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Nuclear Safeguards

Research and Development

Program Status Report
October-December, 1969

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G. Robert Keepin, Editor

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LOS ALAMOS SCIENTIFIC LABORATORY of the University of California LOS ALAMOS • NEW MEXICO

Nuclear Safeguards Research and Development

Pulsed Neutron Research Group, N-6

G. Robert Keepin, Group Leader

Program Status Report October-December, 1969

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Nuclear Safeguards Research and Development

NEUTRON INTERROGATION AND NONDESTRUCTIVE ASSAY OF FISSIONABLE MATERIALS

Delayed-Neutron Assay of Fissionable Material in 1-Gallon Recovery Cans

R. H. Augustson and H. O. Menlove

For the application of delayed-neutron assay techniques to fissionable materials in large sample containers, it is important to determine the effect of the nonfissionable matrix materials on the assay results. This information should make it possible to minimize the sensitivity of the assay technique to the physical form, chemical composition, and spatial distribution of the material. The present investigations are being performed with gallon can containers, because of the large number of these cans containing Pu scrap at the Dow Chemical Facility, Rocky Flats, Colorado. ⁽¹⁾

Due to possible neutron background from $^{17}\text{O}(n, p)^{17}\text{N}$ delayed neutrons (in addition to the $^{19}\text{F}(\alpha, n)$ background), it is desirable to interrogate the fissionable sample using a neutron moderator which will increase the delayed neutron response and decrease the neutron background from ^{17}O . To this end, a moderator with a Pb (2-4 in. thick) core and a graphite (3-6 in. thick) reflector was used. The 1-gallon can sample was placed in the interior of the moderator in a Cd-lined cavity with 1-in. thick steel walls. The purpose of the

steel and Cd was to harden the energy spectrum of the interrogating neutrons, and thus improve penetrability. The Cd ratio in the sample cavity for Pu fission was ~ 1.3 , and the delayed-neutron response from 1 g of Pu was equivalent to the response from 2000 g of oxygen. The presence of the neutron moderator increased the delayed-neutron signal by a factor of ~ 14 over that with no moderator and decreased the ^{17}O background neutrons by a factor of ~ 2.3 , thus improving the signal-to-background ratio by ~ 32 . An additional advantage of the moderating assembly was that the delayed-neutron response was relatively flat throughout the volume of the sample cavity, thus making the assay relatively insensitive to the spatial distribution of the fissionable material.

The interrogating neutrons were produced by Accelerator I with a $\sim 50\%$ duty cycle, ⁽²⁾ and the delayed-neutron response was measured per monitored source neutron using the thin slab detector described on page 13 of this report. In order to measure the effect of the various matrix materials on the Pu fission rate in the can, two types of fission probes were used. A small Pu fission chamber was placed in the moderator cavity along side the gallon can at 90° to the accelerator beam axis. In addition, a 1-mil-thick ^{235}U foil was wrapped

¹ cf. Rocky Flats Trip Report, this Progress Report.

² LA-4070-MS, pp. 4-5 (1968).

around the exterior of the can to act as an "in-situ add-a-gram" probe. The fission detector response indicates the change in neutron flux produced by the matrix material in the can whereas the delayed neutron response from the in situ add-a-gram reflects both the change in the interrogating neutron flux as well as the delayed-neutron absorption by the matrix material.

The different materials that were investigated are listed in the first column of Table I, and the second and third columns list the delayed-neutron response from the add-a-gram foil and the Pu fission detector response, respectively. Both of these responses have been normalized to unity for the case of an empty can; thus the deviation from unity represents the net effect of the added matrix material.

It can be seen that the perturbation in the in situ add-a-gram response is roughly a factor

of two less than for the fission chamber response. This is due to the moderation and absorption in the matrix material of the delayed-neutrons from the add-a-gram foil. The larger matrix effects are caused by the hydrogen content in the samples and the first column of Table I gives the effective hydrogen density (ρ_H) in grams hydrogen per unit volume for some of the samples.

To observe the delayed-neutron response as a function of position in the can, irradiations were performed where a thin metallic Pu disc (0.01 x 2-in. dia) was moved throughout the volume of the can and the can was slowly rotated about its axis of symmetry. Figure 1 shows a plot of the delayed-neutron response as a function of radius for the different matrix materials. It can be seen that there is very little radial variation in the response for cans which do not contain hydrogen. The delayed-neutron response increases for in-

TABLE I
MATRIX MATERIAL EFFECTS ON DELAYED-NEUTRON RESPONSE
OF FISSIONABLE MATERIAL IN 1-GALLON RECOVERY CANS

<u>Matrix Material</u>	<u>Normalized Response</u>	
	²³⁵ U Foil (add-a-gram)	²³⁹ Pu Fission Detector
Empty Can	1.00	1.00
Pb Shot	1.00	1.12
Metal Parts *	1.04	1.11
Carbon Pellets	1.15	1.32
Sand-slag-crucibles *	1.23	1.41
Concrete *	1.30	1.69
String-flo-filters *	1.96	3.13
Poly-chips ($\rho_H = 0.0093$ g/cc)	1.89	2.80
Poly-chips ($\rho_H = 0.027$ g/cc)	4.08	7.75
Plastic-benelux-leadglass *	5.00	10.0
Water ($\rho_H = 0.11$ g/cc)	5.12	13.6

* Matrix materials supplied by Dow Chemical Corporation, Rocky Flats, Colorado.

Fig. 1. Measured delayed neutron response-vs-radius for 1-gallon cans with Pu distributed in various matrix materials (representative of those found in some industrial nuclear processing and recovery facilities).

creasing amounts of hydrogen, tending to saturate for the full density ($\rho_H = 0.11$ g/cc) case. It should be noted that most of the matrix materials contained in gallon cans at Rocky Flats correspond to the region of the lower four curves in Fig. 1. In practical assay applications, a calibrated fission chamber placed next to a gallon can would indicate which response curve (see Fig. 1) should be used to determine the amount of fissionable material in the can. To investigate the uniqueness of the fission response at the exterior surface of a can to the delayed-neutron response in the can's interior, low density polyethylene was added to a can so that the Pu fission chamber response was identical to the case of the slag and crucible can. The delayed-neutron response was then measured throughout the volume of both cans with identical results for both materials.

It was concluded from these investigations that for the materials in gallon cans to be assayed

at Rocky Flats, a fission detector probe at the surface of the can will be adequate to correct for the effects of unknown matrix materials. For example, the delayed-neutron response for the lower four curves in Fig. 1 is quite flat as a function of position, and the fission response at the boundary gives the response in the can's interior, so it should be possible to assay the contents of an unknown can to $\sim \pm 10\%$, and where good standards are available, assay accuracy to within a few percent should be attained routinely.

In view of the demonstrated effectiveness of this delayed-neutron technique for large sample assay, a Pb-C neutron moderating assembly similar to that described in this report is being designed for use in the Mobile Nondestructive Assay Laboratory.

Delayed Neutron Assay of Safeguards Inventory Samples

A. E. Evans, R. H. Augustson, H. O. Menlove, and R. B. Walton

In order to check the accuracy and reliability of the subthreshold-neutron-interrogation, delayed neutron-response technique for assaying small samples, the eight plutonium inventory verification samples received from the New Brunswick Laboratory were assayed several times, using interrogating neutrons of different energies and testing various geometrical arrangements. Some of the results (for PuO_2 and Pu metal oxidized samples) are presented in Table II, together with prior results from passive gamma scanning for ^{239}Pu content and from passive coincidence neutron counting for ^{240}Pu content.

Subthreshold neutrons were produced by bombarding 200- or 400-keV-thick lithium targets with 2.1 or 2.4-MeV protons from the N-6 Van de Graaff. The relative neutron yield was monitored using a BF_3 proportional counter. Samples to be assayed were placed 10-cm in front of the target and were irradiated with 40-msec pulses of neu-

TABLE II

COMPARISON OF ACTIVE AND PASSIVE ASSAY RESULTS FOR
Pu INVENTORY SAMPLES FROM NEW BRUNSWICK LABORATORY

Sample (No.)	Total Weight (g)	²³⁹ Pu Weight		²⁴⁰ Pu Weight
		Delayed-Neutron Yield (g) ^a	414-keV γ (g)	Passive-Neutron Coincidence(g)
PuO ₂ powder (Pu#217)	2.2908	1.84 $\pm$ 0.024	1.95 $\pm$ 0.080	0.244 $\pm$ 0.0037
PuO ₂ powder (Pu#645)	2.0835	0.98 $\pm$ 0.03	0.93 $\pm$ 0.047	0.054 $\pm$ 0.0008
PuO ₂ powder (Pu#646)	2.0821	1.12 $\pm$ 0.03	1.12 $\pm$ 0.061	0.062 $\pm$ 0.0062
Pu metal (Pu#605)	1.3529	1.11 $\pm$ 0.02	1.16 $\pm$ 0.064	0.078 $\pm$ 0.0011

^a The delayed-neutron results were obtained with interrogation neutrons having average energies of both 250 and 450 keV using the N-6 Van de Graaff accelerator.

trons, followed by 10-msec "wait periods" and 50-msec counting periods. Delayed neutrons were counted with a slab detector centered 5-cm from the sample and oriented either at 0° or at 60° to the axis of the beam. Neutrons from spontaneous fission of ²⁴⁰Pu and from (α , n) reactions were measured by preirradiation neutron counting of the samples.

For each assay run, the delayed-neutron count per gram of plutonium per BF₃ monitor count was determined by irradiating a standard sample containing 1.65 g of ²³⁹Pu, encapsulated in the same manner as the inventory samples. The standard was counted at the beginning and at the end of each run.

Similar assay measurements are being carried out on PuO₂-UO₂ pellets, powder and scrap received from the New Brunswick Laboratory. The effect of the depleted uranium on the assay of these samples requires further study, following which complete assay results on all New Brunswick Pu samples will be reported.

²⁵²Cf Source for Neutron Interrogation and Assay

D. B. Smith and H. O. Menlove

A 1-mg ²⁵²Cf source ($\sim 2 \times 10^9$ n/sec) has been acquired by Group N-6 and a program is underway at the LASL Nuclear Safeguards Research Laboratory to investigate ²⁵²Cf as an interrogating neutron source for the nondestructive assay of fissionable materials. A series of measurements and computer calculations have been undertaken to determine the spectral characteristics of ²⁵²Cf neutrons in various moderating assemblies, ⁽³⁾ as well as the possibility of using energy-biased neutron detectors to count prompt fission neutrons induced in the fissionable material by the ²⁵²Cf source.

The absolute neutron yield of the ~ 1 -mg ²⁵²Cf source was recently measured by a direct comparison with a 0.1 μ g ²⁵²Cf source which had been previously calibrated in the Los Alamos

³ LA-4315-MS, pp. 6-7 (1969).

Graphite Pile. (4) A ^{238}U spiral fission chamber was used to compare the two neutron sources giving a ^{252}Cf yield of 1.66×10^9 ($\pm 10\%$) n/sec (on January 8, 1970) which can be compared with the yield of 1.90×10^9 n/sec predicted by ORNL from measurements prior to the source encapsulation.

For many assay applications, it is important to know what fraction of the bombarding source neutrons comes directly from the ^{252}Cf and what fraction comes from "room-return" neutrons. To obtain this information, fission detectors containing back-to-back foils of ^{235}U and ^{238}U were systematically positioned at various distances from the ^{252}Cf source which was located ~ 1 m away from the nearest wall in the hot cell ($3.4 \times 5.2 \times 15.2$ m). If there were no room-return neutrons, the counting rate ratio of the two fission detectors would remain essentially constant as the source-to-detector distance is increased. However, when room-return neutrons (degraded in energy) are present, the counting rate ratio of $^{235}\text{U}/^{238}\text{U}$ increases with increasing source-to-detector distance. The results of the measurements are shown in Fig. 2 and it can be seen that at a distance of 27 cm (no Cd) about half of the total number of fissions in a ^{235}U sample would be induced by room-return neutrons.

Several calculations and measurements have been made to investigate the possibility of detecting prompt fission neutrons resulting from irradiation of a sample by the ^{252}Cf neutron source using a suitable spectrum-tailoring assembly and a biased neutron detector. Two moderating assemblies were considered: (5) a 20-cm radius ^9Be assembly having an internal irradiation position

Fig. 2. Measurements of room-return neutron contribution for ^{252}Cf neutron interrogation source in hot cell facility of the LASL Nuclear Safeguards Research Laboratory.

and a ^9Be -Pb assembly with external irradiation, i.e., in the leakage spectrum of the assembly. The present investigation has included computation of the response of two possible biased detectors (a ^{238}U fission chamber and a ^4He proportional counter) to prompt fission neutrons and background source neutrons in the spectra of both the above assemblies and in the internal spectrum of an additional ^9Be assembly having 0.5 in. of polyethylene on each side of the sample-detector position. The computations assumed one gram of ^{235}U and a 2-cm sample-to-detector distance. The resulting response ratios for the two assemblies having internal sample-detector positions are shown in Fig. 3. The ^{238}U fission chamber is "naturally" biased at the ^{238}U fission threshold, and thus its response ratio does not change with "bias" energy. The ^4He proportional counter takes advantage of the broad resonance in the ^4He cross section at about 1.1 MeV; this produces the peak in the response ratio at a bias energy of about 1.5 MeV as shown in the figure. The response ratio for both detectors is larger in the tailoring assembly having polyethylene adjacent to the sample, since this configuration will tend to increase the low energy

⁴ A. C. Graves and D. K. Froman, "Miscellaneous Physical and Chemical Techniques of the Los Alamos Project", p. 107, McGraw-Hill, New York (1952).

⁵ LA-4315-MS, p. 6 (1969).

Fig. 3. Ratio of prompt neutron response from fission of one gram of ^{235}U relative to the background of source neutrons in a ^{252}Cf moderator assembly for two possible biased detectors. (Schematic of geometry shown below.) The two lower curves refer to a ^9Be moderator (20-cm radius) surrounding the ^{252}Cf source; the upper curves are for the same moderator with one-half inch of CH_2 on each side of the sample-detector position.

flux and thus the fission density in the sample.

To measure the signal/background ratio using a ^{238}U fission detector, a Be pile with a CH_2 reflector was set up in the hot cell which contains the ^{252}Cf source. For the irradiations, the ^{252}Cf source was placed in the center of the Be pile and a thin Pu sample was positioned in the moderating assembly ~ 12 cm from the ^{252}Cf source. This gave a signal/background ratio of 0.78 for a 1-gm Pu sample, which compares well with the calculations.

The curves shown in Fig. 3 include ^{235}U fissions occurring at all neutron energies. The corresponding ratios for epi-cadmium fission only are an order of magnitude or more lower. Thus, it appears that this technique could be practical only for samples in which thermal neutron penetration is not a problem.

The computed detector signal/background ratios were a factor of two or more lower for irradiation in the leakage spectrum of the ^9Be -Pb spectrum-tailoring assembly than for either of the assemblies shown in Fig. 3, and do not indicate a very practical technique. However, irradiation outside the moderating assembly should permit shadow-shielding of the detector from the source neutrons. This possibility will be investigated during the next quarter.

In addition to the above computations, a series of transport calculations has been undertaken to investigate the design of a combined moderator and shield assembly for the ^{252}Cf source. It is proposed to irradiate small samples at some position within a moderating assembly surrounding the ^{252}Cf source. These samples would then be transferred from the assembly for delayed neutron (or activation-gamma ray) counting. The material surrounding the moderator core would serve as personnel (and perhaps also detector) shielding and as a reflector to increase the neutron flux at the irradiation position.

This series of calculations was performed using the one-dimensional neutron transport code DTF-IV with spherical geometry and the 16-group Hansen-Roach cross sections.⁽⁶⁾ Three moderator-shield configurations have been considered. The first consists of a 15-cm-radius beryllium moderator followed by a 10-cm-thick polyethylene reflector-shield; the second is similar but has a 10-mil sheet of ^{235}U between the beryllium and

⁶ G. E. Hansen and W. H. Roach, LA-2543-MS (1961).

the polyethylene to take advantage of neutron production from fission of the ^{235}U and thus increase the fast neutron flux in the moderator. The results of these two calculations are given in Figs. 4 and 5. The two solid curves in Fig. 4 show the total neutron flux per unit source as a function of radius in the assembly for neutrons having energies above and below the cadmium cutoff energy. The vertical line at 15-cm in Fig. 4 shows the position of the Be-CH₂ interface. The dashed curve indicates the sub-cadmium flux in 15-cm of Be with no

reflector. This flux falls off rapidly near the edge of the moderator as expected. The addition of the reflector not only increases, but also very effectively "flattens" the sub-cadmium flux throughout the moderator. The epi-cadmium flux decreases by several orders of magnitude across the moderator. The addition of the ^{235}U foil at the Be-CH₂ interface is expected to increase and "flatten" this epi-cadmium flux near the edge of the Be moderator. The expected effect does occur, but a single 10-mil foil does not sufficiently increase the epi-

Fig. 4. Neutron flux produced by a ^{252}Cf fission source in a moderator-shield assembly of beryllium and polyethylene. The dashed curve indicates the sub-cadmium ($E_n < 0.3$ eV) flux in a beryllium moderator having no polyethylene reflector.

Fig. 5. Neutron flux produced by a ^{252}Cf fission source in a moderator-shield assembly of beryllium and polyethylene having a 10-mil thick layer of ^{235}U between the Be and CH₂ as indicated.

cadmium flux, as seen in Fig. 5. Several foils of ^{235}U separated by layers of moderator should, however, produce a very effective multiplier for the ^{252}Cf source. This possibility, as well as other possible multiplier configurations, will be pursued in the course of further studies on ^{252}Cf as a neutron interrogation source.

The third moderator-shield configuration considered was similar to the first with a 5-cm-radius lead core replacing the first five centimeters of beryllium. The lead was added to provide shielding for the ^{252}Cf spontaneous-fission gammas. The results of the transport calculation indicate, however, that the substitution of lead reduces the sub-cadmium neutron flux in the moderator region by nearly a factor of two, and it appears more profitable to shield the gammas exterior to the Be or perhaps the entire moderating assembly.

Delayed Neutron Assay of Rover Scrap Material

T. D. Reilly, M. M. Thorpe, and R. B. Walton

In the continuing program of developing methods for nondestructive assay of Rover scrap ⁽⁷⁾ a rather large discrepancy between chemical and nondestructive assay arose in the case of one scrap container (recovery can No. 101). Rover scrap, it will be recalled, is a 4-20 wt% mixture of enriched uranium in graphite dust. The amount of uranium in can No. 101 as determined by nondestructive assay of the entire can differed significantly from that obtained by chemical analyses of small samples taken from the can. The contents of the can, which had been carefully blended, were sampled using a twelve-chute riffler. Four of the five samples taken were analyzed chemically and the fifth was assayed using nondestructive techniques. The results on all five of these samples agreed within $\pm 2\%$ in the determination of grams

uranium per gram sample. However, the nondestructive assay of the entire can gave a value which was more than 25% higher than that indicated by the small sample assays.

In an effort to resolve this inconsistency, a passive gamma ray measurement of material from three of the cans was performed. For this experiment, material from cans 101, 102, and 220 was repackaged in a 12 in. diameter by 2 in. thick disc containers which provide good ("thin" in one dimension) geometry for both gamma counting and delayed neutron measurements. Cans 101 and 102 consist of actual Rover scrap, while 220 was a standard prepared by mixing enriched uranium (U_3O_8) with graphite flour.

The intensities of the 0.765 MeV and 1.001 MeV lines from the ^{238}U in each disc container were measured with a 50 cc Ge(Li) detector at a distance of one foot. Typical counting statistics were $\pm 1.4\%$ for the 1.001 MeV gamma ray and $\pm 2.4\%$ for the 0.765 MeV gamma ray. The attenuations of the observed ^{238}U gamma rays were derived from the measured attenuations of 0.511, 0.835, and 1.275 MeV gamma rays from ^{22}Na and ^{54}Mn calibration sources. Corrections for attenuation, which were small, were then applied to the count rates to obtain numbers proportional to the ^{238}U mass in each can.

To determine the total uranium content from ^{238}U assay results, one must know the ^{238}U isotopic fraction in each can. These values were obtained from mass-spectrometer analyses furnished through Group CMB-8, LASL.

The results of the experiment are summarized in Table III. The total mass of uranium in each of the cans, 101 and 102, was obtained by normalizing to the known mass of uranium in the standard, #220. The delayed neutron assay results were obtained by neutron interrogation of the disc containers using the N-6 Van de Graaff accelerator as the neutron source and monitoring neutron

⁷ LA-1162-MS, pp. 10-12 (1969).

TABLE III

STUDY OF ASSAY METHODS FOR ROVER SCRAP

	Chemical Analysis	Passive Gamma	Delayed Neutron Assay (V. D. G. Source) (14-MeV Source)	
U Mass Ratio (Can 101/Can 102)	0.94	1.10 ± 0.06	1.09 ± 0.03	1.13
Mass (Can 101)	974 g	1184 ± 63	1240 ± 30	1326
Mass (Can 102)	1036 g	1074 ± 58	1140 ± 27	1172
Ratio to Gamma Assay:				
Can 101	0.82 ± 0.04	1.00	1.05 ± 0.08	1.12
Can 102	0.96 ± 0.05	1.00	1.06 ± 0.08	1.09

flux with two ^{235}U fission chambers. Also included in Table III are previous delayed neutron assay results ⁽⁷⁾ (14-MeV neutron interrogation) obtained when the Rover scrap was contained in the original recovery cans.

The masses obtained from passive gamma ray and delayed neutron measurements agree within statistics and are consistent with chemical assay for can 102 but differ substantially (~25% larger) for can 101. The good agreement between the small sample assays, as well as the good agreement between the various passive and active methods of nondestructive assay of an entire can,

(note particularly the agreement in the U mass ratio, can 101/can 102) suggest that the riffle-sampling procedure, although quite consistent, may introduce a systematic error or bias into the assay result. More experimentation (perhaps an analysis involving the complete dissolution of the entire contents, 6 kg, of can 101) is required before definitive conclusions can be drawn. Additional cans of Rover scrap will be repackaged in the 2-in. thick disc geometry and assayed nondestructively in order to further examine this problem.

PASSIVE ASSAY TECHNIQUES AND APPLICATIONS

2.6 MeV ^{208}Tl Gamma Ray Signature

T. E. Sampson (W-7), J. T. Caldwell, J. J. Malanify, and C. N. Henry

The possible use of 2.6 MeV gamma rays emitted from Pu and enriched ^{235}U materials for assay purposes is being investigated. This gamma ray results from the de-excitation of the first ex-

cited state of ^{208}Pb which is populated by β decay of ^{208}Tl . Any higher isotope satisfying the relationship $A = 208 + 4n$ (n = an integer) can produce this activity via successive α and β decays. In the U and Pu regime, the possible 2.6-MeV parents would be ^{232}U , ^{236}U , ^{236}Pu , ^{240}Pu , etc. Calculations of relative contributions from

Fig. 6. ^{208}Tl activity vs time after chemical separation of parent ^{232}U .

these parents show that only ^{232}U and ^{236}Pu contribute significantly. The specific activity (disintegrations per gm per sec) of these isotopes as a function of "grow-in" time is shown in Figs. 6 and 7. Assuming ^{232}U or ^{236}Pu concentrations of $\sim 10^{-10}$, an activity of ~ 10 disintegrations per sec per gm total U or Pu material is reached in only 1 to 4 years, with equilibrium values of a few tens of disintegrations per sec per gm. Preliminary measurements using some typical Pu and enriched ^{235}U material verify that the specific 2.6 MeV gamma activity of both materials is ~ 10 per sec per gm.

Both ^{232}U and ^{236}Pu are produced as a result of fast neutron reactions. In the case of ^{232}U which is present as a minor isotope in enriched ^{235}U , it is likely that most, if not all, of the production results from ^{236}Pu decays ($\tau = 2.85$ years) following fast neutron irradiation of ^{238}U in a reactor. This implies a rather complicated production history for the subsequent 2.6 MeV gammas, and possibly a characteristic unique specific activity for each different production batch of Pu and ^{235}U . Since the specific activity is a function of time, some idea of the history of the materials is necessary for proper interpretation of a measurement of specific activity. However, the possible use of this activity as a tag does appear worthy of further investigation, and

Fig. 7. ^{208}Tl activity vs time after chemical separation of parent ^{236}Pu .

work on the 2.6 MeV gamma ray as a possible safeguards signature is continuing.

Passive Assay of ^{235}U in Some Project Rover Discards

R. B. Walton and A. C. Dumrose (CMB-8)

The utility of portable passive instrumentation for field applications has been further demonstrated in the assay of ^{235}U content of miscellaneous discards from a recent overhaul of LASL's CMB-6 press facility for U-graphite fabrication (Project Rover). These measurements were similar to the Pu assay of stainless steel containment spheres reported previously.⁽⁸⁾ An Eberline PRM-5-3 portable unit, which contains an amplifier, single-channel analyzer, high-voltage supply, and a rate meter, together with a 2 x 2 in. NaI probe was employed. Although a smaller NaI

⁸ LA-4162-MS, pp. 12-13 (1969).

crystal (~ 1 in. $\times \frac{1}{2}$ in.) yields optimum signal-to-background for ^{235}U , the available probe proved adequate when appropriately shielded. The probe was placed inside a lead collimator with an angular aperture of $\sim 55^\circ$. In addition to shielding the crystal from background radiation, the collimator defined the scan area, i. e., the area from which gamma rays would be accepted for detection. Because the collimator weighed ~ 20 lbs, it was attached to a steel rod which served as a handle and a "walking-stick" support. The single channel window of the analyzer-rate-meter unit was set to bracket the photopeak of the 185-keV gamma ray from ^{235}U , and the unit was subsequently calibrated absolutely over a range of angles and distances with a 1.7 gm ^{235}U foil. The minimum

sensitivity of this detection system under field conditions (near the fabrication building and adjacent to the large pile of discards under investigation) varied from 30 mg at a sample-detector distance of 1 ft to ~ 0.5 gm at 4 ft.

For most of the discards, which were primarily ducts (of lengths 2 ft to 10 ft), work hoods and filters, the gamma-ray attenuations, as determined with the ^{235}U calibration foil, were small. The complete assay of 24 items was performed in $2\frac{1}{2}$ hours. Only one item contained sufficient ^{235}U to warrant, economically, its removal from the discard batch for future recovery. The total ^{235}U measured in this assay, 57 ± 25 gms, will be credited to the LASL Rover SS materials account.

BASIC DELAYED NEUTRON PARAMETER MEASUREMENTS

Delayed Neutron Yield-vs-Energy Measurements

M. S. Krick and A. E. Evans

Measurements of absolute total delayed neutron yields as a function of the energy of the neutron inducing fission have been continued during this quarter.

An improvement in experimental technique over that used previously has been accomplished by using Li metal targets to increase the neutron yield and by using a double fission counter assembly to allow absolute yield measurements to be calculated more accurately.

The excellent performance of the Van de Graaff accelerator, even at accelerating voltages as high as 3.5 MeV, has been exploited to extend delayed neutron measurements up to 1.8 MeV neutron energy.

Measurements have been made on ^{239}Pu (Fig. 8), ^{233}U (Fig. 9), and ^{235}U (Fig. 10) from 0.1 to 1.8 MeV. The short-term availability of a

highly enriched (99.9%), 25 gm sample of ^{242}Pu enabled the first delayed neutron yield measurements to be made on this isotope. Unfortunately, the large mass, unsuitable geometry, and high neutron background of the sample were not compatible with a precise yield measurement. In addition, the lack of ^{242}Pu fission counter foils necessitated the use of available foils (^{235}U) and

Fig. 8. Absolute total delayed neutron yields from ^{239}Pu .

Fig. 9. Absolute total delayed neutron yields from ^{233}U .

Fig. 10. Absolute total delayed neutron yields from ^{235}U .

appropriate normalization from cross section data. The results are shown in Fig. 11. Relative yield errors are $\pm 10\%$ and the absolute yield error is $\sim 30\%$. The average value (0.016 ± 0.005) obtained for the absolute total delayed neutron yield from ^{242}Pu is consistent with the value (0.021) expected from delayed neutron yield systematics. ⁽⁹⁾

The average of the so-called "front-to-back ratios" in the neutron slab detector were computed to be 2.17, 2.21, 2.16, and 2.18 for ^{233}U , ^{235}U , ^{239}Pu , and ^{242}Pu , respectively, indicating that the average energy of the delayed neutrons from these species is the same within a few tens of keV.

⁹ G. R. Keepin, Physics of Nuclear Kinetics, p. 101, Addison-Wesley, Reading, Mass. (1965). See also Nuc. Sci. Eng. 36, 202 (1969).

Fig. 11. Absolute total delayed neutron yields from ^{242}Pu .

ABSOLUTE DELAYED NEUTRON YIELDS				
ELEMENT	NEUTRON ENERGY			
	THERMAL ^(a)	AVERAGED 0.1-1.8 MeV	FISSION SPECTRUM ^(a)	3.1 MeV ^(b)
^{233}U	0.066 ± 0.003	0.078 ± 0.008	0.070 ± 0.004	0.077 ± 0.008
^{235}U	0.158 ± 0.005	0.171 ± 0.017	0.165 ± 0.005	0.180 ± 0.018
^{239}Pu	0.061 ± 0.003	0.065 ± 0.006	0.063 ± 0.003	0.069 ± 0.007
^{242}Pu	—	0.016 ± 0.005 ^(b)	—	—

(a) See Ref. 9.

(b) 0.7-1.3 MeV

Fig. 12. Summary of total absolute delayed neutron yields from thermal to 3.1 MeV neutron energy for ^{233}U , ^{235}U , ^{239}Pu , and ^{242}Pu .

A summary of measurements of total absolute delayed neutron yields from thermal to 3.1 MeV neutron energies for the elements ^{233}U , ^{235}U , ^{239}Pu , and ^{242}Pu is presented in Fig. 12.

Delayed Neutron Abundances and Half-Lives for 14.7-MeV Fission

R. H. Augustson, L. V. East, and H. O. Menlove

The program to measure delayed-neutron group abundances and half-lives from 14.7-MeV neutron induced fission continues to progress. To date, data have been obtained for ^{232}Th , ^{233}U , ^{235}U , ^{238}U , ^{239}Pu , and ^{242}Pu using Accelerator I as the neutron source. These data are being analyzed using the exponential fitting code de-

scribed previously. ⁽¹⁰⁾ This code is presently being modified to include the necessary correction for neutron pulse non-uniformities that were encountered for some irradiations. Measurements using neutron sources and a random pulse generator have been performed to determine the neutron counter dead time so that counting-loss corrections may be applied to the decay data. Such corrections were not included in the 14.7 MeV fission results for ²³⁵U and ²³⁸U previously given. ⁽¹⁰⁾

Preliminary results for ²⁴²Pu are presented in Table IV. These results were obtained by combining the data from nine independent accelerator runs using irradiation times ranging from 50 ms to 60 sec. The ²⁴²Pu data have been corrected for spontaneous fission neutron background and counting losses, but not for neutron pulse non-uniformities and self-multiplication within the sample.

TABLE IV
PRELIMINARY DELAYED NEUTRON ABUNDANCES AND HALF-LIVES
FOR 14.7-MeV FISSION OF ²⁴²Pu *

Group Index, i	Half-Life, T _i (sec)	Relative Abundance a _i /a
1	53.7 ± 4.3	0.023 ± 0.005
2	21.6 ± 0.5	0.170 ± 0.003
3	5.3 ± 0.3	0.162 ± 0.020
4	2.1 ± 0.1	0.362 ± 0.010
5	0.71 ± 0.07	0.156 ± 0.010
6	0.24 ± 0.01	0.128 ± 0.010

* 99.91% ²⁴²Pu.

NUCLEAR INSTRUMENTATION AND APPARATUS DEVELOPMENT

Delayed-Neutron Detector Optimization

H. O. Menlove and R. H. Augustson

For many assay applications it is desirable to use a delayed-neutron detector whose efficiency is peaked in the region of delayed-neutron energies (~ 0.5 MeV). To this end, a "thin slab" detector making use of previously reported effi-

ciency optimization measurements ⁽¹¹⁾ has been constructed and installed in the Mobile Nondestructive Assay Laboratory. Figure 13 shows the thin slab detector containing 19 ³He counters in a 3 x 20 x 24 in. slab of polyethylene.

¹⁰ LA-4315-MS, pp. 18-19 (1969).

¹¹ LA-4315-MS, p. 20 (1969).

Fig. 13. "Thin slab" neutron detector containing 19^3He Counters in a 3 x 20 x 24 in. slab of polyethylene, and covered with Cd.

Fig. 14. Measured absolute detection efficiencies for "thin slab" and "flat-response" slab-type neutron detectors.

The resulting efficiency was measured using calibrated neutron sources located 1-m from the front face of the detector. D,T neutrons from Accelerator I were used to obtain the calibration point at 14 MeV. Figure 14 shows measured absolute neutron detection efficiencies as a function of energy for both the thin slab detector and the flat-response slab detector described previously. ⁽¹²⁾ It can be seen that the efficiency of the thin slab detector is ~5.2 times greater for delayed neutrons than for the 14 MeV background neutrons from the neutron generator. This decrease in detection efficiency with increasing neutron energies reduces (by a factor of 5) the "on-to-off" ratio required for pulsed neutron interrogation and precise delayed neutron assay.

It should be noted that because of saturation effects in neutron capture by ^3He , a reduction by a factor of two in the number of counter tubes reduces the total counting efficiency by a factor of only ~1.3 for the thin slab detector.

Add-a-gram Apparatus for Scrap Barrel Assay

M. M. Thorpe, T. D. Reilly, and D. B. Smith

A complete description and discussion of the "add-a-gram" apparatus for assay of fissionable scrap in 55-gallon barrels is contained in the previous progress report. ⁽¹³⁾ Several improvements have been made with respect to the mechanical operation of the add-a-gram system. The original magnetic sensors have been replaced with photosensors. This permits the spherical steel casing for the add-a-gram ^{235}U sample to be replaced by a much lighter ball of Kel-F (a hydrogen free plastic similar to teflon but with better machining properties). A change in the orifice system for pneumatic propulsion of the ball has re-

¹² L. V. East and R. B. Walton, "Polyethylene Moderated ^3He Neutron Detectors," Nucl. Instr. and Meth. **72** 161 (1969).

¹³ LA-4315-MS, pp. 8-10 (1969).

sulted in reduced air consumption and smoother operation. A test barrel has been equipped with a set of tubes to facilitate the placement of ^{235}U samples throughout the interior matrix.

Initial measurements have been carried out using an empty barrel and a barrel filled with rags, rubber gloves, etc. Other representative matrix materials are being prepared. Measurements on the empty barrel have dictated a change in the detector arrangement. Originally the apparatus was constructed using 24 ^3He counters imbedded in eight slabs of polyethylene ⁽¹⁴⁾ which completely surrounded the barrel. In this configuration the detector array was also to serve as a neutron reflector, and in fact it provided too great an enhancement of peripheral response. An arrangement which gives a flatter spatial response throughout the barrel is an array of 19 ^3He counters imbedded in a single polyethylene slab (20 x 24 x 3 in.) placed diametrically opposite the 14-MeV neutron generator target.

Another particularly noticeable effect in the barrel filled with rags and rubber gloves was the contribution of delayed neutrons from the $^{17}\text{O}(n, p)^{17}\text{N}$ reaction. Approximately 200 g of natural oxygen produce the same number of delayed neutrons as 1 g of ^{235}U for 14-MeV neutron irradiation energy. Since the threshold for the $^{17}\text{O}(n, p)^{17}\text{N}$ reaction is ~ 8 MeV, a nominal reduction in energy of the 14-MeV source neutrons should greatly reduce the ^{17}O -delayed neutron background without appreciably affecting the penetrability of the interrogating neutrons. A series of DTF-IV neutron transport calculations was undertaken to investigate the magnitude of the background reduction that might be expected from the use of a lead moderator around the 14-MeV neutron source. The ratio of delayed neutrons produced per atom of ^{235}U to delayed neutrons produced in ^{17}O per atom of normal oxygen was computed at a position 12 in.

from a spherical lead moderator surrounding a 14-MeV neutron source. The radius of the moderator was varied from 4 in. to 12 in. ^{17}O was assumed to be 0.037% abundant in normal oxygen, and a value of 21.5 mb ⁽¹⁵⁾ was used for the $^{17}\text{O}(n, p)$ cross section.

The results of these calculations are shown in Fig. 15 as a function of the radius of the moderator. It appears that 8 in. of lead should reduce the ^{17}O delayed-neutron background by more than an order of magnitude. The ratios plotted in Fig. 15 are conservatively low since the cross section for the $^{17}\text{O}(n, p)$ reaction very probably does not retain its 14-MeV value down to threshold (as was assumed in the computations).

Fig. 15. Relative number of delayed neutrons produced in ^{235}U and in normal oxygen (via $^{17}\text{O}(n, p)^{17}\text{N}$) by a 14-MeV neutron source surrounded by various thicknesses of lead.

¹⁴ LA-4315-MS, Figure 9 (1969).

¹⁵ LA-4227-MS, p. 16 (1969); see also H. O. Menlove, R. H. Augustson, and C. N. Henry, Trans. Am. Nucl. Soc. 12 742 (1969).

L. V. East, M. M. Stephens, and J. E. Swansen

The design of several special purpose electronic instruments was completed this quarter. The instruments are NIM-compatible⁽¹⁶⁾ and are individually described below.

Clock Generator

This is a dual width module that provides accurate timing pulses derived from a crystal oscillator. Eight outputs are provided, giving pulse periods from 1 μ sec to 10 sec in decade steps. The oscillator may be externally gated on and off, if desired. This module is intended for use as an accurate time base for the multiscaler channel advance units previously described⁽¹⁷⁾ and other timing applications. In addition, an external input is provided that may be used with pulse sources having an output frequency of 1-MHz or less to give an accurate frequency division of up to 10^7 . The unit contains its own internal power supply (+5V), and requires only 110 V AC from the instrument bin. Integrated circuits are used throughout the module.

Four of these units have been constructed by the electronics fabrication group (ENG-5).

Neutron Coincidence Unit

This instrument incorporates the necessary logic in a single triple-width module to coincidence count spontaneous fission neutrons using the 4 π neutron detectors previously described.⁽¹⁸⁾ A logic diagram of the unit is shown in Fig. 16. In order to reduce pulse pile-up problems, four independent input discriminators are provided. The 0.3 μ sec output pulses of these discriminators are fed to an OR gate, the output of which triggers the coincidence logic. The coincidence logic is

Fig. 16. Logic diagram for neutron coincidence counting circuit.

similar to that used in a neutron coincidence circuit developed at Brookhaven National Laboratory.⁽¹⁹⁾

Output pulses from the discriminators are delayed by the "Pre-Gate Delay" univibrator, which in turn triggers the "Coincidence Gate" univibrator. The ON time of the coincidence gate can be selected by a front panel switch to be 15 μ sec, 25 μ sec, 50 μ sec, or 100 μ sec. Subsequent pulses from the input discriminators arriving during the coincidence gate ON time are passed to the "Real + Accidental" output. The trailing edge of the coincidence gate pulse triggers a chain of nine delay univibrators, and the trailing edge of the last delay triggers an "Accidental Coincidence" univibrator. These univibrators all have the same ON times as the coincidence gate, and recovery times of less than 5% of the gate width. Therefore each input signal that triggers the coincidence gate will also produce an "accidental" gate at a time $10 \times$ (coincidence gate ON time) later. Signals from the input discriminators arriving during the accidental gate ON time are passed to the accidental coincidence output.

The coincidence gate times are chosen to be compatible with the die-away time of the particular neutron detector being used. The length of the "Pre-Gate Delay" is chosen to be roughly equal to the length of the amplifier pulses feeding the input

¹⁶ AEC Report TID-20893 (Revised), (1968).

¹⁷ LA-3859-MS, p. 14 (1968).

¹⁸ LA-4315-MS, pp. 20-22 (1969).

¹⁹ R. L. Chase, private communication.

discriminators. This delay is required in order to compensate for a pile-up effect at high count rates that would otherwise produce an error in the number of "Real + Accidental" events compared to the number of "Accidental" events.

A "Stop Gate" input is provided that can be fed by a timer or other device to control the counting time of the unit. Outputs from each discriminator and the coincidence gates are made available for external use.

The module requires power supply voltages of + 12 V DC, - 12 V DC, and 110 V AC (for an internal + 5 V DC supply). Two prototype units have been built, and four more are being constructed by the LASL electronics group (P-1).

Variable Threshold Logical OR

Several applications exist for a logical OR circuit having a high degree of noise immunity. A single width NIM module was therefore designed with four OR inputs, with a level discriminator on each input. The input thresholds may be adjusted by a single front panel control over the range of 0.5 to 5.0 V. Five of these units have been built to date.

50% Gate Unit

A special single width module has been designed to generate the neutron generator, pre-amplifier, and scaler gating signals required for the delayed neutron yield assay methods utilizing repetitive neutron pulses. (20) The functions performed by this single module previously required two or more modules.

An output pulse timing diagram is shown in Fig. 17. The pulsing sequence may be started

Fig. 17. 50% gate unit timing diagram.

and stopped by a front panel switch, or by an external gate. Timing is such that a complete irradiation-count cycle will be completed when the RUN gate goes to the stop mode. Control pulses are available for the three types of neutron generators presently in use by N-6 (exclusive of the Van de Graaff), as well as gated preamplifiers (21) and scalars.

The irradiation time is continuously adjustable from 10 msec to 11 sec. The delay time between the end of the irradiation and the start of the scaler count is continuously adjustable from 1 msec to 110 msec. The scaler counting time is equal to the sum of the irradiation and delay times (hence the name 50% Gate Unit). If the delay time is set longer than about 80% of the count time, the neutron generator and scaler gates will be held in the OFF mode.

Two prototype units have been constructed, and two more units will be assembled by ENG-5.

²⁰ LA-4070-MS, pp. 4-5 (1969).

²¹ L. V. East, Nucl. Inst. and Meth. **71** 328 (1969).

MOBILE NONDESTRUCTIVE ASSAY LABORATORY

Construction and Instrumentation Procurement

J. H. Menzel, B. R. Dennis, and G. M. Worth

A detailed description of the planned MONAL construction, components, and physical layout has been reported previously. ⁽²²⁾ The successful completion of the major construction phase (in October) made it possible to exhibit the MONAL, (cf. Fig. 18) at the AEC Symposium on Safeguards Research and Development held at Los Alamos October 27-29, 1969, and to perform a complete demonstration of neutron interrogation and assay methods on ²³⁵U graphite systems.

The large-sample ("barrel") handling system (cf. Fig. 19) was demonstrated with eight of its 16 cantilevered sample carriers installed and functioning. The lift and vertical/rotational scan functions of the large-sample (55-gallon barrel)

Fig. 19. Close-up view of the MONAL large-sample (barrel) handling system. The magazine-type conveyor positions the large sample or barrel over the elevator platform (left) which then lifts it into the shielded assay room, located directly above, by passing through the cantilevered sample carrier ("bucket"). The sample carriers and elevator are carried inside the cargo container during transit of the MONAL.

Fig. 18. The MONAL as exhibited during the AEC Symposium on Safeguards Research and Development at LASL on October 27-29, 1969. The conveyor portion of the large-sample (55-gallon barrel) handling system can be seen beneath the center section of the 10 ft wide by 50 ft long cargo container.

²² LA-4162-MS, p. 19 (1969).

Fig. 20. Shielded interrogation chamber (4 ft-wide x 6 ft-long x 7 ft-high) of the MONAL. The elevator (left and below) is used for rotational and vertical scanning of samples for both active and passive assay. Shield penetrations (to right of center) for delayed gamma measurements and a small sample pneumatic transfer tube (upper right) make this interrogation chamber an extremely versatile assay facility. The neutron interrogation source (center) is located at the end of the neutron generator drift tube reentrant into the shield from the rear equipment room (cf. Fig. 21).

elevator (cf. Figs. 19, 20) were also demonstrated during the Symposium; certain improvements such as fail-safe brakes, additional control switches and safety interlocks are now being added.

The neutron shield was leak-tested with pressurized air and then filled with water. A complete neutron and gamma dose mapping of the MONAL and surroundings for the designed neutron source level inside the shielded interrogation chamber (cf. Fig. 20) showed that the main shield gave the expected (calculated) dose levels. However, shielding against ground-scattered neutrons was not sufficient for the designed neutron source level. Consequently, the neutron generator was operated at reduced beam current during the active interrogation demonstration. A thorough calculational and experimental shield study (see following section) resulted in the required design parameters

of the portable ground-shield tanks which are scheduled for completion by the end of February.

Performance tests of the MONAL neutron generator (shown in the rear equipment room of the MONAL in Fig. 21) indicated that the ratings of some of the pulser components were very marginal. Consequently, LASL and Kaman Nuclear have agreed on provisional acceptance of the neutron generator predicated upon delivery of a new pulser. The construction of the new pulser has been completed and new performance tests are scheduled for mid-January; final acceptance of the neutron generator is expected by mid-February. Also during February, the design and construction contract for a post-deflection system to increase

Fig. 21. Rear equipment room of the MONAL. The rear doors of the cargo container have been opened to show the forepump, high voltage supply tank, and target cooling unit (left), the rear of the 14-MeV neutron generator on a movable support (center), and the air conditioning/heating unit (right).

Fig. 22. Control console and data acquisition area, located forward in the MONAL. The four electronic racks contain (from left to right): the multichannel analyzer, data acquisition NIM electronics, neutron generator controls, and radiation interlocks and operational support equipment. The Ge(Li) system (lower right) provides complete on-board capability for high resolution passive gamma nondestructive assay.

the neutron "on/off" ratio to $\sim 10^8$ is expected to be completed by Kaman Nuclear.

Figure 22 shows the control console and data acquisition area of MONAL, as well as the Ge(Li) passive gamma detector mounted on an optical bench (at right).

In addition to the modifications and improvements just described, orders for various electronic equipment and assay apparatus have been placed, including a portable gamma scintillation system, a personnel radiation monitoring system and a complete fire protection system for the MONAL.

Calculational and Experimental Shield Studies

D. B. Smith, J. H. Menzel, and B. R. Dennis

The shielding calculations for the MONAL reported last quarter have been extended to two additional shield-tank configurations, one of which has been constructed and for which experimental

measurements have been made. As previously described⁽²³⁾ the full geometry of the primary neutron shield and ground-scatter shields were modeled in the three-dimensional Monte Carlo code TDMC-1G. Dose rates were calculated at four positions: interlock bar six inches above the floor, operator's position six inches above the floor, center of side door six inches above the floor, and ten feet from the side of the MONAL opposite the door, six inches above the ground. A neutron source strength of 5×10^{10} neutrons/sec was assumed. The calculated dose rates are given in Table V. Dose rate measurements (also in Table V) were made at the interlock bar and at the operator's position for the smaller shield tank. The neutron generator was operated at an average source strength of 1.43×10^{10} neutrons/sec, and the measured dose rates were scaled to correspond to a source strength of 5×10^{10} neutrons/sec. A Mod-6 Neutron Detector with 10 inch ball ("neutron cow") was used for the measurements.

The good agreement between measured and calculated dose rates lends confidence to the calculations for which there are no comparable measurements. It appears that the larger ground-shield tank (3 ft thick) will adequately shield the operator. It is anticipated that additional shield tanks will be used at the side of MONAL to reduce the dose in the vicinity of the door.

Visit to the Dow Chemical Company - Rocky Flats Division

M. M. Thorpe, J. H. Menzel, and R. B. Walton

In preparation for the first field assignment of the MONAL, three members of the LASL Safeguards staff (Group N-6) visited the Rocky Flats Division of the Dow Chemical Co. on December 2, 1969. Extensive discussions were held with representatives of the plutonium process facilities,

²³ LA-4315-MS, p. 24 (1969).

TABLE V
MONAL GROUND SHIELD STUDY

Shield Configuration	Neutron Dose (mr/hr)			
	Bar	Operator	Door	10 ft
Ground Shield 2 x 4 x 9 (ft) - calc.	21 ± 9	3.5 ± 0.6	18 ± 4	28 ± 4
Ground Shield 2 x 4 x 9 (ft) - meas.	21 ± 4	4.2 ± 0.8	--	32 ± 8 ^a
Ground Shield 3 x 4 x 9 (ft) - calc.	5.7 ± 1.4	1.7 ± 0.3	11 ± 2	23 ± 4

^a Interpolated from measurements at adjacent positions.

and with members of the Chemistry Research and Development staff.

Rocky Flats has used passive nondestructive assay systems for 55-gallon barrels and 1-gallon cans for many years and is presently engaged in expanding the capability of these techniques with advanced Ge(Li) detectors and an on-line computer system. Present passive assay techniques are not entirely satisfactory for medium to high Z materials and for inadequately segregated scrap and/or conglomerate materials (e.g., fire residues). Considerable interest was expressed in the assay

potential of the active and passive techniques employed in the MONAL. Members of the Rocky Flats staff provided an excellent summary of all scrap and waste generated in their purification and recovery operations, including material description, average sample weight, plutonium content, etc. This information has already proved very helpful in guiding LASL research on the development of assay techniques for several categories of plutonium scrap, metal-containing fire residues and routine process waste.

AEC SYMPOSIUM ON SAFEGUARDS RESEARCH AND DEVELOPMENT

The AEC Symposium on Safeguards Research and Development, held at the Los Alamos Scientific Laboratory on October 27-29, and at Gulf General Atomic on October 30, attracted an international gathering of over 350 participants from industry, government, and R&D laboratories. Some 29 industrial exhibits presented state-of-the-art equipment related to safeguards and nuclear materials management. A very full two-day schedule

(October 27, 28) of formal presentations and panel discussions provided a forum for fruitful exchange of ideas and information among those groups most directly concerned with nuclear safeguards R&D and practical applications. Complementing the formal presentations, technical briefing tours (on October 29, AM) of the LASL Safeguards Laboratories at Ten Site and at Pajarito Site as well as the Mobile Nondestructive Assay Laboratory (cf.

Fig. 23) gave Symposium attendees a unique opportunity to see demonstrations of the advanced safeguards techniques and methods for nondestructive assay being developed at LASL. A similar tour (October 30) of the Gulf General Atomic facility at San Diego, California, was attended by over

one-third of the Symposium registrants.

Editing of the tapes of the oral presentations and panel discussions, a major step in the preparation of written proceedings for the Safeguards Symposium, has been completed by the LASL Safeguards staff.

Fig. 23. The control room of the MONAL provides a focal point for technical briefings and a tour of LASL safeguards facilities by AEC Commissioner Clarence Larsen (right) and his aide, John Griffin (Center) during the AEC Symposium on Safeguards Research and Development, October 27-30, 1969.

ACCELERATORS, NEUTRON SOURCES, AND FACILITY DEVELOPMENT

Accelerator I Neutron Generator

R. H. Augustson

Two projects using a wide range of pulsing cycles dominated Accelerator I usage this quarter. The α, λ_1 measurements utilized low duty cycles (2% - 20%) while the Pb-C moderator studies were performed using a 50% duty cycle. Throughout the extended running time of both of these projects, Accelerator I performed extremely well, with no machine down time.

Van de Graaff Accelerator Facility

A. E. Evans, M. S. Krick, and J. J. Halvey

During the final quarter of 1969, the N-6 Van de Graaff Accelerator was operated for a total of 98.6 hours. Of this time, 53.5 hours were used for the measurement of delayed-neutron yield of fissionable materials as a function of the energy of the neutron inducing fission. Forty-five running hours were used in developing monoenergetic neutron interrogation methods for assay of the Pu

verification samples from the New Brunswick Laboratory.

The Phillips-ion-gauge circuitry supplied with the accelerator has failed, leaving the accelerator without a vacuum-protective interlock. A new transformer and rectifier are being supplied by the manufacturer, and will be installed immediately upon receipt.

The vertical exit (energy-control) slits in the beam-handling system have been electrostatically shielded. As a result, some improvement in accelerator stability has been observed.

To obtain higher neutron yields and extend target life, an electromagnetic target-scanning system is being built to distribute more evenly the ion beam over the surface of targets. At present, the ion beam is spread by defocusing, with a resultant nonuniform distribution over the target face. Also in the planning stage is a feedback-

controlled beam-centering system which should improve the reproducibility of small sample assays in which the sample must be placed so near the accelerator target that small shifts in beam position can result in significant experimental errors.

Cockcroft-Walton Accelerator Facility

C. N. Henry and T. H. Whittlesey

During the fourth quarter of 1969, the N-6 Cockcroft-Walton was used about 60% of the time for nuclear safeguards research and about 40% for weapons related research. A failure in the light link transmitter and receiver was traced to a defective transmitting diode and phototube receiver, as well as insufficient cooling for the phototube. The diode and phototube have been replaced and increased cooling capability for the phototube is being designed.

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