LA-6849-PR

Progress Report

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**Nuclear Safeguards
Research and Development
Program Status Report
January—April 1977**

Issued: August 1977

**los alamos
scientific laboratory**

of the University of California

LOS ALAMOS, NEW MEXICO 87545

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Compiled by

Joseph L. Sapir

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NUCLEAR SAFEGUARDS RESEARCH AND DEVELOPMENT

Program Status Report

January--April 1977

ABSTRACT

This report presents the status of the Nuclear Safeguards Research and Development program pursued by LASL Safeguards Groups Q-1, Q-2, Q-3, and Q-4. Topics covered include nondestructive assay technology development and applications, international safeguards, perimeter safeguards and surveillance, concepts and subsystems development (e.g., DYMAC program), integrated safeguards systems, training courses, and technology transfer.

Salient features of the program and technical progress during the current reporting period are outlined in the Executive Summary.

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NUCLEAR SAFEGUARDS RESEARCH AND DEVELOPMENT

EXECUTIVE SUMMARY

G. Robert Keepin
Director of Nuclear Safeguards Programs

William H. Chambers
Assistant Division Leader for Nuclear Safeguards

On February 1, 1977, R Division merged with several other LASL groups concerned with energy R&D (both nuclear and nonnuclear) to form a new Energy (Q) Division. Concurrently, the Nuclear Safeguards program was organized into four technical groups, Q-1, Q-2, Q-3, and Q-4 which report to the Q-Division Office for Nuclear Safeguards (Q-DO/SG), headed by the Director of Nuclear Safeguards Programs and the Assistant Division Leader for Nuclear Safeguards. This reorganization of the Nuclear Safeguards program focuses detailed management on specific programmatic projects and tasks while providing overall direction and coordination of the wide-ranging activities and resources within the LASL safeguards program. This structure provides the program management flexibility and "fast response" capability required by our Washington program offices—ERDA/DSS and /DMA.

This program status report is divided into four parts, one for each of the technical groups. Each part begins with a brief summary of the Group's function and major programmatic efforts.

During the present reporting period we pursued several important activities in support of IAEA safeguards. An electronic system to accompany the IAEA portable High Level Neutron Coincidence Counter (HLNCC) was designed, breadboarded, and successfully tested. Three of these systems are being packaged, and the first unit should be delivered to the IAEA in July 1977. Since the fill height of materials assayed in the HLNCC is often unknown, the counting efficiency should be insensitive to height variations. It was determined that 8-

cm-thick polyethylene end plugs significantly flatten the axial efficiency, and these plugs will be included with the HLNCC units being supplied to the IAEA.

Because the IAEA has limited facilities in which to perform NDA measurements, particularly when plutonium is involved, LASL periodically assists the IAEA in obtaining the appropriate materials and performing the required measurements. During this reporting period, LASL assisted IAEA staff members in developing and evaluating NDA techniques for inspections at critical assembly and power reactor installations.

Investigations related to absorption-edge densitometry were pursued on a broad front. The Can Scan, an instrument that adapts this technique to the assay of cans containing plutonium-bearing ash, was designed and fabricated. This instrument has thus far been used to assay cans containing 2 to 75 g of plutonium. An extrapolation method was developed which reduces the matrix sensitivity of uranium assays obtained using K-edge densitometry for high-Z (>20) matrix contaminants. The hardware for an x-ray generator-based L_{III} absorption-edge densitometry assay station was designed and constructed. This station will form the basis for the development of an in-line solution assay system which will measure uranium and plutonium concentrations from 1 to 50 g/l. Studies were initiated to investigate densitometry techniques for assaying special nuclear material (SNM) in highly radioactive solutions. To overcome the high radiation without resorting to excessively intense transmission sources, an x-ray filtering technique

incorporating a Bragg crystal and germanium detector looks promising.

A fourth course entitled "Integrated Safeguards Systems—Concepts and Implementation" was introduced into the USERDA Safeguards Technology Training Program curriculum during the week of March 21, 1977. The three-day course, conducted at LASL by individuals from the four LASL safeguards groups and from Sandia Laboratories, was attended by 35 representatives of the safeguards community.

In the area of perimeter safeguards, evaluation of the Tom Scurry PM-203 portal monitor was completed and the recommendation made that it be approved for ERDA use. This is the fourth commercial monitor that LASL has evaluated and approved. Performance testing of the Direct Rate Monitor (DRM), a new prototype hand-held monitor, is under way. Because its detection sensitivity was designed to be essentially independent of background, the DRM is especially useful in areas having higher-than-normal background radiation levels.

The Dynamic Materials Control (DYMAC) program is proceeding on schedule. During this reporting period the test and evaluation program at the DP-Site plutonium facility (DYMAC, Phase I) was completed. This provided the in-plant working experience that was a necessary prerequisite to installation of the complete DYMAC system at the new TA-55 facility (Phase II). A number of NDA in-

struments, including a thermal-neutron well coincidence counter, a passive gamma-ray plutonium solution assay instrument, load-cell weighing systems, and digital electronic balances were installed and operated in a glovebox environment at DP Site. These NDA instruments, coupled with a sophisticated SNM accounting program, were then integrated into a prototype DYMAC system capable of drawing material balances around individual "unit processes" within a portion of the DP-Site process line.

The focus of the DYMAC effort (Phase II) is now directed toward installation of the plantwide material control system at the new TA-55 facility. The Eclipse C330 computer, which will execute the DYMAC real-time accountability program, has been installed at TA-55 and acceptance tests have been completed.

The final version of the LASL report "Coordinated Safeguards for Materials Management in a Mixed-Oxide Fuel Facility" was published as LA-6536. The report indicates that current measurement technology can provide a means for effectively safeguarding high-throughput fabrication facilities with cost-effective materials management and control systems. Concept definition and conceptual safeguards system design for a spent fuel reprocessing plant, as well as modeling and simulation efforts to assess the effectiveness of the safeguards system, are proceeding on schedule.

PART 1

SAFEGUARDS TECHNOLOGY, INTERNATIONAL SAFEGUARDS AND TRAINING

GROUP Q-1

Roddy B. Walton, Group Leader

H. O. Menlove, Alternate Group Leader

Group Q-1 is responsible for the development and application of a broad range of measurement techniques for the nondestructive assay (NDA) of fissionable materials in the many physical and chemical forms found in the nuclear fuel cycle. These techniques are used to detect and record neutrons and gamma rays emanating from the nuclear material, either naturally (passive assay) or after interrogation with a neutron source (active assay). NDA offers a direct, rapid, and accurate method of detecting, identifying, and assaying nuclear material on a continuing basis as it progresses through a nuclear facility. Specific applications of NDA technology currently under development by Group Q-1 include: plutonium process materials at ERDA facilities, highly enriched uranium enrichment process materials, and spent fuel reprocessing. Prototype instruments for these applications are developed, calibrated, and tested in operating plant environments. The Group Q-1 NDA technology program provides technical support to material control demonstration systems such as dynamic materials control (DYMAC), and a base of measurement technology for integrated safeguards system studies.

To implement the transfer of NDA technology to various types of plants and facilities in the nuclear

community, Group Q-1 is a major contributor to the ERDA Safeguards and Technology Training Program, which is offered to safeguards inspectors (both domestic and international) and qualified government and industry personnel. Additional Q-1 efforts to facilitate technology transfer include briefings, consultations, and issuance of design documentation, operations manuals, interlaboratory assay comparisons, and consensus standards.

A major Q-1 effort is technical assistance to IAEA international safeguards. Development of portable NDA instrumentation to address specific inspection situations, as well as associated calibration and evaluation of this instrumentation, represents a continuing program of assistance to the IAEA. In 1977 we began another effort under guidelines of the U.S. Program for Technical Assistance to IAEA Safeguards, a program funded by the State Department and coordinated through the International Safeguards Project Office at Brookhaven National Laboratory (BNL). This latest effort involves more than instrumentation development; in part, it addresses the IAEA's most urgent needs for technical support (including personnel training). Group Q-1 manages LASL's contributions to the Technical Assistance Program.

I. NONDESTRUCTIVE ASSAY APPLICATIONS AND MEASUREMENT CONTROLS

A. Calibration of the IAEA High-Level Neutron Coincidence Counter for Plutonium Plates in ZPPR Drawers (J. L. Sapir, H. O. Menlove, W. L. Talbert, M. deCarolis,* and N. Beyer*)

Critical assembly mockups of fast breeder reactors, such as the Zero Power Plutonium Reactor (ZPPR) in Idaho and the Fast Critical Assembly (FCA) in Japan, require large inventories of plutonium fuel. The plutonium is usually in the form of thin plates, 1.6 to 6.4 mm thick, 5 cm wide, and up to 22.9 cm long. These plates are interspersed with appropriate nonfissile mockups of fertile, structural, and coolant materials and placed in "trays" or "drawers" approximately 5 cm square and up to 91 cm long (Fig. 1). A typical drawer may contain 0.5 to 1.0 kg of fissile plutonium. The drawers are then stacked into two halves of the critical assembly machine, which are brought together remotely to perform the measurements.

Because of the large number of fuel plates involved and their disposition inside the drawers,

verification of the plutonium content of each plate individually is difficult and time-consuming. A good measurement technique is needed to verify the fissile content of an entire drawer without removal of the individual plates.

The IAEA requested LASL's assistance in obtaining appropriate materials for an investigation into the feasibility of NDA measurements on integral drawers. The Idaho ZPPR facility furnished us with plutonium plates (6.4 mm thick, 5 cm wide, and 12.7, 15.2, and 17.8 cm long) consisting of a Pu-U-Mo alloy fuel material clad in a stainless steel jacket. There were two types of fuel plates. One type was of normal ZPPR fuel containing a 28 wt% plutonium alloy with 11.5% ^{240}Pu ; the other was of SEFOR fuel containing a 20 wt% plutonium alloy with 8.6 wt% ^{240}Pu . We also obtained plates of aluminum, sodium, carbon, iron, and depleted uranium for fertile and filler materials. Appropriate interspersal of these plates with the fuel plates produces a good mockup of a typical FCA configuration.

For the NDA measurements, the portable High Level Neutron Coincidence Counter (HLNCC) was placed on its side, as shown in Fig. 2. Also shown in Fig. 2 is the IAEA intrinsic germanium detector which is being evaluated for plutonium isotopic measurements by using gamma-ray line ratios. The combination of the neutron and gamma-ray detectors gives an independent verification of the total plutonium content in the sample. The initial goal is to develop a measurement technique for assaying the plutonium in drawers from fast critical assemblies. A two-position measurement geometry is being studied to give uniform assay weight to the plutonium regardless of its axial position in the drawer.

The ZPPR-drawer data obtained with the HLNCC and the intrinsic germanium detector are being evaluated and will be reported in the next progress report. The procedures and the ZPPR materials will be used to obtain a calibration curve for use in future IAEA inspections of fast critical assemblies.

Fig. 1.

ZPPR drawer containing two rows of plutonium plates interspersed with plates of depleted uranium, sodium, aluminum, carbon, and iron to mock up structure and coolant materials in a fast critical assembly.

Fig. 2.

Measurement setup for a ZPPR drawer, showing the HLNCC for neutron coincidence counting and the IAEA intrinsic germanium detector with its collimator for measurement of plutonium gamma-ray line ratios.

B. Evaluation of IAEA HLNCC for Mixed-Oxide Fuel-Rod Assemblies (H. O. Menlove, E. Medina, and M. deCarolis*)

For IAEA inspection and verification purposes, a technique is needed for measuring the plutonium content in mixed-oxide fuel assemblies. The new HLNCC (Fig. 3) was designed to accommodate these fuel assemblies as well as small samples and cans.

In cooperation with the IAEA, we made a series of measurements to evaluate the HLNCC for use with mixed-oxide fuel assemblies typical of those encountered during inspections. A 28-rod mockup boiling water reactor (BWR) assembly (Fig. 4) was fabricated at LASL. The fuel rods are easily removed or added to the assembly so that geometric effects and neutron penetration from the interior of the fuel bundle can be investigated. The 11.2-cm-diam assembly contains 3 rings of fuel rods, with 16 rods in the outside ring, 8 in the second ring, and 4 in the center ring. All fuel rods contain 3 wt% PuO_2 .

In making the measurements we placed the fuel assembly in the center of the HLNCC in a vertical

Staff member, IAEA Safeguards, Division of Development.

Fig. 3.

The IAEA portable HLNCC for the verification of high-mass plutonium samples.

orientation. By removing rods from the assembly, we could observe the counter's sensitivity to rod substitutions. The results of the measurements are given in Fig. 5. One missing rod can be detected in a 200-s count, and a rod positioned in the center of the assembly has a sensitivity equal to that of a rod in the outer ring. This good penetrability is attributable to the fast neutrons, which are scattered but usually not absorbed by the PuO_2 - UO_2 rods. The slight increase in the coincidence response as a function of rod loading is caused by neutron multiplication in the assembly and end-effect backscattering into the counter.

These experiments show that the HLNCC can be used to verify the effective ^{240}Pu loading in full fuel assemblies for cases in which the assembly can be lowered into the detector. If the plutonium isotopic

Fig. 4.
Mixed-oxide BWR fuel-rod assembly mockup
for measurement with the HLNCC.

composition is not known, a supplementary gamma-ray ratio measurement must be performed to deduce the total plutonium loading.

C. Matrix Material Effects Study for the Dual Range Thermal-Neutron Coincidence Counter (E. Medina, M. P. Baker, M. L. Evans, and H. O. Menlove)

If neutron coincidence counters are to be used to measure bulk plutonium samples that are combined with large amounts of nonfissionable matrix materials, it is important to understand the influence of the matrix materials on the assay results. Then one can minimize the sensitivity of coincident-neutron assay techniques to the physical form and spatial configuration of such matrix materials. To this end, we made a series of measurements using the "dual range" coincidence counter¹ in

Fig. 5.
Coincidence response measured with the
HLNCC vs number of fuel rods in a $\text{PuO}_2\text{-UO}_2$
fuel assembly mockup.

its two configurations. This neutron counter has two removable cadmium sleeves located on the inner and outer faces of the ring of ^3He detector tubes. The cadmium shortens the neutron die-away time and gives variable efficiency for neutron detection. A cylindrical polyethylene jacket shields the detector from external background neutrons. We made the matrix-effect measurements with the cadmium sleeves in place and with the cadmium sleeves removed.

Several 12-cm-diam by 16-cm-high cans were filled with materials typical of plutonium scrap output. The materials investigated are listed in the first column of Table I. A small ^{252}Cf source of spontaneous-fission neutrons was placed in the center of each can for counting both total and

TABLE I
MATRIX-MATERIAL EFFECTS ON ASSAY
BY NEUTRON COINCIDENCE COUNTING

Dual Range Counter in High-Efficiency Mode

<u>Matrix Material</u>	<u>Density (g/cm³)</u>	<u>Matrix (g)</u>	<u>Normalized Totals (counts)</u>	<u>Normalized Coincidence (counts)</u>	<u>[Totals² Coincidence]</u>
Empty Can	---	---	1.00	1.00	1.000
MgO	0.702	703	1.006	1.012	1.000
Sand	1.753	1549	1.026	1.020	1.032
Ash	1.081	1437	1.021	1.030	1.012
Poly (chips)	0.052	92	1.008	1.004	1.013
Poly ^a (1 cm)	0.916	288	0.983	0.957	1.010
H ₂ O (100%)	1.000	1652	0.723	0.504	1.037

Dual Range Counter with Cadmium Sleeves Inserted

Empty Can	---	---	1.00	1.00	1.000
MgO	0.702	703	1.001	1.008	0.994
Sand	1.753	1549	1.008	1.010	1.006
Ash	1.081	1437	1.007	1.010	1.004
Poly (chips)	0.052	92	0.996	0.986	1.024
Poly ^a (1 cm)	0.916	288	0.967	0.926	1.010
H ₂ O (100%)	1.000	1652	0.647	0.407	1.029

^aThe 1 cm corresponds to CH₂ wall thickness inside the can.

coincident-neutron events. The resulting count rates were normalized to unity for the case of the empty can—thus the deviation from unity represents the net effect of the added matrix material. The matrix effects are small except for the last two materials, which contain large amounts of hydrogen. The detector is designed to exhibit minimum sensitivity to variations in the sample's hydrogenous content.

The last column of Table I gives the ratio of the square of the total counts to the coincidence counts. Because the total counts vary as the detector efficiency and the coincidence counts vary as the square of the efficiency, this ratio is independent of

efficiency to a first-order approximation. Thus the ratio shown in column 5 is relatively independent of detector-efficiency changes due to sample matrix materials. This consistency check can be used as a diagnostic tool in neutron coincidence counting to verify that observed rate changes are not caused by efficiency changes or electronics problems.

We conclude that most carbides and oxides have little effect on the observed coincident-neutron response and that the observed perturbation is approximately the same for all cases except those where large quantities of hydrogenous matrix materials are involved.

D. The CMB-8 Material Balance System (T. R. Canada, D. G. Langner, N. Ensslin, and T. L. Atwell)

An extensive review of the data obtained with the uranium-solution assay system over the past year is under way. Preliminary results show that the properties of the solution standards vary with time and are therefore inappropriate for long-term system monitoring. Uranium-foil secondary standards have been prepared and are being used.

A detailed measurement control program for the CMB-8 material balance system has been written and initiated. It delineates the control procedures, the acceptable ranges of the control parameters, and the action required when control limits have been exceeded.

E. Random Driver for Oak Ridge Operations Office (T. L. Atwell, N. Ensslin, K. Johnson, and W. B. Tippens)

A Random Driver was shipped in April 1977 to ERDA's Oak Ridge Operations Office for in-plant evaluation. The instrument will be used to assay a variety of solid uranium samples by a fast-neutron coincidence-counting technique. The main chassis of this random driver is the one used previously at the Savannah River Plant (Ref. 2, p.6). This chassis uses ^3He tubes to correct for any moderation of the interrogating neutron flux by the sample. The software incorporates a temperature correction and a density correction based on readouts from the temperature sensors and loadcell, respectively (Ref. 3, p.6).

All components of the random driver except the chassis were replaced and upgraded to the present state-of-the-art in minicomputers and fast electronics. Photomultiplier-tube amplifiers were installed to reduce the high voltages required by the phototubes. The instrument is controlled by a PDP-11/04 minicomputer interfaced to a dual floppy disk drive. A line printer and a video terminal were provided. The software operating system is RT-11, multi-user BASIC. Sufficient disk storage space, computer memory, and software are available to

assay samples and store results, analyze data, and run independent programs simultaneously.

All information received from the fast plastic scintillators, the ^3He tubes, the loadcell and the temperature sensors is transmitted to the computer via a CAMAC crate, which contains a 12-channel scaler, a timing module, an ADC, and a relay multiplexer. Information from the loadcell and the temperature sensors is processed by another CAMAC module (built by Group Q-1) before being multiplexed to the ADC. This random driver represents the first application of CAMAC to an in-plant assay instrument. The results of this field test will help in evaluating the flexibility and reliability of such modular systems.

F. Improvements in the Van de Graaff Small Sample Assay Station (SSAS) (J. D. Brandenberger and E. L. Adams)

We continue to work toward improving the accuracy of the SSAS. Because the system had not provided the assay accuracy indicated by counting statistics, long or repeated runs were required for a desired accuracy. Reference 1 describes the modification and tests that brought the observed precision of the SSAS to about 1.2 times the expected statistical uncertainty at a relative standard deviation of 0.5%.

Two additional modifications have improved the accuracy. First, the geometric extent of the neutron source was reduced by slits, and the proton beam on the neutron-producing lithium target was defocused to produce a more uniform neutron source. Second, the output pulses from the detectors were put directly into the PDP-9 data-acquisition system rather than into a discriminator and scaler. Thus the gain can be monitored and a more precise determination can be made of the counts for each assay.

With the above modifications, the SSAS is capable of a precision equal to the expected 0.5% relative standard deviation for a single day's set of runs. For routine assay applications, we plan to use high-mass samples with high count rates as secondary standards to normalize runs from one day to the next.

II. INSTRUMENTATION DEVELOPMENT AND TRAINING

A. Absorption-Edge Densitometry

1. The Can Scan—An Instrument for the Measurement of the Plutonium Mass Present in Ash (T. R. Canada, L. R. Cowder, S. T. Hsue, and J. W. Tape)

An instrument is being developed to adapt the K-absorption-edge densitometry technique to the assay of cans containing plutonium-bearing ash. The instrumental hardware is ready, and preliminary evaluation is under way. The principal component of the device is a drive assembly which translates the can between the transmission source and the detector, while simultaneously rotating it about its cylindrical axis. The transmission sources, 50 mCi of ^{76}Se and 25 mCi of ^{60}Co , are mounted in a large rotatable tungsten shield-collimator.

As described in Ref. 3, p. 21, the plutonium mass is given by

$$M_{\text{Pu}} = \left(\frac{-\pi r^2}{\Delta\mu} \right) \ln \bar{R}$$

where $\bar{R}$ is the ratio of the gamma-ray transmissions from the ^{60}Co (122 keV) and the ^{76}Se (121 keV) averaged over the can radius r , and $\Delta\mu$ is the difference between the plutonium mass absorption coefficients at 122 and 121 keV.

Preliminary results indicate that cans with radii of ~4 cm and containing 2 to 75 g of plutonium can be assayed with this technique. Larger masses of plutonium can be assayed in containers with correspondingly larger radii.

2. Matrix Sensitivity of Uranium Assays Obtained Using K-Edge Densitometry and a ^{169}Yb Transmission Source (D. G. Langner and T. R. Canada)

Ytterbium-169 is a convenient transmission source for uranium assay using the absorption-edge densitometry technique. Although its gamma rays at 110 and 130 keV bracket the uranium K-absorption edge at 116 keV, their relatively large

energy displacement from the edge makes the assay subject to variations in the matrix properties. In Ref. 3, pp. 19-21, an analysis technique is described which can reduce this matrix sensitivity for heavy-matrix contaminants.

A calculation has been made to determine the matrix concentration relative to uranium concentration that will cause a 1% error in a uranium assay obtained by (a) computing simply the ratio of transmissions at 110 and 130 keV and (b) extrapolating the 110- and 130-keV transmissions to the K-edge using the assumption that only high-Z matrix contaminants are present and

$$\log[-\ln T(E_K)] = \log[-\ln T(E_\gamma)] - 2.55 \log \frac{E_K}{E_\gamma}$$

where $T(E_K)$ is the extrapolated K-edge transmission and $T(E_\gamma)$ is the transmission obtained on either side of the edge when a ^{169}Yb source is used.

With method (a), the ratio of matrix concentration to uranium concentration, ρ_m/ρ_u , that will cause a 1% error is given by

$$\frac{\rho_m}{\rho_u} = 0.01 \left[\frac{\mu_u(110) - \mu_u(130)}{\mu_m(110) - \mu_m(130)} \right]$$

where μ_u and μ_m are the mass attenuation coefficient for uranium and the matrix, respectively, at the energies indicated within the parentheses. With method (b),

$$\frac{\rho_m}{\rho_u} = 0.01 \left[\frac{\mu_u(110) - \left[(110/130)^{-2.55} \right] \mu_u(130)}{\mu_m(110) - \left[(110/130)^{-2.55} \right] \mu_m(130)} \right]$$

In Fig. 6 these relationships are plotted against the atomic number Z of the matrix material. The μ values in these calculations were taken from Ref. 4.

The matrix dependencies from the two techniques differ dramatically. For matrix contaminants of all Z , except around 20, the two methods yield significantly different assay results. For $Z > 20$, the extrapolation technique, method (b), yields a less

Fig. 6.

Ratio of matrix concentration to uranium concentration, ρ_m/ρ_u , that will produce a 1% error in a uranium assay vs the atomic number Z of the matrix.

matrix-dependent result. For low- Z , the simpler ratio, method (a), provides the better assay. Thus, use of both methods might provide a means of detecting an unexpected matrix contaminant.

3. L_{III} Absorption-Edge Densitometry (T. R. Canada, D. G. Langner, and J. W. Tape)

Hardware has been designed and constructed for an x-ray generator-based L_{III} absorption-edge densitometry assay station. The station will form the basis for the development of an in-line solution assay system that will measure uranium and plutonium concentrations of 1 to 50 g/l.

Preliminary studies show that simultaneous assays can be made of plutonium and uranium in solutions containing both elements. The x-ray spectrum transmitted through a uranium- and zirconium-bearing solution is shown in Fig. 7. The uranium L_{III} and the zirconium K-absorption edges show clearly. The zirconium K-edge at 17.998 keV is representative of the plutonium L_{III} edge at 18.066 keV.

In some applications, a natural radioisotopic bremsstrahlung source may be preferred to an x-ray generator. Two such sources, ^{90}Sr and ^{147}Pm , have

Fig. 7.

The x-ray spectrum transmitted through a 2-cm-thick solution containing approximately 4.3 g/l of uranium and 1.1 g/l of zirconium.

been ordered, and their applicability as transmission sources will be evaluated.

4. Application of Dispersive X-Ray Filtering Techniques to Absorption-Edge Densitometry (J. W. Tape, T. R. Canada, and E. A. Hakkila)

Densitometry is one technique being considered for application to the problem of special nuclear material (SNM) assay in highly radioactive solutions (Ref. 3, pp. 19ff). In principle, an active assay method such as absorption-edge densitometry has the capability of maximizing the assay signal with respect to the noise (i.e., the background from hot

solutions). However, as the samples become increasingly radioactive, the transmission source must become more intense and a practical limit on source intensity will be reached. An ideal solution to this problem would be a detector that is sensitive only to photons with energies near the absorption edge of interest and completely insensitive to photons of other energies. For x-rays with energies near the L_{III} edges of uranium or plutonium, such an ideal detector can be approximated by combining a curved reflecting Bragg crystal and a germanium detector, as shown in Fig. 8.

The Bragg reflection law, $n\lambda = 2d \sin \theta$, states that photons with wavelength λ and a total angle of deviation of 2θ will be preferentially reflected by a crystal with lattice spacing d (Ref. 5). Thus, for a fixed geometry (Fig. 8), only photons with an energy corresponding to the wavelength λ will be detected by the germanium detector. The spread of energies reflected by the crystal is determined by the range of angles allowed by the slits. For densitometry applications, the use of wide slits makes the resolution of the crystal slit system worse than that of the germanium detector. The crystal then acts as a filter that passes photons in an energy band that is a few keV wide and centered around the absorption edge of interest. The change in transmission across the absorption edge obtained from the high-resolution germanium spectrum yields the detailed assay information.

Figures 9 and 10 illustrate how this technique might be applied. A band of continuous radiation from an x-ray generator is reflected by the curved crystal and detected by the germanium detector, resulting in the spectrum shown in Fig. 9. When a 2-mil niobium foil (K-edge = 18.987 keV) is inserted between the x-ray beam and the initial slits, the spectrum in Fig. 10 is obtained. Note that any background radiation with an energy outside the

Fig. 8.

Schematic of the geometry used for wavelength dispersive x-ray filtering.

Fig. 9.

X-ray spectrum detected by germanium detector after the continuous x-ray beam was reflected by the Bragg crystal.

Fig. 10.

X-ray spectrum detected by germanium detector after the beam was passed through a 2-mil-thick niobium foil and then reflected by the Bragg crystal. Note the niobium K-absorption edge at 19 keV.

range accepted by the crystal will not be detected by the germanium detector.

A small LiF crystal ($d = 2.20 \text{ \AA}$) ground to a 28-cm radius of curvature was used for this measurement. Larger crystals that will increase the efficiency of the system are on order.

B. Technical Assistance to the IAEA (W. L. Talbert)

During this reporting period, an initial program plan has been developed for the Program of U.S. Technical Assistance to IAEA Safeguards. The plan consists of a series of tasks defined by the IAEA in consultation with U.S. representatives. The tasks

range from provision of cost-free experts to development of instrumentation for specific needs of inspectors. The program is monitored and coordinated through the International Safeguards Project Office (ISPO) at Brookhaven for ERDA/DSS. The 15 tasks assigned to LASL are scheduled for a period of a year or two, and some represent extensions of ongoing projects. The ongoing program of technical assistance will evolve on a yearly basis, with new tasks being defined in response to IAEA needs. The participation of the LASL safeguards groups is expected to be prominent throughout the program.

C. Thermal-Neutron Coincidence-Counter Developments

1. Portable Electronic Package for the IAEA High-Level Neutron Coincidence Counter (HLNCC) (J. E. Swanson, P. R. Collinworth,* and M. S. Krick)

The portable HLNCC designed for the IAEA is described in Ref. 3, p. 10. An electronic system to accompany the counter has been designed, breadboarded, and successfully tested.

Five of the electronic systems are being packaged for LASL by EG&G, Inc., and the first unit should be delivered for evaluation in May 1977. We hope to evaluate the first unit and deliver it to the IAEA in July 1977.

The completed package will contain all of the electronic subsystems required to operate the counter: high- and low-voltage power supplies; six channels of preamplifiers, amplifiers, and discriminators; shift register coincidence circuitry; and control and display circuitry. The first units will have a light-emitting diode (LED) display as the only output; future units will contain a microprocessor to control the readout to calculators, terminals, etc.

The package will be operable either from a 120/240-V, 50/60-cycle ac supply or from a dc supply with a 7- to 20-V output; power consumption will be approximately 15 W. The instrument case will be 9 cm high, 28 cm wide, and 41 cm deep. The instrument and a Hewlett-Packard HP-97 programmable calculator will be packaged in a 17-cm-high, 53-cm-

wide, 33-cm-deep instrumentation suitcase for transportation.

2. Variance of Neutron Coincidence Circuits (N. Ensslin)

During the past year, three coincidence circuits were evaluated for use with thermal-neutron well-counters (Ref. 3, p. 10). These circuits are the variable deadtime counter (VDC), shift register, and updating one-shot, delay, one-shot (OSDOS). As reported in Ref. 3, the three circuits yield the same relative error for the assay of a fission source in the presence of a strong external background, provided that all coincidence gate lengths are equal. An example of this is given in Fig. 11 for measurements using 32- μ s gates.

It is somewhat surprising that three different circuits with vastly different deadtime losses should yield the same relative error. A careful analysis of these circuits, taking into account all correlations present in the input data or produced by the circuits, would probably show that their errors are not quite identical. However, for low multiplicity (ν) and low efficiency ϵ , and for the range of count rates over which assays are conducted, the measured errors are the same within 10 to 20%. With these limits in mind, we postulate the following explanation for the equivalence of the relative errors: All three circuits are trying to extract information from

Fig. 11.

Relative error in californium assay (constant strength ^{252}Cf source) as a function of Am-Li background rate.

*Group E-2.

the same distribution of intervals between events. This distribution is formed by T total events, and includes R real events, R_m real events and A accidental events are measured by forming coincidences during a gate length τ . If $A \gg R$, which is usually the case, the relative error in the assay will be roughly $\sqrt{2A}/R$. Since the three circuits have the same τ , they will have the same A and the same relative error. If $A \ll R$, the relative error is given by $1/\sqrt{S_m}$, where S_m is the number of observed fissions. At low rates, if $A \ll R$, $S_m \approx S$ where S is the true fission source strength. Also, for low ϵ and ν , the number of measured coincidences per observed fission, R_m/S_m , is nearly 1 for the three circuits. Then the relative error is given roughly by just $1/\sqrt{R}$. Specific formulas based on this postulate are given below.

For the VDC circuit the relative error is well represented by

$$\frac{\sigma(R)}{R} = \frac{\sqrt{R_m + 2(S_1 - S_2)}}{R_m} \quad (1)$$

The measured coincidences R_m are given by Eq. (4) of Ref. 3. The scalars S_1 and S_2 are strongly correlated, but the difference $(S_1 - S_2)$ is uncorrelated and corresponds to the accidental coincidences A produced by the gate $\tau = \tau_2 - \tau_1$. Equation (1) is simpler than Eq. (3) of Ref. 3 and also fits the measured errors within 10 to 20%.

For the shift register, a very thorough analysis of errors is given by Böhnel.⁶ He finds some correlation between the $(R + A)$ and the (A) registers that tends to decrease the relative error, and some terms of order $\sqrt{1 + \epsilon < \nu(\nu - 1) > / 2\nu}$ due to multiple neutrons per fission that tend to increase the relative error. These correlations are small for practical values of ϵ and ν , and the measured errors are well represented by

$$\frac{\sigma(R)}{R} = \frac{\sqrt{R + 2\tau T^2}}{R} \quad (2)$$

For this circuit the accidental coincidences are given by $A = \tau T^2$ when the counting rate is uniform.

The OSDOS circuit is subject to large deadtimes of order e^{-T} . These reduce the measured counts relative to the shift register, but do not affect the

variance. Equation (2), above, also fits the measured relative error in the OSDOS data to an accuracy of 10 to 20%, with $R = R_m e^{T^2}$. In accord with the postulate that all three circuits operate on the interval distribution in basically the same way, we expect Eq. (2) to be valid for the VDC circuit also. This is true, with R given by Eq. (1) of Ref. 3.

3. Evaluation of Neutron Reflectors for the HLNCC (E. Medina and H. O. Menlove)

The HLNCC is usually used to assay PuO_2 in cans. The fill height of the PuO_2 in the cans will vary with the mass and density in the can, and it is desirable to have the detector's counting efficiency insensitive to these height variations. One way to achieve a flat efficiency is to place end plugs at both ends of the well-counter to reflect back into the detector those neutrons that would otherwise escape through the ends.

To evaluate the effectiveness of end plugs, we made a series of measurements using a ^{252}Cf spontaneous fission-neutron source. The neutron source was moved along the axis of the HLNCC, recording both total counts and coincidence counts. Figure 12 shows the measured profiles with and without the 7.6-cm-thick polyethylene end plugs. The plugs greatly flatten the efficiency. The improvement is greater than for most neutron well-counters because the HLNCC is under-moderated, hence the polyethylene in the plugs is especially effective.

Fig. 12.

Measured axial counting efficiency in the HLNCC, with and without polyethylene end plugs.

As a result of these tests, the three HLNCC units being supplied to the IAEA will have end plugs available for applications in which a flat central response is desired.

4. Effects of Self-Multiplication on Coincidence Counts from PuO_2 Cans (J. Sapir and N. Ensslin)

Introduction. Efforts to calculate and understand the effects of self-multiplication on thermal-neutron coincidence counting continued. Experimental results of measurements on a series of eight PuO_2 standards are described in the previous progress report.¹ The measured coincidence response data, corrected for amplifier deadtime are summarized in Table II and plotted in Fig. 13. The initial self-multiplication corrections reported in Ref. 1 removed only 50% of the curvature exhibited in the experimental response data because only part of the multiplication was being considered. Our recent calculational efforts include the effects of discrete leakage multiplication (probability of 1, 2...n neutrons leaking per source neutron), (α, n) reactions, and reflection by the well-counter.

Definitions. If n neutrons enter the coincidence detector in a time scale short compared to the counter decay time, the coincidence response C to second order will be

Fig. 13.
Coincidence response from PuO_2 standards. Measurement errors are about equal to size of the circles.

TABLE II
SELF-MULTIPLICATION IN CANS OF PuO_2

Standard ID	^{240}Pu (g)	Coincidence Counts $\bar{x}^{240}\text{Pu-g}$	Leakage Multiplication $\bar{M}$	F_{AF}	N_{det}/N_{AF}	F_{det}	F_{net}	Correction Factor F	Corrected Normalized Response
705-002	1.2	2.35 ± 0.02	...	...	...	...	...	...	...
Std. No. 4	3.8	2.42 ± 0.02	1.0048	0.020	3.95	0.022	0.010	1.053	2.30
Std. No. 6	7.7	2.53 ± 0.02	1.0100	0.041	3.28	0.035	0.018	1.093	2.31
Std. No. 9	37.4	2.90 ± 0.03	...	...	...	...	...	...	...
PEO-385	43.4	2.98 ± 0.03	1.0466	0.188	1.36	0.067	0.031	1.286	2.31
PEO-382	54.9	3.05 ± 0.03	1.0388	0.201	1.31	0.070	0.038	1.309	2.31
PEO-381	65.0	3.98 ± 0.03	1.0505	0.208	1.26	0.076	0.042	1.319	2.34
PEO-447	81.3	5.26 ± 0.03	1.0542	0.226	1.29	0.077	0.051	1.353	2.41

$$C \propto \frac{\epsilon^2 n(n-1)}{2!} - \frac{\epsilon^3 n(n-1)(n-2)}{3!} + \dots \quad (3)$$

where ϵ is the detector efficiency.

If more than a single value of n exists (as in the case of spontaneous fissions in ^{240}Pu where n varies between 0 and essentially 6), the coincidence response becomes

$$C \propto \sum_n \left[\frac{\epsilon^2}{2!} P_n(n)(n-1) - \frac{\epsilon^3}{3!} P_n(n)(n-1)(n-2) \right] \quad (4)$$

which can be rewritten as

$$C \propto \frac{\epsilon^2}{2!} \langle n(n-1) \rangle - \frac{\epsilon^3}{3!} \langle n(n-1)(n-2) \rangle, \quad \text{where}$$

$$\langle n(n-1) \rangle = \frac{\sum_n P_n(n)(n-1)}{\sum_n P_n}, \text{ etc.,}$$

where P_n is the probability of n neutrons entering the detector, and $\sum P_n = 1$.

The analysis described in Ref. 1 approximated Eq. (4) for the case of spontaneous fissions in a multiplying medium with

$$C \propto \sum_{\nu=0}^6 P_{\nu} \bar{M}_F (\bar{M}_{\nu} - 1),$$

where P_{ν} is the probability of producing ν neutrons in a spontaneous fission in ^{240}Pu and $\bar{M}$ is the average leakage multiplication in the sample (neutrons leaking per source neutron). In Ref. 1 the second-order term was omitted from the calculations. To simplify the presentation that follows, the second-order terms are omitted from the equations but have been retained in the analysis.

We define a correction factor F as

$$F = 1.0 + F_{SF} + F_{\alpha n} + F_{\text{Ref1}},$$

where

F_{SF} = correction for multiplication of spontaneous fission neutrons

$F_{\alpha n}$ = correction for multiplication of (α, n) neutrons

F_{Ref1} = correction for additional multiplication of both spontaneous-fission and (α, n) neutrons due to reflection from the polyethylene detector.

This correction factor F is the ratio of coincidence response as calculated from Eq. (4) to the response expected if the multiplication were 1.0,

$$C_{M=1} \propto \sum_{\nu=0}^6 P_{\nu}(\nu)(\nu-1).$$

The measured response should therefore be divided by F to correct for multiplication effects.

Discrete Leakage Multiplication. The Monte Carlo code MCN has been modified by Robert Schreudt, Group TD-6, LASL, to yield the discrete leakage multiplication M_n , which is the probability of n neutrons leaking from the sample per source neutron. The average leakage multiplication $\bar{M}$ is related to M_n by

$$\bar{M} = \sum_n n M_n.$$

The MCN modification involves an analog representation of all neutron interactions such that only integral numbers of neutrons are allowed. The integral number of neutrons produced in a fission event is calculated by sampling from a Gaussian distribution about the mean multiplicity $\bar{\nu}$. The source neutron and all of its progeny are followed from birth to death (or escape) while the number of neutrons crossing a specified surface is tallied.

Multiplication of Spontaneous-Fission Neutrons. For spontaneous fissions in a multiplying sample, the quantities P_n appearing in Eq. (4) are obtained by appropriately combining and summing the values of P_n , the neutron emission probability in

spontaneous fission, and M_n , the discrete leakage multiplication. This quantity will be designated as P_n^{sf} . A code for this purpose was written by Mark Johnson, Group Q-13, LASL.

The calculated average leakage multiplication is listed in column 4 of Table II. The contribution to the overall correction factor due to multiplication of spontaneous-fission neutrons is obtained from

$$1.0 + F_{SF} =$$

$$\sum_n P_n^{sf} (n)(n-1) / \sum_\nu P_\nu (\nu)(\nu-1)$$

F_{sf} is shown in column 5 of Table II.

Multiplication of (α,n) Neutrons. In a non-multiplying sample (α,n) neutrons cannot produce net coincidence counts because only a single neutron is produced in each reaction. However, in a multiplying sample, some of the (α,n) neutrons may induce fission and the resulting neutrons might then produce coincidence counts. To account explicitly for (α,n) -initiated coincidences, Eq. (4) may be rewritten as

$$C \propto N_{SF} \sum_n P_n^{sf} n(n-1) + N_{an} \sum_n P_n^{an} n(n-1), \quad (5)$$

where

N_{sf} = spontaneous fission rate in sample

N_{an} = (α,n) neutron production rate in sample.

Because only one neutron is emitted in (α,n) reactions, the quantity P_n^{an} in Eq. (5) is equal to the discrete leakage multiplication M_n .

Dividing Eq. (5) by N_{sf} yields

$$C \propto \sum_n P_n^{sf} n(n-1) + \frac{N_{an}}{N_{sf}} \sum_n P_n^{an} n(n-1)$$

which is the same as Eq. (4) with the addition of the second term to account for (α,n) -initiated coincidence counts. Thus,

$$F_{an} = \frac{\frac{N_{an}}{N_{sf}} \sum_n P_n^{an} n(n-1)}{\sum_\nu P_\nu (\nu)(\nu-1)}$$

The ratio of N_{an}/N_{sf} was calculated for the samples using their known plutonium isotopic composition and the normalized spontaneous fission and PuO_2 (α,n) yields given in Tables 7.4 and 7.3, respectively, of Ref. 7. Because standards No. 4 and No. 6 contain diatomaceous earth, it was not possible to calculate N_{an}/N_{sf} . Instead, the difference between the observed total count rate and the rate expected from the plutonium isotopes present was attributed to (α,n) reactions. N_{an}/N_{sf} was then deduced from this difference. The calculated N_{an}/N_{sf} ratios are shown in column 6 of Table II, and the resulting correction term F_{an} in column 7.

Effect of Reflection in the Well-Counter. The multiplication calculations described thus far were performed for bare (unreflected) samples. Placing the sample inside the well-counter will produce a slight increase in multiplication due to the return of neutrons from the polyethylene ring. The presence of the cadmium liner along the inner polyethylene surface tends to minimize this effect by absorbing the low-energy neutrons. Monte Carlo calculations were run to simulate the well-counter surrounding the sample. The resulting correction term,

$$F_{Ref} =$$

$$\frac{(\text{Response calculated w reflector} - \text{Response calculated w/o reflector})}{\sum_\nu P_\nu (\nu)(\nu-1)}$$

which accounts for the effect of reflection on the coincidence count rate (including both spontaneous-fission and (α,n) neutrons) is shown in column 8 of Table II.

Conclusion. The total correction factor F is obtained by adding the three correction terms to unity

and is shown in column 9 of Table II. The coincidence counts per gram, obtained by dividing the measured response (column 3) by F (column 8) are shown in column 9 of the table, and the corrected coincidence counts are plotted in Fig. 13 as a function of the effective ^{240}Pu mass. For the standards calculated thus far, the application of the multiplication correction terms explains the non-linearity in the measured data quite well.

5. Monte Carlo Simulation of Coincidence Circuits for Thermal-Neutron Well-Counters (M. S. Krick)

Monte Carlo codes were written to simulate two circuits used for neutron coincidence counting—a shift register circuit⁸ and a simple (nonupdating), one-shot, delay, one-shot (OSDOS) circuit.³ Simplified block diagrams for these circuits are shown in Figs. 14 and 15.

Fig. 14.

Block diagram of OSDOS coincidence circuit.

Fig. 15.

Block diagram of shift register coincidence circuit.

The number of neutrons emitted on an average by the spontaneous fission of a ^{240}Pu nucleus differs from the average number emitted in an induced fission and multiplication process involving ^{239}Pu . Because the shift register circuit starts a new gate with each input pulse, it is more sensitive to the number of neutrons emitted per burst than is the simple OSDOS circuit. Therefore, in principle, information concerning the relative contributions of coincidence events arising from spontaneous fission and those from induced fission and multiplication is available from simultaneous measurements with a simple OSDOS coincidence circuit and with a shift register coincidence circuit.

For the Monte Carlo calculations, the thermal-neutron well-counter was assumed to have six channels, each with a $4\text{-}\mu\text{s}$ deadtime. The six discriminator outputs were assumed to be "ORed" together with a resolving time of $0.5\text{ }\mu\text{s}$. The predelay, gate widths, and detector die-away time were assumed to be 6, 32, and $32\text{ }\mu\text{s}$, respectively, for all calculations.

Figure 16 shows the coincidence counts per thermal-neutron-induced fission of ^{239}Pu vs the detector efficiency for the shift register circuit; the shaded area is the uncertainty band. One thousand histories were run at each of ten efficiencies ranging from 10 to 100%; the calculations have poor statistical precision for low efficiencies. Figure 17 shows the corresponding curves for spontaneous fission of ^{240}Pu and ^{252}Cf .

To study the relative response of the shift register and the simple OSDOS circuit as a function of the average multiplicity $\bar{\nu}$ the neutrons were assumed to be emitted according to a gaussian distribution which was centered at $\bar{\nu}$ and which had a width $\sigma = 1.08$. Between 100 and 5000 histories were run for integral values of $\bar{\nu}$ ranging from 2 to 10. The results for a detector efficiency of 50% are shown in Fig. 18; the shaded areas are the uncertainty bands.

The ratio of the responses of the shift register circuit to the simple OSDOS circuit increases with increasing detector efficiency and increasing multiplicity. We do not know if the response ratios are large enough in practical situations to provide useful multiplicity information because no appropriate experimental studies have yet been performed.

Fig. 16.

Results of Monte Carlo calculations for a shift register coincidence counter. Coincidence counts per 10^3 fissions vs the neutron detector efficiency for the thermal-neutron-induced fission of ^{239}Pu . The shaded area is the uncertainty band.

Fig. 17.

Results of Monte Carlo calculations for a shift register coincidence counter. Coincidence counts per 10^3 fissions vs the neutron detector efficiency for spontaneous fission of ^{252}Cf and ^{240}Pu . The shaded areas are the uncertainty bands.

D. Portable Neutron Assay System for Light Water Reactor Fuel Assemblies (J. D. Brandenberger, E. Medina, and H. O. Menlove)

During the current reporting period, work continued on defining the sensitivity of the active neutron portable assay unit for pressurized water reactor (PWR) fuel assemblies. The present configuration of the 15 x 15 rod assembly is shown in

Fig. 18.

Results of Monte Carlo calculations for shift register and simple OSDOS coincidence circuits. Coincidence counts per event vs the neutron multiplicity per event for neutron detector efficiency of 50%. The shaded areas are the uncertainty bands.

Fig. 19. To study the unit's sensitivity to rod substitutions, we removed fuel rods from the indicated sections 1-10 and replaced them with iron rods.

A jig, built to stabilize the positions of the rods in the fuel assembly as they were moved and replaced section by section, prevents skewed positioning of the rods. We measured the assembly three times under separate conditions of active interrogation by the $^{241}\text{Am-Li}$ source. For each set of measurements, the amplifier and/or discriminator for fast neutrons was somewhat different. Data sets 1 and 2 had an effectively lower bias level on the fast neutrons resulting from fission. The count rate decrease and

Fig. 19.

PWR fuel assembly assay system. Small circles represent individual fuel rods in part of a 15 x 15 rod assembly. Circles with an X are possible locations for control rods.

statistical uncertainty were renormalized to the lower count rate of data set 3. These results are summarized in Table III. The results for sensitivity are consistent for all three data sets, indicating that the technique is not dependent on the particular electronic settings so long as they are stable.

When the portable assay unit is used as a verification instrument, a decrease in counts would be expected to reveal that fuel rods were missing. For accurate counting, stable electronics are required, as well as a method of testing the amplifier gain-discrimination level at the test site, perhaps with Am-Li background neutrons.

The assay assembly is being tested for the spatial distribution of outgoing fission neutrons by placing a ^{252}Cf fission-neutron source in various positions in the full PWR bundle. The results of this mapping will determine the final geometry of the assay system.

E. Prototype ^{252}Cf Shuffler for Neutron Interrogation of 55-gal Barrels (T. W. Crane)

A lead-lined, 55-gal-barrel assay chamber was built for the shuffler system¹ and will be used to check the feasibility of assaying small quantities of fissile material in large samples contaminated with up to 100 R/h of fission products.

The background sources of neutrons were identified and measures were taken to reduce their magnitude. When only a cadmium sheet covered the outside of the ^3He tube banks, the observed room background rate was 300 n/s. An additional 20 cm of polyethylene shielding reduced the background to 4 counts/s without the internal lead shielding. The count rate from the source "Off" position was reduced from 120 to 9 counts/s by filling the gap between the shuffler and the well-counter with 30 cm of polyethylene and by stacking polyethylene

TABLE III
SENSITIVITY TEST RESULTS OF PORTABLE PWE ASSAY SYSTEM
FOR A FILLED 15 x 15 FUEL-ROD ASSEMBLY

Region ^a	Decrease in Counts per Rod per 2000 s			Sensitivity at the 2 σ Level	
	Set 1	Set 2	Set 3	Rods Exchanged for Detection	Rods Exchanged (%)
1	202	166	145	2	1.0
2	145	156	153	2	1.0
3	132	185	178	2	1.0
4	58	85	79	< 3	1.8
5	69	161	73	< 4	1.8
6	189	213	154	2	1.0
7	24	35	37	< 10	4.4
8	40	76	44	< 6	2.7
9	73	84	71	3	1.3
10	7	10	12	< 12 ^b	5.3 ^b
11	22	43	17	< 12	5.3
12	31	102	57	< 6	2.7

^aRegion in which iron rods have been substituted for fuel rods.

^bThe 12 rods overlap into the adjacent sections.

under the shuffler unit. The background from cosmic-ray-induced neutrons in the 4800-kg lead shield is 14 counts/s. Only additional overhead shielding would be effective in reducing it.

We performed a short test to estimate the sensitivity of the shuffler system in detecting fissile material in a 55-gal drum, which contained a matrix of graphite scraps and an unknown amount of water (the drum had been outside for several years) and which had a total mass of 223 kg. A 50-mg ^{235}U sample was placed at the center of the barrel. The sensitivity for a 1000-s assay, consisting of a 312-s background count followed by a cyclic interrogation with delayed-neutron counting, was 42 mg ^{235}U at the 3σ confidence level.

Although this low sensitivity is adequate for safeguards and accountancy purposes, the 10 nCi/g fiducial for transuranics could not be met by this prototype unit. For a unit with (1) a more efficient well-counter (a circular geometry, longer tubes, and graphite end reflectors), (2) a stronger interrogation source, 2 mg ^{252}Cf ; and (3) a factor-of-5 improvement in the "Off" shielding, a sensitivity of 4 mg plutonium (80% ^{239}Pu , 20% ^{240}Pu) could be achieved. For a drum loaded with 100 kg of material ($\rho = 0.5 \text{ g/cm}^3$) the corresponding activity would be 4 nCi/g. Thus, the shuffler-type assay of large samples is a reasonable approach for NDA even at the 10-nCi/g fiducial.

F. Nondestructive Assay of Irradiated Fuel

1. Passive Gamma Assay of Irradiated Fuel (S. T. Hsue)

In an irradiated fuel rod, the radial activity distribution may not be uniform because of fission-product migration or nonuniform burnup. The radial activity distribution can be determined by scanning with narrow collimation in a hot-cell environment, but will be difficult to determine in field inspection. If the actual activity distribution is not known, uniform distribution is usually assumed, and an error in the attenuation calculation results.

To estimate the effect of radial activity distribution on attenuation correction, consider the following case. The ^{137}Cs is assumed to migrate radially

toward the fuel periphery to produce a concentration profile described by

$$A(r) = K e^{-B(R-r)}$$

where

$A(r)$ = activity per unit length at radius r

K = arbitrary constant

R = fuel-pin outer radius, and

$B = (1/2) \text{ arcosh}(A_1/A_2)$,

where A_2 is the peripheral activity per unit length and A_1 is the center activity per unit length.

The value of the 661-keV gamma flux received by a detector from a fuel rod of this type was calculated and compared with that received from a fuel rod containing the same amount of ^{137}Cs activity but in which the activity was homogeneously distributed. The calculation was performed both for the well-collimated case (collimator width $\ll R$) and uncollimated case (collimator width $\sim 2R$). A series of curves for fuel rods with a density of 10.5 g/cm^3 (96% of theoretical density of UO_2), and diameters of 0.7, 1.0, 1.3, and 1.6 cm is shown in Fig. 20. In this figure, the degree of activity distribution varies from $A_2/A_1 = 1$ (uniform distribution) to $A_2/A_1 = 100$. The counting rate from the nonuniformly distributed activity is consistently higher. The size of the error made by assuming a uniform distribution depends on the radius of the fuel pin and the activity distribution within the pin.

2. Passive Neutron Assay of Irradiated Fuel (S. T. Hsue)

Passive neutron assay of irradiated fuel has a distinct advantage over gamma assay in that the fuel rods in the inner portion of the assembly contribute a significant portion of the total neutrons at the surface of the assembly. To investigate the correlation between burnup and the total neutrons emitted from irradiated fuel, we calculated the spontaneous-fission neutrons and alpha-induced neutrons from oxides of various uranium, plutonium, americium, and curium isotopes. In the calculation we used the isotopic composition of the above-mentioned

Fig. 20.

Difference in percentage of ^{137}Cs activity detected from a fuel rod as a function of radial migration

isotopes from postirradiation examination of fuel from the second cycle in the Vercellese Reactor⁶ (burnup range of 13 000 MWD/MTU to 27 000 MWD/MTU). The effect of neutron multiplication was ignored.

Figures 21 and 22 depict the neutrons emitted per second per metric ton of initial uranium as a function of cooling time for burnups of 12 859 and 26 884 MWD/MTU, respectively. At discharge, most of the passive neutrons originate from the ^{242}Cm isotope.

Figure 23 depicts the total neutrons emitted per second as a function of burnup. At discharge, the total number of neutrons emitted per second is proportional to burnup from 13 000 to 27 000 MWD/MTU. After a 2- to 3-yr cooling time, the ^{242}Cm isotope is the main contributor of the passive neutrons for burnups that exceed 15 000 MWD/MTU. Figure 23 shows that after 2 yr of cool-

Fig. 21.

Neutrons per second from ^{238}U , ^{240}Pu , ^{242}Cm , and ^{244}Cm isotopes at a burnup of 12 859 MWD/MTU. The total neutrons include contributions from the other uranium, plutonium, and americium isotopes.

ing the total number of neutrons emitted per second is proportional to burnup from 18 000 to 27 000 MWD/MTU. For burnups of less than 18 000 MWD/MTU, the contributions from ^{240}Pu and ^{238}Pu isotopes are significant.

Out of 100 neutrons emitted from ^{242}Cm decay, ~82 come from spontaneous fission and the rest come from (α, n) reactions on oxide. From the ^{244}Cm decay, almost 99% of the neutrons come from the spontaneous-fission process. Since the neutron yield is limited to these two isotopes, neutron coincidence measurement would not have much advantage over simple neutron counting measurement. The calculation indicates that passive neutron assay can be used as a consistency check over the burnup range of 18 000 to 27 000 MWD/MTU for various cooling times.

Fig. 22.

Neutrons per second from ^{238}Pu , ^{240}Pu , ^{242}Cm , and ^{244}Cm isotopes at a burnup of 26884 MWD/MTU. The total neutrons include contributions from the other uranium, plutonium, and americium isotopes.

Fig. 23.

Total neutrons per second vs burnup at the time of discharge and after a 2-yr cooling period.

3. Fission-Product and Actinide Buildup Calculation (W. B. Wilson,* T. R. England,* S. T. Hsue, and W. L. Talbert)

We are calculating the fission-product and actinide buildup as a function of burnup. The purposes of the calculation are twofold: (a) to probe the basis for fuel burnup determination by such indicators as fission-product activities or transuranic-isotope concentrations (especially the curium isotopes) and (b) to determine the transuranic-isotope buildup for enriched uranium oxide fuel compared to that for mixed-oxide fuel. The latter problem is of particular interest in estimating the radioactive waste as well as the associated heat output generated by the two types of fuel.

We are developing a capability for an accurate calculation of temporal concentrations of reactor fission-product and actinide nuclides, using primarily the CINDER-10 computer code.

At present, the CINDER-10 code has two major data libraries. The fission-product library consists of all known fission products (825 currently) and yield data for 6 fissionable actinides (^{233}U , ^{235}U , ^{238}U , ^{239}Pu , ^{241}Pu , and ^{232}Th) at one or more fission-neutron energies. The fission products are included in 423 linear chains with up to ~20 linear nuclides in a single chain and averaging ~5 linear nuclides per chain. These fission-product chains and data are suitable for calculations for time periods associated with problems ranging from fission bursts to waste disposal studies.

The code also has an actinide library, consisting of 58 chains with an average length of ~18 linear nuclides per chain. Currently, the cross-section data in this library are limited to 2200 m/s cross-section values and resonance integrals.

For accurate production calculation, a library of 154-group cross sections for the actinide nuclides will be prepared from the preliminary version of ENDF/B-V actinide files. Preliminary versions of these files have been processed by us with the NJOY code; processing and review have revealed inadequacies and errors in the files. The corrections and improvements have been incorporated in the new preliminary ENDF/B-V actinide files. It will take another month to prepare the actinide cross-section library.

*Group T-2.

**G. Advanced Fast Flux Test Facility (FFTF)
Fuel-Pin Scanning Systems (M. S. Krick, H. O.
Menlove, and R. B. Walton)**

**1. Performance Specifications for FFTF Pin
Scanning Systems**

At the request of the Hanford Engineering Development Laboratory (HEDL), we evaluated two proposals solicited by HEDL from commercial vendors for an advanced, high-speed fuel-pin scanner for the High Performance Fuel Laboratory (HPFL). Also, at the request of ERDA/DSS, we commented on the technical feasibility of constructing an advanced fuel-pin scanner to revised specifications provided by ERDA/DRDD.

Original specifications for the proposed scanner required a fissile uniformity of 0.5% per 0.64 cm of pin length at the 95% confidence level. This requirement was so tight that despite the extreme measures proposed, the uniformity specification might not have been met.

In revised performance objectives prepared by HEDL and DRDD, the uniformity requirement has been relaxed by a factor of 4 (from 0.5 to 2% for 0.64 cm at the 95% confidence level). Table IV shows the performance specifications of the existing LASL/HEDL fuel-pin scanner,¹⁰ the revised HEDL/DRDD goals, and a LASL estimate of realistic performance objectives. This last estimate was obtained essentially by extrapolation of the performance characteristics of the existing LASL/HEDL scanner. Although the present HEDL/DRDD goals are considered to be technically feasible, attainment will be a challenge, particularly for the 2% fissile uniformity requirement, which is considerably better than for existing LWR fuel-rod scanners.

**2. Evaluation of Proposed FFTF Fuel-Rod Scan-
ners (D. A. Close and H. O. Menlove)**

HEDL has inquired about upgrading the FFTF fuel-rod scanner to obtain a higher throughput and more precise pellet-to-pellet measurements. One approach proposed by instrument vendors is to use a thermal-neutron system with the fuel surrounded with a layer of cadmium or thin B₄C to harden the

neutron spectrum and obtain better neutron penetration in the fuel pin. Monte Carlo calculations were performed in cylindrical geometry for the following feasibility studies. The fuel rod (0.241-cm radius) was surrounded by a 1-cm-thick annulus, allowing for the insertion of a layer of cadmium or B₄C. This annulus was in turn surrounded by a 10.16-cm-thick annulus of polyethylene. A point neutron source having the energy spectrum of a ²⁵²Cf fission spectrum was assumed to be midway through the polyethylene moderator.

The FFTF fuel is a UO₂-PuO₂ mixture, using natural uranium, with an overall density of 10 g/cm³. For the calculations, the plutonium in the mixed oxide was varied between 10 and 50 wt%. To observe the sensitivity to the ²⁴⁰Pu content, calculations were performed assuming no ²⁴⁰Pu as well as the nominal 3% ²⁴⁰Pu (in the total U + Pu). The isotopic composition of a typical fuel mixture containing 23 wt% plutonium and 3% ²⁴⁰Pu is shown in Table V. Figure 24 shows the number of fissions in the fuel rod as a function of the percentage of plutonium when no cadmium or B₄C is present. The lower response for the 3% ²⁴⁰Pu compared to the 0% ²⁴⁰Pu case is caused by parasitic capture of thermal neutrons by the ²⁴⁰Pu.

Figure 25 shows the corresponding results when a 0.081-cm- (0.7 g/cm²-) thick cadmium layer or a 0.1-cm- (0.2-g/cm²-) thick B₄C (natural boron) layer is inserted around the 3% ²⁴⁰Pu fuel. Also shown in Fig. 25, for direct comparison, is the case with no neutron absorber layer, replotted from Fig. 24. The cadmium and B₄C absorbers reduce the response by about two orders of magnitude.

Least squares fits to the data were performed using the functional form

$$R = A(1 - e^{-BM}) \quad (6)$$

where R is the number of fissions in the fuel rod per source neutron, M is the percentage of plutonium (plutonium loading), and A and B are constants. The smooth curves in Figs. 24 and 25 are the least squares fits to the data. These fits are summarized in Table VI. The large errors on the parameters for the cadmium and B₄C cases reflect the low responses when these absorbers are present in the Monte Carlo calculations.

TABLE IV
PERFORMANCE OBJECTIVES FOR AN
ADVANCED, HIGH-SPEED FUEL-PIN SCANNER

Item	Existing LASL/HEDL System	HEDL/DRDD Goals	LASL Estimate of Realistic Performance Objectives
Scanning rate	15 pins/h	50 pins/h	50 pins/h
Fissile content per pin			
Precision (2σ)	0.8%	---	0.5%
Accuracy (2σ)	1.2%	0.75%	0.75%
Fissile uniformity			
Precision of fissile measurement per 0.64 cm (2σ)	1 %	---	0.5%
Probability of false detection of nonuniformity per 0.64 cm	0.0002	0.0002	0.0002
Relative deviation of fissile content per 95% detection probability (with 97% acceptance probability for nondeviant pin)	6 %	2 %	2 - 3%
Accuracy of fuel-column length measurement	---	0.25 cm	0.25 cm
Minimum detectable gap at 95% confidence level	---	0.025 cm	0.025 cm

The parameter A is a normalization parameter controlling the magnitude of the response, and the parameter B controls the shape of the curve. The larger the absolute magnitude of B, the sooner the response saturates for larger values of plutonium. Conversely, the smaller the magnitude of B, the more linear is the response. When there is no cadmium or B_4C surrounding the fuel rod, the number of fissions per source neutron saturates for plutonium loadings greater than 20%. Although this saturation effect is reduced when cadmium or B_4C

surrounds the fuel rod, there is still appreciable non-linearity, as can be seen in Fig. 25.

From Fig. 25 it would appear that a cadmium absorber has the most nearly linear response for the three cases presented. Also the magnitude of the nonlinearity parameter B is the smallest for the cadmium case, indicating that the response for the cadmium case is the most nearly linear. However, judging by the results in Table VI, the difference between the nonlinearity factor for the cadmium and

TABLE V

FFTF FUEL COMPOSITION

Isotope	Weight Fraction ^a	Atom Fraction ^b
¹⁶ O	0.1187	0.6674
²³⁸ U	0.0048	0.0018
²³⁰ U	0.6478	0.2447
²³⁹ Pu	0.1979	0.0745
²⁴⁰ Pu	0.0266	0.0100
²⁴¹ Pu	0.0042	0.0016

^aThe fractions correspond to a fuel pin with 23 wt% plutonium.

^bTotal atom density = 0.0670×10^{24} atoms/cm³.

Fig. 24.

Number of fissions in the fuel rod per source neutron as a function of the percentage of plutonium in the rod for no B_4C or cadmium surrounding the fuel rod. The solid curves are least squares fits to the data.

B_4C absorbers [$(1.35 \pm 2.72) \times 10^{-2}$] is not significant.

Another way to quantify the nonlinearity of the response curves presented in Fig. 25 is to consider the error buildup in determining the plutonium loading caused by an uncertainty in measuring the response. The following relationship between the absolute uncertainty in the plutonium loading

Fig. 25.

Number of fissions in the fuel rod per source neutron as a function of the percentage of plutonium in the rod for 0.081-cm-thick cadmium and 0.1-cm-thick B_4C surrounding the fuel rod. The solid curves are least squares fits to the data. Also shown for comparison is the case where no thermal neutron absorber surrounds the fuel rod.

(dM/M) and the absolute uncertainty in the response (dR/R) is derived from Eq. (6),

$$\frac{dM}{M} = - \frac{(1 - e^{BM})}{BM e^{BM}} \frac{dR}{R}$$

Thus the absolute uncertainty in the plutonium loading depends on the plutonium loading and on the nonlinearity parameter B, but is independent of the normalization parameter A. Table VII lists the error buildup in determining the plutonium loading for the three cases shown in Fig. 25. It shows that the nonlinearity of the calibration curve can cause significant error buildups. For the case with no thermal-neutron absorbers, the large errors in the plutonium content for high plutonium loadings indicate that the calibration curve has nearly saturated. The B_4C and cadmium cases indicate less saturation of the calibration curve. For a linear calibration curve passing through the origin, a 1% error in the response results in a constant 1% error in the plutonium loading, independent of the plutonium loading. If the errors on the calibration

TABLE VI
PARAMETERS RESULTING FROM LEAST SQUARES
FITS TO THE FFTF FUEL-ROD DATA

Neutron Absorber and Fuel	Curve Fitting Parameters	
	A	B
No B ₄ C or Cd 3% ²⁴⁰ Pu	8.098×10^{-3} $\pm 4.248 \times 10^{-4}$	-1.016×10^{-1} $\pm 1.696 \times 10^{-2}$
No B ₄ C or Cd 0% ²⁴⁰ Pu	8.332×10^{-3} $\pm 4.228 \times 10^{-4}$	-1.136×10^{-1} $\pm 1.975 \times 10^{-2}$
0.081 cm Cd 3% ²⁴⁰ Pu	2.391×10^{-4} $\pm 2.014 \times 10^{-4}$	-1.840×10^{-1} $\pm 2.004 \times 10^{-2}$
0.1 cm B ₄ C 3% ²⁴⁰ Pu	1.004×10^{-4} $\pm 3.753 \times 10^{-5}$	-3.186×10^{-2} $\pm 1.841 \times 10^{-2}$

TABLE VII
PLUTONIUM ERROR BUILDUP AS A
FUNCTION OF PLUTONIUM LOADING

Plutonium ^a Loading (%)	Plutonium Error Buildup Factor		
	No B ₄ or Cd	B ₄ C	Cd
0	1.000	1.000	1.000
5	1.303	1.084	1.047
10	1.734	1.178	1.098
15	2.356	1.282	1.152
20	3.262	1.399	1.209
25	4.598	1.529	1.270
30	6.586	1.675	1.335
35	9.568	1.838	1.404
40	14.076	2.022	1.478
45	20.940	2.228	1.556
50	31.452	2.460	1.641

^aThe mixed-oxide fuel contains 3% ²⁴⁰Pu.

parameters are ignored, the cadmium case exhibits the least error buildup. All three response curves are less linear than the present FFTF fuel-rod scanner in operation at HEDL.

H. CdTe Gamma-Ray Detector Investigations (J. W. Tape)

Safeguards inspectors who must transport their NDA equipment to different locations need a small, efficient, high-resolution gamma-ray detector that will operate at room temperature. Of a number of binary semiconductors investigated as possible candidates for such a room-temperature detector, only CdTe has the potential for high efficiency at the >100-keV energies required for safeguards measurements.

Development of CdTe for gamma-ray detection has been under way for some time both in the U.S. and abroad. To ascertain the current level of development of CdTe and to better appreciate its possible uses and limitations, we purchased a small detector from Radiation Monitoring Devices, Inc. (RMD). If the CdTe detector is to be applied to absorption-edge densitometry, good resolution and high efficiency for gamma rays at energies up to 122 keV will be required.

The RMD detector is 10 mm in diameter and 2 mm thick (157 mm³) and when used with a Canberra 970D low-noise preamplifier has a resolution of 8% at the 122-keV gamma-ray energy from ⁶⁰Co. A preamp specifically designed for CdTe is not available commercially, but some improvement in

resolution might be gained by a better matching of detector and preamp. The detector resolution increased about 10% at high coun. rates (50 000 counts/s).

The most serious drawback of this detector is its poor charge collection characteristics. The amplified and shaped signals from a more energetic gamma-ray source exhibit two distinct pulse shapes, one with short and the other with long rise- and fall-times (for amplifier time constants of $\sim 1.0 \mu\text{s}$). Longer time constants will cure the "two pulse-shape" problem, but the good resolution will be lost. The slow rising and falling component of the amplified pulse is presumably due to slow or incomplete charge collection and results in a long "tail" associated with each gamma-ray peak in the spectrum. The spectra labeled PSA OFF in Figs. 26 and 27 show the effects of the poor collection. The long tails on the peak become a serious problem for complex gamma-ray spectra when one tail piles up on another.

Standard pulse-shape analysis (PSA) techniques can minimize the tails associated with each peak. A block diagram of the PSA electronics is shown in Fig. 28. Figures 26 and 27 show a considerable improvement in spectral quality when PSA is used to reject pulses with the improper shape (curves labeled PSA ON).

Fig. 26.

^{57}Co spectra from CdTe detector with and without pulse-shape analysis. Note that NaI (Tl) detectors normally used in safeguards will not resolve the 122- and 136.5-keV lines from ^{57}Co .

Fig. 27.

Highly enriched uranium spectra from CdTe detector with and without pulse-shape analysis. The 2-mm thickness of this detector results in low efficiency for the 186-keV ^{235}U gamma ray.

CdTe detectors can satisfy some safeguards needs, but development work is still required. Larger volume and area detectors with good resolution are needed. In addition, if the charge collection characteristics cannot be improved, PSA circuits will have to be incorporated in the instrumentation to be used with CdTe detectors (e.g., the next-generation stabilized portable assay meter).

Fig. 28.

Block diagram of the pulse-shape analysis setup. A window on the "time spectrum" output by the PSA box is used to gate the analyzer on only when pulses of a particular fall time are present at the ADC input.

I. Upgrading of the MEGAS Box Counter (T. W. Crane, D. F. Jones, and E. R. Martin)

In upgrading the MEGAS box counter we are concerned primarily with incorporating an intrinsic germanium detector into the system so that combustible waste containing fission products can be assayed for transuranic contamination at the 10-nCi/g fiducial.¹ Here we discuss modifications to the scanning hardware and the collimator design.

The MEGAS box counter uses rotation and vertical translation to improve the response uniformity for the waste boxes. However, with the present system the mid-portion of the box is weighted more heavily than the ends. This effect arises because in the vertical scan the mid-portion of the box passes by the entire detector whereas the ends of the box pass by only half of the detector. The effect of incorporating a pause in the vertical scan at the ends of the box is shown in Fig. 29. The data indicate an improvement of approximately a factor of 3 in the response uniformity. A new motor controller is being developed for the upgraded box counter that will incorporate a pause at the ends of the vertical scan. In principle, this improvement could be retrofitted to existing systems.

Fig. 29.

Response variation for a 57-L combustible waste box. The optimum time indicated is approximately 22% of the vertical scan time.

Computer calculations were used to model collimator configurations for the upgraded MEGAS box counter. The code takes into account the rotation and vertical translation of the standard waste boxes to calculate the photon-detection probability and its variation as a function of position within the box. The matrix material is assumed to have the attenuation properties of carbon. The intent of the study was to select a collimator that maximizes the solid angle (photon-detection probability) while maintaining a reasonable response uniformity over the entire volume of the box.

The calculations show that no single aperture for the collimator can be optimum for the entire energy spectrum used in the analysis. In general the L x-rays (13 to 20 keV) require a smaller aperture than the higher energy gamma rays (≥ 30 keV) to maintain a uniform response. Since the more accurate assays will be based on the gamma rays for which a large solid angle is desirable, a wider opening is preferred. However, the assays near and below the 10-nCi/g fiducial will use the low-energy L x-rays. The variation with a large aperture is about a factor of 2.3 at 17 keV; thus, to be on the safe side an artificial 4-nCi/g fiducial must be used to ensure that the 10-nCi/g level will be met. The additional counting time is a factor of 5 for this case. Thus, it is important for the response to be uniform for both the L x-rays and the gamma rays.

One way to achieve uniformity is to incorporate a variable aperture collimator in the system—a solution that would complicate the apparatus and require a separate counting time for each energy region. A second solution is to make the collimator aperture energy-dependent, with a larger effective opening for the higher energy photons. One way to achieve this behavior is to cover part of the aperture with a thin metal foil. With this technique, the higher energy gamma rays pass through the thin foil while the L x-rays are stopped. Computer simulations were made with different foil materials, including tin, niobium, copper, and iron, and with a number of configurations. To avoid possible interference from x-ray fluorescence in the foil and for structural strength, iron (actually spring steel) foil was chosen.

If we choose a $\pm 30\%$ limit for systematic errors on a quantitative assay, the calculations can be used to determine the mass range of boxes that can be assayed by a photon of a given energy. In Table VIII

TABLE VIII
BOX MASS RANGES FOR
±30% ACCURACIES

Photon Energy (keV)	Low and High Limits (kg)		
	No Foils	One Foil	Two Foils
13	0-1.3	0-2.2	1.2-2.5
15	0-2.3	0.9-3.8	1.9-4.2
17	0-2.7	0.9-4.6	2.1-5.1
20	0-3.9	0.9-6.7	2.3-6.1
30	0-6.7	0-10	0-10
60	0-9.8	0-11	0-11
200	0->12	0->12	0->12

the box mass limits are given. The collimator without the additional iron foils has by far the lowest usable range of box masses. The choice between the one- and two-foil configurations is less clearcut. The two-foil system allows a slightly greater high-mass limit for low-energy photons; however, the low-mass limit could be a nuisance when a lightly packed box is assayed. Therefore, the single-foil configuration will be used for the up-graded box counter.

J. The USERDA Safeguards Technology Training Program (T. R. Canada, J. W. Tape, and J. L. Parker)

A fourth course, "Integrated Safeguards Systems—Concepts and Implementation," was introduced into the USERDA Safeguards Technology Training Program curriculum during the week of March 21, 1977. The course was conducted by individuals from the four LASL safeguards groups Q-1, Q-2, Q-3, and Q-4, and from the Sandia Laboratories. The 3-day course was attended by 35 representatives of the safeguards community. A list of those attending and their affiliation is given in Table IX.

The course included an overview of real-time dynamic materials accounting and control (DYMAC) concepts and the techniques for their incorporation into safeguards plants. Lectures, demonstrations, and tours of LASL facilities were given. A detailed course manual, the table of contents for which is shown in Table X, was prepared by the participating lecturers and is available for distribution.

The growth in the scope of the courses offered in the training program and in the number of participants has been dramatic since the program's inception in 1973 (see Fig. 30). The continued vitality of this program demonstrates its important role in the transfer of safeguards technology to the nuclear community.

TABLE IX

**INTEGRATED SAFEGUARDS SYSTEMS—
CONCEPTS AND IMPLEMENTATION:
COURSE ATTENDEES**

March 22-24, 1977

<u>Organisation</u>	<u>Representative</u>
Allied General Nuclear Services	Michael Ehinger
Argonne National Laboratory	Richard E. Combs Fred W. Thalgott George Winslow
Battelle, Columbus Laboratories	Joseph F. Dettorre Homer M. Faust Harley Toy
Brookhaven National Laboratory	James R. Lemley
DuPont, Savannah River Plant	W. T. Dickenson J. E. Samsel Carl M. Schaumann William C. Scotten Robert V. Studley
Exxon Nuclear Co.	Joseph Ryden, Jr.
Goodyear Atomic Corp.	James R. Griggs Edward Jordan James R. Miller
IAEA	Carlos Buchler
Lawrence Livermore Laboratory	Nicholas J. Roberts
Oak Ridge National Laboratory	Edward J. Allen Edward D. Blakeman
National Bureau of Standards	B. Stephen Carpenter
Sandia Laboratories	Drayton D. Boozer Donald J. Gould
Westinghouse Hanford Co.	Robert L. Carlson
United Kingdom	Douglas Gordon
USERDA	Stephen M. Baloga James P. Hammond, Jr. Raymond E. Lang E. D. Marshall
USNRC	Stanley L. Dolins Donald S. Thurman Philip Ting L. F. Wirfs Edward F. Woltner

TABLE X

**INTEGRATED SAFEGUARDS SYSTEMS—
CONCEPTS AND IMPLEMENTATION:
COURSE CONTENTS**

<u>Tab</u>	<u>Title</u>	<u>Speaker and Affiliation</u>
A	Integrated Safeguards Systems— An Overview	D. D. Cobb, LASL
B	Conceptual Design of Coordinated Safeguards Systems—An Example	J. P. Shipley, LASL
C	A Review of Available NDA Techniques and Instrumentation	T. R. Canada, LASL
D	The Segmented Gamma Scanner	E. R. Martin, LASL
E	The Uranium Solution Assay System	J. W. Tape, LASL
F	The Neutron Well Coincidence Counter	M. Baker, M. Evans, LASL
G	Conventional Analytical Chemistry and Nuclear Accountability	E. A. Hakkila, LASL
H	Error Propagation and Measurement Control	W. R. Severs, LASL
I	Material Accountability—Modeling, Simulation, and Evaluation	D. B. Smith, LASL
J	Real-Time Data Base Management Hardware and Software	R. F. Ford, LASL
K	A Review of Available Physical Security Techniques and Equipment	J. D. Williams, Sandia
L	Perimeter Safeguards and Emergency Search Capabilities	C. N. Henry, LASL
M	Experience with Real-Time Nuclear Material Control Systems	R. H. Augustson, LASL

Fig. 30.

Number of participants in the USERDA Safeguards Technology Training Program courses by year.

PART 2

DETECTION, SURVEILLANCE, VERIFICATION, AND RECOVERY

GROUP Q-2

Carl Henry, Group Leader

E. J. Dowdy, Alternate Group Leader

The activities of Group Q-2 are directed toward the development of compact and highly sensitive instruments for the detection and surveillance of SNM. The group participates actively in the Nuclear Emergency Search Team (NEST), ERDA's program to provide immediate response to nuclear emergencies involving accidents, lost or clandestine materials, and terrorist threats. Q-2 provides personnel, techniques, and procedures for NEST exercises, and design data for new and/or improved portable and mobile nuclear search systems in support of the NEST program. Because much of the effort in this area is classified or sensitive, the work is reported separately in classified progress and topical reports.

Group Q-2 also applies nuclear detection and surveillance techniques to the problem of perimeter

safeguards at domestic and international nuclear facilities. This effort includes the design of hand-held personnel and vehicle monitors for searches at exits from SNM access areas, and the continued development, testing, and evaluation of SNM portal monitors and associated test methods and standards to evaluate performance of portal installations.

A substantial program is also under way to develop techniques for automatic and continuous monitoring of SNM in storage vaults. For long-term storage of significant quantities of SNM, such techniques may provide assurance in real time that no diversion has occurred and reduce the frequency or eliminate the need for periodic physical inventories.

I. SPECIAL NUCLEAR MATERIAL (SNM) MONITORS

A. Delta Rate Monitor (W. E. Kunz, C. N. Henry, and D. R. Millegan)

Performance testing of a new prototype hand-held monitor for SNM searches is under way. This monitor, the Delta Rate Monitor (DRM), operates in a manner similar to our previously described Personnel and Vehicle Monitor (PVM).¹¹ Each monitor accepts gamma-ray pulses from a NaI(Tl) detector, integrates for a preset counting interval (typically 0.3 s), and produces an audible alarm whenever the counts in the interval exceed the trip level. The

DRM and PVM logics differ only in the method of trip-level determination. The PVM's trip level is a preset multiple (typically 1.3 or 1.4) of the stored background count; the DRM's trip level is the stored background count plus a preselected number of counts, called the delta count.

The DRM is designed for operations in areas having higher than normal background radiation levels. With a fixed delta count added to the stored background to produce the trip level, the DRM's detection sensitivity is almost independent of background, whereas for a given increase of the

background, the PVM requires a corresponding increase in the source count to maintain the same detection probability.

With a 10-g sphere of ^{235}U , detection probabilities were measured for the PVM and DRM at different background levels. The PVM trip level was set at 1.4 times the mean background count per 0.3-s interval, while the DRM trip level was set at a delta count of 16 above the mean background count per interval. The ^{235}U source was moved past the monitors at 0.5 m/s, with the distance of closest approach equal to 0.25 m. Detection probability was defined as the number of passages during which detection occurred (alarm sounded during source passage) divided by the total number of passages. The results of the measurements are shown in Table XI. Whereas the detection probability of the DRM remains constant as the background increases, the detection probability of the PVM decreases significantly with increasing background.

Operation of the DRM at the high detection sensitivity obtained with a delta count of 16 is limited to background levels that do not produce an excessive false-alarm rate. An increase in the delta count reduces the false-alarm rate but also reduces the detection sensitivity.

B. ERDA Portal Monitor Evaluation (P. E. Fehlau)

We previously evaluated the IRT Corporation Model PRM-110 Doorway Monitor¹² and the National Nuclear Corporation Model DM-2

Doorway Monitor¹³ against the ERDA portal monitor standard¹⁴ and recommended that they be approved for ERDA use. During this reporting period we completed the evaluation of the Tom Scurry PM-203 portal monitor and forwarded a draft report¹⁵ to ERDA/DSS, recommending approval of the monitor for ERDA use.

C. Doorway Monitor Test Sources (P. E. Fehlau and T. E. Sampson)

1. Investigation of a ^{133}Ba Source as a Substitute for the Standard 1-g Plutonium Sphere

We compared a ^{133}Ba source and the 1-g plutonium-sphere ERDA portal monitor test standard to determine if a ^{133}Ba source of suitable activity might substitute for the 1-g plutonium sphere as a doorway monitor test source. Such a change would have advantages from both cost and health and safety standpoints.

Because there are no "standard" doorway monitors and no "standard" methods of operation, a substitute standard must be an accurate mockup for any detector type and any reasonable energy window.

The count rate in several typical energy windows was tabulated for an 8.8- μCi ^{133}Ba source and the 1-g plutonium sphere. These data were tabulated for NaI detectors of three different thicknesses: 1.27, 2.54, and 3.81 cm. The ratio of the count rate from the ^{133}Ba source to the 1-g plutonium sphere is tabulated in Table XII. This ratio is not expected to be unity because the ^{133}Ba source was not designed to be an exact intensity mockup. However, for the ^{133}Ba source to be considered a suitable substitute, these ratios must be constant both as a function of detector thickness and energy window. The two energy windows tabulated represent a typical wide-open plutonium window (70 to 452 keV) and a typical shielded plutonium window (330 to 452 keV). Americium-241 at 60 keV was deliberately excluded from the wide-open window because it varies with time in any plutonium source.

As a function of detector thickness, the wide window (70 to 452 keV) varies a total of 17%. For the narrow window, the variation is only 4%. These variations in themselves are not too serious, but the

TABLE XI

DETECTION PROBABILITY OF PVM AND DRM MONITORS

Mean Background (counts/interval)	Detection Probability	
	PVM	DRM
40	1.00	1.00
82	0.50	0.99
123	0.08	0.99

TABLE XII
COMPARISON OF ^{133}Ba SOURCE
WITH STANDARD
1-g PLUTONIUM SPHERE
(USING NaI DETECTORS)

NaI Detector Diameter x Thickness	Count Rate Ratio $\left[\frac{8.8 \mu\text{Ci } ^{133}\text{Ba}}{1\text{-g Pu sphere}} \right]$	
	Energy Window (keV) 79 to 452	330 to 452
12.7 cm x 1.27 cm	1.67	4.44
12.7 cm x 2.54 cm	1.83	4.27
12.7 cm x 3.81 cm	1.95	4.26

difference between the narrow- and wide-window ratios at a given detector thickness is more than a factor of 2, which means that this type of substitute source would work at only one window setting.

Similar measurements were performed with a 5.08-cm-diam by 0.91-m-long NE 102 plastic scintillator. Here, typical ^{238}U and ^{239}Pu windows were used. Results are given in Table XIII. The 11% difference between these numbers is not significant, but their disagreement with the NaI ratios in Table XII shows that the ratios are not constant between detector types (plastic scintillator and NaI).

TABLE XIII
COMPARISON OF ^{133}Ba
SOURCE WITH STANDARD
PLUTONIUM SPHERE
(USING NE102 PLASTIC SCINTILLATOR)

Energy Window	Count Rate Ratio $\left[\frac{8.8 \mu\text{Ci } ^{133}\text{Ba}}{1\text{-g Pu sphere}} \right]$	
	^{238}U	^{239}Pu
	2.12	2.36

If a substitute ^{133}Ba source were to be used, a specific source would be required for each detector type and window—clearly an undesirable goal.

2. ^{238}U Test Sources

During our evaluation of the Tom Scurry Associates PM-203 portal monitor, we made static measurements with a 10-g ^{238}U sphere (the ERDA ^{238}U standard source) and a 0.20-g ^{239}Pu sphere. The monitor had eight 5.08-cm-diam by 2.54-cm-thick NaI detectors and its energy window was set for detection of ^{238}U . These measurements and accompanying calculations indicate that a 0.25-g ^{239}Pu sphere would give the same static count rate as the ^{238}U standard source. Thus a monitor that meets the detection requirements with the standard ^{238}U source probably will meet the requirements with a 0.25-g ^{239}Pu source. We expect to receive a 0.25-g ^{239}Pu sphere from Group CMB-11 and will investigate its equivalence with the standard ^{238}U source in as many monitors as possible. If they are equivalent, then the 10-g ^{238}U sphere could replace the 1-g ^{239}Pu ERDA test source, or a 0.5-g ^{239}Pu source as required by NRC in R.G.5.27, or the 0.5-g ^{239}Pu source that appears in our proposed revision of R.G.5.27.

The Scurry monitor and the plastic NE 102 scintillator mentioned earlier were also used to make static measurements on 93% enriched U_3O_8 samples. This material is available from NBS as a standard reference material U-930 and is packaged and sold commercially as a doorway monitor test source by IRT Corporation. The measurement shows that self-attenuation is very important in this fine-powder sample and for consistent results one must package it in such a way that the material cannot move around. A 4-g U_3O_8 sample packaged inside a small cylindrical vial surrounded by a 0.3175-cm-thick brass container simulates the commercial source. The ratios of the U_3O_8 source count rate to the 10-g ^{238}U sphere count rate were 0.92 in the Scurry monitor and 1.02 for the plastic detector. Thus, such a source containing 4 g or less of U_3O_8 should be suitable for use by a licensee or contractor in routine doorway monitor performance checks. An IRT portal monitor is now available at TA-18 and

will be used to verify this conclusion. However, because the two detectors have different responses and the U_3O_8 source strength differs from the stan-

dard ERDA ^{235}U source, performance verification by NRC inspectors should be done with the standard 10-g ^{235}U sphere.

II. VAULT AUTOMATIC INVENTORY SYSTEMS

A. Real-Time Inventory (N. Nicholson, T. H. Whittlesey, T. H. Kuckertz,* and C. D. Ethridge*)

1. Introduction

The objective of the Real-Time Inventory task is the development of an integrated vault storage system that will measure and verify all incoming and outgoing SNM materials, detect unauthorized removals from the vault, and continuously monitor material while stored inside the vault. As presently envisioned, the system consists of a verification chamber, gamma-ray and metal detector portal monitors, an array of shelf detectors, and a computer for data storage and analysis.

All SNM material entering or leaving the vault must pass through the verification chamber which incorporates a scale, gamma-ray and neutron detectors, and possibly label-reading equipment. As the material enters the vault, a composite signature of the sample's weight, gamma spectra, and neutron emissions is obtained and stored in the computer. Whenever the sample is withdrawn from the vault its signature is remeasured and compared with the initial signature to determine whether the contents have changed during the storage interval.

Personnel entering or leaving the vault must pass through the portal monitors which are installed to detect passage of SNM. Inside the vault, an array of shelf detectors continuously monitors the sample's weight and gamma-ray emissions and transmits the information to the central computer which compares it to the expected values. Any discrepancy between the transmitted and the expected values will cause the computer to sound an alarm condition.

The Real-Time Inventory task has three phases. The first phase, currently in progress, consists of the

design, assembly, and checkout of an experimental 16-can system at Pajarito Site (TA-18). During Phase II a demonstration system will be installed in the operating vault at DP Site (TA-21) and expanded to monitor several hundred cans. During Phase II we will investigate the system's capability to perform satisfactorily in a real vault environment. During Phase III a vault system will be designed and integrated into the DYMAC system at the Plutonium Reprocessing Facility (TA-55).

2. Phase I--Experimental System at TA-18

The NOVA 3/12 computer arrived and has been checked out. The ORTEC CAMAC crate controller unit, which is to feed data from the verification station to the computer, arrived but had to be returned to the vendor because of an electronic malfunction. An IRT Doorway Monitor arrived and is to be used in conjunction with the verification station at the entrance to the vault.

Attempts to reduce the cost and complexity of the shelf monitors resulted in a new version (MOD 4), which uses an INTEL microprocessor. Three months ago, INTEL introduced the MCS 48 single chip microcomputer that is a 40-pin chip containing an 8-bit CPU (central processing unit), 27 I/O lines, an internal timer/event counter, oscillator and clock driver, reset circuit, and an interrupt circuit. In short, this chip is a complete computer. Two versions are being considered. The 8748 is the EPROM (erasable programmable read only memory) version that contains 1024 words of ultraviolet erasable programmable ROM (read only memory). The 8048 is the nonerasable equivalent of the 8748 and is available in quantity at \$10 per copy. In comparison with the MOD 3 design, the 8048 reduces the chip count from 40 to 6 and reduces the electronic component cost from \$75 to \$35. The MOD 4 version still

*Group E-3.

retains the use of the UART (universal asynchronous receive and transmit) although a MOD 5, to be developed, will eliminate the need for the UART, thereby reducing the component cost another \$7.

Twenty-five 8748 microprocessors have been ordered and will be used in the 16-can demonstration system that is being assembled at TA-18. The use of the 8748 version in this system permits software development and modifications within the shelf detector without the problem of hardware changes that would accomplish the same thing. The 8048 will be used in the DP-21 detectors.

The 8748s will be delivered by May and we anticipate that the MOD 3 detectors can be completely bypassed in favor of the superior MOD 4 version without seriously delaying our schedule. In fact, the MOD 4 circuitry, which will be easier to debug than MOD 3, should result in time savings.

A new design for the lead collimator surrounding the GM tube improves the shielding of the tube from unwanted gamma radiation, improves the directional view of the detector so that it sees only the material directly above it, and reduces the fabrication cost.

The rest of the components required for the MOD 4 shelf detector, such as the parallel plate weight sensor, GM tubes, rainboxes, etc., are being collected. Detector assembly will begin as soon as the new boards incorporating the 8748 microprocessors are available. The estimated cost of the weight sensor is about \$15, and the total cost for the MOD 4 detector is approximately \$75.

B. Hanford Verification Station (N. Nicholson)

The present plutonium protection system being designed by Sandia Laboratories for Atlantic Richfield Hanford Co. (ARHCO) does not verify the presence or amount of SNM within the sealed containers as they enter the vault. DSS has requested LASL to provide a verification station specifically designed for Sandia's 280-can demonstration system to identify (Phase I) and quantify (Phase II) the presence and amount of SNM entering and leaving the vault. This verification station will furnish assurance that an omission or substitution, either

deliberate or accidental, has not occurred in shipping, handling, or packaging of the SNM in the interval between the last process assay and the entry into the Sandia-designed plutonium protection system.

The Phase I verification station, comprised of inexpensive instrumentation capable of detecting gamma and neutron radiation, was completed and delivered to Sandia. The station will provide operating experience and demonstrate the utility of a verification station at ARHCO.

We are now working toward a more sophisticated Phase II station which will provide measurements as close to a bona fide assay as is practicable. In preparation for the design of the Phase II verification station, we conducted a survey of selected facilities involved with the storage and safeguarding of SNM to determine whether a generic storage safeguards system could be designed to accommodate a large number of facilities. Ten of the thirteen facilities have responded to the questionnaire that was sent out in September 1976 (the last reply arrived late in February 1977). Two of the facilities that did not respond were concerned about the sensitive nature of the information they were asked to supply.

In October 1976 we visited five storage facilities, including HEDL, ARHCO, Pacific Northwest Laboratories (PNL), and Exxon (all in the Richland, WA, area), and GE Vallecitos, to gain some insight into the variety of chemical and geometrical SNM storage configurations. We gathered other useful information, such as unique problems and attitudes, storage inventory, and sealing techniques, which will be helpful in the design considerations of the Phase II verification station. In essence, the survey shows that the factors involved in the design of a portal security system or a real-time inventory shelf system vary substantially from facility to facility. Because of these variations, a universal safeguards storage system cannot be designed unless significant compromises are made on the acceptable accuracy of the proposed system.

When applied to the proposed verification station for the 280-can Sandia demonstration system program, this conclusion suggests that the design of the station should be directed specifically at the Sandia/ARHCO demonstration system.

PART 3
SAFEGUARDS SUBSYSTEM DEVELOPMENT AND EVALUATION

GROUP Q-3

Ronald H. Augustson, Group Leader

Group Q-3 is responsible for developing, implementing, demonstrating, and evaluating the DYMAC program. DYMAC is an in-plant materials control system that uses a plantwide network of in-line NDA instruments coupled with automatic data-processing equipment to continuously pinpoint the quantity, form, and location of SNM throughout an operating nuclear plant.

The DYMAC concept was developed because of the shortcomings of present in-plant material control methods that lack timeliness, are relatively insensitive, and levy a severe economic penalty in frequent plant shutdowns and cleanouts. DYMAC overcomes the shortcomings of present methods by measuring the material in-line, by using modern NDA technology, and by transferring the data to an on-line computer system that provides real-time data-base management.

DYMAC subsystems are: NDA instrumentation, data acquisition, data-base management, and real-time accountability. The NDA instrumentation subsystem provides rapid, quantitative on-line measurement for SNM as it moves from one unit process accounting area to another. The data-acquisition subsystem provides accurate and reliable material control information to the central computer. The data-base management subsystem accepts and organizes incoming data into files for efficient retrieval of specific information. The real-time accountability subsystem draws on the data base for continuous status monitoring of the nuclear material within the facility. On detecting an abnormality, it signals the condition and provides inventory data to aid plant personnel in determining the appropriate response.

A DYMAC system is being integrated into the new LASL plutonium processing facility at TA-55 to serve as a working demonstration (DYMAC/TA-55). DYMAC/TA-55 goals are to demonstrate the reliability and operational feasibility of in-line NDA instrumentation in a production environment, accurate and efficient data generation, sensitivity to missing material, and compatibility with production control and quality assurance in a cost-effective manner.

The DYMAC/TA-55 program is divided into three phases. Phase I was a pilot program of research and development and in-plant testing of NDA instruments coupled with an interactive computer system at the LASL DP-Site facility. This phase has been completed.

In Phase II, currently under way, the technology and equipment developed in Phase I will be integrated into the new plutonium facility. DYMAC/TA-55 will be a full-scale, plantwide DYMAC system with NDA measurement stations in each unit process accounting area, interactive computer terminals, and real-time SNM accountability and control. In Phase III the data collected during operation of all subsystems will be evaluated in depth, focusing on effectiveness of material control, reliability, operational ease, costs, and applicability to other facilities.

DYMAC/TA-55, in addition to demonstrating feasibility, will provide guidance for the direction of research and development in nuclear material control technology and operating data for the modeling studies that are used extensively in integrated safeguards systems designs. In serving as a working example, it provides the basis for transferring

technology to the nuclear industry. To meet this objective, Group Q-3 will prepare complete technical documentation (e.g., reports and manuals), develop training courses, and serve as liaison with organizations and facilities seeking DYMAC technology and capability.

During this reporting period, the test and evaluation program at the DP-Site plutonium facility was completed and the focus of the DYMAC effort turned toward the system being installed at the new

TA-55 facility. For the next year and a half, DYMAC personnel will work toward procuring, installing, and checking out the equipment and software necessary to implement dynamic materials control at the new facility. In addition, they will continue to develop technology in materials control and to exchange technical information with the safeguards community. Typical of such exchanges are two technical gatherings hosted by the LASL safeguards staff in January and March 1977.

I. CONCEPTS AND SUBSYSTEM DEVELOPMENT

A. NDA Instrumentation

1. Studies of Thermal-Neutron Coincidence-Counting Techniques (N. Baron)

Reported here are the results of further studies of passive thermal-neutron coincidence-counting techniques that could be used to perform accurate NDA of spontaneously fissioning material. These data were obtained with a shift register neutron coincidence module⁸ and a well-counter¹⁰ with two concentric rings of ³He tubes placed symmetrically about the well. The "coincidence" and "single" count rates for 16 different standards were measured for the inner ring alone, for the outer ring alone, and for the simultaneous response of both inner and outer rings. These standards were either relatively pure plutonium oxide, or plutonium oxide mixed with either ash, diatomaceous earth, uranium oxide, or magnesium oxide. Total plutonium in the samples ranged from 55.7 to 779 g, and the effective mass of ²⁴⁰Pu ranged from 3.51 to 81.4 g. Table XIV shows the matrix and plutonium loading of each standard along with the measured coincidence and single count rates. Using these data, one could empirically define a function of the coincidence and single count rates measured in the inner and outer rings that is linearly proportional to an empirically adjusted power of the effective mass of spontaneously fissioning material. This relation is

$$\frac{\sqrt{\dot{N}^0} \left(\frac{\dot{I}^I}{\dot{I}^O} \right)^2}{(M_E)^{0.53}} = \text{constant} \quad (7)$$

where $\dot{N}^0$ is the coincidence count rate measured by the outer ring of tubes, $\dot{I}^I$ and $\dot{I}^O$ are the single count rates of the inner and outer rings respectively, and M_E is the effective mass of ²⁴⁰Pu. The measurement uncertainty of $\dot{N}^0$ was about 1%, and for $\dot{I}^I/\dot{I}^O$ about 0.1%. The proportionality constant is given in Table XIV for each of the measured standards. The standard deviation of the mean value of the proportionality constant for the 16 measured standards is 1.38%. The relative standard deviation of the specific response rate (coincidence counts/gram of ²⁴⁰Pu effective) calculated using the relationship of Eq. (7) is 2.60%. Assumptions of a linear relation between the effective mass and the coincidence count rate from the simultaneous response of all the detector tubes, $\dot{N}^0$, results in a 7% relative standard deviation in response per gram. The empirical expression shown in Eq. (7) is more straightforward than previously reported,¹ yielding an improvement in the measurement accuracy. The expression linearizes the calibration curve, thus reducing the variety of standards needed to perform nondestructive assay. Table XIV data indicate that one can expect the specific response rate of effective ²⁴⁰Pu to have a relative standard deviation of less than 3% for the measurement of PuO₂ powders ranging in plutonium mass from several to about 1000 g and including a variety of matrix materials.

Theoretical studies in Group TD-6, LASL, are now being made to gain greater insight into the physical significance of such an analysis using a double-ring well-counter.

TABLE XIV

THERMAL-NEUTRON COINCIDENCE-COUNTING DATA

Source ID	Material	Mass of Plutonium (g)	Effective N_{eff} of ^{239}Pu N_1 (g)	Coincidence Count Rate			Single Count Rate		$\frac{N_1}{N_2}$	$\frac{\sqrt{N_1}}{N_2} \left(\frac{t_1}{t_2} \right)^2$
				Inner Ring, N_1	Outer Ring, N_2	Both Rings, N_3	Inner Ring, N_1	Outer Ring, N_2		
AB 24647	Ash	55.20	1.507	10.54	20.77	62.51	1.779	2.408	17.82	1.197
STD 4	Distomaceous earth	60.02	1.017	11.27	22.29	65.62	774.5	1.005	17.10	1.179
STD 5	Rq0	60.01	1.010	12.51	24.75	74.41	1.706	2.599	19.39	1.152
STD 1	Distomaceous earth	60.29	4.651	17.00	27.73	78.45	921	1.788	16.72	1.107
STD 11	Rq0	66.40	5.460	18.75	35.71	104.10	701	1.001	19.07	1.100
AB 24624	Ash	110.6	7.392	23.77	40.29	141.07	4.160	5.840	19.45	1.210
STD 6	Distomaceous earth	120.0	7.672	27.83	46.33	135.77	1.364	1.920	17.64	1.165
STD 7	Rq0	120.0	7.672	28.50	55.27	151.04	2.781	1.909	19.69	1.192
STD 20	Rq0	100.1	9.410	32.17	61.39	191.27	1.006	1.530	19.26	1.201
STD 14	Rq0	143.0	13.01	40.71	94.97	275.07	1.741	2.575	19.92	1.199
STD 8	Rq0	280.3	15.35	57.61	111.57	311.47	1.731	5.310	21.14	1.193
STD 9	none	480.4	37.39	137.06	260.00	771.44	4.610	6.634	20.61	1.172
PIO 005	none	459.0	41.39	136.44	302.71	808.69	5.050	7.174	20.02	1.169
PIO 302	none	556.0	54.91	204.61	398.71	1.120.2	6.520	9.777	20.39	1.100
PIO 001	none	615.0	64.97	231.47	471.81	1.157.6	7.527	10.702	20.80	1.170
PIO 447	none	779.0	81.36	321.07	611.70	1.714.6	9.819	13.402	21.07	1.194
Standard Deviation									6.99%	1.30%

2. Straightforward Measurement of Die-Away Time for a Thermal-Neutron Coincidence-Counting System (N. Baron)

The die-away time τ of a coincidence-detection system represents the time interval at the end of which the detection probability of the second member of a coincident pair is reduced by e^{-1} . The die-away time can be measured readily by using the electronics necessary for a typical coincident count rate measurement. Recognizing that the detection probability for the second member of a coincident pair at time t seconds after detecting the first member is given as

$$P = e^{-t/\tau},$$

Then, over a period of time, the number of true coincidences N expected if the members of the pulse pair are separated by a time interval ≥ 0 and $\leq t$ is given by

$$N(\text{gate width} \approx t) = \int_0^t N(0)e^{-t/\tau} dt$$

$$= \tau N(0) (1 - e^{-t/\tau}). \quad (8)$$

The expected number of true coincidences for an infinitely long gate is

$$N(\infty) = \tau N(0).$$

Using the simultaneously measured number of true coincidences, $N_1(t_1)$ and $N_2(t_2)$, for gate widths t_1 and t_2 , respectively, in Eq. (8), we have

$$\frac{N_1(t_1)}{N_2(t_2)} = \frac{1 - e^{-t_1/\tau}}{1 - e^{-t_2/\tau}},$$

which allows a solution for the die-away time τ .

II. DYMAC IMPLEMENTATION

A. DP-Site Test and Evaluation Phase (R. H. Augustson)

As previously reported,¹ the effort at DP Site was directed toward gaining working experience preparatory to installing a DYMAC system at the new TA-55 facility. The program proceeded in four areas. First, a nuclear materials accounting system, which emulated the present LASL system, was programmed to run on a NOVA 840 computer located at DP Site. Material transactions (i.e., records describing material movement and/or processing) were entered into the data base via interactive terminals, in contrast to the present LASL system which uses handwritten paper transaction forms. A series of parallel runs was made in which transactions were entered into both systems and the resultant book inventories were compared. These parallel runs verified that the terminal system did indeed exactly duplicate the paper system.

During this development, NDA instruments installed and operated in a glove box environment included a solution assay instrument (SAI) based on gamma-ray measurements, a thermal-neutron well coincidence counter (TNWCC), load cell weighing systems, and a digital electronic balance.

The NDA instrumentation was integrated, along with a more sophisticated version of the NOVA 840 accounting program, into a prototype DYMAC system capable of drawing unit process balances around a portion of the DP-Site process line. This prototype system entered transactions from terminals located adjacent to the process lines.

The fourth area covered in the DP-Site test and evaluation program was measurement control. All data, in particular all measurement data entering the system, must have a high level of integrity if the overall system is to be sensitive to missing material. A manually recorded measurement control program for the NDA instruments was designed and implemented to assure data integrity. To date, these control procedures have detected two problems. In one case, the problem was noisy electronics; in the other, an improper background subtraction procedure. Operation of the control program for the equipment in the DP-Site counting room will con-

tinue even though the other activities at DP Site have terminated. At TA-55 these control concepts will be extended to include computer monitoring and record-keeping of the program's day-to-day operation.

1. DP-Site Prototype System (J. Hagen, W. M. Olson,* and R. P. Wagner*)

The DYMAC team at DP Site has completed a 2-1/2-month effort to provide a prototype process and accountability system for the new FFTF oxide production process. The computer was dedicated to transaction entry on a full-time basis. Two team members followed the process and entered transactions from terminals in the plant. The primary goal was to keep an up-to-date inventory record using off-line NDA instrumentation and an on-line terminal connected to the NOVA 840 computer. Because of the on-line inventory feature, it was possible to minimize the information that the operator has to enter at the terminal. For example, in most cases he supplies the item name, his own name, the processing step, and the amount of material to be transferred. Other information, such as the percentage of ²⁴⁰Pu, is supplied from the inventory file. Transaction formats have been developed so that the information needed at a given process step is specifically requested by the computer.

The first process flow definitions were presented to the software support people in early December 1976. Subsequently the process underwent several revisions. The DYMAC team was finally able to define four accounting areas based on the FFTF oxide production unit processes. They are the receiving, cutting, dissolution, and screening balance areas.¹

a. Receiving Balance Area. Input is the shipper's value for the weight of the ingot. Measured output: are the weight of the oxide scraped from the

*Group CMB-11.

ingot, the clean ingot weight, and the amount of plutonium (determined by the segmented gamma scanner [SGS]) remaining in the shipping materials. Any material remaining on the books after these transactions (i.e., the difference between the input and output values) is assigned to the shipper-receiver difference account.

b. Cutting Balance Area. Input is the clean ingot weight. Measured outputs are the weights of the four quarters, the metal sawdust, and the plutonium (determined by the SGS) in the worn-out sawblades. The process has changed from sawing ingots to turning shavings on the lathe; it no longer generates metal sawdust and worn-out sawblades. Material remaining in the inventory record is assigned to the material-in-process (MIP) account for the cutting box.

c. Dissolution Balance Area. Input is the weighed metal in the form of chunks, sawings, or turnings. Pieces from several ingots frequently are mixed in a dissolution lot. Measured outputs are the weight of the nitrate solution filter residues, the weight of undissolved metal, the plutonium in the peroxide and oxalate filtrates (determined by the SAI), and the weight of the calcined oxide. The remaining inventory is assigned to the MIP account for the specific dissolution area.

d. Screening Balance Area. Input is the weighed calcined oxide. Output is the measured PuO_2 product. Differences between input and output SS values* were assigned to a MIP account for this screening area.

Oxide product is weighed, canned, and then measured with the neutron coincidence counter (NCC). The DYMAC prototype system provides transactions to adjust the SS value of the oxide product when the chemistry results become available at a later time. Thus, the NCC method provides an immediate measurement which can subsequently be upgraded by the analytical chemistry value.

Process transactions are of two types: transactions used during normal processing, designed to be entered by technicians, and transactions that deal

with nonphysical accounts, to be entered by supervisors. Supervisory transactions access MIP and shipper-receiver difference accounts, make adjustments to NDA-measured items, and correct transactions. Figure 31 shows option lists for both types of transaction. Technicians can access the first type only; supervisors can access both.

- OPTIONS
- 1= ENTER BUTTON
 - 2= SCRAPED OXIDE
 - 3= CLEAN INGOT
 - 4= SHIPPING PS
 - 5= QTR INGOTS
 - 6= CUTTING PS
 - 7= QTRS STOR TO PROC
 - 8/9= NS SOLN (ALL/PART)
 - 10= RESIDUES
 - 11= UNDISSOLVED METAL
 - 12= FS SOLN
 - 13= FS TO PEROXIDE CAKE
 - 14= FS REDISSOLVED
 - 15= FS TO OXALATE CAKE
 - 16= CALCINE
 - 17= PEROXIDE FILTRATE
 - 18= OXALATE FILTRATE
 - 19= FILTRATE TO ION-EXCHANGE
 - 20/21= SCREENING (521,523 / 528)
 - 22/23= SAMPLES (CMB-1 / ARCHIVE)
 - 24= CANNED OXIDE (NCC)
 - 25= REJECTED OXIDE
 - 26= PROCESS SOLIDS TO DRUM
 - 27= *** GENERAL TRANSFER ***
 - ENTER OPTION *
- (a) Option list for technician.
- SUPERVISORY TERMINAL OPTIONS
- 1 = SHIP/REC DIFFERENCE
 - 2 = CUTTING MIP
 - 3 = MF/FS MIP
 - 4 = SSA CORRECTION
 - 5 = SCREENING MIP
 - 6 = NCC CORRECTION
 - 7 = TRANS REVERSAL
 - ENTER OPTION *
- (b) Option list for supervisor.

Fig. 31.

Transaction option lists for the FFTF oxide production process. Technicians can access only the list displayed in (a). Supervisors can access either list.

*The SS value is the amount of nuclear material contained in the material undergoing processing.

Process transactions lessen the amount of information the operator must enter to write a transaction. They perform diagnostics on his entries which result in fewer data input errors. The process transaction approach can respond quickly and relatively easily to process modifications. Likewise, the computer program can be modified in a remarkably short time. For example, the chemical process changed considerably from mid-December 1976 to mid-February 1977 during DYMACE system testing. Most process changes required a corresponding modification to the software system. The option lists went through three major and several minor revisions in the 2-month period.

After the system is installed and in full operation, modifications to a transaction format will be given to the DYMACE programmers with enough lead time to allow implementation and checkout during non-production hours. There are two procedures for emergency situations: a change can be made during regular working hours (which takes 15 to 30 min of computer down-time) or the technicians can temporarily use a general transfer* until the change is implemented. Neither procedure would seriously affect those who depend on the DYMACE system to make transactions.

The prototype system has begun providing for corrective transactions in response to the need for a straightforward way to "back out" a transaction. The prototype saves the supervisor from having to look up such information as accounts, receipt areas, and item descriptions. A REVERSE option was implemented which allows the supervisor to scan the TRAN (transaction activity) file, identify a particular transaction, and edit it, after which the system reverses the transaction. The capability to scan the TRAN file has proved useful in other instances, too. All corrective transactions are flagged with a special transaction number to make them easily distinguishable.

Accountability and process personnel benefit from proper setup of accountable balance areas. For example, MIPs have been categorized according to

their cause. Separate "bag-out" of shipping materials and cutting materials gives more accurate values for the shipper-receiver difference and the cutting MIP. Accurate measurement of calcined oxide into the screening area, and of finished product out of the area should give a good MIP value for that box. That value is, in turn, important to the criticality limit for the screening box.

In-line measurements greatly benefit the process by helping to keep the inventory record current. A prime example is blending of nitrate solutions to form feed solutions. The present accounting system must wait until all metal is either dissolved or removed and weighed, and residues are completely filtered, dried, and then weighed before any transactions can be entered for the combining of nitrate solutions into a feed solution. At present, both DYMACE and the paper system calculate the SS value of the solution transferred as a ratio of the volume transferred to total volume of the filtrate. Even though processing continues through precipitations of the blended feed solution, the inventory record must wait for a dried residue weight and a total filtrate volume. A better method, one which would keep the inventory record current, would be to provide a plutonium value from the in-line SAI for each transfer to a feed solution lot.

Processing and accountability must go hand in hand. Good processing procedures are essential to accurate accounting and timely measurement. Up-to-date accountability data definitely benefit the process.

A total of 56 000 g of plutonium was processed during the period in which the prototype system was operating. The cumsum material balances in the four accounting areas at the end of operation were: receiving area, 15 g; cutting area, 10 g; dissolution area, -48 g; and screening area, -566 g. The screening balance was due to a bias in the NCC measurements (which is being corrected) and the fact that the chemistry values were not available at the time the balances were drawn.

The operation of the prototype system will be described in a final detailed report now being written.

*A general transfer requires the technician to enter all the information to make a transaction; this is the long form which can substitute for any process transaction.

2. Solution Assay Instrument (SAI) (J. L. Parker, W. R. Severe, and R. P. Wagner*)

The SAI^{3,17} was originally designed to measure ash-leach solutions with concentrations ranging from 0.5 to 20 g of plutonium per liter of sample. During the processing of FFTF plutonium oxide, we needed to measure peroxide filtrate containing lower concentrations of plutonium but high concentration of americium. While attempting these measurements, we discovered that the 208-keV peak from ²⁴¹Am and ²³⁷U was so intense as to interfere with the 414-keV ²³⁹Pu peak through the pulse pile-up mechanism. Further development is needed before this type of solution can be measured practically in the present assay system.

3. Experience with the Measurement Control Program (W. R. Severe)

The measurement control program, previously described,¹ has been officially implemented as an operating procedure for the DP-Site counting room. Many determinations of the control parameters will be required before the program can be thoroughly evaluated. However, preliminary analysis indicates that the control parameters are behaving as expected. The measurement control program for the TA-55 facility probably will be based on the measurement control program for the DP-Site counting room.

Procedures similar to the measurement control program are being used to evaluate instruments in other areas at DP Site. In particular, we have been applying control procedures and evaluation techniques to load cells and electronic balances which are located inside gloveboxes.

B. DYMAC for the New LASL Plutonium Facility

1. DYMAC Instrumentation (R. S. Marshall, V. S. Reams, N. Baron, and M. M. Stephens)

A number of instruments are being developed and tested for on-line use with the DYMAC system. In

the development phase are the SAI and the fast-neutron coincidence counter. The thermal-neutron well coincidence counter and electronic balances are being tested

Thermal-Neutron Well Coincidence Counter (TNWCC). The first TNWCC for the new plutonium facility has been fabricated and assembled. The unit was designed for installation in the Recycle Wing at TA-55. It is currently being used to test and evaluate mechanical, electronic, and neutronic parameters which will apply to most of the TNWCCs installed at the new facility.

Figure 32 illustrates the layout of major TNWCC components. As mentioned in Ref. 1, the counter has a double-range capability. Some counting samples have a sufficiently high neutron flux to "swamp" the electronics when all 20 ³He tubes are used. In such cases, only 4 of the 20 tubes (one per quadrant) may be used as active detectors. The polyethylene, which houses the tubes, and the collection ring are divided into quadrants to facilitate assembly underneath the gloveboxes.

New amplifier and preamplifier designs are being used for initial testing of the TNWCC. The shift register now in use is an updated version of the model described in Ref. 8. It is being replaced by a new unit with much faster electronics which will be capable of processing rates of 100 000 counts/s.

Mechanical components for three TNWCCs to be located in the ²³⁹Pu R&D Wing are being fabricated.

Fig. 32.

Thermal-neutron well coincidence counter designed for placement underneath gloveboxes.

*Group CMB-11.

and machine work for the ^{239}Pu R&D Wing and Metal Fabrication Wing counters will go out for bid in a few weeks.

Data analysis electronics are in the process of specification and design.

Solution Assay Instrument. On the basis of DP-
Site experience, a final design has been established
for the detector, sample holder, and data acquisition
and analysis electronics. Group Q-1 has taken the
responsibility for the electronics design.

Fast-Neutron Coincidence Counter. On the
basis of measurements made with a random-driver-
type device, Group Q-1 is developing the measure-
ment system for high-density samples with large
(α, n) backgrounds.

Electronic Balances. We evaluated electronic
balances from three manufacturers and were able to
remote all three balances without difficulty.
(Remoting refers to the separation of the weight-
sensing portion and electronic readout portion of the
balances.) Balance performance was not adversely
affected by remoting with 8 m of coaxial cable.
When the balances are installed, the weighing por-
tion will be located in the corrosive glovebox en-
vironment and the electronic portion will be moun-
ted outside.

One of the balances uses knife edges. The unit was
installed in a working glovebox and used for 5
months. During that period it required servicing on
five occasions. We observed two malfunctions:
failure to return to zero after a weight was removed
from the balance pan, and increase in bias when a
weight was placed progressively farther away from
the pan's center. We attribute both malfunctions to
knife-edge problems. Eventually, errors up to 10 g
for a 20-kg object were observed. The balance was
dropped from consideration and will not be used for
the DYMAC system at the new facility.

After a review of the future users' (Group CMB-
11) requirements, final specifications were written
for balances to be used at the new facility. Table XV
lists requisite weighing ranges and precisions.

TABLE XV

BALANCE REQUIREMENTS FOR TA-55

Weight Range (kg)	Precision (g)	Quantity Needed
0-5.5	0.1	26
0-5.5	0.01	6
0-15	0.1	3

2. TA-55/DYMAC Communication System (R.
B. Glascock* and K. A. Lindsey)

The communication system that will interconnect
instruments and terminals with the Eclipse C330
computer has been described in detail.¹ During this
reporting period we were planning the cable system,
buying terminals, and designing and testing 20-mA
current-loop interface circuits.

The number of terminals and instruments to be
connected has fluctuated, causing a slight modifica-
tion in the number of cable pairs to be installed.
Present plans are summarized in Table XVI. To ex-
pedite installation, approved drawings are being
made available to the installers. These drawings will
be revised to reflect changes made during the in-
stallation period. As soon as the installers are
available, construction can begin.

The status of various purchases, construction
work, and equipment installation is:

- The terminal boxes have arrived.
- The 2-pair cable has been shipped.
- The 24-pair cable is to be shipped on May 6, 1977.
- The jackfield is installed in the rack.
- Most of the new floor penetrations have been drilled.
- The components for DYMAC computer-room junction boxes have been shipped.
- Cables with connectors for the computer room have arrived.

*Group E-5.

TABLE XVI

TA-55/DYMAC INSTRUMENT LOCATIONS BY WING

Wing	Wire Pairs	Possible Locations	Number of Instruments	Number of Terminals		Spare Locations
				CRT	Hard Copy ^a	
1	144	72	11	8	1	52
2	120	60	6	2	1	51
3	144	72	19 ^b	8 ^c	3 ^c	42
4	168	84	34	8 ^d	3 ^d	39
Total	576	288	70	26	8	184

^aHard-copy units will all be 30-character/s terminals.

^bIncludes four spare outlets in Room 124 of the cold support building.

^cIncludes terminals in cold support building.

^dIncludes terminals at the basement vault.

- The connectors for one end of the cables that connect the rack to the CPU in the computer room are to be shipped on May 19, 1977.

- The WIRERUN computer program should be complete by May 23, 1977.

- Revised drawings have been received from Group ENG-2.

3. Computer Procurement for TA-55 (R. F. Ford,* J. Hagen, and C. A. Slocomb*)

The Eclipse C330 computer system arrived and was accepted. Figure 33 is a photograph of the system components. Provisions are being made for additional air conditioning. Data General has signed a maintenance contract to provide 4-h response to service calls.

*Group E-5.

4. Program Development at TA-55 (R. F. Ford,* J. Hagen, C. A. Slocomb,* and T. M. Sandoski*)

The accountability program that was running on the NOVA 840 at DP Site is now running on the Eclipse C330 at TA-55. The data-base management portion of the program has been converted to run with the INFOS system, and is now being debugged. The contents of each record in the inventory, transaction, and incomplete transaction files have been defined. The methods of producing and reporting TA-55 transactions to the laboratory-wide accounting system have been defined, and programs are being written to demonstrate the ability of the DYMAC system to send information to that accounting system.

Traffic simulations have begun in order to determine the maximum volume that the Eclipse C330

*Group E-5.

Fig. 33.
Eclipse C330 computer at TA-55.

and the network can sustain before degraded performance results. This maximum traffic point must be established so that the proper trade-offs on future requirements can be made. We simulated a situation in which 24 lines transmit constant data at 1200 baud, 24 different lines receive it (the equivalent of 24 people typing at a rate of 120 key-strokes/s), and one person types in data at a terminal. For this simulation we detected no degradation of the system and plan to conduct more tests at increased rates and with more lines.

Process sheets for the various process lines are being drawn up. When all the information is filled in, we will be able to design and code the various packets and image files.

5. Backup Procedures at TA-55 (R. F. Ford,* C. A. Slocomb,* and J. Hagen)

The failure rates for all system components, where known, have been calculated and are shown in Table XVII. In addition, several computer maintenance groups were canvassed for more heuristic failure rates. This, coupled with our local experience, has dictated the following initial hierarchy of back-up. During daily operation, identical sets of data files are maintained on separate disks and an additional copy of the TRAN file is maintained on magnetic tape. At the end of the day the two disks are compared and, if they match, they are backed

*Group E-5.

TABLE XVII
COMPONENT FAILURE-RATE MATRIX

<u>Component</u>	<u>Number in System</u>	<u>Mean Time Between Failures (h)</u>	<u>Failures/ million h</u>	<u>Failures/ 50-h-wk</u>	<u>Failures/yr^a</u>
Computer Subsystem					
C330 with 32k of core	1	2 300	435	0.0218	1.09
38.4-cm computer board	18	28 570 ^b	35 ^b	0.0018	0.0875
Expansion chassis	1	26 238	38	0.0019	0.0950
5-million-word disk unit	2	4 000 ^b	250 ^b	0.0125	0.625
Mag tape unit	2	2 000 ^b	500 ^b	0.0250	1.25
Line printer	1	2 000	500	0.0250	1.25
Communications chassis	2	16 800 ^b	60 ^b	0.0030	0.150
System CRT	1	2 000	500	0.250	1.25
Terminals					
Telray	27	6 350	157	0.0079	0.393
T1	7	---	---	---	---

^aBased on 52 wk of 50 h each.

^bEach.

up on two separate magnetic tapes. If they differ, the Safeguards Officer and CMB-11 personnel will determine which file is correct. If this determination cannot be made quickly, both sets of files will be saved until it can be ascertained which set is correct. At the end of a reporting period (still to be determined) the TRAN file will be backed up by magnetic tape and a new tape will be started.

If a disk drive fails during the day, the activity will continue with use of the other disk. If they both fail, or if the computer fails, the only loss will be the transactions currently in progress because we will have yesterday's ending files and today's activity to date (TRAN) on a magnetic tape. If both disks fail and the remainder of the system is operative, a

nondisk-based program will be loaded from magnetic tape which will enable personnel to continue to make transactions. This system will be very similar to the current paper system in that it will only gather the transaction information without immediately updating the files. Some diagnostic capability will still exist, but file-searching will be temporarily lost. Recovery involves inputting the magnetic tape copies of today's transactions into yesterday's ending activity and updating the files.

In the event that the DYMAC computer is inoperative, eight terminal lines (for example, two in each wing) will be patched to a similar transaction-gathering program on the NOVA 840 which normally is used as the training computer. If the NOVA

840 fails while we are in this mode, the eight terminal lines will be patched to the background of the building backup computer. In these backup modes we will also have lost the ability to read NDA instruments on-line and will have to rely on manually entered data. As before, recovery involves inputting the collected transactions back into the DYMAC system and updating the files. After this type of

failure, a "book" will immediately be distributed to all custodians for book physical inventory difference (BPID) check because the information was obtained in a reduced diagnostic mode. Long-range plans call for continued evaluation of failures which may entail update or modification of these backup procedures.

III. TECHNOLOGY TRANSFER

(D. C. Amsden and R. H. Augustson)

Two technical meetings were held during this reporting period: a topical workshop and the Integrated Safeguards System Training Course which has been described in Part I-II-J of this report.

The DYMAC group hosted a safeguards topical workshop at LASL on January 18 and 19, 1977. Its purpose was to allow an exchange of technical information and in-plant experience with safeguards techniques and instrumentation. Twelve visitors came from Rocky Flats, Oak Ridge Y-12, ERDA, and NRC. Table XVIII contains a list of participants.

The real-time accountability programs at Rocky Flats, Y-12, and LASL were discussed. The three programs have similar problems, such as developing operating concepts, defining algorithms for instrumentation, defining material balance areas, planning the interaction between the system and operating personnel, and establishing data diagnostics and the appropriate action to be taken in the event of an error. The program approaches differ somewhat because of the type of material processed, the stage of evolution from a paper accounting system, the type of communications and computer hardware and software already in existence, and differences in who will be using the terminals, i.e., floor-level operators or supervisory personnel.

The LASL contingent gave detailed lectures of the present accounting system at DP Site and the projected system for the new plutonium facility at TA-55, reinforced with tours of the two facilities.

TABLE XVIII
TOPICAL WORKSHOP PARTICIPANTS

<u>Organization</u>	<u>Representatives</u>
Rocky Flats	Jack Bartlett Dick Chanda Jim Frederick Jim Lawless
Oak Ridge Y-12	Deanis Darby Phil Mason Jim Treadwell Bill Wilson
USERDA Oak Ridge	Ray Lux
USERDA Albuquerque	Ray George
USERDA Washington, D.C.	Don Sikes
USNRC Washington, D.C.	Dan Smith
LASL	Tom Atwell, Q-1 Ron Augustson, Q-3 Norton Baron, Q-3 Tom Canada, Q-1 Jane Hagen, Q-3 Dave Hall, QDO Bob Marshall, Q-3 Bill Severe, Q-3 Charlie Slocumb, E-5 Whitney Thorpe, ADASF

The entire conference proceedings were taped and an informal transcript was prepared for the participants.

The attendees concurred that the workshop was productive and agreed to meet again in late summer, 1977.

PART 4

INTEGRATED SAFEGUARDS SYSTEMS AND TECHNOLOGY TRANSFER

GROUP Q-4

R. J. Dietz, Group Leader

A major effort of Group Q-4 is to establish design and performance requirements for cost-effective, integrated safeguards systems for typical nuclear fuel cycle facilities and to assess the impact of these requirements on future plant design criteria. Functions of these facilities include spent fuel reprocessing, plutonium nitrate-to-oxide conversion, plutonium recycle-fuel fabrication, scrap recovery, and waste handling and disposal. In addition to conceptual design and analyses, this effort involves the engineering design and testing of prototypical systems in operational environments to identify actual performance, vulnerability characteristics, and operational compatibility. Results of these studies will provide working guides and performance criteria for material control and accountability systems that can be integrated with physical protection and process control systems in future fuel cycle facilities.

By maintaining a high level of participation in NDA and systems technologies through interactions

with R&D organizations, industry, and government agencies, Q-4 staff members serve as a focal point to coordinate and transfer safeguards technology throughout the nuclear community. By virtue of their overall cognizance of safeguards technology and needs, Q-4 continues to be available to provide technical information and assessments to ERDA on a high-priority, as-needed basis.

The efforts of Q-4 are usually reported in journal papers, consultations, briefings, and topical reports at the conclusion of specific tasks.

During the past four months the Q-4 effort was maintained at an acceptable two-to-one ratio between the two line items: Safeguards Concept Definition for Fuel Cycle Facilities, and Technology Transfer and Support to ERDA/DSS, their subcontractors, and the safeguards industry. Much of this work will be presented at the 18th Annual Meeting of the Institute of Nuclear Materials Management (INMM), Washington, DC, June 29-July 1, 1977.

I. SAFEGUARDS CONCEPTS DEFINITION FOR FUEL-CYCLE FACILITIES

The concept definition for a fuel reprocessing plant is proceeding on schedule. Four major appendices have been drafted; they include critical reviews of NDA Capabilities for Fissile Solutions, Flow Measurement Technology, Minor Isotope Safeguards Techniques, and Conventional Analytical Techniques--all as they pertain to safeguarding a spent-fuel reprocessing facility. As part of the fuel reprocessing study, five pivotal experiments were suggested to ERDA to resolve measurement ambiguities in the model reprocessing facility.

Modeling and simulation efforts for the chemical separations facility advanced during this period. A computer model of the purification process (post-partition through plutonium product storage) was developed along with a model process flow simulation code, FLOSIM. Measurement points and materials accounting strategies are being identified.

The development of a comparable computer model for a plutonium nitrate-to-oxide conversion facility was also initiated. It will simulate the operation of the process by calculating materials flows and in-process inventories. Measurement models,

which simulate the characteristics and uncertainties inherent in selected measurement strategies, are being developed to provide measurement overlays for both the reprocessing and conversion facilities so that the effectiveness of each safeguards strategy can be evaluated.

"Coordinated Safeguards for Materials Management in a Mixed-Oxide Fuel Facility," was published in February.¹⁸ The baseline facility for this study was the projected Westinghouse-Anderson Recycle Fuels Plant with an assumed annual throughput of 8000 kg of plutonium. The study assumed only minor extrapolations of existing fuel fabrication technology and state-of-the-art NDA technology. The safeguards system was designed to be highly automated, reliable, and unobtrusive, interacting passively with process operations. Feasibility and effectiveness of the system were assessed by applying diversion detection algorithms to results of computer-simulated models of process operations and SNM measurements. Typical diversion-detection sensitivity levels in major plant process areas are: 100 to 200 g of plutonium (about 2% of throughput) for a single theft; 200 to 400 g of plutonium (about 0.2% of throughput) for cumulative multiple-small-thefts over any 1-wk period; and 400 to 800 g of plutonium (about 0.1% of throughput) for cumulative multiple-small-thefts over any 4-wk period.

II. INTERNATIONAL SAFEGUARDS

Safeguarding foreign fuel cycle facilities, either under the auspices of the IAEA or through a bilateral agreement with the host country, is an important aspect of the U.S. policy on nonproliferation. In particular, the Japanese reprocessing plant at Tokai is about ready to begin receiving irradiated fuel; the operation of this facility could have far-reaching consequences for future U.S. involvement in the foreign fuel cycle.

Accordingly, on February 28, 1977, representatives of ERDA/DSS and their contractors came to LASL/Q-4 to discuss the problems and possibilities of safeguarding foreign facilities, and to address the question of how the U.S. could, within its agreements with the Japanese, assure itself and the world

The capital cost of the materials control system is expected to be under 5% of total plant cost, and the safeguards staff (excluding guards) is projected at approximately 8% of the total plant operating staff. This study indicates that current measurement technology can provide effective safeguarding of high-throughput fabrication facilities through cost-effective materials management and control systems.

The LASL-Sandia integrated version of this report, "A Preliminary Concept Definition of a Combined Safeguards System for a Mixed-Oxide Fuel Fabrication Facility," has been jointly written and submitted to ERDA/DSS for review. It will be published as a Sandia report.

As part of Q-4's general charter to improve materials accounting for all fuel cycle facilities, work has begun on constructing a logical framework, based on decision theory, for analyzing materials accounting data to detect diversion. The new framework is flexible enough to include all currently used methods of data analysis, including the cusum and V-mask technique as well as Kalman filtering. The work extends and clarifies the results of ORNL on Kalman filtering, allows for the possibility of using nonparametric tests, and discusses some new schemes such as spatial filtering.

that the reprocessing plant at Tokai is adequately safeguarded. On March 4, 1977, a draft report entitled "International Nuclear Safeguards Inspection: Maximizing the Effectiveness of International Safeguards for a Chemical Reprocessing Plant" was sent to USERDA. The report proposed a more comprehensive safeguards system than had heretofore been envisioned, and one that is based on current technology as viewed by LASL, Sandia, and the Idaho Chemical Processing Plant. This report is presently being reviewed within the ERDA complex for its applicability as a technical guideline for impending interactions.

Major Q-4 responsibility has been identified for three tasks in direct support of IAEA activities,

These are (1) the definition of materials accounting data required by the IAEA for safeguarding modern fuel-cycle facilities; (2) a review of dynamic materials control concepts for IAEA personnel, with emphasis on applications to international safeguards; and (3) techniques for materials control in critical assembly facilities. Task 1 will be completed in FY 1978 while Tasks 2 and 3 extend at least through FY 1979. Tasks 1 and 2 are currently being defined in detail. Discussions in all three task

areas are scheduled for August at IAEA headquarters. The ZPPR facility at Idaho Falls was visited in mid-February for discussions of details of operations, materials control, and inventory procedures at a typical critical assembly facility, under Task 3. On the basis of these discussions, preliminary concepts for materials control measurements and statistical data evaluation techniques have been identified.

III. TECHNOLOGY TRANSFER AND DSS SUPPORT

In addition to briefings, visits, and presentations to DSS and its contractors, we performed several identified tasks. Four technical lectures were prepared and delivered for the new ERDA/DSS/LASL training course, "Integrated Safeguards Systems--Concepts and Implementation." They were:

- "Integrated Safeguards Systems--An Overview" --D. D. Cobb,¹⁹

- "Conceptual Design of Integrated Safeguards Systems--An Example" --J. P. Shipley,²⁰

- "Conventional Analytical Chemistry and Nuclear Accountability" --E. A. Hakkila,²¹

- "Materials Accountability--Modeling, Simulation, and Evaluation" --D. B. Smith,²²

IV. MISCELLANEOUS

Concerning the impact of the new national nuclear nonproliferation policy on Safeguards Research and Development, considerably increased responsibilities are anticipated, especially in the area of safeguards evaluation of alternative fuel cycles. This will undoubtedly increase our modeling and simulation efforts, which will, in turn, further

tax the overloaded computational capability available to us. Accordingly, we have initiated a program to enhance this capability with distributed computing via an on-site minicomputer and with increased central computing facilities dedicated to safeguards design and effectiveness modeling.

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GLOSSARY

ARHCO	Atlantic Richfield Hanford Co.
BNL	Brookhaven National Laboratory
BPID	book physical inventory difference
BWR	boiling water reactor
cusum	cumulative summation
DMA	Division of Military Application, ERDA
DRDD	Division of Reactor Development and Demonstration, ERDA
DRM	delta rate monitor
DSS	Division of Safeguards and Security, ERDA
DYMAC	dynamic materials control
EPROM	erasable programmatic read only memory
FCA	Fast Critical Assembly
FFTF	fast flux test facility
HEDL	Hanford Engineering Development Laboratory
HLNCC	high level neutron coincidence counter
HPFL	High Performance Fuel Laboratory
IAEA	International Atomic Energy Agency
INMM	Institute for Nuclear Materials Management
ISPO	International Safeguards Project Office
LED	light-emitting diode
MIP	material in process
NCC	neutron coincidence counter
NDA	nondestructive assay
NEST	Nuclear Emergency Search Team
OSDOS	one-shot, delay, one-shot
PNL	Pacific Northwest Laboratories
PSA	pulse-shape analysis
PVM	personnel and vehicle monitor
PWR	pressurized water reactor
RMD	Radiation Monitoring Devices, Inc.
SAI	solution assay instrument
SGS	segmented gamma scanner
SNM	special nuclear material
SSAS	small sample assay system
TNWCC	thermal neutron well coincidence counter
UART	universal asynchronous receive and transmit
VDC	variable deadtime counter
ZPPR	Zero Power Plutonium Reactor