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HAZARDS CONTROL PROGRESS REPORT NO. 51
July through December 1975

February 16, 1976

MASTER

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Progress Reports

RADIATION PROTECTION

Electrochemical Etching of Recoil Tracks in Polycarbonate for Fast Neutron Personnel Dosimetry

About a year ago, Sohrabi¹ reported on the use of electrochemical etching of polycarbonate for detection of recoil particles following neutron irradiation. His results held a great deal of promise that the technique could be used as a fast neutron personnel monitor. We have done preliminary work using electrochemical etching, and are also encouraged by its potential.

Enhancement of nuclear particle tracks by electrochemical etching² is a simple technique. After exposure to a neutron source, a piece of plastic is placed between two volumes of caustic solution. Electrodes are placed in the solution, and a pulsed field is established across the sample. We use an alternating current, but pulsed direct current fields could also be used. The apparatus is shown schematically in Fig. 1. After a few hours, particle tracks such as recoiling nuclei from neutron interactions are enhanced considerably and become visible with the unaided eye (some magnification is required for counting).

We have built an etch chamber with positions for four samples. The chamber is shown in Fig. 2, together with a new chamber which requires much less caustic electrolyte and can accommodate up to nine 1-in. discs. The first chamber held 1.8 litres of electrolyte on each side of the sample. The new chamber contains 0.5 litre on each side, and is much easier to handle. The data presented here, however, were taken from samples etched in the original chamber. Dimensions of both are given in Table 1.

Fig. 1. Schematic representation of electrochemical etch system.

Fig. 2. Electrochemical etch chambers.

Table 1. Etch chamber parameters

	Original chamber	New chamber
Inside diameter (mm)	114	114
Inside length (one side) (mm)	170	53
Overall length (excluding valves) (mm)	396	151
Electrode composition (mm)	Nickel	Nickel
Electrode diameter (mm)	73	102
Electrode separation (mm)	250	88
Exposed sample area (mm ²)	924	924 ^a
		285 ^a

^aNew chamber uses interchangeable sample holders.

Fig. 3. (a) Recoil tracks in polycarbonate produced by fast neutrons (122X), (b) Artifact tracks along a scratch (160X), (c) "Rosette" artifact, thought to outline the perimeter of a polycarbonate particle pressed into the surface (122X), (d) Damage region caused by high electric field around the perimeter of the air bubble during etching (19X).

Table 2. Sensitivity of electrochemically etched foils to ^{252}Cf neutrons

Voltage	Background	Tracks $\text{cm}^{-2} \cdot \text{rem}^{-1}$		
		None	Moderator	
			10-cm CH_2	20-cm Al
800 V	1.4 cm^{-2}	19	18	8
1000	5	23	23	10
1200	23	34	30	10
1500	210	36	30	10

The electrochemical etching technique involves a number of variables including chamber frequency, voltage, electrode diameter, electrode spacing, electrolyte concentration, etching time, sample thickness and exposed sample area. A significant part of future work will be the study of the effect of changes in these parameters on the etching sensitivity. The data presented here are for samples etched at a frequency of 2 kHz at voltages from 800 V to 1500 V for 5 hours. The polycarbonate samples are 0.25 mm thick. Because the etch time increases significantly with sample thickness,¹ material thicker than 0.4 mm is not useful for routine work.

One variable we had not anticipated was the quality of the plastic surface. The process is very effective for enhancing imperfections in the plastic. These imperfections seem to be the major contribution to the sample background. The etching process amplifies any surface penetration, and produces either a single track or an artifact which is easily distinguished from a true track. Etched tracks in a sample of polycarbonate exposed to ^{252}Cf neutrons are shown in Fig. 3(a). For comparison, artifacts from three surfaces are shown: 3(b) a surface scratch; 3(c) a fine particle of polycarbonate pressed into the surface when the sheet was rolled; and 3(d) the etch pattern around a bubble on the film.

The material we have found to have the lowest background is 0.25-mm polycarbonate film from Roland Plastics in Kensington, Connecticut.* It was shipped to us with a thin protective layer of polyethylene film which we leave on to protect the surface until the sample is etched. We should note that the quality of material can be expected to vary from batch to batch and on how it has been handled, so it is worthwhile to obtain samples from a number of suppliers before making a selection.

One set of samples was exposed to each of two neutron spectra from a moderated ^{252}Cf source. The moderation was provided in one case by a sphere of polyethylene with a thickness of 10 cm, and in the other case by a 20-cm-thick sphere of aluminum. A third set of discs was exposed to the source with no moderation. After etching, the samples were counted with an optical microscope at 35X. The results are presented in Table 2. It should be noted that the magnification was low enough that by focusing between the surfaces of the sample, both surfaces could be counted simultaneously. The results presented are the total for both surfaces of a 1-cm² area of the sample.

*Reference to a company or product name does not imply approval or recommendation of the product by the University of California or the U.S. Energy Research & Development Administrator to the exclusion of others that may be suitable.

The results show that in the range 800 V to 1500 V, the background seems to double every 100 V, but the response increases only by a factor of two. This suggests the need to use low voltages with longer etching times to achieve equivalent track size and lower limit of detection. It will be necessary to see how variation of some of the other parameters affects this situation.

Preliminary results from our use of the electrochemical enhancement of neutron-induced recoil tracks in polycarbonate support the optimism for its value as a fast neutron personnel dosimeter. It has the advantages that the material is cheap, is insensitive to gamma rays, and has good latent track stability. Although it does not appear to be as sensitive as TLD albedo techniques,³ it would be very useful in conjunction with an albedo system to provide a combination of high sensitivity and improved accuracy at dose levels in excess of a few hundred mrem.

The technique has major drawbacks: (1) long etch times are required; and (2) it is a process that does not lend itself in an obvious way to automation. However, if it is used with an albedo dosimeter, only a small fraction of the dosimeters used, i.e., those having significant exposure indicated by the albedo component would require processing. Work must now be done to (1) characterize the effects of the various parameters in the etching process, (2) select a set of parameters that yield the best sensitivity, and (3) make irradiations with monoenergetic neutrons to define the energy response (particularly from 0.5 to 5 MeV).

Realistic Torso Phantom Development

Introduction. In the previous Hazards Control Progress Report,⁴ we reported on the initial stages of construction of a realistic phantom for internal measurement of ^{239}Pu and other heavy elements *in vivo*. In this report we review the current status of the construction, and discuss the work to be done.

Initial work on the phantom⁴ included: (1) determination of the external dimensions that most closely represent the stature of a typical radiation worker; (2) construction of a clay model to be used as a guide for contouring the torso; (3) acquisition of the rib cage and major organs from a male cadaver of the same approximate size; (4) molding the torso and interior void of the cadaver with organs removed; and (5) preliminary selection of tissue equivalent (T.E.) materials.

The work to be discussed here includes: (1) casting the organs in plaster; (2) casting the torso void to provide a cavity for organ arrangement; (3) more detailed selection of tissue-equivalent materials; and (4) development of a method for filling the rib cage bones with T.E. material to simulate lost bone marrow.

Organ Casting. With the exception of the lungs, it was a simple task to make plaster casts of the major organs. Individual organs were suspended in a box with hypodermic needles (Fig. 4). The box was then filled with polyurethane to make the mold. Non-shrinking plaster was used for the model organs. A sample mold and plaster cast is shown in Fig. 5.

The heart, liver, spleen, and kidneys were left

sufficiently rigid by the embalming technique to be cast in this way. The lungs, however, began deflating after removal from the cadaver, and could not have been cast without serious distortion. As a result, we used a simple specimen preparation technique⁵ to make the lungs rigid for casting. The lungs were reinflated and dried by passing a tube into the trachea which was still attached to the lungs, and then flowing compressed air through them for a period of about two weeks (Fig. 6). Normally, animal lungs prepared in this way are hard and feel as if they were made of styrofoam. In this case, the glycerin in the embalming fluid left most of the lungs somewhat spongy, but rigid enough to be cast.

The volumes of the organ casts are presented in Table 3, together with average values for an adult male.⁶ At the beginning of the project, we were very concerned with trying to produce "Standard Man" organs. However, a tour of the dissection laboratory at the University of California Medical Center Anatomy Department graphically demonstrated to us individual variability in organ shape and size. This variability is reflected in the ranges shown in Table 3.

Fig. 4. Spleen in position for molding.

Fig. 5. Sample mold and plaster replica of the heart.

Fig. 6. Air reflation and drying of lungs.

Table 3. Organ volumes for plaster casts and reference man

	Volume (ml)	
	Plaster casts	Reference man
Heart	865	750(630-870) ^a ml
Liver	2050	1700(1320-2200) ^b
Spleen	155	181(109-300) ^b
Kidneys	150 (right), 170 (left)	149(112-182) ^b
Lungs	1410 (right) 740 (left)	5000 (total volume)

^aValues in parentheses represent standard deviation for samples measured.

^bValues in parentheses represent 80% range for samples measured.

As a result, the main concern now is that the existing organs fit together properly. We do feel, however, that the lungs must be modified because they appear to have shrunk when compared with the cast of the torso cavity.

Torso Cavity Cast. The original torso cavity cast was made by filling the cadaver void with expanding silicon foam. That model was reproduced in plaster in a manner similar to that used for the organs. Because there was some shrinkage in the original foam cast, it was necessary to modify the plaster cast until the rib cage fit properly (Fig. 7).

Fig. 7. Plaster mold of torso cavity with ribs in place. Connection between fourth rib and sternum is a calcified bridge V on the cartilage that develops.

Selection of Tissue-Equivalent Material. Our first preference for a muscle-equivalent material to be used for the heart, liver, spleen, kidneys and torso was based on the relative transmission factors of available T.E. materials.⁴ The depolymerized rubber, TEMEX, seemed to have the best simulation of the x-ray transmission properties of lean tissue. However, as we studied samples of depolymerized rubber available to us, there was concern that thin layers such as that containing the ribs and sternum would crack or split easily. We also learned that the attenuation coefficient at 10 keV differed from that for muscle by more than 25%.⁷ People in our plastics shop proposed use of polyurethane as a base for our own T.E. composition. Polyurethane is nominally 65% carbon, 9% hydrogen, 5% nitrogen and 21% oxygen, and is generally free of materials having higher atomic numbers. Specific composition depends on the supplier. In addition to the strength of the material, the availability of foamed polyurethane as a possible lung material may be an advantage.

The attenuation coefficient for polyurethane not having significant levels of high Z materials is less than that for lean tissue. The transmission of 17-keV x rays through 4 cm of Scotchcast 221 is four times that of an equivalent thickness of fresh beefsteak. In fact, polyurethane without filler may be a good simulant for adipose tissue. To simulate lean tissue with

Fig. 8. Transmission of ²³⁸Pu L x rays through muscle-equivalent materials. 1, MIX-D (LLL); 2, Rando T.E. material; 3, Temex; 4, LLL T.E. material; 5, Fresh beefsteak.

polyurethane, it is necessary to add a higher Z filler. In selection of a proper filler, it is best to select materials with atomic numbers lower than about 20. If high Z fillers are used, it is more difficult to match the attenuation coefficients of tissue over the range of energies from 10 to 100 keV. However, if very low Z materials are used, the quantities required to achieve the proper attenuation may exceed the capacity of the polyurethane.

A number of commercial polyurethane formulations are available in a variety of compressibilities, so there is a large selection of potentially useful combinations of plastics and fillers for T.E. applications. We selected Scotchcast 221 and added 3.5% sulfur. Calcium carbonate is also under consideration as a filler. It does not have the odor associated with sulfur, but more material is required to achieve a given degree of attenuation. The narrow beam attenuation curve for our T.E. plastic material using ^{238}Pu x rays is shown in Fig. 8, together with curves for TEMEX, MIX-D, RANDO muscle-equivalent material and beefsteak. Attenuation curves for other x-ray energies have not been made, but the use of polyurethane appears to meet our needs as a base to simulate soft body tissues.

Bone Marrow Replacement. When the rib cage was cleaned and degreased, all of the bone marrow was removed. A technique to replace the lost marrow was suggested by Dr. Donald Newton.⁸ Because of the ease of handling and its low viscosity after melting, MIX-D⁹ was selected as the marrow simulant. The process involves suspending the bone over a pot of molten MIX-D in a vacuum chamber. The bone is lowered into the fluid MIX-D and the chamber is brought to atmospheric pressure, forcing the T.E. material into the trabeculae of the bone. The mixture is cooled, and the bone is removed just before the MIX-D solidifies. After the bone is removed from the chamber, excess MIX-D is scraped off. The whole process takes about a day to assure proper evacuation and penetration of the MIX-D; however, only about an hour of the operator's time is required to make a run.

We have built and tested such a chamber. Figure 9 shows a neutron radiograph of two ribs acquired from an educational supply firm. The top bone is unfilled, and the bottom bone has been filled as described. There are small voids in the filling, particularly one path near the tip. That particular void connects with a hole in the bone surface. We suspect that the MIX-D drained from that hole after being removed from the molten solution. Redipping the same bone did not improve the situation. A recent run with a new bone has been made, allowing the mixture to cool more

Fig. 9. Neutron radiograph of ribs. Top: Without marrow simulant replacement; bottom: marrow replaced by vacuum filling with MIX-D.

before removing the bone. That bone has been photographed, and the filling is much better. We now consider the technique successful even with vertebrae, and feel small voids are tolerable.

Summary. Recent work on the realistic phantom includes preliminary casts of major organs, plaster casts of the torso and organ cavity, formulation of muscle-equivalent material, and development of a successful technique for replacing lost bone marrow in the rib cage. Considerable effort lies ahead. We must modify the organs to establish a closely packed array in the torso cavity. The torso exterior must be modified, based on ultrasonic chest scans of living subjects, so that the chest wall is 10 to 15 mm thick. The rib cage must be assembled and cast in the torso shell using T.E. plastic. The organs must be cast in T.E. plastic, and we must find a lung material simulant.

Acknowledgment. We wish to thank Norm Boyer and the people in the plastics shop for their support, and Robert Taylor and Charles Harder for making and testing the apparatus for filling the bones.

Respirator Cartridge Efficiency Studies

Since the last Progress Report,¹⁰ we have summarized our experimental results into the following guidelines:

- The service life of the respirator cartridge was found to be inversely proportional to the flowrate, and no differences were observed between steady-state and pulsating flow.
- The log of the service time at any breakthrough percent was inversely proportional to the log of the concentration over a concentration range of 50 to 3000 ppm.
- Cartridge effectiveness was decreased at storage or use humidities above 65%, with use humidity exhibiting the dominating effect.
- The service time was found to be directly proportional to the amount of carbon available, but did not depend appreciably on temperature.
- Volatile material, especially the first members of each homologous group (i.e., methane¹, methyl amine², methyl chloride, etc.), exhibited relatively short breakthrough times.

From our experimental data we have developed the empirical expression to estimate the service lifetime. The 10% breakthrough time can be calculated from

$$t_{10\%} = \frac{2.4 \times 10^6 W_c (a+b)}{QMC^{2/3}}$$

where

- W_c = weight of carbon in cartridge (g),
- i = normal boiling point of gas or vapor ($^{\circ}\text{C}$),
- a, b = coefficients for a particular solvent class (see Table 4),

Table 4. Solvent class coefficients

Solvent type	Boiling point range ($^{\circ}\text{C}$)	Coefficients a	Coefficients b
Acetates	50 to 190	-0.050	0.0038
Alcohols	60 to 160	-0.46	0.0071
Alkanes	20 to 200	0.095	0.0022
Alkyl benzenes	80 to 220	0.12	0.0074
Amines	-10 to 220	0.037	0.0033
Ketones	50 to 220	0.034	0.0029
Monochlorides	-30 to 250	0.032	0.0033
Dichlorides	40 to 250	-0.092	0.0048
Trichlorides	60 to 200	-0.080	0.0056
Tetrachlorides	70 to 200	0.19	0.0049

Q = average breathing rate (litres/min),

M = molecular weight (g/mole),

C = concentration (ppm).

This equation is a good approximation in the 0-40 $^{\circ}\text{C}$ temperature range, but a factor must be added for unusually humid conditions or if the cartridges have been stored in a damp environment (see Ref. 11).

A System for Measuring Respirator Leakage

To function properly, a respirator must form a good seal between the facepiece and the wearer's face. If the respirator fits poorly, 30% or more of the inhaled air can be leaked into the mask without purification.

A system for quantitatively testing respirator fit has been installed in the respirator facility at Bldg. 324. This system is designed to measure accurately facepiece-to-face leakage while the wearer performs specified exercises that simulate working conditions. The result is an improved respirator program since personnel are assigned the make of respirator which provides the best fit.

The basic components of the system are:

- A diethylphthalate (DOP) aerosol generator,
- a test chamber,
- an aerosol sampling system consisting of a light-scattering photometer and a recorder.

Before entering the test chamber, the subject dons the respirator. Each respirator is equipped with a special probe, inserted through the facepiece at a point that does not disturb the facepiece fit. The probe is placed close to the nose and mouth to detect leaks into the interior of the mask (Fig. 10). Each respirator is equipped with a standard high-efficiency particulate filter.

The subject enters the chamber and connects the respirator to the sample system with a section of plastic tubing. DOP aerosol is introduced through a plenum located in the ceiling of the chamber. The aerosol generator is a Naval Research Laboratory four-jet model which produces a liquid droplet aerosol of 0.7 μm aerodynamic mass median diameter (AMMD).

The generator operates by forcing compressed air through calibrated nozzles into the DOP liquid. Aerosol is introduced into the chamber until a concentration of approximately 100 mg/m^3 is achieved. The generator is turned off, and the aerosol concentration is measured with the photometer.

To begin the test, the subject performs a series of exercises while air is sampled continuously from inside the facepiece at a flow rate of 8 litres/min. The chamber operator instructs the test subject via an intercom. The exercises include normal breathing, deep

Fig. 10. Full-face respirator with sample probe.

breathing, turning the head from left to right and up and down, smiling, frowning, reciting the alphabet, and a repeat of normal breathing. Each exercise is

Fig. 11. Respirator test chamber.

continued for approximately 30 sec. Figure 11 shows the test chamber in operation.

The aerosol sample system consists of a forward light-scattering photometer, Air Techniques Inc., Model TDA-2C, equipped with a strip chart recorder. It is capable of detecting face piece leakage from 0.01 to 100%.

Tests can be repeated with different model respirators until an acceptable fit is achieved. The chamber is used most often for performing routine fit tests on employees who wear full-facepiece respirator. Employees must be refitted annually since facial shape can change. In addition, the chamber is a valuable tool for determining fit characteristics of new respirators which are under evaluation. The system can also be used for leak testing of half-mask and air-supplied respirators, as well as air-supplied suits.

INSTRUMENT DEVELOPMENT

Absolute Calibration of de Pangher Long Counter

A de Pangher long counter is a very efficient neutron detector whose response is essentially constant from 0.5 keV to 20 MeV.¹²⁻¹⁴ In the past, a great deal of work has gone into demonstrating that the response is highly reproducible from one detector to another,¹⁴ and to verify its independence of energy.¹³⁻¹⁵ Less work has gone into measuring the absolute neutron sensitivity.

Absolute neutron sensitivity can be measured with neutron sources whose emission rates are well known provided the detector count rate due to scattered neutrons can be determined. Monte Carlo or analytical calculations may be carried out to estimate the intensity of scattered neutrons at the counting

position. However these calculations are not only imprecise, they are not particularly applicable since the long counter response is highly anisotropic. Therefore, experimental methods must be used.

The absolute neutron sensitivity S is defined as follows:

$$S = \frac{A}{\phi_0} \quad (1)$$

where

A = detector count rate (counts/sec),
 ϕ_0 = neutron flux at the effective center due to neutrons incident on the front face of the long counter ($n/cm^2/sec$).

The count rate in the long counter due to a point, isotropic source is given by Eq. (2)

$$A = \frac{SQ}{4\pi (R+C)^2} + B(R) \quad (2)$$

where

S = absolute neutron sensitivity,

Q = neutron emission rate of the source,

R = distance to detector face,

C = distance between face and effective center,

B(R) = count rate due to scattered neutrons.

Location of the effective center varies with neutron energy, but is well known.¹⁶ Thus if the background count rate due to scattered neutrons B(R) and the absolute neutron sensitivity S are known, either the emission rate of an unknown source or the neutron flux may be determined directly from the detector count rate. The discussion which follows will describe the methods used to determine for a ²³⁸PuBe neutron spectrum the background contribution to the count rate, and measurement of the absolute neutron sensitivity.

One method of determining B(R) is suggested by multiplying Eq. (2) by (R + C)² to obtain Eq. (3):

$$(R + C)^2 A = \frac{S}{4\pi} Q + (R + C)^2 B(R). \quad (3)$$

It may be argued on the basis of first principles that an isotropic point source emitting neutrons in an evacuated spherical chamber will produce a uniform intensity of scattered neutrons within the chamber provided scattering in the walls is nearly isotropic. The low-scatter cell in the LLL calibration facility approximates this situation crudely, except for the contribution due to air scatter. Consequently, background count rate in the cell due to scattering would not be expected to vary greatly with location. If we measure detector count rates at many radii and plot the results, i.e., (R + C)² X A vs (R + C)², the data should resemble a straight line with Y-intercept SQ/4 π and slope B₀, the background count rate. This technique is potentially precise, but is very time-consuming and requires great care in data analysis.

The second method consists of directly measuring B(R) using a shield which obscures the source from the detector. Such a shield has been designed and constructed. Details are given below, together with a determination of the absolute neutron sensitivity S and the background count rate due to scattered neutrons.

Design of a Neutron Cone Shield. Optimum design

of a neutron shield for the purpose described above is guided by two basic considerations:

1. The shield should completely obscure the source from the detector. Its transmission must be low for neutron energies above the detector threshold.
2. The shield should not contribute significantly to the background intensity or prevent the detector from quantitatively measuring it.

At the outset, we chose borated polyethylene containing 5% by weight natural boron as the shield material. That choice was based on cost and maximum attenuation per unit weight. We decided on a conical shape with apex at the source to minimize the shield's contribution to the background intensity. Since we also desired to minimize the apex angle, we selected an angle of 22° so that the shield would have an angular width (viewed from the source) which was just large enough to obscure the face of the long counter at R = 1 m.

Optimum axial length of the neutron shield represents a compromise between a very thick shield which has low transmission, but may obscure much of the background intensity from the detector's field of view, and a thin shield which has greater transmission. The minimum thickness required for adequate attenuation is a function of incident neutron energy. Consequently, we designed a shield of variable axial length.

Our cone shield consists of a horizontal stack of 5.1-cm-thick pieces of borated polyethylene. A cross section of one piece is shown in Fig. 12. All are

Fig. 12. Cross section of cone shield.

supported on an aluminum frame to form the completed assembly shown in Fig. 13. We obtained increasing thicknesses by installing additional pieces up to a maximum length of 70 cm. Each additional piece has an increasing cross section so that the envelope of the shield resembles a conical shape.

To establish the minimum required thickness as a function of incident neutron energy, a series of calculations was carried out using the MORSE-L Monte Carlo neutron transport code. Monoenergetic neutrons were assumed to be emitted isotropically at the apex of the cone. For each source energy and shield length, the total intensity of neutrons emerging from the shield at energies above 1 keV were compared to the intensity with no shield. In all cases, the contribution due to air scatter was included. A partial summary of the results is shown in Fig. 14. A family of curves

Fig. 13. Neutron cone shield.

Fig. 14. Shield thickness required to attenuate neutrons at given energy. Each curve gives thickness for a particular value of neutron transmission.

shows the appropriate shield length to provide an indicated neutron transmission for each source energy.

Measurements. The long counter absolute neutron sensitivity was calibrated using both background determination methods described above. Preparatory to the measurements, the counter was carefully set up according to the Standard Calibration Procedure.* First, we measured the count rate due to a calibrated source at distances of 1.0 to 4.5 m. The $^{238}\text{PuBe}$ source used (ID #300094) was calibrated at the National Bureau of Standards on February 19, 1973. Its emission rate, corrected for decay on December 5, 1975, is $Q = 3.93 \times 10^6$ n/sec ($\pm 2.3\%$). The count rates observed are plotted in Fig. 15 in the form $(R + C)^2 \times A$ vs $(R + C)^2$ according to Eq. (3). Linear regression has been used to fit a straight line to the data. In no case do the data deviate from the straight line by as much as $\pm 1\%$. This supports the expectation that the background intensity is fairly uniform. Analysis of the error matrix shows the uncertainty in the Y-intercept to be $\pm 1\%$, while the uncertainty in slope is $\pm 8\%$.

Using the known source strength and the fitted line of Fig. 15 we determined the absolute neutron sensitivity and the background count rate from Eq. (3):

$$\begin{aligned} S &= 3.23 \text{ counts/n/cm}^2 & (\pm 3\%) \\ B_0 &= 3.10 \text{ counts/sec} & (\pm 8\%) \end{aligned}$$

The background count rate has also been measured using the cone shield. Figure 16 shows the results. Examination of the figure indicates that the cone shield may increase the background intensity in the vicinity of the shield by as much as 25%. However, the resulting uncertainty in net count rate (signal less background) is generally less than $\pm 3\%$ at all radii. With this in mind, the absolute neutron sensitivity of the

*In principle, the long counter is used to determine the ratio of count rates from an unknown source to that from an internal calibrated source. Use of the long counter as a precision absolute monitor then requires meticulous attention to the setup and calibration procedure before an intensity measurement is made:

1. The detector output should be RC shaped with time constants: 1- μ sec integration, 1- μ sec differentiation.
2. Anode bias = 1750 V.
3. Calibration source rod (ID #300044) should be fully inserted and the lower level discriminator on the output pulses adjusted to provide 6.06×10^{-3} ($\pm 1\%$) counts per source neutron. On December 5, 1975, the source strength was 1.80×10^6 n/sec, so the proper count rate on that date is 108.2 cps ($\pm 1\%$).
4. The detector count rate should never exceed 250 cps during a measurement.

Fig. 15. $(R + C)^2$ times count rate versus $(R + C)^2$.

long counter has been derived from calibrated source measurements using the cone shield to determine the background contribution. These measurements are summarized in Table 5. Neutron sensitivities determined from two calibrated sources using both methods are included in the table. Sensitivities obtained agree within errors.

Fig. 16. Wall scatter count rate measured using a cone shield. Source was $^{238}\text{PuBe}$ (3.93×10^6 n/sec).

Finally, the cone shield has been used to map the background contribution to the count rate along several radii in the low-scatter cell. For this purpose, the source was a $^{238}\text{PuBe}$ with an emission rate of 3.93×10^6 n/sec. Figure 17 shows the results of the measurements where the background count rate is given as a fraction of the total count rate.

Conclusion. The de Pangher long counter may be used for absolute neutron emission rate or intensity

Table 5. Absolute neutron sensitivity of the de Pangher long counter for a $^{238}\text{PuBe}$ spectrum

Source ID	Calibration date	Method of measuring scatter count rate	R (cm)	Sensitivity, S (counts/n/cm ²)
300094 ^a	2/19/73	$1/R^2$	—	3.28 ($\pm 3\%$)
300094 ^a		Cone shield	100	3.25 ($\pm 3\%$)
300094 ^a		Cone shield	150	3.21 ($\pm 3\%$)
300094 ^a		Cone shield	200	3.34 ($\pm 5\%$)
300044 ^a	8/7/69	Cone shield	150	3.10 ($\pm 3\%$)
30084	2/28/73	Cone shield	300	2.97 ($\pm 5\%$)
3000101	3/7/73		150	3.10 ($\pm 5\%$)
Mean of first = 3.24 ± 0.10 five entries				

^aCalibrated at NBS.

Fig. 17. Floor plan of low-scatter cell. Numbers circled are fraction of total long counter count rate due to scattered neutrons when counter is pointed toward source.

measurements with a precision of $\pm 3\%$ provided it is used as prescribed in the Standard Calibration Procedure. It is necessary, however, to accurately determine the count rate contribution due to scattered neutrons. This determination has been made very simple and precise through the use of a cone shield. It is nevertheless important to note that the long counter may not be used to directly determine the total intensity due to scattered neutrons since its response is not isotropic.¹⁶ Cone shield measurements provide only a lower bound on the intensity of scattered neutrons.

A Portable Liquid Scintillation Counting System for Tritium Analysis

A complete liquid scintillation counting system has been designed for application in emergency field response such as Hot Spot. The system, including batteries, is housed in a weather-resistant carrying case, shown in Figs. 18 and 19. A tritium sensitivity of 90 nCi/l is obtained with two RCA 8575 photomultipliers in coincidence. The sample handling mechanism uses standard vials, and it is not necessary to remove high voltage when changing samples. When the liquid

Fig. 18. Complete liquid scintillation counting system. Pancake probe and sample box are normally well behind control panel.

Fig. 19. Batteries and electronics under hinged control panel.

scintillation system is not in use, a G. M. pancake probe, sharing the electronics, is available either with a holder for swipe counting or for external monitoring. Space is also available for a box to hold standards and samples and, incidentally, to prop up the carrying case lid in the vertical operating position.

System Description. The system includes two single-channel analyzers (SCA), a choice of manual timing or a preset timer 0.1 to 80 min using a crystal-controlled time base and a light-emitting diode readout. The readout is switched to whichever channel is desired - time or either of the SCAs. The readout is turned on with a pushbutton, and automatically goes off after 30 sec to conserve battery life.

The normal operating time from batteries is 8 hr using the two 1.5 AH, 12 V gel cells. The batteries can be recharged in 8 hr using the internal charging and control circuits. Also, the unit may be operated on-line indefinitely.

The system can be operated over the temperature range of -10° to 50°C . The high-voltage power supply is well regulated from 900 to 2000 V with a temperature coefficient of less than $-0.02\%/^{\circ}\text{C}$. The HV automatically adjusts to 900 V when the pancake probe is used.

A Packard Instrument Company light pipe enhances light collection from the sample vial. Transparent silastic is used to couple the light to the PM photocathode windows. The PMs are magnetically shielded and vacuum potted in silastic for protection against shock and moisture.

A double slide mechanism keeps the counting chamber light-tight even if the top slide is accidentally moved. When the vial cover is in place both slides move together, and the cover cannot be removed except in the right-hand position. In this position, the sample vial is loaded as shown in Fig. 20. The cover is used as a handle to move the sample over the counting chamber. The front panel knob attached to a cam is then turned to operate the lever that lowers the sample into the counting chamber, as shown in Fig. 21.

The carrying case is $340 \times 530 \times 195$ mm in size, and the complete system weighs 17.5 kg.

The circuitry for the first unit is neat breadboard construction. Printed circuit boards will be designed for the next unit, but the outside appearance will be similar. It will be necessary in bad weather to find shelter in which to operate this unit as we do not plan to moisture-seal the panel and controls.

Fig. 20. Sample vial placed in load position without cover.

Fig. 21. Lever has lowered sample into counting chamber.

Application. The channel ratio method was used to obtain calibration curves to compensate for quenching. We obtained the setting for Channel 2 by measuring the best signal-squared-to-background ratio and the

Fig. 22. Tritium calibration curves.

setting for Channel 1 by varying the discriminator settings until we found the best straight-line calibration curve. The best calibration curves for two different count rates are shown in Fig. 22. The discriminator settings are 40-140 for Channel 1 and 40-200 for Channel 2. To use the curves, the operator calculates the ratio of net counts in Channel 1 to Channel 2 and uses this ratio to obtain the efficiency from the curve. The dps in the sample is then calculated using the efficiency and the net count in Channel 2.

The quench curves in Fig. 22 are for standards of 197 and 39,000 dps. It can be seen from these curves that the highest efficiency for the 197 dps curve is 20.0% and for the 39,000 dps curve is 17.3%, which shows that there is a counting loss of 13.7% at the high count rate.

The average background using 15 ml of scintillation cocktail and a plastic vial is 2.1 cps in Channel 2. This is about ten times that obtained for a shielded counter (this system does not have shielding), but it is enough for field use.

Further work will be done to attempt to increase efficiency and perhaps reduce counting loss at high rates.

Tritium Detector Calibration Sources

The purpose of this project was to devise a method for calibrating the existing Tritium Surface Contamination Detectors.¹⁷ Considering the very short range of the weak beta particles emitted by tritium ($E_{\text{max}} \approx 18 \text{ keV}$) it was decided that a thin plastic film containing tritium offered the best possibility. We abandoned preliminary attempts at in-house tagging of Formvar plastic and sent a sample of Formvar to ICN Isotope and Nuclear Division.

The method would be to fabricate a thin film of the plastic using the same technique employed in producing the windows used on the detector probes. This is done by immersing a glass plate in a solution of Formvar dissolved in ethylene dichloride. The glass plate is then withdrawn very slowly, leaving a very uniform thin film of plastic on the plate. The edges of the plate are then scored with a scalpel, and the plastic film is floated off onto the surface of water. The film is then lifted from the surface with a suitable holder and dried in an oven to remove moisture and allow the film to stretch firmly onto the holder. The detector probe can then be placed against the film and the count rate recorded. The film is then cut out of its holder, dissolved in a suitable solvent and the total activity obtained accurately by liquid scintillation. The detector efficiency is then calculated from these data.

The solvent used must be capable of dissolving Formvar, be miscible with the liquid scintillator used,

Fig. 23. Effect of solvent on count rate.

and have a minimum quenching effect on the liquid scintillator.

Small amounts of water containing tritium were added to 15-ml portions of liquid scintillator, and the effect on count rate was observed for the addition of small volumes of three Formvar solvents:

ethylene dichloride
p-dioxane
dimethyl formamide

The resulting graphs are shown in Fig. 23. Other Formvar solvent were not used because they were either uncommon, hazardous or otherwise impractical.

A consistent volume of solvent was used to dissolve the plastic film each time. The effect of 1 ml of each solvent on the count rate of the liquid scintillator is shown in Table 6.

From these data it is apparent that p-dioxane is the best of those solvents tested, with only a 3% decrease in count rate at 1 ml.

Next, the possible effect of the dissolved Formvar on the liquid scintillator was studied. Several dilutions of Formvar in p-dioxane were made, and 1 ml of each was added to liquid scintillator containing tritium. The results are shown in Table 7.

Table 6. Effect of 1 ml of solvent on count rate

Solvent	Percent of initial count rate
Ethylene dichloride	70
dimethyl formamide	87
p-dioxane	97

Table 7. Effect of dissolved Formvar on count rate

Percent Formvar (w/v)	Percent count rate reduction
0	3
0.0001	3
0.001	3
0.01	3
0.1	5
1.0	3

The data show that with the dilutions used, the p-dioxane alone is responsible for the quenching effect on the liquid scintillator. For this reason each time a source was dissolved, 1 ml of p-dioxane was used.

The Formvar sample returned from ICN with a tritium concentration of 2.48 $\mu\text{Ci}/\text{mg}$. A 4% solution in ethylene dichloride was made of this for fabrication of the sources. The source holders are stainless steel, and are shown in Fig. 24.

Two sources were then fabricated as described earlier. It should be mentioned here that the films are fabricated such that each side of the glass plate has a film identical to the other. If both are carefully floated onto the surface of the water and then great

care is taken to see that the films are stretched uniformly onto the holders before drying, two identical sources can be produced.

One of the sources was then counted with each of three detectors. The source was placed on the probe windows so that only the window area of the source was counted. The second source was also counted and then stored for 30 days to determine if the source activity changes with time. When this source was recounted, the count rate was observed to be unchanged.

Each source was dissolved after counting and the total activity was determined by liquid scintillation. The results are shown in Table 8.

To calculate the significance of any existing self-absorbance within the sources, the following equation¹⁸ was used:

$$\rho R = 0.54E - 0.13(1 - e^{-3.2E})$$

where

$E = \beta$ energy in MeV,

$R = \beta$ range in cm, and

$\rho =$ density in g/cm^3 (1.228 for Formvar).

The specific activity of the tagged Formvar and the total activity of the source was used to calculate the thickness of the sources. The results are shown in Table 9.

Table 8. Detector efficiency results

	Type of detector	LLL No.	Background (counts/min)	Source (counts/min)	Efficiency
Source No. 1 1.656×10^6 dis/min	PAC-4G	164102	500	2.0×10^4	1.18%
	PAC-4G	164103	300	1.5×10^4	0.89%
	PAC-3G	94149	200	5.5×10^3	0.32%
Source No. 2 2.038×10^6 dis/min	PAC-4G	164102	500	2.4×10^4	1.18%
	PAC-4G	164103	300	1.8×10^4	0.86%
	PAC-3G	94149	200	6.0×10^3	0.29%

Stainless steel source holder

Fig. 24. Stainless steel source holder.

Table 9. Source self-absorption data

ρ range in Formvar	
Range E-max (18 keV)	2000 nm
Range of E-avg (6 keV)	600 nm
Thickness of Formvar	30 nm

Conclusion. The low efficiency of the detectors most likely results from parameters such as the tritium beta first traveling a distance in air to reach the detector probe, and the thickness of the Formvar window on the detector probe having a linear density of 50-100 $\mu\text{g}/\text{cm}^2$.

An actual contaminated surface counted with a 95% confidence (2-sigma) would require a total activity of 4×10^3 dis/min for detection.

Four more sources have been fabricated, and are being stored for use when future calibration becomes necessary.

FIRE SAFETY

Calibration and Preliminary Use of High Flux Heater for the NBS/ILL Smoke Chamber

Introduction. Shortly after Gross *et al.*¹⁹ developed a smoke density chamber for measuring the opacity of smokes under flaming and nonflaming exposure, we built a similar apparatus and have used it with modifications for examining the opacity of smokes from pyrolyzing and combusting materials. Our modifications have consisted of adding a ventilating capability, an improved pilot flame burner, and a gas sampling and analytical system. This work has been reported elsewhere.²⁰⁻²²

During the course of our work, we developed the concept of bracketing fire parameters in which a specimen of a material of interest could be exposed to (1) a low radiation flux versus a high flux, (2) no flame or a pilot flame, and (3) no ventilation vs some "standard" ventilation condition in the chamber. The idea is illustrated in Fig. 25, taken from Ref. 23. By using these extreme conditions in various combinations, we believe that the boundaries of smoke production (evolution of particulates and gases) can

be determined. If these results can be correlated with large-scale fire tests such as we are planning,²⁴ it may be possible to simulate fire conditions in a laboratory-size apparatus with a considerable saving in time and money.

For the past several years, we have endeavored to develop a heater capable of yielding a high radiant flux on the face of the sample (10-20 W/cm^2). Our difficulties to date have been with the stability of such a heater. Most of those tried burned out after a few tests. During the past two years we have been working with the Mellen Company of Penacook, New Hampshire to develop a high-flux heater which could be used with confidence. At the same time Chien and Seader²⁵ obtained a commercial heater which developed a heat flux of 7.5 W/cm^2 , but which failed after several uses at 10 W/cm^2 . In the last year we worked out a successful prototype, and have recently received and calibrated a unit which we feel will provide a high radiant flux and which will also last.

Description of Heater. The heating element consists of a half cylinder of ceramic 75 mm i.d. $\times$ 85 mm long, containing a number of internal slots parallel to the axis in which is laid a continuous helical coil (4.8-mm diameter) of 50/50 Pt/Rh wire 1.07 mm in diameter. The total wire length is 9.16 m; its cold resistance is 1.4 Ω .

The element is mounted vertically in a cubical stainless steel box 125 mm on edge; it is separated from the walls of the box with high-temperature ceramic block and blanket insulation. One end of the box has an 85-mm square opening so that the heat from the inside of the element can radiate onto the test specimen; the opposite end is closed with an asbestos cement board slab, pierced with holes for electrical connections to the element wire and for a Pt/Rh thermocouple which has its active junction located in the center of the face of the element.

The outside box has sheet metal ears welded on. These ears are pierced with slotted holes for adjusting the orientation and position of the heater to make it parallel to the face of the test specimen and with a distance of 38.1 mm between the specimen and the heater opening.

Fig. 25. Boundary conditions of a fire.

Fig. 26. Front view of new square heater showing interior arrangement.

Other Modifications. The original circular heater used a 38.1-mm air-cooled Gardon foil radiometer to establish the heat flux on the specimen face. This in turn was calibrated against a 38.1-mm blackened round copper disc as a primary calibration standard. We felt it more realistic to use a 65-mm square copper disc (this is the actual size of the exposed specimen), and have done so. Further, because the air-cooled radiometer is unusable at our high irradiances, we use a Medtherm water-cooled radiometer with a double CaF_2 window for normal control of the heat flux. This radiometer has an included viewing angle of 30° .

Because we are developing a load cell system to determine specimen weight loss during a test, it is necessary to fix the sample holder and slide the heater back and forth in front of the specimen and the radiometer, respectively. In the original chamber, the furnace position was fixed and the specimen and radiometer were slid back and forth in front of the

heater. The assembled heater in actual operation is shown in Figs. 26 and 27.

Calibration of Heater. Using the Medtherm water-cooled radiometer, we have compared the use of both the round and square blackened copper discs for the old circular heater and the new square heater. The data are shown in Figs. 28 and 29, and Table 10. It is evident from these figures that a good correlation exists between the results obtained using the round copper disc and the square copper disc, probably because the Medtherm radiometer sees only the center of the heater. However, we feel that the values using the square copper disc as a primary standard are more realistic since the dimensions of the square disc are the same as those of the exposed specimen.

Preliminary Results. We checked the new square heater against the old circular heater by exposing samples of α -cellulose paper and ABS plastic to various

Fig. 27. Side view of new square heater.

heat fluxes. (These materials are supplied by the Bureau of Standards as check materials.) We compared the results of the two heaters using as criteria the maximum specific optical densities obtained and the times to a specific optical density of 16 (obscuration time). The results are shown in Fig. 30 and Table 11. From these data it appears that the correlation is fair to good at a heat flux of 2.0 W/cm^2 (below "standard" exposure), but is somewhat better at the "standard" 2.5 W/cm^2 . In any event, the shapes of the curves are quite similar.

Future Work. We propose to examine specimens of Douglas fir and white pine at various radiant fluxes up to the limit of the new heater, $\sim 13 \text{ W/cm}^2$, under piloted and nonpiloted exposure and under various degrees of ventilation. We may compare these same materials using the old circular heater within its limited heat flux. Following this, we shall examine a selected group of those materials we have tested in the past to evaluate the smoke particulate and gas evolution under the higher irradiances.

Fig. 28. Calibration of round heater using round Cu disc square Cu disc, and water-cooled radiometer.

Fig. 29. Calibration of Mellen heater using Medtherm radiometer and blackened copper disc primary standards (least-squares fit).

Table 10. Calibration data for special model 10 Mellen Heater-2500-4 X 4^a

Heater		Radiometer output ^b (mV)	Radiant flux on blackened Cu discs	
Voltage (V)	Current (A)		38-mm circular (W/cm^2)	65-mm square
40	10.5	3.80	1.99	1.92
60	13.25	8.20	4.71	4.11
80	16.0	14.0	7.31	7.15
100	18.8	20.3	11.37	10.96
110	20.0	21.4	12.92	12.28

^aMellen Co. Inc., Penacook, New Hampshire 03301.

^bMedtherm Model TGRW-22-30-330-Double CaF_2 window.

Fig. 30. Effect of heat flux on D_m and T_{16} for NBS paper and ABS plastic using standard (round) and new (square) heaters. D_m = maximum specific optical density; T_{16} = time in min to $D_s = 16$; D_s = specific optical density. (a) NBS paper-round heater. (b) NBS paper-square heater. (c) ABS plastic-round heater. (d) ABS plastic square heater.

Table 11. Salient smoke density results obtained on NBS cellulose and ABS plastic using standard circular heater and new square heater

Circular heater		Square heater		Heat flux (W/cm ²)
Dm ^a	T-16 ^b	Dm ^a	T-16 ^b	
NBS paper (nonflaming exposure)				
164	2.6	129	3.3	2.0
165	1.7	155	1.9	2.5
178	1.1	—	—	3.0
—	—	160	1.1	3.5
—	—	92	0.5	5.0
ABS plastic nonflaming exposure)				
153	2.0	247	2.1	2.0
297	1.6	249	1.5	2.5
393	1.1	—	—	3.0
—	—	563	1.6	5.0
—	—	558	0.3	7.5
ABS plastic (flaming exposure)				
571	0.5	516	0.5	2.0
558	0.5	536	0.4	2.5
549	0.4	—	—	3.0
—	—	605	0.3	5.0

^aDm = Maximum specific optical density obtained.

^bT-16 = Time (min) to reach a specific optical density of 16 (this is usually taken to mean time to visual obscuration of smoke).

- Notes: 1. Round heater was calibrated against round and square Cu discs using Medtherm radiometer.
2. Square heater was calibrated against square Cu disc using Medtherm radiometer.

DECONTAMINATION

Enhanced Filtration

The major objective of the enhanced filtration project is to extend the effective service life of high-efficiency particulate air (HEPA) filters. This project is currently being funded by ERDA's Nuclear Fuel Cycle and Production Division (previously the Division of Waste Management and Transportation).

A sodium chloride test aerosol is produced from a system similar to the one used by the British to determine filter efficiency.^{26,27} Figure 31 shows a photograph of the testing apparatus. Basically, the system consists of an air mover, aerosol generator, aerosol modification section, filter holder, and service life diagnostic equipment. The experimental set-up is described more fully in Ref. 28. The test aerosol generated is 0.8 μ m aerodynamic mass median diameter (AMMD) with a geometric standard deviation of 3.9.

Most of our experiments have centered around high-voltage techniques to induce particle agglomeration. If small particles can be coalesced into larger agglomerates, they can be removed more efficiently with less pressure drop. A typical agglomerator is shown in Fig. 32. It consists of a series of charged wires and rods (5-10 kV). Particles passing through the grid are charged, and agglomeration is more likely to occur.

Figure 33 shows some preliminary results from our particle agglomeration experiments.

Samples were taken just downstream from the wire and rod assembly. These figures show the relative abundance of both mass of sodium and condensation nuclei (particles between 0.0025 and 0.1 μ m) as a function of agglomerator voltage. This was done with both ac and dc voltage with negative charge on the wires. These experiments indicate that the most

Fig. 31. Photograph of sodium chloride aerosol generator.

pronounced agglomeration and collection are observed when dc voltage is used with the negatively charged wires and with the rods grounded. As the voltage is raised, little change is noted in either the particle or mass concentration until 5 kV. As the voltage is further increased, the number of particles begin to decrease faster than the mass concentration, indicating that some agglomeration is taking place. However, the dominant effect is collection of the salt on the rods. The goal of future experiments is to decrease the particle number without appreciably reducing the mass concentration. The rods and wires must agglomerate rather than precipitate if long-term field usefulness is to be perpetuated.

Fig. 32. Agglomerator showing alternately charged wires and rods.

Fig. 33. Relative abundance of particles and salt penetrating the wire and rod agglomerator as a function of voltage for negatively charged wires. (a) dc; (b) ac.

OTHER

NUTSPEC, A Code for Unfolding Neutron Pulse-Height Data

A series of earlier progress reports has chronicled the progress of a neutron spectroscopy capability at the LLL calibration facility.²⁹⁻³⁵ This report will give an update on the data reduction capability now available, and some operational aspects of the system. All pulse-height data are acquired on either a Nuclear Data ND 4420 computer analyzer or a Nuclear Data ND 100 pulse-height analyzer with dual parameter module. Raw data consist of dual or single parameter pulse-height spectra on 9-track magnetic tape which is decoded and transposed to a disc file on the CDC 7600. Any of a large array of detectors and detector types may be used including methane, hydrogen, or ³He proportional counters or stilbene and NE 213 scintillators. The code NUTSPEC reads the data files, obtains the necessary detector parameters from an internal library, and then unfolds the data to produce a neutron energy spectrum from each data file. Each detector type requires a unique unfolding strategy, and these are summarized briefly below.

Scintillators. Stilbene crystals and NE 213 liquid scintillators produce light output due to proton recoils and Compton electrons. Analysis of the latter is prevented by pulse-shape discrimination. Unlike gamma-ray spectrometers, these neutron spectrometers

produce a pulse-height spectrum which is related to the integral of the neutron spectrum. That is,

$$P(E) = \int_E^{\infty} \frac{N \sigma_s T \phi(E') dE'}{E} \quad (4)$$

where

$P(E)$ = proton recoil distribution,
 N = number of hydrogen atoms in detector active volume,
 σ_s = n,p elastic scattering cross section,
 T = duration of exposure,
 E = neutron energy,
 ϕ = incident neutron flux.

Neutron spectral intensities are obtained by differentiation,

$$\frac{dP(E)}{dE} = - \frac{N \sigma_s T \phi(E)}{E}$$

or

$$\phi(E) = - \frac{E \frac{dP(E)}{dE}}{N \sigma_s T} \quad (5)$$

Fig. 34. Neutron spectrum for D(d,n)He neutron source.

Fig. 35. Neutron spectrum for $^{238}\text{PuBe}$ source.

Unfortunately, the derivative of the proton recoil distribution is not easily obtained from the detector pulse-height spectrum. Some of the complications are listed below, more or less in order of importance.

- The relation between proton energy and pulse height is significantly nonlinear.
- Statistical fluctuations in the data tend to be amplified when the spectra are differentiated.
- Distortions in the pulse-height spectra arise from multiple scattering of neutrons in the detector and recoil protons which escape the active volume with significant energy.
- Scattering in carbon, a major constituent of the detectors, distorts the neutron spectrum in the vicinity of the detector.

The code NUTSPEC deals with all of the above difficulties to some extent. First, B-splines are used to relate scintillator light output to proton energy. The analytical representation is smooth so as not to inject noise into the slope of the data, and is accurate over the range of data, falling well within experimental uncertainties. Second, digital filtering techniques are used to obtain the slope while smoothing the statistical fluctuations. The code allows the user to choose the degree of smoothing as the data are processed. Spectral distortions due to multiple scattering and escape are partially compensated for by using an empirical relation due to Broek and Anderson,³⁶ and to some extent, they are reduced by poor detector resolution (7-10%). Finally, since neutrons scattered in carbon are superimposed on the incident neutron spectrum, their contribution to the detected intensity may be

calculated and subtracted from the unfolded results. This is done by the code.

Figure 34 shows a neutron spectrum obtained near the neutron generator in the LLL calibration facility. Incident neutrons are produced by the D(d,n) reaction and have energies $E_n \approx 2.8$ MeV. An attempt has been made in the code to subtract the contribution due to neutrons scattered in carbon. It is clear that the subtraction process does not completely remove the scattered neutron contribution. An investigation is now underway to find out why. Spectral measurements have also been made on unmoderated PuBe and fission source. Figure 35 shows results obtained for a $^{238}\text{PuBe}$ neutron source.

Operationally, the procedure required for spectral measurements with organic scintillators is fairly simple. Energy is calibrated using a ^{22}Na gamma-ray source. Pulse-shape discrimination is adjusted in the presence of the neutron source while acquisition of a dual-parameter pulse-height spectrum is observed. Then with analysis of electron events suppressed, a single-parameter pulse-height spectrum is obtained and unfolded.

Proportional Counters. Because of their much lower densities, proportional counters are less sensitive and more vulnerable to distortion due to particle escape than are the scintillators. Consequently, spectral distortions play a large role in the unfolding process. In NUTSPEC, a response matrix is calculated for each proportional counter. A neutron spectrum solution is obtained iteratively by estimating a neutron spectrum,

Fig. 36. Calculated response functions for 3 atm, 3.8 cm diameter hydrogen proportional counter.

calculating the corresponding pulse-height spectrum, and then comparing the calculated and measured pulse-height spectra. The iterative process is continued until chi-squared per degree of freedom for the calculated pulse-height spectrum is less than some value supplied by the user. Figure 36 shows a plot of selected rows of the response matrix for one measurement, and Fig. 37 shows the unfolded neutron spectrum obtained from a 250-keV neutron source.

Fig. 37. Unfolded neutron spectrum for a mono-energetic 250 keV neutron source obtained via the ${}^7\text{Li}(p,n)$ reaction.

In the case of hydrogen-recoil proportional counters, the iterative unfolding process is greatly accelerated by a good first guess for the neutron energy spectrum. This good first guess is obtained using the same derivative technique as for scintillator spectra. Generally, chi-squared per degree of freedom is reduced to the range 2-3 in only a half dozen iterations.

Operationally, spectral measurements with proportional counters are more complicated than with scintillators. Energy is calibrated using a tail pulser and calibrated coupling capacitor at the preamplifier input. The pulser is adjusted to deposit a known charge at the detector anode. The electronics gain is adjusted to establish a certain pulser location in the pulse-height spectrum. Gas multiplication characteristics curves³⁵ are then used in conjunction with the anode voltage and test pulser calibration to infer proton energy deposition from the pulse-height data.

To determine and subtract the response due to gamma rays, data are generally acquired in dual parameter. A computer program is then used to set an acceptance area in the rise-time/pulse-height space, and a single-parameter spectrum is constructed for unfolding. The code NUTSPEC makes use of the test pulse, input capacitor, and anode voltage data to establish an energy scale. Proton recoil response functions are first calculated for neutrons whose energies are above the useful energy range of the detector. An input must be supplied which describes this portion of the spectrum. Then the response matrix is used with the input spectrum to calculate the pulse-height response due to the high-energy neutrons. These events are subtracted from the measured pulse-height spectrum. A response matrix is then calculated for neutrons within the useful energy range of the detector, and the residual pulse-height spectrum is unfolded to produce the final neutron energy spectrum at low energy.

Results. Results of the unfolding process are given in three forms. First, a tabulation of cross sections and detector parameters that were used are given, followed by a table containing the unfolded results. An example is shown in Fig. 38. For each neutron energy bin, the result table gives the neutron flux per unit lethargy, kerma in units of erg/g-sec-MeV, and neutron dose equivalent in units of mR/hr-MeV. In addition, three integrals are given: the neutron flux due to neutrons at energy greater than E as a function of E , integral kerma, and integral dose equivalent. Second, a file is produced which contains only the differential flux $\phi(E)$ which may be preserved for later input to calculations of low-energy spectra. Third, a plot file is produced containing graphical display of detector

TIME DATE 16 18 53R 12/22/75
CODE VERSION 12/21R 13 58 09

HYDROGEN TUBE, 250KEV NEUTRON SOURCE

24 JAN 03

CHI-SQ PDF FOR FINAL FIT 3.21E+00 AFTER 6 ITERATIONS

DIFFERENTIAL RESULTS

RESULTS INTEGRATED FROM E TO INF

CHANNEL	ENERGY	FLUX DENSITY	KERMA	DOSE EQUIV	INTEG FLUX	INTEG KERMA	INTEG DOSE
	(MEV)	(NEUTRONS/CM ² -SEC-LETH)	(ERGS/CM ² -SEC-MEV)	(MRAD/HR-MEV)	NEUTRONS/CM ² -SEC-ABOVE E	(ERGS/CM ² -SEC-ABOVE E)	(MRAD-HR)
1	1.69E-02	5.801E-01	4.030E-05	1.14E-01	1.112E-05	3.572E+00	
2	2.39E-02	4.434E-01	3.699E-05	5.739E-02	1.026E-05	3.912E+00	
3	3.09E-02	4.160E-01	3.677E-05	4.005E-02	1.052E-05	3.51E+00	
4	3.79E-02	4.056E-01	3.491E-05	3.076E-02	1.026E-05	3.423E+00	
5	4.49E-02	4.320E-01	3.135E-05	2.245E-02	1.004E-05	3.592E+00	
6	5.19E-02	3.997E-01	2.674E-05	1.631E-02	9.821E-06	3.368E+00	
7	5.89E-02	3.312E-01	2.181E-05	1.178E-02	9.533E-06	3.330E+00	
8	6.59E-02	3.437E-01	2.209E-05	8.61E-03	9.40E-06	3.312E+00	
9	7.29E-02	3.142E-01	1.894E-05	6.12E-03	9.356E-06	3.282E+00	
10	8.01E-02	3.122E-01	1.973E-05	4.627E-03	9.185E-06	3.253E+00	
11	8.71E-02	2.495E-01	1.584E-05	3.43E-03	9.046E-06	3.221E+00	
12	9.41E-02	2.412E-01	1.13E-05	2.515E-03	8.935E-06	3.191E+00	
13	1.01E-01	2.246E-01	1.1E-05	1.77E-03	8.856E-06	3.160E+00	
14	1.08E-01	8.204E-02	5.022E-06	1.619E-03	8.725E-06	3.127E+00	
15	1.15E-01	5.017E-02	2.930E-06	9.495E-04	8.689E-06	3.116E+00	
16	1.22E-01	4.328E-02	2.433E-06	7.886E-04	8.70E-06	3.124E+00	
17	1.29E-01	1.088E-01	1.5E-05	1.825E-03	8.547E-06	3.067E+00	
18	1.36E-01	1.607E-02	1.820E-06	2.744E-04	8.459E-06	3.044E+00	
19	1.43E-01	3.985E-03	1.383E-06	6.579E-04	8.475E-06	3.046E+00	
20	1.50E-01	4.182E-02	2.026E-05	6.759E-03	8.488E-06	3.03E+00	
21	1.57E-01	2.542E-02	1.1E-06	3.89E-04	8.546E-06	3.003E+00	
22	1.64E-01	3.025E-02	1.3E-06	4.663E-03	8.573E-06	2.970E+00	
23	1.71E-01	7.820E-02	4.5E-06	1.04E-03	8.572E-06	2.937E+00	
24	1.78E-01	3.45E-02	1.5E-06	1.495E-03	8.577E-06	2.894E+00	
25	1.85E-01	1.161E-01	4.934E-06	1.695E-03	8.53E-06	2.819E+00	
26	1.92E-01	3.166E-02	1.6E-06	1.317E-03	8.653E-06	2.819E+00	
27	1.99E-01	3.519E-02	1.37E-06	1.435E-03	8.573E-06	2.787E+00	
28	2.06E-01	7.680E-02	3.0E-06	3.065E-03	8.573E-06	2.752E+00	
29	2.13E-01	1.347E-02	1.1E-06	5.297E-05	8.573E-06	2.676E+00	
30	2.21E-01	9.685E-02	1.4E-06	3.129E-03	8.573E-06	2.546E+00	
31	2.28E-01	2.008E-02	1.7E-06	1.069E-03	8.503E-06	2.453E+00	
32	2.35E-01	1.964E-02	9.0E-07	7.168E-04	8.503E-06	2.426E+00	
33	2.42E-01	4.250E-02	1.1E-06	1.573E-03	8.566E-06	2.411E+00	
34	2.49E-01	6.557E-02	1.1E-06	2.396E-03	8.675E-06	2.444E+00	
35	2.56E-01	8.278E-02	1.1E-06	3.888E-03	8.541E-06	2.434E+00	
36	2.63E-01	4.675E-02	1.4E-06	1.13E-03	8.573E-06	2.38E+00	
37	2.70E-01	4.45E-02	1.7E-06	1.581E-03	8.582E-06	2.391E+00	

Fig. 38. Typical tabular output from the neutron spectrum unfolding code NUTSPEC.

response functions, pulse-height data, unfolded neutron energy spectra, and dose-equivalent spectra.

Finally, since many of the spectral measurements are made near slightly moderated PuBe or fission sources, the spectral results are decomposed by linear regression into as many as three spectral components. Characteristic PuBe, fission, and 1/E slowing down spectra are calculated, and an attempt is made to fit these three components to the unfolded results. Amplitudes are given for the best linear superposition. They correspond to the integrated neutron intensities due to PuBe, fission, and 1/E spectral components. The plot file contains a graph in which the unfolded spectrum is compared to the computed superposition of three component intensities.

Conclusion. At this point, data reduction for spectra obtained with scintillators is fully operational. Proportional counter data may also be unfolded, but with some restrictions. First, there is no determination of the gamma-ray response as yet. This means that

spectra may be obtained only at energies greater than that deposited by electrons in the proportional counters (10-50 keV). The construction of a gamma-ray response function will soon remove this restriction. Second, the response matrix used to unfold the proportional counter data is somewhat crude, but will be improved as a result of recent measurements using monoenergetic neutrons.

The code is now available on the LLL octopus network, and may be used to unfold data obtained with all of the detector types mentioned. Thus far, spectral results have been in good agreement with other measurements and with expectation. Integrated intensities have been found to agree with long counter measurements in most cases. In a few cases the integrated intensities were less than that observed in the long counter, but these were cases where the low-energy contribution to the long counter count rate were not well known. A preliminary User's Manual has been written and is available. Full documentation will be available within a few months.

Technical Notes

RADIATION PROTECTION

Measurement of Neutron Spectrum at LLL Fenceline

Introduction. The Insulated Core Transformer (ICT) at LLL is used to produce 15-MeV neutrons with great abundance via the $T(d,n)^4He$ reaction. Although the neutron source is located below ground level, its close proximity to the LLL site boundary results in a measurable neutron intensity outside the exclusion area. To assess the accuracy of neutron dose rates measured at the site boundary, an effort has been made to determine the neutron energy spectrum there. The data reported below are incomplete, but summarize the results of this first measurement.

Measurements. During the month of June 1975, the ICT scheduled several long runs at maximum intensity. While these runs were in progress, the LLL portable neutron spectrometer³⁷ was set up in the bed of a pickup truck and the truck was parked against the fence at the point nearest the ICT neutron source. Total neutron intensity was found to be low for spectroscopy purposes, on the order of $0.3 \text{ n/cm}^2\text{-sec}$, but it was possible to carry out some measurements successfully. Neutron energy spectra were obtained using an NE213 liquid scintillator ($5.1 \times 11.4\text{-cm}$ diam) for $E_n > 1 \text{ MeV}$ and hydrogen proportional counters for $E_n < 1 \text{ MeV}$. Unfortunately, detector failure and inadequate sensitivity led to questionable results at low energies, and those data have been discarded. However, the scintillator data yielded useful neutron spectra even for the low intensities involved here. These will be reported.

The energy scale for the NE213 data was calibrated using a ^{22}Na gamma-ray source and the well-known relation between electron and proton scintillation output. Pulse-shape discrimination was used to suppress analysis of gamma-ray interactions during the neutron spectrum measurements. A pulse-height spectrum containing 3.6×10^5 recoil proton events was acquired during a 10 sec count. The resulting 256-channel pulse-height spectrum was unfolded using the LLL neutron spectrum unfolding code, NUTSPEC.³⁸ Figures 39 and 40 show the pulse-height spectrum and the unfolded neutron energy spectrum.

Results. Examination of Fig. 40 reveals that the neutron energy spectrum above 2 MeV resembles a $1/E$ spectral shape, i.e., $\phi(E) \sim 1/E$ or $E \times \phi(E) \sim \text{constant}$. This would be expected for a monoenergetic source viewed through several metres

Fig. 39. Pulse-height spectrum obtained with NE213 detector.

Fig. 40. Neutron energy spectrum at LLL fenceline; near ICT.

of moderating material. There is a noticeable rise above the $1/E$ spectrum near 3.5 MeV, but its magnitude exceeds the measurement errors by only a factor of two. Above 8 MeV, the low statistics result in increasingly large error bars which prevent useful

intensity determination. However, the low statistics rule out any abundant intensity due to "streaming." Setting an upper bound on the detector count rate due to 15-MeV neutrons indicates that the intensity of unscattered neutrons is less than $0.05 \text{ n/cm}^2\text{-sec}$ at the fence. The data below about 1.5-2.0 MeV must be viewed with some skepticism since the pulse-height spectrum in this region is very sensitive to minor variations in the adjustment of the pulse-shape discrimination. A future measurement using proportional counters should establish the spectral shape below 2 MeV.

Calibration of Building 212 Stack Monitor

Radioactive effluent releases from the LLL insulated core transformer (ICT) facility consist primarily of tritium gas. During normal operations, the gas release rate from the target (tritium-bound titanium substrate) into the accelerator vacuum system is 2.5 Ci/hr. The ion pumps trap the majority of this gas, and the remainder is pumped into the effluent stack where it

is diluted with the exhaust from the target pit. The maximum capacity of the three vacuum pumps is $7 \times 10^{-3} \text{ m}^3/\text{sec}$. This is small compared to the target vault exhaust of $0.275 \text{ m}^3/\text{sec}$ which is exhausted via a HEPA-filtered glove box.

The monitoring equipment at the ICT draws a sample from the effluent stack at a constant rate and then discharges it back into the stack. Figure 41 is a schematic diagram showing the monitoring and tritium source systems.³⁹ A source bottle prepared by tritium facility personnel⁴⁰ contained 2.84 Ci of tritium gas diluted with argon under a pressure of 0.344 MPa (usable pressure of 0.241 MPa). A flowmeter coupled to the tritium source bottle was calibrated to give the desired release time.

The tritium gas was introduced into a glove box at an average rate of 30.7 mCi/min for a total release time of 65 min. The tritium gas was continuously sampled from the stack via the stack monitors' ion chamber. This current was read out from a picoammeter at intervals of 1 min, and was also recorded on a permanent chart.

Fig. 41. Diagram of stack monitoring system.

Fig. 42. Response of picoammeter to 2 Ci of tritium released over a 65-min period.

Figure 42 shows the results of the release (total of 2 Ci of tritium gas in 65 min with a stack flow rate of $0.275 \text{ M}^3/\text{sec}$). The area under the curve was approximated numerically by using the trapezoidal approximation method, i.e.,

$$\text{Area} = \frac{1}{2}(y_0 + y_{j0})\Delta x + \sum_{n=1}^{69} y_n \Delta x$$

$$= 1.16 \times 10^{-9} \text{ A-min}$$

where

$$y_n = \text{current in A,}$$

$$\Delta x = 1 \text{ min.}$$

Therefore, for a release of 2 Ci in a period of 65 min, the stack calibration constant is $1.73 \times 10^9 \text{ Ci/A-min}$. It should be noted that this constant is good only for this flow rate: a variation in flow rate can result in an inaccurate estimate of tritium discharges. The largest contributors to a deviation in the air flow rate are the filters and the damper. To alleviate the flow rate problem, the damper on the hood will be locked in the open position and the hood door will be left with an opening of 140 mm. To monitor the conditions of the filters, a flow rate meter will be installed to monitor the air flow rate up the stack.

In the future, we will calibrate the system as a function of flow rate.

Measurement of the Responses of Multisphere Detectors with Monoenergetic Neutrons, 0.1 to 18.5 MeV

Introduction. In October 1975, experimenters from the Hazards Control Department and the Nuclear

Engineering Department of the University of California, Berkeley Campus conducted a series of detector calibrations using the LLL cyclograph accelerator. $^7\text{Li}(p,n)^7\text{Be}$, $^3\text{H}(p,n)^3\text{He}$, and $^3\text{H}(d,n)^4\text{He}$ reactions were used to produce neutrons with energies from a few keV to 18.9 MeV. Our portion of the work was devoted to measuring the response of two sets of spherical detectors used for multisphere spectrum measurements.⁴¹

In our method of using the multisphere spectrometer, responses of detectors in each of the spheres are provided as input to the LOUHI unfolding code.⁴² The code requires a set of energy-dependent response functions to complete the unfolding. All of our recent spectral measurements have been made with a set of seven polyethylene moderators (5.1-, 12.7-, 20.3-, 25.4-, 30.5- and 45.7-cm-diam) using a $1.27 \times 1.27 \text{ cm } ^6\text{LiI}$ scintillator. We use the calculated responses of Sanna,⁴³ adjusted for differences in polyethylene density for the spectrum unfolding. One objective of this experiment was to compare measured responses of our detector set with the calculated values. There was particular interest in the largest sphere because experimental results for a sphere that size have not been published.

A second set of spheres has been made which use ^{235}U track etch spark foil detectors. The sphere sizes are the same as the original set, but the four largest spheres are water-fillable shells of stainless steel. These spheres can be used either for pulsed field measurements or measurement of long-term, low-level environmental spectra. Unlike our primary detector set, there are no published responses for spherical moderators with uranium track etch detectors. We planned to measure the fast neutron responses directly, and to use these values to normalize published low-energy data for spheres using other detectors to obtain full-energy-range responses. The low-energy responses can be refined by future transport calculations using existing computer codes.

Experimental. The irradiations were carried out in a concrete room $12.2 \times 12.2 \times 12.5 \text{ m}$ high with a steel false floor 4.6 m above the room floor. The targets were placed on the end of the beam pipe which is 152 cm above the false floor. The targets were made at the Oak Ridge National Laboratory by plating the target material (LiF or $^3\text{H-Erbium}$) on 0.76-mm-thick discs of tungsten, 21 mm in diameter. The target material was made as thin as possible to reduce the neutron energy spread and still have adequate yield.

All of the measurements were made with the center of the detecting sphere $100 \pm 2 \text{ cm}$ from the target. The angle between the beam line and detector-target line was chosen based on the angular dependence of

Table 12. Cyclograph reaction summary

Reaction	Particle energy (MeV)	Target thickness	E-neutron energy (MeV)	η -angle with beam	γ -neutron yield relative to $\eta = 0$	$dI/d\theta$ (keV/deg)	$dY/d\theta$ (%/deg)
${}^7\text{Li}(p,n){}^7\text{Be}$	1.95	200 $\mu\text{g}\cdot\text{cm}^{-2}$	0.10	46°	0.79	2.3	1.3
	2.10	200 $\mu\text{g}\cdot\text{cm}^{-2}$	0.20	69°	1.68	3.2	0.4
	2.30	200 $\mu\text{g}\cdot\text{cm}^{-2}$	0.50	40°	0.63	4.5	2.0
${}^3\text{H}(p,n){}^3\text{He}$	3.5	2 $\text{mg}\cdot\text{cm}^{-2}$	1.0	100°	0.37	15	1.2
	3.5	2 $\text{mg}\cdot\text{cm}^{-2}$	2.0	51°	0.28	23	3.6
	6.5	10 $\text{mg}\cdot\text{cm}^{-2}$	3.0	81°	0.27	41	1.4
	6.5	10 $\text{mg}\cdot\text{cm}^{-2}$	5.0	35°	0.70	37	0.7
	6.5	10 $\text{mg}\cdot\text{cm}^{-2}$	13.0	128°	0.42	42	0.1
${}^3\text{H}(d,n){}^4\text{He}$	2.5	10 $\text{mg}\cdot\text{cm}^{-2}$	18.5	25°	0.85	32	1.0

the neutron energy. A summary of the reaction data for these irradiations is shown in Table 12.

We monitored the neutron intensity with a De Pangher precision long counter⁴⁴ placed with the counter face 200 $\pm$ 1 cm from the target. Again the angle with beam line was selected based on the angular dependence of the reaction used. Calibration of the long counter response was part of the irradiation program, and is discussed elsewhere in this report.⁴⁵ Corrections for the difference in angular flux distribution between the long counter and detector positions were made from published data⁴⁶ and computer calculations using simple kinematics. The long counter was used at a few different angles for each reaction and particle energy to check the validity of those distributions.

A low-energy component in the neutron field which resulted from room scatter could not be eliminated. To determine the contribution of that scatter to the long counter response, a polyethylene shadow shield was built⁴⁵ which could be placed between the target and long counter (Fig. 43). The count rate of the long counter with the shadow shield in place was taken to be the room scatter contribution. This contribution ranged from 2-3% for the ${}^7\text{Li}(p,n)$ irradiations at forward angles to 20% for the ${}^3\text{H}(p,n)$ irradiations with 6.5-MeV protons and the long counter at 80° to the beam line. We did not use the shadow shield to measure the scatter backgrounds for the spheres because it would have required doubling the number of measurements in an already full schedule. In addition, the spheres were closer to the source so that the relative level of the source neutrons was increased by a factor of four, suggesting a background level of a few percent. Although the 2.5-, 7.5- and 12.5-cm spheres were covered with a 0.05-cm cadmium sheet, we did see evidence of a significant background

contribution during irradiation with neutrons having energies above 3 MeV.

Additional monitoring was provided by Faraday cup measurements of the charge collected on the target from the beam. We monitored the ratio of the long counter to charge ratio as the irradiations progressed to assure ourselves that the beam did not wander off the target and, for higher energy reactions, that we were not getting lower energy neutrons from reactions with carbon or deuterium building up on the target. In previous experiments and the first part of this experiment, we experienced a great deal of difficulty with neutrons from the ${}^{12}\text{C}(d,n){}^{13}\text{N}$ reaction. Careful redesign of the target pipe and vacuum system was necessary to overcome the problem. An NE213 scintillator and an ${}^3\text{He}$ proportional counter were used before the ${}^3\text{H}(d,n)$ irradiations and ${}^3\text{H}(p,n)$ irradiations with 6.5 MeV protons to determine that

Fig. 43. DePangher precision long counter with shadow shield for room scatter determination.

there was not a group of low-energy contaminating neutrons from the target in excess of a few percent of the total emission.

The pulses from the ^6LiI detector used for one set of sphere measurements were processed with multichannel pulse-height analyzer located in the cyclograph control room. The analyzer, long counter and Faraday cup charge measurement were started a few seconds before a beam stop was removed to begin the irradiations. Immediately after irradiation with the beam stop ended, the data were recorded and the counters were reset. The analyzer was readout on a slow-speed printer.

Data reduction for the ^6LiI response measurements involved summation of the counts under the $^6\text{Li}(n,\alpha)^3\text{H}$ capture peak with compensation for gamma rays and other neutron interactions. To do this, we selected a channel on each side of the reaction peak which appeared to represent the extent of the peak. The channels between the two boundary channels were summed. The continuum was estimated by multiplying the log average of the boundary channels by the number of channels in the peak, and was then subtracted from the peak sum. The net peak area in counts divided by the neutron fluence was then taken to be the detector response. Experience with low-intensity irradiations has shown us that a log average is better approximation than a linear continuum. In a large majority of the cases, the continuum represents less than 10% of the peak area.

The ^{235}U track detector used for the second set of spheres were assembled by placing an

8×10^{-3} -mm-thick film of polycarbonate film between two 0.025-mm-thick foils of 93% enriched ^{235}U . Each ^{235}U was separated from the polycarbonate surface by a 1-mm-thick plastic ring. One of the plastic rings was used with a concentric stainless steel ring to hold the polycarbonate film. Each detector was assembled and placed in the sphere just before and disassembled immediately after irradiation to prevent detection of low levels of neutrons from the high-flux ICT accelerator which is housed in the same building. The polycarbonate film were etched for 1-1/2 hr in 6 N NaOH at 60°C. Counting was done by the now familiar spark technique.⁴⁷ The responses of these detectors are then equal to the number of spark counts per foil divided by the neutron fluence. The foil area is 2.85 cm², and remembering that two uranium foils are used, the detector records fissions from 5.7 cm² of ^{235}U .

Results. The responses for both sets of spherical moderators are shown in Tables 13 and 14. For comparison, we have included the ratios of the measured values to those obtained from Sanna's calculations.⁴³ The original calculations were published as responses in an energy group structure. Since that makes direct comparison difficult, we took the liberty of estimating the equivalent values at specific energies by assigning a single log average energy to each band and using the energy values that we interpolated to arrive at an estimate for the response for each calibration energy. Although some data were taken at energies below 0.10 MeV, we do not have enough confidence to report them here.

Table 13. Measured responses of polyethylene multisphere moderators with 1.27×1.27 cm ^6LiI scintillator

Sphere diam (cm)	Counts/N-cm ⁻²						
	5.1	7.6	12.7	20.3	25.4	30.5	45.7
Polyethylene density (g-cm ⁻³)	0.912	0.948	0.930	0.961	0.924	0.918	0.925
Neutron energy (MeV)	Response						
0.10	0.48(1.10)	1.17(0.92)	1.55(0.92)	0.76(1.11)	0.48(1.06)	0.168(1.24)	0.051(0.98)
0.20	0.30(0.81)	0.90(0.80)	1.52(0.84)	0.77(1.02)	0.49(1.09)	0.159(0.84)	0.0092(1.16)
0.50	0.185(0.78)	0.77(0.79)	1.63(0.86)	1.17(0.95)	0.83(1.11)	0.38(1.03)	0.026(1.11)
1.0	- ^b	- ^b	1.64(0.92)	1.58(0.96)	1.26(1.05)	0.72(1.02)	0.078(0.87)
2.0	0.123(1.54)	0.47(1.11)	1.43(0.97)	1.98(1.09)	1.77(1.15)	1.23(1.12)	0.24(0.96)
3.0	0.139(2.79)	0.48(1.64)	1.19(1.02)	1.56(0.89)	1.44(0.89)	1.06(0.83)	0.28(0.67)
5.0	0.064(2.13)	0.22(1.32)	0.81(1.17)	1.27(0.82)	1.25(0.80)	1.13(0.80)	0.56(0.83)
13.0	0.092(10.3)	0.27(4.5)	0.76(2.16)	1.06(1.28)	1.11(1.12)	1.06(1.04)	0.92(1.16)
18.5	0.058(10.1)	0.186(4.9)	0.54(2.47)	0.79(1.46)	0.79(1.23)	0.77(1.13)	0.79(1.41)

^a Values in parentheses are the ratio of measured responses to interpolated responses based on the calculations of Sanna.⁴³

^b Equipment malfunction prevented acquiring data for these responses.

Table 14. Measured responses of multisphere moderators with ^{235}U track etch detectors

Sphere diam (cm)	Spark counts/ $\text{N}\cdot\text{cm}^{-2} \times 10^3$						
	5.1	7.6	12.7	20.3	25.4	30.5	45.7
Moderator material	CH_2^a	CH_2^a	CH_2^a	H_2O	H_2O	H_2O	H_2O
Neutron energy (MeV)	Response						
0.10	3.5	11	17	13	6.4	3.4	0.30
0.20	2.1	7.9	16	13	7.9	4.7	- ^b
0.50	1.6	6.2	17	14	10	6.5	0.77
1.0	0.80	4.4	16	15	13	8.3	1.6
2.0	0.79	3.8	13	15	17	13	4.2
3.0	1.1	4.0	9.7	15	15	13	5.1
5.0	0.51	1.7	7.0	10	10	11	7.0
12.0	0.79	2.1	5.4	7.8	7.8	9.2	7.6
18.5	0.40	1.7	4.8	6.5	6.9	6.7	5.7

^aSphere covered with 0.51-mm Cd shell

^bData not taken.

An estimate of the errors in responses should include the following contributions: measurement of neutron fluence by the long counter (including location error) - 10%, uncertainty in sphere-to-target distance - 4%, and error in flux distribution from long counter to detector - variable, but average about 5%. The errors in counting statistics for spark counting are variable, but average about 7%. The inherent reproducibility of the spark counter is about 10%. Counting statistical errors with the ^6LiI scintillator are less than 1% in most cases. The error in background contribution estimate is about 3%. Assuming that the overall error in the measurement is the square root of the sum of squares of these errors, the estimated error in measurements using the ^6LiI detector is about 12%. The equivalent error in responses for the ^{235}U track etch detectors is 17%. Errors in neutron energy due to the solid angle subtended by the detector have not been included, but can be estimated from the detector diameter and the data in Table 12.

Conclusions. One objective of the experiment was to determine the validity of the calculated responses for a 1.27×1.27 -cm detector used in our multisphere spectrometer system. Although there are some points where measurements with the four largest spheres differ by more than the expected error from calculated values, we feel the calculations are valid and will continue to use them. The responses obtained from the 6.5-mm protons on ^3H (3- and 5-MeV

neutrons) appear to be too low, and the data require further analysis.

As for the three small spheres at energies below 2 MeV, the responses seem to be lower than calculated values by about 15-20%. This is not unexpected, however, because these spheres have detector holes which could not be simulated by the one-dimensional discrete ordinates code. It does appear that we should consider reducing our small-sphere responses to account for this. At values above 2 MeV, the measurements obviously suffered from room scatter, but the problem occurred at energies where the detectors have reduced sensitivity to the primary neutrons, and accurate data are not essential to spectrum unfolding problems.

The second objective of the experiment was to obtain responses for multispheres with ^{235}U track etch detectors from 0.1 to 18.5 MeV, and we have done that. At points where there are obvious problems with the responses such as the 3- and 5-MeV data points, we can use responses for the ^6LiI detector for normalization. In the same way, we can now obtain low-energy responses by normalizing these results to calculated data for the ^6LiI detector.

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ENVIRONMENTAL PROTECTION

Gloved Box Expansion and Pressure Test

In a recent experiment in Building 241, an inert hard vacuum gloved box was used to perform experiments with unstable hydrides. Theoretically, if the hydrides were to disassociate, a large volume of gas under pressure would expand into the gloved box. A 2-psf pressure relief valve was deemed necessary for safe relief of the entire system. The relief valve would be adequate if all the gloved ports covered with the gloves inside the box were properly secured. However, during the times when the gloves were in use, the weak link in the system would be the bursting pressure of the gloves.

A quick experiment was set up to determine the bursting pressure and the approximate free volume of the two standard types of gloves normally used at LLL. Glove A is fabricated from 30-mil neoprene, and Glove B is fabricated from 30-mil butyl rubber with a duralac coating. Both are manufactured by Charleston Rubber Co.

Each glove was attached to a standard 8-in. gloved port and was secured with an aerosol clamp. An aluminum plate was attached to the gloved port with the proper hose-barb connection.

Both gloves were quickly filled with air to simulate a sudden expansion. A manometer was used to measure the resultant pressure on the glove. Both gloves reached an equilibrium pressure within a few seconds. However, the free volume obtained by the butyl glove was approximately twice that of the neoprene glove before bursting.

The results of the tests are given below:

Glove	Bursting pressure	Approximate volume
Neoprene	1.1 psi	0.5 m ³
Butyl	0.5 psi	0.2 m ³

In conclusion, the bursting pressure of the gloves should be considered in designing a pressure relief valve for the system.

The following photographs show a sequence of the glove expansion. The black glove is the neoprene glove (Fig. 44).

Fig. 44(A). Neoprene glove expansion test.

Fig. 44(B). Butyl rubber glove expansion test.

Publications

B. N. Odell, *Supporting Documents for NTS 5100 and 5360 Safety Analysis Reports*, Lawrence Livermore Laboratory, Rept. UCRL-51963 (1975).

The following appendices are common to the LLL Safety Analysis Reports Livermore Site and are included here as supporting documents to those reports:

- Appendix A Description, Population Distribution, and Land Use
- Appendix B A Geological and Seismological Investigation of the Lawrence Livermore Laboratory Site
- Appendix C Meteorology
- Appendix D Tornado Criteria for the LLL Site (To be supplied)
- Appendix E Characteristics of California Tornados
- Appendix F Surface Water Hydrology
- Appendix G Ground Water Hydrology
- Appendix H Lawrence Livermore Laboratory Quality Assurance Program
- Appendix I Environmental Monitoring at the Lawrence Livermore Laboratory 1974 Annual Report

C. L. Lindeken, J. H. White, A. J. Toy, and C. W. Sundbeck, *Ambient Environmental Radiation Monitoring at the Lawrence Livermore Laboratory*, Lawrence Livermore Laboratory, Rept. UCRL-77106 (1975). (Presented at the 9th Midyear Topical Symposium on Operational Health Physics, Denver, Colo., Feb. 9-12, 1976.)

Thermoluminescence dosimetry is the principal means of measuring ambient gamma radiation at the Lawrence Livermore Laboratory. These dosimeters are used at 12 perimeter locations and 41 locations in the off-site vicinity of the Laboratory, and are exchanged quarterly. Control dosimeters are stored in a 75-mm-thick lead shield located out-of-doors to duplicate temperature cycling of field dosimeters. Effect of dosimeter response to radiation in the shield is determined each quarter. Calibration irradiations are made midway through the exposure cycle to compensate for signal fading. Terrestrial exposure rates calculated from the activities of naturally occurring uranium, thorium, and potassium in Livermore Valley soils vary from 3 to 7 $\mu\text{R/hr}$. Local inferred exposure rates from cosmic radiation are approximately 4 $\mu\text{R/hr}$. TLD measurements are in good agreement with these data. Off-site and site perimeter data are compared, and differences related to Laboratory operations are discussed.

Dale E. Hankins, *The Effect of Energy Dependence on the Evaluations of Albedo Neutron Dosimeters*, Lawrence Livermore Laboratory, Rept. UCRL-77198 (1975). (Presented at the 9th Midyear Topical Symposium on Operational Health Physics, Denver, Colo., Feb. 9-12, 1976.)

Albedo neutron dosimeters have proven to be effective as a method of measuring the dose from low-energy neutrons that other types of personnel neutron dosimeters do not detect. Many albedo neutron dosimeters have been designed, and some are being used extensively. All albedo neutron dosimeters are energy-dependent, becoming less sensitive as the energy of the neutrons is increased. In this paper we discuss the energy dependence and the effect it has on the evaluation of the dose. Numerous measurements are shown which were made under field conditions at five reactors, two accelerators, a radiochemistry facility, two plutonium facilities, a low-scatter neutron irradiation facility, three storage vaults, and at the N.B.S. low-energy neutron beams. These measurements show the limits imposed by the energy dependence on the evaluation of the dose determined by albedo neutron dosimeters. In most cases evaluation of an individual's dose occurring in a single facility can be reasonably accurate, but it must be known in which facility the person was exposed. We discuss procedures using the incident thermal neutron response of TLDs located on top of the dosimeter to correct for the energy dependence; and we evaluate errors caused by wearing a dosimeter improperly.

T. A. Gibson and D. N. Montan, *Simplified Scheme for Radioactive Plume Calculations*, Lawrence Livermore Laboratory, Rept. UCRL-77092 (1975). (Presented at the 9th Midyear Topical Symposium on Operational Health Physics, Denver, Colo., Feb. 9-12, 1976.)

A simplified mathematical scheme to estimate external whole body gamma radiation exposure rates from gaseous radioactive plumes was developed for the Rio Blanco Gas Field Nuclear Stimulation Experiments. The method enables one to calculate swiftly, in the field, downwind exposure rates knowing the meteorological conditions and gamma radiation exposure rates measured by detectors positioned near the plume source. The method is straightforward and easy to use under field conditions without the help of mini-computers. It is applicable to a wide range of radioactive plume situations. It should be noted that the Rio Blanco experiment was detonated on May 17,

1973, and no seep or release of radioactive material occurred.

Gary O. Nelson and Charles P. Richards, *Enhanced Filtration, July-September Progress Report*, Lawrence Livermore Laboratory, Rept. UCID-16949-75-3 (Oct. 1975).

The service life of high efficiency particulate air (HEPA) filters can be extended if 1) particles can be agglomerated prior to removal, or 2) if prefilters can be induced to behave more efficiently toward the smaller particles. Two electronic techniques are being devised to explore these possibilities.

Gary O. Nelson, A. Nicholas Correia, and Charles A. Harder, *Respirator Cartridge Efficiency Studies: VII Effect of Relative Humidity and Temperature*, Lawrence Livermore Laboratory, Rept. UCRL-77390 (Oct. 1975).

We have determined the effect of relative humidity on organic vapor respirator cartridges which contain activated carbon as the adsorbing media. Cartridges were tested with 1000 ppm at 53.3 liters/min with seven solvent vapors. Cartridges preconditioned at 50% humidity and tested at 65, 80 and 90% humidity showed a service life reduction of 1, 12 and 17% respectively when compared to those tested at 50% humidity. Those preconditioned at 90% exhibited a 52% reduction in effective service life when compared to those cartridges preconditioned and tested at 50% humidity. Temperature is not a dominant factor when estimating service life. Calculations indicate that the breakthrough times are diminished from 1-10% with each 10°C rise in temperature. (Submitted to the American Industrial Hygiene Association Journal.)

Gary O. Nelson, Robert D. Taylor, and Harry B. Johnson, *A System to Produce Low Concentrations of Solvent Vapor in Air*, Lawrence Livermore Laboratory, Rept. UCID-16918 (Oct. 1975).

We have designed a system to continuously produce known concentrations of either gas or solvent vapor in a moving air stream. It is a positive displacement device with a dynamic concentration range of approximately 1 to 10,000 ppm at flowrates varying from 1 to 16 liters/min. The system is capable of handling any gas which is compatible with glass and Teflon, and any vapor which is compatible with aluminum and stainless steel.

Gary O. Nelson, Charles A. Harder, and Bruce E. Bigler, *Respirator Cartridge Efficiency Studies: VI. Effect of Concentration*, Lawrence Livermore Laboratory, Rept. UCRL-76184 (1975).

The effective service life of organic vapor cartridges has been determined for nine solvent vapors and one gas at concentrations ranging between 50 and 3000 ppm. Test procedures typically include passing the vapor and air mixtures through a pair of respirator cartridges containing activated carbon at 53.3 liters per minute, 50% relative humidity, and at a temperature of 20 to 25°C. All results showed that the time to any chosen breakthrough percent (t_b) conformed to the empirical expression $t_b = aC^b$, where C is the challenge concentration in ppm and a and b are constants for a given set of experimental conditions. The average solvent vapor values for b at 10% breakthrough was determined to be -0.67 ± 0.17 . This value is not appreciably affected by flowrate; however, it is affected by humidity, especially at relative humidities above 50%. The values for a, however, vary widely depending on the conditions encountered. Additional tests also showed the service life to be inversely proportional to flow rate for acetone (100 to 2000 ppm) and benzene (250 to 2000 ppm). (Submitted to American Industrial Hygiene Association Journal.)

R. G. Purington, *Grass Fire at Site 300 on May 31, 1975*, Lawrence Livermore Laboratory, Rept. MISC-2101 (1975).

This report contains the basic facts on the grass fire at Site 300 on 31 May 1975. The fire involved about 380 hectares (ha) (950 acres) of grassland. There were two injuries, one minor and one lost time; fire loss was zero.

D. S. Myers, W. J. Silver, D. G. Coles, K. C. Lamson, D. R. McIntyre, and B. Mendoza, *Evaluation of the Use of Sludge Containing Plutonium as a Soil Conditioner for Food Crops*, Lawrence Livermore Laboratory, Rept. UCRL-77318 (1975). (Presented at the International Symposium on Transuranium Nuclides in the Environment, Washington D.C., November 17-21, 1975.)

During the three-week period from 25 May to 16 June 1967, approximately 32 mCi of ^{239}Pu was inadvertently released from Lawrence Livermore Laboratory into the sanitary sewer system. It was concluded at the time from an investigation conducted by LLL in cooperation with ERDA-SAN and the California Department of Public Health that the release did not present a hazard to treatment plant personnel or the community.

Helen Shaver, *Health and Safety Education Course Catalog*, Lawrence Livermore Laboratory, Rept. MISC-2095 (1975).

A health and safety education program shall be provided to disseminate information on preventing injury to employees. Hazards Control Department has developed a number of Health and Safety courses of interest to a broad spectrum of Laboratory personnel. Some are required to qualify to perform certain tasks or jobs, while others are of general interest to various groups at the Laboratory. All are Laboratory-wide in nature and essentially ready for presentation.

Alan L. McGibbon, *A 1024-Channel Portable Gamma-Ray Spectrometer*, Lawrence Livermore Laboratory, Rept. UCRL-77498 (1975). (Presented at the Measurement Science Conference, California Polytechnic University, San Luis Obispo, California, December 5, 6, 1975.)

An instrument is described which is designed to determine radioactive isotope spectra in the field under adverse environmental conditions. The instrument is battery-powered, stores and compares multiple spectra, and performs computations upon the resulting displayed graph.

A. J. Toy and C. L. Lindeken, *The Implications of Sampling from a Log-Normal Population*, Lawrence Livermore Laboratory, Rept. UCRL-76936 (1975). (Presented at the 3rd ERDA Environmental Protection Conference Chicago, Ill., Sept. 23-26, 1975.)

The environmental radioactivity surrounding the Lawrence Livermore Laboratory is typically log-normally distributed. The geometric standard deviation of radioactivity in all types of samples — air, water, soil, sewage, and vegetation — is about two. Hypothesis testing was done using the mean geometric standard deviation of these samples. Parametric studies demonstrated the effect of the number of background samples, the magnitude of tolerable errors, the variability of the sample population, and the precision of the analysis on the detectability of differences from natural and global fallout radioactivity.

Robert E. Strout II, *Use of the LLL Fire Data-Base*, Lawrence Livermore Laboratory, Rept. UCID-16916 (1975).

Through the use of the "MASTER CONTROL" program, we have had created a data-base containing references to literature concerning different aspects of Fire Technology.

The Data-base allows a researcher to use the MASTER CONTROL program to retrieve references concerning only those parts of Fire Technology that he is interested in. MASTER CONTROL also allows the researcher to add references to the data-base for

retrieval at a later time. However, it does not give the actual literature, just a reference to it.

References were collected from Engineering Index/The Journal of Fire & Flammability.

Robert E. Henry, *Evaluation of Floor Coverings for Explosives Work Areas*, Lawrence Livermore Laboratory, Rept. UCID-16887 (1975).

In an attempt to find an acceptable and inexpensive substitute for the currently employed sprayable polyurethane floor covering, Urapol (used in explosives work areas), two widely available vinyl floor coverings were evaluated. The floor coverings tested were 'Brigantine' and "Quiet Zone." Resilience and surface-friction characteristics were tested by impacting samples with hemispherical, 21-kg (46-lb) billets of plastic-bonded explosive which were dropped from various heights. Test results indicated that either Brigantine or Quiet Zone is an acceptable substitute for Urapol.

Thomas R. Crites, *An Experimental Study of the Spectral Dependence of an Albedo-Type Neutron Dosimeter*, Lawrence Livermore Laboratory, Rept. UCRL-77028 (1975). (Submitted to Health Physics Journal.)

A simple albedo-design badge was exposed to various neutron spectra available at the Lawrence Livermore Laboratory (LLL) calibration facility. Results of thermoluminescent dosimeters within the badge were compared in an attempt to develop a method of determining the neutron dose equivalence independent of the incident neutron spectra.

Yen Mee Kwok, *Evaluation of the Spectrophotometric Determination of Traces of Selenium with 2,3-Diaminonaphthalene*, Lawrence Livermore Laboratory, Rept. UCRL-77017 (1975).

The selenium content in atmospheric particulates is presently measured by a spectrophotometric procedure that requires removing foreign elements from the reaction solution. Although this technique is applicable to the routine analysis of selenium in sewage water, the results lack both reproducibility and accuracy. We therefore evaluated the technique in detail and suggest modifications that increase the accuracy and reproducibility of the results. (Submitted to Analytical Chemistry.)

Gary O. Nelson, *Safety Limits of Half-Mask Cartridge Respirators for Organic Solvent Vapors*, Engineering and Technology Review, Lawrence Livermore Laboratory, Rept. UCRL-52000-75-10 (Oct. 1975).

Recent studies of the effective service life for half-mask respirators have shown these devices to be unsuitable for certain organic vapors. Methanol, methylamine, vinyl chloride, and dichloromethane vapors all rapidly penetrate cartridges containing activated carbon. The experimentally determined service life for many vapors is shorter than predicted by adsorption theory.

David R. McIntyre and J. W. Silver, *Trace Element Uptake by Field-Grown Food Plants Fertilized With Waste-Water Sewage Sludge*, Lawrence Livermore Laboratory, UCRL-76820.

Garden vegetables and grains were field-grown in San Ysidro loam amended with municipal wastewater sewage sludge at 235 Mt/ha (84 dry tons/acre). Crops were grown without additional fertilizer in a sprinkler-irrigated garden plot maintained at moisture levels near field capacity. Trace elements from the sludge were utilized by the plants and transferred to edible portions of the root, stem, leaf, flower and fruit. Crops grown in adjacent soils provided the control data. Both soil and plants were analyzed for 35 elements by neutron activation and atomic absorption spectrometry. Plants grown in soils enriched with sludge contained elevated levels of Cd, Zn, Pb, Cu, Mg, Ni, K and Ag. Cd and Zn were elevated in tissues of lettuce, corn, radish and turnip leaves, radish bulbs, and wheat seeds. There was an indication Cd had a more effective uptake by the plants at lower Cd soil concentrations. In general concentrations of metals were higher in leaf tissues than in roots. Levels of zinc in radish leaves were higher (290 ppm) than in radish roots (145 ppm). (Presented at the 67th Annual Meeting of The American Society of Agronomy, Knoxville, Tenn., Aug. 24-29, 1975.)

Jerry A. Roby, *Sabotage*, Lawrence Livermore Laboratory, Rept. UCRL-76914.

The history and various aspects of sabotage are described and discussed. (Submitted to Fire Chief Magazine.)

B. N. Odell, *Site 300 Blast Forecasting*, Lawrence Livermore Laboratory, Rept. M-064 (1971).

Blast forecasting at Site 300 is performed at the weather station, and consists of four operations. These operations are:

- Balloon tracking (radar or pibal)
- Temperature plotting
- Computer operation (G-machine, PDP-11, 7600)
- Determining H.E. weight limit.

W. L. Eneidi and R. D. Taylor, *Air-Purifying Powdered Respirator Pack Utilizing a Miniature Two-Stage Air*

Mover, Lawrence Livermore Laboratory, Rept. UCRL-75183 (1975). (Submitted to American Industrial Hygiene Journal.)

Design and construction of a moderate-weight, battery-powered, positive-pressure respirator pack is described. Design detail of a two-stage air mover is given, along with its adaptation for use as a portable air sampler. (Submitted to Health Physics Journal.)

R. G. Purington, *Hose Streams and Nozzles*, Lawrence Livermore Laboratory, Rept. UCRL-76640 (1975).

The main fire-fighting objective of a fire department is to place water on a fire. The application of water is usually accomplished through the use of pumps, hose lines and nozzles. For effective applications both pump operator and nozzleman, as well as fire officers, must have a working knowledge of the hydraulics of a hose-line system. This task involves understanding of flows through various types, diameters and lengths of hose, and the discharge of water through nozzles. In addition, a knowledge of losses due to elevation changes and friction is needed. Finally, a thorough knowledge of the operation and capability of pumping apparatus and water supply system is required. (To become a chapter in the National Fire Protection Association Handbook.)

Byron N. Odell and Arthur J. Toy Jr., *Safety Analysis Report for Building 231 Central Vault*, Lawrence Livermore Laboratory, Rept. UCRL-51815 (1975).

This Safety Analysis Report (SAR) for Building 231 Central Vault provides a comprehensive review of the facility and the operations conducted in it. All credible accidents that may have off-site consequences have been treated in this report. Accidents having no off-site consequence are generally not discussed in this document since they are covered by other documents. The intent of this SAR is to provide the LLL Plant Manager with a written objective analysis of the facility as well as furnishing him with recommendations for corrective action in any area that appears deficient. This report has been prepared for the LLL Plant Manager to meet a specific request from the ERDA SAN office.

Building 231 Central Vault is used for storing plutonium, uranium, and other radioactive isotopes. The approximate cost to build the building (not including equipment) was over \$500 000. The approximate cost of the equipment presently in the building is over \$100 000.

Byron N. Odell and Arthur J. Toy Jr., *Safety Analysis Report for Building 331*, Lawrence Livermore Laboratory, Rept. UCRL-51816 (1975).

This Safety Analysis Report (SAR) for Building 331 (the Gaseous Chemistry Facility) provides a comprehensive review of the facility and the operations conducted in it. All credible accidents that may have off-site consequences have been treated in this report. Accidents having no off-site consequence are generally not discussed in this document since they are covered by other documents. The intent of this SAR is to provide the LLL Plant Manager with a written objective analysis of the facility as well as furnishing

him with recommendations for corrective action in any area that appears deficient. This report has been prepared for the LLL Plant Manager and to meet a specific request from the ERDA SAN office.

Building 331 is used primarily for experimental work with radioactive gases and their compounds. The approximate cost to build the building (not including equipment) was \$1.2 million. The approximate cost of the equipment presently in the building is \$1.1 million.

References

1. M. Sohrabi, "The Amplification of Recoil Particle Tracks in Polymers and its Application in Fast Neutron Personnel Dosimetry," *Health Phys.* **27**, 598 (1974).
2. L. Tommasino, "Electrochemical Etching of Damaged Track Detectors by H. V. Pulse and Sinusoidal Waveforms," *Comitato Nazionale Energia Nucleare*, Rept. RT/PROT(71) (1970).
3. D. Hankins, *A Small, Inexpensive Albedo-Neutron Dosimeter*, Los Alamos Scientific Laboratory, Rept. LA-5261 (1973).
4. *Hazards Control Progress Report No. 50*, Lawrence Livermore Laboratory, Rept. UCRL-50007-75-1 (1975), p. 5.
5. Milton Hildebrand, *Anatomical Preparation* (University of California Press, 1968).
6. International Commission on Radiation Protection, Publication 23, *Report of the Task Group on Reference Man* (Pergamon Press, Oxford, 1975).
7. Donald R. White, St. Bartholomew's Hospital, London, England, private communication.
8. Donald Newton, Environmental and Medical Sciences Division, AERE, Harwell, Oxfordshire, England, private communication.
9. D.E.A. Jones and H.C. Raines, *Brit. J. Radiology* **22**, 549 (1949).
10. *Hazards Control Progress Rept. No. 50*, Lawrence Livermore Laboratory, Rept. UCRL-50007-75-1 (1975), p. 9.
11. G.O. Nelson, A.N. Correia, C.A. Harder, *Respirator Cartridge Efficiency Studies: VII. Effect of Relative Humidity and Temperature*, Lawrence Livermore Laboratory, Rept. UCRL-77390 (1975).
12. J. De Pangher and L.L. Nichols, *A Precision Long Counter for Measuring Fast Neutron Flux Density*, Battelle Northwest Laboratory, Rept. BNWL-260 (1966).
13. I.M.G. Thompson and A. Lavender, "Calibration of the de Pangher Long Counter," *Symp. Neutron Monitoring for Radiation Protection Purposes, Vienna 11-15 Dec 1972*, IAEA-SM-167-42.
14. *National Bureau of Standards Handbook 72*, p. 32.
15. International Commission on Radiation Units and Measurements Publication 20, *Radiation Protection Instrumentation and its Application* (Complete bibliography).
16. M. Heinzlmann and F. Rohloff, *Nucl. Instr. Meth.* **109**, 205 (1973).
17. *Hazards Control Progress Report No. 48*, Lawrence Livermore Laboratory, Rept. UCRL-50007-74-1 (1974), p. 19.
18. H. Henry, *Fundamentals of Radiation Protection* (Wiley-Interscience, New York, 1969).
19. D. Gross, J. Loftus, and A.F. Robertson, "A Method of Measuring Smoke from Burning Materials," STP 422 in *Fire Test Methods: Restraint and Smoke*, (American Society for Testing and Materials, Philadelphia, 1967), p. 166.
20. *Hazards Control Progress Report No. 25*, Lawrence Livermore Laboratory, Rept. UCRL-50007-67-1 (1967), p. 41.
21. J. R. Gaskill and C.R. Veith, "Smoke Opacity from Certain Woods and Plastics," *Fire Technol.* **4**, 185 (1968).
22. J.R. Gaskill, "Smoke Development in Polymers During Pyrolysis or Combustion," *J. Fire and Flammability*, **1**, 183 (1970).
23. J.R. Gaskill, "Smoke Hazards and Their Measurement," *J. Fire and Flammability*, **4**, 1/9 (1973).
24. *Hazards Control Progress Report No. 50*, Lawrence Livermore Laboratory, Rept. UCRL-50007-75-1 (1975), p. 20.
25. W.P. Chien and J.D. Seader, "Smoke Development at Different Energy Flux Levels in an NBS Smoke Density Chamber," *Fire Technol.* **10**, 187 (1974).
26. "Methods of Test for Low Penetration Air Filters," *British Standard 3928* (1965 Revision), (British Standard Institute, 1965).
27. *Sodium Flame Test Regulation*, Development Office, Volks Ltd., Guildford, England, Rept. 293 (1965).
28. G.O. Nelson and C.P. Richards, *Enhanced Filtration, July-September 1975 Progress Report*, Lawrence Livermore Laboratory, Rept. UCRL-16949-75-3 (1975).
29. *Hazards Control Progress Report No. 46*, Lawrence Livermore Laboratory, Rept. UCRL-50007-73-2 (1973), p. 11.
30. *Hazards Control Progress Report No. 47*, Lawrence Livermore Laboratory, Rept. UCRL-50007-73-2 (1973), p. 14.

31. *Hazards Control Progress Report No. 48*, Lawrence Livermore Laboratory, Rept. UCRL-50007-74-1 (1974), p. 20.
32. *Hazards Control Progress Report No. 49*, Lawrence Livermore Laboratory, Rept. UCRL-50007-74-2 (1974), p. 13.
33. *Hazards Control Progress Report No. 50*, Lawrence Livermore Laboratory, Rept. UCRL-50007-75-1 (1975), p. 10.
34. *Hazards Control Progress Report No. 50*, Lawrence Livermore Laboratory, Rept. UCRL-50007-75-1 (1975), p. 12.
35. *Hazards Control Progress Report No. 50*, Lawrence Livermore Laboratory, Rept. UCRL-50007-75-1 (1975), p. 36.
36. H.W. Brock and C.E. Anderson, *Rev. Sci. Instr.* **31**, 1063 (1960).
37. *Hazards Control Progress Report No. 50*, Lawrence Livermore Laboratory, Rept. UCRL-50007-75-1 (1975), p. 12.
38. While the code is not yet fully documented, the best currently available descriptions may be found in the following two documents: *Hazards Control Progress Report No. 48*, UCRL-50007-74-1 (1974), p. 20. *Hazards Control Progress Report No. 47*, UCRL-50007-73-3 (1973), p. 14.
39. *Hazards Control Progress Report No. 40*, Lawrence Livermore Laboratory, Rept. UCRL-50007-71-2 (1971), p. 11.
40. G. Morris and L. Christensen, private communication.
41. R.L. Bramblett, R.J. Ewing and T.W. Bonner, "A New Type of Neutron Spectrometer," *Nucl. Instr. Methods* **9**, 1 (1960).
42. J.T. Routti, "Mathematical Considerations of Determining Neutron Spectra from Activation Measurements," *Proc. Second Intern. Symp. on Accelerator Dosimetry and Experience*, Stanford University, November 5-7, 1969, paper CONF 691101.
43. R.S. Sanna, *Thirty-One Group Response Matrices for the Multisphere Neutron Spectrometer Over the Range Thermal to 400 MeV*, USAEC Health and Safety Laboratory, Rept. HASL-267 (1973).
44. J. de Pangher, *A Reproducible Precision Polyethylene Long Counter for Measuring Fast Neutron Flux*, Hanford Laboratories, Richland, WA., Rept. HW-SA-2140 (1961).
45. D.R. Slaughter and D.W. Rueppel, "Calibration of a de Pangher Precision Long Counter," this Rept.
46. H. Liskien and A. Pruksen, "Neutron Production Cross Sections," *Nuclear Data Tables*, **11**, No. 7, 1973, pp. 580-615.
47. W.G. Cross and L. Tommasino, "Electrical Detection of Fission Fragment Tracks for Fast Neutron Dosimetry," *Health Phys.* **15**, 196 (1968).