Beta Measurements at Department of Energy Facilities

**L. A. Rathbun
K. L. Swinth
D. L. Haggard**

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FOREWORD

This report presents measured beta dose rates and spectra at U.S. Department of Energy (DOE) facilities. This and the companion report, "Characteristics of Beta Detection and Dose Measurement at Department of Energy Facilities" (Mulvehill et al. 1987), present information to characterize the beta fields and protection practices at DOE facilities.

Beta measurements are technically difficult, and innovative efforts must be expended to improve their accuracy. The most important problem is that the beta dosimetry and instrumentation used are highly dependent on energy, angular incidence, and source geometry. Many health protection specialists believe that beta exposures are adequately controlled because beta-to-photon ratios are assumed to be low. This assumption is not always valid as the data in this report indicate. Significant beta doses exist where personnel are exposed to mixed fission products; for example, chemical reprocessing plants, reactor accidents, or where uranium metals or powders are processed. Consequently, accurate measurements of beta fields are important to effect proper control of beta exposures to workers.

This report is part of an effort to increase the DOE awareness of this technically difficult area of radiation protection. Findings and recommendations for improvements are noted. A cooperative effort among the various DOE contractors is needed to improve upon the present situation, and your comments and suggestions will be welcomed.

E. J. Vallario
Acting Director
Radiological Controls Division
Office of Nuclear Safety
U.S. Department of Energy

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SUMMARY

BACKGROUND

As part of a program to upgrade beta measurements at U.S. Department of Energy (DOE) facilities, Pacific Northwest Laboratory (PNL) performed a two-step process to characterize the current beta measurement practices at DOE facilities. First, PNL issued a survey questionnaire on beta measurement practices to DOE facilities, the results of which have been documented.^(a) Second, PNL measured beta doses and spectra at seven selected DOE facilities and compared selected measurement techniques in the facility environment. This report documents the results of the radiation field measurements and the comparison of measurement techniques at the seven facilities.

Data collected by PNL included beta dose and spectral measurements at seven DOE facilities that had high beta-to-gamma ratios. The measurement systems used were a silicon surface barrier spectrometer, a plastic scintillator spectrometer, and a multielement beta dosimeter. Other dosimeters and survey meters representative of those used at DOE facilities or under development were also used for comparison.

Field spectra were obtained under two distinct conditions. In the first case, silicon- and scintillation-based spectrometer systems were used under laboratory conditions where high beta-to-gamma dose ratios made the beta spectra easier to observe and analyze. In the second case, beta spectrometers were taken into actual production and maintenance areas of DOE facilities. Some of the facilities had large quantities of unshielded uranium in various forms leading to high beta-to-gamma dose ratios. However, where fission products were found or where uranium was shielded, low beta-to-gamma dose ratios made beta spectra difficult to interpret. Low-energy (less than 200 keV) beta fields caused additional difficulties in interpretation due to the nonlinear energy response of the beta spectrometers at low energies.

(a) Mulvehill, J. M., L. W. Brackenbush, K. L. Swinth, and E. J. Vallario. 1987. Characteristics of Beta Detection and Dose Measurements at Department of Energy Facilities. PNL-5960, Pacific Northwest Laboratory, Richland, Washington.

At facilities where mixed fission products are present, handling and control procedures to limit internal exposures and exposure to penetrating radiations are also effective in controlling beta dose rates to workers. At facilities where large amounts of uranium (^{234}Th - $^{234\text{m}}\text{Pa}$ and ^{231}Th) are handled in the metal form, the observed beta dose rates were higher.

Analyses of beta and gamma spectra showed that ^{234}Th - $^{234\text{m}}\text{Pa}$, ^{231}Th , ^{137}Cs , and $^{90}\text{Sr}/^{90}\text{Y}$ were the major nuclides contributing to beta doses at the facilities visited. Beta doses from other fission products and ^{60}Co were also measured, but the potential for exposure was less significant.

Because of superior handling and control procedures at facilities with mixed fission products, beta dose rates to those workers were lower than beta dose rates to workers at facilities where unshielded ^{234}Th - $^{234\text{m}}\text{Pa}$ and ^{231}Th existed. In the latter facilities, rapid accumulation of black uranium-decay product dust on the outside of workers' gloves (leather gloves worn for mechanical protection) and the opportunity for glove interiors to become contaminated made dose rates to the hands potentially large. Dose rates greater than 800 mrad/h were measured inside one glove.

The beta dose rates determined with five different commercial survey meters were typically slightly lower than the accepted values from spectral measurements when the center of the entrance window was taken as the reference point and correction factors were applied for energy and geometry. However, if a single correction factor was used without consideration of beta energies or geometry, measured beta doses could be in error by a factor of 2. In extreme instances, the measured doses could be two orders of magnitude lower than the actual doses.

CONCLUSIONS

The analysis of the measurements and comparison of measurement techniques led to the following five major conclusions:

- Beta survey meters should be calibrated for beta energies and detector-source geometries similar to those encountered in the facility.

- The best results from survey meters were obtained when the center of the sensitive volume was used as the reference point. However, such a reference point is not useful when measured dose rates are required near (<5 cm) the source.
- Tests of a prototype beta-plus-gamma dose rate meter developed at the Idaho National Engineering Laboratory (INEL) showed superior performance based on agreement with our standard techniques.
- Doses measured by many dosimeters used at DOE facilities were only one-half of the doses measured by PNL's eight-element dosimeter.
- When a thin window covered the beta measuring element of a tested dosimeter, the response was much closer (80% to 90%) to that of PNL eight-element dosimeter, provided that an appropriate algorithm was used to resolve the dose components.

RECOMMENDATIONS

Based on the results of the site measurements and the analysis of measurement techniques, the following list of recommendations was developed.

- The INEL prototype beta-plus-gamma dose rate meter should be further developed.
- Further studies of detector response at energies below 200 keV should be undertaken to determine the degree of detector nonlinearity and its influence on dose calculations.
- Studies are needed on the influence of backscatter from detectors and detector windows on beta spectra and beta dose calculations.
- Further studies should be performed on the eight-element dosimeters to determine their response in mixed fields dominated by low-energy photons.
- A dosimeter similar to the eight-element unit, but using stacked elements should be developed. This would eliminate the concern about the spatial response of the present multielement dosimeter and permit the study of spatial variations in radiation fields.

- Dosimeters used in beta fields at DOE facilities should have a thin window (less than 20 mg/cm²) over the beta element.
- Conventional 235 mg/cm² TLDs should be replaced by thin TLDs (17 mg/cm²) in dosimeters used to monitor beta exposures.
- Dosimeters should be calibrated to beta energies similar to those encountered in the work areas. Yttrium-90 calibrations provide a beta spectrum with too high an average energy for many applications.
- The influence of source geometry and source-detector distance on measurements should be known for each instrument used in a facility. Whenever possible, measurements should be made to determine the correction factors.
- Personnel should be trained to use correction factors for source energy, source geometry, and source-to-detector distance on dose rate measurements.
- The variation in response of instruments with beta energy and angle of incidence should be documented.
- A modified ionization chamber instrument with a large-area, shallow chamber should be available for beta-dose-rate measurements.
- Workers who handle uranium metal products and associated processing materials should be made aware of the potentially high contact beta dose rates.
- Radiation protection procedures should be improved for workers in facilities where uranium decay products present beta dose problems. Contamination control, protective clothing, and dosimeter placement should be addressed.

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BETA MEASUREMENTS AT DEPARTMENT OF ENERGY FACILITIES

1.0 INTRODUCTION

Health physicists and health physics researchers need a good working knowledge of the beta spectra encountered by personnel at U.S. Department of Energy (DOE) facilities to ensure that personnel dosimeters and survey instruments are properly calibrated, designed, and used. Such information is important for the accurate measurements of personnel beta doses. In many cases, health physicists have assumed low beta-to-gamma ratios. Thus, beta (nonpenetrating) doses have not been considered the limiting factor in radiation exposures because permissible skin doses are three times the whole body limit. Plant modifications, inspections, maintenance, and repair may require workers to enter radiation environments where beta-to-gamma ratios are large. Then accurate monitoring of the dose equivalent to the skin and lens of the eye from beta radiation requires use of techniques different from those used in routine monitoring of the whole body dose equivalent.

In order to improve beta measurement capabilities at DOE facilities, one must first characterize the current practices. The characterization must include an assessment of current measurement practices and an assessment of the beta spectra and dose rates encountered at facilities. As the first step in characterization, a questionnaire was issued to survey the beta measurement practices at DOE facilities, and the survey results were analyzed (Mulvehill et al. 1987). The second step involved onsite measurements and a comparison of measurement techniques.

Pacific Northwest Laboratory (PNL) measured beta doses and spectra at seven DOE facilities. Thirty field locations, eleven liquid samples, and three wipe samples were evaluated with spectrometers and multielement dosimeters. Also, beta monitoring instruments and dosimeters used by plant health physics staffs were evaluated at several of the sites to compare their beta dose measurements with the PNL measurements. Measurements were also made with experimental dosimeters and instruments for evaluation of their performance in

an operational environment. This report presents the results of the measurements of radiation fields made at seven DOE facilities that had high ratios of beta-to-gamma dose rates and discusses the measurement techniques.

Section 2.0 describes the methodology used to obtain the data. Detailed descriptions of the measurement systems are presented. The methods used to collect the data in the field are discussed. A description of data analyses, including dosimeter and instrument responses and calibration, is contained in Section 3.0. The study findings are presented and discussed in Section 4.0. Conclusions and recommendations based on the study findings are presented in Section 5.0.

2.0 METHODOLOGY

This section describes the instrumentation and dosimeters used to measure beta radiation at selected DOE facilities. The measurement systems are documented below. Data collection and analysis procedures are described.

2.1 MEASUREMENT SYSTEMS

The measurement systems selected for this study were the silicon surface barrier spectrometer system (Fox and Borkowski 1962), the plastic scintillator system (Wohn et al. 1972), the germanium photon spectrometer system, and the multielement beta dosimeter (Scherpelz, Endres, and Rathbun 1983). These systems are described below. The equipment arrangement used for the germanium photon and silicon surface barrier spectrometers is the same and is shown in Figure 1. The arrangement for the plastic scintillator is also the same except that the preamplifier is not used.

Initially the silicon surface barrier detector was used as the spectrometer. It could measure the full energy range up to 2 MeV at dose rates of several hundred mrad per hour, and it was sufficiently rugged for use in the field. Later it was found that the plastic scintillation detector provided more accurate spectral measurements at low energies due to lower backscatter. Thereafter the plastic scintillator was used unless high dose rates dictated the use of the surface barrier detector.

FIGURE 1. Equipment Arrangement for the Beta and Photon Spectrometers

2.1.1 Silicon Surface Barrier Spectrometer System

The silicon surface barrier spectrometer system used a 100-mm² circular detector that was depleted to a depth of 5 mm at a bias of 1800 V. The detector was mounted, via a rear high-voltage mount, on a low-noise, charge-sensitive preamplifier. The detector was enclosed within a light-tight cardboard tube with two, 0.006-mm-thick, aluminized Mylar windows. Cables for the signal, bias voltage, and preamplifier power connected the preamplifier to a multichannel analyzer (MCA) containing an amplifier, bias voltage source, and preamplifier power source. The spectral data were collected with the MCA and stored on magnetic tape in a cassette recorder. Figure 2 illustrates a spectrum taken with the silicon detector showing the beta spectrum and conversion electrons from a ¹³⁷Cs source.

FIGURE 2. Cesium-137 Beta and Conversion Electron Spectrum Collected with a Silicon Surface Barrier Spectrometer

2.1.2 Plastic Scintillator Beta Spectrometer

A plastic scintillator detector was used for measuring low-dose-rate field spectra. It had a diameter of 5.08 cm and a thickness of 2.54 cm. The detector and a 1-cm-thick, nonscintillating light guide were optically coupled to a photomultiplier tube, the base of which incorporated a voltage divider network. The scintillator was fitted with a 0.003-mm, aluminized Mylar window.

The main problem with the plastic scintillator (and the silicon detector) is that they cannot accurately measure low-energy electron (less than 200 keV) fields. This is because both detectors must be covered with a metallic film to prevent interference from ordinary light. The metallic film prevents penetration of low-energy electrons. Also, if a light leak occurs, the low-energy portion of the spectrum is affected.

A phenomenon, called backscattering, occurs when a beta particle entering a detector undergoes sufficient deflection so that it re-emerges from the same surface through which it entered. The backscattered beta particles do not deposit all their energy in the detector, and therefore the detector will not measure the total energy of the original beta particle.

Backscatter for selected materials is shown in Figure 3, where T is the energy of the incident beta particle and n is the fraction of beta particles backscattered relative to the total number incident. The backscatter for silicon (not shown) would be slightly greater than the backscatter for aluminum, based on atomic numbers of 13 and 14 for aluminum and silicon, respectively. Backscatter is a problem for silicon even without a metallic window. The backscatter is much worse at lower beta energies than at higher energies. Based on Figure 3, the backscatter for plastic scintillators does not appear to be serious. However, the graph in Figure 3 applies only to normally incident beta particles. Information regarding the backscatter for beta particles that enter plastic at shallow angles is not readily available in the literature. It should be noted that plastic is very similar to human tissue for purposes of backscatter. Therefore, beta particles backscattered from the surface of plastic would similarly not deposit energy in the basal layer of skin.

FIGURE 3. Backscatter of Electrons from Selected Materials

The backscatter from plastic scintillators is primarily of importance when spectra obtained with the scintillator are compared with theoretical spectra or when doses are compared with ion-chamber measurements. Based on preliminary studies at PNL, it appears that the backscatter (outscatter) for low-energy beta particles incident at shallow angles is much more severe than Figure 3 would indicate. Additional studies are needed to resolve this question.

Simons et al. (1982) and Birks (1964) have reported that the plastic scintillator has a nonlinear response at low, beta particle energies (less than 125 keV). This response is thought to be due to quenching from damaged molecules along the track of the beta particle. The higher ionization density associated with lower energy beta particles enhances the effects of quenching, lowering the scintillation efficiency.

2.1.3 Germanium Spectrometer

In some cases, a portable, high-purity germanium detector was used to obtain photon spectra. In mixed fields, these were useful in interpreting beta spectra. The photon spectrometer system used a hyperpure germanium crystal. The remainder of the system is the same as described for the silicon detector.

Raw data were in the form of pulse-height distributions. The detector signal was proportional to the photon energy only when the full energy was collected. The detector did not collect the full energy for most photons because of the high probability for scattered photons to escape. The pulse-height distributions were corrected for detector efficiency and scattering loss to yield energy spectra.

2.1.4 Multielement Beta Dosimeter

The multielement dosimeter developed at PNL (Scherpelz, Endres, and Rathbun 1983) uses thermoluminescent dosimeter (TLD) chips placed under aluminum filters of varying thicknesses. Each filter attenuates the beta radiation corresponding to the thickness of the filter. An indication of the beta-energy spectrum is obtained by comparing the relative responses of the TLDs under the different filters (elements) to those for dosimeters that have been exposed to calibrated sources. The relative responses allow for the selection of a calibration factor for converting TLD response to dose.

The multielement beta dosimeter used in this study has eight elements. Each element contains three TLD chips covered by a filter made of either Mylar or aluminum. Table 1 describes the filter materials and thicknesses used in the dosimeter. Figure 4 is a photograph of the eight-element dosimeter.

TABLE 1. Multielement Dosimeter Filter Specifications

<u>Filter Material</u>	<u>Filter Thickness</u>		
	<u>in.</u>	<u>mm</u>	<u>mg/cm²</u>
Aluminized Mylar	0.00025	0.0063	0.65
Aluminum	0.001	0.025	7
Aluminum	0.005	0.127	34
Aluminum	0.010	0.254	69
Aluminum	0.020	0.508	137
Aluminum	0.032	0.813	219
Aluminum	0.064	1.626	439
Aluminum	0.125	3.175	857

FIGURE 4. Eight-Element Dosimeter

The dosimeters used were actually double-sided: two eight-element dosimeters were placed back-to-back, separated by a 3.175-mm aluminum sheet. Because the aluminum separator prevents beta particles that strike one face of the dosimeter from entering the other side, this dosimeter package can be used to measure the beta radiation coming from opposite directions. Thus, a dosimeter placed against a wall could determine separate doses for beta particles coming from wall contamination and for beta particles emitted by sources in the room.

The thickest dosimeter filter (3.175 mm) will stop beta particles with an incident energy of 1.9 MeV or less. The 0.127-mm filter will stop beta particles with energies below 180 keV. The Mylar filter will stop beta particles with energies below 20 keV. Photons with energies above 40 keV are not significantly attenuated by any of the filters.

Throughout this report the dose and dose rates determined with the multielement dosimeters are considered to be true or reference quantities. The multielement dosimeter was used as a reference because it is well characterized and has an energy-dependent calibration for beta measurements. Detailed descriptions and examples of uses of the multielement dosimeter can be found in several references (Scherpelz, Endres, and Roberson, 1984; Scherpelz, Endres, and Rathbun 1983; Rathbun et al. 1984). The error of the photon measurements made with the dosimeters will be about 9.5%. The error considers the standard deviations of the readings and of the calibration factors. For beta measurements, the errors will be about 11% for high beta-to-gamma ratios and will increase as the beta-to-gamma ratio decreases. The accuracy of the beta readings can be influenced by several factors:

- Nonuniform irradiation of the dosimeter elements by a gradient in the radiation field results in erroneous readings (the analysis routine flags these readings with a large error component).
- Significant exposures to low-energy photons (<40 keV) are recorded as beta exposures.

- Low beta-to-photon ratios may result in large errors for beta readings because the photon reading is subtracted from the beta reading.

2.1.5 Other Measurement Devices

In addition to the measurement tools described above, survey meters and commercial-type and prototype thin dosimeters were also used in field measurements. This provided an opportunity to evaluate the performance of typical survey meters or dosimeters and to test development dosimeters. In most cases, the survey meters used belonged to the facility where measurements were being made. A few commercial survey meters available at PNL were also included. Two-element dosimeters, prepared at PNL to simulate commercial dosimeters, were used at nearly all sites where measurements were made. A commercially available two-element dosimeter was used at some sites and a commercially available four-element dosimeter was used at other sites. In most cases, the PNL dosimeters contained the conventional 235 mg/cm^2 lithium fluoride TLDs. However, a few 100 mg/cm^2 lithium fluoride TLDs and 17 mg/cm^2 graphite-backed lithium fluoride TLDs were used. The latter were developmental dosimeters provided by Kansas State University (KSU) under a subcontract with PNL.

2.2 DATA COLLECTION

At each facility, PNL scientists reviewed routine survey data and discussed possible beta exposure sources with onsite health physics personnel. This information was used in conjunction with known equipment capabilities and potential for personnel exposure to choose the specific beta measurement locations.

To prevent contamination, all measuring devices were wrapped in plastic prior to entering radiation zones. The plastic covering acted as an additional filter for beta radiation, except for a few cases when it was removed at the measurement location. However, the plastic was less than 7-mg/cm^2 thick in all cases. Thus, the projected skin dose was not affected.

Field spectra were obtained under two distinct conditions. In the first case, silicon and scintillator spectrometer systems were set up in laboratory fume hoods. Small liquid sample vials and other contaminated materials were brought to the detector and placed immediately below the detector (Figure 5). The beta spectrum was obtained by subtracting the photon contribution (Figure 6) to the spectrum from the beta-plus-gamma spectrum (Figure 7). The photon contribution to the spectrum was determined by placing beta shields of plastic, ranging from 7-mg/cm² to 1000-mg/cm² thick (ICRP 1977), over the detectors. The detectors were exposed to beta radiation from the sample 0.4 to 1 cm in front of the detector. In some laboratories the ambient photon radiation was low, with the major photon dose resulting from the sample. This led to high beta-to-gamma ratios and made the beta spectra easier to analyze. However, it may not be representative of the relative beta versus gamma dose to the worker.

In the second case, beta spectrometers were taken into actual production and maintenance areas at the facility. Some facilities had large quantities of unshielded uranium in various forms. This made the ambient beta-to-gamma ratios high. However, where fission products were found, or where the uranium was shielded in the work locations, low beta-to-gamma ratios resulted. This made the beta spectra difficult to interpret.

FIGURE 5. Surface Barrier Detector and Liquid Sample

FIGURE 6. Cesium-137 Gamma Spectrum

FIGURE 7. Cesium-137 Beta-Plus-Gamma Spectrum

FIGURE 6. Carbon-13 EDS Spectrum

FIGURE 7. Carbon-13 EDS Spectrum

3.0 ANALYSIS TECHNIQUES

3.1 SILICON AND PLASTIC SCINTILLATOR DETECTORS

The following five sections describe how the data from the silicon and plastic scintillator detectors were analyzed. Beta detection and response, response to photons, average beta energies, beta dose calculations, and calibration are discussed.

3.1.1 Beta Detection and Response

Silicon detectors act as a solid-state analog of the ion chamber. Their structure is basically the same as that of normal semiconductor diodes for electronic applications, with a p-type region and an n-type region separated by a junction or depletion layer. The depletion layer is the sensitive part of the detector, where the concentration of charge carriers is practically zero. The detector is biased in the reverse direction. When a charged particle penetrates the depleted region, the electrons liberated there will move to the n-type zone (which has been given a positive voltage), while the positive charge carriers ("holes") move to the p-type zone. The average energy (ϵ) necessary to create an electron-hole pair in a given semiconductor is independent of the type and the energy of the ionizing radiation. Therefore, the number of electron-hole pairs produced is proportional to the incident energy of the incident particle, provided the particle is fully stopped within the active volume of the detector. The value of ϵ for silicon at room temperature is 3.62 eV, compared to about 33 eV needed to create an ion pair in a typical gas-filled detector.

When a beta particle interacts with a plastic scintillator, other electrons are set in motion. As these electrons travel through the crystal, they excite and ionize the atoms of the crystal and produce many flashes of light. The photons, which originate in the track, are emitted in all directions. The photons that reach the outside surface of the crystal are mostly returned by surface reflection and by reflection from the covering. As long as they are not absorbed in this covering, they will eventually pass through the light guide and strike the photosensitive surface of the photomultiplier. Upon striking the photosensitive surface, the light photons then eject low-energy

photoelectrons, which are collected and amplified. The number of electrons ejected from the photosensitive surface will be proportional to the amount of light that reaches this surface. This amount of light is in turn proportional to the energy absorbed from the beta particles incident on the crystal.

For both the silicon and plastic scintillator detectors, the output pulse of electrons is proportional to the initial energy of the incident beta particle. After amplification, the output pulses are fed into a pulse-height analyzer, which sorts them into equal energy intervals. If the detector systems are calibrated to the endpoint of a beta spectrum of known maximum energy, the pulse-height spectrum can be interpreted in terms of energy. Corrections are still necessary for backscatter and the portion of the spectrum lost due to the discriminator setting of the analyzer.

For the thicknesses stated (in Section 2.0), both detector systems have good detection efficiencies for beta particles between 125 keV and approximately 2.5 MeV. Below 125-keV, backscatter of beta particles from the surface of the detectors (especially silicon) causes a much lower detection efficiency. The response of the detectors is also nonlinear with respect to energy in this region due to a rapidly changing stopping power. In addition, electronic noise and x-ray interference normally prevent the analysis of beta particle interactions in the silicon detector below 70 keV and in the plastic scintillator below 40 keV. Considerably better results for low-energy particles are possible under optimum laboratory conditions. However, for radiation protection purposes, it is not necessary to analyze beta particles below 70 keV.

Based on the discussion above, the plastic scintillator is a better choice for beta particle measurements in low-intensity fields. The scintillator has lower backscatter, responds to lower energies, and is nearly equivalent to tissue in its radiation absorption characteristics. However, the upper limit for this detector is approximately 100 mrad/h. The silicon detector can be used up to approximately 1 rad/h. Beyond these limits, pulse pileup becomes a problem.

Figure 8 illustrates the response of the silicon detector to three laboratory beta sources. These spectra exhibit significant departures from the bell-shaped curve typically thought to be characteristic of beta spectra.

FIGURE 8. Comparative Detector Responses to Laboratory Beta Sources

The bell shape arises from probability considerations about the electron and neutrino sharing the available nuclear disintegration energy. When a nucleus disintegrates by beta-ray emission, the most probable division of energy between the beta particle and neutrino is that each will carry roughly the same energy. Departures from the bell shape (such as those seen above) depend on whether more than one endpoint energy is involved ($^{90}\text{Sr}/^{90}\text{Y}$) and whether the beta particles leave the nucleus tangentially or radially.

In a pure beta field emanating from a single beta emitter, or even two beta emitters if their spectra have widely differing maximum energies, it is possible simply by observation to identify the nuclides responsible for the beta field. This is important because it helps to determine endpoint and average energy and calculate doses. To aid in identifying nuclides, theoretical spectra can be generated based on the quantum mechanical theory of beta

decay developed by Fermi. These spectra are made more accurate by the application of relativistic and self-screening corrections. Figure 9 shows typical beta spectra calculated from theory. In Figure 9, the horizontal scales give energies in million-electron volts and the vertical scales give the number of beta particles emitted per million-electron volts for one transition (Cross et al. 1984).

FIGURE 9. Examples of Beta-Ray Spectra, Illustrating the Wide Variety of Shapes

3.1.2 Photon Response

For typical nonlaboratory conditions, radiation fields are seldom caused by pure beta emitters. Therefore, the response of the detectors to photons is important. Although photon contributions to the spectrum can ideally be subtracted out, a one-to-one subtraction often does not completely remove the photon-generated portion, especially in high photon flux measurements. Virtually all the photon interactions of importance in the detectors are Compton scattering events, with recognizable Compton edges. Compton scattering is an elastic collision between a photon and a free electron. The Compton electron dissipates its kinetic energy in the same manner as a beta particle in the silicon detector. The Compton electron can have any energy up to a maximum value, which approaches (but never equals) the energy of the incident photon. An "edge" or sudden decrease in pulse height occurs in the spectrum at these maximum electron energies. Compton edges are useful in identifying nuclides and calibrating the detector system (Attix and Roesch 1968).

3.1.3 Average Beta Energies

One property of a beta-ray spectrum that is of great importance in dosimetry is the average beta-ray energy per disintegration ($\bar{E}_\beta$), which is defined as

$$\bar{E}_\beta = \frac{\int_0^{E_0} EN(E) dE}{\int_0^{E_0} N(E) dE} \text{ MeV/dis} \quad (1)$$

where $N(E)$ is the relative number of beta particles in the spectrum with energy between E and $E + dE$, per unit energy interval and E is the beta energy. The importance of $\bar{E}_\beta$ is that it represents, on the average, the beta-ray energy per disintegration that can be absorbed by tissue from radioactive decay within the tissue. The average beta energy is approximately one third of the maximum beta energy.

It follows from Equation (1) that $\bar{E}_\beta$ can be computed by graphical integration of a beta spectrum.

3.1.4 Dose Calculations

Figure 10 shows the silicon detector, whose detection volume is the depletion region around the junction. It can be seen that the average dose absorbed in this region is

$$\frac{\sum E}{m} \quad (2)$$

where $\sum E$ is the energy lost by beta particles while travelling in the depletion region with mass, m .

Dose can be measured by relatively simple pulse-height analysis of the charge pulses observed.

The beta dose calculation is based on the concept that the mass stopping power for a particular energy beta particle multiplied by the fluence of beta particles with that energy is equivalent to the dose-rate contribution from particles of that energy. The energy spectrum is actually a histogram of energy versus frequency, so the fluence in each energy bracket is multiplied by the median mass stopping power for that energy bracket. The median mass

FIGURE 10. Reversed Biassed Silicon Junction Beta Dosimeter
(Jones 1984)

stopping power is determined using Bethe's continuous slowing down approximation. The integral value of this multiplication is equal to the absorbed dose per unit time. However, this product must be multiplied by 1.6×10^{-6} ergs/MeV to give standard dosimetric units (Murphy et al. 1984). Also, because of the nonlinear response for low-energy beta particles, an empirical correction factor of 1.33 must be applied.

3.1.5 Calibration

In order to perform accurate measurements, calibrations must be performed for energy response and efficiency. Energy calibration and efficiency calibration are discussed in the following sections.

Energy Calibration

Because of the easily recognizable conversion electron peak and the well-defined Compton edges, ^{137}Cs and ^{207}Bi are the radionuclides of choice for the calibration of silicon detectors and plastic scintillators.

Internal conversion involves the emission of an electron in the de-excitation of a nucleus by direct coupling between the excited nucleus and an extranuclear electron, usually one in the K, L, or M shell. The ejected electron, called a conversion electron, has kinetic energy equal to the difference between the transition energy and the binding energy of that electron. Conversion electrons are observed as a line spectrum easily distinguishable from the continuous beta-ray spectrum when beta rays are present.

An International Atomic Energy Agency (IAEA) ^{137}Cs source with very thin encapsulation was used as the primary source of calibration. This source is traceable to the National Bureau of Standards. Both detectors discussed in Section 3.1 can also be calibrated to the endpoint energies of sources such as ^{90}Y , ^{36}Cl , and $^{121\text{m}}\text{Sn}$. Calibration to ^{147}Pm is not recommended, because of the low-energy nonlinearity of the detectors.

Efficiency Calibration

In order to make accurate measurements of doses and dose rates the efficiency of the detectors and dosimeters must be known for the conditions in which they are used. The efficiency calibration is affected by the beta

energy, the source geometry, and the source to probe distance. Energy calibration factors were determined by using the sources recommended by the International Organization of Standardization (ISO 1981) and sources of ^{85}Kr , nat_U , and $^{106}\text{Ru}/^{106}\text{Rh}$, which are calibrated and maintained at the PNL calibrations laboratory.

Beta doses and dose rates are often measured near sources of various geometries. This requires corrections to the instrument response because efficiency calibrations generally assume a uniform irradiation of the detector volume. Such correction factors vary with energy and instrument in addition to the geometrical variables. Complete data for such corrections are not available, and data determined for a few instruments in use at PNL were used as appropriate. The calibrations are discussed in more detail in the following sections, which discuss the specific measurement tools.

3.2 PACIFIC NORTHWEST LABORATORY MULTIELEMENT DOSIMETER

The analysis of multielement dosimeter measurements is described below in Sections 3.2.1, TL Chip Readout and Dose, and 3.2.2, Calibration.

3.2.1 Thermoluminescent Chip Readout and Dose

Thermoluminescent chips taken from the dosimeters are read out in a TLD reader which provides results in units of nanocoulombs per unit of light output. The TLD responses are then used as raw data in a computer program that converts the TLD responses to absorbed doses.

The reading for the TLDs under the thickest filter (3.175 mm) is used to indicate the gamma dose, and this value is subtracted from the other element readings to obtain the response due to beta particles alone. The program then determines the response ratios of aluminum-covered elements to the Mylar-covered element. It uses these ratios, which are an indication of the beta-energy spectrum, to select the appropriate beta-energy correction factor. This correction factor is then multiplied by the response of the Mylar-covered element to determine the beta dose at 0.65 mg/cm^2 .

3.2.2 Calibration

The multielement beta dosimeters were calibrated by exposing them to known radiation sources at the PNL calibrations laboratory. The beta sources used in these calibrations and the maximum energies of the emitted beta particles are presented in Table 2.

The dosimeters were also exposed to calibrated sources of x rays with effective energies ranging from 15 to 78 keV, and to a calibrated ^{137}Cs source emitting 662-keV gamma rays. These measurements characterized the multielement dosimeter response to photons and showed that photon attenuation is measurable only for energies less than about 40 keV. For photons with higher energies, the TLD response is fairly uniform for all elements. This dosimeter characteristic permits using the response of the element with the thickest aluminum filter as an indicator of photon dose.

The data from these beta and photon calibrations were used to derive calibration factors for determining the beta doses from exposed dosimeters (Scherpelz, Endres, and Rathbun 1983).

For each dosimeter exposed to a calibrated beta source, a beta "Calibration Factor" was determined by subtracting the gamma component (the reading of the element covered by the 0.318-cm Al shield) from the TLD response of the Mylar-covered shield, and dividing this value (in nanocoulombs) into the beta dose (in rads) absorbed by the Mylar-covered TLDs during the calibration measurement. The beta dose delivered by the calibration source is defined as the dose to tissue at a depth of 7 mg/cm^2 . This factor could then be used to

TABLE 2. Characteristics of Beta Calibration Sources

Nuclide	Maximum Beta, Energy, MeV	Dose Rate, mrad/h	Distance, cm
^{147}Pm	0.23	10.2	20
^{85}Kr	0.62	1150	50
^{204}Tl	0.76	64.1	30
$^{90}\text{Sr}/^{90}\text{Y}$	2.3	164	50
$^{106}\text{Ru}/^{106}\text{Rh}$	3.5	200	50

convert the responses of other dosimeters to a 7-mg/cm² dose, as long as the dosimeters were exposed to a field of beta particles with the same energy distribution as the beta calibration source. For each of the five elements with aluminum shields between 0.013- and 0.163-cm thick, the ratio of the element's beta response to the Mylar-covered element's beta response was found for each calibration measurement. This ratio was a function of the calibration source's energy distribution: high beta energies resulted in high values of these ratios.

3.3 COMMERCIAL DOSIMETERS AND OTHER DOSIMETERS

Commercial dosimeters and other dosimeters similar to those used at various DOE facilities were placed at many of the locations where spectral and survey meter measurements were obtained. Both two- and four-element dosimeters were exposed during field measurements. Thermoluminescent (TL) materials in the dosimeters included lithium fluoride, lithium borate, and calcium sulphate. The TL materials ranged in thickness from approximately 17 mg/cm² to 240 mg/cm². Filters covering the TL materials ranged from 0.006-mm (0.65-mg/cm²) aluminized Mylar to 0.102 cm of tin and 0.08 cm of plastic. This tin-plastic filtration is equal to a mass thickness of 900 mg/cm².

In a two-element dosimeter, the dose is determined from the following expression

$$\text{Calibration Exposure} \times \frac{(R_1 - K_2 R_2) \text{ Field Exposure Chip Readings}}{(R_1 - K_2 R_2) \text{ Calibration Exposure Chip Readings}}$$

where R_1 and R_2 are the net readings of chips 1 and 2 after subtracting their respective chip backgrounds, and K_2 is a constant nearly equal to 1.0 representing the differences in ¹³⁷Cs photon response between R_1 and R_2 . The penetrating dose calculation is most direct, depending only on the net response of chip 2 relative to the calibration response. The evaluation of four-element dosimeters follows the procedure indicated for the PNL multi-element dosimeter (Scherpelz, Endres, and Rathbun 1983).

Both the two-element dosimeters designed at PNL and the commercial two-element dosimeters consisted of a set of TLDs under a thin absorber (88 mg/cm² for PNL and 30 mg/cm² for commercial) and a set of TLDs under a thick absorber, approximately 350 mg/cm² in both cases. The PNL two-element dosimeters were calibrated to ¹³⁷Cs gamma rays and ⁹⁰Y beta particles.

The calibration factors for the various TLDs in the PNL-design and commercial two-element dosimeters are shown in Table 3. The commercial dosimeters were calibrated and processed by the supplier.

Because the trend is to calibrate dosimeters with high-energy beta sources, i.e., ⁹⁰Sr/⁹⁰Y with a 100-mg/cm² attenuator, nonconservative beta dose results are often obtained. Causes for this situation are twofold: the covering material is too thick and the dosimeters are too thick. The covering material can easily be made thinner. However, the effect of a thin covering is largely lost when the TLD behind the thin window is covered by a security badge. Thin TLDs were used by KSU to obtain experimental verification of the strong energy dependence of TLDs exposed to beta radiation. Table 4 presents a brief summary of how much the doses would be underestimated for different maximum energy beta sources, relative to the ⁹⁰Sr/⁹⁰Y dose measured at 50 cm in air. For a commercial 0.89-mm- (235 mg/cm²-) thick TLD covered with 1.45 mg/cm² of Mylar, the reported dose would be high by 290% for ²⁰⁴Tl beta

TABLE 3. Calibration Factors for Two-Element Dosimeters

TLD Thickness, mg/cm ²	Gamma Calibration Factor, rad/nC	Beta Calibration Factor, rad/nC
235	1.89	2.67
100	4.59	5.70
17 ^(a)	28.2	33.9

(a) The calibration factors for the 17 mg/cm² TLDs apply only to the average of the group. Each TLD in the group had an additional correction factor to compensate for variations in thickness. This additional factor was usually between 0.9 and 1.1.

TABLE 4. Measured Factor for How Much a LiF TLD Would Underestimate the Actual Beta Dose Relative to the PTB $^{90}\text{Sr}/^{90}\text{Y}$ ($E_{\text{max}} = 2.27 \text{ MeV}$) Dose

Beta Source	Maximum Beta Energy, MeV	Cover Thickness, mg/cm^2	Average TLD Thickness, mg/cm^2				
			235 (LiF)	18.9 ± 5.3 (LiF)	13.1 ± 4.2 (LiF)	19.3 ± 2.8 (^7LiF)	24.7 ± 7.8 (^6LiF)
^{147}Pm	0.225 ^(a,b)	0	-	2.91	2.24	2.80	3.44
		1.4	27.4	4.27	3.31	3.89	4.89
^{204}Tl	0.763 ^(b)	0	-	1.13	1.04	1.09	1.12
		1.4	3.93	1.17	-	-	-
		14.5	5.15	-	1.44	-	-
		27.6	-	-	-	1.97	-
^{85}Kr	0.690	1.4	3.38	1.11	-	-	-
$^{106}\text{Ru}/^{106}\text{Rh}$	3.54	1.4	1.04	1.01	-	-	-

(a) The maximum beta energy was much less than this value at the TLD position.

(b) Physikalisch-Technische Bundesanstalt (PTB) sources.

particles. The corresponding error for a nominal 18.9 mg/cm^2 TLD would be 17%. The ^{147}Pm data are difficult to discuss because 200 mm of air and a beam flattener were between the source and the TLDs. This configuration would provide extremely low-energy beta particles. Nevertheless, typical errors for thick and thin TLDs were factors of 27.4 and 4.27, respectively.

3.4 SURVEY METERS

The survey meters found at each DOE facility visited were studied for performance under field conditions. All the survey meters (except two) were of the air ionization type. The survey meters tested are referenced by a randomly assigned alphabetical character to avoid comparison of specific products. The exceptions were a cylindrical GM tube (Instrument D) and a thin scintillator survey instrument (Instrument E). Detector window thickness ranged from 3.5 mg/cm^2 to 30 mg/cm^2 . All instruments were designed to measure both beta and gamma radiation. In two cases, the instrument design favored photon monitoring over beta monitoring because the length of the detector exceeded the width of the detector. However, for one ion chamber the width (diameter) was about 1.7 times the length of the detector chamber. Even though this ratio is far from optimal, it tends to favor beta monitoring. The advantages of beta monitoring of a "pancake" type of ion chamber were described by Roesch and Donaldson (1956), but this type of instrument has not been commercially manufactured in the United States.

The response of a survey meter to beta fields is imperfect because it uses a large volume to represent the dose to a small volume of tissue. Beta fields will lead to energy-dependent response of meters because low-energy beta particles cannot uniformly irradiate the detector volume (Table 5). Similarly when the geometry of the source or the source-to-detector distances lead to nonuniform irradiation of the detector volume, the actual dose rate will be underestimated (see Tables 6 and 7, where Instrument C is an ion-chamber survey meter in a "Cutie Pie" configuration). In order to correctly estimate the dose rate, correction factors similar to those in Tables 5, 6, and 7 must be applied to the instrument readings.

TABLE 5. Energy Calibration Factors for Survey Instruments^(a)

Source	¹⁴⁷ Pm	⁸⁵ Kr	²⁰⁴ Tl	⁹⁰ Sr/ ⁹⁰ Y	U _{nat}	¹⁰⁶ Ru/ ¹⁰⁶ Rh
Maximum Energy, MeV	0.23	0.62	0.76	2.3	2.3	3.5
Distance, cm ^(a)	20	50	30	50	contact	50
A (air ion, 7 mg/cm ²) 208 cc	2.8	2.6	2.5	1.2	3.2	1.8
D (GM, 30 mg/cm ²) 17.3 cc	1.8				6.4	
E (thin scintillator dose rate meter)	1.0	1.6	1.0	1.0		0.8

(a) Rathbun et al. (1984).

TABLE 6. Examples of "Cutie Pie" (Instrument C) Correction Factor for Disk Sources^(a)

Source Diameter	Correction Factors			
	Source - Window Distance			
	Contact	1/2 in.	1 in.	6 in.
1/2 in.	90.0	9.0	1.9	1.0
1 in.	35.0	5.0	2.0	1.0
6 in.	4.5	2.6	2.2	1.0
1 ft	4.5	2.5	2.3	1.3
6 ft	4.5	2.5	2.5	2.3

(a) In order to obtain the beta exposure/dose rate at the detector window, multiply the meter reading by the applicable correction factor (from Selby et al. 1983).

A thin scintillator survey meter was developed by the Idaho National Engineering Laboratory (INEL) and described in a paper by Johnson et al. (1983). The detector system consists of a very thin plastic scintillator (NE-105) covered by a light-tight, aluminumized Mylar entrance window. The scintillator is backed by a 1-cm-thick, lucite, light pipe, attached to a photomultiplier tube.

TABLE 7. Examples of "Cutie Pie" (Instrument C) Correction Factors for Contact with Cylindrical Sources^(a)

<u>Cylinder Diameter</u>	<u>Correction Factor</u>
1/2 in.	31.0
3/4 in.	17.0
1 in.	11.0
2 in.	5.7
3 in.	5.0

(a) Correction factors are for cylinders greater than 3-in. long (from Selby et al. 1983).

For all the survey meters, except the dual chamber instrument and the thin scintillator device, discrimination between nonpenetrating and penetrating radiation was determined as follows:

- penetrating dose rate = meter reading with filter covering the window
- nonpenetrating dose rate = (meter reading with filter removed - meter reading with filter in place) x correction factor.

The dual-chamber survey meter measured beta-plus-gamma fields and gamma fields simultaneously. Correction factors were determined by placing the instruments on a uranium slab covered by 7-mg/cm² filtration and by following the specified procedures for the ISO beta sources (ISO 1981). Although Instruments B and F were not calibrated by PNL, typical calibration factors for similar instruments were previously determined by PNL and are listed in Tables 5, 6, and 7. Instrument B should have a calibration factor that is close to that of Instrument A, and Instrument F had a calibration factor of 1 in the configuration used.

Typically, beta survey meters at DOE facilities are calibrated to either a uranium slab or a ⁹⁰Y point source and the face of the instrument is usually the point of calibration. The calibration factors thus derived (3.2 and 1.2

for uranium and ^{90}Y , for Instrument A) are applicable only when the instrument is exposed to field conditions that are similar to the calibration conditions. Because the instrument readings are commonly used to determine potential doses to individuals, the instrument must be in the same proximity to the source as the person might be. In some cases, the beta energies, geometry, and filtering media that occur in a personnel exposure situation adequately match the conditions under which the instrument was calibrated. For example, the calibration factor derived for a ^{90}Y point source at 50 cm can be used for much larger sources at distances greater than 50 cm. Also, due to a fortunate coincidence, the calibration factor for a survey meter with its face in contact with a uranium slab is often only slightly larger (Table 5) than the calibration factor for the same instrument with its face at 20 cm from a ^{147}Pm point source. The average factor (Instrument A) for the uranium slab and ^{90}Y source is

$$\frac{(1.2 + 3.2)}{2} = 2.2 \text{ (Table 5)} \quad (3)$$

The highest calibration factor for the face of Instrument C in Table 6 is 2.5. Based on the foregoing discussion, a calibration factor of 2.5 is considered conservative for all except contact and near-contact situations, and it is used by many DOE contractors for instruments similar to A and C. However, calibration factors ranging from 1 to 5 have been observed. As seen in Tables 6 and 7, large correction factors are needed for survey meters when near-contact readings are desired. These factors vary with the type of instrument, the calibration parameters adopted, and the source geometry.

For many popular survey meters, if the calibration point is taken as the center of the instrument chamber rather than the face, the calibration factor will be between 1 and 2 for all conditions except very close to small or low-energy sources. Using the center of the chamber volume as a reference makes it impossible to obtain dose rates within a few centimeters of the source. This information is often important because workers work close to the material.

4.0 RESULTS

4.1 FIELD SPECTRA MEASUREMENTS

This section discusses the results of field measurement conducted at each of the seven DOE facilities (Sites A through G).

4.1.1 Site A: Separations Facility

Field measurements were conducted at Site A on two different occasions to provide an indication of beta-plus-gamma fields, which are routinely encountered at a DOE separations facility. Both sets of measurements included a process sample and a waste sample. The first measurements at Site A were taken in a laboratory hood in a low-background area. The samples were contained in small glass vials. A silicon detector was placed in the same hood at distances of 30, 60, and 90 cm from the sample.

Figure 11 shows the beta-energy spectrum for the process sample at 30 cm. The spectrum was obtained by placing 412 mg/cm² of aluminum absorber in front of the detector and subtracting the resultant spectrum from the beta-plus-gamma spectrum (Figure 12). Thus, Figure 11 approximates the beta spectrum; approximately 10% of the beta particles at higher energies will penetrate the aluminum and thus be subtracted. Also, backscatter of incident low-energy beta particles and x-ray attenuation in the absorber cause a possible distortion of the low-energy portion of the spectrum. A comparison of the two spectra reveals that only about one-half of the electrons generated in the detector were due to beta radiation. Compton scattering of photons dominated the spectrum below 200 keV. Only gamma radiation was evident above approximately 1000 keV.

The same procedure just described was used to obtain spectra for a waste sample during the first measurements at Site A (Figures 13 and 14). Although these spectra appear to be somewhat similar to those of the process sample, the relative intensity of the beta field is much smaller and the endpoint of the beta spectrum is at a lower energy (approximately 620 keV).

The second set of measurements at Site A was made nearly two years after the first set and under somewhat different conditions. The process sample in

FIGURE 11. Beta Spectrum for the First Process Sample at 30 cm

FIGURE 12. Beta-Plus-Gamma Spectrum for the First Process Sample at 30 cm

FIGURE 13. Beta Spectrum for the First Waste Sample at 30 cm

FIGURE 14. Beta-Plus-Gamma Spectrum for the First Waste Sample at 30 cm

this instance was in the form of a drop of liquid on a gauze pad. It was placed in a ventilated hood in a relatively high gamma background area. The measurement area was immediately adjacent to a process area, separated by a wall containing a shielded pass through. Figure 15 is the beta-plus-gamma spectrum for the process sample. This spectrum was obtained with a plastic scintillator placed 30 cm from the sample. Although this spectrum looks similar in shape to the process sample spectrum obtained during the first Site A visit, there is a much more dominant beta component (Figure 16); the gamma component is negligible above approximately 600 keV. This was verified by a gamma spectrum obtained with a portable high-purity germanium detector near the source. The number of photon interactions beyond 800 keV was negligible compared with the number occurring at less than 800 keV. The 796-keV peak (^{134}Cs) seen in the germanium spectrum (Figure 17) would occur as a 600-keV Compton edge in the plastic scintillator spectrum.

The second waste sample was measured at the same location as the process sample just discussed. However, the waste sample was in a glass vial within a lead pig. The beta-plus-gamma spectrum that was obtained with a silicon (lithium) surface barrier detector is shown in Figure 18. The window of the silicon(lithium) detector was 7.5 cm away from a rubber-like membrane covering the top of the sample bottle. Although this membrane would not drastically attenuate beta particles with energies greater than a few hundred kiloelectron volts, the glass bottle would. This is important, because the bottle subtended a much larger solid angle at the detector than did the small membrane at the top. The ratio of solid angles is greater than an order of magnitude. Hence, Figure 18 is essentially a gamma spectrum below the Compton edge for the highest photon energies. This occurs at approximately 600 keV, as seen in Figure 19 where only the beta spectrum above the Compton edge remains after subtracting out the heavily filtered spectrum. This result is due to the Compton electrons leaving the lucite filter and striking the detector in numbers greater than the original number of beta particles. The subtraction of spectra then "erases" the beta spectrum up to the Compton edge of the photon spectrum. This phenomenon only occurs at small beta-to-gamma ratios. The visible portion of the beta spectrum in Figure 19 is similar to that for the process sample.

FIGURE 15. Combined Beta-Plus-Gamma Spectrum from the Second Process Sample

FIGURE 16. Beta-Only Spectrum from the Second Process Sample

FIGURE 17. Germanium Photon Spectrum from the Second Process Sample

FIGURE 18. Beta-Plus-Gamma Spectrum from the Second Waste Sample

FIGURE 19. Beta-Only Spectrum from the Second Waste Sample

4.1.2 Site B: Fuel Fabrication Facility

Field measurements taken at Site B provide an indication of beta-plus-gamma fields, which are routinely encountered with clad fuel at a DOE fuel fabrication facility. Two 50-cm-long, 7-cm-dia fuel element sections, one 0.95% enriched in ^{235}U and the other 1.25% enriched in ^{235}U , were measured to provide typical spectral information. A plastic scintillator, a silicon surface barrier detector, and a high-purity germanium detector were used for this study.

Measurement instruments and fuel elements were arranged on carts in the building containing the fuel fabrication operations. All beta spectroscopy measurements were performed with detectors at one-half inch from the surface of the fuel elements, midway between the ends.

The beta and gamma spectra are compared in Figures 20 through 22. Most of the counts in the mixed spectrum (Figure 20) are due to the Compton scattering of photons. When the beta particles were removed by a plastic filter, the remaining spectrum (Figure 21) was virtually the same as Figure 20. The subtraction of spectra shows the beta spectrum (Figure 22), which is very

FIGURE 20. Beta-Plus-Gamma Spectrum for the 1.25%-Enriched Fuel Element

FIGURE 21. Gamma Spectrum for the 1.25%-Enriched Fuel Element

FIGURE 22. Beta Spectrum for the 1.25%-Enriched Fuel Element

low in intensity due to attenuation by the fuel element cladding (2 mm of zirconium).

Figures 23 and 24 show the gamma spectra of the 0.95% and 1.25% enriched fuel elements. Both spectra are practically identical except for the 185.7-keV gamma ray from ^{235}U being more prominent in Figure 25.

4.1.3 Site C: Separations Facility

At Site C, another separations facility, beta spectral measurements were taken in an analytical laboratory on samples from eight liquid waste tanks. All but one of the spectra were characterized with a scintillation detector. The liquid samples were stored together in one section of the laboratory. Each liquid sample was contained in a small (less than 10 ml) glass or plastic bottle ($\approx 1.6\text{-cm}$ dia $\times 5.0\text{-cm}$ high). These bottles, typically one-half full, were stored inside small lead pigs with two bottles per pig.

Prior to beta analyses, each lead pig was taken to a low-background portion of the laboratory and placed inside a hood where the beta analysis

FIGURE 23. Germanium Gamma Spectrum for the 1.25%-Enriched Fuel Element

FIGURE 24. Germanium Gamma Spectrum for the 0.95%-Enriched Fuel Element

system was set up. The face of the beta detector was suspended 10 cm above the floor of the hood. This allowed the liquid samples to be placed, one at a time, into a glass beaker and slid under the detector. Metal tongs and long tweezers were used to place the sample bottles into the beaker and remove the lids. As soon as the lids were removed, "Cutie Pie" (CP) readings were taken for comparison with spectral results. With the arrangement described, the face of the beta detector was 2.5 cm from the top of the sample bottles, and typically 5 cm from the liquid surface, although there was a 2-cm variation between the lowest and highest liquid level. The outside of the bottles were slightly contaminated. In some cases, the photon contribution from the sample still in the lead pig complicated the spectrum from the sample being analyzed. Even with these variables, the relationship between the detectors and the samples constituted a situation seldom seen in field studies, i.e., a small beam of beta particles nearly perpendicular to the detector face.

Figure 25 shows the combined beta-plus-gamma spectrum, and Figure 26 shows the beta-only spectrum from the first tank sample (taken from Tank 105AW). Although the Compton-scattering of photons contributes more than half

FIGURE 25. Beta-Plus-Gamma Spectrum for the Sample from Tank 105AW

FIGURE 26. Beta Spectrum for the Sample from Tank 105AW

of the events at energies less than 500 keV, the beta spectrum from ^{137}Cs is very evident in this same region. The conversion electron peak from ^{137}Cs is seen in Figure 26 between 500 and 700 keV. There are high-energy beta particles in the spectrum; some of the high-energy counts may arise from pulse summing due to the high count rates (dead time = 45%).

Figures 27 and 28 show the combined and beta spectra, respectively for the sample from the second waste tank (Tank 107AN). When the photon contribution (probably ^{137}Cs) is subtracted, the beta spectrum is very well defined and intense.

However, to date, no nuclide with such a high-energy spectrum has been identified. There are many beta emitters with endpoint energies greater than 4 MeV, but all have very short half-lives and none of these have a long-lived precursor. The most plausible explanation for the observation is pulse pileup in the analysis system causing two or more lower energy events (less than 2.5 MeV) to sum and appear as a higher energy event. This would explain events above 3 MeV. The spectrum is thought to be accurate below 3 MeV.

FIGURE 27. Beta-Plus-Gamma Spectrum for the Sample from Tank 107AN

FIGURE 28. Beta Spectrum for the Sample from Tank 107AN

The spectra for the sample from the third waste tank (Tank 105AN), Figures 29 and 30, show ^{137}Cs and a very small contribution from a higher energy beta emitter, probably ^{90}Y . The photon contribution at lower energies (less than 800 keV) is at least four times as great as the beta contribution.

Figures 31 and 32 show the spectra for the sample from the fourth waste tank (Tank 102AZ). Although the conversion peak for ^{137}Cs is evident, the subtraction of the photon component also removes most of the lower energy beta spectrum. A small contribution from higher energy beta particles is evident.

The sample from the fifth tank (Tank 105AW) has no measurable beta spectrum. The unfiltered spectrum, primarily from ^{137}Cs , is shown in Figure 33.

Another ^{137}Cs spectrum is seen in Figures 34 and 35, which are based on analyses of the sample from the sixth waste tank (Tank 105AN154). The beta contribution to the total spectrum is approximately one-half of the total amplitude of the spectrum. The conversion electron peak is especially well defined in this case (Figure 35).

The spectra from the sample from the seventh waste tank (Tank 104AN272) are also ^{137}Cs spectra. The presence of ^{239}Np , known to be present due to a

FIGURE 29. Beta-Plus-Gamma Spectrum for the Sample from Tank 105AN

FIGURE 30. Beta Spectrum for the Sample from Tank 105AN

FIGURE 31. Beta-Plus-Gamma Spectrum for the Sample from Tank 102AZ

FIGURE 32. Beta Spectrum for the Sample from Tank 102AZ

FIGURE 33. Gamma Spectrum for the Sample from Tank 105AN77

FIGURE 34. Beta-Plus-Gamma Spectrum for the Sample from Tank 105AN154

FIGURE 35. Beta Spectrum for the Sample from Tank 105AN154

gamma spectral analysis, is not noticeable in the beta spectrum, but the ill-defined conversion electron peak from ^{137}Cs may be influenced by the presence of the ^{239}Np . The beta contribution (Figure 36) is only about one quarter of the total amplitude of the curve (Figure 37).

In the last liquid sample spectra (Tank 102AW), the ^{137}Cs spectrum (beta-plus-gamma) is complemented by the beta spectrum from a higher energy beta emitter with an endpoint energy of approximately 2.2 MeV (Figure 38). The beta-only spectrum is shown in Figure 39. This sample was analyzed with a silicon detector at 7 cm from the liquid due to the intense beta-plus-gamma field. The silicon detector spectrum is similar to that for the scintillator used earlier, but there is more backscatter of electrons at lower energies with the silicon detector.

FIGURE 36. Beta Spectrum for the Sample from Tank 104AN272

FIGURE 37. Beta-Plus-Gamma Spectrum for the Sample from Tank 104AN272

FIGURE 38. Beta-Plus-Gamma Spectrum for the Sample from Tank 102AW

FIGURE 39. Beta-Only Spectrum for the Sample from Tank 102AW

4.1.4 Site D: Uranium Foundry

Site D is a depleted uranium metal-working and foundry facility. Field spectra were obtained by taking a plastic scintillator spectrometer into actual production areas. Each beta spectrum was obtained by subtracting the photon portion of the beta-plus-gamma spectrum. To obtain the gamma-only spectra, beta shields of plastic, ranging in thickness from 60 mg/cm² to 1000 mg/cm² were placed over the detector. Because low-energy (less than 20 keV) photon fields and intense high-energy photon fields caused distortions in the subtracted beta spectra, it was necessary to characterize the photon fields where beta measurements were made.

In Figures 40 through 44 the beta spectra seen are very similar in shape to a typical uranium spectrum. However, the average energies derived from the spectra are slightly higher than the 800 keV average energy expected for uranium. This apparently stems from two sources: 1) pulse pileup, and 2) the subtraction of distorted gamma spectra. The pulse pileup occurs at high count rates when two simultaneous pulses are summed to appear as one higher energy event. The distorted gamma spectra result when a thick plastic filter is placed on the face of the detector. Under broad-beam conditions, the number

FIGURE 40. Beta Spectrum for the End Cut from a Uranium Billet

FIGURE 41. Beta Spectrum for the Top of a Uranium Electrode Pour

FIGURE 42. Beta Spectrum at 1 m from a Charge Makeup Crucible

FIGURE 43. Beta Spectrum at 1.9 cm from a Charge Makeup Crucible

FIGURE 44. Corrected Beta Spectrum at 1.9 cm from a Charge Makeup Crucible

of electrons and secondary photons due to Compton scattering in the plastic filter plus the uncollided photons incident upon the detector can exceed the number of photons that would impact the detector in the absence of the plastic filter. Thus, there are too many counts in the gamma spectrum. When the subtraction is made, the remaining "beta" spectrum has too few counts. This effect is seen in Figure 42, but Figure 43 is a more extreme example. If it were not for the effect described above, Figure 43 would look similar to Figure 44. Both the pulse pileup and photon spectra subtraction problems tend to raise the calculated average beta energies and lower the calculated dose. In most cases, the combined effect is no more than a few percent of the total. However, when the beta-to-gamma ratio is low, as in Figure 43, the effect is more significant.

A gamma spectrometer was used to aid in the interpretation of the beta spectra. The spectra generated with this system are shown in Figures 45 through 48. Significant gamma peaks are seen at approximately 90 keV, 770 keV, and 1000 keV.

FIGURE 45. Germanium Gamma Spectrum for an End Cut from a Uranium Billet

FIGURE 46. Germanium Gamma Spectrum for a Fresh Cast Electrode

FIGURE 47. Germanium Gamma Spectrum for an Arc Melt Cast

FIGURE 48. Germanium Gamma Spectrum for a Charge Makeup Crucible

4.1.5 Site E: Uranium Facility

Site E is a uranium facility that fabricates depleted uranium metal products. Field measurements were conducted at this site twice in 1985.

For the first set of measurements, a small carbon crucible, a pallet with hollow metal fuel elements, and 5-gal buckets containing green salt and pressed uranium metal briquettes were placed in a laboratory. Field spectra were obtained by placing a plastic scintillator at a distance of 10 to 25 cm from the various samples. Plastic filters of 5 up to 860 mg/cm^2 were used to obtain gamma spectra for subtraction from the total radiation spectra. The beta spectra (Figures 49 through 52) for Site E measurements are very similar to those just described for Site D. The only noticeable difference is the lack of pulse pileup problems in the first set of measurements due to lower dose rates. The distortion of the low-energy portion of the beta spectra by photon subtraction is again noted, but it is less severe than at Site D.

FIGURE 49. Beta Spectrum 15 cm Above a Uranium Crucible

FIGURE 50. Beta Spectrum for Green Salt

FIGURE 51. Beta Spectrum for Briquettes

FIGURE 52. Beta Spectrum for Hollow Fuel Rods

During a second visit to Site E, beta spectra of a crucible, a contaminated glove, a fuel element, and a smear from a uranium ingot were obtained (Figures 53 through 58). These spectra are noticeably different from the earlier spectra: there is a prominent peak below 200 keV. This is a beta peak and it is caused by the close proximity of the scintillation detector (0.6 cm) to the sources of radiation. Also, the covering on the scintillator was very thin (less than 2 mg/cm^2). This allowed the beta particles from ^{234}Th ($E_{\text{max}} = 101$ and 193 keV) to be detected by the scintillator. These low-energy beta particles were not evident in previous spectra due to the usual separation between the detector and the source: 15 to 25 cm of air, in combination with a thicker detector window, which stopped all ^{234}Th beta particles. The detector window plus 15 cm of air will stop all beta particles below 130 keV.

4.1.6 Site F: Laboratory and Fuel Processing Facility

The field spectra obtained at Site F were based on two separate measurement configurations. The first was taken in a laboratory where a soil sample containing fission products had been placed. The second set of measurements was taken at an opening designed for sampling in a spent fuel processing facility.

FIGURE 53. Beta Spectrum for a Graphite Crucible at 0.6 cm

FIGURE 54. Beta Spectrum for the Outside of a Used Glove

FIGURE 55. Beta Spectrum for the Inside of a Used Glove

FIGURE 56. Beta Spectrum for a Briquette at 0.6 cm from the Detector

FIGURE 57. Beta Spectrum for an Ingot Smear

FIGURE 58. Beta Spectrum for a Fuel Element at 0.6 cm from the Detector

Figure 59 shows the beta spectrum taken with a plastic scintillator at 30 cm from the sample. This spectrum shows a strong influence from ^{90}Y . The gamma spectrum (Figure 60) reveals that ^{137}Cs was also present based on the Compton edge near 500 keV.

The spectra taken at the sample opening also indicated the presence of ^{137}Cs and ^{90}Y (Figures 61 and 62). Due to the attenuation of the beta particles by air and the material covering the blister port opening, the ^{137}Cs gamma component is more evident than the beta component.

4.1.7 Site G: Plutonium Production Reactor

The final spectral measurements were taken at Site G, a plutonium production reactor. Used fuel-tube caps had been placed in a rack in a low-background area of the building that housed the reactor. A plastic scintillation detector was suspended at 8-1/4 cm above the midpoint of an array of 21 fuel-tube caps. Spectra were taken with and without a 1000-mg/cm² plastic cover on the detector. The spectra, shown in Figures 64 and 65, indicate that the radiation field was largely due to ^{60}Co contamination.

FIGURE 59. Beta Spectrum for a Contaminated Soil Sample

FIGURE 60. Gamma Spectrum for a Contaminated Soil Sample

FIGURE 61. Beta Spectrum at 60 cm from a Blister Port

FIGURE 62. Gamma Spectrum at 60 cm from a Blister Port

FIGURE 63. Beta-Plus-Gamma Spectrum for Fuel-Tube Caps

FIGURE 64. Beta Spectrum for Fuel-Tube Caps

4.2 DOSE RATE MEASUREMENTS

4.2.1 Site A: Separation Facility

At Site A, beta dose rates and doses were obtained from the spectrometer, survey meter, and eight-element dosimeter for each measurement location except that the waste sample spectrum for the second visit did not provide a useful beta dose; any beta dose was completely masked by variations in the gamma component. The results are shown in Table 8. There is good agreement between the dose rates derived from the three devices for the first process sample, especially at 60 and 90 cm. Because of the hard beta spectra produced by encasing the sources in glass bottles, the spectral doses were not corrected for low-energy backscatter (Rathbun et al. 1984) and a small correction factor (1.2) (Table 5) was used for the survey meter. However, a correction factor of 1.33 (Section 3.1.4) was used for the spectral dose when the process sample data from the second visit were analyzed. This was based on the very low attenuation of beta particles provided by the gauze pad that contained the sample. For the same reason, a correction factor of 2.5 (Section 3.4) was

TABLE 8. Site A Beta Dose Rates Measured with Spectrometers and Survey Meters

Radiation Source	Spectral Analysis, mrad/h	Survey Meter A, mrad/h	PNL Eight-Element, Dosimeter mrad/h	Beta:Gamma Ratios	
				Survey Meter	Eight-Element Dosimeter
Process Sample ^(a)					
at 30 cm	610	840	730	34:1	38:1
at 60 cm	130	130	120	22:1	21:1
at 90 cm	50	40	50	13:1	23:1
Waste Sample ^(a)					
at 30 cm	270	360	110	12:1	11:1
at 60 cm	100	120	50	24:1	13:1
at 90 cm	30	40	20	13:1	4:1
Process Sample ^(b)					
at 30 cm	360	440	580	8:1	120:1

(a) Samples analyzed during first trip. Samples were in glass bottles.

(b) Sample analyzed during second trip. Sample was on gauze pad.

used for the survey meter. Even with these corrections, the results from the survey meter and spectral measurements were low during the second trip relative to results from the eight-element dosimeter.

The beta doses shown for the first process sample decrease with distance as expected based on an inverse distance squared relationship (plus air attenuation). The relationship of dose versus distance for the waste sample in Table 8 is not so straightforward. Although the doses at 90 cm versus those at 60 cm follow the expected inverse distance squared (plus attenuation) relationship, the doses at 30 cm appear to be too low relative to those at 60 and 90 cm. The same correction factors used for the first process sample were used for the waste sample, based on a similar configuration. This caused the spectral doses to be approximately 75 to 85% as high as the survey meter doses. The eight-element dosimeter doses were only 30 to 50% of the survey meter doses. It is not clear why this should be the case; it does not follow the trend of the process samples.

Gamma doses were also determined for each sample with a survey meter and an eight-element dosimeter. These gamma doses generally follow an inverse distance squared relationship with distance except for the eight-element dosimeter with the waste sample. In the latter case the gamma doses at 30 and 60 cm appear to be too low.

Beta-to-gamma ratios were also determined at a distance of 30 cm with the survey meter and eight-element dosimeter. These results are listed in Table 8. All the beta-to-gamma ratios are high, but the eight-element dosimeter beta-to-gamma ratio is much higher than that of the survey meter during the second visit.

4.2.2 Site B: Fuel Fabrication Facility

The results of measurements with the three survey meters, the scintillation detector, and the multielement dosimeters are shown in Tables 9 and 10. The multielement dosimeters and scintillation detector may be assumed to provide good estimates ($\pm 50\%$) of the contact beta dose rates for the fuel elements. Although contact beta readings are also listed for the survey meters there is no direct correspondence between the contact survey meter readings and the actual contact dose. Although the window of the chamber is in contact with the source, the sensitive volume of the chamber is separated from the source. Meter readings represent the average ionization throughout the sensitive volume. Hence, the survey meter readings underestimate the beta dose in close proximity to the source, unless a correction factor is applied. Based on previous experience at PNL (Table 7), a correction factor of 5.3 was used with the 7-cm-dia fuel element readings to determine contact doses with the survey meters. Even with this adjustment, the beta contact doses and total doses measured with the survey meters were low relative to the corresponding values for the multielement detectors. The readings for the 208-cm³ CP were especially low, but they were still within 50% of the multielement contact doses. A factor of 1.2 was used for all distances other than contact, based on the same rationale discussed for Site A. It is apparent from the preceding tables that the whole body dose rate in the vicinity of a fuel element (not in contact with it) is typically no more than 1 mrad/h. This is borne out by data from other multielement dosimeters placed in various work

TABLE 9. Site B Dose Rates from a 0.95%-Enriched Fuel Element, mrad/h^(a)

Instrument	Open Window	Closed Window	Gamma	Beta	Total
Ion Chamber "A" Contact	2.5	1.9	1.9	3.2	5.1
15 cm	0.8	0.5	0.5	0.4	0.9
30 cm	0.5	0.3	0.3	0.2	0.5
60 cm	0.3	0.2	0.2	0.1	0.3
CP "B" (208 cm ³) Contact	2.1	1.6	1.6	2.6	4.2
15 cm	0.7	0.5	0.5	0.2	0.7
30 cm	0.45	0.3	0.3	0.2	0.5
60 cm	0.3	0.25	0.25	0.1	0.35
CP "C" (463 cm ³) Contact	2.9	1.9	1.9	5.3	7.2
15 cm	1.0	0.85	0.85	0.2	1.0
30 cm	0.55	0.5	0.5	0.1	0.6
60 cm	0.35	0.3	0.3	0.1	0.4
Plastic Scintillator (E _{ave} = 550 keV)					3.6
Multielement Dosimeter Contact	ND ^(b)	ND	3.0	5.2	8.2

(a) TLD statistical variation on gamma readings are ± 10 to 20%, and on beta reading $\pm 50\%$ for the above data.

(b) ND = not determined.

TABLE 10. Site B Dose Rates from a 1.25%-Enriched Fuel Element, mrad/h^(a)

Instrument	Open Window	Closed Window	Gamma	Beta	Total
Ion Chamber "A" Contact	3.1	1.85	1.85	6.6	8.45
15 cm	0.5	0.25	0.25	0.4	0.65
30 cm	0.1	0.0	0.0	0.1	0.1
CP "B" (208 cm ³) Contact	2.3	1.6	1.6	3.7	5.3
15 cm	0.75	0.5	0.5	0.4	0.9
30 cm	0.4	0.25	0.25	0.2	0.45
45 cm	0.2	0.15	0.15	0.1	0.25
CP "C" (463 cm ³) Contact	2.5	1.5	1.5	5.3	6.8
15 cm	0.8	0.5	0.5	0.4	0.9
30 cm	0.4	0.3	0.3	0.1	0.4
60 cm	0.25	0.1	0.1	0.2	0.3
Plastic Scintillator (E _{ave} = 400 keV)					5.8
Multielement Dosimeter Contact	ND ^(b)	ND	3.2	5.4	8.6

(a) TLD statistical variation on gamma readings are ± 10 to 20%, and on beta reading $\pm 50\%$ for the above data.

(b) ND = not determined.

locations in the fuel fabrication building (Table 11). The maximum exposure for a person standing at location 1 (intermittent fuel storage area) for 8 h/day could not exceed 1600 mrad for the year. If it were not for the zirconium cladding (2 mm) on the fuel elements, beta doses would be much higher.

TABLE 11. Dose Rate in Various Locations in the Uranium Fuel Fabrication Facility (Site B)

Location	Description	Dose Rate, mrad/h		
		Beta	Gamma	Total
1	Intermittent Fuel Storage Area	0.19	0.61	0.80
2	Inspection Station	0.17	0.21	0.38
3	Sandblast Cabinet	ND ^(a)	0.02	ND
4	Ultrasonic Testing	0.10	0.13	0.23
5	Near Autoclaves	ND	0.11	ND
6	Northwest Entrance	ND	0.03	ND

(a) ND = not determined.

4.2.3 Site C: Separations Facility

After the lid was removed from each waste tank sample at Site C, survey meter readings were taken with a CP instrument. The beta dose rates were obtained by subtracting the closed window reading from the open window reading and multiplying by a factor of 2.5. This typical factor should be adequate, provided only beta particles from inside the sample bottle were detected. The measurement of contamination on the outside of the bottle would require a larger calibration factor due to the close proximity of the source to the detector. The survey meter dose rates are listed along with spectral dose rates, in Table 12. In three out of four liquid sample cases compared, the spectral dose rates were larger than the CP dose rates. Also, a small portion of the spectrum near the origin was lost, due to the discriminator setting. Actual beta dose rates may be even larger than those listed based on a correction factor of 1.33.

TABLE 12. Site CDose Rates for Waste Tank Samples Measured with CP "C", mrad/h

Sample	Survey Meter		Beta = Difference x 2.5	Spectral Beta Dose, mrad/h	Spectral Average Energy
	Open Window Reading, mrad/h	Closed Window Reading, mrad/h			
105AW	30	10	50	130	980
107AN	700	90	1525	1140	1400
105AN	180	90	235	310	650
102AZ	200	80	300	ND ^(a)	1000
104AN272	50	15	87	270	420
102AW	800	100	1750	ND	750

(a) ND = not determined.

It is known from gamma analyses of the samples that the primary nuclide was ^{137}Cs . The 32-keV x rays from the ^{132}Cs decay product (^{137}Ba) would be attenuated more by the 1000-mg/cm^2 covering on the scintillation detector than the 400-mg/cm^2 beta window of the survey meter. However, in this case it would change the reported beta doses by only a few percent, at the most. At Site A, where similar samples were measured with greater attenuation, the survey meter dose rates were larger than the spectral dose rates.

4.2.4 Site D: Uranium Foundry

At Site D, three of the site's routine survey meters were used for comparison with spectrometer results. These survey meters are typical of those used at other DOE facilities. Many of the survey meter and spectrometer dose rates (Table 13) exceeded 400 mrad/h with typical correction factors of 2.5 for survey meters and 1.33 for the spectrometer.

The published value for a uranium slab, with a 7-mg/cm^2 cover, is about 230 mrad/h (Shleien and Terpilak 1984). This difference can be explained by the nonuniform deposition of uranium decay products. Apparently the short-lived progeny of ^{238}U and ^{235}U were depleted from the center of a molten mass of uranium and concentrated at the ends by chemical/physical processes. This resulted in a very high beta-to-gamma ratio and little attenuation for the

TABLE 13. Site D Beta Dose Rates Measured with Spectrometers and Survey Meters

Radiation Source	Spectral Analysis, mrad/h	Survey Meters			PNL Eight-Element, mrad/h	Beta-to-Gamma Ratios	
		A, mrad/h	D, mrad/h	E, mrad/h		Survey Meter	Eight-Element Dosimeter
End Cut from Uranium Billet	390	400	560	ND ^(a)	150	13:1	26:1
Top of Uranium Electrode Pour	560	1200	ND	ND	790	10:1	84:1
Uranium Electrode Near Midpoint	90	280	200	130	ND	12:1	ND
18 in. Above Open Uranium Crucible	490	ND	530	ND	ND	10:1	ND

(a) ND = not determined.

surface material. However, the beta-to-gamma ratios show considerable variation both between the survey meter and the dosimeter and between the different locations. Much of this variation can be attributed to the position and size of the source relative to the measurement device.

The data in Table 13 result mostly from measurements taken at the surface of uranium metal. The uranium crucible is an exception. Also, the survey meters, dosimeters, and spectrometer were not placed at exactly the same position relative to the uranium metal. With these variables considered, the agreement among the results is reasonably good.

4.2.5 Site E: Uranium Facility

As at Site D, Site E exposures were also due to uranium. However, measurements made during the first visit to Site E (Table 14) revealed much lower dose rates than those recorded at Site D. This was partially due to a greater distance between the detectors and uranium surface (greater than 15 cm) during the first measurements at Site E. In addition, the green salt at Site E did not exhibit an increased concentration of uranium decay products at the surface. The residual uranium decay products on the inner surface of the crucible had apparently had time to decay. The measurement at 1.3 cm yielded about the same dose rate as the measurement at 15 cm.

TABLE 14. First Site E Beta Dose Rates Measured with Spectrometers and Survey Meters

Radiation Source	Spectral Analysis, mrad/h	Survey Meters		PNL Eight-Element, mrad/h	Beta-to-Gamma Ratios	
		D, mrad/h	E, mrad/h		Survey Meter	Eight-Element Dosimeter
15 cm Above Open Uranium Crucible	80	50	90	80	ND ^(a)	160:1
Green Salt	50	40	60	55	ND	70:1
Briquettes	110	70	120	110	35:1	60:1
Hollow Fuel	130	90	180	155	24:1	40:1
1.3 cm Above Inner Surface of Crucible	80	60	150	ND	ND	ND

(a) ND = not determined.

The new thin scintillator prototype survey meter (survey meter E) used at Site E seemed to be very consistent and accurate. It measured slightly higher dose rates than the other devices, as one would expect. Because the thin scintillator is only 5-mg/cm² thick and is nearly tissue equivalent, it is a good approximation of the outer layer of human skin. Thus, measurements made with the thin scintillator can closely approximate skin doses.

The second set of measurements at Site E produced much higher dose rates (Table 15). The measurements in this case were all taken at 0.6 cm from the surface of the uranium and/or decay products. The fuel element dose rate was close to that expected of a uranium slab; the briquette dose rate was similar to those seen at Site D; and the ingot wipe and inner glove surface dose rates were even higher. The crucible and outer glove surface dose rates approached those one would expect for an infinite plane source of ²³⁴Th-^{234m}Pa (Cross et al. 1982).

TABLE 15. Second Site E Beta Dose Rates Measured with Spectrometer and Survey Meter

	Spectral Dose Rate, ^(a) mrad/h	Average Energy, keV	Survey Instrument C, mrad/h	Beta-to-Gamma Ratio
Graphite Crucible				
No filtration	2320	827	2090	14:1
20 mg/cm ²	2070	776		
Used Glove (outside surface)				
No filtration	2448	867	ND ^(b)	ND
20 mg/cm ²	2110	833		
Used Glove (inside surface)				
No filtration	835	638	680	18:1
20 mg/cm ²	620	623		
Briquette				
No filtration	402	770	280	13:1
20 mg/cm ²	370	730		
Wipe from Ingot				
No filtration	876	642	370	15:1
20 mg/cm ²	697	663		
Fuel Element				
No filtration	200	580	90	6:1
20 mg/cm ²	164	560		

(a) Dose rates calculated at 0.6 cm from surface of material based on beta radiation spectra for beta energies greater than 70 keV.

(b) ND = not determined.

4.2.6 Site F: Laboratory and Fuel Processing Facility

Dose rate measurements at Site F were taken in cooperation with a team of non-PNL scientific investigators. Measurements were made in two different general locations using three commercial survey meters and two similar plastic scintillator spectrometers. The measurement results are shown in Table 16. It is important to note that for these measurements, the center of the survey instrument (rather than the face) was used as the reference point. For the particular survey meters used, this allowed the beta dose to be read directly from the instruments rather than calculated with correction factors. With

TABLE 16. Beta Measurements at Site F

	Beta Dose Rate		Survey Meters					
	Scint. #1, mrad/h	Scint. #2, mrad/h	A		B		E	
			Beta, mrad/h	Gamma, mrad/h	Beta, mrad/h	Gamma, mrad/h	Beta, mrad/h	Gamma, mrad/h
Soil in HP Laboratory	69	67	58	7	68	0.5	64	0.7
60 cm from Blister Port, Directly in Front	126	190	150	-	160	6	-	-

respect to the contaminated soil, the results show very good agreement for the various measurement devices and good accuracy if the extrapolation chamber result is taken as the true dose.

The results for the sampling port measurements (Table 16) are much more scattered. The position of the instruments relative to one another may have been an important factor here. Large changes in readings were noted with a vertical change of 5 cm. Extrapolation chamber measurements at the sampling port are not available due to an equipment malfunction.

4.2.7 Site G: Plutonium Production Reactor

At Site G, only one survey instrument, "G", was used. With the center of the instrument at 11.4 cm above the array of fuel-tube caps, it read a beta dose rate of 390 mrad/h and a gamma dose rate of 32 mrad/h. This compares to a beta dose rate of 230 mrad/h derived from the scintillation detector spectrum with the face of the detector at 8.3 cm above the array of fuel-tube caps. The average energy was 120 keV. Under these conditions, the scintillation detector received beta particles from four fuel-tube caps, while the survey instrument received additional beta particles from fuel-tube caps that were out to the side of the four directly below the instrument chamber.

4.3 SUMMARY OF MEASUREMENT RESULTS

The following sections summarize the dosimeter and instrument measurements.

4.3.1 Dosimeter Results

Commercial and prototype dosimeters representative of those used at DOE facilities were placed at or near locations where radiation fields were measured with other devices. These dosimeter results along with those for the PNL eight-element dosimeters are listed in Table 16. The data in Table 16 result mostly from dosimeters placed at various distances from some form of uranium. The exceptions occurred at Sites A and F; these are discussed last.

At sites where doses were from uranium, the eight-element dosimeters generally recorded higher dose rates than the other dosimeters. A notable departure from this trend occurred from a measurement of the end cut from a uranium billet at Site D. Although there was a variation in dose rate at different points on the billet end cut, the variation was not as high as indicated by the dosimeter results. In addition, a survey meter reading taken where the eight-element dosimeter was placed indicated that the location had a higher dose rate than other points on the surface. Thus, the result for the eight-element dosimeter must be accepted as anomalous in this case. At Site D, the most accurate dosimeter results (compared to the eight-element results) were provided by a PNL two-element dosimeter containing either 17-mg/cm² or 235-mg/cm² TLDs. A commercial four-element dosimeter with a modification of the thinnest filter and a revised algorithm was the most accurate at Site E, followed by the two-element PNL dosimeter with 235-mg/cm² TLDs. At Sites D and E, the two-element PNL dosimeter with 100-mg/cm² TLDs responded similarly to the same dosimeter with 235-mg/cm² TLDs, but the thinner TLD results were more erratic. The second commercial four-element dosimeter and the commercial two-element dosimeters tended to have low responses (approximately one-half) relative to the eight-element results. At Site F, only the two-element dosimeters with 100-mg/cm² and 235-mg/cm² TLDs were exposed at the same point as the eight-element dosimeter; they had the low response just discussed. At Site A, the 235-mg/cm² TLD response was

TABLE 17. Response of Dosimeters to Beta Radiation at DOE Facilities

Site	Location	PNL Eight- Element, mrad/h	Two-Element 17 mg/cm ² , mrad/h	Two-Element 100 mg/cm ² , mrad/h	Two-Element 235 mg/cm ² , mrad/h	Four-Element Commercial, mrad/h ^(a)	Four-Element Commercial, mrad/h	Two-Element Commercial, mrad/h
A	Process Sample	580.0	430.0	380.0	920.0	ND ^(a)	50.0	230.0
D	End Cut from Uranium Billet	150.0	390.0	480.0	510.0	ND	200.0	280.0
	Top of Uranium Electrode Pour	790.0	270.0	160.0	740.0	ND	ND	340.0
	Charge Makeup at 1 Meter	1.1	0.9	0.7	0.7	ND	0.5	0.4
	General Area #1	0.5	0.4	0.4	0.4	ND	0.2	0.2
	General Area #2	4.8	2.4	2.3	2.5	ND	2.5	1.6
	General Area #3	2.9	1.4	1.4	1.0	ND	1.0	0.6
	General Area #4	0.6	0.6	0.5	1.0	ND	0.1	0.3
	General Area #5	5.9	1.0	1.6	3.4	ND	0.8	0.5
	General Area #6	1.0	1.2	0.9	0.9	ND	0.6	0.6
E	15 cm Above Open Uranium Crucible	80.0	ND	80.0	65.0	65.0	11.0	31.0
	Green Salt	55.0	ND	45.0	45.0	50.0	28.0	27.0
	Briquettes	110.0	ND	ND	ND	85.0	50.0	62.0
	Hollow Fuel Rods	155.0	165.0	220.0	160.0	120.0	65.0	66.0
F	Contaminated Soil	92.0	ND	ND	38.0	ND	43.0	ND

(a) ND = not determined.

abnormally high, the four-element commercial dosimeter response was very low, and the other dosimeters followed the trend described for Sites D and E.

4.3.2 Instrument Results

In addition to measurements made with the silicon diode and scintillation spectrometer, commercial and prototype instruments were compared with spectral measurements when available. Selected measurements are summarized in Table 18 and discussed in more detail in the sections on the specific site measurements. Table 18 also shows the results for the eight-element dosimeter when measurements were made in the same location.

As one can see from the table, the measurements made with survey meters tend to be higher than dose rates determined by analysis of the spectral data. Ion chamber instruments (A, B, and C) show reasonable agreement with spectral data, particularly in the high-energy beta exposure conditions (Sites D, E, and F). The developmental thin scintillator (Johnson et al. 1983) showed good agreement with the spectral analysis results in the limited testing performed.

TABLE 18. Response of Instruments to Beta Radiation at Department of Energy Facility

Site	Location	Spectral Analysis, mrad/h	Survey Meter Readings, mrad/h					Eight- Element Dosimeter, mrad/h
			A ^(a)	B ^(a)	C ^(a)	D ^(b)	E ^(c)	
A	Process Sample	610	840	ND ^(d)	ND	ND	ND	730
A	Waste Sample	270	360					110
B	Fuel Element (contact) ^(e)	3.6	5.1	4.2	7.2	ND	ND	8.2
B	Fuel Element (contact)	5.8	8.5	5.3	6.8	ND	ND	8.6
C	Waste Sample	130	ND	NO	50	ND	ND	ND
C	Waste Sample	270	ND	ND	87	ND	ND	ND
D	Uranium Billet	390	400	ND	ND	560	ND	150
D	Uranium Electrode	90	280	ND	ND	200	130	ND
E	Uranium Crucible	80	ND	ND	ND	50	90	80
E	Hollow Fuel Element	130	ND	ND	ND	90	180	155
F	Contaminated Soil	69	58	68	ND	ND	64	ND
F	Blister Port	126	150	160	ND	ND	ND	ND

(a) Ion chamber instruments.

(b) Geiger-Mueller detector and count rate meter.

(c) Thin scintillation detector survey meter.

(d) ND = not determined.

(e) Readings are total dose at contact.

5.0 CONCLUSIONS AND RECOMMENDATIONS

5.1 BETA EXPOSURES

Seven different DOE facilities were visited during the course of the field measurement described in this report. This represents one-fourth of the DOE facilities that have beta measurement programs (Mulvehill et al. 1987). Five of these facilities were selected based on the probability that beta radiation dose rates exceeded photon dose rates. Thus, the results reported may be biased towards higher beta dose rates than typically experienced at DOE facilities.

In laboratories or field conditions where large quantities of unshielded material were present, high beta-to-gamma ratios were observed. The beta dose rates under these conditions were often hundreds of mrad per hour, and thus, beta spectra were easy to observe and analyze. Where fission products were found, or where uranium was shielded in the actual work locations, the resulting low beta-to-gamma ratios made interpretation of beta spectra difficult. Low-energy (less than 200 keV) beta fields caused additional difficulties due to the nonlinear energy response of beta spectrometers.

Analyses of beta and gamma spectra showed that ^{234}Th - $^{234\text{m}}\text{Pa}$, ^{231}Th , ^{137}Cs , and $^{90}\text{Sr}/^{90}\text{Y}$ were the major nuclides contributing to beta doses. In addition to ^{137}Cs and $^{90}\text{Sr}/^{90}\text{Y}$, other fission products and ^{60}Co were present in smaller quantities.

At facilities where mixed fission products were present, handling and control procedures developed to control internal exposures and exposure to penetrating radiations were also effective in controlling beta dose rates to workers. At facilities where large amounts of uranium were handled (^{234}Th - $^{234\text{m}}\text{Pa}$ and ^{231}Th) in the metal form the beta dose rates were higher. Even though the contact dose rates from fission product samples often exceeded the contact dose rates from uranium decay products, the opportunity for routine contact with the uranium decay products appeared to be much greater.

Workers exposed to mixed fission product fields typically wore at least two pairs of gloves taped to full-length outer protective clothing. Eye protection was usually available and small samples were handled in fume hoods

or glove boxes. In larger enclosures where fission products existed, workers wore full-face masks. The use of long-handled tongs and lead containers also reduced exposures to fission products. Existing equipment and procedures for personnel contamination monitoring at facilities where fission products were present minimized the opportunity for undetected skin contamination.

At facilities where short-lived uranium decay products were the major source of beta exposure, workers typically wore heavy leather work gloves but no inner gloves or tape. These workers often stood close to large pieces of cast uranium metal, furnaces, and graphite crucibles. The surfaces of the uranium and the surfaces of associated processing and handling equipment were typically covered with a black film containing carbon from the crucibles and uranium highly enriched in ^{234}Th - $^{234\text{m}}\text{Pa}$ and ^{231}Th . When uranium is melted, the thorium that has grown in from ^{238}U and ^{235}U apparently moves preferentially to the outside of the molten solution and remains there while the metal resolidifies. This results in a thin film of much higher beta activity than the typical uranium slab, in which the uranium decay products are homogeneously mixed with the uranium. Because ^{231}Th has a half-life of approximately one day, it will virtually decay away after one week. However, ^{234}Th (half-life 24 days) will continue to provide significant beta emissions for up to six months. Protactinium-234m is always in equilibrium with ^{234}Th . Protactinium-234m has a 2.3-MeV beta particle (maximum energy) that can penetrate several meters of air.

The other beta emitters associated with the immediate decay products of uranium are important only for skin contact doses and internal doses. Rapid accumulation of black, uranium decay product dust on the outside of workers' gloves (worn for physical protection) and the opportunity for the interior of the gloves to become contaminated made dose rates to the hands potentially large. Dose rates of greater than 800 mrad/h were measured on the inside of one glove (see Table 14).

5.2 SURVEY METERS

Beta dose rates measured by five different commercial models of survey meters were typically higher than the accepted values when the face of the instrument was taken as the reference point and a carefully considered correction factor was applied. For a hardened beta spectrum (^{90}Y) a correction factor near unity was adequate. However, typical correction factors of 2.5 were often not large enough for CP-type instruments used to measure lightly filtered beta fields.

The best survey meter results for the commercial instruments were obtained when the center of the instrument was taken as the reference point. Because of the availability of spectral information and the opportunity for careful analyses, the correction factors used in this report were probably more accurate than a single correction factor that might be applied at a typical DOE facility.

The prototype beta-plus-gamma dose rate meter developed at INEL was used briefly at Site E. Based on data from this limited application, the prototype survey instrument appeared to produce very good results.

5.3 DOSIMETERS

Typical dosimeters usually had a lower response to beta radiation than the PNL eight-element dosimeter. Two commercial dosimeters (two-element and four-element) responded approximately 50% less than the PNL eight-element dosimeter. However, a second four-element commercial dosimeter and PNL-prototype two-element dosimeters containing three different thicknesses of TLO typically responded within 70 to 80% of the eight-element dosimeter. A thinner filter on the "beta" (PNL prototype) dosimeter and, in the case of the four-element dosimeter, an improved algorithm appeared to cause the better response. The dosimeter containing the KSU thin (17-mg/cm^2) TLO responded equally to the same type of dosimeter containing 235-mg/cm^2 TLDs. This indicates that most beta fields the thin TLDs were exposed to consisted of higher energy beta particles.

In some cases, the results from the PNL eight-element dosimeter appeared to be erratic. The influence of low-energy photons may have caused these erratic results. Because the eight-element dosimeter was used to compare other dosimeter results, any deviation in the eight-element dosimeters results affected the evaluation of the other dosimeters.

5.4 RECOMMENDATIONS

Based on the measurements made at the DOE facilities and observations during the measurements, several recommendations can be made. These recommendations are listed below.

- Further studies of detector response at energies below 200 keV should be undertaken to determine the degree of detector nonlinearity and its influence on dose calculations.
- Studies are needed on the influence of backscatter from detectors and detector windows on beta spectra and beta dose calculations.
- Further studies should be performed on the eight-element dosimeters to determine their response in mixed fields dominated by low-energy photons.
- A dosimeter similar to the eight-element unit, but using stacked elements should be developed. This would eliminate the concern about the spatial response of the present multielement dosimeter and permit the study of spatial variations in radiation fields.
- Dosimeters used in beta fields at DOE facilities should have a thin window (less than 20 mg/cm^2) over the beta element.
- Conventional 235 mg/cm^2 TLDs should be replaced by thin TLDs (17 mg/cm^2) in dosimeters used to monitor beta exposures.
- Dosimeters should be calibrated to beta energies similar to those encountered in the work areas. Yttrium-90 calibrations provide too energetic a beta spectrum for many applications.

- The influence of source geometry and source-detector distance on measurements should be known for each instrument used in a facility. Whenever possible, measurements should be made to determine the correction factors.
- Personnel should be trained to use correction factors for source energy, source geometry, and source-to-detector distance on dose rate measurements.
- The variation in response of instruments with beta energy and angle of incidence should be documented.
- A modified ionization chamber instrument with a large-area, shallow chamber should be available for beta-dose-rate measurements.
- The thin scintillator should be further developed for field measurements.
- Workers who handle uranium metal products and associated processing materials should be made aware of the potentially high contact beta dose rates.
- Radiation protection procedures should be improved for workers in facilities where uranium decay products present beta dose problems. Contamination control, protective clothing, and dosimeter placement should be addressed.

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