L
      DO 1 I=1,L
      TOT=TOT+SIN(TS)*SIGOND(Z,EBETA,TS)
1  TS=TS+DT
      SIGINT=THDP1*TOT*DT
      RETURN
      END

```

```

      FUNCTION SIGOND(Z,EBETA,TMETA)
      DATA CERAD/2.8E-13/, EMCSO/0.511/
      COMPUTE DIFFERENTIAL CROSS SECTION FOR SCATTERING OF MOVING BETA BY A NUCLEUS
C  BETA ENERGY IN MEV, TMETA IN RADIANS
C  SEE E. MARA, N.I.M., V.65, P.85 (1968)
      G=1.+EBETA/EMCSO
      GSQ=G*G
      BSQ=1.-1./GSQ
      SS=SIN(0.5*TMETA)
      SSO=SS*SS
      ZK=(Z*CERAD*0.5)**2
      DSIG=ZK*(1.-BSQ*SSQ)/(BSQ*G*SSQ)**2
      SIGOND=DSIG
      RETURN
      END

```

```

      FUNCTION BRESP(A,X0,SIG,SA,X)
C  BETA RESPONSE IN NE-110 - CAF2 "PHENOMENON". A GAUSSIAN + LOW-ENERGY
C  "TAIL" DETERMINED EMPIRICALLY
C  REVISED "TAIL" SEPTEMBER 1976
      DIMENSION T(45)
      DATA T/0.,0.00230,0.00477,0.00715,0.00954,0.01192,0.01431,
X  0.01669,0.01897,0.02145,0.02384,0.02731,0.03099,0.03457,
Y  0.03815,0.04172,0.0457,0.04967,0.05364,0.05811,0.06256,
Z  0.06705,0.07152,0.07740,0.08344,0.0894,0.09536,0.10132,
A  0.10720,0.11324,0.1192,0.12377,0.12705,0.13021,0.13213,
B  0.13341,0.12862,0.12069,0.11106,0.09814,0.075,
C  0.04650,0.024,0.00364,0.0/
      (NOTE - SUM OF "DATA" IS 2.75305.)
      DATA RT2PI/2.5066282/
      EQ=0.
      B=(X-X0)/SIG
      IF (ABS(B) .GT. 7.0) GO TO 2
      G=0.5+0.8
      EQ=EXP(G)
2  Y0=A/SIG/RT2PI
      Y=Y0*EQ
      IF (X-GE,X0 .OR. X-LE,0.405405*X0) GO TO 1
C== ASSUME "TAIL" INTEGRAL IS APPROX SA=A; ADJUST TAIL TO SUIT
C== (19-SEPT)
      TC=SA*SIG*0.6/X0
      XX=75.63*(X-0.405405*X0)/X0
      IXX=FIX(XX*0.495)
      IF (IXX.LT.1 .OR. IXX.GT.45) IXX=1
      TAIL=Y0*TC*Y(IXX)
      Y=Y+TAIL
1  BRESP=Y
      RETURN
      END

```


APPENDIX G. ABSOLUTE NORMALIZATION OF UNFOLDED SPECTRA
AND COMPARISON WITH CALCULATION;
THE CODES HEAT4 AND HEAT5

These codes process the unfolded data from FERD. The number of fissions, n_f , is input via teletype and the unfolded data are read in. If there is a comparison file (of calculated data) these may also be read in. The code prepares a graph of either or both of the differential spectra either for display or hard copy, and computes the two important integrals (total number and total energy) from the differential data. The last two data sets from FERD are the same two integrals obtained using appropriate "window functions" (see Section 6.D). These 4 integrals are typed out. In addition 2 partial integrals for $E_{\text{output}} \leq 0.28 \text{ MeV}$ are computed and typed out. This option was originally included to provide a direct comparison of our gamma-ray data with earlier ORNL²⁴ results, and has proved to be valuable in assessing the low-energy character of all of the present data, for example in the study of the ^{20}F beta decay shown in Fig. 29.

Details of the plotting code are outside of the scope of this report and are not presented here.

```

C THIS IS FILE HEAT4.F4
C PURPOSE TO INTEGRATE AND DISPLAY UNFOLDED OUTPUT
C ALSO DISPLAY ORIGIN TYPE CALCULATIONS IF AVAILABLE
C ALSO THIS ONE TO DIFFERENTIATE BETWEEN 176X176 AND 176X170
      DIMENSION E(170),DL(170),DU(170),TTITLE(5),T2(2)
      DIMENSION DORI(170),DC(170),DE(170),TTITLE(5),T23(2)
      COMMON /CRPA/ ICR,IPA
      DOUBLE PRECISION FILNAM,ORIGEN,FOLANK
      DATA FOLANK/10M /
      NWIN=1
      ICR=0
      IPA=0
      IEE=0
      WRITE (5,115)
115  FORMAT(' FOR 176 BY 170 DATA INPUT. EXTRA WINDOWS!'/)
      WRITE(5,116)
116  FORMAT(' IF NO EXTRA WINDOWS TYPE -1'/)
      READ (5,111) NW
      IF (NW.EQ.-1) NWIN=NW
      NTHEO=0
      WRITE (5,118)
118  FORMAT(' ORIGIN CAL FILNAME (A10) = 'S)
      READ (5,91) ORIGEN
      IF (ORIGEN.EQ.FOLANK) GO TO 1
      NTHEO=1
      WRITE (5,119)
119  FORMAT(' ORIGIN CALCULATIONS FOR'/)
      CALL IFILE(20,ORIGEN,*10,IEE)
      READ(20,92) TTITLE,T23
      WRITE (5,87) TTITLE,T23
      READ (20,93) NDATA
      READ (20,94) (E(I),DORI(I),I=1,NDATA)
      84  FORMAT(2E12.5,12X,2E12.5)
      1  WRITE (5,90) IEE
      90  FORMAT(14/' INPUT DATA FILNAME (A10) = 'S)
      READ (5,91) FILNAM
      91  FORMAT(A10)
      CALL IFILE(21,FILNAM,*10,IEE)
      WRITE (5,88)
      88  FORMAT(' INPUT NDELETE, NO-OF-FISSIONS AND TIRRAD (1/2E)'/)
      READ (5,89) NDELETE,ENF,TII
      89  FORMAT(1/2E)
      IF (ENF.LE.0.0) ENF=1.
      WRITE (5,102) ENF,TII
102  FORMAT(' ENF,TII =0P2E11.3/)
      11  READ (21,92) TTITLE, T2
      92  FORMAT(5A5,11X,2F6,1)
      WRITE(5,87) TTITLE,T2
      87  FORMAT (2X,5A5,' T1='F5.0.', T2='F6.0.' SECS'/)
      READ (21,93) NDATA
      93  FORMAT(110)
      READ (21,94) (E(I),DL(I),DU(I),I=1,NDATA)
      94  FORMAT(6E12.5)
      WRITE (5,79)
      79  FORMAT(' TYPE N=0 IF OKAY, OR N=1 FOR NEXT CASE'/) N= 'S)
      READ (5,111) NEN
      IF (NEN.EQ.1) GO TO 11
      N8=1
      IF (NDELETE.LE.0) GO TO 14
      N8=NDELETE+1
      DO 2 I=1,NDELETE

```

```

      DL(I)=0.
      DU(I)=0.
14   NDM2=NDATA-1-NWIN
      NDM1=NDM2-1
      DO 12 I=1,NDM1
12   DE(I)=(E(I+1)-E(I))/2.
      DE(NDM2)=DE(NDM1)
      K200=0
      SHAX=0.
      SUME=0.
      SUMN=0.
      DO 3 I=NDM,NDATA
      DL(I)=DL(I)/ENF
      DU(I)=DU(I)/ENF
      IF(NDM.GE.NDATA-NWIN) GO TO 3
      IF(NVNEQ.EQ.1) DC(I)=DORI(I)/ENF
      IF (DU(I).GT. SHAX) SHAX=DU(I)
      IF (K200.EQ.0 .AND. E(I).GT.0.20) GO TO 15
      GO TO 16
15   K200=1
      WRITE (5,86)
      86   FORMAT(' DATA FOR EGAMMA .LE. 200 KEV ')
      WRITE (5,295) SUMN,SUME
295   FORMAT(' SUM GAMMAS='1PE11.4,' SUM ENERGY='E11.4/)
16   CONTINUE
      SUMN=SUMN+DE(I)*(DL(I)+DU(I))
      SUME=SUME+DE(I)*(DL(I)+DU(I))*E(I)
3    CONTINUE
      WRITE (5,81) SUMN,SUME
      81   FORMAT(' INTEGRATED SUM GAMMAS='1PE11.4,'/FISSION'/
X      11X,' SUM ENERGY='1PE11.4,' MEV/FISSION'/)
      IF (NWIN.EQ.-1) GO TO 17
      SUMN2=0.5*(DL(NDATA-1)+DU(NDATA-1))
      SUME2=0.5*(DL(NDATA)+DU(NDATA))
      SUMNM=DU(NDATA-1)+SUMN2
      SUMEN=DU(NDATA)+SUME2
      WRITE (5,95) SUMN2,SUMNM,SUME2,SUMEN
      95   FORMAT(' WINDOW SUM GAMMAS='
Y      1PE11.4,' *OR= '1PE11.4,' /FISSION'/
Z      11X,' SUME ENERGY='1PE11.4,' *OR= '1PE11.4,' MEV/FISSION'//)
17   TT=0.5*(T2(2)+T2(1)+T11)
      TD=T2(2)-T2(1)
      ST=TT+SUMN/TD
      SE=TT+SUME/TD
      WRITE (5,105) TT,TD,ST,SE
105  FORMAT(' TPAR='F9.2,' SECS. TCRUNT='F8.1,' SECS'/5X
X      'WEIGHTED NUMBER='E13.4,'/FISSION'/5X
Y      'WEIGHTED ENERGY='E13.4,' MEV/FISSION'/)
      NT=10.
      N=0.0
4    WRITE (5,96) ICR,IPA
      96   FORMAT(' ICR='I2,' , IPA='I2,' , NEW ICR,IPA= '5)
      READ (5,97) ICR,IPA
      97   FORMAT (2I)
      IF (ICR.EQ.0 .AND. IPA.EQ.0) GO TO 11
5     T2=1.2*SHAX
      KE=1
      Y1=0.001*Y2
      NY=1
      X1=0.0
      X2=0.0

```

```

      KX=0
      IF (ICR.NE.2) GO TO 6
10     KE=2
      ICR=1
      CALL VMP0(1)
      CALL XTAKES(X1,X2,KX,Y1,Y2,KY,MY,-N)
      IF (KE.EQ.2) GO TO 21
      CALL LABPLT(2,44,'                      YIELD (PHOTONS/MEV/FISSION)')
      GO TO 19
21     CALL LABPLT(2,21,'          PHOTONS/FISSION')
19     CALL LABPLT(1,32,'          ENERGY (MEV)')
      PS=Y1
      DO 7 I=ND,NDM2
      YL=DL(I)
      YU=DU(I)
      IF (KE.EQ.1) GO TO 22
      YL=YL+E(I)
      YU=YU+E(I)
      IF (NTHED.EQ.1) DC(I)=DC(I)+E(I)
22     YB=0.5*(YU+YL)
      XL=E(I)+DE(I)
      XU=E(I)+DE(I)
      IF (YB.LT.PS) GO TO 8
      YSV=YU
      YU=YL
      YL=YSV
      CALL LINE(1,E(I),YU)
      CALL LINE(2,E(I),YL)
      CALL LINE(1,XL,YB)
      CALL LINE(2,XU,YB)
      PS=YB
      IF (NTHED.NE.1) GO TO 24
      CALL CURVE(E(1),DC(1),NDDATA=1,NDDATA=0,0,0,04)
24     CALL LABCRV(1,1,15,TITLE,0,0,0,14)
      CALL LABCRV(2,1,10,TITLE(4),0,0,0,14)
      IF (KE.EQ.2) CALL LABCRV(4,1,13,' ENERGY CURVE',0,0,0,14)
      CALL PLOTXX(W,X1,X2,KX,Y1,Y2,KY,NDB)
      IF (NDB.GE.1) GO TO (5,6,11,20), NDB
20     WRITE(5,110)
110    FORMAT(' N=0 TO OUT'// ' =1 NEXT GROUP'//
      X ' ' =2 FOR ENERGY PLOT'/4X'N=? 'S)
      READ(5,111) N
111    FORMAT(11)
      IF (N=1) 10,11,18
10     CALL VNELL0
      STOP
      END

```

APPENDIX H. CALCULATIONS OF SHAPES OF BETA-RAY SPECTRA;
THE CODE ELECSP

The electron (β^+ or β^-) spectrum for an n -times forbidden transition can be written as:⁶⁷

$$N(W)dW = KF(Z,W) p W (W_0 - W)^2 S_n(W)dW \quad (H.1)$$

where $W = 1 + E_\beta/mc^2$, W_0 corresponds to the maximum energy beta, $F(Z,W)$ is the Fermi function which is important for all but the lightest elements, p is the β -ray momentum and S_n is the shape factor. K is a constant, which is treated here as a normalizing parameter. For allowed transitions $S_1(W) = 1.0$ for all W . In this case a plot of

$$[N(W)/F(Z,W)pW]^{1/2} \text{ vs } (W_0 - W)$$

yields a straight line. A curve constructed in this fashion is called a Kurie (or Fermi, or Fermi-Kurie) plot, and any deviation from a straight line is attributable to S_n being not constant with W . For certain types of transitions S_n has an analytic expression. These are called Unique Forbidden and are characterized by $\Delta J = n + 1$. For these cases the shape factors are⁶⁸

$$S_1 = (W^2 - 1) + (W_0 - W)^2 \quad (H.2)$$

$$S_2 = (W^2 - 1) + (W_0 - W)^4 + 3.33(W^2 - 1)(W_0 - W)^4 \quad (H.3)$$

$$S_3 = (W^2 - 1)^3 + (W_0 - W)^6 + 7(W^2 - 1)(W_0 - W)^2 [(W^2 - 1) + (W_0 - W)^2] \quad (H.4)$$

The Fermi function takes into account the Coulomb force on the electron. Treating the electron nonrelativistically, an approximate

expression is obtained⁶⁹ for $F(Z,E)$:

$$F(Z,E) = \frac{2\pi X}{1 - \exp(-X)} \quad (\text{H.5})$$

where $X = Ze^2/\hbar v$ for electrons, v being the speed of the electron far away and Z the atomic number of the daughter (product) nucleus. This expression is satisfactory (to $\pm 3\%$) for $Z < 30$ and electron energies a few MeV or less. A more generally useful approximation⁷⁰ can be written as

$$F(Z,E) \approx F(Z,E) \frac{W^2(1 + 0.000852 Z^2) - 1}{4}^S \quad (\text{H.6})$$

where

$$S = (1 - 0.0000533 Z^2)^{1/2} - 1. \quad (\text{H.7})$$

This approximation is satisfactory (to $\pm 2\%$) for $Z < 75$. This additional factor is included in the routine ELECSP.

```

C THIS IS FILE ELECSP.F4
C PURPOSE IS TO COMPUTE BETA-RAY SPECTRUM
1  WRITE (5,7)
7  FORMAT(' EMAX(MEV) = '5)
   READ (5,9) EMAX
8  FORMAT(F)
   IF (EMAX.LE.0.0) GO TO 4
   WRITE (5,17)
17  FORMAT(' UNIQUENESS, (!-FORMAT) ? '5)
   READ (5,18) II
18  FORMAT(I)
   WRITE (5,16)
16  FORMAT(' Z(DAUGHTER)= '5)
   READ (5,9) Z
   WRITE (25,9) EMAX,II,Z
9  FORMAT(' EMAX= '0PF0.3,' MEV, FERRID= '12,', Z (DAUGHTER) = '
X  F7.1/' P EN(P) '
H  ' E=BETA EN(E) SUM EN(E) * E '
Y  ' SUM(EN(E) * E)')
   DE=0.04*EMAX
   UO=1.0*EMAX/0.511
   E=0.5*DE
   S=0.
   SE=C.
2  M=1. *E/0.511
   WM=WM-1.0
   WA=(M0-M)*0.2
   ENP=WM*FZE(Z,E)*WA
   P=SQRT(WM)
   JJ=JJ+1
   GO TO (20,21,22,23) JJ
20  FRRB=1.0
   GO TO 30
21  FRRB=WM*(M0-M)*0.2
   GO TO 30
22  FRRB=WM*WM*WA*WA*10.0*WM*WA/3.0
   GO TO 30
23  FRRB=WM*0.3*WA*0.3*7.0*WM*WA*(WB+WA)
30  CONTINUE
   ENE=ENP*WM*FRRB/P
   S=S+ENE
   EE=ENE+E
   SE=SE+EE
   WRITE (25,10) P,ENP,E,ENE,S,EE,SE
10  FORMAT(1P2E12.5,2X.5E12.5)
   E=E+DE
   IF (E.LT.EMAX) GO TO 2
   WRITE (25,11)
11  FORMAT(//)
   GO TO 1
4  END FILE 25
STOP
END
FUNCTION FZE(Z,E)
C COMPUTE COULOMB CORRECTION TO FIRST ORDER. SEE BLATT-WEISKOPF
C PG 680. GOOD FOR Z<30 AND E> "A FEW HUNDRED KEV"
DATA ESQW/2.18F0/C/2.988E+10/
GAMM=1.0*E/0.511
G=2/137.0
S=SQRT(1.0-G*G) -1.0
VEE=C*SQRT(1.0-1.0/GAMM**2)
CC=(GAMM*GAMM*(1.0+4.0*G*G) -1.0)/4.
Z2=6.2832*7*ESQW/VEE
FZE=Z2*CC**S/(1.0*EXP(-Z2))
RETURN
END

```

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