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Safeguards and Security Progress Report

January–December 1989

Compiled by
Darryl B. Smith
Genara R. Jaramillo

MASTER

Los Alamos
Los Alamos National Laboratory
Los Alamos, New Mexico 87545

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ABSTRACT

From January to December 1989, the Los Alamos Safeguards and Security Research and Development (R&D) program carried out the activities described in the first four parts of this report: Science and Technology Base Development, Basic Systems Design, Onsite Test and Evaluation and Facility Support, and International Safeguards. For the most part, these activities were sponsored by the Department of Energy's Office of Safeguards and Security.

Part 1 covers development of the basic technology essential to continuing improvements in the practice of safeguards and security. It includes our computer security R&D and the activities of the DOE Center for Computer Security, which provides the basis for encouraging and disseminating this important technology. Part 2 treats activities aimed at developing methods for designing and evaluating safeguards systems, with special emphasis on the integration of the several subsystems into a real safeguards system. Part 3 describes efforts of direct assistance to the DOE and its contractors and includes consultation on materials control and accounting problems, development and demonstration of specialized techniques and instruments, and comprehensive participation in the design and demonstration of advanced safeguards systems. Although, in several cases, the implementation of technology is described, the implementation parts of these projects were funded by DOE Program Offices, DOE Field Offices, or by the facilities themselves, rather than the DOE/OSS; they are included in this report as the natural and necessary result of our R&D activities. Part 3 also reports a series of training courses in various aspects of safeguards that makes the technology more accessible to those who must apply it. Finally, Part 4 covers international safeguards activities, including both support to the International Atomic Energy Agency and bilateral exchanges. All of these efforts provide substantial returns on our investment in technology transfer, not only in raising the level of safeguards effectiveness throughout the world, but also in our benefitting from field experiences in operating environments.

Part 5 reports several safeguards-related activities that have sponsors other than the DOE/OSS. The final part of this report lists titles and abstracts of Los Alamos safeguards R&D reports, technical journal articles, and conference papers that were published in 1989.

ACRONYMS

accountability function	ACF
automated data processing	ADP
Audit Log Analysis Package	ALAP
Advanced Nuclear Fuels Corporation	ANF
American Society of Testing and Materials	ASTM
active well coincidence counter	AWCC
Argonne National Laboratories-West	ANL-W
Argonne West/Unified Safeguards project	ARGUS
atomic vapor laser isotope separation	AVLIS
British Nuclear Fuels plc	BNFL
boiling water reactor	BWR
consequence analysis	CA
Central Alarm Monitoring and Assessment System	CAMUS
controlled balance area	CBA
Center for Computer Security	CCS
Commissariat a l'Energie Atomique	CEA
Commission of European Communities	CEC
continuous flow analysis	CFA
Comissao Nacional de Energia Nuclear	CNEN
computer security enhancement review	CSER
central storage facility	CSF
Computer Security System Managers	CSSMs
Computer System Security Officer	CSSO
containment and surveillance	C/S
Central Training Academy	CTA
chemical weapons	N
database management system	DBMS
Department of Energy	DOE
DOE's Albuquerque Operations Office	DOE/AL
DOE's Central Training Academy	DOE/CTA
DOE/Office of Safeguards and Security	DOE/OSS
DOE Savannah River Operations Office	DOE/SR
Deutsche Gesellschaft fuer Wiederaufarbeitung von Kernbrennstoffen mbH	DWK
Comitato Nazionale per la Ricerca e per lo Sviluppo dell'Energia Nucleare e delle Energie Alternative - Italian National Commission for Research and Development of Nuclear and Alternative Energy	ENEA
Enrichment Plant Safeguards Review Group	EPSRG
European Atomic Energy Community	EURATOM
fast breeder reactor	FBR
Franco-Belge Fabrication de Combustible	FBFC
figure of merit	FOM
Fixed Energy, Response Function Analysis, with Multiple Efficiency	FRAM
Federal Republic of Germany	FRG
Gaseous Diffusion Plant	GDP
graduate research assistant	GRA
highly enriched uranium	HEU
high level neutron coincidence counter	HLNC-II
high performance capillary electrophoresis	HPCE
high-purity germanium	HPGe
hexapartite safeguards project	HSP
International Atomic Energy Agency	IAEA
instrument control function	ICF
isotope dilution gamma-ray spectroscopy	IDGS
induced fission	IF
isotope dilution mass spectroscopy	IDMS
inspection and evaluation	I&E
Idaho National Engineering Laboratory	INEL
inventory sample	INVS
isotopic analysis by high-resolution gamma-ray spectroscopy	ISO
individual unit	IUN
local area network	LAN
LArge SCAle Reprocessing plants	LASCAR
Los Alamos Vulnerability/Risk Assessment	LAVA
lump-corrected segmented gamma-ray scanning	LCSGS
low enriched uranium	LEU
Lawrence Livermore National Laboratory	LLNL
large size dried	LSD
limited-frequency unannounced-access	LFUA
light water reactor	LWR
materials accounting with sequential testing	MAWST
Materials Accounting Safeguards System	MASS
material balance area	MBA
materials control and accounting	MC&A
Monte Carlo code for neutron and photon transport	MCNP
multi-channel scaler	MCS
multi-element bottle	MEB
modular integrated video system	MIVS
molecular laser isotope separation	MLIS
mixed uranium-plutonium oxide	MOX
molten salt extraction	MSE
Master Safeguards and Security Agreements	MSSAs
material unaccounted for	MUF
New Brunswick Laboratory	NBL
neutron coincidence counter	NCC
National Computer Security Center	NCSC
nondestructive assay	NDA
Network Design System	NDS
neutron multiplicity counting	NMC
Nuclear Material Storage Facility	NMSF
near-real-time accounting	NRTA
New Special Recovery	NSR
Nevada Test Site	NTS
Nuclear Materials Accounting System	NUCMAS
protect as restricted data	PARD
passive/active neutron coincidence counter	P/A NCC
process flow diagram	PFD
plutonium fuel production facility	PFPF
portable multichannel analyzer	PMCA
photomultiplier tube	PMT
Power Reactor and Nuclear Fuel	PNC

Development Corporation	
PNC/Tokai Reprocessing Plant	PNC/TRP
Program of Technical Assistance for the IAEA	POTAS
plutonium processing facility	PPF
Power Reactor and Nuclear Fuel	PRNF
prototype graphical representation model	PROGREP
Processing and Fuel Facilities	PROFF
pressurized water reactor	PWR
random access memory	RAM
research and development	R&D
relational database management system	RDBMS
Rocky Flats Plant	RFP
remote mechanical c-line	RMC
relative standard deviation	RSD
solution assay instrument	SAI
software change request	SCR
solution enrichment system	SES
spontaneous fission	SF
segmented gamma scanner	SGS
special isotope separation	SIS
Sandia National Laboratories, Albuquerque	SNLA
special nuclear materials	SNM
structured query language	SQL
Savannah River	SR
Shipper/Receiver Confirmation System	SRCS
Savannah River Site	SRS
State Systems of Accounting and Control	SSAC
scrap and waste	S&W
test and evaluation	T&E
Technikum für Grosskomponenten	TEKO
Thermal Oxide Reprocessing Plant	THORP
thermal neutron multiplicity counters	TNMCs
Tokai Reprocessing Plant	TRP
transuranic	TRU
University of New Mexico	UNM
University of California-Davis	UCD
vulnerability assessment	VA
Wireless Alarm Transmission of Container Handling	WATCH
Westinghouse Hanford Co.	WHC
Westinghouse Idaho Nuclear Co.	WINCO
Waste Isolation Pilot Project	WIPP
Wisdom and Sense	W&S
Westinghouse Savannah River Site	WSRS
x-ray fluorescence	XRF

PART 1. SCIENCE AND TECHNOLOGY BASE DEVELOPMENT

I. MATERIALS CONTROL AND ACCOUNTING (MC&A)

A. Nuclear Materials Detection and Surveillance.

The objectives of this project are to develop nuclear materials control technology by improving radiation-monitoring techniques for the detection and surveillance of nuclear materials (for example, portal monitors, verification stations, digital image analysis, and remote surveillance instruments), and to continue to help Department of Energy (DOE) facilities specify, evaluate, calibrate, and properly use portal monitors.

1. Portal Monitor Technology Development (P. E. Fehlau, K. L. Coop, H. F. Atwater, and K. S. Allander, N-2). We continue to pursue methods to detect and identify special nuclear materials (SNM) by sensing its emitted radiation and have transferred this technology through our applications guides[1,2] and other reports. We also continue to transfer technology for improving the availability of effective commercial SNM monitors as we develop and evaluate new equipment. This year our neutron-detection-based monitors[3] approached commercial availability after successful in-plant tests. Also during this year, our hand-held neutron verification instruments[4] became commercially available. These topics and some new initiatives that are underway are discussed below.

a. Neutron-Detection-Based SNM Portal Monitors. The first in-plant evaluation of the pedestrian version of our neutron-detection-based portal monitor was completed at the Savannah River Site (SRS). After the evaluation results reported by SRS confirmed the performance predicted from laboratory tests at Los Alamos, SRS asked for assistance with preparing purchase specifications for the monitor. SRS plans to use commercially produced neutron-detection-based monitors for detecting gamma-ray-shielded plutonium, a much more effective and convenient detection means than the alternative of attempting to detect gamma-ray shields with metal detectors. Westinghouse Hanford became interested in our neutron portal for the same purpose during the year. We demonstrated the portal for them at Los Alamos, and then loaned them our second portal for an evaluation at Hanford. After the portal was put into operation at Hanford, we assisted them in developing a test plan for their evaluation, which is scheduled to begin early in 1990.

Familiarity with our neutron portal prompted SRS to suggest another use for our detector design in a low-level waste monitor. During the year, we assisted SRS in developing a design for the monitor and carried out Monte Carlo calculations on the expected sensitivity for detecting bare and neutron-shielded plutonium. We will continue to assist as needed with developing the monitor.

b. Hand-Held SNM Verification Instruments. Hand-held SNM verification instruments are an outgrowth of our work with hand-held SNM search instruments. We modified the search instrument design to produce two types of prototype verification instruments;[4] a stabilized gamma-ray spectrometer for verifying the presence of penetrating plutonium gamma rays and a neutron-detection-based instrument to verify the presence of plutonium by its neutron emission. These instruments duplicate some of the types of confirmation measurements used at Pantex for both war reserve weapons and nonnuclear test assemblies. During the year, we participated in a program with Sandia National Laboratories, Albuquerque (SNLA) and the DOE's Albuquerque Operations Office (DOE/AL) to evaluate the instruments in pre-flight verification that Air Force test warheads are nonnuclear test assemblies (Fig. 1). The neutron instrument's reliability and ease of use resulted in SNLA purchasing 30 commercially produced, upgraded neutron instruments (TSA Systems* NNV470) for use in further studies in the Air Force and the other military service test programs. Near the end of the year, we proposed that the instruments also be applied to SNM confirmation measurements at the Nevada Test Site (NTS) on weapon components shipped there for assembly. (Portions of these activities were supported by DOE/AL and SNLA funding.)

2. SNM Monitors at DOE Facilities (P. E. Fehlau and M. C. Lucas, N-2). We continue to provide information on SNM monitoring and SNM monitor applications to DOE, its contractors, and others. Those who made use of this service during the year included Westinghouse Hanford Co. (WHC), Westinghouse Idaho Nuclear Co. (WINCO), the Portsmouth Gaseous Diffusion Plant (GDP), the Oak Ridge Y12 Plant, the Rocky Flats Plant (RFP), Mound Laboratory, Lawrence Livermore National Laboratory (LLNL), Los Alamos National Laboratory, SNLA, Brookhaven National Laboratory, Pacific Northwest Laboratory, Naval Seas Systems Command, and the DOE's Central Training Academy (CTA). We also provided more substantial assistance as follows.

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*TSA Systems, Ltd., 1820 Delaware Place, Longmont, CO 80501.

Fig. 1. The lightweight, hand-held verification instruments can provide a last minute, pre-flight verification that Air Force test warheads are nonnuclear. This step, together with other measures, guarantees a very high level of confidence that no mistakes will be made.

a. Entry-Control Workshops. We gave an invited lecture on SNM monitoring at an SNLA Entry Control Workshop[5] and have accepted an invitation to give a presentation at an American Society for Testing and Materials (ASTM) Symposium on Access Security Screening[6] to be held in 1990. The SNLA workshop offered an excellent opportunity to discuss modern monitoring capabilities with an audience from across the DOE complex.

b. SNLA Portal Monitor. We have been helping SNLA remedy problems with the SNM portal monitors that they had been using and to instrument SNM monitoring booths in a new entry-control facility. At the beginning of this year, we measured detector response to simulate a monitoring booth and predicted Category II performance (detects 10 g of highly enriched uranium (HEU) or 0.29 g of low-burnup plutonium) for the new six-detector entry-control monitoring booths (Fig. 2). At the same time, we helped to put a renovated walk-through portal into operation at the existing entry-control station and then wrote and published a calibration manual[7] covering both the renovated and the new booth monitors. Later, when the new monitoring booths were completed, we helped Sandia put the SNM monitoring booths into operation and calibrate them.

c. Pantex SNM Monitor Calibration Manual. In response to a request from the plant, we wrote a draft manual for calibrating and testing the TSA Systems, Ltd., SNM portal monitors used at the plant for monitoring pedestrians and vehicles. This type of monitor is widely used, but it is an inexpensive monitor that is relatively difficult to calibrate using the manufacturer's maintenance manual. We visited the plant to review calibration and testing with the personnel from the electronics repair, metrology, and security engineering departments and to participate in a trial calibration. Following that, we revised the manual and provided the plant with final draft copies. We expect the manual to be published as a report[8] early in 1990.

d. Pantex Material Tracking System. We assisted SNLA with a technical review of a subcontractor's proposal for a material tracking system at the Pantex plant. Our conclusion that the proposed system had serious shortcomings supported an independent review by plant

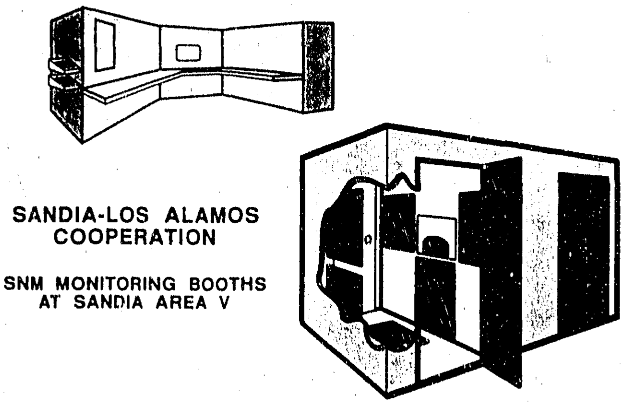


Fig. 2. The Sandia monitoring booths use six plastic scintillation detectors that surround a pedestrian being identified and cleared for passage. A security inspector seated at a console is inform d of the outcome of SNM monitoring.

personnel. A second proposal by the subcontractor was not much better and the project is now on hold.

e. Shielding Test Objects for Metal Detectors. We helped SNLA create suitable objects for testing the capabilities of metal detectors to detect metallic SNM gamma-ray shielding materials. We based the size of the test objects on the amount of SNM that would have to be contained under the assumption of protracted diversion over a period of 40 days to remove two formula quantities of either metallic HEU or metallic low-burnup plutonium (DOE order 5633.3). The smaller HEU shield weighing 144 g was used as Sandia's shielding test object. The low-burnup plutonium shield weighed 2.9 kg and might be impractical to carry.

3. Technical Evaluation of Commercial SNM Monitors (P. E. Fehlau, N-2). As part of an effort to track the performance of commercially available SNM monitors and to transfer state of the art technology to the industry, we evaluate promising new equipment in a laboratory environment. We evaluated three monitors this year: a Los Alamos-designed monitor produced by a new manufacturer, and two others that are high-sensitivity contamination monitors now being offered as SNM monitors.

a. Jomar Systems, Inc.,* Model JPM-22 SNM Portal Monitor. This monitor is manufactured to a Los Alamos specification and will replace an earlier, Gull Engineering (now out of business) version, which approximated our specification. The evaluated monitor, which had a 29-inch portal width and plastic scintillators in side cabinets only, uses a sequential probability ratio monitoring method that allows it to serve as either a walk-through or wait-in monitor. It achieved Category II performance (detecting 10 g of HEU or 0.29 g of low-burnup plutonium) as a walk-through monitor and Category III performance (detecting 3 g of HEU or 0.08 g of low-burnup plutonium) as a nominal 2-s wait-in monitor as long as the pedestrian faced the detectors.[9] Otherwise, the wait-in performance was Category II. In both cases, the nuisance alarm rate was 1 per 2500 passages. If necessary, the monitor's sensitivity could be increased by using a lower alarm threshold that would also increase the nuisance alarm rate, using an HEU region of interest for uranium monitoring, using a narrower portal width, or increasing the passage time.

We completed this evaluation in two weeks with no delays and needed less than 30 minutes to initially operate the monitor and check its calibration. Los Alamos has purchased this type of monitor to replace its existing portals. SNLA uses the basic design in its monitoring booths. The Portsmouth GDP also purchased it, in a form that integrates its remote control unit with a metal detector, to replace its existing portal monitors.

b. National Nuclear Corporation* Model DM-60. This monitor, which has a 24-inch portal width, is designed to detect contamination on body surfaces with uniform sensitivity. It has more than twice the detector area of the Jomar monitor, including very large side detectors and additional head and foot detectors. The monitor achieved Category II operation as a walkthrough monitor and Category III performance as a 3-s wait-in monitor.[10] The nuisance alarm rate was 1 per 2500 passages in walk-through operation and much less in wait-in operation.

This monitor is not designed for SNM monitoring and, as a result, its detector design and signal conditioning are inefficient for the purpose. Calibrating and maintaining the monitor is difficult because there is no way to observe detector signals. A calibration scheme using the monitor's internal computer is time consuming. The

*Jomar Systems, Inc., 110 Eastgate Drive, Los Alamos, NM 87544 (505) 662-9811.

*National Nuclear Corp., 1904 Colony St., Mt. View, CA 94043 (415) 962-9220.

evaluation period was extended by needed repairs, and an electrical safety problem was encountered while making the repairs.

c. TSA Systems, Ltd., Model SPM-904. This monitor, which is designed for use as a wait-in contamination monitor, also has a very large detector area with large side detectors on both sides of the portal and smaller overhead and underfoot detectors. The evaluated monitor had a 24-inch portal width and was delivered assembled and in operating condition, needing only minor calibration adjustments before beginning routine use. The monitor achieved Category II operation as a 1-s wait-in monitor at a nuisance alarm rate of 1 per 1100 passages.[11] The monitor's performance could be improved by repositioning a poorly placed occupancy sensor that forces a pedestrian to stand with part of his body in a relatively insensitive area. Another worthwhile improvement would be to adopt a form of detection logic that would increase the monitor's sensitivity when using longer wait-in times.

We completed the evaluation without incident. Our only problems were inaccuracies in the manual, the need to dismount the electronics enclosure so that the cover could be opened for calibration, and investigating reputed errors in software routines that the monitor uses during calibration.

4. Technical Collaboration with France (P. E. Fehlau, N-2). For several years, we have been cooperating on matters pertaining to research, development, and testing to improve the physical protection of nuclear material with the French Commissariat a l'Energie Atomique (CEA) center at Fontenay-aux-Roses. The basis for the cooperation is a memorandum of understanding between the CEA and the US DOE. This year we studied a proposed French monitoring method based on enhancing the performance of their walk-through portal monitors with a recursive digital filter.[12] Our study compared the proposed French method with moving-average scalers and sequential probability ratio testing methods used in this country to enhance performance.

We used a computer-operated counter and software that we wrote to simultaneously analyze real-time data from a portal monitor with each of the three detection methods. First, we used software with a range of operating parameters for each method to observe nuisance alarms over a long time. That allowed us to select operating parameters for each method to give the same statistical alarm probability with 95% confidence. We then used those parameters in detection sensitivity tests. We conducted walkthrough tests using nine individuals to carry a test source through the portal in 450 total passages. The resulting detection probabilities for each of the three methods were also identical with 95% confidence. Hence, we concluded that the three detection methods are equivalent and only secondary considerations would govern the choice of which one to use. There may be a significant disadvantage to the digital filter because the influence of past high- or low-radiation intensity in the portal may persist longer than with the other methods. The recursive nature of the filter leads to an exponentially decreasing influence from past intensity measurements, whereas the other methods use only a few intensity measurements from the past. We will prepare a report on our study.

5. Device Verification (K. L. Coop, G. S. Brunson, and G. Arnone, N-2). We successfully tested the large modular multiplicity counter for device verification using large quantities of D38 and ^{6}LiD surrounding target materials of either D38 or ^{235}U. We placed a small 14-MeV neutron generator inside the counter with the materials to be irradiated. We measured the multiplicities of emitted neutrons between interrogating pulses from the generator and related them to the type and amount of target material. The major problem with this technique is that the neutron generator irradiates the entire assembly, and the 14-MeV neutrons cause fission in the large quantities of D38 surrounding the target material, which interferes with the measurement.

A better interrogation method, such as using a bremsstrahlung beam from a linear accelerator, would ameliorate this problem, because such a beam can be collimated to exclude much of the D38 from the field of view. We have not done these experiments however, because such measurements are no longer needed to solve verification problems at either Pantex or NTS. For this reason, we will terminate this project at its present stage, and redirect our efforts to a problem of more immediate concern: the detection of concealed SNM in packages and waste containers. Solutions to this type of problem may involve the modular multiplicity counter and similar interrogation techniques.

References 13 and 14 contain recent reports on the module that we use to collect and sort the data from the modular multiplicity counter.

6. Monitoring Packages and Containers – Passive SNM Package Monitor for Uranium, (P. E. Fehlau, J. M. Ortiz, and K. S. Allander, N-2). The question occasionally arises of whether something more effective than hand-held monitoring, but still simple, can be done at entry-control stations for routinely monitoring hand-carried items for HEU. Last year we described a laboratory prototype uranium package monitor, which is IBM-PC-controlled and based on incorporating radiation detectors in and beside a load-cell weighing device.[15] This year we used the prototype to compare the detection sensitivity of the monitor using two different gamma-ray energy regions for detection: one a narrow region optimized for bare HEU, and the other a very broad region (with lower HEU sensitivity) that would also include energetic, penetrating radiation from ^{232}U and its daughters. We compared detecting shielded weapons-grade uranium, which contains some ^{232}U, contained in hand-carried packages weighing up to 15 kg, using the two regions. The broad region, in which the more penetrating radiation is detected, approximately tripled the thickness of lead needed to shield the HEU adequately. We also improved the mechanical design of the prototype so that it can be moved to the Portsmouth GDP for additional measurements of uranium that contains less of the ^{232}U isotope daughters.

B. Materials Control Subsystem Definition and Development.

The objective of this project is to provide the DOE complex with improved techniques for controlling and tracking material that can be interfaced with materials accounting and physical protection subsystems. It emphasizes defining and developing materials control components and subsystems for integration with other safeguards subsystems to provide defense-in-depth and to maximize total safeguards system effectiveness.

Digital Image Analysis for Materials Control (C. A. Steverson, N-4). We are focussing our attention on improvements in imaging hardware, software development systems, and imaging software. We have been investigating infrared technology, optical frequency charge coupled device camera technology, and other imaging technology that will enhance our capability to apply digital image processing to safeguarding nuclear materials at DOE facilities.

We purchased a SUN 3/260 computer system and configured it to provide a software development environment for imaging projects; this environment supports the requirements of multi-person programming projects and provides the tools necessary for properly engineered software. Imaging software, previously written for IBM 286/386-based computers, has been ported to the new platform, and additional software has been added to our imaging library. New software that we developed during this fiscal year includes additional change-detection software, image registration capabilities, and histogram analysis capabilities. Detailed information on software development may be found in Reference 16.

C. Nondestructive Assay (NDA) Measurement Technology.

The objectives of this project are to develop and adapt state-of-the-art NDA techniques and instruments to meet the needs of DOE and commercial nuclear facilities, as well as those of safeguards inspection authorities for inventory verification.

1. Design of a New High-Efficiency Small-Sample Neutron Coincidence Counter (NCC) (M. C. Miller, H. O. Menlove, and P. A. Russo, N-1). The coincidence count rate of sufficiently small (that is, nonmultiplying) samples of plutonium is a direct measure of the sample's effective ^{240}Pu mass. For representative samples of known mass and plutonium isotopic composition, this measured ^{240}Pu effective mass can be used to determine the total plutonium content of the bulk item from which the sample was taken. The precision of this method improves with increased counter efficiency. The accuracy relies on the reproducibility of the counter efficiency with the position of the small sample in the counter well. Assay uncertainties of 1% or better can be achieved.

Experience with our original inventory sample (INVS) coincidence counter[17] has shown that under some field conditions, the sample cannot be accurately positioned. Under these circumstances, variations in counter response caused by axial position can cause a measurement bias. To eliminate this potential source of measurement error, we are upgrading the original INVS counter. The focus of the new design is a flattened axial response to minimize the effects of positioning, and an increased efficiency for improved precision. We used the Los Alamos Monte Carlo code for neutron and photon transport (MCNP),[18] which provides great flexibility for such parametric studies to help optimize the design.

The MCNP geometry (Fig. 3) included the ^3He detector tubes (including dead space), the high-density polyethylene moderator, and the graphite/polyethylene endplugs. We made tallies to estimate both totals and coincidence count rates (Fig. 4). A ^{252}Cf point source, in various positions within the sample cavity, estimated the response function of the counter. The sample cavity is 5 cm in diameter by 15 cm high. Efficiency is increased by incorporating two additional ^3He tubes and by optimally placing the tubes within the high-density polyethylene moderator, as well as by removing cadmium from the original design. We extended the axial flat zone—compared to the original INVS counter—by using a combination of graphite and high density polyethylene and increasing the active length of the tubes from 30 to 40 cm. Table I compares the original and the new INVS counter designs.

We plan to investigate the use of the ratio of totals in concentric rings of ^3He tubes to determine the neutron

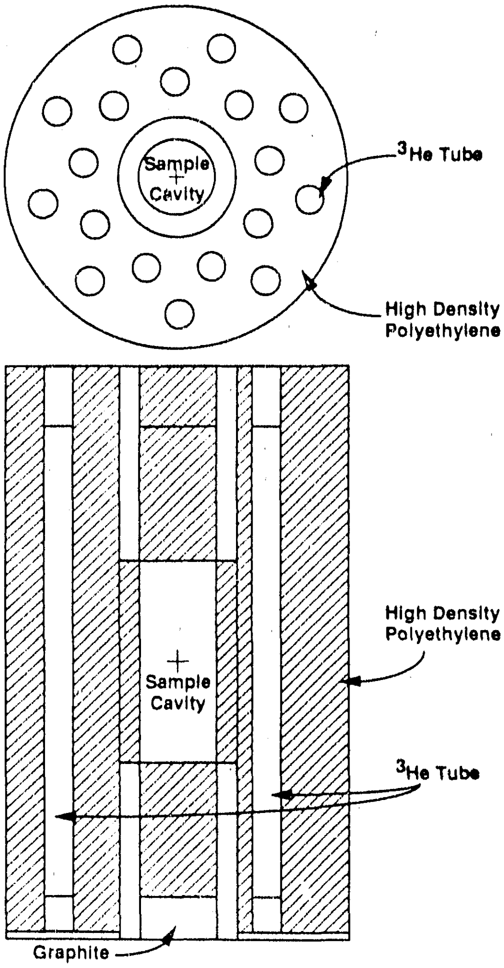


Fig. 3. Geometry plot of the MCNP model of the new INVS counter showing ^{3}He tube layout and material composition of the endplugs.

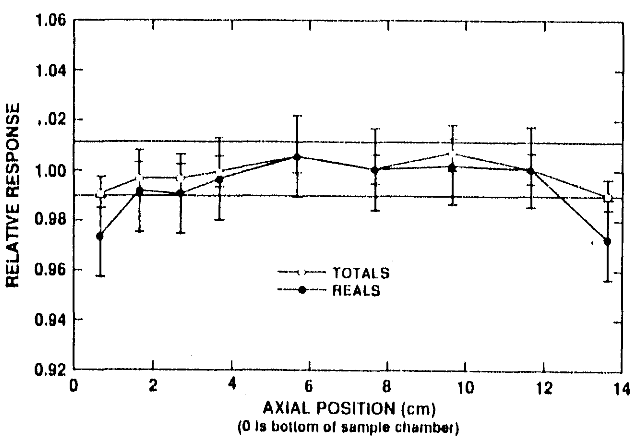


Fig. 4. Calculated totals and reals response to a ^{252}Cf point source as a function of axial position within the sample cavity of the new INVS counter.

energy and to estimate the (α,n) neutron production rate. We also will be comparing the benefits (higher efficiency) vs drawbacks (higher background) of a third ring of ^{3}He tubes.

2. Neutron Coincidence Counting Software (M. S. Krick and E. A. Kern, N-1). As part of our efforts to standardize software, we have developed two neutron coincidence counting programs and released them for use by NMT and OS Divisions at Los Alamos. One is designated NCC and is designed for passive neutron coincidence counting of plutonium samples; the other is designated active well coincidence counter (AWCC) and is designed for active neutron counting of uranium samples. They are general-purpose codes and are not intended for a specific facility, although some custom modifications will be needed for specific applications. The NCC code also is used at the Savannah River Plant. Presently, the NCC and AWCC programs run on IBM PC/AT or compatible computers.

The two programs are very similar and use a windowed environment for user interactions. User menus can be selected as short form or long form, although a password is required to access the long-form menus, which allow changes to be made in the calibration parameters, reference data, etc.

Table I. Comparison of Original and Upgraded INVS Counter

Item	Original INVS	New INVS[a]
^{3}He Tubes		
Number	16	18
Active length	30 cm	40 cm
Counter height	48 cm	58 cm
Counter diameter	27 cm	30 cm
Efficiency	30%	40%
Cadmium	Yes	No
Graphite	No	Yes
Flat Zone[b]	5.7 cm	12 cm

[a]Based on MCNP calculations.
[b]Defined as reals response variation of < ± 2% relative to center of cavity.

The NCC and AWCC programs are used for background, measurement-control, calibration, assay, verification, and general-purpose measurements. Data can be entered from coincidence electronics, from the keyboard, or from disk files. All measurement data and results can be printed as they are produced.

The calibration measurements produce total and coincidence count rates corrected for deadtime, background, and normalization. For plutonium measurements, these can be with or without multiplication correction. A least-squares fitting code (not part of the NCC and AWCC codes) must be used to obtain the calibration coefficients and associated uncertainties; these values are entered from the keyboard into the database using the NCC and AWCC codes.

The database contains all of the parameters required for the measurements, including detector, source, calibration, isotopic, and test parameters. These parameters are easily entered or edited through the data-entry windows available in the long-form menus. Raw data and results also are archived in the database and can be recalled and displayed as desired.

Any practical number of detectors, sources, calibration curves, etc. can be stored simultaneously in the database. A calibration curve can be created using one detector and then used to perform assays with a different detector of the same type; the necessary cross-referencing is done automatically by the program.

The main identifier for archiving measurement data from the NCC and AWCC codes is the Measurement Series identification number (ID), which is a name assigned by the user that tags all measurement data. An example might be "HEU oxide verif. meas.: Oct. 89." A Measurement Series ID remains in effect until changed by the user. Measurement series data are automatically saved on the hard disk. Any practical number of measurement series can be stored simultaneously on the disk and any series can be used at any time.

A measurement series can be copied to a floppy disk for storage or for transfer to another computer. Conversely, a measurement series can be recalled from a floppy disk for additional measurements, reanalysis, or review. When a measurement series is copied to a floppy disk, the parameter database also is copied, so the floppy disk contains a complete record of the measurements and analysis. The results also are transferred to the floppy disk in text format, so the results can be accessed easily by popular spreadsheet and database software.

3. Californium Ion Chamber as a Neutron Source for Active Interrogation (C. R. Hatcher, N-1). We are using a ^{252}Cf ion chamber that emits 3000 n/s to measure ^{235}U samples by the differential die-away technique.[19] Such measurements require a neutron detector with high counting efficiency and short die-away time. The neutron well counter originally developed for multiplicity measurements[20] is well suited for this application because it has an efficiency of 16.5% and a die-away time of 11.5 µs. Figure 5 shows the geometry of a polyethylene insert designed to fit into the well of the

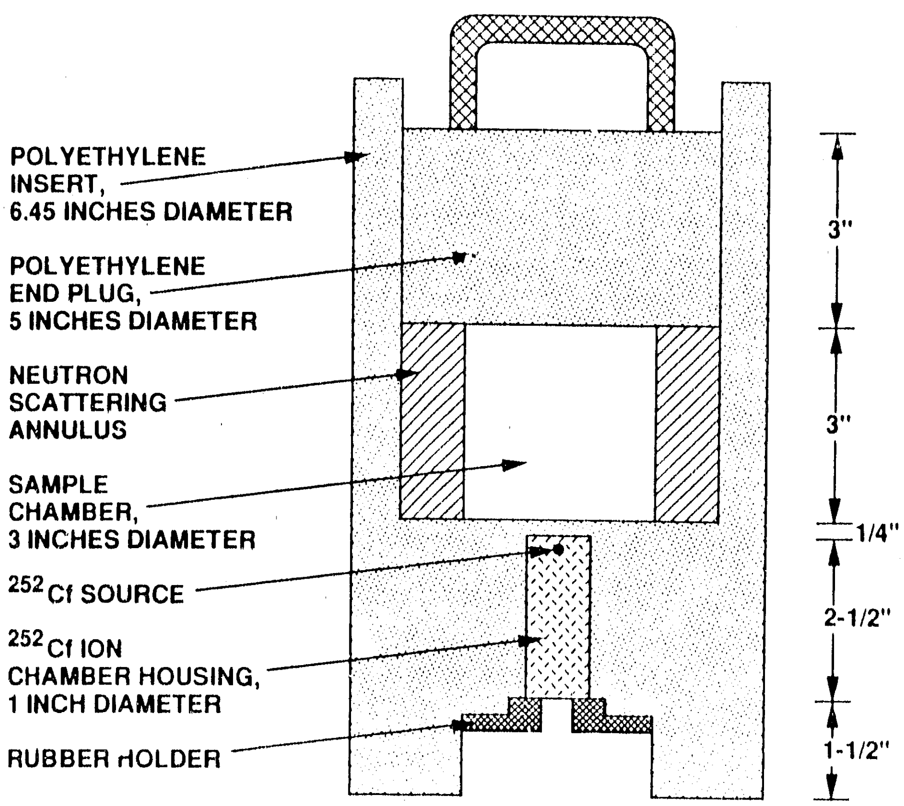


Fig. 5. Cross-section of polyethylene insert, neutron scattering annulus, and californium ion chamber designed to fit inside a neutron well counter.

multiplicity counter for performing differential die-away measurements. Near the center of the insert is a neutron scattering annulus, and inside the annulus is a 3-in.-diam by 3-in.-high sample chamber. The annulus can be changed easily to vary the neutron scattering material. The californium ion chamber is mounted below the sample chamber, with the californium source near the upper end of the ion chamber housing.

Neutrons detected by ^3He detectors in the well counter are recorded with a multichannel scaler (MCS). The sweep of the MCS is started when a fission fragment is detected in the ion chamber, which indicates that a spontaneous fission has occurred in the ^{252}Cf source. Some of the neutrons emitted by the ^{252}Cf source enter the detector, where they die away rapidly. Other neutrons enter the polyethylene insert and neutron scattering annulus; these neutrons die away more slowly, and are used to interrogate uranium samples placed in the sample chamber. Figure 6 shows a differential die-away curve taken by the MCS with a 103-g HEU standard in the measurement chamber. In the time interval from zero to 50 μs, we observe the expected 11.5-μs die-away time. In the interval from 75 to 250 μs, the die-away time is much longer (136 μs), which is characteristic of the polyethylene insert and neutron scattering annulus. To obtain a measure of ^{235}U mass, the area under the curve in Fig. 6 is integrated over the time window from 70 to 250 μs. With no ^{235}U in the sample chamber, there is a significant count rate in the selected time window because of accidental counts, that is, counts caused by neutrons that are not correlated with the spontaneous fission event that started the MCS sweep.

We calibrated the instrument using HEU standards in the shape of 2-in.-diam disks, which were held on edge in the center of the sample chamber by a thin aluminum holder. Calibration curves of count rate (reals plus accidentals) vs ^{235}U mass have a steep slope for the mass region from zero to 10 g, and then begin to flatten out. In the data analysis, we used a linear approximation for the calibration curves in the region from zero to 10 g and in the region from 40 to 240 g. Table II shows the relative probable error in the measurement of ^{235}U mass caused by counting statistics for several neutron scattering annuli and

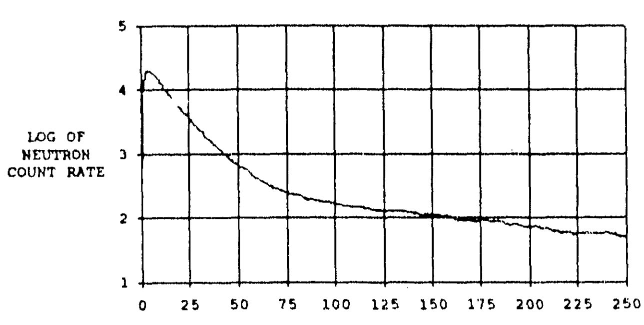


Fig. 6. Differential die-away curve obtained with a CH_2 annulus and a 103-g HEU sample.

for two ^{235}U masses, one in each linear region of the calibration curves. The polyethylene annulus gives the best precision for a 5-g sample and one of the best for a 100-g sample. Other neutron scattering materials appear to harden the neutron spectrum, but also reduce the count rate.

An MCS is not the ideal recording device for this application because of its long dead time (10^{-2} s) between sweeps. If we could replace the MCS with a cross-correlation coincidence counter based on shift-register technology, the dead time would be largely eliminated. This change would increase the count rate by a factor of ten, and would reduce the statistical uncertainties quoted in Table II by a factor of three. Even with this improvement; however, applications for such an instrument would be limited because of its high cost and low precision compared with other active neutron instruments.

4. Measurement of Uranium Isotopic Ratios with the FRAM Plutonium Isotopic Code (T. E. Sampson, N-1). The FRAM (Fixed Energy, Response Function Analysis, with Multiple Efficiency) plutonium isotopic code[21] has been used routinely in the Los Alamos Plutonium Facility (TA-55) for the past year to measure a wide variety of materials. We are continuing to develop its capability and have demonstrated that the same code that is in routine use for plutonium analysis also can measure the ratio of ^{235}U to ^{238}U in uranium samples of arbitrary geometry and composition *without any code modifications.* To our knowledge this capability has never been demonstrated in any other plutonium isotopic code.

We are investigating this capability with a variety of samples having ^{235}U fractions ranging from 0.3% to 91%.

TABLE II. Relative Probable Error in ^{235}U Mass for a 1000-s Measurement Using a Cf Ion Chamber

Neutron Scattering Annulus	M = 5 g %	M = 100 g %
CH_2	12	21
C	18	22
Pb	21	21
Ni	54	51
CH_2 + Cd*	78	95
W	102	57
Fe	103	45

*Cd cylinder inside CH_2 annulus.

5. Intrinsic Densitometry of High-Burnup Plutonium Solutions (S.-T. Hsue, N-1). Although we developed the intrinsic densitometry technique[22] for nondestructively assaying low-burnup solutions at Los Alamos, we are extending it to high-burnup solutions. Seven solutions with concentrations from 50 g/L to 300 g/L were prepared from high-burnup plutonium samples. Six of the solutions were from one mix of plutonium isotopes; the seventh had a different isotopic distribution. The plutonium concentrations were characterized by titration and by K-edge densitometry, and the isotopic distribution was determined by mass spectrometry. The intrinsic densitometry technique required 6 mL of solution from each sample. Each of the seven samples was measured in 7 to 16 runs (1 h per run) to determine repeatability. Four of the solutions were removed from the sample holder and measured later to determine reproducibility. We found the assay precision for plutonium concentration using this technique to be about 2% when the samples are counted for 1 h. Further data analysis is underway.

6. Nondestructive Assay of Pyrochemical Process Residues. Plutonium in the bulk forms generated by scrap recovery operations is often chemically impure and physically and chemically heterogeneous. Pyrochemical residues, in particular, are lean and highly impure chloride-salt-based materials in which the plutonium can coexist in both metallic and salt forms, the americium content is typically high (from a few to tens of weight percent relative to plutonium), and the typical residue consists of heterogeneous and nonrepresentative chunks of various sizes. Minimizing the handling of such highly radioactive materials requires assigning accountability values to these residues without removing them from the process line so that they can be immediately routed either to the next stage of processing or to waste disposal.

a. Assay of Pyrochemical Residues with the FRAM Plutonium Isotopic Code (T. E. Sampson, N-1). One of our main reasons for developing the FRAM[21] code was to give the Los Alamos Plutonium Facility an additional method to measure samples with heterogeneous Am/Pu ratios. Conventional plutonium isotopic techniques do not give the correct Am/Pu ratio if it is not the same in all of the plutonium in the sample. This is the case with many types of pyrochemical residues, which often contain americium in a salt phase (as a chloride, for example) while the plutonium is in a finely divided metal phase.

A method developed by Fleissner,[23,24] which greatly reduces the errors involved in measuring these materials, has been incorporated into the FRAM code. We have tested the FRAM code, as part of a larger effort, on the salt residues from the molten salt extraction (MSE) process. Several of the same samples that were measured by FRAM also were shipped to the RFP where they were measured by Fleissner. Subsequently, the samples were crushed, blended, sampled, and analyzed with traditional analytical chemical and mass spectrometric methods by group CLS-1 (Analytical Chemistry) at Los Alamos. The results are compared in Table III for the effective specific power in watts/gram of plutonium as determined from the the nondestructive FRAM and Fleissner measurements on the original heterogeneous samples and by destructive chemical analysis on the crushed and blended samples (denoted CLS-1).

The second and third columns in Table III show that the analyses of FRAM and Fleissner give similar results with the FRAM bias being, perhaps, slightly smaller. The average bias for all five FRAM results, a little over 2%, is consistent with that found in a similar study by Fleissner and Hume.[24] Of greatest importance, however, is the last column. This shows what the bias would have been if conventional nondestructive isotopic analysis, which doesn't account for the heterogeneous Am/Pu, had been performed. The heterogeneous analysis in the FRAM code reduced biases as large as 60% to about 2% — a significant improvement in materials accounting.

b. Geometry-Based Multiplication Correction for NCC (D. G. Langner and P. A. Russo, N-1). Multiplication corrections for neutron coincidence based assays depend on knowledge of the (α,n) production rate of the sample.[25] There are many categories of materials, however, for which neutron coincidence counting would be a desirable method of assay, but the (α,n) rate is unknown. Pyrochemical residues are such a class of materials.

We have developed a new method of analysis that corrects for multiplication effects using knowledge of the sample geometry. This technique assumes that the samples being measured are of approximately constant chemical composition but vary in plutonium fraction, sample density, and container loading. For cylindrical cans, the technique assumes that the sample multiplication, M, is given by

$$M = k \frac{{}^{239}Pu\text{-}effective}{r\,(r+h)} + 1$$

where r and h are the sample radius and fill height, and k is a constant that depends primarily on the sample chemistry, but also is a function of the neutron energy spectrum of the sample.

When this expression is substituted into the equations for neutron count rates (reals and totals) from the point model for neutron coincidence counting,[26] and the assumption is made that the constant k is small enough that terms involving powers of k greater than one can be ignored, an expression results that can be fitted to reals and totals data obtained by measuring samples of known plutonium loading and isotopic composition. This procedure gives calibration constants that are particular to the class of materials being measured, but are independent of the

TABLE III. Plutonium Isotopics Measurements Compared to Chemistry			
Sample ID	**Peff: ratio of FRAM/CLS-1**	**Peff: ratio of Fleissner/CLS-1**	**Bias without Heterogeneous Analysis**
MSE-1	0.9806	0.9612	1.27
MSE-2	0.9451	0.9338	1.58
MSE-3	no FRAM data	0.9929	
MSE-4	0.9523	0.9472	1.44
MSE-5	no FRAM data	0.9934	
ARF876642	no FRAM data	0.9866	
XBLP121	1.0049		1.23
XBLP278	0.9905		1.09

sample geometry. The calibration constants are functions of neutron detection efficiency, induced fission (IF) moments, spontaneous fission moments, and coincidence gate fraction, as well as the multiplication constant k. The derivation of this procedure will be described in detail in a forthcoming LAMS report.

We have applied the technique to crushed MSE residues[27] and impure oxide materials from two different sources. The results for the residues are shown in Fig. 7, the results for the impure oxides are shown in Fig. 8. The technique also has been applied to pure oxide materials. Figure 9 shows these results.

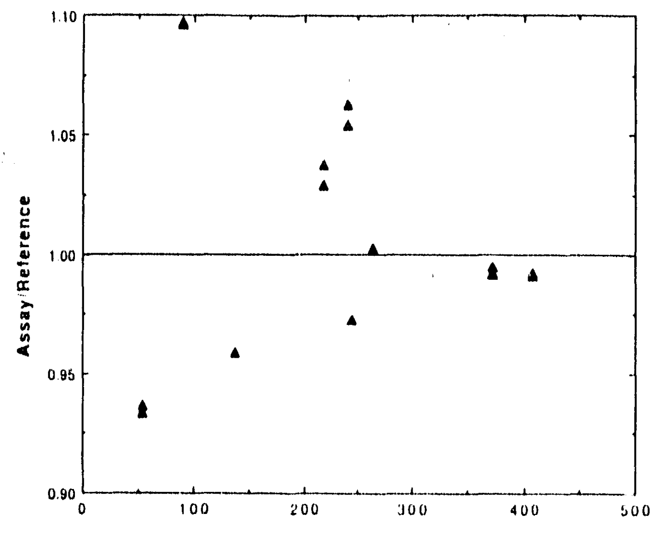


Fig. 7. Assay results obtained using the geometry-based multiplication correction technique divided by the reference value vs ^{239}Pu effective for crushed MSE spent salts.

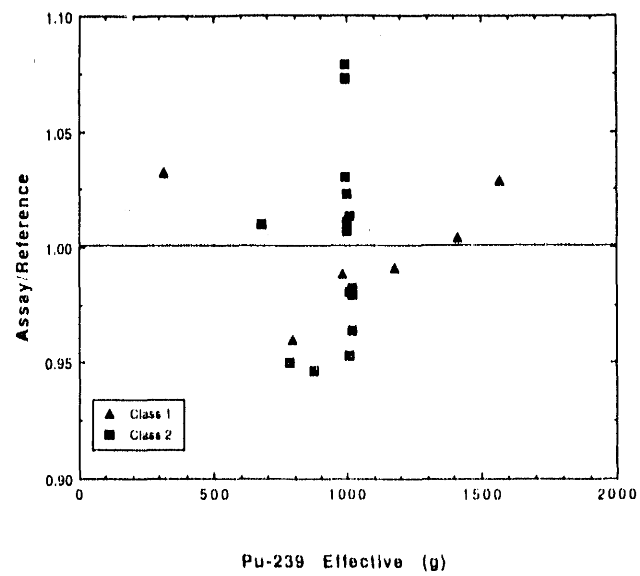


Fig. 8. Assay results obtained using the geometry-based multiplication correction technique divided by the reference value vs ^{239}Pu effective for two classes of impure plutonium oxide samples.

c. Assay of Pyrochemical Residues with the Advanced Segmented Gamma Scanner (SGS) (J. K. Sprinkle, Jr. and S.-T. Hsue, N-1; V. L. Longmire, NMT-4). Conventional NDA methods are used to measure quite accurately the plutonium content of many forms of relatively pure, homogeneous bulk items. However, physical and chemical heterogeneities combined with high and variable impurity levels, which are present in many categories of processing scrap, produce biases in conventional NDA results. These categories of scrap also present a significant challenge to the assignment of reference values to selected items for evaluating new NDA methods.

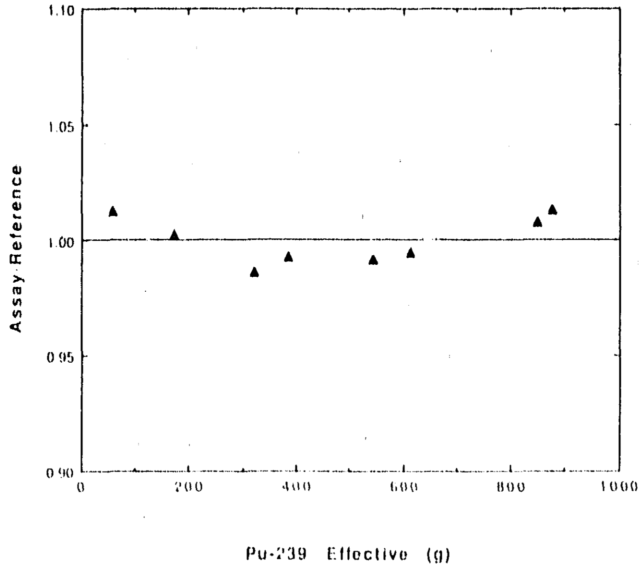


Fig. 9. Assay results obtained using the geometry-based multiplication correction technique divided by the reference value vs ^{239}Pu effective for pure plutonium oxide samples.

We recently evaluated several NDA methods in a study[28] using pyrochemical residues from MSE of the americium. This category of scrap contains lumps of metallic plutonium in a salt matrix; the americium has been partially extracted from the plutonium. We chose these samples, which present one of the most difficult measurement challenges, to study a large-bias case in conventional NDA measurements and to help evaluate the new measurement corrections in our advanced SGS.[27] We also felt that any measurement techniques that were effective for this category of scrap would perform adequately for less difficult assay problems. Once reference values were obtained for several selected samples, it became clear that the advanced SGS[29] showed significant improvement over the traditional SGS.[30] The bias in the results (summed over 14 samples) decreased from 9% to 4%. The samples ranged from 52 g of ^{239}Pu per item to 384 g. Most of the correction was obtained by the advanced SGS in a few segments, because the metallic lumps usually were concentrated in a small volume in each of the samples.

7. Assay of Impure, Plutonium-Rich Process Materials — Plutonium Solution Assay Instrument (SAI) with Isotopic Capability (S.-T. Hsue, S. M. Simmonds, and T. Marks, N-1). Plutonium SAIs often are used to determine plutonium concentrations for accountability purposes. At the request of the Los Alamos Plutonium Facility, we have developed an advanced in-plant system with a wide density range (1 - 300 g/L) that determines both plutonium concentration and isotopic distribution simultaneously. The facility has installed a continuous-feeding dissolver in the plutonium recovery process to speed up the dissolution of scrap and waste, and, because this is not a batch process, the concentration and isotopic distribution of the product solution are unknown and must be measured.

We developed the original plutonium SAI[31,32] ten years ago and installed it in Los Alamos's Plutonium Facility. Although this instrument has served its purpose by providing the capability of determining solution concentration in the process line, we also have addressed two operational problems in the new design. Some of the solutions at Los Alamos contain abnormally large amounts of ^{237}Np. In the original SAI, assays of plutonium solutions contaminated with ^{237}Np and ^{233}Pa can be excessively biased because of the presence of the 415.76-keV gamma ray from ^{233}Pa. Also, some mechanical parts, which worked well in the beginning, began to develop problems in the hostile acid environment of the glove box after several years. The tungsten shutter, although it is gold plated, has a tendency to corrode in the acid atmosphere, and after a period of time has difficulty in rotating and shuttering the transmission source. One purpose of the new SAI is to overcome these problems as well as to include a plutonium isotopic determination capability.

Protactinium 233 emits a 415.76-keV gamma ray that interferes with the 413.71-keV gamma ray of ^{239}Pu, which is the main peak used in ^{239}Pu assay with the region-of-interest method of peak area determination. We solved this problem by response-function fitting the overlapping peaks. Peak fitting also allows for the presence of a potential pileup peak at 2 x 208 or 416 keV, although the shape of the pileup peak may be different from the regular photopeak. Because the fitting technique can tolerate the pileup peak, the 0.76-mm-thick tungsten filter in front of the detector is not necessary, and the 129.29-keV peak also can be used to assay ^{239}Pu; this will improve the precision at low concentrations. We use the 129.29-, 345.014-, 375.01-, and 413.72-keV gamma rays of ^{239}Pu for the assay.

To make quantitative assays at these energies, sample self-absorption corrections are necessary when assaying over a wide concentration range and a variety of solution types. We selected ^{75}Se as the transmission source; transmissions are measured at 136.00, 279.53, and 400.65 keV. These transmissions are interpolated to other energies by means of quadratic fits.

To determine the plutonium isotopic distribution in a relatively short time (<500 s), we use the multi-group analysis 2 technique.[33] This technique uses the gamma-ray information from 60 keV to 208 keV to determine the plutonium isotopic distribution.

The hardware of the new SAI (Fig. 10) that resides in the glove box comprises three segments. The top segment contains the transmission source, the shutter, and the shutter motor. The shutter is driven by a dc gear motor; the direction of the shutter is controlled by a reversing actuator. This segment of the SAI will be filled with

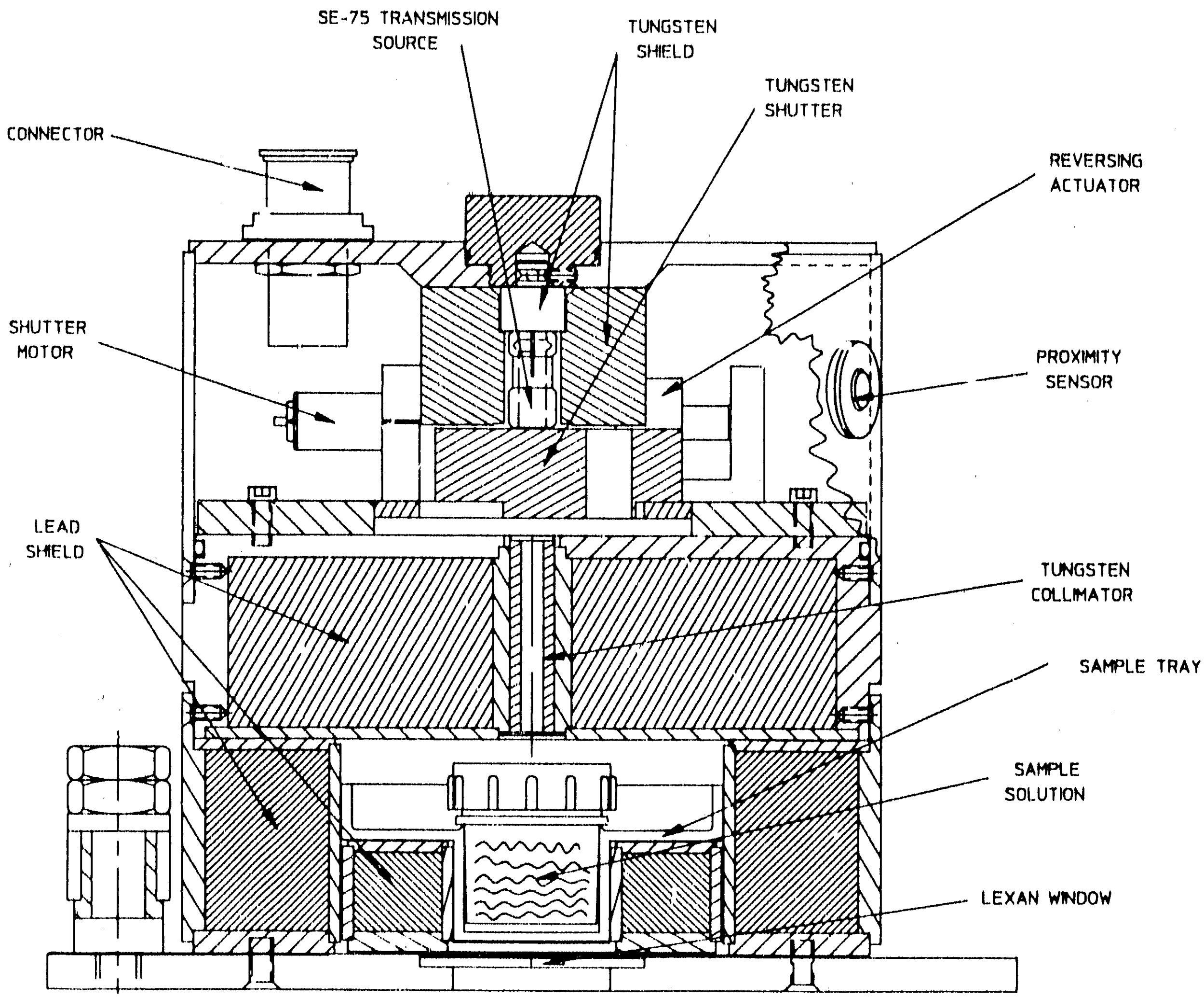


Fig. 10. Sample chamber of the advanced SAI residing in the glove box. The top portion contains the transmission source, the tungsten shutter, and the shutter motor.

clean air to prevent acid corrosion. The second segment is primarily a stainless steel housing filled with lead to shield the detector from background radiations; the center of this segment is a tungsten collimator for the transmission source. The bottom segment is a plastic well that holds the sample solution during assay. This segment is hinged and can be opened for sample loading and unloading.

The new system is scheduled to be completed by April 1990 and will be installed soon afterward in the plant.

8. A Method for Accurate Moisture Corrections to Passive Neutron Coincidence Assays of Bulk Mixed-Oxide (MOX) Powders (J. E. Stewart and H. O. Menlove, N-1; M. Aparo and F. Troiani, ENEA)

Introduction. Experiments were completed at Saluggia, Italy to establish quantitatively the effects of entrained sample moisture on High Level Neutron Coincidence Counter (HLNC-II) assays of uranium-plutonium MOX powders. The experiments also tested and calibrated two prototype detectors designed for independently measuring the amount of moisture in MOX and PuO_2 samples.

Previous calculational studies on HLNC-II measurements of moist PuO_2 powders[34] quantified the components of the assay bias and indicated that if the wt% of water in the sample were known, a correlation could be used successfully to correct HLNC-II assays.

The Collaboration. As part of the Italian Support Program to the International Atomic Energy Agency (IAEA), ENEA-Cassaccia made arrangements with the EUREX plant at Saluggia to make available ~1100 g of MOX for the experiments. Before the exercise, the plant developed an innovative approach for adding controlled amounts of water to the powder to produce a homogeneous mixture. Representatives of the IAEA, the Commission of European Communities (CEC) Safeguards Directorate (Luxembourg), the CEC Joint Research Centre

(Ispra), and Los Alamos, participated in the exercise along with those from ENEA (Cassaccia and Saluggia).

Moisture Monitors. Los Alamos provided two prototype moisture monitors for the MOX measurements at Saluggia. Both monitors include a standard HLNC-II. The first monitor consists of three additional ^{3}He detector tubes in polyethylene moderator blocks, which are attached to the exterior of the HLNC-II to form a Girdle. The second prototype is essentially an HLNC-II with no polyethylene moderator and only six ^{3}He detectors (instead of the standard 18). This unit is called the Air Counter/HLNC-II because of the absence of a moderator. Both prototype monitors are shown in Fig. 11, and the HLNC-II/ Girdle is shown in Fig. 12. The Monte Carlo simulation geometry of the HLNC-II/Girdle is shown in Fig. 13.

Results. Both prototype units and a EUREX-plant detector similar to the HLNC-II were used to measure all of the MOX samples. Tightly controlled moisture concentrations between 0 and 9 wt% were obtained inside a glove box for two MOX masses (~600 g and ~1100 g). Calibration curves were developed for both moisture monitor prototypes.

Multiplication-corrected coincidence assays were computed for all samples both with and without corrections for moisture. The performance figures for the two prototype moisture monitors are compared in Table IV as average biases in HLNC-II assays of ^{240}Pu-effective masses for all samples. Four cases are represented: (1) no correction for moisture, (2) a correction based on the known moisture concentration, (3) the correction obtained from the HLNC-II/Girdle moisture measurement, and (4) the correction determined from the Air Counter/HLNC-II moisture measurement.

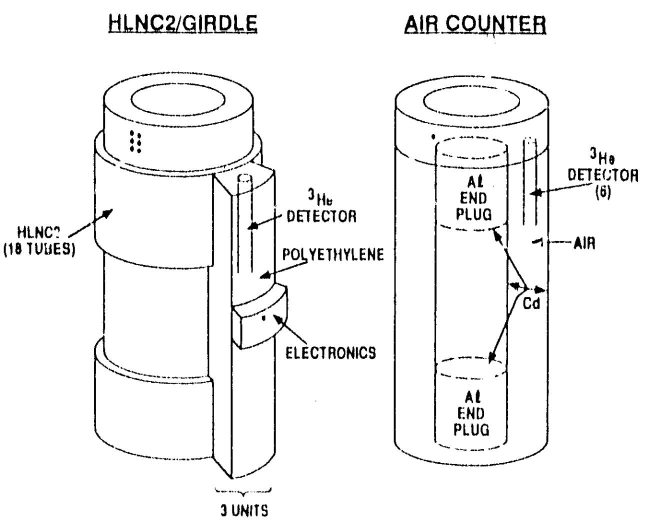


Fig. 11. Partial schematics of two prototype moisture monitors.

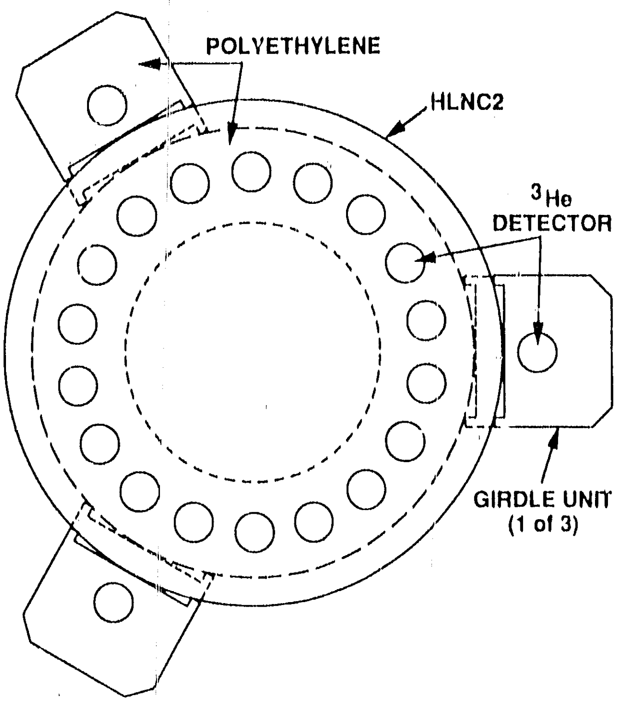


Fig. 12. Full section view of the HLNCII/ Girdle moisture monitor.

Table IV shows that the HLNC-II/Girdle measurement is only slightly less accurate than the case in which the water concentration is known from weighing. The Air Counter/HLNC-II results are certainly an improvement over the no-correction case, and show a smaller bias but more scatter than results from the HLNC-II/Girdle. Figure 14 shows the HLNC-II multiplication-corrected real coincidence rate divided by the ^{240}Pu(effective) mass vs water content for both sample masses. The data have been corrected for the moisture bias using the moisture content determined from the HLNC-II/Girdle totals ratio in combination with a correlation described in Ref. 34.

Conclusions

- Results of the Saluggia experiments establish the feasibility of using differential neutron-moderation ratios to determine moisture levels in relatively small MOX samples. The ratios also depend on sample mass.
- A procedure for moisture corrections[34] of HLNC-II assays has been validated experimentally. An iterative modification of this procedure for removing the sample mass effects on the moderation ratio has been proposed and tested. The modified procedure should be tested for MOX samples with larger masses.
- The HLNC-II/Girdle provides good sensitivity to MOX moisture levels above 1% by weight. Below this concentration, the moisture bias is not significant.

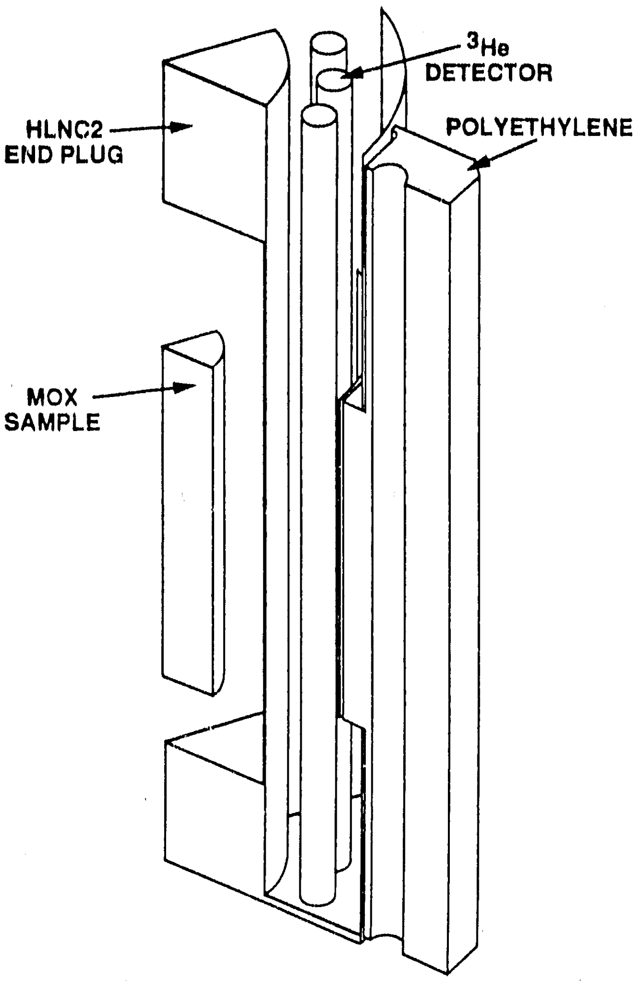


Fig. 13. Monte Carlo simulation geometry for the HLNCII/Girdle.

Table IV. Performance Summary for Moisture Monitors		
Case	**Moisture Correction**	**Avg. Assay Bias for All Samples (%)**
1	None	3.6 ± 3.0
2	Known weight % H_2O	-0.7 ± 0.7
3	HLNC2/Girdle	-1.5 ± 1.2
4	Air Counter/HLNC2	-0.4 ± 2.3

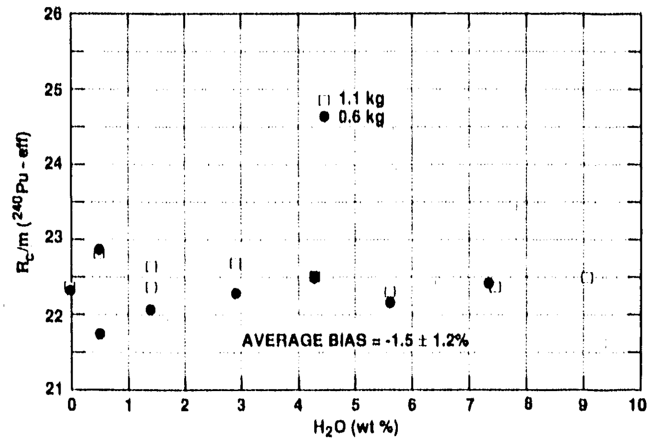


Fig. 14. HLNC-II multiplication-corrected reals per gram of ^{240}Pu (effective) vs H_2O concentration (0 to 10 wt %) for two MOX samples (0.6 and 1.1 kg). The average bias and its uncertainty also are shown.

- The Air Counter/HLNC-II provided good sensitivity to the MOX moisture level for the 1.1 kg MOX sample. For the 0.6 kg sample, sensitivity was marginal.
- For inspectorates, the moisture level of every MOX sample to be verified would not necessarily be measured. However, outliers should definitely be measured in the moisture monitor and the expanded data analysis should be performed. Random, periodic moisture measurements also appear to have value.
- For plant operators concerned with criticality safety limitations, the moisture level of every sample must be verified.

Summary. Because of the efforts of ENEA (Cassaccia) and EUREX plant personnel, a unique set of experiments was made possible. The resulting data, which were not previously available, are essential for evaluating methods for removing moisture bias effects from HLNC-II verification measurements of MOX and PuO_2.

The results shown in Table IV establish the feasibility of using the HLNC-II/Girdle for moisture determination in 0.6- to 1.1-kg MOX samples with up to 10 wt% of water. Most of the scatter in the Air Counter/HLNC-II results comes from poor counting statistics on the 0.6 kg sample. For the 1.1-kg sample, the Air Counter performed as well as the Girdle, but the potential for implementation is smaller because integration with the existing HLNC-II is more costly.

Finally, with these moisture measurements, the validity of a procedure (described in Ref. 34) for correcting HLNC-II assays has been established experimentally. Further experiments and calculations are planned to verify the approach for larger sample masses.

9. Assay Methods for Other Nuclear Materials (J. K. Sprinkle, Jr. and E. L. Adams, N-1). The ability to assay neptunium is of interest to

DOE facilities that are involved in the production of ^{238}Pu heat sources because ^{237}Np is the feed for reactor production of ^{238}Pu. In exploring techniques for the assay of ^{237}Np,[35] we have placed small containers bearing uranium, plutonium, or neptunium in a barrel shuffler. The relative responses agree quite well with those predicted by Monte Carlo calculations.[18] With a cadmium-lined shuffler, the response per gram from ^{237}Np is similar to that from ^{238}U, which is an order of magnitude less than the response per gram for ^{235}U or ^{239}Pu. Thus, an accurate assay for ^{235}U or ^{239}Pu in the sample would require corrections for any ^{238}U or ^{237}Np that is present.

10. Determination of Plutonium Isotopic Composition and Plutonium Concentration by Isotope Dilution Gamma-Ray Spectroscopy on Resin Beads (T. K. Li, N-1; Y. Kuno, K. Nakatsuka, and T. Akiyama, TRP/PNC, Japan). We have developed a new technique -- isotope dilution gamma-ray spectroscopy (IDGS) -- for simultaneously determining the plutonium concentration and the isotopic composition of highly irradiated fuel dissolver solutions, such as the input to a chemical reprocessing plant. The IDGS technique combines the high-resolution, low-energy gamma-ray spectroscopy technique, the isotope dilution technique, and the resin bead technique. It involves adding a well characterized plutonium isotope (spike) to the unknown solution and then extracting the plutonium from the spiked (mixed) samples on resin beads and subsequently measuring the beads with high-resolution gamma-ray spectroscopy. The isotopes ^{236}Pu, ^{238}Pu, ^{239}Pu, and ^{240}Pu are all good candidates as a known spike for the IDGS technique. However, for reasons of cost and availability, ^{239}Pu is the best choice. We used a large size dried (LSD) spike[36] of ^{239}Pu for our experiments. Its certified isotopic composition (in atom %) is 0.002461% ^{238}Pu, 97.93026% ^{239}Pu, 2.05199% ^{240}Pu, 0.013984% ^{241}Pu, and 0.001304% ^{242}Pu.

The concentration of plutonium in the unknown input dissolver solution, (or other unknown sample), C_u, can be determined as follows:

$$C_u = \frac{M_s}{V_u} \cdot \frac{W_s^9}{W_u^9} \cdot \frac{R_m - R_s}{R_u - R_m} , \quad (1)$$

where

M_s = mass of plutonium in the spiked sample,

R_u = ^{240}Pu/^{239}Pu ratio in the unspiked (unknown dissolver solution) sample,

R_m = ^{240}Pu/^{239}Pu ratio in the spiked (mixture of dissolver solution and spike) sample,

R_s = ^{240}Pu/^{239}Pu ratio in the spiked sample,

W_s^9 = Weight percent of ^{239}Pu in the spiked sample,

W_s^9 = Weight percent of ^{240}Pu in the unspiked dissolver solution, and

V_u = Volume of dissolver solution taken.

In this equation, the values of M_s, W_s^9, R_s, and V_u are known. Therefore, only values of R_u and W_s^9in the unspiked sample of dissolver solution and R_m in the spiked sample are to be measured by gamma-ray spectroscopy.

The measurement method is based on high-resolution, low-energy gamma-ray spectroscopy. Details of the measurement technique and of the resin bead sample preparation procedure are described in Refs. 37 and 38. Four aliquots were prepared after the LSD spike was mixed well with the precisely known volume of the input dissolver solution. Each LSD-spiked aliquot originally contained approximately 4.5 mg of plutonium from the LSD spike and 1 mg of plutonium from the dissolver solution; approximately 60% of the plutonium is lost during sample preparation because fission products are necessarily well rinsed out. Because the LSD spike is expensive and difficult to obtain, each LSD-spiked aliquot was diluted to X2 (half of the original concentration, designated LSD1), X4 (one quarter of the original concentration) (LSD2), X8 (LSD3), and X16 (LSD4) to determine an optimum dilution. The plutonium masses contained in the LSD1, LSD2, LSD3, and LSD4 diluted samples were approximately 1.1 mg, 0.55 mg, 0.28 mg, and 0.14 mg, respectively. After washing to remove fission products, uranium, and americium with 8M HNO_3, the plutonium was eluted, its acidity was adjusted with 8M HNO_3, and it was absorbed in a small gauze bag filled with resin beads. The resin bead samples were bagged out of the glove box and placed directly in front of a high-purity germanium (HPGe) detector for the IDGS measurements.

Figure 15 shows low-energy gamma-ray spectra of the (a) LSD spike, (b) unknown dissolver solution, and (c) the LSD-spiked resin bead samples. The ^{240}Pu/^{239}Pu atomic ratios obtained from both IDGS and traditional isotope dilution mass spectroscopy (IDMS) for LSD-spiked samples (part A) and for the dissolver solution sample (part B) are summarized in Table V. Columns 2 to 5 show the IDGS results for various dilution factors (X2, X4, X8, X16) of four LSD-spiked samples. Most of the data (ratios) shown are averages of two or three independent measurements. A few samples were contaminated by fission products during preparation in a hot glove box; data from these are not included in the averages. Direct interferences of the K x-rays from fission products with the plutonium low-energy gamma rays will affect the accuracy of the measurement. Furthermore, the dramatically increased continuum background produced by higher energy gamma rays from fission products will reduce the precision

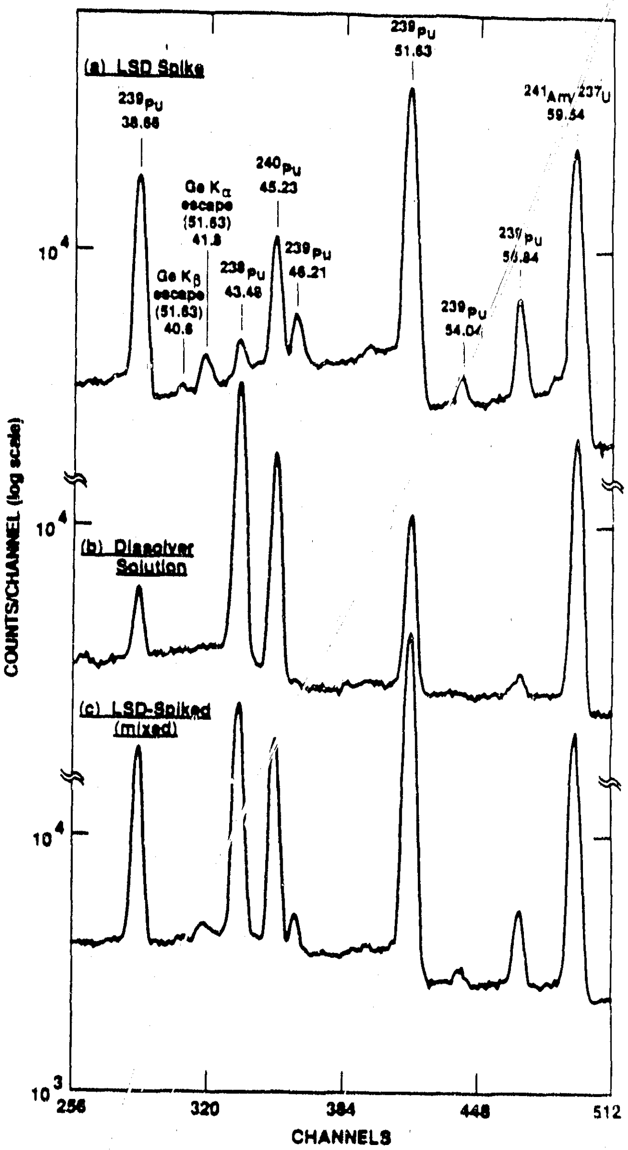


Fig. 15. Low-energy gamma-ray spectra of (a) LSD spike, (b) unknown dissolver solution, and (c) LSD-spiked (mixed) resin bead samples.

of the measurement. Figure 16 compares the gamma-ray spectrum from a fission-product-contaminated resin bead sample (dotted spectrum) and the spectrum from the same sample after it was rewashed to remove fission products. In Table V, the last column shows the ratio of the averaged ^{240}Pu/^{239}Pu ratio obtained from IDGS (column 6) and the ^{240}Pu/^{239}Pu ratio obtained from IDMS (column 7) for each of the LSD-spiked samples. The average of the IDGS/IDMS ratio of ^{240}Pu/^{239}Pu is 0.9840 with a relative standard deviation (RSD) of 0.22%. The bias of 1.6% between the IDGS and IDMS results arises because the gamma-ray system has not yet been calibrated with this type of sample. Because the measurement is affected by relative efficiency determination, peak integration, background subtraction, branching ratios, and half-life selections this value probably can be used as a calibration factor. Additional measurements and extensive analysis are needed to confirm this.

The total plutonium concentration (in g/L) of the unknown dissolver solution can be determined from Eq. (1) by using the measured ^{240}Pu/^{239}Pu values for the LSD-spiked sample (R_m) and for the unspiked dissolver solution (R_u) in Table V and certified values for W_s^9 and ^{240}Pu/^{239}Pu (R_s) for the LSD spike. The volume of the dissolver solution was 0.9958 ml for each spiked sample. The mass of the LSD spike (M_s) was 4.5021 mg for LSD1, 4.49898 mg for LSD2, 4.48541 mg for LSD3, and 4.49349 mg for LSD4. The measured weight fraction, of ^{239}Pu, W_u^9 in the dissolver solution was 0.66876 for IDMS and 0.657419 for IDGS. The results of the total plutonium concentration determined from IDGS and IDMS are compared in Table VI. The average plutonium concentration obtained by IDGS agrees with that obtained by IDMS within 0.042% with an RSD of 0.384% (the last column in Table VI). The ^{240}Pu/^{239}Pu ratios show a slight bias between IDGS and IDMS; the excellent agreement for plutonium concentration may be a result of the systematic errors in the calibration factor canceling in Eq. (1) when we calculated the concentration. Further examination is underway.

The estimated precision (1σ) for the ^{240}Pu/^{239}Pu ratio of the spiked sample, as a function of dilution factor and count time, is given in Table VII. The count time required to give a precision better than 1% when measuring the ^{240}Pu/^{239}Pu ratio in the spiked sample is about 1 h for the X2 dilution, 2 h for the X4 dilution, and 4 h for the X8 dilution. For rapid routine measurements, either X2 dilution or X4 dilution is recommended.

The IDGS technique may rapidly and accurately verify input and intermediate process plutonium samples, which are very important for near-real-time accounting at reprocessing plants. It also is a potential on-site verification method for IAEA inspections. By implementing this new technique, the IAEA could significantly reduce the number of samples sent to Vienna for IDMS analysis. Although the results of the first IDGS measurement are very promising, further development work is underway to improve the gamma-ray analysis and the resin bead preparation. Improvements to gamma-ray analysis would include peak fitting, interpolation of background using a smoothed step function, and an efficiency calibration for a defined sample-to-detector geometry. Modifications to the sample preparation would include reducing preparation time with a simplified procedure and reducing potential fission product contamination by using an automatic sample preparation system. Future experiments are planned that use different plutonium concentrations in dissolver solutions and different spike-to-dissolver-solution ratios.

TABLE V. Comparison of the $^{240}Pu/^{239}Pu$ Ratio (wt%) by IDGS and IDMS

	$^{240}Pu/^{239}Pu$ ($\times 10^{-3}$)						
	IDGS					IDMS	IDGS/IDMS
	x2	x4	x8	x16	Average		
A. LSD							
LSD 1		60.320	60.591		60.456	61.613	0.9812
LSD 2	60.532	60.589	60.834		60.651	61.672	0.9834
LSD 3	60.033	61.137	61.036		61.069	61.937	0.9860
LSD 4		60.943	60.472	61.315	60.910	61.817	0.9853
Average							0.9840
RSD(%)							0.22
B. Dissolver Solution					333.4	336.552	0.9906

D. Detector and Electronics Development for NDA.

To meet demands for more sophisticated measurements and more compact measurement equipment for both materials control and materials accounting, we continue to develop new NDA detector configurations and to study and improve detector technology.

1. Compact NaI(Tl) Detectors (P. A. Russo, M. M. Stephens, and S. C. Bourret, N-1). The development of compact NaI(Tl) detectors is important to several applications in NDA for nuclear safeguards. For about ten years, portable low-resolution gamma-ray spectroscopy with the Davidson Model 4096 portable multichannel analyzer (PMCA) has been in widespread use, but has been limited to operation with standard commercial NaI(Tl) detectors that are significantly larger than is often required (6-cm diam, 23- to 26-cm long) and too heavy to be conveniently portable when the required shielding and collimation are attached. For many portable applications, including holdup measurements, a commercially available detector that is both smaller and compatible with the PMCA is needed.

Some stationary low-resolution gamma-ray measurements require large (7.6-cm-diam, 7.6-cm-long) NaI(Tl) crystals to detect gamma rays at high energies, as well as thick lead shields to limit background contributions to the spectra at the same high energies. Typical detector lengths are 30 cm for commercial equipment. An example is the four-detector measurement station for the HEU Shipper/Receiver Confirmatory System[39] (SRCS) at the Oak Ridge Y-12 Plant, which requires 2000 kg of lead shielding. Because the SRCS concept will be reproduced at other facilities with more than a single station at some facilities, a commercially available (7.6-cm by 7.6-cm) NaI(Tl) detector that requires significantly less shielding is urgently needed.

A small NaI(Tl) detector,[40] which we developed, has recently been commercialized in a lead-collimated (2.5-cm-diam by 2.5-cm-long collimator) shielded package. This compact unit is shown in Fig. 17 along with its unshielded commercial predecessor. For HEU applications [2.5-cm-diam by 1.3-cm-thick NaI(Tl) crystals], the weight of the collimated, shielded detector is only 1.8 kg, and the length of the complete unit (18 cm) is significantly shorter than that of the unshielded predecessor. We are testing this detector and evaluating the digital gain drift compensation techniques that will substitute for the analog approach that requires americium-seeded NaI(Tl) crystals.

We also have tested a prototype commercialized version of a large NaI(Tl) detector with compact electronics. The new prototype uses a 7.6-cm by 7.6-cm NaI(Tl) crystal and a 7.6-cm-diam photomultiplier tube (PMT) whose length (5.6-cm) is much shorter than the lengths of the PMTs used in previous commercial detectors. The voltage divider, designed at Los Alamos, uses a 7.6-cm-diam

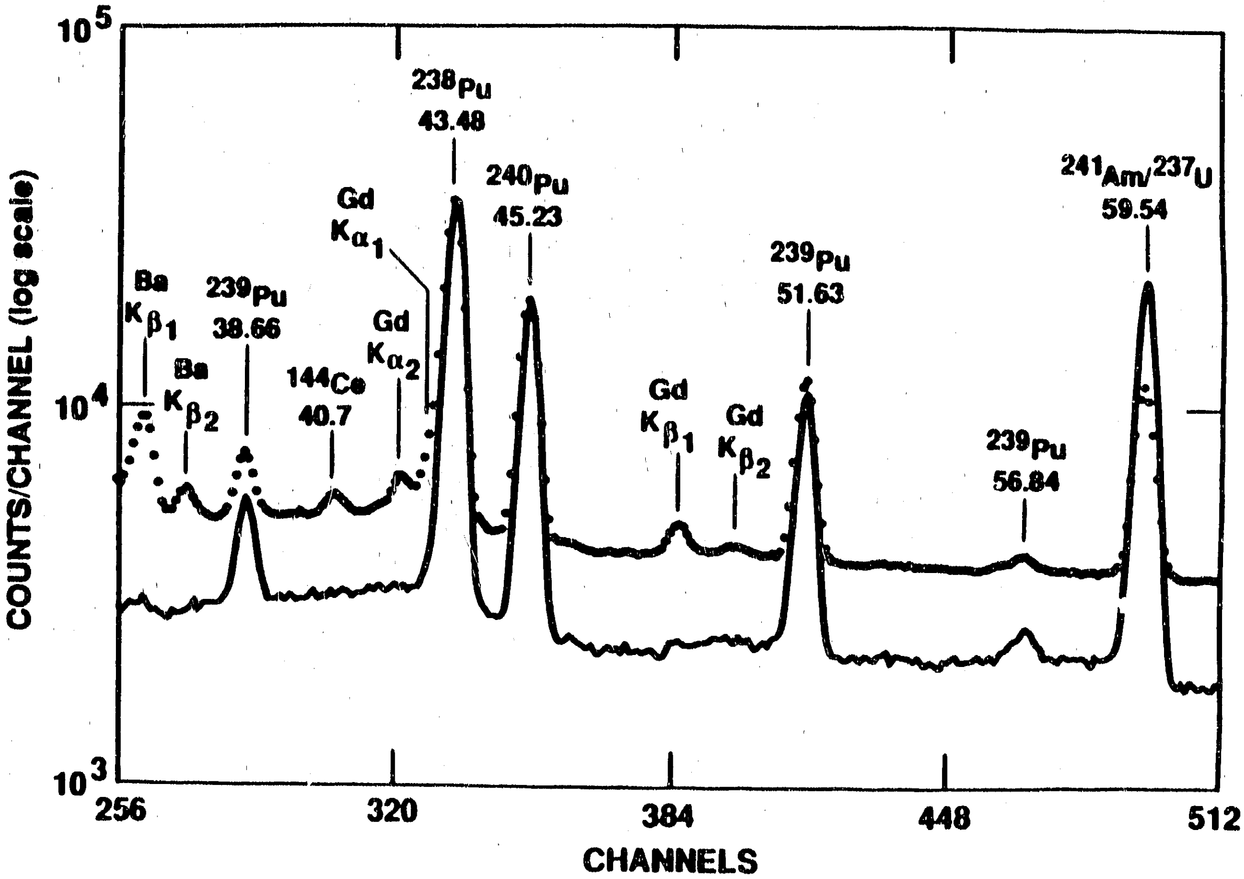


Fig. 16. Comparison of the low-energy gamma-ray spectrum from a fission-product-contaminated resin bead sample (dotted spectrum) and from the same sample after rewashing the fission products.

TABLE VI. Comparison of Total Plutonium Concentration of Dissolver Solution by IDGS and IDMS

	Total Plutonium Concentration (g/ℓ)		
	IDMS	IDGS	IDGS/IDMS
LSD 1	0.97686	0.97242	0.99545
LSD 2	0.97782	0.97725	0.99942
LSD 3	0.98217	0.98610	1.00400
LSD 4	0.98062	0.98337	1.00281
Average			1.00042
RSD(%)			0.384

TABLE VII. Estimated Precision (%) for LSD-Spiked (Mixed) Samples

	$^{240}Pu/^{239}Pu$				
Dilution	1/2H	1H	2H	4H	10H
X2	1.3	0.9	0.6		
X4	1.8	1.2	0.8	0.6	
X8			1.4	1.0	
X16			4.1	3.0	2.1

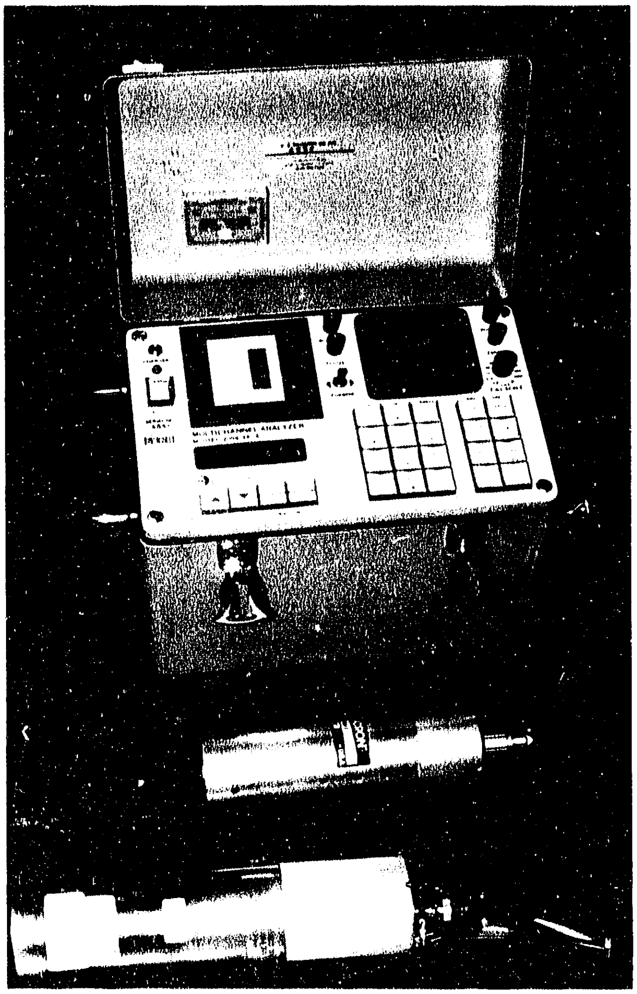


Figure 17. The commercialized, collimated, shielded, compact NaI(Tl) detector (2.5-cm-diam by 1.3-cm-thick crystal) is shown directly in front of the PMCA. In the foreground is its commercial predecessor (5.1-cm-diam by 1.3-cm-thick crystal) with no shield or collimator.

printed circuit board that mounts close to and parallel to the pin end of the PMT. We recently tested the commercial prototype with the preamplifier that was designed for the portable NaI(Tl) detectors and have placed an order for a completely commercialized integral detector. The overall length of this unit is expected to be less than two-thirds that of its commercial predecessor, which will result in a significant decrease in shielding weight for measurement systems such as the SRCS.

2. Upgraded Shift-Register Electronics (M. S. Krick, S. C. Bourret, P. R. Collinsworth, J. K. Halbig, and B. Strait, N-1; W. Baird, MEE-3). We are developing an upgraded shift-register coincidence electronics package. The shift-register circuit is designated the SR4 because it uses a 4 MHz shift-register clock. We designed a printed circuit board for the SR4 circuit, and several units have been fabricated as nuclear instrument modules for test and evaluation (T&E). We have used these SR4 modules to check detectors and measure coincidence standard-deviations, detector die-away profiles, and coincidence scans.

The SR4 circuit has a useful random count-rate limit of 2.6 MHz, gate lengths from 0.25 to 4096 µs in 0.25 µs steps, predelays from 0 to 4095.75 µs in 0.25 µs steps, and a long delay of 16 milliseconds. All settings are under computer control using serial communication.

The final SR4 unit will be a self-contained instrument for neutron coincidence counting; it will include high- and low-voltage power supplies and a user-interface panel so that it can be operated without a computer, if desired. Two prototypes are being developed for test and evaluation; these should be completed in 1990.

E. Technology Development for Holdup Measurements.

We have an ongoing task to develop and document specialized instrumentation for in-plant holdup, waste, confirmatory, and inventory verification measurements to meet current and anticipated needs of the DOE and commercial nuclear facilities.

1. Compact Neutron Instrument for Portable or In-Line Measurements of SNM (P. A. Russo, B. G. Strait, H. O. Menlove and N. Ensslin, N-1). Several applications for compact neutron detectors for in-line measurements of SNM have been described previously.[41-43] Some of these (for example, those involving fluorinated chemical forms of plutonium) are well-suited to total neutron count rate measurements, while others (such as the measurements of plutonium holdup in tilt-pour furnaces) may be more appropriate for neutron coincidence counting.

Upgrades of the hydrofluorination process at the Los Alamos Plutonium Facility will require the ability to monitor this process to improve the product and thereby minimize failures in the subsequent reduction process.[42] Such failures complicate materials accountability in that larger amounts of plutonium are incorporated into the residues from the reduction process. At the request of the facility operator, we will design two neutron detectors for continuous readout of the total neutron count rate from the two hydrofluorinator processes that operate simultaneously. The two detectors will monitor the production of PuF_4, and will be used in a calibrated measurement to signal completion of the conversion of batches of PuO_2 to PuF_4. This method also requires inferring the influence of neutrons produced by each process on the neutron detector that monitors the other process. An in-plant test of the principles of the proposed measurements of the hydrofluorination process has been documented.[41]

Initially, the method will use a hardware approach similar to the Swansen self-contained ^{3}He neutron discriminator hardware.[43,44] However, as the compact measurement needs at Los Alamos and elsewhere extend to neutron coincidence counting, we will incorporate the new compact shift register into the design for a compact neutron counter that will allow counting of total and coincident neutrons for in-line and portable applications in a highly-reliable, compact, and self-contained unit. Long-term requirements for the coincidence counter include (1) self-contained analog/logic circuits for pulse-height discrimination and analysis of time-correlated neutron events, (2) self-contained high voltage and low voltage power, (3) self-contained readout of total and coincident neutron count rates and elapsed count time, (4) self-contained counting setup capability, (5) self-contained hardware setup capability, and (6) standard interfaces for readout to data storage, external processor, or hard-copy units. The self-contained features of the "totals" neutron counter will be a subset of those listed for the coincidence unit. The "totals" detection unit will be coupled to external commercial units (to scale count rates, for example), and these will be equipped with standard interfaces to link to an external processor, data storage unit, or hard-copy unit.

The totals counters for the Los Alamos hydrofluorinator monitoring application will be implemented in late summer 1990. The design for the compact, self-contained coincidence unit will be underway at that time. Part of the hydrofluorinator monitoring effort is funded by the facility.

2. Technologies for Portable Gamma-Ray Holdup Measurements (P. A. Russo, J. A. Painter, D. C. Garcia, and S. M. Simmonds, N-1). The ability of many DOE facilities to meet enhanced requirements from regulatory agencies for quantitative measurements of nuclear materials holdup will depend in part on improvements in technologies that support these measurements. Because of the age, size, and complexity of many of the facilities, large numbers of measurements using portable instruments may be routinely required. Expressions of user needs in these areas are increasing. Combining these with a growing base of experience in gamma-ray holdup measurements, we have placed emphasis on compact NaI(Tl) detectors, versatile mechanical equipment, compact and more capable instrumentation, and enhanced automation for the Los Alamos portable gamma-ray holdup assay system.

a. Design of Compact Detector Systems. The design of compact NaI(Tl) detectors for low-resolution gamma-ray measurements using the Davidson PMCA has recently been transferred to the commercial sector (see Part 1, Section D.1). Commercial HPGe detectors customized for increased portability in holdup applications also are documented elsewhere.[45] (see also Part 3, Section I.B.1) Further efforts toward increased compactness are underway.

The engineering drawings for the versatile mechanical equipment that was field tested during measurements of plutonium holdup in glove box process equipment[45] (see also Part 3, Section I.B.1) were completed recently[46] and have been provided to users on request. Additions to the equipment design will be included as updates in this drawing package. Figure 18 is a photograph of the equipment (with detector and data acquisition instrumentation in place) represented by the current design package.

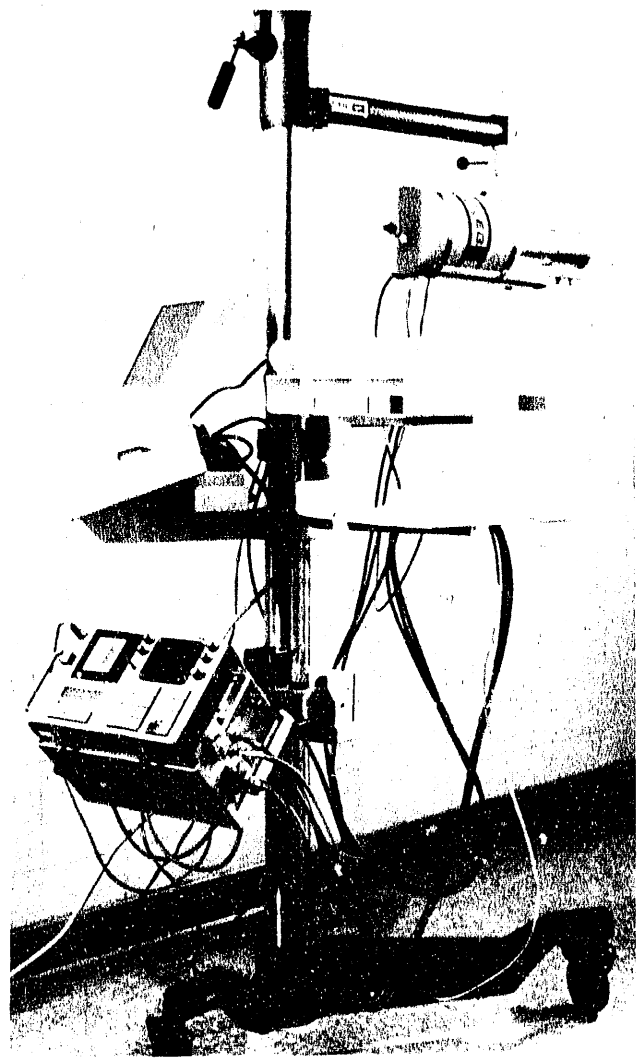


Fig. 18. Portable gamma-ray holdup measurement system with mechanical equipment represented by the current design package.

b. Improved Compact (PC-compatible) Instruments. We have incorporated a variety of compact instruments into the holdup assay system. Laptop computers with 80286 or 80386 processors, 40-Mbyte fixed disks, floppy disk drives, and improved liquid crystal (or gas plasma) displays have been implemented and tested. The testing included verification of the automation capability for the Davidson model 2056 PMCA along with transfer of spectral data between the PMCA and the PC. The fixed disk for mass storage of spectral data is particularly important for high-resolution applications. It also allows for mass storage of files of measurement results that are to be used off line with data base, spread sheet, or graphing software packages. Improvements in the traditional liquid crystal displays are important for measurements in dimly-lit areas. We are testing automated input of a measurement (sample or location) identification number with interfaced bar code reader hardware. This is important for accomplishing large numbers of measurements in rapid succession at prescribed (coded) locations, for accurately entering location or sample information, and for subsequently retrieving stored measurement results for specified location or sample codes.

c. Enhanced Automation. Automation of the portable gamma-ray holdup assay system is now available in the new software package written in the C programming language. The new code provides automated electronics setup, the choice of (low- or high-resolution) gamma-ray detectors, digital gain drift compensation, quality checking of all spectra, an automated bias check, assays at multiple gamma-ray energies, and an automated calibration procedure for generalized-geometry holdup assays, all of which were included in the prototype software written and compiled in Microsoft Quick Basic 3.0.[45,47] (see also Part 3, Section I.B.1) The new C language software adds automated rate-loss corrections, the ability to assay at preset measurement configurations, automated corrections for equipment attenuation effects, and the ability to perform rapid assays in succession with automatic documentation and storage of the results. The user interface for the new C code is designed to emulate the industry standard, and user input has been reduced to a minimum. Software written in the C language is more transportable for enhanced hardware compatibility. The capability for mass storage of raw data has been greatly enhanced, and the ability to retrieve data (for reanalysis or re-examination) has been greatly simplified. In addition, mass storage of results files has been added for future review capabilities. This new software is presently undergoing testing in the laboratory. Facility testing will begin in late spring of 1990. Enhancements to this package are planned, including correcting assays for self-absorption, special analysis functions, automatically incorporating results files into commercial spread sheets, and adapting the automation to other commercial compact multichannel analyzers.

F. Computer Modeling and Simulation for NDA Development.

Mathematical modeling and simulation are powerful tools to improve our understanding of NDA techniques for measuring nuclear materials, to increase our assay ability, and to reduce the cost of designing NDA instruments. We have an ongoing program to improve and use these tools and to generate and maintain the physics data libraries necessary for realistic modeling and simulation of NDA instruments and measurements.

1. Measurement Performance Optimization for Thermal Neutron Multiplicity Counters (J. E. Stewart, R. R. Ferran, and M. S. Krick, N-1). We have performed a series of neutronics calculations using the MCNP[18] code to suggest optimum characteristics of practical designs for Thermal Neutron Multiplicity Counters (TNMCs). This new generation of instruments will be applied to verification measurements of plutonium samples for which the magnitudes of neutron emissions from (α,n), spontaneous fission (SF), and IF reactions are unknown. Results of the study may be used directly to guide design choices for specific applications and may also stimulate further optimization studies.

a. Counter Parameters. Parameters that affect measurement **precision** are:

- ε - counting efficiency for neutrons from a point source of ^{240}Pu centered in the sample cavity, and
- τ - die-away time or average time to ^{3}He capture for neutrons from a point source of ^{240}Pu centered in the sample cavity.

Counter parameters that affect measurement **accuracy** are:

- $\varepsilon(z)$ - the distribution of ε (for a point source of ^{240}Pu) as a function of position along the z-axis of the sample chamber, and
- $\varepsilon(E)$ - the distribution of ε (for a centered point source) as a function of the neutron energy.

The radial variation in efficiency was not considered to be an important influence on accuracy.

b. Design Assumptions. Ground rules for the study were:

- **Hexagonal geometry** - This geometry allowed a constant moderator thickness between ^{3}He detectors, eased problem setup, and produced an acceptable range in the size of cavity openings. (Fabrication of an actual device would not necessarily be complicated by this geometry.) Equivalent cylindrical geometries should yield nearly identical results.
- **Polyethylene moderator (high density) and 2-in. side shield** - This option was chosen from experience to lower material and fabrication costs.
- **^{3}He detectors** - One hundred and thirty two tubes were used in four rows (standard 4 atm, 28-in.-active-length Reuter-Stokes model).
- **Graphite end plugs** - Seven-in.-tall plugs were chosen from experience for better reflection properties than aluminum or polyethylene.

- **Cadmium** - Sixteen-mil cadmium sheet lined the cavity and was placed between the moderator and shield.
- **Cavity height.** A cavity height of 17.5 in. was chosen based on a survey (at TA-55) of container sizes.

c. **Varied Parameters.** Counter parameters chosen for variation were:

- **Moderator thickness.** The thickness of the moderator between the ^{3}He detectors was varied from 1/4 to 3/4 inches in 1/8-in. steps.
- **Moderator (and shield) poison.** Values of 0, 0.2, 0.5, and 1.0 wt% borated polyethylene were used for the moderator and shield (these are available from Reactor Experiments*).

Table VIII gives dimensions for the cases calculated. For brevity, we have used the term "Hexaplicity Counter" to mean the hexagonal TNMC. The cavity openings range from 7.74 in. to 10.338 in. For comparison, the cavity diameter in the HLNC-II is 6.89 in., in the AWCC it is 8.66 in., and in the new Passive/Active Counter for TA-55 it is 9.7 in.

Figure 19 shows an elevation view of the counter with 3/4 in. moderator between tube holes. Figure 20 is a cutaway view of the same design with the ^{3}He detectors exposed. Figure 21 is a plan-view section showing the configuration of tube holes and external shield.

d. **Results.** A summary of the results of the study is shown in Table IX. Columns 4 and 5 list calculated values of ε and τ, respectively, for all cases considered. Uncertainties (only from the counting statistics of the calculations) are ~0.2% on ε and ~ 0.3% on τ. An analytical point model[48] of assay variance based only on counting errors has been proposed. Generally speaking, the highest ε and lowest τ (down to ~6 µs) give the best measurement precision. The highest ε calculated was 55.8% for the case with 0.625 in. of pure polyethylene between tube channels (the cell with darkened border in column 4). The lowest τ calculated was 12.0 µs for the case with 0.25 in. of 1 wt% B-polyethylene between channels (cell with darkened border in column 5). Graphs of ε vs τ are shown in Fig. 22. Clearly, the goal of high efficiency and low die-away time is not easily achieved. Also shown in the figure are points for the high- and low-efficiency modes of the existing Al/poly/Cd five-ring counter.[49]

Column 6 of Table IX gives s_E -- the percent standard deviation for each case calculated of cavity-centered, point-source efficiencies [ε(E) values] for neutron energies of 0.5, 1.0, and 2.0 MeV. The effects of variable and unknown amounts of impurities (for example, water, fluorine, sodium, chlorine, and magnesium) will require that ε(E) variations in the 0.5-2.0 MeV range be minimized for the best measurement accuracy. The calculational precision of individual ε(E) values is 1% or better. The calculations showed that the 3/4-in. moderator spacing gives the most uniform ε(E) distribution. This is shown numerically as the 3.1% standard deviation (Table IX cell with darkened border in column 6) of ε(E) values at 0.5, 1.0, and 2.0 MeV.

Column 7 of Table IX gives s_z -- the percent standard deviation for each case calculated of point-source (neutron energy spectrum from ^{240}Pu spontaneous fission) efficiencies [ε(z) values] for 22 equally spaced positions along the axis of the sample chamber. For measurement accuracy to be minimally influenced by sample positioning, the axial efficiency profile should be uniform, or nearly so. The calculational precision of individual ε(z) values is 1% or better. Column 7 of Table IX shows that the two smallest pure-polyethylene moderator spacings give the most uniform distributions, but not much is lost by increasing the spacing to 1/2 in. or even 3/4 in. The ε(z) distributions were not calculated for the borated-polyethylene cases. In Table IX they are assumed to be the same as for the pure-polyethylene cases.

Finally, column 8 of Table IX gives a simple figure-of-merit (FOM) for each case. An overall quantitative index of measurement-performance quality was desired for each design. This FOM must include both measurement precision and accuracy, at least as far as they are influenced by the neutron physics. The model described in Ref. 48 includes only measurement precision, and the code that implements the model is not generally available. Furthermore, because we know of no rigorous measurement-error model that includes both random error components and biases for multiplicity counting, we chose a simple

Table VIII. Hexaplicity Counter Dimensions (inches)

Moderator Thickness Between Tube Holes	0.250	0.375	0.500	0.625	0.750
Cavity Opening (across flats)	7.740	8.390	9.039	9.689	10.338
Moderator Thickness (across flats)	4.763	5.304	5.846	6.387	6.929
Overall Width (across flats)	21.266	22.998	24.730	26.463	28.195
Shield Thickness (across flats)	2.0	2.0	2.0	2.0	2.0
Cavity Height	17.5	17.5	17.5	17.5	17.5
Overall Height	31.5	31.5	31.5	31.5	31.5

*Reactor Experiments, Inc., 963 Terminal Way, San Carlos, California 94070.

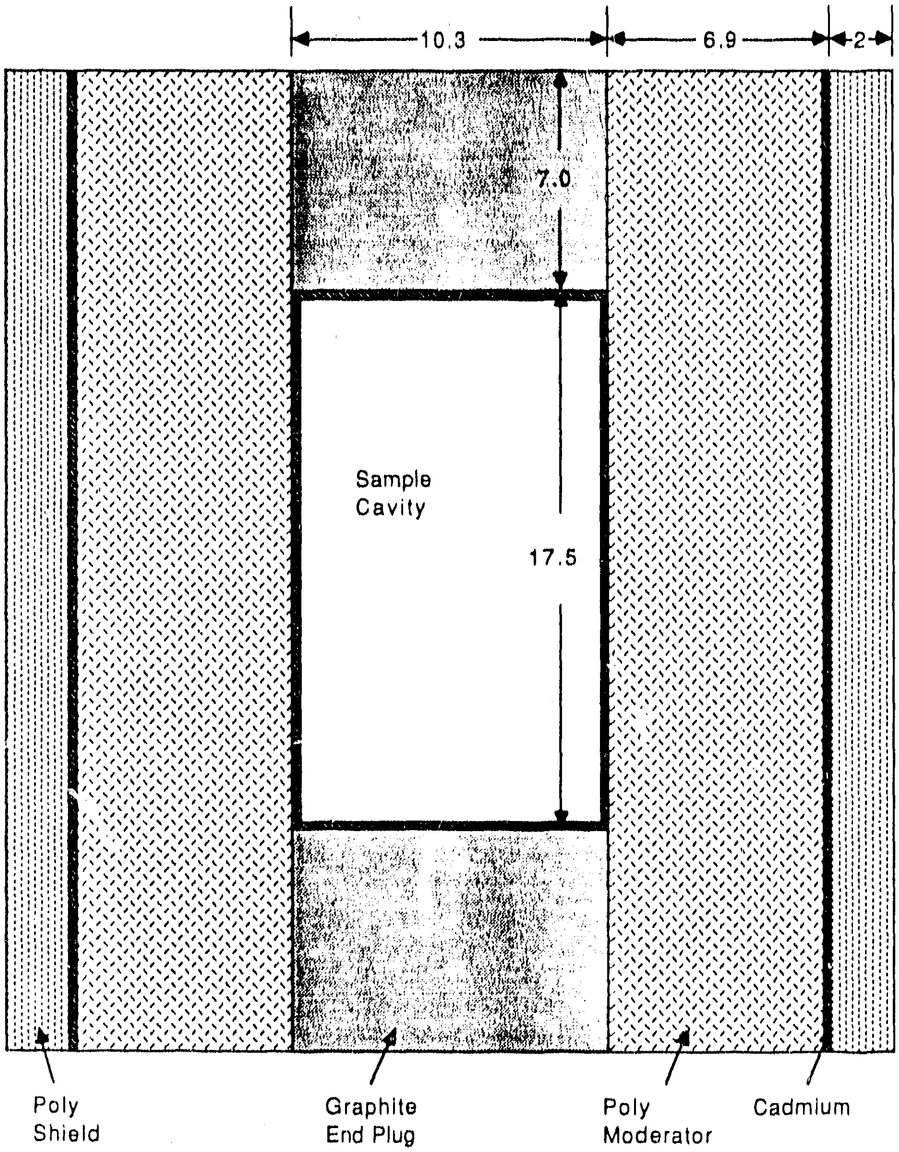


Fig. 19. Hexaplicity counter elevation view (0.75 in. poly between tubes).

FOM that incorporates qualitatively, at least, the desirable features of an optimum counter:

$$FOM = \varepsilon^2/(\tau\ s_r\ s_z).$$

Efficiency is more heavily weighted because we think it is generally more important than τ in improving assay precision.[48,49] Use of this FOM should lead to prediction of an optimum design that is not very different (in terms of measurement performance) from one that would be predicted by a more rigorous analysis.

Column 8, then, shows the pure-polyethylene moderator case with the largest amount of polyethylene between tube holes to be the best choice for this study based on the simple FOM.

e. Conclusions. The optimum design (3/4 in. pure-polyethylene spacing) has an efficiency of 54.1% and a die-away time of 47.5 μs. Table X shows the predicted measurement precisions for four specific samples measured in this counter (designated hexaplicity-optimum, or HO). Also shown, for comparison, are values for the high-efficiency mode of the existing counter described in Ref. 49 (designated Krick high-efficiency, or KH). These values were obtained from the results in Ref. 48 for similar counters. Table X shows the precision of the KH measurements to be better than that of the HO, but the KH counter was not designed for the uniformity of the $\varepsilon(E)$ and $\varepsilon(z)$ distributions. Also, from Table X it is seen that high accuracy may not be a strong consideration for

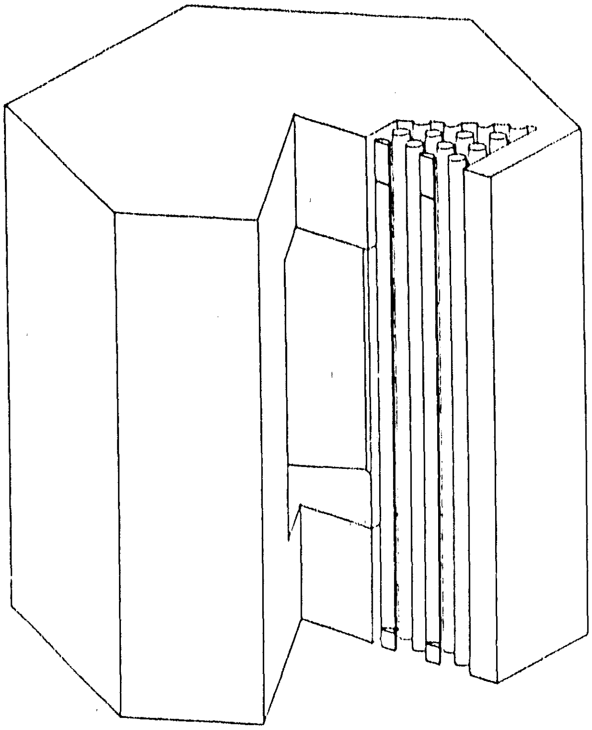


Fig. 20. Hexaplicity counter cutaway view.

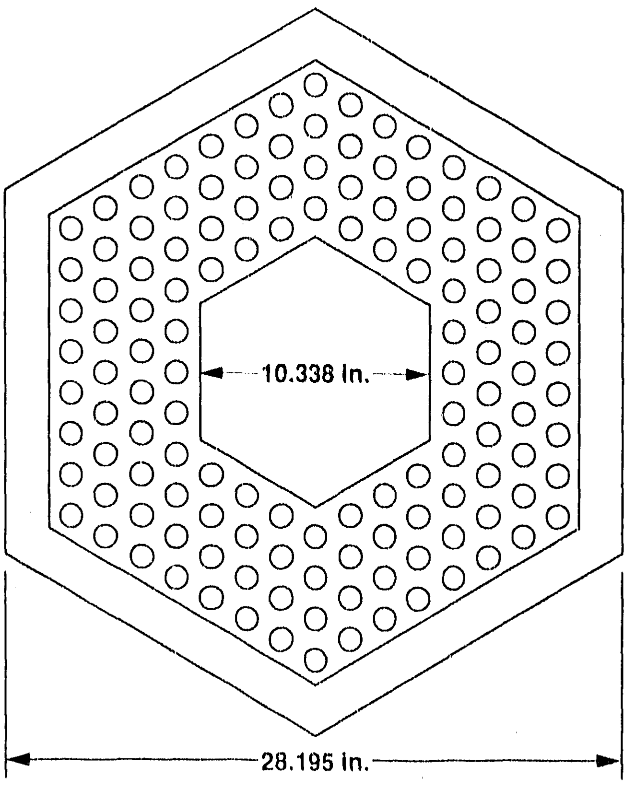


Fig. 21. Hexaplicity counter plan view (0.75 in. poly between tubes).

samples like A. For these items, 12- to 24-h count times may be required of the high-efficiency designs to obtain even 5% precision. Here the lower die-away time (and efficiency) designs will perform better. In Table X, HL is the 1.0% B-polyethylene design with 0.25-in. moderator spacing (see Table IX). The KL design is the low-efficiency mode of the existing counter described in Ref. 49. The HL design has the best precision for sample A, and the KH design has the best precision for sample B.

For samples and measurement times in which 5% precision is achievable, accuracy begins to become an important design consideration. Here, the high-efficiency designs will be superior performers. For samples like A, a fundamentally different approach, for example the combination of standard active and passive neutron coincidence counting, could yield superior measurement performance.

2. Neutron Multiplicity Counter Design for Pyrochemical Process Materials (D. G. Langner, N. Ensslin, and M. S. Krick, N-1). Materials from the pyrochemical process are difficult to measure using conventional neutron counting methods because of significant self-multiplication and variable (α,n) reaction rates. Multiplicity counters that can measure the first three moments of the neutron multiplicity distribution make it possible to determine sample mass even when multiplication and the (α,n) rate are unknown. We have used Monte Carlo simulations to design a new multiplicity counter suitable for in-plant measurement of pyrochemical process materials. Our goal was to produce a counter with a high neutron detection efficiency, low die-away time, a flat spatial efficiency profile, and insensitivity to the neutron energy spectrum.

We performed Monte Carlo calculations for several prototype models consisting of 4 rings of 71-cm active length ^{3}He detector tubes in a polyethylene body. The cadmium-lined sample well is 24.1-cm in diameter to accommodate a wide variety of in-plant sample containers. The counter can be used in a free-standing mode or an in-line mode without mechanical modification. We used the calculations to determine design criteria for several configurations of detector tube spacing, cadmium liners, and sample height. Calculations also were performed (for distributed sample sources) to understand the integrated effects of variable neutron spectra on the counter.

Figure 23 shows a vertical cross section of the counter for a 1.5875-cm tube spacing. Figure 24 shows a horizontal view of the same configuration. For this tube spacing there are 126 tubes. Monte Carlo calculations predict a total neutron counting efficiency of 56% and a die-away time of 49 μs. The axial response of this configuration is shown in Fig. 25. The energy response is shown in Fig. 26. For nine distributed samples with fill heights varying from 1 to 35 cm, plutonium contents varying from 25 to 900 g, and mean energies of the neutron source spectra varying from 1.2 to 2.7 MeV, the calculated total counting efficiency varied only 2.4%.

Table IX. Summary of MCNP Results for Hexaplicity Counter Design Optimization

Column 1	Column 2	Column 3	Column 4	Column 5	Column 6	Column 7	Column 8
Moderator and Shield	Moderator Thickness Between Tube Holes (in.)	Cavity Size (in. across flats)	ε efficiency (%)	τ die-away time (μs)	s_E eff. vs energy std. dev. (%)	s_Z eff. vs Z std. dev. (%)	FOM
Pure Poly	0.25	7.66	42.8	21.7	19.9	1.4	3.0
	0.375	8.31	51.6	27.4	14.5	1.4	4.6
	0.5	8.96	55.6	34.0	9.4	1.8	5.5
	0.625	9.61	55.8	40.4	5.7	1.8	7.4
	0.75	10.26	54.1	47.5	3.1	2.1	9.6
0.2% B Poly	0.25	7.66	34.3	18.1	22.2	1.4	2.0
	0.375	8.31	39.2	21.3	16.7	1.4	3.0
	0.5	8.96	40.0	24.4	10.9	1.8	3.4
	0.625	9.61	37.8	26.8	8.6	1.8	3.4
	0.75	10.26	34.7	29.2	5.5	2.1	3.6
0.5% B Poly	0.25	7.66	28.7	15.1	21.9	1.4	1.7
	0.375	8.31	30.6	16.5	16.8	1.4	2.3
	0.5	8.96	29.6	17.9	11.3	1.8	2.4
	0.625	9.61	26.7	18.6	8.6	1.8	2.4
	0.75	10.26	23.6	19.4	5.3	2.1	2.6
1.0% B Poly	0.25	7.66	22.8	12.0	22.3	1.4	1.4
	0.375	8.31	23.0	12.5	17.0	1.4	1.7
	0.5	8.96	21.2	12.8	12.1	1.8	1.6
	0.625	9.61	18.6	13.0	8.2	1.8	1.8
	0.75	10.26	16.0	13.2	5.7	2.1	1.6

Notes:

Columns 4 & 5 Calculated for centered point source with ^{240}Pu spontaneous-fission energy spectrum

Column 6 Standard deviation (% of mean) of centered point-source efficiencies at 0.5, 1.0, and 2.0 MeV

Column 7 Standard deviation (% of mean) of 22 axial point-source positions with ^{240}Pu spontaneous-fission spectrum: poison cases not actually calculated; assumed to be same as for pure poly

Column 8 $FOM = e^2/(t s_E s_z)$

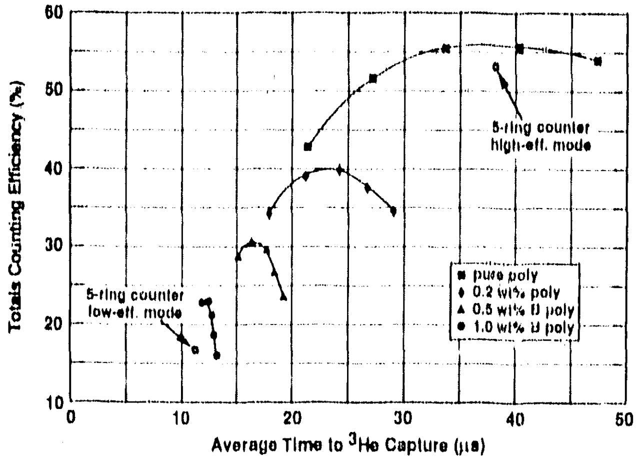


Fig. 22. Efficiency vs die-away time for hexapilcity counter designs.

The mechanical design of the counter is underway, based on the results of these calculations. A paper describing this work has been submitted for presentation at the annual INMM meeting in July 1990.

G. Chemical and Isotopic Measurement Technology.

Chemical analysis methods provide the reference base for materials accounting and product quality assurance in the nuclear industry. We have an ongoing task to develop new, highly reliable methods of chemical analysis for determining plutonium and uranium in all varieties of nuclear fuel-cycle materials with emphasis on automated chemical analyzers to improve turnaround time and reduce analyst bias and error. Developing and refining this technology for the wide range of plutonium and uranium compositions is essential to timely, accurate safeguards accounting.

1. Advanced Automated Plutonium Titration System (M. D. Randow, D. Temer, D. D. Jackson and L.E. Wangen). Accurate and timely plutonium assay is vital for nuclear materials accountability. The method used for assay of plutonium by the Analytical Chemistry Group at Los Alamos and elsewhere in the DOE complex is titration of Pu(III) to Pu(IV) with Ce(IV) titrant to a photometric end point. We have developed an automated instrument for this titration that eliminates analyst bias in locating the end point and approximately doubles sample throughput with much less stress on the analyst.

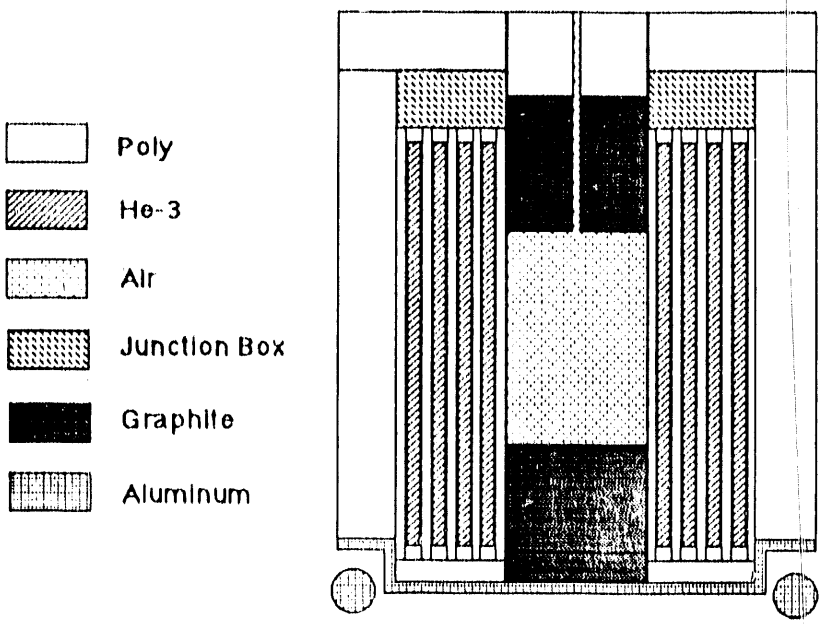
In the analysis, it is necessary to completely reduce the plutonium to Pu(III) before titration. Originally, this was done by passing the plutonium solution through a lead reductor column. Because this is not amenable to automation, we developed a method for reducing the plutonium using a Ti(III) solution. Ti(III) effectively reduces all the plutonium to Pu(III), and the excess Ti(III) is oxidized with nitric acid. Unfortunately, the oxidation of the excess Ti(III) by nitric acid generates a large quantity of gas that can interfere with the absorbance measurements used to locate the end point. We had to develop a procedure to reduce the effect of the gas bubbles on locating the end point.

An apparent degradation of precision and accuracy caused by changes in ambient temperature was confirmed by controlled experiments covering the range from 20 to 35°C, and we have developed a correction based on a linear regression equation relating measurement error for known standards to temperature. The temperature of the titrant is

Table X. Hexaplicity Counter Measurement Precision Compared with Existing Design

			Precision of a single 1000 s measurement (%)				
Sample	^{240}Pu (g)	Multiplication	Alpha*	H O	K H	H L	K L
A	200	1.16	10	45	35	22	33
B	4	1.02	10	7.6	6.8	10	16
C	200	1.16	1	2.1	1.6	1.5	2.5
D	4	1.02	1	0.5	0.5	1	1.7

*Ratio of (α,n) to spontaneous fission neutron emission.



Cavity Is 24.1 cm x 37.5 cm

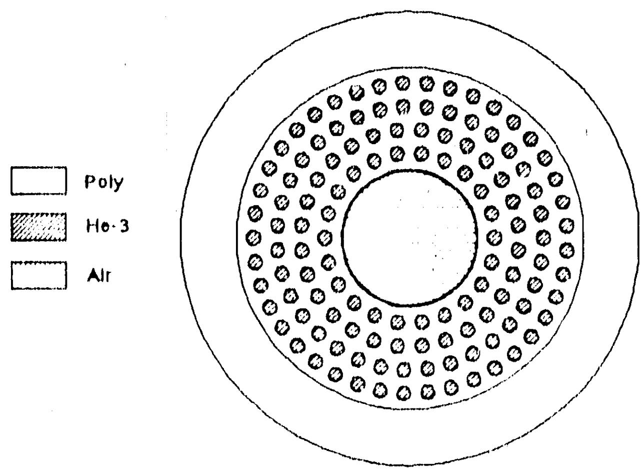
and Is Cadmium Lined

Fig. 23. Vertical Cross Section of Pyrochemical Multiplicity Counter.

measured automatically for each titration, and the temperature correction is calculated. Long term precision and accuracy of the assay was improved using the temperature correction.

The present automated titration system uses a 250-mg plutonium sample and generates considerable liquid waste (as does the original hands-on analysis). For both safeguards and reprocessing reasons, it would be desirable to reduce the sample size and quantity of liquid waste produced, so we have modified the method to use a smaller sample. Precision remains excellent (about 0.03% RSD) down to a 40-mg plutonium sample, and the amount of liquid waste is reduced by a factor of five from about 175 ml to 35 ml per sample.

The relatively constant reagent blank value becomes more significant at small sample size. The reagent blank value is primarily due to the amount of ferroin indicator in the solution. By changing to a more dilute ferroin indicator solution, we were able to increase the volume dispensed to 1 mL instead of the usual 200 μL. The larger volume is much easier to deliver accurately, improving our blank precision. We are using Hamilton automatic dispensers to deliver all of the reagents directly into the sample cell. Tests of these dispensers gave excellent precision of 0.02% RSD, which is a significant improvement over that obtained by manual addition using a micropipette. We developed the necessary control software and interfaced the two automatic dispensers into the system



Tube Spacing Is 1.5875 cm

126 Tubes

Fig. 24. Horizontal Cross Section of Pyrochemical Multiplicity Counter.

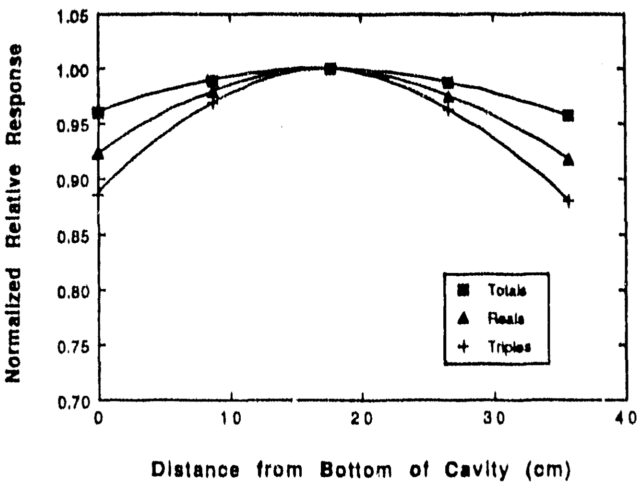


Fig. 25. Calculated Axial Response for Pyrochemical Multiplicity Counter for ^{252}Cf Point Source.

and control. Because our original HP-85 controller did not have enough interface ports to control and measure all necessary items, we added an HP-IB Data Acquisition/ Control System, which does provide the required input/ output capability. We also added an interface and software to provide control of the electric stirrer by the computer. A pneumatic cylinder mechanism removes the titrant delivery tip from the sample solution during stirring to prevent diffusion of the ceric solution into the sample.

Design and construction of the Advanced Automated Plutonium Titration system is now complete, and evaluation tests are underway. Weight aliquots of an iron solution containing the equivalent of 40 mg of plutonium have been processed in the automated mode. Over a two week period, during which 25 aliquots of the iron solution were processed, the precision was 0.028% RSD for a single determination, which compares favorably with the large-sample instrument currently in use. We are installing the new instrument in a glove box for evaluation with plutonium.

2. High Precision Spectrophotometric Method for Plutonium Assay (H. L. Nekimken, P. Mendoza, E. Lujan, D. Temer, D. D. Jackson and L. E. Wangen). Spectrophotometric methods of assay often have the advantages of being relatively fast and requiring few operations. They also are often particularly suitable for automation because of the types of operations that must be performed.

We initially—and unsuccessfully—investigated a method based on the tetrapropylammonium plutonyl trinitrate ion-pair extraction system using uranium as a stand-in for plutonium. In this method the trinitrate complex is extracted from an aluminum nitrate salting solution into an organic phase in which the absorption of the complex is measured. The method is highly specific for plutonium and uranium and is capable of good precision. For uranium, we achieved a precision of 0.1% RSD consistently for a single determination based on 10 measurements at the 10-mg uranium level.

Applying this extraction method to plutonium assay requires oxidation of the plutonium to Pu(VI) before it is extracted into the organic phase. Silver(II) oxide effectively oxidizes plutonium to Pu(VI) and was selected as the oxidant. Because the silver oxide must be added as a solid, it reacts vigorously with the plutonium solution generating gas that can cause sample loss. We developed techniques for manually adding the silver oxide and performing the oxidation without loss of sample. However, results with plutonium were erratic with precision ranging from 0.1 to 0.3% RSD. Automation of the oxidation step is difficult because of the vigorous reaction of the solid silver oxide with the solution. We could find no suitable solution to oxidize the plutonium.

We now are developing a continuous flow analysis (CFA) automated spectrophotometric method for determining plutonium based on the Pu(III) chloride spectrum. A peristaltic pump is used to mix an ascorbic acid solution in hydrochloric acid with the plutonium sample solution to dilute and reduce the plutonium to Pu(III) prior to spectrophotometric measurement. A computer-controlled flowing system composed of a peristaltic pump, Teflon mixing coil and tubing, and a flow-through spectrophotometer cell performs the necessary operations. The absorbance spectrum is measured with an HP photodiode array spectrophotometer. Our initial experiments indicated that a precision better than 1% should be attainable.

Initial precision studies using praseodymium as a stand-in for plutonium gave poorer results than expected. This was attributed to variations in flow rates through peristaltic pump tubing, and memory effects in the tubing and a three-way valve used in the instrumental setup. Flow studies using different sizes and types of peristaltic

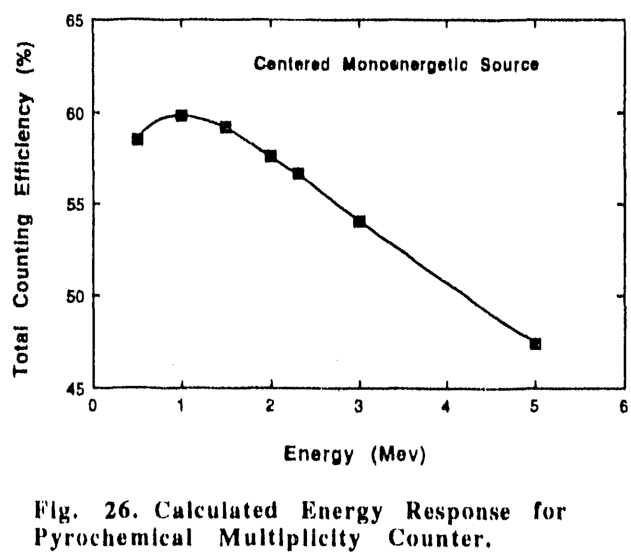


Fig. 26. Calculated Energy Response for Pyrochemical Multiplicity Counter.

pump tubing verified the presence of within-day and between-day variations, and the drift was different for different sizes and types of tubing. Tygon tubing exhibited pump tubing verified the presence of within-day and between-day variations, and the drift was different for different sizes and types of tubing. Tygon tubing exhibited the least variation and was selected for use in the automated system. Additional improvements included reducing the total amount of Tygon tubing, eliminating the three-way valve and one of two mixing coils, and changing from a 10-mm to a 2-mm path-length flow cell. The latter change decreased the dilution factor required to obtain absorbances in the linear range from 25 to 5, thereby eliminating the need for the largest dilution tube and reducing the drift problem. The changes reduced memory effects, analysis time, and amount of liquid waste generated.

We used the system to evaluate the effectiveness of an internal standard in the diluting solution for calculating actual sample dilution to compensate for flow rate drift. We used neodymium as the internal standard. Average daily precision was reduced from 1% without the internal standard to 0.4% for the praseodymium system using the internal standard. These results are based on calibrating once each day before the analyses and performing an analysis about once an hour. Using this system, sample analysis time is about 7 minutes, and only 9 μL of waste is produced.

We evaluated four lanthanides and two organic dyes as potential internal standards for Pu(III). Least squares curve fitting techniques were employed to quantify each internal standard and the Pu(III) in test solutions. Praseodymium was selected for further evaluation as an internal reference for plutonium analyses. Only one Pr(III) absorption peak interferes significantly with any of the Pu(III) peaks, and this can be corrected using the curve fitting procedure.

The CFA system was set up in a glove box for testing with plutonium using praseodymium as an internal standard. Replicate determination of a plutonium standard solution gave precisions of less than 0.5% RSD on four different days of operation both with and without the internal standard. The system has been evaluated further for plutonium determinations involving praseodymium using both dissolved plutonium oxides and feed and product samples from the Los Alamos SNM chloride process line. Results using the automated system were compared with those of a manual Pu(III) spectrophotometric system used routinely in the Analytical Chemistry Group for plutonium assay of plutonium oxide materials. For three plutonium oxide materials the differences were all within 0.1 mg/L at the 18 mg/L level. The agreement for process product samples also was satisfactory. However, the results for the process feed samples showed greater differences. This may be caused by particulate material and other interferences in feeds containing high levels of dissolved salts and plutonium. This is being investigated.

II. Reference Material Characterization (T. K. Marshall, T. R. Hahn, A. P. Lovell, J. W. Dahlby, and D. D. Jackson, CLS-1).

Physical standards (reference materials) must be available to evaluate and calibrate both NDA and chemical analysis methods, and a variety of standards is needed to meet the requirement of DOE Order 5633.3 that all measurement methods be calibrated using certified reference materials. We have an ongoing task to characterize and package suitable certified reference materials as requested by the New Brunswick Laboratory (NBL).

We currently are evaluating alternative primary packing containers for NBL CRM 126 Plutonium Metal Standard, which is used as an assay standard. The containers must protect the metal from air oxidation and also be of a configuration amenable to secondary packing containers; the present primary containers are too long. We sealed small pieces of clean, shiny plutonium metal into a variety of small containers in an inert atmosphere. Any oxidation of the plutonium metal is easily noted by change in the appearance of the metal. Most of the packaging systems have been unsatisfactory as shown by oxidation of the metal after a few months. However, a shortened, modified version of the original system using evacuated, heat-sealed glass tubing shows no evidence of reaction of the metal after many months. The evaluation is continuing.

I. Measurement Control and Calibration.

We have an ongoing task to develop physical standards and proven methods of measurement control and calibration to give confidence that NDA measurements are within the accuracy and precision limits necessary for materials accounting. We also participate in the preparation of consensus standards as part of our technology transfer activities.

1. NDA Calibration Standards. The accuracy of NDA measurements depends on the attenuation, multiplication, and absorption effects of the contained SNM. An acceptably small number of physical (calibration) standards can be obtained by considering carefully the range of materials to be measured with specific instruments and by providing correction factors for known effect, based upon an understanding of the measurement physics and laboratory measurements.

a. Design and Fabrication of Plutonium SGS Can Standards (S.-T. Hsue, S. M. Simmonds, N-1; V. L. Longmire, NMT-4; S. M. Long, NMT-4). We are developing the plutonium can standards needed to calibrate SGS's to assay low-density scrap and waste. Each of the standards has to satisfy the measurement physics of the SGS assay principle:

- The SNM should be uniformly distributed.
- The diameter of the standard should be small so that the gamma-ray transmission through the standard is reasonable ($T > 0.2$).
- The height of the standard containing SNM should be at least ten times the height of the collimator used in the SGS measurement to minimize the end effect. Because the typical collimator height of an appropriate SGS is 1.27 cm, the height of the SGS standard should be at least 13 cm.
- The particle size of the SNM should be small so that the self-absorption in the particles is negligible down to 203 keV. These standards then can be used for the recently developed lump-corrected SGS technique.[29]

In addition, the standards must meet DOE shipping requirements so that they can be shipped to other sites throughout the DOE complex. This means the standards must be doubly contained and the container must be welded or canned.

Can Selection: We tried to locate commercially available cans satisfying the above requirements. The ideal inner can would be approximately 10 cm in diameter and 25 cm high, with the outer can slightly larger. These cans should be sealable either by canning or welding. After an extended search, we concluded that appropriate cans would have to be custom designed and fabricated. Our design group designed the cans, and the Los Alamos main shop fabricated them.

After the cans were fabricated, we tried to weld the inner can in a glove box. It was difficult to weld the lid onto the can; it tended to pop out. We modified the lid and the welding equipment, and, after several practice runs, successfully welded the lid onto the can.

Matrix Selection: Low-Z matrix materials must be blended with the SNM in the standard. Both graphite and diatomaceous earth have been used to make SGS standards. However, graphite is flammable and requires a special work permit before it is allowed in a glove box; diatomaceous earth is used routinely in glove boxes as a filtering agent. Furthermore, the density of diatomaceous earth is lower ($d = 0.26$ g/cm^3) than that of graphite ($d \sim 1.25$ g/cm^3). The lower density means that the same matrix can be used for drum standards, which we need to develop next.

Standard Preparation: Each standard was prepared by putting the plutonium oxide in the inner can, adding the diatomaceous earth, and welding on the lid. After sealing it in the outer container, we mixed the standard on a V-blender. In the first standard that we prepared, we filled 80% of the inner can with diatomaceous earth; the mixing was not complete after blending for longer than 8 hours, and further blending was required. In the rest of the standards, we reduced the diatomaceous earth to only 60%, which reduced the blending time substantially.

Characterization: After the standards were prepared, they were characterized by NDA. We found:

- The standards are reasonably uniform up to 13 cm.
- There is no evidence of plutonium and diatomaceous earth agglomerating into clumps.

Four sets of standards are being prepared with plutonium loading that ranges from 10 g to 250 g in each set. One of the sets is being prepared for the Westinghouse Savannah River Site (WSRS). (Facility funding was provided for this set of standards.) The actual mass of plutonium in each standard is determined by chemical analysis and confirmed by NDA.

b. Monte Carlo Simulation of 55-Gal Drum Standards for the SGS (M. C. Miller and S.-T. Hsue, N-1). To assay scrap and waste in drums with the SGS technique, it is important that the SNM in the calibration standards be uniformly distributed in a low-Z matrix. Because of the size of a 55-gal drum, it is difficult, if not impossible, to mix the SNM directly in the drum to achieve a uniform distribution. A more practical approach is to mix the SNM with the low Z-matrix in 4-L bottles and distribute these in the drum so that they represent a uniform distribution. However, there are questions as to how to position the bottles in the drum, and whether the void between the bottles should be left empty or filled with the matrix material. Because it is expensive and time consuming to prepare standards, we used the Los Alamos Monte Carlo transport code MCNP[18] to study the best configuration for the standards.

To compare various methods of positioning bottles containing SNM to the case of SNM homogeneously distributed in the drum, we simulated a geometry that included the drum, the SNM, and SiO_2 (diatomaceous earth) matrix material (Fig. 27). For the uniform base case, we dispersed the SNM throughout the drum and removed the stacks of 4-L bottles. Estimates of the count rate in a detector viewing a rotating drum at a distance equal to one half the drum diameter from the surface of the drum were made by the Monte Carlo technique. We also estimated the transmission of gamma rays of the same energy through the sample at the same distance; this enabled us to calculate a correction factor for self-attenuation and the total corrected count rate.

Table XI gives the results of the calculations. We considered three cases: (a) SNM distributed uniformly in the drum, (b) SNM contained in seven columns (polyethylene bottles) as shown in Fig. 27 with voids between the columns, and (c) SNM contained in seven columns with diatomaceous earth filling the voids between the columns. In all cases, the drum contained 105 g of plutonium and the gamma-ray energy was 414 keV. In case (b) the total corrected count rate was 4% higher than the uniformly distributed case (a). This was caused primarily by an increase in the normalized count rate of 25% relative to the uniform case. As can be seen from Table XI, the situation most closely resembled

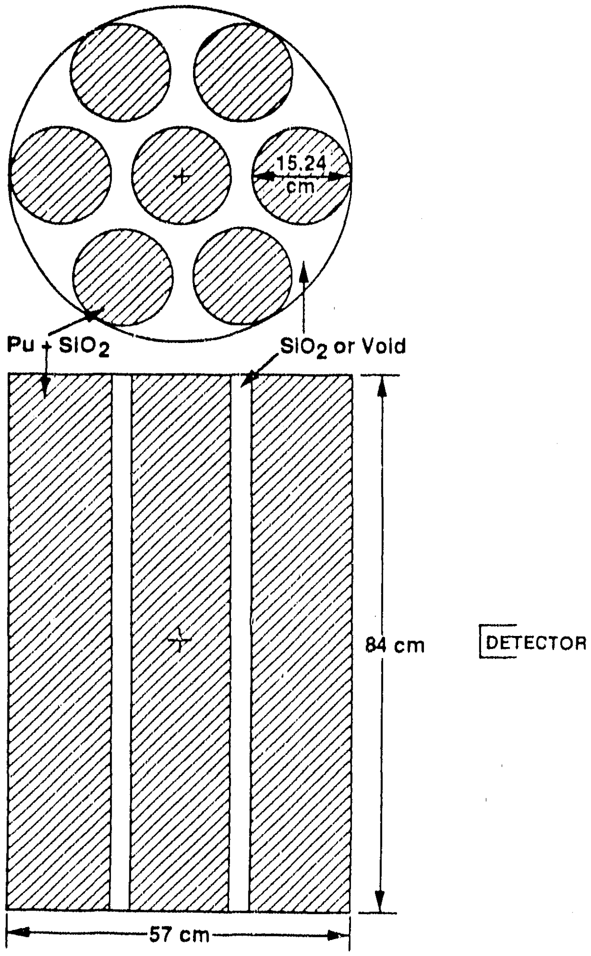


Fig. 27. Plot of MCNP geometry used in the simulation of 55-gal. drum standards for the SGS.

the uniform case when the gaps between the columns of bottles containing the SNM were filled with the matrix material; the total corrected count rate was within 1% of the uniform drum value.

We repeated the calculation with another drum containing 210 g of plutonium and obtained similar results. We concluded that the drum SGS standards could be prepared using 4-L bottles with the gaps between bottles filled with matrix material.

c. Examination of Los Alamos Measurement Control Practices (T. E. Sampson, N-1). Los Alamos (operational safeguards and plutonium processing) is examining its application of measurement control, NDA instrument certification, and measurement method error statements to ensure compliance with DOE regulations, while applying technically sound practices that do not unnecessarily hold up the certification of new NDA instruments or place unnecessary measurement control burdens (with resultant productivity loss and increased radiation exposure to workers) on the NDA measurement staff.

We are participating in all phases of the discussions on these issues.

2. Consensus Standards. A writing committee for consensus standards is made up of representatives from the user community, commercial manufacturers, and research and development (R&D) laboratories; a standard can be written only after all members of the writing group understand the technology. As MC&A technology matures and moves from R&D for individual applications into well-established methods used at many facilities, it joins the ranks of traditional assay techniques documented by the ASTM. Because of the broad range of the members' interests and backgrounds, an ASTM consensus standard is approved only after widespread understanding and agreement on the technique is achieved.

a. NDA Consensus Standards (J. K. Sprinkle, Jr., N-1). After NDA instruments are fielded, evaluated, and in routine use, the next step is to document the possible applications and performance of the instruments by means of the consensus process. We participate in ASTM subcommittee C26.10's preparation of NDA standards.

An SGS standard (designated C1133-89) was approved and published in 1989, and an NCC standard for

Table XI. SGS Drum Standard Calculational Study Results

Case	Normalized Count Rate	Transmission	Correction Factor[a]	Total Corrected-Count Rate (Normalized)
a. Uniform	1.00	0.1975	1.769	1.769 (1.00)
b. Columns + Void	1.25	0.3448	1.476	1.845 (1.04)
c. Columns + SiO_2	1.01	0.1973	1.769	1.787 (1.01)

[a]Correction factor defined as $$CF = \frac{-\ln T^{\pi/4}}{(1 - T^{\pi/4})}$$

plutonium bearing scrap and waste is nearing final form, with the technical content agreed upon. We provided a technical review of additional standards based on gamma-ray assay techniques and made suggestions about the scope, significance, and use for a holdup standard. All of these standards address techniques developed in the Los Alamos Safeguards and Security R&D program.

b. Portal Monitor Standards (P. E. Fehlau, N-2). We also participate in ASTM subcommittee C26.12's efforts to develop performance standards for perimeter safeguards devices. The subcommittee's guide to SNM monitors, designated C1112, was published in 1989, and development continued on guides to laboratory evaluation, in-plant evaluation of pedestrian SNM monitors, and pedestrian monitor calibration. The subcommittee also held a workshop on metal detectors and began drafting metal-detector standards. One of these will be a guide to installing metal detectors and another a guide to their in-plant testing.

J. Automated MC&A Technology.

The goal of this project is to provide the DOE with the technology for collecting and automating an accurate database and managing the information.

1. A Relational Database System for MC&A (J. B. Marchi and W. J. Whitty, N-4; R. C. Bearse, University of Kansas). We have developed a relational database application for MC&A, using a popular relational database management system (RDBMS), from the hierarchical file handling system (ctree) used in the Argonne West/Unified Safeguards project (ARGUS) project.[50] We developed this relational database to determine its portability using economical PC hardware and software, and to determine if the system could respond quickly and have the potential for enhancement. This application depends on NDA measurements and data entry to maintain an up-to-the-minute computerized database of all the SNM in a facility.

We used the ORACLE RDBMS with the structured query language (SQL) interface and the Microsoft C language on an IBM/AT. The SQL and C languages were chosen for their portability and compatibility. We determined that this RDBMS and application, using standard PC hardware and software, is efficient, cost-effective, portable, has the potential for growth, can perform well with careful data modeling, and is the best generalized database management system model for a distributed environment. In addition, ORACLE runs the same software on all platforms, which include all the machines where MC&A applications occur.

2. Network Design System (G. L. Barlich, N-4). As more MC&A functions are computerized, the size of the associated databases also will increase and perhaps even span several machines connected in a network. The WSRS has several such systems in the development phase, including systems at the New Special Recovery (NSR) facility. Database designers for such applications must carefully plan table structures and allocation assignments to minimize retrieval times while restricting the need to maintain duplicate tables for every task. This problem lies in the computational complexity class of nondeterministic polynomially hard (NP-hard) problems, meaning that an optimal solution will become exponentially more difficult for larger and larger problem sets. Even the smaller sets may be beyond the capability of a human working for many days. To meet the difficulty of designing these large systems, a computerized analysis program called the Network Design System (NDS)[51] has been developed cooperatively between Los Alamos and the University of New Mexico (UNM). The NDS uses computationally efficient heuristics to analyze proposed allocation schemes and to construct an optimal scheme from limited data. It is oriented toward common functions in systems used for safeguarding nuclear material.

To test the NDS, we used it to analyze the allocation of data fragments and transactions in the NSR network at WSRS. This network consists of a process control computer with slaved microprocessors, an accountability computer, a laboratory control computer with slaved instrumentation controlled by microprocessors, and a vault control computer. The system topology was characterized using 4 large network nodes, 16 terminals, and 7 instrument nodes. Approximately 250 data fragments were defined to represent the global view of variables required to run the process with 74 different transactions possible. Frequencies and combinations of transactions were defined. Although relatively few fragments had to be restricted to certain nodes, it was assumed that process and accountability information should not be available to the nodes used for laboratory and vault control, for security and integrity reasons. Edge costs were designed to reflect approximate distances required for network and terminal connections. After defining the system for the NDS, we ran the option to analyze an existing topology.

As expected, allocations to the laboratory and vault computers were found to be very close to optimal. These functions run fairly independently of the other nodes and require only infrequent network access to obtain information, such as container reference values, and to update process and accountability values. In practice, both of these machines run efficiently and seem able to handle their loads well. However, the NDS indicated a number of inefficiencies in the allocation of data to process and accountability systems. Many of the fragments required for process control are equally important for accountability. Maintaining separate but identical copies of these fragments required significant overhead on both machines, yet keeping the data on only one machine and requiring the other to retrieve needed values was equally expensive. In practice, the accountability and process control computers are very burdened and seem to spend a great deal of effort communicating with each other to maintain the defined functions.

Our analysis showed that process control and accountability should be treated as one function even though implementation might still require multiple nodes with a more optimal allocation of fragments. The NDS indicated that one efficient configuration would be to use a large central computer for maintaining process data, rough trial balances, and accountability balances made with variance propagation. Smaller secondary computers could collect process data and operator interactions for transfers and generate accountability and process reports.

We plan to enhance the NDS to better model arrival and processing times with different probability distributions. Some events, like the arrival of shipping containers at the vault, are very batch-oriented while other events, like monitoring process vessels for holdup, occur on a very strict timetable. The reliability of certain operations is also involved. The containers in a shipment must be assayed quickly and surely to obtain shipper/receiver records. Monitoring plutonium values in the laboratory catch tank may have lower priority because rough calculations based on assayed values of dumped samples are available. Queuing analysis also will be added to model the problems associated with very busy nodes such as the proposed central process control/accountability node.

K. Data Analysis and System Evaluation Technology.

We have an ongoing project to develop and adapt statistical, mathematical, and decision analysis tools to support improved MC&A at DOE facilities. The results of the project are for use by DOE field offices and contractors in evaluating the effectiveness of existing systems and proposed upgrades, and in the analysis of MC&A data to detect and resolve anomalies.

1. Evaluation and Analysis for Small Computers (W. D. Stanbro, N-4). The proliferation of small computers has opened the possibility of producing evaluation and analysis tools for them to aid the safeguards and security community. We have developed a computer code called THIEF that is an interactive simulation of the insider threat at nuclear facilities. THIEF runs on IBM compatible computers with 640K of RAM and a VGA or EGA graphics card.

In THIEF, the user takes the role of the insider seeking to steal SNM. The computer runs the safeguards and security system, which consists of MC&A and physical protection elements. Facilities are described in terms of their physical layout and process details. The insider is characterized by his access to information and areas of the facility. He also is characterized by the times he is authorized to be in the facility. All scenario elements can be modified by the user to represent specific situations. The computer keeps a log of all of the insider's actions and the movement of SNM. This log is in a Lotus 1-2-3 compatible format so that it is available for further analysis.

To make the simulation available to the widest range of users, we implemented a graphical interface to make it easier to enter scenario elements and interactions with the simulation. The simulation is useful both for training and for systems evaluation and analysis.

2. Computer Code for Selecting Optimum Safeguards Upgrades (A. Zardecki, J. T. Markin, N-4; and S. P. Pederson, A-1). Designing a materials control and accounting system to safeguard SNM against the insider threat is a complex process requiring tradeoffs among many candidate system components. As an aid to the analyst in designing new safeguards systems or upgrading existing ones, we are developing PC-compatible design tools that use mathematical optimization methods to select the most effective combination of system elements.

Previous work in developing the computer program RAOPS[52] resulted in a dynamic programming algorithm that optimizes the safeguards design for a system protecting a single SNM location, with a constraint on the total safeguards budget. Modeling the system as a series of concentric boundaries with the SNM at the center and considering the design problem as the selection of safeguards elements at each boundary to optimize a measure of protection, the mathematical statement of the optimization problem is as follows.

At each stage (boundary) k calculate

$$I(x,k) = \min_{u \in U} \{L(x,u,k) \cdot I[g(x,u,k), k+1]\} ,$$

where $x(k)$ is the total resource spent on safeguards elements through stage k, $u(k)$ is the resource spent at stage k, and U is the total resource budget. The total resource spent through stage $k + 1$ is $x(k + 1) = g[x(k), u(k), k]$, the measure of protection at stage k is $L[x(k), u(k), k]$, and the optimized measure of protection through stage k is $I[x(k), k]$.

The multiple location design problem is modeled as a collection of points, each representing an SNM location, which are contained in a hierarchy of concentric boundaries. There are options for selecting safeguards elements at each boundary, and the assumed insider adversary has alternatives for crossing the boundaries to accomplish a theft or diversion action. The safeguards designer chooses that combination of safeguards elements that simultaneously optimizes a measure of protection for the SNM at each location, considering all of the actions available to the adversary and a limited safeguards budget.

We have completed a nonserial dynamic programming and software algorithm for selecting safeguards elements that protect against an insider adversary gaining unauthorized access to SNM. The extension to include the selection of elements to protect against removal of SNM is underway. When completed, the algorithm and associated software will allow system designers to address fully the problem of selecting safeguards elements for protecting multiple SNM locations.

II. COMPUTER SECURITY

The DOE Center for Computer Security (CCS) at the Los Alamos National Laboratory is responsible for developing, collecting, organizing, and disseminating computer security information to the DOE and DOE contractors. Fulfilling this responsibility involves operations and field support, computer security education and awareness, and R&D in several areas of computer security.

A. Computer Security Education.

We have an ongoing project to develop, deliver, and maintain a comprehensive curriculum designed to meet the computer security education needs of DOE and to deliver the education through appropriate means such as video tapes or live classes.

1. Train-the-Trainer Classes (H. C. Rosenblum, N-4). The DOE CCS continued to provide Computer Security System Officer (CSSO) training during 1989. Two CSSO training seminars, in the train-the-trainer format, were presented at the DOE Central Training Academy (DOE/CTA) in Albuquerque, New Mexico.

The modules that compose the CSSO training seminar are:

DAY 1 - WHAT IS DOE COMPUTER SECURITY ALL ABOUT?

Introduction to Computer Security Training presents the five modules to be discussed in the seminar:

1. What is DOE computer security all about?
2. Site policies and procedures.
3. How to write an automated data processing (ADP) Security Plan.
4. How to test your ADP Security Plan.
5. How to write, implement and test a contingency plan.

Threats, Vulnerabilities, and Penetration Techniques describes the threats, vulnerabilities, and penetration techniques the computer security professional is likely to encounter on the job. During this session the participants learn the requirements of the DOE and site Statements of Threat. They write the site Statement of Threat for a fictitious DOE computer system processing classified information. (We use a fictitious system to eliminate classification problems.)

DOE Orders discusses the DOE Orders applicable to computer security, with special emphasis on DOE Order 5637.1. A tree structure is used to identify the DOE Orders that are applicable to computer security.

Risk Management discusses the DOE risk management program and describes the risk assessment requirements in DOE Order 5637.1. Various methods of risk assessment are described, and the participants complete a risk assessment of the fictitious computer system.

DAY 2 - SITE POLICIES AND PROCEDURES

The participants are taught how to present the policies and procedures written for their own sites. The following five major elements of DOE computer security are discussed:

Personnel Security discusses the personnel security requirements of DOE Orders 5637.1 and 5631.2A:--Personnel Security Program.

Physical Security discusses the protection requirements for single and multi-level classified ADP systems, including protection of storage media, electronic protection requirements, and visual access requirements.

Administrative Security discusses the administrative controls required of a classified ADP system. The following elements require controls: user validation and authentication; passwords; database management system access; accountability and audit trails; classified data marking; user training and guidelines; hardware clearing and sanitizing; destruction procedures; remote diagnostic services; handling of protect as restricted data (PARD) information; protection index determination; assurance testing; acquisition specifications; protection of ADP systems from waste, fraud, and abuse; and backup procedures.

Telecommunications Security discusses the telecommunications and emission security requirements, as outlined in the DOE Orders, with special emphasis on the TEMPEST requirements for classified ADP systems.

Hardware and Software Security discusses the protection requirements, based upon the protection index and the features and assurances associated with each protection index.

Management Procedures discusses the additional procedures required of a CSSO—those procedures and duties not covered by other modules. These procedures include:

- Computer Security Incident Reporting—a discussion of the procedures for recording, reporting, investigating, documenting, and responding to computer security incidents.

- Computer Security Planning—a discussion of the long and short range planning requirements of DOE Order 5637.1 and the DOE/Office of Safeguards and Security (DOE/OSS).
- Computer Security Program Evaluations—a discussion of the evaluations required to ensure that the Classified Computer Security Program management process continues to meet the requirements of the policies and procedures of the DOE.
- Configuration Management—a discussion of the requirements for hardware and software inventories at all classified ADP facilities and a description of the tool developed by the DOE CCS.

Each participant writes a section of the policies and procedures for the fictitious company.

DAY 3 - HOW TO WRITE AN ADP SECURITY PLAN

The ADP Security Plan module begins with an outline of the DOE requirements for the ADP security plan for classified systems and classified networks. The participants are given a security plan template that describes the requirements of each section of the security plan. Then they write an ADP security plan for the fictitious system.

DAY 4 - HOW TO TEST YOUR ADP SECURITY PLAN.

This module describes ADP systems security tests and certification of classified ADP systems and networks. The participants evaluate the ADP security plan written above and describe the steps taken to test the plan and to verify that the ADP system has been implemented as described in the plan. During this session, the DOE accreditation and reaccreditation procedures are described.

DAY 5 - HOW TO WRITE, IMPLEMENT, AND TEST A CONTINGENCY PLAN.

The **Contingency Planning** module describes various methods of contingency planning, testing, and recovering from disaster. The audience participates in writing a contingency plan for the system described above.

CSSO Duties and Responsibilities recaps and discusses the duties of the CSSO.

The Center also conducted a management seminar at Sandia National Laboratories, Albuquerque. Over 30 first- and second-level line managers received computer security training at the seminar.

2. Escort Training Procedures (H. C. Rosenblum, N-4). The Center distributed a training videotape, "The Outsider—Training for Escorting Uncleared Personnel into a Classified DOE ADP Facility," to all Computer Security Office Managers and Computer Security System Managers (CSSMs).

The Center also is creating two new training aids to be distributed during 1990 — an on-line computer security training program designed to identify system vulnerabilities and a video presentation describing threats to classified ADP systems.

B. Computer Security Research and Development.

The objective of this project is to identify present and future DOE computer security needs and to conduct R&D activities in these areas to provide practical, cost-effective solutions.

1. Computer Security Review Package (W. J. Hunteman, N-4). The computer security review assistant tools are designed to help a human reviewer conduct the various reviews required in DOE Order 5637.1. Prototype tools for reviewing the CSSO and the CSSM positions have been developed. Another prototype tool, Security Plan Assistant, has been developed to aid the developer or the reviewer of the ADP Security Plan required by DOE 5637.1 for each computer system. The tools use expert system techniques as an experiment to evaluate the use of these techniques in computer security. The tools are not intended to be a replacement for human reviews; rather they are an aid to ensure that the reviewer covers all of the details involved in a review.

A primary benefit of each tool is the automated collection of the reviewer's responses and comments that relieves the reviewer of making notes for a written report. Upon completion of a review, a hard-copy report can be printed from the information and collected comments.

Each tool supports a variety of approaches to the review requirements:

Formal Review. Each tool can be used during a formal review, with a portable computer if appropriate. The reviewer can observe the particular area or topic and then enter the appropriate responses and comments. Once the review is completed, the report can be printed.

Formal Review with Spot Check. The tools can be a component in a comprehensive review by asking an individual (for example, the CSSO or CSSM) to respond to the questions posed by the tool and then return the results to the reviewing organization. The reviewing organization can then spot-check the results to validate the information. This approach can produce a comprehensive review with little increase in personnel resources.

Self Review. The tools can be used as a self-evaluation to assist in periodic checks of the computer security program and activities.

Preparation for an Inspection and Evaluation (I&E) Review. The tools can be used to prepare for an I&E review by conducting a self-review and then correcting problems that the tools identify.

2. Computer Security Advisor (W. J. Hunteman, N-4). The CSSO is required to integrate the user's needs for improved productivity with security policies that are, at best, complex, difficult to interpret into a local environment, and often apparently in conflict with each other. Today's CSSO is faced with a large quantity of written and unwritten policies, large software systems, and complex hardware. The CSSO is required to interpret the information into a set of safeguards for a particular computer system with the goal of "securing" the system against some typically poorly specified threats.

The lack of relevant education and guidance for the CSSO frequently requires the individual to resort to reliance on tradition or folklore to "secure" a system. Many CSSOs are assigned previously "secured" systems and are expected to continue the status as the system evolves, with little explanation or understanding of the original approach to securing the system.

CSSOs who have access to computer security knowledge through their own experience or through experts are able to implement and maintain better, more comprehensive, computer security programs. The principle characteristic seems to be an ability to integrate the different components of computer security (i.e., physical, personnel, communications, administrative, hardware, and software) into a balanced program that establishes a secure environment for the computing activities. As the computer security field matures and these experienced CSSOs retire or move to other jobs, some mechanism must be developed to make the expertise readily available to the new CSSO.

Our development of a knowledge-based Advisor system with methodologies to incorporate uncertain or incomplete information and manage the large quantities of knowledge will provide the necessary information to any CSSO on demand.

The Advisor is designed to provide an integrated collection of policy requirements and expert knowledge. The design goals for the system are to:

- produce a knowledge-based system to advise the security officer,
- define a comprehensive list of possible attack scenarios,
- define a comprehensive list of safeguards and associate the safeguards with policy requirements,
- collect a detailed description of the local computing environment,
- produce a list of safeguards that are applicable to the local environment with guidance on the required implementation approach for each safeguard,
- manage the use of uncertain or incomplete information in the knowledge bases,
- support "what-if" experimentation to adjust the local computing environment description or reject proposed safeguards because of resource limitations,
- provide, on request, justification or explanation of each decision throughout the process,
- provide a user interface oriented to the CSSOs needs and environment, and
- generate the security and test plans necessary to support system certification and accreditation.

The Advisor architecture contains several knowledge bases and inference engines designed to manage the information and provide an appropriate interface for the CSSO. Each of the knowledge bases—Policy, Safeguards, Attack Scenarios, and Computing Environment—resides in independent data files. These files allow the rapid updating of the knowledge to reflect policy changes, technology advances, or adjustments in the attack scenarios. Two additional components that are critical to determining the security of a computing system are estimates of the data exposure and the sensitivity level of the information on the system.

3. Security Issues in Computer Operating Systems (M. Steuerwalt, N-4). We wrote a guideline for assessing the security of Unix systems and began implementing the test plan and building the tools described in it. The guide outlines the factors that influence Unix security; presents a schema for assessing security; describes the scope of the study's technical security functions (identification, authentication, access control, privilege, and audit mechanisms), system administration, and communications; and suggests tools to help assess and maintain system security. To ensure the widest application for the study, we follow the efforts of POSIX* wherever possible. The first tools address issues of identification, authentication, and file integrity.

C. Network Research and Development.

We have an ongoing task to examine and identify DOE needs and perform R&D for computer networks to provide technological advice and assistance for the growing demands of data and network security.

Generic Model of DOE Computer Networks (J. S. Dreicer, N-4). We are developing a prototype graphical representation model (PROGREP) to support the graphical display and security characterization of computer networks. At this time, the establishment of stand-alone computer systems is limited to special situations; the trend for the development of new, and the modification of many existing, computer systems is toward

* POSIX is the group developing IEEE Portable Operating Systems Interface Standards.

networking. This is a result of the benefits that networking provides, such as economies of scale, enhanced productivity, efficient communication, resource sharing, and increased reliability. Inherent in the desire to network is the implicit acceptance of increased interconnection with other computers, and these may be interconnected to other unknown computers or networks. This increased interconnection or connectivity can result in a combinatorially explosive number of computers that attain the capacity to communicate. Although this capability has advantages, it also presents a challenge to and is potentially disadvantageous for ensuring and maintaining the security of computer systems that are networked to other, unknown computer systems and users. Furthermore, it presents a challenge in managing network resources and in understanding and remembering the specifics of each of these resources. This is of particular concern to the DOE because of the nature and sensitivity of the data that are processed and stored on DOE and DOE contractor computer systems.

The DOE has a large number of local area networks (LANs) and subnets (small LANs connected to larger networks), is connected to a variety of national and international networks (i.e., BITNET, HEPNET, ARPANET), and operates several wide-area networks for DOE use only (i.e., NWCNET). For DOE contractors to do their work efficiently, it is necessary to promote and continue using computer networks. However, there is a positive correlation between the increased dependence on computer networks and the need to determine methodologies and procedures that graphically represent and ensure the security of these networks. Some relatively recent events that demonstrate the need for applied research and development in network security are the November 1988 Morris worm, the German Chaos Club infiltration of computer systems at various US government organizations, and various attacks on the LLNL network. While the rapid emergence of networks already has had many beneficial impacts, research into network security has just begun. Thus there is a lack of knowledge, tools, and the capability to understand and address the problem adequately. The applied research undertaken for the PROGREP model effort is the first step in developing a research program, tools, and methodologies to investigate network security.

The purpose of the PROGREP research effort and system development is to (1) develop a better understanding of computer networks, which is required for future R&D, (2) provide a tool capable of graphically representing any computer network, which is required by computer security personnel, (3) develop methodologies that detect and indicate security relevant information and events, (4) develop methodologies that check the security of proposed network topologies, and (5) expand the means to conduct further network security and graph theory research.

The primary goals of the PROGREP research effort and software system are to help system security personnel check the security of existing networks, to determine the security of proposed networks, and to conduct applied research into graph theoretical problems. Therefore, it was our goal to produce a network representation that is realistic and valid rather than the ultimate network representation system, and to provide a useful tool to computer security personnel that would enhance their understanding and the security of an actual computer network.

The first phase in the development of the PROGREP model was to establish and determine an analytical basis by which to define computer networks as generically as possible. An additional constraint on the analytical foundation of this work was that it provide flexibility in any representation and characterization of real systems (e.g., computer networks). During this phase of the effort, it became apparent that confusion existed concerning the technical description of a computer network.

In the PROGREP model, the definition of a computer network is very general; a computer network is any collection of interconnected autonomous computers or components of slave hardware (e.g., printers, disk storage components, or plotters). If two or more computers or components of slave hardware are able to exchange information, they are interconnected. This definition of a computer network complements the definition of a graph — a structure $G = (V, E)$ that consists of a finite set of vertices V and a finite set of edges E (an edge is specified by an unordered pair of distinct vertices). In the PROGREP model, computer networks are represented and characterized in terms of graph theory and graph structures. The components (computer or slave hardware) of a computer network (e.g., computer, gateway, printer, or disk storage) are defined in terms of vertices, and the interconnections or network links are defined in terms of edges. Computer network security requirements and risks are modeled as constraints at the vertices and across the edges of the represented network (graph structure).

The PROGREP model research effort has provided great insight into the proper means by which to approach the modeling of graph structures in general and computer networks in particular. The PROGREP model permits quick and efficient representation of network components, interconnections, and interrelationships. Important features of the model are its intelligent and graphical interfaces. The intelligent interface aids the user in creating a dynamic network by providing logical control of the specification of the computer characteristics, parameters, properties, and security factors through the use of text and graphics. The graphical interface allows the user to display the topology of the configured network and analyze its security.

We use several approaches to answering network security related concerns and issues. The first approach is the stand-alone security checks and data capture. These security checks ensure compliance with policy and regulations concerning the use of various operating modes and the necessary hardware and software functions associated with particular evaluated-products list levels. The second approach is the network interconnection security checks and data capture. These security checks ensure data transfer compatibility over a link, operating mode compatibility between machines, an indication of the creation of a multilevel system, and the indication of a possible cascade problem between machines.

Although the PROGREP effort was intended originally to address computer network security, the model also appears to be applicable to nuclear safeguards because of the parallels between the basic principles of computer security and safeguards systems. In computer security the intent is to protect the data and information on computer systems; in safeguards the intent is to protect the SNM and the inventory data associated with the material. With modifications, the PROGREP model could represent SNM process lines, which are fundamentally graph structures. The PROGREP model is capable of representing process lines (directed graphs) but will have to be modified to allow for real world characterizing and modeling of safeguards systems.

D. Database Security (L. M. Harris, N-4).

Database security addresses the issues of data confidentiality, integrity, and availability, as well as the protection of data from accidental or intentional disclosure, destruction, or modification. Because DOE installations frequently rely on databases containing classified information, the intentional or inadvertent disclosure, alteration, or destruction of that information could cause significant harm and affect both the continuation of facility operations and national security.

The question of how best to ensure the security of information is fundamental to the notion of database security. Because databases store more information in a single location than most nondatabase files, and because database technology was developed to facilitate information sharing among users, it is clear that databases are a potential target for insider threats. When a database is used to store sensitive or classified information, its potential as a threat target increases, as does both its importance to the facility and the facility's need to protect it. These factors, along with the relative newness of the technology, make database security a new concern for computer security.

Database management system (DBMS) security research to date, although primarily theoretical in nature, has followed a course of action similar to that taken by secure operating system research. Both operating and database management systems have similar sets of security problems, and in many cases the approaches and solutions to problems associated with database security have evolved from those used to ensure the security of operating systems. However, database system security poses additional challenges because security must be accomplished at a finer level of granularity. For example, operating systems may protect information at the file level, while database management systems must protect information at the table, tuple, and data element levels.

Many DBMS researchers are concerned that existing security models developed for operating systems are not entirely applicable to database management systems. As there is a recognized need for a formal security model to define secure database management systems, DBMS researchers are attempting to define exactly what "secure" means for database management systems. Innovative approaches to the application of security policies are being tested. However until such time that a DBMS-specific security policy is constructed, Department of Defense regulations for processing classified data will be used.

Disclosure and integrity are the major concerns of database security. Disclosure issues address confidentiality and protect data from accidental or intentional release. To protect against disclosure, hardware and software safeguards focus on preventing information from being revealed to a user or process that is not authorized to have access to it. Integrity issues concern the correctness and validity of data, as well as the protection of that data from accidental or intentional destruction and modification. The focus of integrity is not who is allowed to access the information, but rather who is allowed to modify it. Protection can be accomplished by limiting both the set of users allowed to modify the database and the transactions they are able to perform.

Classifying database security issues into the categories of disclosure and integrity is a somewhat fuzzy activity. However, to simplify the understanding of database system security, the topics of access control, covert channels, data labelling, inference, aggregation, and sanitization are discussed as disclosure issues. Unauthorized modification problems (both direct and indirect threats) and valid database state problems such as integrity constraints, recovery, and concurrency problems, as well as polyinstantiation, are discussed as integrity problems.

We have examined the security issues associated with database management systems, current research in the field, and available or soon-to-be-released security products and prepared a report. In addition, we created a library of database security publications, articles, and references. Although primarily an information gathering/educational activity, the results of this task have been converted into training aids for CSSO's and those individuals responsible for database administration and database development or use. These tools meet the training requirements specified in DOE Order 5637.2A. In addition, we are developing field support applications of DBMS security technology.

E. Computer Security Technology Transfer and Support.

We have an ongoing task to provide technical support and assistance in computer security to DOE/OSS and to the DOE user community. This support includes reviews and evaluations, information collection and dissemination, and liaison with industry, academia, and government agencies.

1. Computer Security Newsletter (K. G. Redle, N-4). During calendar year 1989, we published two issues (April and August) of the *DOE Center for Computer Security News* and distributed them to nearly 1200 people. Activities related to newsletter production included planning, contacting potential authors, researching computer security periodicals for articles, writing and editing articles, ensuring timely classification review, submitting articles to the Center staff and DOE/OSS for review, and collecting feedback on how to improve the newsletter.

2. Computer Security Enhancement Reviews (W. J. Hunteman, N-4). The DOE/CCS conducts independent computer security reviews for DOE or DOE contractors that process classified information. This service, known as the Computer Security Enhancement Review (CSER) program, provides DOE facilities with a friendly but critical assessment of the site's computer security program. The program has evolved over the last four years to meet the changing computer security requirements in the DOE. During 1989, the CCS conducted CSER's at the Savannah River Operations Office, the Oak Ridge Operations Office, the Albuquerque Operations Office, and at DOE/OSS. In addition to the traditional CSER's, the CCS participated in a variety of surveys or reviews of specific computer security issues.

The CSER program has resulted in a number of benefits at all levels of the DOE's classified computer security program. Computer security officers at the sites have gained an improved understanding of DOE orders and regulations and have learned about good computer security practices at other sites.

The CCS has gained an enhanced understanding of the issues, problems, and practic lity of existing computer security solutions, which is shared with DOE headquarters through general discussions, research activities, participation in working groups on specific issues, and the CCS report, "Lessons Learned in the DOE Computer Security Enhancement Review Program."[53]
Furthermore, the CCS R&D program is based on findings and needs identified during CSERs.

The CSER process begins with a site's request to the CCS for a review. The request always is voluntary; neither the CCS nor the DOE computer security program management ever initiates a CSER. If the CCS agrees to provide the CSER, the CCS and the site jointly agree on the time, duration, and focus of the review. The previsit discussions may include such items as computer systems to be included in the review, areas of emphasis, and CSER team members. The CSER team consists of experts from the CCS and a representative from the site's computer security organization.

The actual CSER begins with a briefing by the site's computer security organization to acquaint the CSER team with the site. The briefing also identifies any special issues that must be addressed during the CSER. The CSER team then briefs the site management on the CSER process and the extent and timing of this particular CSER. A tentative schedule of facility and individual visits is developed during the discussions. After the in-briefing, the CSER team begins the review.

During the review discussions, the team probes the implementation of the site's classified computer security program. The discussions also address each interviewee's understanding and the implementation of the local program. The discussions are characterized by friendly, unconstrained sharing of information.

The visits and discussions continue until the CCS members of the CSER team have developed a thorough understanding of the site's computer security program. The team's findings are then presented at an out-briefing. The team reviews its findings with the site management and computer security organization at the out-briefing. Attendance at the out-briefing is always controlled by the site.

CSER findings are not routinely documented or disclosed to any audience without the approval of the site's computer security organization. All notes collected by the CSER team are treated as classified information. The notes may be left at the site or destroyed following the procedures established for destruction of classified information.

F. Audit Log Analysis Package (D. P. Martinez, M. Steuerwalt, and C. A. Steverson, N-4; K. G. Redle, ADP-4)

The objective of this project is to develop tools for use by the DOE computer security community in examining audit trails of computer use for evidence of intrusion into the computer system.

We have finished developing and testing the Audit Log Analysis Package (ALAP), which processes VAX/VMS image accounting data for auditing and analyzing DOE VMS computer system activities to detect and analyze abnormal computer system/user behavior. The software runs on VMS operating systems.

The ALAP software incorporates methodology (known as "Wisdom and Sense (W&S)") that was developed for detecting anomalous data entry in nuclear materials accounting databases. The concept is that anomalous data can be detected based on rules (i.e., a rulebase) formulated from historical data that reflect normal behavior for the given system. ALAP is designed to use VMS image accounting data as its historical database.

In 1988, we moved the ALAP software from a prototype development environment (i.e., IBM RT) into its product target environment on a Digital Equipment Corporation VMS operating system in preparation for local testing at the CCS and alpha testing at the Los Alamos National Laboratory.

Software revisions and refinements, identified during the tests, were addressed systematically by managing and testing the software. First, the ALAP principal investigator developed standard ALAP software change request (SCR) forms to describe the necessary revisions. The SCRs were distributed to the appropriate ALAP programming staff. The revised software was moved into a local test environment. The principal investigator developed a test plan to be administered by him and performed by other members of the ALAP test team. If additional software refinements were needed, we continued to follow the same configuration management procedures until the software passed local tests.

The ALAP User Guide was revised along with the software, again based on the test results and recommendations of the ALAP team, and distributed for review and comment.

After local testing was completed and the revised software and ALAP User Guide were accepted, both the software and the documentation were subjected to coherence testing. All inconsistencies identified during testing were resolved and, through additional testing, subsequently validated as coherent.

During the first half of FY 1989, the software and documentation were revised several times, which significantly improved the functionality, reliability, and ease of use of ALAP. Software modifi ations included a much improved audit-session logging facility, an improved user interface, and a more reliable software interface. Major improvements to the documentation included a glossary of ALAP terminology, an ALAP practice-session section, and a logging facility documentation section.

After completing the revision, refinement, and testing phase, the Center distributed ALAP to selected DOE organizations and DOE contractors for formal beta testing. Potential beta test sites were identified through programmatic interactions among the Center staff, the DOE, and DOE contractor personnel and in response to a paper[54] presented at the 1989 DOE Computer Security Group Conference in Amarillo, Texas. Many people expressed interest in obtaining ALAP once it became publicly available, and a few expressed interest in becoming beta test sites.

Beta testing began in April 1989 and continued through most of the remainder of FY 1989. Beta test results revealed that ALAP was easy to use, effective, reliable, capable, but limited. Comments on the usefulness of ALAP were generally positive. Although most sites initially required time to set it up and learn it, they indicated that ALAP was effective at detecting anomalous behavior on their systems and useful as a tool for analyzing VMS audit trails. Several beta testers indicated that auditing with ALAP increased their awareness of activities performed on their systems.

Comments on the ALAP User Guide also were generally positive. Beta testers commented that it was well written and organized and found the "Getting Started" and "Running a Practice Session" sections to be extremely valuable and helpful.

From the outset, we recognized that ALAP would have some limitations. We also recognized that auditing with ALAP would require a moderate amount of user resources. Many of the problems reported by beta testers emphasized concerns we had already identified. Descriptions of the major problems reported by testers, together with our evaluation and recommendations regarding those problems, follow:

- Software Implementation Problems—Two beta test sites reported problems running ALAP; it crashed during rulebase generation.

  After evaluating the problem, we discovered that the beta testers who experienced crashes had not set the account memory parameters as described in the ALAP User Guide. To increase awareness of this issue, we revised the guide to place more emphasis on ALAP account configuration during installation. The need for proper account configuration also is emphasized in the ALAP release notes.
- Logging Facility Needs Improvement—Beta testers indicated a need for more effective logging of anomalous data.

  The ALAP team evaluated the beta ALAP logging facility. We agreed it provided inadequate logging of ALAP audit session activities, so we redesigned it and incorporated the improved logging facility into the ALAP 1.2 release. We recommend that future audit systems consider using a database to extract and summarize selected audit information from the audit session log in a more effective manner.
- ALAP Tuning Problems—Testers reported that ALAP flagged too many normal transactions as anomalous.

  ALAP was designed to incorporate an already existing methodology for detecting anomalies based on rules derived from historical data. If the data being audited by ALAP are not reflected in the historical data (i.e., in the rulebase), then ALAP will appropriately flag that data as abnormal. This creates a problem when events do not occur frequently and yet are normal; for example, when a new user is added to the system after the rulebase has been generated from the historical data set, but before the selection and processing of the current audit data set. Review and analysis of such "false alarms" does significantly increase the time required to audit system activity, particularly when working with large data sets.

Several issues are associated with anomaly tuning. The anomaly detection algorithm used in ALAP represents the methodology employed at the time ALAP was initially developed. We are continuing to improve this methodology. A second issue is that of rulebase editing and updating. It would be useful to have mechanisms that would allow ALAP users to edit, update (e.g., add a new user's profile), and tune a rulebase to their site-specific needs. Such features would significantly reduce the "false alarm" rate and increase ALAP's effectiveness. We have improved the research version of ALAP, but more research is required to resolve related issues, such as rulebase integrity, before the methodology can be implemented in a product version of the software.

- Resource Problems—Large test sites reported that ALAP was too resource intensive with prohibitive memory and processing requirements.

  Auditing large systems is inherently resource intensive. ALAP was designed to use standard VMS image accounting data generated by the VMS accounting utility. However, some records not used by ALAP are included in the data automatically generated by the accounting utility. This combination of unused (by ALAP) data with the necessary image accounting data results in large storage requirements.

  Auditing large systems with ALAP can be resource intensive. We recommend that ALAP auditing for such systems be performed on a separate dedicated VMS system, such as a VAX workstation, to keep from overloading the host system.

- User-Interface Deficiencies—Some test sites recommended that the historical data processing, and rulebase generation processes be performed in batch mode, particularly when working with large data sets. Other sites indicated that they would like a version of ALAP that would run in the background and trigger an alarm whenever an anomaly occurred.

  ALAP was designed to be an interactive audit/analysis tool. However, we agree that the user interface should have the flexibility to allow the user to perform both the historical data and rulebase generation processes in batch mode. Accomplishing these tasks on large data sets takes hours and requires that the user be present at or near the terminal. Revising the current version of ALAP to incorporate this option will require moderate modifications of the ALAP user interface and the ALAP User Guide.

  We considered having ALAP run in real time or near real time in the background as an option during the initial design stages, but decided against it. To obtain the VMS audit data in real time would require unacceptable modifications to the standard VMS operating system software. For ALAP to be effective in real time, it would have to have "learning" capabilities to update and tune the rulebase automatically.

Based on the recommendations derived from reviewing and analyzing the problems reported during beta testing, we developed a revised version of the ALAP software and documentation and tested both at the CCS. The revised ALAP (version 1.2) is now available for distribution to interested sites.

G. Academic Collaborations and Interactions (J. S. Dreicer, N-4).

This effort resulted from a proposal that stressed the need to establish liaison and relationship with academic institutions and industry to keep pace with the rapidly changing technologies in computer science, in particular, computer security. Five primary objectives were associated with this proposal: to establish technical collaboration between the US DOE CCS and university researchers, to create a recruitment mechanism for future staff, to encourage computer security education at the graduate level, to direct applied computer security research, and to support CCS efforts.

We collaborated and interacted with UNM and University of California-Davis (UCD). During the past year, under the direction of the principal investigator, Dr. McCabe, UNM began applied research into network monitoring and formal models. At UCD, Dr. Topkis conducted research into network security constraints and routing alternatives; in addition, he is attempting to identify a qualified graduate student to carry on applied research.

During the first six months of the academic liaison program, we determined which institutions to target, established working relationships, set up collaborative exchanges, identified graduate students, trained the graduate students, and identified specific research topics. These university associations are firmly established with UNM and UCD and continue to evolve and mature. Recently, Dr. Luger, another professor at UNM, has become involved with Dr. McCabe on the UNM research project. We expect that the collaborative relationships with UNM and UCD will continue.

We also are taking advantage of the Los Alamos National Laboratory's graduate research assistant (GRA) program. It is organized to allow a Laboratory staff member the opportunity to select technically qualified graduate students from universities around the country and to hire them for three or four months. This provides both the GRA and the Laboratory mentor with a mutual evaluation period. The Laboratory mentor can evaluate the technical education and background, quality of knowledge, maturity, and capability of the GRA. If the GRA has technical potential and is a good prospect for future employment, the mentor can provide specific technical training and education, encourage areas of educational study, and encourage thesis or dissertation work to be done at the Laboratory. In essence, the GRA program allows the expeditious hiring of capable technical staff with low risk of lost resources. Last summer, the CCS hired two GRAs, one from the University of Illinois and one from Washington State University. These two GRAs spent four months at the CCS working on computer security related problems.

Although the potential for the academic interaction program seems positive, its progress and ultimate success depends on the constraints, expectations, and resources that are placed on it during the first several years. We are confident that a consistent program in this area will result in our continued capability to solve computer security problems and to assess computer security technologies as they relate to DOE.

PART 2. BASIC SYSTEMS DESIGN, INTEGRATION, AND EVALUATION

I. SAFEGUARDS SYSTEMS METHODOLOGY DEVELOPMENT

A. Basic Systems Design and Integration.

The objectives of this task are to develop and transfer to field personnel a methodology for designing safeguards systems that emphasizes integration of materials control, materials accounting, physical protection, and operational activities such as process monitoring. It is intended to improve the effectiveness of safeguards resources through an integrated approach to DOE policy, orders, and implementation of safeguards activities.

Some of the projects described here are supported, in part, by facility implementation funding.

1. Special Isotope Separation (SIS) Project. The SIS production plant will convert fuel-grade plutonium to weapons-grade plutonium using the atomic vapor laser isotope separation (AVLIS) process; it is to be located at the Idaho National Engineering Laboratory (INEL). We are working with the SIS Project Management Office and the SIS Safeguards Task Force to deve'op an integrated system of accountability measurements and nuclear materials control.

a. Safeguards Systems Design Concepts Applied to SIS (R. S. Leonard and R. B. Strittmatter, N-4). In support of safeguards planning for the proposed SIS plutonium processing facility, the Los Alamos Safeguards Program has provided input on the materials accounting needs of the new facility. This effort, supported by both OSS and the SIS Project Office, carefully studied the process flow, identified important locations for measurements, and quantitatively analyzed the variance propagation for the process. We conducted a systems study of the proposed SIS facility and developed a set of recommendations for creating a cost-effective safeguards system. We identified NDA instruments and techniques that can provide useful safeguards, accountability, or process control information and estimated measurement uncertainties for some process materials. Through periodic reporting to and review by the SIS Safeguards Task Force, the SIS project benefitted from technical support and guidance in systems design, safeguards instrumentation, and measurement technology developed through the DOE safeguards R&D program. The safeguards community will, in turn, benefit by applying newly developed analysis and design methodologies (related to variance propagation) to an actual facility, studying the results, and by developing and characterizing NDA methods applicable to pyrochemical processing techniques and difficult to measure plutonium and americium wastes.

Based on work done in 1988 (for the SIS project) on the conceptual design of an integrated safeguards system, we further analyzed the process flow streams using the LANL variance propagation code PROFF (PROcessing and Fuel Facilities). The amount of data from transfer streams, material composition, measurement methods, and associated errors, coupled with the need to rearrange or create new groupings of data, required manipulating the data with both spreadsheets and databases.

Transferring the results of variance calculations made using PROFF to the spreadsheet program EXCEL facilitated reporting and comparing results. Having the results in a spreadsheet rather than a BASIC output file allowed the data to be easily plotted as a function of either throughput or instrument uncertainties. By these comparisons, we identified the need to develop an improved set of instrument uncertainties that could be associated with various material streams. This list made possible more precise variance calculations and a clearer identification of process steps that are potentially attractive theft or diversion targets for an insider. In particular, the more detailed analysis highlighted the possibility of protracted diversion of the small amounts of material being discarded into the waste streams, although concerns had centered on the abrupt theft of significant quantities of material.

We analyzed the system performance of the SIS safeguards design in detail and from the top down. This approach required grouping a significant amount of numerical and descriptive data in various ways. The material balance variances must be propagated for each data grouping and, for trade studies, the results stored for later use. The results then have to be recalled, compared, and ranked. Given the constraints, such as impact on operations, dose limits, and funds available, decisions can then be made on the best configuration or grouping of individual unit (IUN) processes, either controlled balance area (CBA) or material balance area (MBA), that will meet the DOE Orders and performance criteria. The top down detailed analysis is best suited to new facilities, major upgrades, changing threat environments, and revised orders that require improved performance from existing systems.

Toward the end of the fiscal year, we realized that productivity would be enhanced if the data were available in a database on which variance calculations could be made without leaving the database program. Foxbase+/Mac was chosen because it is available for both Macintoshes and MS-DOS machines, it can incorporate graphics as fields, and "C" programs can be included as subroutines. The basic tables (1 thru 5, mentioned below) that make up the database are shown in Fig. 28.

Table 1 contains data about the processes. As a design or analysis progress›s, additional detail can be added. The table identifies which MBA, CBA, and IUN and NDA station that a process belongs to. By sorting the 80 plus process flow diagrams (PFD's) in various ways, such as by sub-MBA (CBA) or NDA station, we identified various conflicts or poor groupings of materials.

Table 2 contains information about the process streams. Typical fields in a record are the stream number, type of material in the stream, quantity of material, whether the material is a transfer in or out or an inventory item, and if the material is measured.

Table 3 documents the chemicals making up the material stream and their quantities. The quantities are summed and posted back to the quantity field in Table 2. Table 3 is linked, as are Tables 4 and 5, to Table 2 by PFD identifier and stream number.

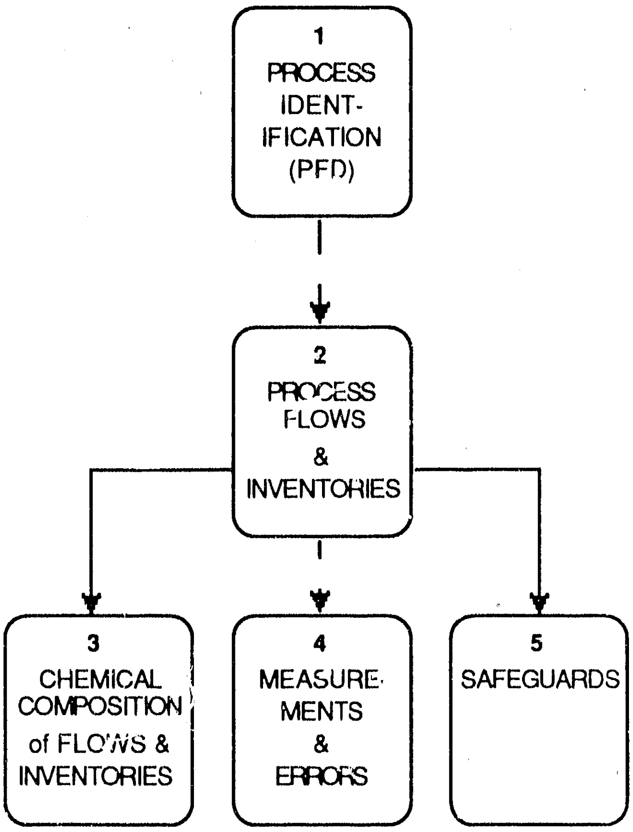


Fig. 28. Hierarchy of relational databases.

Table 4 contains information about the instrument used to measure a particular process stream or inventory item. Instrument uncertainty, tied to the material and quantity being measured, is also included in this table.

Table 5 contains safeguards data that are a combination of calculated and posted information. Typical data will include the measurement limit-of-error for the material, probability of detection, attractiveness, and notes. This table was deliberately set up to be a stand-alone table that could be isolated.

Correlations between measurement streams were identified by sorting either Table 1 or Table 2 by NDA station. This information was then used in setting up the input files for PROFF.

Sensitivity studies were conducted on such variables as throughput, size of instrument uncertainties and configuration of sub-MBA's. Progress in conducting the sensitivity analysis and developing concepts was reported to the project and task force in April at Idaho and in October at Livermore.

Work planned for FY 1990 includes completing the sensitivity analysis using the new database analysis tool. A final report will summarize the results as well as describe the development of the integrated information.

The use of a relational database program with a C-based variance calculation module can greatly facilitate systems design and analysis. The results of a development effort, carried out in support of the safeguards task force for the SIS project, are reported. The work has resulted in a tool, which when refined, may be used by facility accountability clerks or DOE reviewers to either plan upgrades to existing facilities or design new accountability systems. The set of generic tables used to organize the data is described along with a standard set of reports created for presenting the results of the analyses and trade studies.

Although the work was done using a Macintosh, the database program, Foxbase+, is available for the PC user. The work, because of the ease of manipulating data within the structure of a relational database and the associated help or tutorial files, will be useful to the analyst and accountability clerk who wishes to develop a better understanding of how to improve their MC&A system in a cost-effective manner.

b. Potential NDA Options and Accuracies for the SIS Process (N. Ensslin, T. K. Li, P. A. Russo, N-1). In support of safeguards planning for the proposed SIS plutonium processing facility, the Los Alamos Safeguards Program has provided input on the materials accounting needs of the new facility. This effort has carefully studied the process flow, identified important locations for measurements, and quantitatively analyzed the error propagation for the process. As part of this effort, we have identified NDA instruments and techniques that can provide useful safeguards, accountability, or process control information, and we have tried to estimate the expected measurement uncertainties for some process materials.

The SIS process would require the destructive or nondestructive measurement of plutonium in a variety of solid or liquid forms. Because the facility would employ pyrochemical processing techniques and would generate plutonium and americium wastes in a variety of forms, many of the materials would be difficult to measure accurately. For nondestructive measurements, we considered the following potential techniques: calorimetry, isotopic analysis by high-resolution gamma-ray spectroscopy (ISO), conventional neutron coincidence counting (NCC), neutron multiplicity counting (NMC), lump-corrected segmented gamma-ray scanning (LCSGS), plutonium

SAI, x-ray fluorescence (XRF), and gamma-ray-based portable process and holdup monitoring. For each technique, we identified the minimum number of instruments or instrument locations that should be considered for the facility if the technique was considered sufficiently accurate, timely, or labor-saving to be useful.

Table XII is an example of the type of information that we prepared for SIS. This table includes four of the above-mentioned techniques and lists a number of pyrochemical, scrap, or waste process materials. The uncertainty estimates were obtained by combining the expected random and systematic errors in quadrature. The random errors were obtained from the expected counting precision. The systematic errors were obtained either from past user experience or from NDA instrument developer estimates of the expected effects from the sample matrix or the inhomogeneities on the assay uncertainty. In general, user experience provides more conservative estimates, whereas instrument developer experience reflects goals that can reasonably be achieved if the SIS facility takes advantage of forthcoming improvements in NDA technology.

We expect significant improvements in NDA accuracy to be available for the new SIS facility. These improvements include use of the known-M approach or the sample self-interrogation technique to supplement conventional NCC, or the use of new NMC techniques to measure impure plutonium samples with unknown multiplication and (α,n) yields. Gamma-ray assay improvements include the development of LCSGS, and better algorithms for isotopic measurements of heterogeneous materials. Other potential improvements in neutron NDA accuracy can result from process changes that reduce the (α,n) neutron reaction rate in the sample. In some cases it may be

Table XII. Estimated Uncertainty of Some NDA Techniques for Various SIS Process Materials (quoted as relative standard deviation, in %)

SIS Process Materials	NCC[a]	NMC[b]	LCSGS[c]	ISO[d]	
Incoming drums	5-10	-	-	0.1 - 0.2	0.5 - 1
Receipt cans	1-10	1-3	-	0.1 - 0.2	0.5 - 1
Byproduct oxide	5	2-3	-	0.1 - 0.2	0.5 - 1
DOR input salt cake	5	-	5-10	0.3 5	
DOR spent salts, sweeps	5-10	-	2-10	0.3 5	
DOR scrubbed salts	5	-	2-5	0.3 5	
DORSS metal buttons	10-20	3-5	-	0.3 5	
DOR metal product	10	2-3	-	0.3 5	
MSE spent salts, sweeps	5-10	-	2-10	0.3 5	
MSE scrubbed salts	5-20	-	2-5	0.3 5	
MSESS metal buttons	10	-	-	0.3 5	
MSE metal castings	1-2	-	-	0.3 5	
MSE skull & crucible	20	5	15		-
ER metal input	3-10	2-4	-	0.1 - 0.2	0.5 - 1
ER anode heels	10-20	5	-		-
ER spent salts	5-10	5	1-5	0.2 - 0.3	1-2
ER scrubbed salts	5	-	2-5	0.2 - 0.3	1-2
ERSS metal buttons	10	3-5	-	0.2 - 0.3	1-2
ER metal product	1-2	-	-	0.1 - 0.2	0.5- 1
ER skull & crucible	20	5	15		-
Failed furnace parts	5	-	-		-
Passivated scrap	5-10	5	5-10		-
Sweeps/reactive scrap	5	2-5	-		-
Combust. waste (low-α)	5-10	-	2-5		-
Combust. waste (high-α)	10-100	-	2-5		-

[a]NCC - conventional neutron coincidence counting
[b]NMC - neutron multiplicity counting
[c]LCSGS - lump-corrected segmented gamma-ray scanning
[d]ISO - isotopic composition of homogeneous plutonium by high-resolution gamma-ray spectroscopy

possible to improve NDA accuracy by blending or homogenizing process materials before measurement. Standardizing can sizes and material handling procedures can also help. The assay uncertainty estimates in Table XII reflect the above-mentioned improvements in NDA techniques, but not the potential improvements in processing or material handling.

c. Applications of FacSim to Inventory Difference Calculations for SIS (D. Stirpe and C. A. Coulter, N-4). The FacSim (Facility Simulation) program can be used to analyze expected inventory difference variances at facilities that process nuclear materials.[55] We previously developed a preliminary FacSim data file for SIS that described the major material flows in the facility to evaluate the proposed measurements and MBA structures. Results from the application of FacSim to this data file were presented at an SIS Safeguards and Security Task Force meeting in Idaho Falls. The data file has now been greatly expanded to include the description of processing and measurements on *all* facility materials, including low-level scrap and waste streams. This enhanced data file will be used in calendar year 1990 for a detailed evaluation of measurement and MBA alternatives for SIS.

2. Integrated System for Westinghouse Hanford (K. E. Thomas, J. S. Ballman, and R. M. Tisinger, N-4). An integrated safeguards system demonstration–Insider Demonstration–is planned for the Plutonium Finishing Plant at Westinghouse Hanford at the end of 1990. SNLA, Westinghouse Hanford, and Los Alamos are participating in this demonstration. The demonstration will be run in a laboratory but will simulate operations in the processing plant. PC-DYMAC, the accounting system developed for Argonne National Laboratory-West (ANL-W), will be the accounting system for the demonstration. Sandia will apply the WATCH system in the demonstration. New integrated safeguards concepts that will be demonstrated include personnel monitoring (Sandia and Los Alamos). In addition, two computers will be used to demonstrate connecting classified and unclassified computers using the Hanford RDTC (restricted data transmission controller) switch.[56,57]

B. Computer Simulation Development – Enhancements to FacSim (C. A. Coulter, R. Whiteson, and A. Zardecki, N-4).

Computer-based modeling and simulation has historically been a useful, but cumbersome, tool for safeguards systems design and evaluation. Recent advances in commercial software and hardware have made it possible to develop much more flexible modeling and simulation tools to support safeguards. We have developed a generic simulation model for safeguards systems (FacSim), which is designed to simulate both processing and safeguards functions for facilities that process nuclear materials. We are enhancing this model to facilitate system design and analysis tasks. The several enhancements are described below.

User interface. We added a menu-driven interface to execute the model. New data-entry checks were incorporated to help the user guard against entering erroneous information. Color displays are now used to improve readability and video appearance.

Material flow sheet display. Material flows in nuclear material processing facilities usually involve many recycle and scrap-recovery streams, and can be complex and difficult to visualize. For this reason, we enhanced FacSim to provide the ability to display the materials flow sheet for the simulated facility on the computer screen. This capability not only provides additional information on facility operation, but aids in developing and debugging model data files as well.

To achieve an efficient and understandable flow sheet layout, the flow sheet routines first construct an approximate arrangement of vault areas and unit processes from the facility data file by simple algorithms, and then use a simulated annealing process to optimize the arrangement by reducing the lengths of the material flow lines. The vault areas and unit processes are then displayed on a rectangular array of screens, each of which can display up to nine process/vault areas arranged in three rows and three columns. These process/vault areas are connected to one another by directed lines that show the material flows. The lines are color coded to distinguish among material flows that go between two locations on the current display screen, flows that go between one location on the current screen and one that is not on the screen, and flows that cross the screen between two locations neither of which is on the screen. A list of all materials that flow across the current display screen appears on the screen, with numerical keys relating materials on the list to labels on the flow lines. The user may move from one display screen to another by using the cursor keys. To provide continuity, adjacent display screens overlap by one row or column.

Development of anomaly detection algorithms. We are enhancing FacSim to permit it to describe the actions of insider adversaries and the response of the facility safeguards system to these actions. A part of the anomaly detection ability of the safeguards system will be a suite of statistical tests that can be applied to the inventory difference values generated by material balance closures at both the MBA and sub-MBA level. We have begun developing and testing the algorithms and computer procedures for various single-decision and sequential tests that may be included in FacSim for this purpose. Tests that are being developed include the classical Shewhart test, the Wald sequential probability ratio test, and various versions and combinations of Page's test. In addition, we are evaluating various pseudorandom-number generators for use in the simulation tests, to try to assure that there are no inherent nonrandom biases in the generators used for the simulations that could lead to misleading conclusions about the effectiveness of any of the tests considered.

Description of insider threat scenarios. We are enhancing FacSim so that it can model insider adversary actions and the response of the facility safeguards system to these actions. This will allow one to evaluate the operation of the safeguards system within the context of facility operation and to obtain detection time estimates that might otherwise be difficult to determine. To provide the flexibility required for simulating operator adversary actions, the description of process operators in the simulation program has been totally redesigned and reprogrammed, and the new program elements have been tested. The operator description is now completely object oriented. Facility elements requiring operator actions send messages requesting these actions and await operator response. Each operator defined in the facility data file is represented in the simulation program by his or her routine, which allows the operator to analyze requests, check on the status of the facility and of other operators, and take appropriate actions. In addition, each operator can pass requests to other operators and coordinate actions with these operators when necessary. This new operator representation provides a versatile framework for describing the kinds of independent and informed operator actions that are required for adversary scenarios. The approach that we are developing to generate adversary scenarios within the program uses artificial intelligence methods to provide the adversary with simulated intelligent behavior that is both goal-directed and nondeterministic. This approach also is opportunistic, in the sense that it takes advantage of transient facility situations to aid in achieving adversary goals when possible. The adversary formulates flexible, modifiable strategies that make use of his/her authorizations and facility knowledge. The effort to achieve the adversary's goals can be carried out by a series of steps performed over an extended period of time, with "backtracking" from unpromising situations when necessary. The formulation also provides some ability for the adversary to learn from his successes and failures and thus to improve his probability of success in future actions.

Run-time animation. We have added new run-time animation features to FacSim that allow the user to display one or more floor-plan views of facility areas with unit processes and/or vault areas displayed at appropriate locations. The information displayed for each unit process is:

- process name,
- total SNM in the feed queue,
- unit process status (indicated by a color code),
- amount of SNM in process, and
- total amount of SNM in the product queue.

For each vault area, the vault area name and the total amount of SNM in the vault area are displayed. These floor-plan displays can give the user additional perspectives on the time-sequenced interactions of related processes.

Simulation model language developments. FacSim is currently coded in a simulation language that runs under the MS/PC-DOS operating system on IBM PC-class computers. A new version of this simulation language that runs under the OS/2 operating system has become available, and we are developing a version of FacSim that uses this new version. Unfortunately, the initial release of the new language contained a number of errors that made its immediate use impossible. A corrected version should be available soon, and the FacSim conversion will then be completed. Run times of the OS/2 version of FacSim should be significantly shorter than run times of the current DOS version.

As FacSim has become larger and its development has become a multi-person effort, we have felt a strong need to re-express the model program in a simulation language that is more modular and object-oriented than that currently in use. Fortunately, a very attractive new simulation language that has the desired features and which operates under the OS/2 operating system on IBM PC/AT-compatible computers has just become available. We have, therefore, begun converting FacSim to this new simulation language. We are making some long-planned enhancements to FacSim, and also are developing generic simulation tools that will be useful in developing future models. These generic tools are contained in a "kernel" of procedures that manages user definition and revises model features by means of menu-driven input and maintains the disk files that contain the facility description. Other program modules are able to access these capabilities by making a few straightforward definitions and procedure calls, and can in this way avoid the necessity for managing their own user input and data file manipulation requirements. This will obviate most of the "nuisance" programming that has traditionally been necessary in every module of a simulation program, and thus speed both prototyping and full model development.

II. SAFEGUARDS SYSTEMS APPLICATIONS

Support for Headquarters Initiatives (K. E. Thomas, N-4)

The objective of this task is to provide technical support to DOE/OSS in understanding IDs and MC&A practices at operating facilities, to review existing orders, to support development of the MC&A portions of Master Site plans and Master Safeguards and Security Agreements (MSSAs), and to address DOE-complex-wide cost/risk benefits of applying proposed safeguards measures.

Throughout this year, we have continued to revise the MC&A Guides based on comments from the field and experience gained from site visits. We participated in site visits at Rocky Flats and Oak Ridge. In June, a workshop of the performance requirements was held at Brookhaven National Laboratory; we assisted in the preparation and helped conduct the workshop.

In December, we participated in a working group at the Central Training Academy that developed a guide for preparing MC&A plans.

PART 3. ONSITE TEST AND EVALUATION AND FACILITY SUPPORT

I. FIELD CONSULTATION AND DEMONSTRATION

The tasks described in this part of the report are designed to demonstrate, test, and evaluate onsite improved safeguards technology and to assist in upgrading materials measurement, control, and accounting capabilities and practices throughout the DOE complex. Many of the projects described here are partially supported by facility implementation funds.

A. Instruments and Measurement Systems for Uranium and Plutonium Processing Facilities.

A wide variety of measurement problems are still extant at facilities that process nuclear materials. The goal of this task is to improve NDA measurement technology for uranium and plutonium processing facilities and to demonstrate each technique in an operating process line. It includes improving existing measurement capability, in-line T&E of new measurement technology, and developing and demonstrating integrated measurement systems.

1. Assistance to Los Alamos

a. Plutonium Isotopics Assay System (T. E. Sampson N-1, T. A. Kelley C-3, G. W. Nelson, University of Arizona). Two FRAM[21] plutonium isotopic assay systems have been completed and delivered to Los Alamos users. The count room facility at TA-55 operated by group NMT-4 has been using a single-detector FRAM system for about nine months. This system is capable of performing almost all of the necessary nondestructive isotopic assay measurements and can be expanded easily to increase throughput. The second system (Fig. 29), which can measure two samples simultaneously, was delivered to the count room at Los Alamos' Nuclear Material Storage Facility.

Both systems were delivered with complete documentation -- hardware manual, software manuals, and user manual. A description of the system and its performance also has been published in Ref. 21. FRAM represents a significant improvement over the previous plutonium isotopic assay system used at TA-55.

b. A Versatile Passive/Active Neutron Coincidence Well Counter (P/A NCC) for In-Plant Accountability Measurements of Plutonium and Uranium (J. E. Stewart, H. O. Menlove, S. W. France, J. Baca, and R. R. Ferran, N-1; J. R. Wachter, NMT-4). We have designed a new well-type NCC for the NDA of a wide variety of items containing plutonium, uranium, or both. The instrument will be used at several measurement locations in the Los Alamos plutonium facility (TA-55).

Features of the design include:

- removable sample-well inserts for passive and active modes;

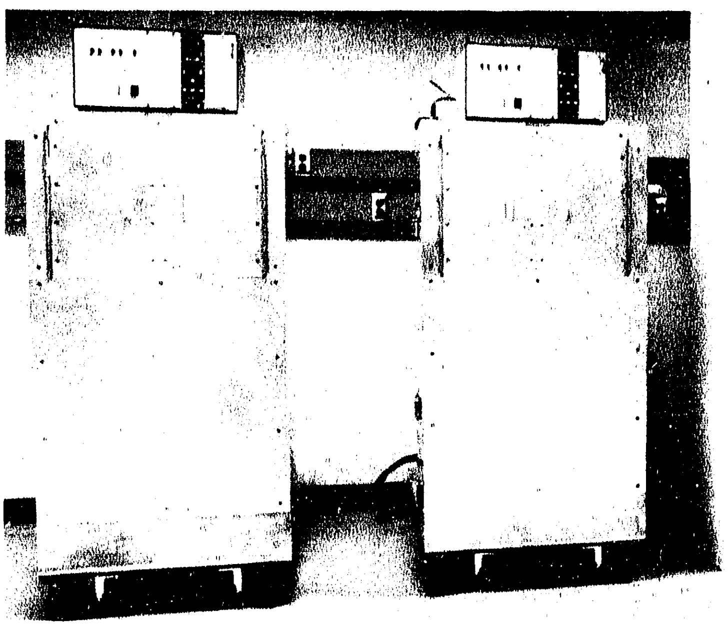


Fig. 29. The FRAM plutonium isotopic system installed at Los Alamos' Nuclear Material Storage Facility. This system is capable of measuring two (expandable to four) samples simultaneously.

- both fast and thermal-mode inserts for active measurements;
- removable AmLi sources for combined passive/active measurements;
- AmLi source intensity (two sources at 5 x 10^5 n/s each) sufficient to override passive coincidence background for mixed uranium/plutonium and/or impure plutonium samples;
- external shielding to reduce effects of plant neutron backgrounds;
- uniform efficiency over the sample cavity to minimize SNM positioning effects;
- low sensitivity to sample matrix;
- passive efficiency of 35.7%, active efficiency of 31.0%;
- large sample cavity to accommodate a wide range of containers; and
- inherent safety, reliability, and maintainability features.

Figure 30 is an elevation layout view of the instrument, showing the shielded sample well and the active insert suspended from the hoist arm. Figure 31 shows details of the active insert on the left and the passive insert on the right.

The active insert contains AmLi sources above and below the sample cavity; each is easily removed from its polyethylene end plug using a handling rod. The active cavity will accommodate a sample container up to 8.5 in. in diameter and 9 in. tall.

The passive insert features graphite end plugs for optimum reflection of neutrons originating in the plutonium-bearing sample. The top plug allows placement of a ^{252}Cf source in the center of the sample cavity for routine measurement-control checks. The passive cavity will accommodate a sample container up to 8.5 in. in diameter and 16 in. tall.

The neutron moderator is an annulus of polyethylene 4.75 in. thick and 31.5 in. tall with a 9.7-in. opening. Forty-eight ^3He proportional-counter tubes are placed in two circular rings of holes in the moderator body.

We used Monte Carlo[18] calculations to optimize the placement of tubes and the moderator thickness. This resulted in a calculated passive efficiency of 35.7% for a centered ^{252}Cf source and a calculated active (fast mode) efficiency of 31.0% for a centered source of ^{235}U fission neutrons. These high efficiencies provide very good sensitivities for both uranium and plutonium measurements.

The calculated axial efficiency profile for a ^{252}Cf source is shown in Fig. 32. This profile is equivalent in uniformity to that of the flat-squared counter,[58] which has the same cavity diameter, a single ring of 24 ^3He tubes, and a cadmium insert to flatten its profile.

Figure 33 shows the efficiency of the passive/active counter as a function of neutron energy for a centered point source. For comparison, the efficiency of the HLNC-II[59] is shown, as is the normalized emission spectrum for

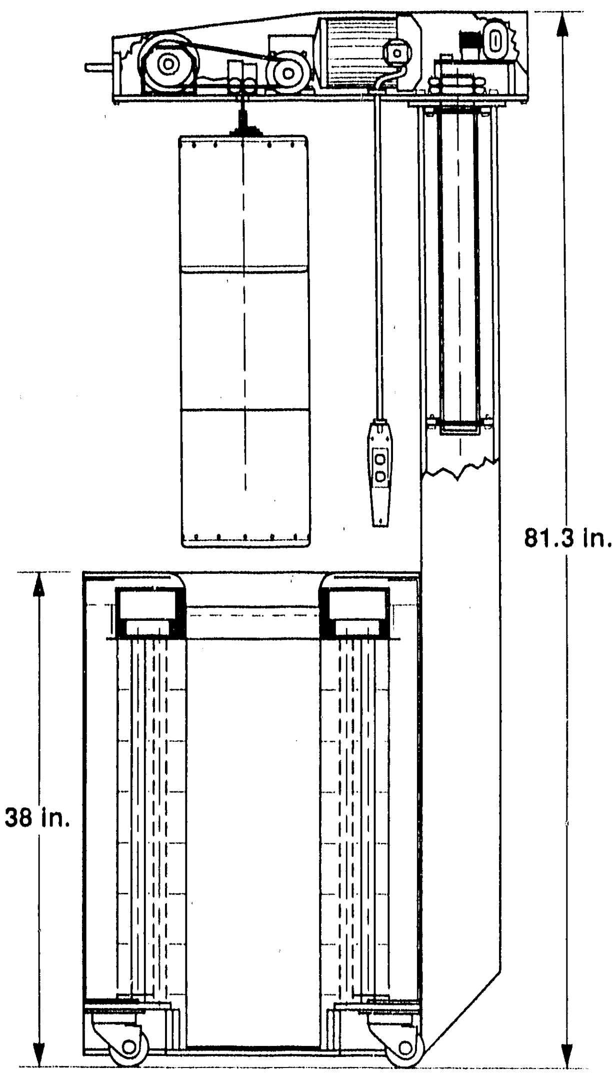


Fig. 30. Elevation layout view of passive/active neutron coincidence well counter (P/A NCC).

^{240}Pu. The profile of the new counter is much flatter than that of the undermoderated HLNC-II, and conforms to the shape of the ^{240}Pu emission spectrum quite well.

Intended routine uses for the counter include passive measurements of plutonium-bearing samples, active thermal-mode measurements of samples with low uranium concentrations (<50 g ^{235}U), and active fast-mode measurements of samples with high uranium concentrations (>50 g ^{235}U). Also planned is an evaluation of combined passive/active measurements of plutonium-bearing pyrochemical residues.

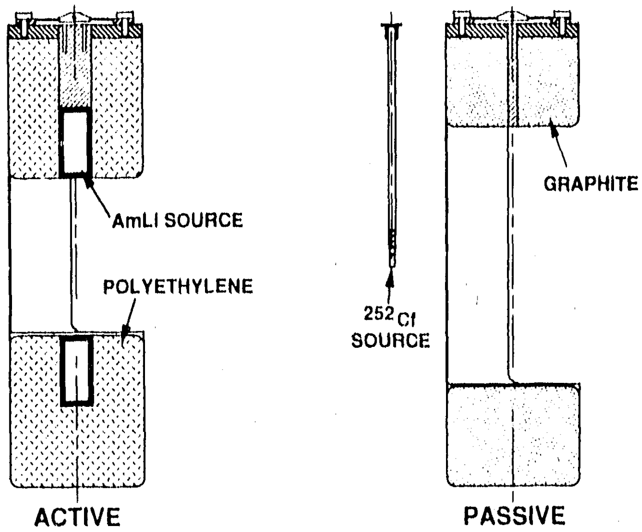


Fig. 31. Active and passive inserts for the P/A NCC.

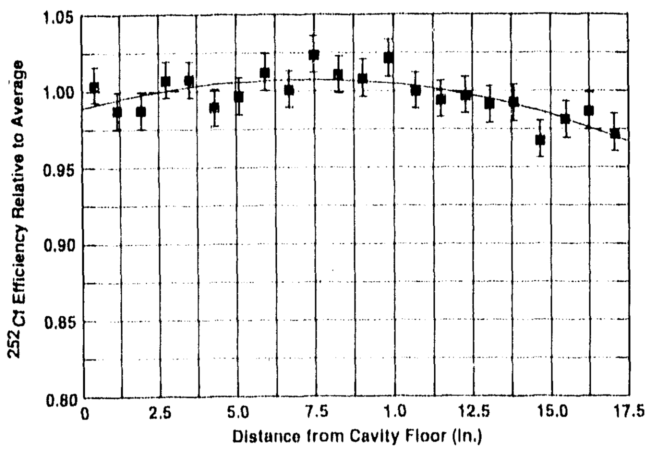


Fig. 32. The axial efficiency profile (in the passive mode) for the P/A NCC.

2. Assistance to Savannah River

a. Conceptual Design Report on NDA Instrumentation for the Transuranic (TRU) Waste Facility at the SRS (N. Ensslin and C. B. Bjork, N-1). A new TRU Waste Facility is being designed at the SRS that will accept exhumed TRU waste and process it for shipment to the Waste Isolation Pilot Project (WIPP). We have prepared a report that provides conceptual designs for neutron- and gamma-ray-based NDA instrumentation for this facility. The measurement categories considered were boxed waste, 55-gal. drums, in-process holdup, and certification of waste for shipment to WIPP. For each measurement category, we considered several different instrument options and recommended instruments and conceptual designs that, based on our understanding of the process, would be most useful to the facility.

For large incoming wood or steel boxes containing a wide variety of plutonium-bearing waste, we recommended an archway neutron monitor that measures the passive total neutron response and an archway gamma-ray monitor that measures the passive gamma-ray response. These archway monitors would straddle the conveyor line that brings the boxes into the facility, as shown in Figure 34 for the gamma-ray monitor. The monitors can be designed to have a detection sensitivity of about 100 nCi/g and reasonably flat efficiency profiles over the large waste boxes as they pass through. Segregating the incoming waste by point of origin and chemical compound, when possible, will improve the accuracy of the neutron monitor. The passive neutron and passive gamma-ray measurements will be used together to arrive at a weighted value for the assay and its uncertainty.

For the assay of incoming 55-gal. drums, we recommended two separate passive assay instruments: an NCC and an SGS, each large enough to accommodate 83-gal. drum overpacks. The passive neutron counter will have enough sensitivity to screen waste at the 100-nCi/g level if the counter is installed in a moderately well-shielded area. As in the case of boxed waste, the passive neutron and passive gamma-ray measurements will be used together to determine the drum loading and its uncertainty. The SGS assay also will be used to estimate the isotopic composition of the plutonium and to identify the presence of other isotopes that might affect the neutron assay.

For process holdup measurements, we recommended that the TRU Waste Facility acquire portable gamma-ray and neutron holdup monitors for use throughout the process area during periodic holdup measurement campaigns. We also considered a fixed gamma-ray or neutron holdup monitoring array that could provide more information and

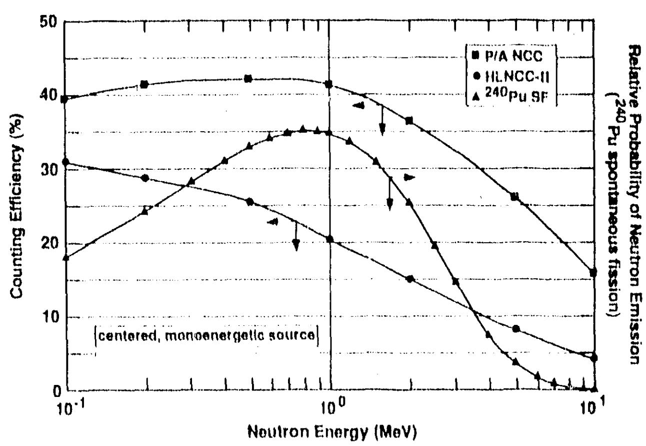


Fig. 33. Counting efficiency as a function of neutron energy for the P/A NCC and the HLNC-II. Also shown is the neutron emission spectrum for ^{240}Pu spontaneous fission.

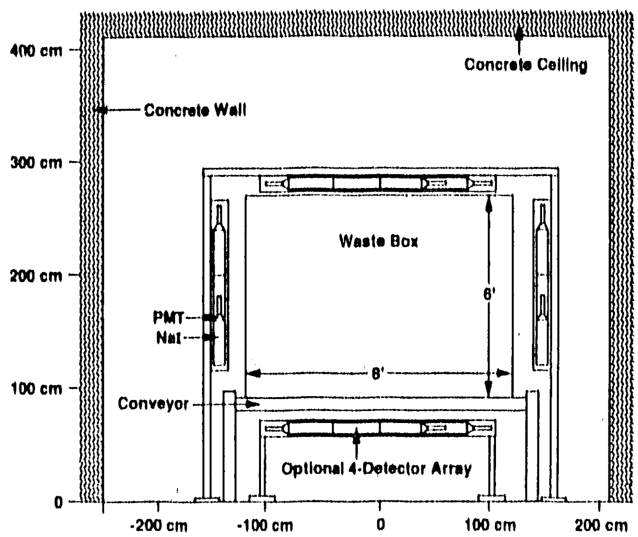


Fig. 34. Passive gamma-ray box monitor (elevation view) (section through passive NaI counters) (each NaI counter is 10 cm by 10 cm by 40 cm).

substantially reduce the need for manual holdup measurements, but is unlikely to be very accurate for the wide range of waste materials present throughout the facility. Assuming no waste segregation within the process area, gamma-ray measurements will be more quantitative and more useful, with the neutron measurements providing some assurance against large, clumped deposits.

For certification of waste for shipment to WIPP, we considered two options: scanning the waste in the process area before it is loaded into 55-gal. drums or standard waste boxes, and assaying drums or boxes after they are loaded. Each option has advantages or disadvantages in terms of assay accuracy, operator exposure, and the expected measurement accuracy as a function of waste segregation. For waste scanning, we recommended a fixed scanning setup with small neutron and gamma-ray archway monitors. The fixed geometry would decrease errors caused by source-to-detector distances, and would provide higher detection sensitivity and shorter counting times. For assaying output waste in 55-gal. drums or standard waste boxes, we recommended passive NCC and SGS as the most cost-effective, penetrating, and matrix-insensitive approach.

In the conceptual design report for each NDA assay option, we review the items to be measured, describe the assay technique and method of operation, describe mechanical and electrical features of the instrument, recommend appropriate electronics and software, estimate cost and schedule, and recommend a procurement route. Also for each option, we estimate the measurement accuracy, the detectability limit, the measurement control requirements, and the degree of waste segregation required.

b. Assay of Heterogeneous Scrap and Waste (J. K. Sprinkle Jr., G. E. Bosler, S. T. Hsue, M. P. Kellogg, M. Miller, S. M. Simmonds, and A. R. Smith, N-1). The assay of scrap and waste for plutonium content is difficult because the material to be measured usually is heterogeneous. Segregation of the scrap and waste into categories makes it somewhat easier to measure; however, development of suitable techniques has been hampered by the lack of appropriate standards for calibration and the evaluation of measurement bias. We now have characterized 25 scrap and waste items for plutonium content to 2% or better, covering three distinct material types—incinerator ash, sand slag and crucible, and MSE residues—all of which exhibit different problems to NDA measurement methods. We used these characterized scrap and waste items along with fabricated calibration standards (pure or diluted oxide) to evaluate two state-of-the-art NDA instruments: an advanced SGS and an NCC.[60] We demonstrated that these material categories can be measured with less than 5% bias, but caution users that each new category of scrap and waste requires evaluation for measurement bias. When both the SGS and NCC are available, comparing their assay results is a stringent test for measurement bias, because the two measurement methods are not sensitive to the same causes of bias. If the results agree, the user can be confident that the assay has little or no bias. Because these scrap and waste materials are more difficult to measure, we expect that these techniques will be applicable to many other scrap and waste categories.

c. Uranium Billet Shuffler (P. M. Rinard and K. E. Kroncke, N-1). We have completed the design of and begun fabricating a shuffler to assay the uranium content of billets before they are extruded into fuel tubes for the SRS production reactors. In addition to facilitating materials accounting, reactor safety will be enhanced because the fuel tubes will be assured to have the correct uranium content.

Special features of this shuffler include a billet loading device that forces the billet to be loaded with the proper orientation before the door can be closed and large shielding containers, which are basically cylinders into which shielding material will be poured to harden. The material contains boron for neutron absorption and eliminates the need for gamma-ray absorbing lead on the outside of the shield. This simple design has led to a shorter fabrication time and a great reduction in cost.

The billet shuffler should be installed by the fall of 1990.

d. Passive Drum Coincidence Counter for the SRS FB-Line (N. Ensslin, D. M. Miller, S.-T. Hsue, M. S. Krick, and E. Kern, N-1). We have developed a new passive NCC for assaying 55-gal. drums; the drum counter is now installed and operating at the SRS FB-Line. The specifications and conceptual design of the drum counter were prepared jointly by

SRS and Los Alamos, and the mechanical and electrical design and fabrication of the counter were carried out by Jomar Systems, Inc.* After the counter was fabricated, it was delivered to our laboratory for a regimen of performance measurements and acceptance testing, calibration, software installation and checkout, and preparation of documentation.

The new drum counter is the first passive NCC designed for the assay of 55-gal. drums in over 15 years. It incorporates several state-of-the-art features, such as fast Amptek preamplifiers/discriminators mounted inside the ^{3}He junction boxes for low deadtime and low electronic noise pickup, indicating desiccant holders mounted in the junction boxes, and a motor-driven door with built-in safety interlocks.

Figure 35 is a photograph of the drum counter showing the motor-driven door and the Jomar door controller box. The assay chamber is 71 by 71 by 96 cm and can easily hold a standard 55-gal. drum, which is loaded by sliding it along the fixed platform of rotating wheels. The counter has six banks of ^{3}He tubes -- one in each of the four sides and one on the top and bottom. The four vertical side banks each contain ten 91-cm-active-length ^{3}He tubes, and the top and bottom horizontal banks each contain ten 51-cm-active-length ^{3}He tubes. Ten Amptek Model A-111 preamp/discriminator boards are used to read the ^{3}He tubes, which permits counting at rates above 1.3 M counts/s (equivalent to 80 kg of PuO_2).

Each of the six banks of ^{3}He tubes is embedded in a 10.2-cm-thick slab of high-density polyethylene (CH_2) that is covered on all sides with 0.40 mm of cadmium sheeting. Each bank also is shielded on the outside with another 10.2-cm-thick slab of polyethylene. Within the six detector banks, the ^{3}He tubes are centered 4.16 cm from the inside edge of the polyethylene, which is slightly less than the distance of about 4.57 cm that gives "optimum moderation." A 0.23-cm-thick polyethylene liner that will be inside each waste drum will provide partial compensation for the difference, but the counter remains slightly undermoderated.

The FB-Line Drum Counter is operated from an IBM Personal System/2 Model 70 386 computer, has a color monitor for program operation, and has an IBM Proprinter II for hard-copy printout. The computer also is interfaced to a Jomar Model JSR-11 coincidence electronics package and to the custom-made Jomar door controller box. Data are collected and analyzed by a new menu-driven software package developed at Los Alamos. The software package provides a complete range of assay, measurement control, calibration, parameter modification, data storage, and data transfer options.

The new drum counter design has a good neutron detection efficiency of 15% and a moderate die-away time of 74 µs. The detectability limit for plutonium oxide (6% ^{240}Pu) at 3 standard deviations above background is given in Table XIII for totals and coincidence counting. The background rates used in the calculations are those actually present at SRS. Table XIII also gives the detectability limit in nanocuries per gram for a drum loading of 100 kg, showing that the counter can segregate waste at the 100 nCi/g fiducial.

The efficiency profiles inside the assay cavity are reasonably flat because of the uniform placement of ^{3}He tubes around the cavity. If these profiles are overlapped with the volume of a 55-gal. drum centered in the cavity, the average integrated response across the drum volume is within 1% of the response at the center. The average uncertainty in response caused by plutonium location is approximately ±5% RSD for totals and ±10% RSD for coincidence counts. Within the drum volume, the upper extreme is 8% higher than the response at the center, and the lower extreme is 17% lower than the response at the center.

To determine the effects of matrix materials on the coincidence response of the drum counter, we measured a californium source in the center of several drums filled with various matrix materials. The measured matrices were iron, raschig rings, polyethylene slivers, polyethylene shavings, polyethylene tubes, and polyethylene chunks. The total neutron response is within several percent of nominal for polyethylene (or water) loadings up to 0.1 g/cm^3.

These performance characteristics will make this type of counter useful for a wide variety of waste materials with either very low or very high plutonium loadings. At SRS, the instrument is used to measure the plutonium content of 55-gal. drums before shipment to a burial site. The drum counter measurement can be used either as the actual accountability measurement or as a final verification of previous values for safeguards purposes.

e. NDA Conceptual Design Report for the Pyrochemical Development Laboratory (T. K. Li, N. Ensslin, and G. Walton, N-1). The Pyrochemical Development Laboratory is being designed at Savannah River for research and development on pyrochemical treatment processes. The process will involve direct oxide reduction, molten salt extraction, electrorefining, pyroredox, hydriding, and salt scrubbing. We are preparing a conceptual design report on NDA instruments for feed, product, and waste assay. The instruments include a neutron multiplicity counter, an LCSGS, and a plutonium isotopic analysis system (ISO). Our report will provide recommendations for the instrument options, feasibility studies of a common detector for both the LCSGS and the ISO, and performance and cost estimates.

*Jomar Systems, Inc., 1143 18th St., Los Alamos, NM 87544 (505) 662-9811.

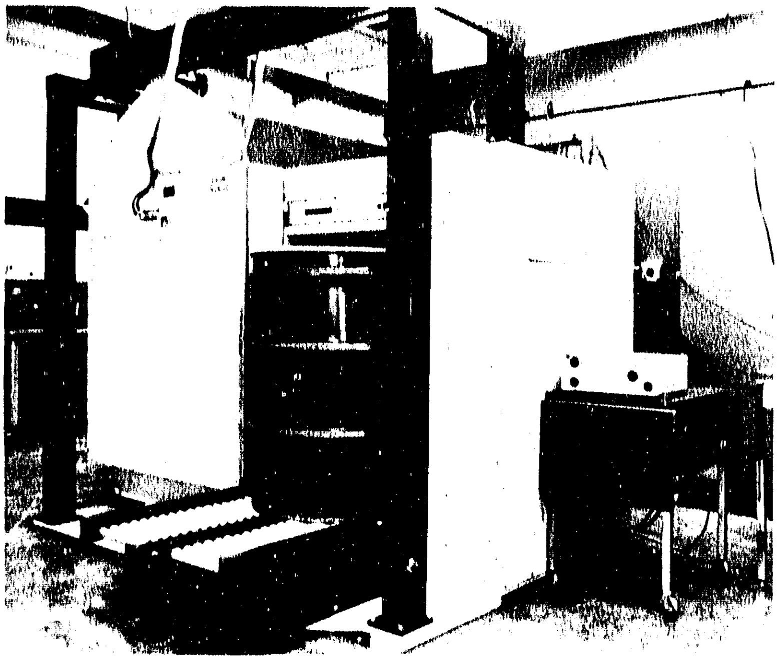


Fig. 35. Passive 55-gal. drum counter for the Savannah River Site FB-Line.

3. Assistance to the Portsmouth GDP

a. Californium-252 Shuffler for Measuring Enrichment Plant Scrap and Waste (J. K. Sprinkle, Jr., E. L. Adams, J. Baca, L. R. Cowder, H. R. Dye, D. C. Garcia, D. L. Garcia, C. R. Hatcher, E. C. Horley, H. O. Menlove, P. Polk, C. M. Schneider, and B. G. Strait, N-1). The uranium enrichment plant at Portsmouth is a large facility that generates commensurate quantities of scrap and waste. This scrap and waste might be contaminated with uranium enriched in ^{235}U to between 0.2% and 97% . We designed[61] and built a ^{252}Cf shuffler to assay 55-gal. drums of waste that typically contain less than 100 g ^{235}U. The design capitalized on our previous experience,[62] but we added the flexibility to use a dual-mode assay. The dual-mode assay interrogates the sample with thermal and epicadmium neutrons to change the ratio of the ^{235}U to ^{238}U. We plan to use the dual mode to correct the ^{235}U measurement for bias caused by the changing ^{238}U content.

We calibrated and tested the shuffler at Los Alamos,[63] then delivered it to Portsmouth, where it was installed in November 1989. Its performance agreed with our predictions based on Monte Carlo calculations,[18] at least for the limited testing that we were able to carry out. Because Portsmouth required prompt delivery of the system, we decided to finish developing the dual mode assay after delivery to Portsmouth. However, the partial evaluation at Los Alamos was sufficient to determine that this mechanical/electrical design is suitable for technology transfer. We are writing a specification package for commercial procurement of the hardware and plan to switch our programmatic emphasis from hardware design to software and data reduction techniques for measuring drums of uranium contaminated scrap and waste with shufflers.

b. Portsmouth Solution Enrichment System (T. K. Li, J. L. Parker, T. E. Sampson, L. R Cowder, E. C. Horley, and G. Walton, N-1). We are developing two identical automated uranium solution enrichment systems (SES's) for the NDA of both ^{235}U and total uranium concentration in solutions for the Portsmouth GDP. Each system will consist of two measurement stations and an automatic sample changer controlled by a single multichannel analyzer/computer system. One measurement station will assay ^{235}U concentrations by a passive gamma-ray technique and the

Table XIII. Detectability Limit Calculations for the 55-gal. Drum Counter for PuO_2 (6% ^{240}Pu)

Passive Technique	Assumed Background (counts/s)	Count Time (s)	^{240}Pu Limit (g)	Nanocuries/g in 100 kg
Totals	20	300	0.003	34
Coincidence	0.5	300	0.010	100

other station will determine the total uranium concentration by an XRF technique. The automatic sample changer will consist of one commercial laboratory robot with its controller, one bar-code reader, one sample rack, and one support shelf. An air-conditioned enclosure for the automatic sample changer will physically isolate the highly accurate and sensitive robot from the room. The robot will move samples among the sample rack, the bar-code reader, and the sample chambers of the measurement stations. The bar-code reader will identify the assay requirements of each sample by reading the sample ID and other information into the computer.

In the past several months, we have investigated whether an x-ray generator or a discrete source should be used for XRF, whether one head (single detector) or two heads (two detectors) should be used for passive gamma-ray and XRF measurements, whether one size or two sizes of sample container are necessary for both types of measurements, and which type of robot will satisfy our requirements. We also have been studying the extraction of peak areas of x rays without bias when they appear on background continuums of marked curvature that are very high relative to the x-ray peaks themselves. This condition holds when doing XRF measurements on low-concentration solutions of uranium, a condition that will be frequently encountered in the SES's. The problem is primarily one of accurately delineating the background continuum and predetermining the peak shape parameters with sufficient accuracy to fix many of them during the peak extraction process. We have been studying both problems with the aim of meeting the SES requirements for accuracy as well as determining the practical lower limits of sensitivity for this type of x-ray fluorescence instrument. We have not yet decided the final procedures and are continuing the development.

4. Assistance to the Idaho National Engineering Laboratory (INEL)

Test and Evaluation of a Uranium Monitor for Raffinate Lines (P. M. Rinard, C. M. Schneider, and E. L. Adams, N-1). We have completed a ^{252}Cf shuffler to monitor the uranium concentration in raffinate lines,[64] and it has passed the acceptance test. However, installation at the WINCO Idaho Chemical Processing Plant has been delayed while the plant upgrades its general facilities.

Installation will follow the scheme shown in Fig. 36, with most of the instrumentation in a clean corridor. Only the assay enclosure and some shielding will be mounted in the hot cell. A shielded plug inside a wall of the hot cell will carry electrical and mechanical cables.

The instrument will operate continuously for months at a time without a human operator. It will measure the ^{235}U concentration in fission-product waste streams and will generate warning and alarm signals at certain concentrations of ^{235}U set by the operator. Several features have been built into the monitor to assure the accuracy of the results during the long period of unattended operation.

- A special assay is taken with a sample of ^{235}U driven into the assay region. The frequency of this check is set by the plant operator. If this assay is outside the expected range of values,

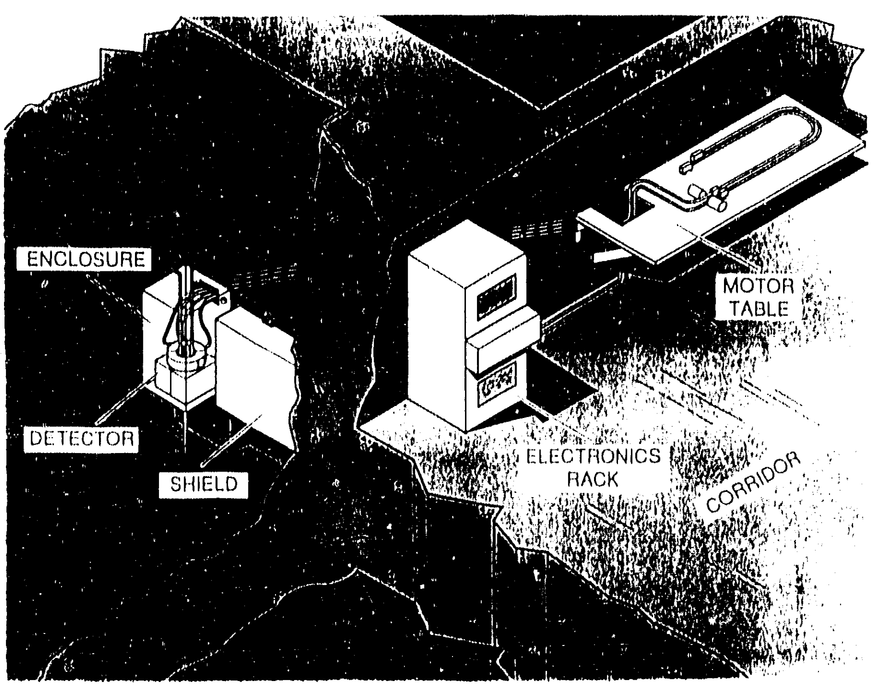


Fig. 36. The assay chamber of the instrument is mounted on a hot cell wall with a raffinate line entering from the bottom and leaving through the top. The guide tubes for capsules containing ^{252}Cf and ^{235}U pass through a plug in the wall into a corridor; electrical cables travel the same route. An electronics rack and stepping motors for the capsules are in the corridor.

subsequent assays are normalized, and the operator is warned about the anomalous condition.

- Background counts are taken at a different frequency (also set by the operator) and compared to an expected range of values. A rate outside the preset limits causes a warning to be issued, and the new background rate is used in the data analysis.
- The counts in two separate detector banks are compared as part of each assay. If the ratio of counts deviates from an expected value by more than a preset amount, a warning is issued to the operator. If one of the banks gives no counts, it is assumed that the bank is dead and the concentration is calculated on the basis of the counts in the good bank only. Should the count in the suspect bank of a later assay not be zero, it is assumed that the bank is once again functioning.
- Flux monitors may be used to compare the counts during the irradiation phase to the expected values. The monitors could detect changes in the nature of the liquid that would affect the neutron transport properties and hence the assay process. If the flux monitors are selected for use, adjustments to the count rates of delayed neutrons are made based on the flux monitor count rates.

All assay results are transmitted to a plant computer. Should no data from the monitor be received after a certain period of time, the plant computer will assume the monitor is inoperative and inform the plant operator.

The delayed neutron count rate is corrected for the decay of the ^{252}Cf source, the background, flux monitor variations (if selected for use), the flow rate of the raffinate through the monitor, and small variations in the times of the assay activities.

The correction for timing variations has always been inconsequential (less than 1%), but the flow rate correction is very important. The correction (Fig. 37) was deduced from measurements at 0.48 g/L with flow rates from 0 to 100 L/h. The most common flow rate is expected to be 80 L/h; if a calibration is performed with static liquids, uncorrected assays at 80 L/h would be in error by 14%. The monitor receives the current flow rate from a meter as part of each assay.

We developed a calibration curve using nitric acid solutions of ^{235}U with six concentrations and different flow rates. This curve is shown in Fig. 38; it is a straight line as expected for these low concentrations. Subsequent measurements with the 0.02 g/L solution indicated that the small bias at the origin may be caused by a small error in the flow-rate correction. All of these measurements will be repeated after the monitor is installed in the plant.

The precision of the monitor (Fig. 39) was determined from 137 consecutive measurements on a nitric acid solution with 0.034 ($\pm$0.002) g/L of ^{235}U. The calibration curve in Fig. 38 produced an average value of 0.0348

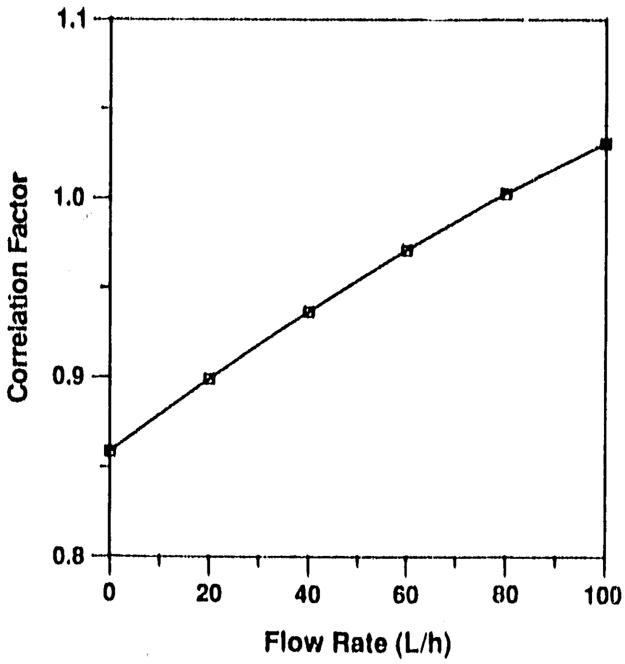


Fig. 37. This correction factor is applied to the delayed neutron count rates, according to the flow rate of the raffinate stream. The correction is relative to a rate of 80 L/h, the rate most commonly expected.

g/L with a standard deviation of 0.0031 g/L (or 9% RSD). Two other sets of similar data gave averages of 0.0341 and 0.0323 g/L with 10% and 11% RSD. The precision desired by the plant is 10% so it appears that this goal will be met.

B. Holdup Determination and Analysis for Uranium and Plutonium Processing Facilities—Assistance to Rocky Flats.

Quantifying nuclear materials holdup in processing equipment and piping continues to pose problems because of complex and uncertain geometries, difficult access, large and variable backgrounds, and lack of appropriate standards for calibration. The goal of this task is to demonstrate methods for measuring or otherwise estimating holdup, calibrating holdup instruments, and analyzing holdup data. It includes developing dedicated instruments for selected process equipment and portable instruments for *ad hoc* measurements, with test and evaluation (T&E) at operating facilities. Portions of some of the projects described here are supported by facility implementation funding.

1. Quantifying Solid Plutonium Holdup in High-Throughput Scrap Recovery Processes (P. A. Russo and R. Siebelist, N-1). We have made high-resolution gamma-ray measurements of plutonium holdup on the Rocky Flats calciner, and have documented new approaches to determining the very large

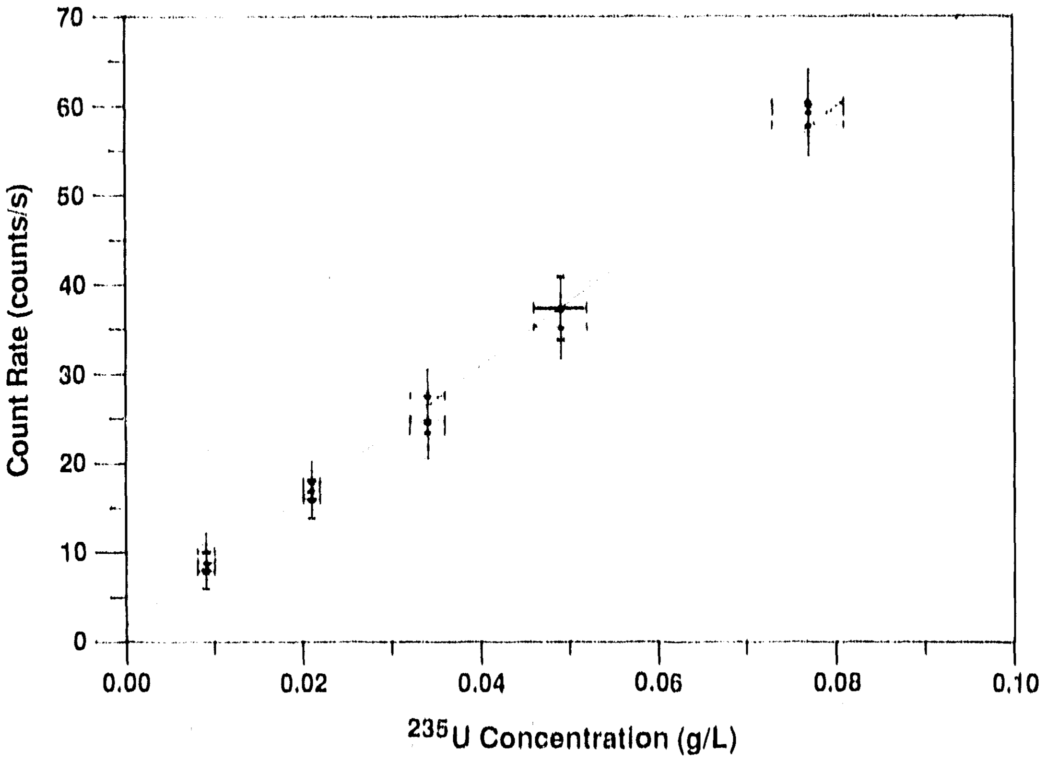


Fig. 38. A calibration curve was developed using nitric acid solutions of ^{235}U. Uncertainties in the concentrations are from chemical analyses. At each concentration there are corrected count rates from flow rates of 40, 60, 80, and 100 L/h. The concentrations of major concern are below 0.05 g/L; data points at 0.48 g/L were also taken but could not be included in this figure without obscuring the more important data points at lower concentrations.

attenuation of the solids bulk-processing equipment, along with the measurement results.[45,65] Recent experimental efforts that have significantly improved the correction factors for equipment attenuation have been extended to the Rocky Flats hydrofluorinator. Finally, we evaluated the use of a point transmission source for determining self-attenuation effects in large, localized holdup deposits for these applications.

Remeasurement of Attenuation Correction Factors for Calciner Equipment. We have documented the procedures and results of the original measurements of the correction factors for calciner equipment attenuation.[45,65] The original measurements were performed with a significant (~6-kg) amount of plutonium held up in the calciner process line; gamma-ray backgrounds from these deposits resulted in poor precisions for the measured transmissions used to obtain the correction factors, which, along with their relative uncertainties, are applied directly to the assay results for holdup. For this reason, we remeasured the correction factors following a cleanout and reassembly of the calciner. The remeasurements used ^{239}Pu gamma-ray peaks at seven (rather than the original five) different gamma-ray energies for each of the five calciner assay locations. The PuO_2 line source inserted into the calciner tube was used, as described previously[45,65] to perform the transmission measurements. In this case, positioning the line source at the radial center of the calciner tube improved the accuracy of the correction factors over those of the original measurements in which the line source was laid at the bottom of the 15-cm-diam tube. The remeasured equipment attenuation correction factors for the calciner, along with their improved (by a factor of 10 or more) precisions are given as CF_{EQ} (1σ) in Table XIV. Agreement (within 2 or 3σ for the original measurements) between the original values and the remeasured values is generally observed. Larger discrepancies are probably the result of the different positioning of the transmission source for the two sets of measurements.

The magnitudes of the CF_{EQ} values can be compared to the calculated minimum correction factors determined from the gamma-ray attenuation of the known uniform components of the calciner structure. These known components, described previously,[45] include the calciner tube that extends over the full length of the calciner and the heat shield that covers the heater section of the tube. Table XIV also lists these minimum expected equipment attenuation correction factors, $CF_{EQ,MIN}$, and gives the ratios of the measured to the minimum expected values for

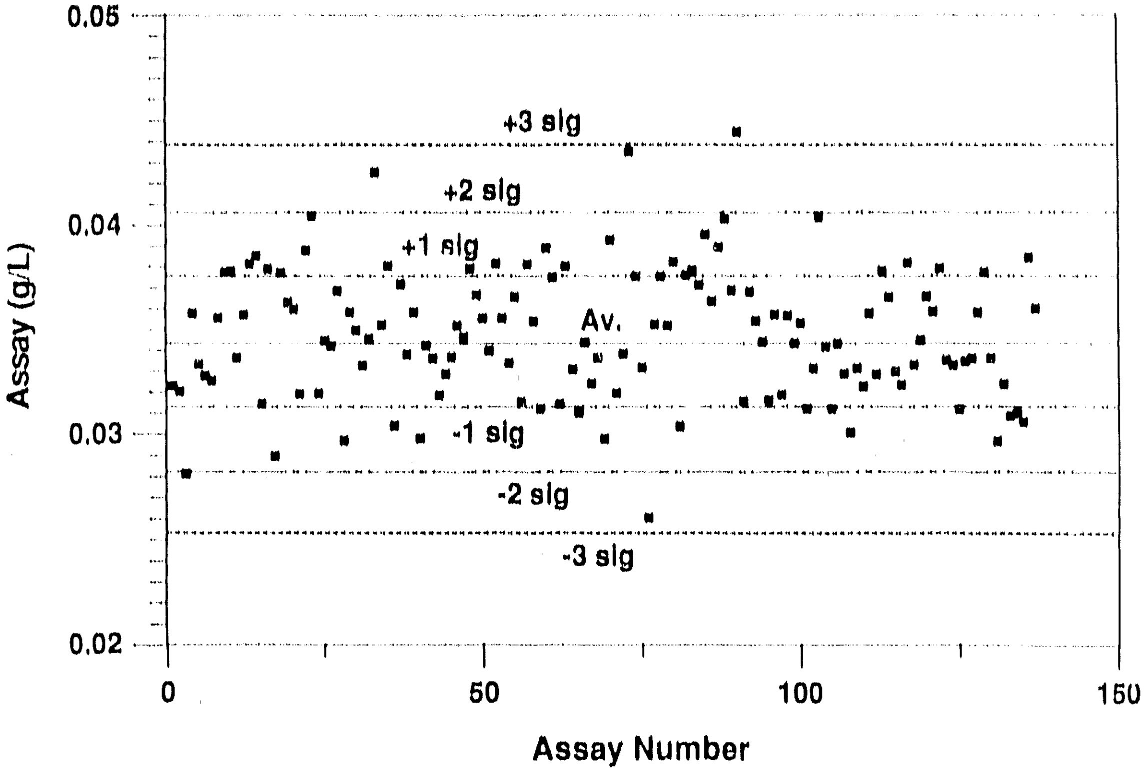


Fig. 39. With a solution of 0.034 g/L flowing through the instrument at 80 L/h, we took 137 consecutive assays. The sequence of results is shown along with the average value and multiples of the standard deviation.

TABLE XIV. Calciner Equipment Attenuation Correction Factors

Calciner Assay Location	Heater (H) or Tube (T) Section	CF_{EQ} (1σ) [$CF_{EQ,MIN}$] $CF_{EQ} + CF_{EQ,MIN}$ (1σ)						
		129.3 keV	203.5 keV	345.0 keV	375.0 keV	413.7 keV	451.5 keV	646.0 keV
1	T	9(1) [6.85] 1.3 (0.2)	4.5 (0.4) [3.16] 1.4 (0.1)	4.20 (0.13) [2.34] 1.79 (0.06)	4.23 (0.06) [2.26] 1.87 (0.03)	4.09 (0.05) [2.17] 1.88 (0.02)	3.97 (0.18) [2.11] 1.88 (0.09)	5.4 (1.8) [1.88] 2.9 (1.0)
2	T* (and H)	18(6) [>6.85] <2.6 (0.8)	9.0 (2.0) [>3.16] <2.8 (0.2)	6.01 (0.27) [>2.34] <2.57 (0.08)	5.69 (0.10) [>2.26] <2.52 (0.03)	5.35 (0.08) [>2.17] <2.46 (0.03)	5.22 (0.31) [>2.11] <2.47 (0.10)	5.3 (0.8) [>1.88] <1.7 (0.4)
3	H	90 (130) [53.9] 1.7 (2.4)	37 (32) [18.4] 2.0 (1.7)	13.35 (2.10) [9.99] 1.33 (0.15)	12.24 (0.51) [9.22] 1.33 (0.06)	11.50 (0.39) [8.42] 1.37 (0.03)	10.22 (1.19) [7.81] 1.31 (0.15)	9.5 (5.7) [5.77] 1.6 (1.0)
4	H** (and T)	79 (69) [<53.9] >1.5 (1.3)	24 (8) [<18.4] >1.3 (0.4)	13.61 (1.01) [<9.99] >1.36 (0.10)	12.46 (0.33) [<9.22] >1.35 (0.04)	10.87 (0.23) [<8.42] >1.29 (0.03)	9.78 (0.80) [<7.81] >1.25 (0.10)	10.5 (6.1) [<5.77] >1.9 (1.0)
5	T	11(1) [6.85] 1.6 (0.2)	4.2 (0.3) [3.16] 1.3 (0.1)	3.52 (0.08) [2.34] 1.50 (0.03)	3.40 (0.03) [2.26] 1.50 (0.01)	3.22 (0.02) [2.17] 1.48 (0.01)	3.34 (0.11) [2.11] 1.58 (0.05)	3.8 (0.8) [1.88] 2.0 (0.4)

*Tube CFEQ,MIN values will underestimate the minimum corrections at this location.
**Heater CFEQ,MIN values will overestimate the minimum corrections at this location.

the seven gamma-ray energies at each assay location. The ratios are greater than one because the calciner structure also includes numerous bulky flanges, sleeves, shields, and brackets mounted along the calciner tube, which are not taken into account in computing the $CF_{EQ,MIN}$ values. The ratios are typically between 1 and 2, an entirely reasonable result for the calciner structure.

Extension of Measurements to the Hydrofluorinator. The Rocky Flats hydrofluorinator is similar in design to the calciner. The greater overall length (3.5 m compared to 2.3 m for the calciner) increases the number of measurement locations required for experimentally determining holdup. Figure 40 is a simplified drawing of the hydrofluorinator hardware showing the tube and heat shield with various major components labelled. The feed portion of the hydrofluorinator (shown by the dashed line) is removed during the holdup measurements, as it is for the calciner.

Figure 41 shows the hydrofluorinator assembly with (superimposed) glove ports, projected from the operations side of the glove box, where the detector is positioned for the holdup measurements. The assay measurement glove ports are marked with a square grid. For holdup measurements, we recommend that background spectra be obtained at the six upper assay ports (on the heater section) with the detector tilted upward to exclude the hydrofluorinator tube from the detector field of view. For the measurements of the CF_{EQ} values, the background measurements are performed in the same geometry as for the transmission source measurements except that the source is removed (as for the calciner CF_{EQ} measurements).

We measured the CF_{EQ} values with the PuO_2 line source positioned at the radial center of the hydrofluorinator tube by a mechanical fixture similar to that used in the calciner exercises. We obtained results for the seven gamma-ray energies at each of the nine hydrofluorinator assay locations. The measurements were performed after the hydrofluorinator was cleaned out and reassembled. The resulting CF_{EQ} and 1σ values are given in Table XV for locations 1 through 9, corresponding to right-to-left numbering of the nine hydrofluorinator assay locations shown in Fig. 41. The calculated minimum correction factors for equipment attenuation, $CF_{EQ,MIN}$, analogous to those for the calciner, also are given in Table XV, along with the ratios of the measured correction factors to the calculated minimum correction factors for the seven gamma-ray energies at each assay location. Overall, the magnitudes of the CF_{EQ} values and the ratios are similar to the calciner results at similar locations.

Point Source Transmission Measurements for Determining Self-Attenuation Effects. The bottoms of the Teflon bellows at the feed ends of both the

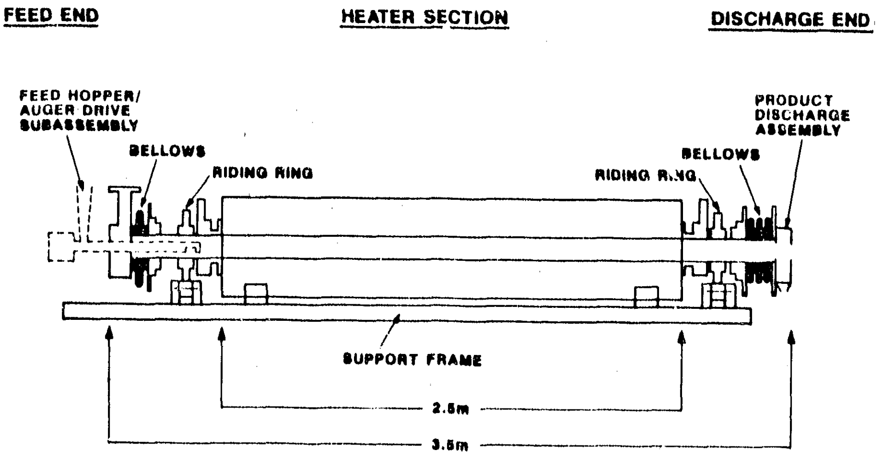


Fig. 40. Simplified drawing of hydrofluorinator assembly (approximately to scale).

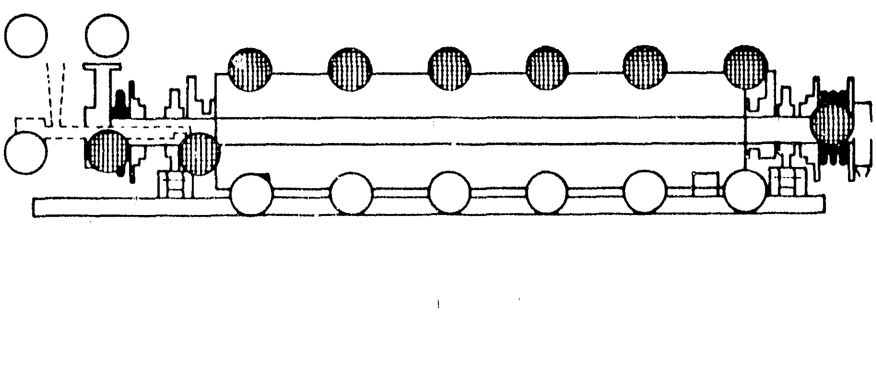


Fig. 41. Hydrofluorinator assembly with outlines of glove ports projected from the operations side of the glove box. The gridded circles correspond to the glove ports at which the detector is positioned for holdup measurements. The measurements of the equipment self-attenuation correction factors were performed at these nine locations.

TABLE XV. Hydrofluorinator Equipment Attenuation Correction Factors

Hyrdro-fluorinator Assay Location	Heater (H) or Tube (T) Section	CF_{EQ} (1s) [$CF_{EQ,MIN}$] $CF_{EQ} + CF_{EQ,MIN}$ (1s) 129.3 keV	203.5 keV	345.0 keV	375.0 keV	413.7 keV	451.5 keV	646.0 keV
1	T	8.46 (1.59) [6.85] 1.2 (0.2)	5.27 (0.61) [3.16] 1.7 (0.2)	4.36 (0.14) [2.34] 1.86 (0.06)	4.04 (0.05) [2.26] 1.79 (0.02)	3.95 (0.05) [2.17] 1.82 (0.02)	3.83 (0.20) [2.11] 1.82 (0.09)	2.20 (0.47) [1.88] 1.2 (0.3)
2	T, H* (~equal)	-308 (1231) [12.1] -25 (101)	21.9 (7.6) [5.39] 4.1 (1.4)	10.9 (0.6) [3.79] 2.9 (0.2)	10.1 (0.2) [3.63] 2.78 (0.06)	9.28 (0.17) [3.45] 2.69 (0.05)	7.60 (0.58) [3.32] 2.3 (0.2)	4.8 (1.5) [2.84] 1.7 (0.5)
3	H	-103 (91) [53.9] -1.9 (1.7)	20.5 (4.7) [18.4] 1.1 (0.3)	16.9 (0.5) [9.99] 1.69 (0.05)	14.2 (0.3) [9.22] 1.54 (0.03)	12.7 (0.3) [8.42] 1.51 (0.04)	12.9 (1.4) [7.81] 1.7 (0.2)	9.6 (5.8) [5.77] 1.7 (1.0)
4	H	-308 (923) [53.9] -6 (17)	12.7 (20.3) [18.4] 0.7 (1.1)	13.6 (0.9) [9.99] 1.36 (0.09)	12.3 (0.3) [9.22] 1.33 (0.03)	10.8 (0.2) [8.42] 1.28 (0.02)	10.4 (0.9) [7.81] 1.33 (0.11)	6.9 (3.0) [5.77] 1.2 (0.5)
5	H	308 (1026) [53.9] 6 (19)	25.4 (8.2) [18.4] 1.4 (0.4)	13.7 (0.9) [9.99] 1.37 (0.09)	11.9 (0.3) [9.22] 1.29 (0.03)	10.7 (0.2) [8.42] 1.27 (0.02)	8.94 (0.80) [7.81] 1.14 (0.10)	6.0 (2.3) [5.77] 1.0 (0.4)
6	H	3077(11281) [53.9] 57 (209)	42.4 (28.3) [18.4] 2.3 (1.5)	13.4 (0.9) [9.99] 1.34 (0.09)	13.2 (0.4) [9.22] 1.43 (0.04)	11.7 (0.2) [8.42] 1.39 (0.02)	13.5 (1.8) [7.81] 1.73 (0.23)	9.6 (7.7) [5.77] 1.7 (1.3)
7	H	-54 (45) [53.9] -1 (1)	57.8 (63.1) [18.4] 3.1 (3.4)	14.9 (1.5) [9.99] 1.49 (0.15)	14.5 (0.5) [9.22] 1.57 (0.05)	12.9 (0.4) [8.42] 1.53 (0.05)	12.2 (1.9) [7.81] 1.56 (0.24)	4.4 (2.0) [5.77] 0.8 (0.3)
8	T** (plus H)	19 (6) [>6.85] $<$2.8 (0.9)	11.0 (2.3) [>3.16] $<$3.5 (0.7)	8.18 (0.44) [>2.34] $<$3.5 (0.2)	7.05 (0.07) [>2.26] $<$3.12 (0.03)	6.74 (0.11) [>2.17] $<$3.11 (0.05)	6.99 (0.65) [>2.11] $<$3.3 (0.3)	4.8 (2.4) [>1.88] $<$2.6 (1.3)
9***	T	8.46 (1.59) [6.85] 1.2 (0.2)	5.27 (0.61) [3.16] 1.7 (0.2)	4.36 (0.14) [2.34] 1.86 (0.06)	4.04 (0.05) [2.26] 1.79 (0.02)	3.95 (0.05) [2.17] 1.82 (0.02)	3.83 (0.20) [2.11] 1.82 (0.09)	2.20 (0.47) [1.88] 1.2 (0.3)

*Assume that $CF_{EQ,MIN} = 2\left(\frac{1}{CF_{T,MIN}} + \frac{1}{CF_{H,MIN}}\right)^{-1}$

**Tube $CF_{EQ,MIN}$ values will underestimate the minimum corrections at this location.

***Results for location 1 are used here because of the possibility that the background at location 9 was measured incorrectly.

calciner and hydrofluorinator have traditionally accumulated large amounts of solid plutonium during operation. This material is outside the calciner tube (and hence the gamma rays are immune to the large equipment attenuation effects), but is subject to large gamma-ray self-attenuation effects. We investigated the use of a 10 mCi ^{137}Cs gamma-ray source to measure the transmission of 662-keV gamma rays through these localized deposits of plutonium to subsequently determine the gamma-ray self-attenuation effects of the deposits with additional measurements performed on the cleaned-out calciner and hydrofluorinator.

The ^{137}Cs point source was mounted on a thin rod 1 m in length, which was inserted into a glove on the side of the glove box opposite the detector. The source was positioned in the glove fingertip and the fingertip was positioned to touch the bottom of the Teflon bellows at the alleged location of the large holdup deposits, with the deposit located between (and in line with) the source and the detector. The alleged deposit location was centered in the field of view of the detector. The 662-keV transmission measurements were repeated three to five times; the ^{137}Cs source was repositioned between measurements. At the calciner, the standard deviations in the 662-keV count rate for the multiple measurements were 4.5% and 5.3% (1σ) at the feed end and discharge end bellows, respectively. At the hydrofluorinator, the ability to position the ^{137}Cs source was severely constrained by the equipment geometry inside the glove box and by the more limited access space outside the glove box. The corresponding precisions for the hydrofluorinator were 22% and 24% (1σ), respectively.

This approach to obtaining corrections for gamma-ray self-attenuation effects does not give a reliable result because we lacked the hardware to reproduce the position of the transmission source. We are examining the energy dependence of the holdup assay in an effort to correct for these effects.

2. Assistance with Measurements for Plutonium Holdup (P. A. Russo, D. C. Garcia, J. A. Painter, and R. Siebelist, N-1). Many of the techniques that we have developed for portable neutron and gamma-ray holdup measurements have been first tested at the RFP before implementation elsewhere in the DOE community.

Recent heightened interest in performing portable holdup measurements on process equipment in many locations (for a variety of reasons including accountability, safety, materials control, and process control) has resulted in significant efforts to transfer our newest holdup measurement technology to RFP during the past year. This has included:

- obtaining, documenting, and transferring experimental results and procedures required for quantitative measurements of holdup in the Building 771 calciner and hydrofluorinator;[45] (see also Part 3, Section B.1)
- documenting the mechanical design of the portable holdup "cart" or "tree"[46] and transferring the design along with detailed photographs of the equipment to RFP;
- providing exclusive laboratory training (at Los Alamos) and documentation to Rocky Flats personnel on the principles and procedures for the generalized-geometry holdup calibration and assay and on the design of the software that automates the measurements;
- providing an executable software package written in Microsoft Quick Basic 3.0 for automating the generalized-geometry holdup calibration and assay;[47] (This is the same software package developed by Los Alamos and used for the Building 771 calciner and hydrofluorinator measurements in 1988 and 1989. The source listings with comments also were provided for documentation.)
- providing procurement specifications for the commercial (some customized) hardware (detectors, electronics, and some mechanical components) required to perform the automated generalized-geometry gamma-ray holdup measurements; and
- providing the electronics and mechanical design package, procurement information, and performance data on the recently commercialized Los Alamos compact NaI detectors.

As the new software (see Part 1, Section E.2) is tested at Los Alamos, more personnel training will be required as a result of transferring this more flexible and user-friendly program to RFP.

C. Improved Materials Accountability for Bulk Uranium and Plutonium Processing Facilities—Assistance to Los Alamos.

The goal of this task is to test and demonstrate the advanced materials accountability methods and software developed in the base technology tasks in operating facilities. It includes the development, demonstration, and T&E of improved measurement control and calibration procedures, variance propagation, and inventory difference evaluation, as well as automated PC-based materials accountability systems that can be used by small facilities or in a single MBA. Some of the projects described here were supported in part by facility implementation funding.

T&E of Automated Error-Detection and Analysis Software (H. S. Vaccaro, N-4; B. Hoffbauer, L. McGavran, C-9). This year our R&D on automated anomaly detection systems for safeguards and security focused on seven areas:

- human-entered rules,
- multiple knowledge domains,
- analysis and replacement of scoring algorithms,
- user accessible tuning,
- software robustness
- portable data communications, and
- events and threads subsystem design,

We met our objectives; new concepts were developed, designed, and tested in each area. The most important and successful improvements were in human-entered rules, multiple knowledge domains, and new scoring algorithms. Both the portable data communications software and the events and threads subsystem design efforts proved more difficult than we expected, but they have been completed.

Each of the areas is discussed further below. We briefly explain the problems the R&D addresses and summarize the progress.

Human Entered Rules--*Problem*. Most research and development teams working on automated anomaly detection have used one of two approaches thought to be alternatives: detection of anomalies with respect to expert rules or detection of departures from statistical ranges derived from normal activity. In fact, neither approach is sufficient for most applications. Both expert rules and historically normal activity patterns are needed for anomaly detection (as well as other information discussed under multiple knowledge domains).

The anomaly detection team for the Stanford Research Institute Intrusion Detection Expert System project is now working on a system that incorporates both approaches; however, the two techniques are not integrated. Our W&S anomaly detection system has always been a hybrid; it uses rules similar to the expert rules in some systems, but the rules have been automatically generated from historical data like the ranges in the statistical approaches. Thus, its rules were human-readable but not human-modifiable.

The seriousness of this shortcoming became obvious during two tests of W&S at Los Alamos. Working with TA-55 nuclear materials inventory data showed that many of the anomalies that the staff wished to detect with W&S resulted from changed accounting practices, such as designation of new materials and project codes. Because many of the current inventory items used the old codes, W&S generated rules indicating that the codes were normal. Similarly, in working with audit data from the Los Alamos Network Security Controller, W&S categorized common but nonetheless incorrect activity as "uninteresting" when it generated rules including those activities regarded as historically normal. (Note that examination of the rules generated by W&S quickly identified many of these problems. We learned that there is great value in studying such automatically generated rule bases) Though W&S generated very good rule bases about historical activity patterns, additional human input was clearly needed.

Solution. We developed and tested W&S capabilities to examine, print, add, modify, and load plain text rules into W&S-generated rule bases. We further extended W&S's rule syntax to accept a negation operator (the complement of a list of values in a rule), to detect rule conflicts and side-effects, and to perform requested rule merging and replacement. These features are fast, reliable, and easy to use; they met all of our objectives in enhancing the power of W&S. However, they uncovered a new dilemma described in the next section.

Multiple Knowledge Domains—*Problem.* If a rule base contains both expert and historical rules, it co-mingles two different "knowledge domains." When we detect an anomaly against such an "impure" rule base, we can no longer say whether the anomaly was abnormal, or in violation of expert knowledge, or perhaps both.

In a typical application there will be numerous sources of information on what to look for when detecting anomalies. Some examples are

- policy (for example, no SNM will be left in area Z overnight);
- expert knowledge (for example, material W should be measured by calorimetry);
- physical constraints (for example, glovebox 172 is not connected to 171);
- administrative information (for example, person X works half days, Z is uncleared);
- recent historical activity (for example, what was normal over the past 60 days); and
- longer-term historical activity (for example, what was normal over the past two years).

Our experimentation with rule conflict resolution and impure rule bases showed that each information source should be kept separate and be updated independently.

Solution. We redesigned W&S to handle an arbitrary number of rule bases, together with all associated information (such as data value dictionaries, tuning parameters and detection thresholds, data mapping functions, comments, etc.). We refer to this collection of information as a knowledge domain. Each knowledge domain can be loaded and applied independently, and each produces its own assessment of an audited event.

Thus, an MC&A materials transfer event may be found to be consistent with "longer-term historical activity," but in violation of "policy." "Expert" rules might indicate that an activity could be data leakage, and "administrative" information might confirm that the user's account is supposedly revoked.

All software to handle multiple knowledge domains was written, debugged, and tested this past year. We believe the capability is extremely important, and it is an innovation for automated anomaly detection systems. The downside is that multiple knowledge domains add to the complexity of an anomaly detection system and can produce more assessments of an event than a human can readily digest.

As a result, we are working on concepts for resolvers that will further digest anomaly indications from multiple domains and (a) present the results in a more palatable form and (b) control the collection and analysis of new events through a feedback loop. If successful, this will lead to an integrated analysis of data streams relevant to safeguards or security. The feedback loop will ensure that the monitoring and analysis system is not overwhelmed and that it adapts its analysis and data collection to rising levels of concern. The additional processing should ensure that humans (or machines) tasked with interpreting anomaly indications are not desensitized by a large amount of numeric scoring data.

Analysis and Replacement of Scoring Algorithms—*Problem.* W&S's scoring algorithms apply a rule base to an event and compute a numeric figure of merit for each event feature (field), for the event as a whole, and for a thread of activity related to the event. The quality of these algorithms directly affects the rate of false and missed anomaly indications.

This year we made significant progress on analyzing W&S's scoring functions and suggesting improvements. Because W&S's rules are based on a finite amount of historical data (which itself includes unidentified anomalies) or upon uncertain human expertise, its rules are uncertain and sometimes conflicting. W&S has always been designed to accommodate this real-world situation. As a result, W&S scoring algorithms were based more on art than solid mathematical foundations. However, that is changing.

Solution. After analyzing the behavior of W&S's scoring method, we made two major changes. We now use only the deepest applicable rules (those with the most left-hand-side conditions) to reduce rule interdependence. And the figure of merit for each event feature is now computed by means of a binomial probability model consistent with the way rules are derived from historical data.

User Accessible Tuning—*Problem.* Beta software testing re-emphasized the need for user accessible tuning of the rule base generation process. For many historical data sets, the default pruning parameters overgenerate rules. (That is, W&S generates an inappropriately large number of instantiated rules.) We have long sought to achieve fully automated tuning, but so far have not devised a method for making W&S self-tuning.

Solution. W&S has always contained a rich set of internal tuning parameters. We added additional parameters this year, significantly enhancing control over rule base generation and anomaly detection. These adjustments can now be made from the user interface at run time and are supported by help screens. Parameter settings for each knowledge domain are remembered across W&S sessions.

Software Robustness—*Problem.* W&S is intended to support a wide variety of R&D features and also operate at client sites for extended testing and evaluation. We previously kept its user interface and data input/output functions primitive, in the belief that software changes to core algorithms would be easier if the input/output functions were minimal. An unavoidable side effect was that W&S was difficult for those outside the R&D team to use. Furthermore, as we added features to W&S, such as human-entered rules and multiple knowledge domains, it became increasingly difficult for even experienced team members to avoid loading incorrect data into W&S.

Solution. We have rewritten much of W&S's user interface. It is now consistent throughout all menus and fully supports the user through help screens for every possible user input. Inputs and outputs now take place through scrollable windows, unless the information is known to fit on a single line.

In addition, we implemented new file input and output routines for W&S data, along with comprehensive data sanity checking routines. These have eliminated all problems with loading corrupt or inconsistent data into W&S. The user, whether naive or experienced, no longer crashes software due to "user errors." However, it does indeed take us much longer now to change W&S because of the amount of user-interface and file-input/output code affected by feature changes.

Portable Data Communications—*Problem.* As we applied W&S to various MC&A and computer security audit trails, we repeatedly ran into the same problem—how to move data from the source to the computer running W&S and where to store the huge quantities of historical data. Each test seemed to use different hardware and software protocols. Furthermore, security or performance issues kept us from running W&S on the computers generating the data.

Solution. We developed a portable communications software module, optimized for moving data from various source computers to a Unix host computer running W&S. The software uses either Ethernet or serial port hardware and software. Configuration parameters invoke commonly available encryption algorithms when necessary. (We are not currently using this capability.)

Overall, this module is important to our test and evaluation efforts, and we will continue to develop it. In particular, we want to improve its performance and add needed features for record queue control and status reporting. The communications module is an essential component of the multi-audit-trail, multi-host, networked anomaly detections system that we are currently developing.

Events and Threads Subsystems Design—*Problem.* The W&S design differed from many other approaches by analyzing individual transactions. We created the concept of "threads" and "thread members" to allow W&S to aggregate data, such as the number of file access errors in a user's computing session. Threads are a very powerful concept, and allow W&S to handle fine-grained analysis (which most other systems cannot) and coarse analysis at the same time. But W&S's thread design was just a proof-of-concept and was not adequate for continued R&D.

Solution. We have now completely redesigned transaction handling in W&S. Transactions now generate what we call events that in turn have threads and thread methods that operate on the data in both the event and the thread member data structure to produce a class feature vector. It contains state and state change data. The feature vector is what a knowledge domain describes, so rule bases analyze the contents of the feature vector.

Several different formats of incoming transactions might generate the same event in a network of process control and MC&A computers. The event will be handled in the same way, regardless of its source. Furthermore, a single transaction might generate more than one event for W&S to analyze. Finally, each event can have several threads, each creating a feature vector, and those feature vectors can be analyzed by multiple knowledge domains.

When the new design is fully implemented and tested, W&S will be capable of beginning the task of integrated analysis of nuclear materials control and accountability data and of networked computer security data. The systems integration task will eventually require a "back-end" data archiving and retrieval capability (to store both "raw" event data and analysis results), but we expect to make significant progress using the software we are now completing.

D. Demonstration of Safeguards System Integration and Evaluation.

The objective of this project is to demonstrate integrated systems technology developed under base technology tasks at small and large production facilities. It emphasizes the integration of materials control, materials accounting, physical protection, and process monitoring through efficient acquisition, organization, and analysis of safeguards and facility information.

1. Assistance to Savannah River

a. Enhancements to NSR Software (G. L. Barlich, N-4). The Instrument Control Function (ICF) at the NSR facility integrates data management and collection for three rooms of instruments in the process area and connects process control, accountability, and vault computers via a network. The Sample Assay Room provides wet process control analysis, the Feed Assay Room provides feed preparation measurement, and the Receipts Assay Facility Laboratory supports vault area operations. The integrated measurement system was a joint project among Los Alamos, EG&G Mound Laboratory, Lawrence Livermore National Laboratory, and WSRS and was delivered in its first phase in 1986.

Minor additions were made to the ICF software to enhance the tracking of standards in the feed preparation and vault areas. System-wide testing with process control and vault operations was successful, and we have completed the software design manual[66] for this system. In 1990, a minor upgrade is planned to accommodate more instruments in the vault area. We also are assisting Westinghouse personnel with a proposal to replace the Digital VAX 750 computer with a faster, more expandable machine without major software changes.

b. Application of NSR Software at Savannah River F Area (W. J. Whitty, R. C. Bearse, R. D. Sutherland, R. M. Tisinger, and R. S. Leonard, N-4). The NSR facility at Savannah River is a highly automated plutonium dissolution/ purification process incorporating computer-controlled operations, on-line instrumentation, and an NDA laboratory. The facility is designed to process plutonium-contaminated scrap and transfer the product (plutonium nitrate solution) and waste to F Canyon.

The NSR facility includes an extensive network of on-line and laboratory-based NDA devices. Most of these microprocessor-controlled devices are interfaced directly to a network of larger computers that acquire, transmit, and archive data related to the amounts and locations of nuclear materials within the NSR facility. The NSR computerized Accountability Function (ACF) is the primary means for providing near-real-time accountability. In its simplest abstraction, the ACF works as follows: data about process operations (such as tank volumes, concentrations, and temperatures) are received either from the ICF or manually as messages. Computer programs on the ACF process these data into transactions. These are handled by a transaction processing system, which handles shipper-receiver differences and updates the book and physical inventories. The transactions and related message operations are the front-end of the entire accountability system. The back-end produces reports and handles Explainable Inventory Differences and Book-to-Inventory Differences.

We have designed and developed a prototype accounting system for the ACF. The major elements of the accounting system software are (1) a real time, data acquisition subsystem that captures automated and manual messages, (2) a message-processing subsystem that combines related messages and updates inventory and related data, (3) a variance-calculation subsystem for calculating material balances and their uncertainty, and (4) an interface subsystem that selects measurement data and writes an input file for use by the variance-calculation subsystem, which employs the variance propagation code MAWST (Materials Accounting with Sequential Testing).

In a parallel effort, Westinghouse personnel also have developed accounting software, the Nuclear Materials Accounting System (NUCMAS), which replicates many of the functions in the Los Alamos accounting program but lacks any facility for calculating the variance-covariance matrix of a sequence of material balances or for testing such sequences for evidence of material loss.

At the request of the Savannah River Operations Office (DOE/SR) and Westinghouse personnel responsible for materials accounting, the Los Alamos and Westinghouse accounting software will be combined to produce a single program that will enable NSR accounting personnel to maintain near-real-time accounting for SNM. This program will include variance propagation and statistical decision procedures for timely detection of accounting anomalies that could indicate loss of nuclear material. The variance propagation program will be based on the Los Alamos MAWST code.

Because the DOE/SR wants to develop a uniform approach to variance propagation in many of the process areas at Savannah River, we are now developing generic interface software to permit MAWST to be used with NUCMAS to analyze accounting information at NSR, FB-line, and other process areas.

This project demonstrates a successful collaboration between the DOE Operations Office, Westinghouse contractor personnel, and Los Alamos to identify and solve operational safeguards problems.

2. Assistance to ANL-W

Argonne West Unified Safeguards Project (J. S. Ballman, R. C. Bearse, and R. M. Tisinger, N-4). In collaboration with SNLA and ANL-W, we have completed the alpha test version of the ARGUS. ARGUS is a safeguards system that combines materials control and materials accounting functions using a network of IBM-PCs and comprises three major subsystems.

The materials control subsystem, known as Computer Augmented Materials Access System (CAMUS), was developed by SNLA. Motion-detection devices, called WATCHes (wireless alarm transmission of container handling), are attached to objects that are to be protected. Movement of a WATCH causes an RF signal to be sent to a receiver at the CAMUS computer. The computer also is coupled to a bar code reader and keypad that allow the user to deactivate the appropriate WATCHes and prepare information about movement of SNM.

The second subsystem, which is distributed over one central and three peripheral computers, is the materials accounting system. This system is a direct descendant of the Los Alamos-developed PC-DYMAC, which was itself a descendant of the Los Alamos' Materials Accounting Safeguards System (MASS). The current version of the accounting system uses the C language and three C-based support libraries: the Faircom* ctree and rtree packages for file handling and report writing, and the Vermont Creative Software** windowing system, Vermont Views, for screen handling. The operating system is Santa Cruz Operations XENIX. A plant inventory is maintained and updated either through a keyboard or by transaction messages sent by the CAMUS system.

The third part of the system, the communications package, ties the CAMUS computer and the three peripheral computers to the Central Computer. Messages are sent from each peripheral and the CAMUS to the Central Computer, and may be forwarded to other peripherals. The Central Computer does not transmit to the CAMUS computer. RS-232 serial communications are used with a Los Alamos-developed communications protocol.

The key to the efficient development of this system was the creation of clean interfaces between the three modules. This allowed almost completely separate development of each of the three packages.

The format of the messages to be generated by the CAMUS system for forwarding to the materials accounting system (PC-DYMAC) was decided very early in the project. Five types of messages and the meaning and range of each field of the message were specified. It was thus possible to test the PC-DYMAC system easily with prototype messages without requiring access to the completed CAMUS system. Similarly, the CAMUS system, needing no input from PC-DYMAC, was developed by SNLA without constant interaction with Los Alamos.

Our experience with an earlier version of PC-DYMAC made it possible to define the information that would have to be transferred from one computer to another to maintain consistent inventories in each of the accounting computers.

The major new problem in this task was developing an efficient and reliable communications package. We originally planned to use the communications packages supplied as part of the XENIX system. We found, however, that because these were designed to be highly flexible, they did not handle our needs efficiently. There can be a delay of up to several minutes between the time the operating system receives a message and the time the message is forwarded. This was not acceptable for our application.

We therefore designed a communications system from scratch, implementing it in the C language. It efficiently sends messages of constant length from one computer to another. We can now send messages every 2 seconds between the accounting system computers with a maximum delay of 15 seconds. The protocol ensures reliable transmission.

An accounting database of all SNM residing within one room is maintained on each of the peripheral computers, while a complete database for all three rooms is maintained on the Central Computer. Every time material is moved between rooms, the inventory of the room of origin is decremented, while the inventory of the receiving room is incremented. The transaction generated by PC-DYMAC to effect these changes in inventory must be transmitted to the appropriate computers. It is written to a file that is placed in a special directory. The name of this file is made unique by incorporating the system time, source, and destination in the filename. This also guarantees proper chronological processing by the receiving computer.

The operating system provides buffering through its file system in the event a computer goes down. When the computer returns to service, the files are processed automatically when PC-DYMAC is restarted. As long as the file system remains intact, no information is lost.

Outgoing files are transmitted by an independent process called SENDWIRE on the sending computer. Another independent process on the receiving computer called READWIRE is activated when an incoming message is detected. Proper transmission is assured by use of checksum calculations and handshaking. The file is then deposited in a special directory in the receiving computer. When PC-DYMAC becomes aware of the presence of this file, it updates its database with the new transaction. Communication from the CAMUS to the Central Computer is handled similarly, although no messages are sent in the opposite direction.

In summary, the peripheral computers have three processes running simultaneously: PC-DYMAC, READWIRE, and SENDWIRE. The Central Computer has eight processes running: PC-DYMAC, a program for reading CAMUS messages called READCAMUS, and a SENDWIRE/READWIRE process pair for each of the three peripheral computers. This multiprocessing would be difficult or impossible on a DOS-based PC.

The READWIRE and SENDWIRE programs are insensitive to the actual contents of the messages sent. The messages are of a fixed, configurable length and can include any ASCII character. The communications processes do a checksum analysis and automatically retransmit any message that becomes garbled. Although the system was difficult to develop, it has been working flawlessly. Our goal was absolute reliability, which can be elusive in a real-time, multitasking system like XENIX. In addition, several nearly undocumented features of the XENIX device drivers had to be understood before the final bugs could be eliminated.

The PC-DYMAC system operates quite well in its testbed at ANL-W. It is about 25 times faster in updating databases than the version currently operating in the plant. The Vermont Views windowing system used for data entry

*Faircom, 2606 Johnson Drive, Columbia, MO 65203.

**Vermont Creative Software, Pinnacle Meadows, Richford, VA 05476.

is much more informative and user friendly than the one used in the current version.

The system will be installed at ANL-W in October 1989 with initial testing leading to a beta test version by March 1990. Final pre-operational testing will occur between March and June 1990, with on-line adoption of the system planned for July 1990.

E. Confirmatory Measurements and Shipper/Receiver Difference Analysis.

In this task, we demonstrate and evaluate confirmatory measurement techniques for several types of nuclear materials in operating production and processing facilities. We address problems associated with shipments and receipts of nuclear materials and the verification of the SNM content or configuration of nuclear devices or both. Portions of some of the projects described here are supported by facility implementation funding.

1. NDA Measurements for Confirmation of HEU Shipments to Oak Ridge (J. K. Sprinkle Jr., P. A. Russo, J. E. Stewart, N-1). The test of a prototype Los Alamos gamma-ray instrument for confirming HEU shipments has been completed at the Oak Ridge Y-12 plant. The in-plant evaluation of this SRCS for HEU demonstrated an excellent ability to discriminate among similar items.[67] Figure 42 shows the first

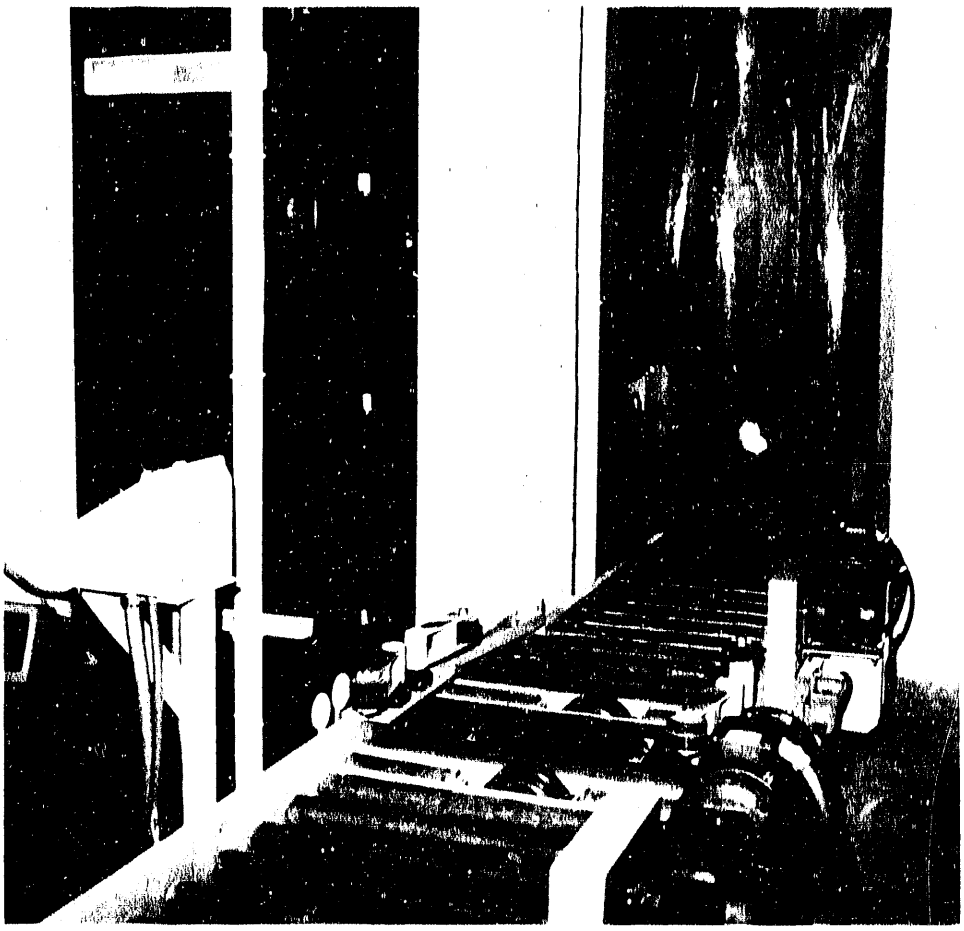


Fig. 42. A view of the third measurement station at Y-12. The rotation section of the conveyor is in front of the detector stand, which contains four collimated detectors.

measurement station (of three planned stations) installed in the Y-12 warehouse. The rotation station in front of the detector assembly and the conveyor are interfaced to the instrument computer. Shipping containers of HEU of various enrichments were evaluated. Four chemical forms and 20 different uranium masses were represented. Figure 43 shows multiple measurements for each of the 20 items. The relative responses at the three gamma-ray energies are plotted on the three axes. The combination of the different gamma-ray responses gives unique fingerprints, as shown by the distinct clusters for each symbol. Even items of the same chemical form, packaging, and uranium mass

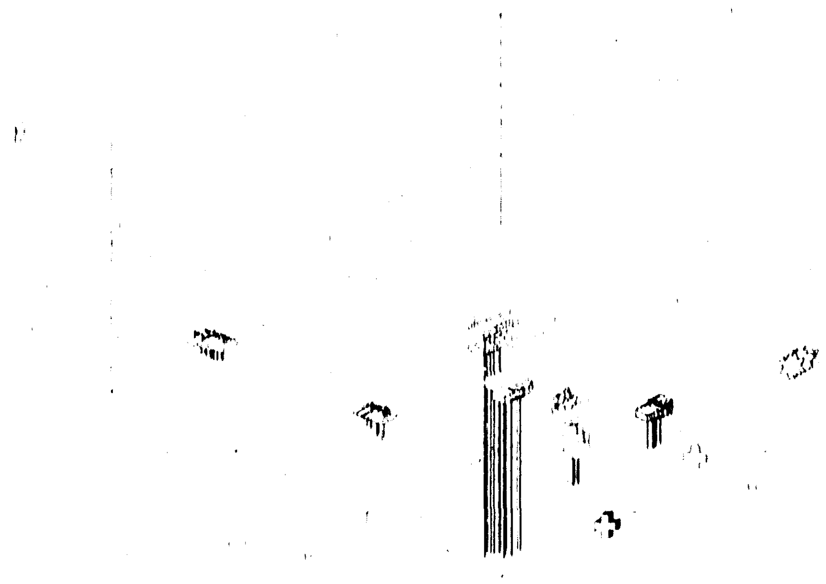


Fig. 43. Multiple responses from 20 shipping containers. The three axes are the relative intensities of three gamma rays: 186 keV, 1001 keV, and 2614 keV. Different shapes indicate different chemical forms of HEU. The distinct clustering of each sample's response illustrates the discrimination this fingerprint provides.

usually could be distinguished—a capability that exceeds the requirements for a shipper/receiver verification.

The Y-12 plant will start receiving UF_6 cylinders in the near future. Using NDA methods to confirm these shipments and possibly using them as the accountability measurement is being discussed. Although it is not yet clear whether NDA methods will be allowed for the input accountability values, we are consulting with Y-12 on some of the NDA options. These include using NCC methods such as the self-interrogation technique[68] and passive gamma-ray methods including ^{235}U enrichment as well as the multi-energy approach used in the SRCS.

2. HEU Shipper/Receiver Confirmatory Instrumentation for Rocky Flats (J. K. Sprinkle, Jr., G. G. Ortiz, R. Siebelist, T. R. VanLyssel, and G. Walton, N-1). After the successful evaluation of the prototype SRCS instrument at Y-12, we are evaluating the insensitivity of the SRCS to the "cereal box" effect—the settling of the contents of shipping containers of HEU during their transport between facilities. This effect could alter the signature used by the SRCS as a fingerprint. Rocky Flats was chosen as a second facility because nuclear material travels both ways between RFP and Y-12. We altered the SRCS design to use a solitary barrel rotator instead of a conveyor. A careful search for commercially available rotators indicated that none were suitable. Consequently we designed a compact, inexpensive solution. Installation at Rocky Flats is expected in February 1990.

3. Shipper/Receiver Confirmatory System Specification for Technology Transfer (J. K. Sprinkle, Jr., R. Siebelist, T. R. VanLyssel, and G. Walton, N-1). If the "cereal box" effect is minimal for the SRCS, we plan to transfer the SRCS technology to private industry. A specification is being started, and the next customer has been located. We have identified 10-20 additional potential customers.

4. Device Verification at Pantex Plant (K. L. Coop, C. L. Hollas, and C. E. Moss, N-2). We completed our development work on the project this year by making additional measurements on a number of secondaries, of two types, during a trip to Pantex in June. We successfully demonstrated a method for each type of secondary that unambiguously permits verification of their composition. To verify the smaller secondary, we used a passive gamma-ray measurement in a well defined and repeatable geometry. For the larger type of secondary, we interrogated the device with a 9-MeV linac, which produces photofissions in uranium. Pulses from the linac, the delayed fission neutrons, and fission neutrons induced in the fissile material by those neutrons, are detected with ^3He and ^4He proportional counters.

We later made a trip to Pantex to demonstrate the neutron method and train Pantex personnel on how to make the verification measurement routinely. We also transferred the detection system to them. Because of delays at Pantex in completing a holder for the gamma-ray detector, we were not able to complete the demonstration and transfer of the gamma-ray method by the end of the year; however, we expect to have this completed early in 1990 and issue a final report soon after.

F. Materials Control Technology Test and Evaluation.

The objective of this task is to demonstrate, test, and evaluate in DOE production facilities the nuclear materials control methods developed under basic technology tasks. Parts of the projects described here are supported by facility implementation funding.

1. Evaluation of Neutron-Detection-Based Vehicle Portal Monitor at Pantex (K. L. Coop and P. E. Fehlau, N-2). We completed an in-plant evaluation of our neutron-detection-based vehicle portal at Pantex. The portal will soon be moved to a permanent location for routine operation. During a visit to Pantex for detection sensitivity testing, we found that the neutron vehicle portal achieved the same level of sensitivity for detecting neutron and gamma-ray shielded plutonium in drive through operation that we expect for our vehicle monitoring stations, which require a vehicle to stop for monitoring.

2. T&E of Digital Image Analysis Techniques for Materials Control (C. A. Steverson, N-4). During 1989, we pursued a variety of image processing T&E projects including the use of infrared sensors to detect holdup in some plutonium processing areas at Los Alamos' plutonium processing facility (PPF) at TA-55; an inventory verification system hosted by a robot stacker-retriever system at Los Alamos' central storage facility (CSF); interim storage materials surveillance in the countroom of Los Alamos' PPF; the use of infrared sensors to monitor stored ingots on the remote mechanical c-line (RMC) metal line at Westinghouse Hanford; and real-time and static image subtraction techniques to monitor materials in long-term and medium-term storage in vaults at INEL facilities.

We found that to detect holdup in plutonium processing areas using infrared sensors provides no great advantage over methods currently in use at the Los Alamos facility. The physical access to problem areas is extremely limited, and background thermal noise in most process lines is prohibitive. Detailed information on this project may be found elsewhere.[69]

An inventory verification system for the CSF, which is in the early conceptual stages of development, has been delayed because of other priorities at the CSF. This system will automatically verify the contents of a stacker-retriever system, ultimately interfacing with the MASS materials accounting database. We will continue this project at a later date.

We have done a preliminary study in the count room of the PPF to develop an imaging-based surveillance system to provide additional protection for items in open, interim storage.

The RMC metal line at Hanford contains an interim storage glovebox for metal ingots produced by the line. We are designing an infrared imaging system to monitor and perhaps track these ingots while they are in temporary storage.

Vault surveillance projects at INEL are in the early stages of development. These projects include the use of real-time and static change detection methods for surveillance of several vault areas, with the desired goal of reducing the inventory requirements for the facility.

II. SAFEGUARDS TECHNOLOGY TRAINING

The DOE Safeguards Technology Training Program is a major vehicle for technology transfer to both the domestic and foreign nuclear communities. Since 1973, the program has grown from a single course to the present curriculum (which includes 10 formal course offerings and a special lecture series) and has serviced nearly 2000 students. The program is very successful both in informing participants of the latest nuclear material control and measurement technology and in keeping the R&D program abreast of the needs and experiences of facility operators and safeguards inspectors. The training program enjoys an excellent reputation throughout the nuclear community.

A. Safeguards Technology Training (H. A. Smith, N-1; K. K. S. Pillay and K. E. Thomas, N-4).

Our 1989 DOE Safeguards Technology Training Program comprised three formal courses and one workshop:

- Materials Accounting for Nuclear Safeguards
- Gamma Spectroscopy for Nuclear Materials Accounting
- Fundamentals of Nondestructive Assay of Nuclear Material
- Variance Propagation and Systems Analysis Workshop

An early presentation of the *Fundamentals* course had been scheduled for March; however, increased physical security requirements could not be implemented in time, so the course was cancelled (see below). The other courses, together with the NDA school for IAEA inspectors and the Nuclear Nonproliferation course on State Systems of Accounting and Control (SSAC) of Nuclear Materials accounted for six offerings in the 1989 training schedule (See Fig. 44). (See Part 4 for a report on the SSAC training and Part 5 for discussion of the IAEA Inspector Training Course.)

The *Materials Accounting for Nuclear Safeguards* course was presented to 30 participants during the period April 10-14. A series of 17 lectures and 5 workshops constituted the core of this year's course. Topics of lectures ranged from fundamentals of materials accounting to site-specific MC&A systems. The use of measurement, analyses, records, and reports to maintain knowledge of the quantities of nuclear materials present in a defined area of a facility and the use of physical inventories and material balances to verify the presence of SNM were emphasized as key elements of good materials accounting for nuclear safeguards.

The workshops offered the attendees the opportunity to participate in exercises that illustrate the advantages and limitations of NDA instrumentation and to carry out exercises designed to demonstrate the roles of real-time materials accounting, measurement control, and variance propagation to nuclear materials accounting. The course culminated in an MC&A systems design exercise in which the participants brought together their personal experiences and what they learned during the course to design an MC&A system for an example facility. This year, a special course manual[70] was prepared for the exclusive use of this course. Sixteen topical discussions prepared by 25 professionals from the nuclear safeguards field cover a spectrum of topics of relevance to materials accounting for nuclear safeguards.

CY 1989 LANL/DOE
Safeguards Technology Training

January	February	March	April
		13-17 Fundamentals Cancelled	10-14 MC&A Course
May	**June**	**July**	**August**
1-19 SSAC Santa Fe, Los Alamos, Richland, WA			2-11 IAEA School
September	**October**	**November**	**December**
18-22 Advanced Y	24-26 Variance Propagation	13-17 Fundamentals	

Fig. 44. Summary of safeguards training activities during calendar year 1989. Shaded entries are the DOE-sponsored courses. The unshaded entries correspond to the IAEA Inspector training courses (see Part V) and the Nuclear Nonproliferation Act (SSAC) Training Course (see Part IV) that were part of the International Safeguards activities.

In addition to the contributions of 20 instructors from Los Alamos, this year's course benefited from the generous participation and contributions of Glenn Hammond (DOE/OSS), Gary Carnival (RFP), and Tom Williams (DOE/SR). The course participants included DOE employees from six regional centers, an NRC employee, an NRC licensee employee, and DOE contractor employees from 14 different sites.

The first presentation of the *Variance Propagation and Systems Analysis Workshop* occurred during the period October 24-26. Participation was by invitation only. Lecture topics included: Terms in the Materials Balance Equation, Theory and Application of Variance Propagation, Estimating Uncertainties, Problems with Variance Propagation, Detection Sensitivity and Decision-Making, and Description of the Example Facility. Working groups derived variance equations for example materials balance terms and solved several problems applied to an example facility. The course was well received, and numerous constructive comments on the content and conduct of the workshop will be incorporated into the next workshop offering, which is scheduled for August 27-29, 1990. Attendance will be limited to 14 persons (2 per computer).

The *Gamma-Ray Spectroscopy for Nuclear Materials Accounting* course was given on September 18-22, 1989 for 24 experienced students interested in becoming familiar with the advanced NDA techniques typically used in in-plant instruments. The course emphasized the use of high-resolution, computer-based gamma-ray spectroscopy systems in applications such as uranium and plutonium isotopics measurements, bulk and segmented transmission-corrected assay, absorption-edge densitometry, and x-ray fluorescence. The course concluded with three topical lectures:

- The Poor Man's Densitometer
- Cold Fusion Measurements in the Safeguards Assay Group
- A Summary of Gamma-Ray NDA Capabilities: A Comparison of Techniques

The course, *Fundamentals of Nondestructive Assay of Nuclear Materials*, was held on November 13-17 for 32 students, most of whom had not been admitted to the October 1988 offering because of space limitations or had been scheduled for the March 1989 course that was cancelled. As usual, this course provided an introduction to neutron and gamma-ray NDA of nuclear materials. Although designed primarily for professional scientists and engineers with little or no background in NDA, the course also is useful to materials accounting supervisors, NRC inspectors, and NDA technicians. The course consisted of lectures and laboratories in gamma-ray and neutron interactions, uranium enrichment measurements, transmission-corrected gamma-ray assays, and neutron singles and coincidence counting. The course concluded with three lectures on

- Specialized Neutron Coincidence Counters for Verification of SNM in an Automated MOX Facility
- Shufflers: History and Application
- The Rest of the Story: Other NDA Applications

The 1989 offering of the *Fundamentals* course also marked a new era in the physical facilities used for the Los Alamos/DOE Safeguards Technology Training Program. During 1989, we acquired additional, temporary classroom space that allowed us to continue certain courses under enhanced SNM physical security requirements. This new training space is within an existing security area where the proper physical security measures can continue during course sessions. This additional training space within an existing protected area has the added benefit that the SNM used in the class sessions is no longer transported with an escort to the classroom, over public roads. Instead, the material is brought to the training area, as needed, from secure vault storage within the same protected area. This space is, however, only temporary until our new Nuclear Safeguards Technology Laboratory is constructed at TA-55.

Table XVI summarizes the training course attendance for 1989. Although each training course has been given in previous years, the courses usually are updated each year to include the latest in measurement techniques, commercially available instruments, and materials accounting procedures.

Both the *Fundamentals* and the *Gamma Spectroscopy* courses have shown significantly increased demand in the past two years, to the point where a single offering of these courses satisfies only approximately half of the demand. In addition, we have received an increased number of requests for courses on neutron NDA techniques, inventory difference analysis, and in-plant holdup measurements. Accordingly, the calendar year 1990 and 1991 schedules are being studied with a view toward dealing with this increased demand. In particular, the former *Attributes* NDA course is being revised and will be presented in FY 1990 as an *In-Plant Holdup* NDA course.

B. Training Course on the Nondestructive Assay of In-Plant Holdup (P. A. Russo, N-1)

We are developing a new course on the principles and techniques for NDA of special nuclear materials holdup. This course will replace the one entitled *Verification Measurements of Nuclear Material Attributes*, which was last presented in August 1987. The 4.5-day *Attributes* course included 1.5 days of laboratory exercises on the calibration and quantitative assay of holdup. Low-resolution gamma-ray measurements used 5-cm-diam by 5-cm-deep NaI(Tl) detectors and PMCAs to measure HEU holdup in pipes, ducts, tanks, and filters. Holdup was simulated by placing HEU samples of various ^{235}U masses and geometries into the appropriate pieces of ducts, piping, tanks, etc. in eight separate "stations" to provide a plant-like environment of deposits for quantitative assay. Calibration of the holdup assay was done with HEU point

Table XVI. Summary of Attendance at LANL/DOE Safeguards Technology Training Courses, 1989

Attendee Affiliation	Materials Accounting 4/10-14	Gamma Spectroscopy 9/18-22	Variance Propagation 10/24-26	Fundamentals of NDA 11/13-17	Totals
Argonne National Laboratory	3				3
Babcock & Wilcox, Lynchburg				1	1
Brookhaven National Laboratory			1		1
DOE (All field offices)	2		1	3	6
DOE Headquarters	3				3
EG&G (Idaho)				1	1
G.E. Wilmington	1				1
LLNL	2	1			3
Los Alamos (Pu Facility)	3	4	2	3	12
Los Alamos (Other areas)	1	2		3	6
Martin Marietta, Oak Ridge	3	3		1	7
Martin Marietta, Piketon	2	4	1		7
Mason & Hanger, Pantex				2	2
New Brunswick Laboratory	2				2
NFS, Irwin		1			1
NRC	1			1	2
Rockwell, Golden	3	1	1	2	7
Sandia Labs, Albuquerque	1	3		2	6
Westinghouse Hanford			1	4	5
Westinghouse Idaho			1		1
Westinghouse Savannah River Co.	2	4	1	6	13
Westinghouse WID, Carlsbad				1	1
Aldermaston/AWE (U.K.)				2	2
CNEA, Argentina		1			1
Japan	1				1
TOTALS	30	24	9	32	95

sources for generalized (point, line, or area) holdup geometries. The HEU assay exercises, along with two exercises involving plutonium holdup assay using neutron totals counting and thermoluminescent dosimetry, constituted the holdup portion of the attributes course. These exercises were quite well received by DOE participants.

The current version of the DOE orders, which became effective since the last offering of the *Attributes* school, describes holdup as a quantity of SNM contained within processing equipment, which, for accountability purposes, may be measured in situ. The recent heightened interest in the ability to perform portable holdup measurements of SNM for accountability purposes as well as for materials control, process control, and safety has increased the interest in an extended offering of a course in this area. In addition, recent advances in technologies[45,47,72] (see also Part 3, Section I.B.1, Part 1, Section I.D.1, and Part 1, Section I.E.2) for portable detectors and improved hardware and automation (see Part 1, Section I.D.1) have provided possibilities for enhancements in such an extended course. The new schoolhouse at TA-18 (Part 3, Section II.A) also provides the opportunity for significant course enhancements in the area of plutonium holdup assay. The new four-day course entitled "Nondestructive Assay of Special Nuclear Materials Holdup" will be given at Los Alamos in the late summer of 1990. The three-day laboratory portion of this course will emphasize the calibration and measurement procedures for assay of holdup using gamma-ray methods in the generalized geometry approach. The simulation of holdup deposits will be accomplished as described previously. Low-resolution gamma-ray detectors and PMCAs will be used in the HEU exercises at the TA-35 schoolhouse. Plutonium holdup deposits will be measured using high-resolution detectors and PMCAs at the new schoolhouse. The high-resolution calibrations and the measurements will be performed with automated electronics setup, data analysis, data reduction, and calculation of calibration and assay results. The high-resolution calibrations and assays will be performed at multiple gamma-ray energies. Both low-and high-resolution exercises will include a quantitative treatment of the effects of equipment attenuation. The advantages of the high-resolution measurements in diagnosing self-attenuation effects will be examined.

For the (low-resolution) HEU exercises, the new compact NaI(Tl) detectors (see Part 1, Section I.E.2) will replace the detectors used in the *Attributes* school. The shielded-detector weight is reduced to less than one fifth that of the larger detectors. Because the smaller NaI(Tl) crystals are not seeded with ^{241}Am to stabilize the analog gain, automated technologies to compensate for the digital-gain drift will be demonstrated as an alternative. The plutonium gamma-ray holdup measurements will use custom-designed portable HPGe detectors (see also Part 1.B.1).[45,71] The plutonium neutron holdup measurements will use portable, polyethylene-moderated ^{3}He detectors.

The new course is intended to instruct both those with experience in the use of low-and high-resolution gamma-ray detectors and multichannel analyzers and those with a direct interest in the measurement of holdup. The three days of laboratory exercises will be preceded by a half day of introductory presentations on holdup measurement principles, practices, equipment, and techniques (including the generalized geometry technique), and radiation safety. The laboratory exercises will be divided into two 1.5-day sessions (for HEU and plutonium measurements). Each 1.5-day session will conclude with a summary discussion to compare measured results with reference values and to draw conclusions on the effectiveness of the assay techniques. The HEU and plutonium sessions will be separated by a half day of presentations on topics related to quantifying holdup of special nuclear materials. The four-day course is scheduled for September 17-20, 1990.

PART 4. INTERNATIONAL SAFEGUARDS

I. SYSTEMS STUDIES AND SAFEGUARDS DESIGN

We have an ongoing task to support international safeguards by improving methodologies, instrumentation, and system designs for major processing facilities. It includes transfer of state-of-the-art safeguards technology to the international community.

A. International Safeguards for Enrichment Plants (M. C. Miller, T. K. Li, and E. L. Adams, N-1; R. B. Strittmatter, N-4).

This subtask assists the IAEA and Member States in developing safeguards equipment and approaches for large commercial enrichment plants.

1. Safeguards Approaches for Enrichment Plants. The US DOE is developing the AVLIS technology for enriching uranium on a production scale as part of the US enrichment enterprise. It is anticipated that the production scale AVLIS enrichment plant would be subject to the US/IAEA agreement (recorded in INFCIRC/288) to voluntarily offer all peaceful US nuclear activities to IAEA safeguards. Under this agreement, the AVLIS plant would be eligible for IAEA safeguards as specified in INFCIRC/153 after being placed on the inspection eligibility list by the US government.

The implementation of INFCIRC/153 safeguards at a laser-isotope-separation uranium enrichment plant is expected to parallel the inspection procedure at centrifuge enrichment plants based on similarities in safeguards concerns for the two technologies. Inspections at centrifuge enrichment plants include detecting the diversion and production of a significant quantity of uranium at an enrichment level higher than declared. In the context of centrifuge enrichment plants, the goal to detect production of enrichment levels higher than declared was implemented as a go/no-go measurement at the 20% enrichment level for facilities with declared maximum enrichment levels of 5%.

Routine inspection activities outside the centrifuge cascade area consist of

- examining records,
- evaluating operator's measurement systems,
- verifying nuclear material flow, and
- verifying physical inventories.

Activities inside the cascade area that contribute to the verification of material production in the range of declared enrichment include:

- visual observation either directly by inspectors or indirectly by optical surveillance devices, and
- technical measures such as radiation monitoring and NDA measurements, sampling, and using seals.

For centrifuge enrichment plants with enrichment capacities of 1 MSWU/yr (million separative work units per year) or less the Limited Frequency Unannounced Access (LFUA) strategy according to the Hexapartite Safeguards Project (HSP)[72] report is to be implemented through inspections that would occur 4-12 times per year. The necessary number of inspections inside the cascade area would, however, be plant specific. On these visits the inspector is allowed in the process building within two hours after the facility is notified by the IAEA that an LFUA inspection is to occur.

2. Measurement Techniques for AVLIS Plants. Under the guidance of the Enrichment Plant Safeguards Review Group (EPSRG), we have focused on evaluating methods of detecting radiation indicative of HEU production and have evaluated a variety of active and passive neutron and gamma-ray measurements for possible application at AVLIS plants. Benchmark measurements at Los Alamos were made to evaluate the various detection methods using HEU and depleted uranium metal disks. We prepared a draft report[73] describing our results, which were discussed at the EPSRG meeting in June.

Neutron Measurements. We obtained both active (AmLi source) and passive data using plastic scintillator and ^{3}He detectors to evaluate ^{235}U and ^{238}U neutron responses. From the benchmark experiments, we concluded that coincidence neutron counting is not feasible for the separator geometry, and only total counting should be pursued. Additional experiments were performed at the Mars facility, which is a half-scale separator development facility at LLNL.

Both the ^{3}He and plastic scintillator detector systems appear capable, in principle, of being used in an active mode for ^{235}U determination and in the passive mode for ^{238}U measurement. However, our experience with these detectors in the separator shows problems with both systems. The basic difficulty is that the sample-to-detector coupling is not favorable, and the problem apparently cannot be overcome by the large mass of material being measured. The problem is even greater with active techniques. In general, the ^{3}He system worked better because of its greater stability, ease of use, background-to-noise ratio, and insensitivity to gamma rays, which makes data interpretation less difficult. Although a ^{4}He system might appear to be better than ^{3}He for discriminating the AmLi source from the IF neutrons, our comparison of ^{3}He and ^{4}He data under similar detector/sample configurations indicates that the efficiency of the ^{4}He detector is too low for this application. Taking into account all of these factors leads to the conclusion that external separator measurements are not practical with any of the neutron systems that we evaluated, primarily because of the unfavorable geometry. We therefore recommend that neutron methods not be further pursued in the case of external separator measurements. When coincidence counting methods could be used (for example, in measuring billets), a ^{3}He system would be the detector of choice.

Gamma-Ray Measurements. We considered two approaches for confirming enrichment. Gamma-ray measurements of uranium enrichment are typically based on measuring 185.7-keV gamma rays emitted as a result of the alpha decay of ^{235}U to ^{231}Th. The standard nondestructive ^{235}U enrichment assay by gamma-ray spectrometry employs the enrichment meter principle.[75] If the sample to be measured has a thickness that is large compared to the mean-free path of 185.7-keV gamma rays, then the 185.7-keV gamma-ray flux emitted from a unit area is directly proportional to the ^{235}U enrichment. In practice, the attenuation of the measured gamma ray through container walls also must be determined. The penetrability of this gamma ray through commonly used materials of construction, such as stainless steel, is low enough that material thickness would have to be confirmed.

Another approach that has been used to determine enrichment based on gamma-ray measurements is to combine measurements of the ^{235}U and total uranium concentrations. The total uranium concentration has been determined by transmission measurements using external sources, XRF, and other emitted gamma rays. Gamma radiation resulting from the decay of ^{238}U daughter products includes the 766- and 1001-keV gamma rays emitted from the decay of ^{234m}Pa. Other gamma rays from daughters in the decay chain of ^{238}U include the 63-, 92-, and 93-keV gamma rays from ^{234}Th. However, because these gamma rays are the result of a decay chain, decay equilibrium must be established for accurate determination of ^{238}U concentrations.

Preliminary results of gamma-ray measurements can provide information to verify the enrichment of the feed material (if it is homogeneous) by employing the enrichment principle. Similar measurements may be suitable to monitor the tails and product accumulators. Our gamma-ray measurements indicate a potential for application as an external measurement if there is assurance that the internal geometry has not changed (that is, the location of product and tails, the material thickness, etc.). However, there are still questions regarding application of the enrichment meter principle for separator measurement and further study is needed. Although the external measurement of the separator using a gamma-ray-based method may be technically feasible and would provide a direct confirmation of the enrichment range in the separator at the time of inspection, the severe dependence on separator geometry poses serious problems.

Additional Measurements. Other potential measurement strategies include measurements of the internal separator and feed, product, and tails. Internal separator measurements would be gamma-ray based and would directly confirm the enrichment range in the separator at the time of removal, but would require assurance that this enrichment indicates its previous operation. This approach poses moderate difficulty for technical success and implementation. The biggest concern would be sampling adequacy and availability, with associated potential problems regarding impact on the operator and on the control of sensitive/classified information.

Evaluating the feasibility of such an approach would require field measurements at the test facility. Measuring the feed, product, and tails would be relatively straightforward. Technical success would be highly probable but implementation could be a moderate problem. A gamma-ray measurement employing the enrichment meter principle could enrich feed, product, and tails; however the relatively high accuracy of the measurements might raise questions about the security of sensitive/classified information. Verification of the material flow probably would be necessary to assure sample representativeness.

We recommend that a series of measurements be made on the separator intervals, concentrating on gamma-ray methods. As a result, our investigations in the upcoming year will be directed toward measuring feed, product, and tails material with active neutrons. The study will be facilitated by MCNP calculations. The MCNP model will guide the detector design and aid in calibration and analysis once the prototype assay system is built.

At the same time, operational issues and the protection of sensitive/classified information (regarding the measurement of separator feed, product, and tails streams using gamma-ray or neutron techniques) should be addressed.

B. International Safeguards for Reprocessing Plants.

The goal of this subtask is to assist the IAEA and Member States in developing safeguards equipment and approaches for large commercial reprocessing plants.

1. Instrumentation for Monitoring Light-Water Reactor (LWR) Spent Fuel Movement (G. E. Bosler and S. F. Klosterbuer, N-1). An unattended system for monitoring the movement of spent-fuel storage casks is being installed in the transfer channel between the receiving pool and the main storage pool at the Thermal Oxide Reprocessing Plant (THORP) in Sellafield, UK. The system uses radiation detectors to trigger a Modular Integrated Video System (MIVS) that records images of the activity in the transfer channel. The radiation detection part of the system has been developed by Los Alamos. The MIVS will be provided by SNL.

The radiation detection equipment consists of a GRAND I electronics unit[76] and four ion chambers. A computer retrieves information about each detected event stored in the memory of the GRAND I. Submerged ion chambers are mounted in pairs on either side of the underwater TV camera. Signal levels in the ion chambers are continuously monitored by the GRAND I. When multi-element bottles containing clusters of either boiling water reactor (BWR) or pressurized water reactor (PWR) spent-fuel assemblies, or skips containing advanced gas-cooled elements are moved through the channel, gamma-ray levels increase and exceed preset threshold levels. The GRAND I software determines that a fuel movement is underway and sends an appropriate signal to the MIVS. The TV camera and recording system are turned on to record images of the activity in the channel. With pairs of detectors, the direction of motion can be determined by the GRAND I software. The direction information is sent to the MIVS and displayed along with other information such as date and time in an annotation at the bottom of the images.

Sense switches attached to the track of the equipment that move the storage casks also trigger the MIVS. In this manner, all movements in the transfer channel are recorded. The radiation monitors confirm that radioactive materials are being moved.

The radiation detection system has been installed in the transfer channel; in March 1990, the MIVS will be installed, and a test and evaluation period will follow. Eventually, the system will be used by European Atomic Energy Community (EURATOM) for routine inspections at THORP.

2. Safeguards Approaches for a Large Reprocessing Plant (R. G. Gutmacher and J. W. Barnes, N-4). We completed a study with personnel from DWK in the Federal Republic of Germany (FRG) on the loss-detection sensitivity of a 500-MT/yr reprocessing plant. The study used error estimates from key measurement transfer and inventory points as provided by DWK and as arrived at by Los Alamos personnel. In some cases, Los Alamos estimates of measurement uncertainties were larger than those of DWK; in some cases, our values were smaller. The study showed that loss detection sensitivity depends on how well transfers can be measured and how well process material in tanks, contactors, and concentrators can be measured or estimated. Our final report considered comments received from DWK on the draft report, and was accepted by DWK as final.

3. Verification Approaches for Reprocessing Plants (E. A. Hakkila, J. W. Barnes, and R. G. Gutmacher, N-4; R. R. Picard, A-1). We completed a report[76] that reviews approaches for verifying transfers into and out of a reprocessing plant and for verifying the in-process inventory. The report was submitted as input to the Safeguards for LArge SCAle Reprocessing plants (LASCAR) project.

4. Relationships Between Process Monitoring and Materials Accounting for Safeguards (E. A. Hakkila, N-4). We have begun a study to determine if process monitoring can be used to verify some aspects of the design of reprocessing plants.

Process monitoring has been suggested as a safeguards measure to ensure that a facility is operating as designed, or as a surveillance measure to ensure that material is not removed from the facility and not declared. In a process-monitoring system, the facility operator monitors process operations such as tank levels, solution densities and temperatures, process flows, and physical parameters such as valve positions to ensure that the operations performed are both desired and required. At many facilities (for example, at INEL), the process-monitoring system also is an important safety feature to prevent criticality.

Verifying facility design is necessary for applying safeguards in a reprocessing plant. However, verifying all pipes and valves by comparing blueprints with the as-built facility is an almost impossible task with the IAEA's limited inspection resources. We propose applying process monitoring to verify the design of the internationally safeguarded facility. By carefully selecting process-operating variables, it may be possible to verify that plant flows are as described and that key measurement points are not bypassed.

We are continuing the study using process monitoring data from an operating facility.

5. Near-Real-Time-Accounting (NRTA) in Reprocessing Plants (E. A. Hakkila and R. G. Gutmacher, N-4). One of the concerns in applying NRTA techniques in reprocessing plants is measuring and verifying the inventory in solvent extraction contactors. One proposed method uses process operating parameters to estimate the inventory. This technique could involve proprietary or sensitive information, and it is important to define the minimum set of data that would be required.

The TEKO facility, operated by DWK in Karlsruhe, FRG, has experimental pulsed columns operating with uranium for which extensive data have been accumulated to measure concentration profiles in the columns as a function of operating parameters.

A Los Alamos consultant visited the DWK headquarters in Hannover and the TEKO reprocessing experimental facility in Karlsruhe to obtain experimental data from the TEKO pulsed columns. We reviewed pulsed column data from three campaigns. One campaign consisted of normal operation and the others of perturbed operating conditions. For one perturbed run, acid was deleted in the scrub stream, and in the other, the organic/aqueous phase ratio was changed. The data were transmitted to Los Alamos for analysis. DWK also requested that Los Alamos evaluate the column density technique for determining nuclear material inventory using DWK-supplied density data. These data are being analyzed by a consultant from Clemson University.

6. Resin Bead Measurements Using Low-Energy Gamma Rays (T. K. Li, N-1). As part of the Specific Memorandum of Agreement between the US DOE and the Power Reactor and Nuclear Fuel Development Corporation (PNC) of Japan, it was agreed to carry out a joint research and development study of plutonium isotopic and concentration measurements for input accountability tank solutions at chemical processing plants. The technique under study is the gamma-ray measurement of resin bead samples. PNC is responsible for the plutonium separation and the resin bead preparation for unspiked and spiked samples. Los Alamos is responsible for developing a gamma-ray spectroscopy technique to measure the resin beads. This method, which may facilitate the application of safeguards and IAEA inspections at chemical reprocessing plants, is to be demonstrated and possibly implemented at the PNC's Tokai Reprocessing Plant (TRP). The measurement technique and the results of the first joint measurements, which were performed in January 1989 at PNC/TRP, are reported in Part 1, Section I.C.10.

7. Rapid Chemical Separation of Uranium and Plutonium in Dissolver Solutions (V. T. Hamilton, W. D. Spall, B. Smith, and D. D. Jackson, CLS-1). We are developing methods for separating and measuring plutonium and uranium in dissolver solutions. An approach using reversed-phase liquid chromatography followed by post-column colorimetric reaction and spectrophotometric detection has given promising results. Alpha-hydroxyisobutyric acid in the mobile phase complexes Pu(IV) and U(VI) and provides good separation. Following the separation, Arsenazo III is added to form color complexes, and the absorbance is measured. Plutonium and uranium are separated from the lanthanides, and preliminary investigation indicates that transition metals will not interfere with the detection. The separation of uranium and plutonium from fission products appears to be good. We are planning experiments with dissolver solutions.

We also investigated high-performance capillary electrophoresis (HPCE) to separate and measure Arsenazo III complexes of uranium and plutonium. HPCE is attractive; it needs very small reagent and sample volumes and produces little waste. The instrumentation has few components and can be very compact, making it suitable for field use. The metal-Arsenazo III complexes are electrophoretically loaded, moved through the column, and separated based on total charge. The separated complexes are measured spectrophotometrically as they exit the capillary column. We investigated electrophoresis buffers, sample and buffer conditions, and instrument settings to optimize the resolution and detection of the complexes.

Although the method may be useful for separating other complexes, we were unable to separate thorium and uranium complexes; they behave as though they were neutral species. Different buffers, complexants, and separation columns might be able to resolve the metals, but such a study was beyond the scope of this initial investigation. Sample loading techniques and certain other instrumental factors are not satisfactory for hot samples, but this is an emerging technique and should be monitored closely as it develops for applicability in this type of analysis.

8. LASCAR (E. A. Hakkila, N-4 and G. E. Bosler, N-1). LASCAR is an international group to study safeguards approaches for reprocessing plants. Membership comprises the IAEA, EURATOM, Japan, France, the UK, the FRG, and the US.

Los Alamos staff members provided a major portion of technical input to LASCAR Working Group 1 (Safeguards Measures for the Spent Fuel Storage Pool) and Working Group 2A (Safeguards Measures for the Headend of Reprocessing Plants, Chop-Leach to the Input Accountability Tank), and participated as the US technical representative to meetings of these working groups in Sellafield, the UK, La Hague, France, and Hanover, FRG, and in the second Plenary meeting of LASCAR in York, UK.

C. International Safeguards for MOX Fuel Fabrication Facilities.

The goal of this subtask is to develop appropriate safeguards approaches for MOX fuel fabrication facilities, including nuclear materials measurement techniques, and methods for processing and interpreting MOX materials accounting data.

Remote-Controlled NDA Systems for International Safeguards at an Automated MOX Facility (H. O. Menlove and M. C. Miller, N-1). We have developed NDA measurement systems for use in an automated MOX fabrication facility. Unique features of these NDA systems accommodate robotic sample handling and remote operation. In addition, the systems have been designed to obtain IAEA inspection data without the need for an inspector at the facility at the time of the measurements. The equipment will operate continuously in an unattended mode while storing data for up to one month. The systems include a canister counter for assaying MOX powder at the input to the facility and a capsule counter for assaying complete liquid-metal fast breeder reactor (FBR) fuel assemblies at the output of the plant.

Although the HLNC-II for measuring bulk samples of plutonium powder was based on previous work at Los Alamos, the special features of new automated fabrication facilities required new NDA technology. These facility features included:

- automated robotic fuel handling,
- remote operation,
- large sample masses with the possibility of high radiation levels, and
- inaccessibility of most fuel for sampling.

These operational constraints are common to many of the modern facilities that have been designed for fabricating and processing plutonium fuel.

To accommodate these facility features and to reduce the inspector's workload, we designed the NDA equipment to be automated, amenable to unattended operation, and have a size and fuel-mass capability to match the robotic fuel manipulators. Authentication techniques have been incorporated into the NDA systems so that the data can be used by independent inspectors such as the IAEA.

The authentication features include

- detector head under inspector seal,
- visible and unbroken cable runs between detector head and electronics cabinet,
- sealed electronics cabinet,
- modular electronic components that can be replaced with standard IAEA equipment,
- continuous data collection,
- software replaceable by the inspectors,
- software diagnostics to detect interruption of or tampering with the signal,
- californium-252 check sources and normalization sources for verification of total system performance, and
- containment and surveillance (C/S) system overview of the detector and electronics cabinet.

These measures give an in-depth redundancy in authenticating the NDA system.

Continuous monitoring of the room background gives a record of any movement of MOX in the room. Because recording MOX movement also is part of the C/S system, the detector gives an independent method for partially authenticating the system.

The NDA systems have been designed to measure the feed powder input, the in-process MOX, the pellets, the fuel pins, and the finished fuel assemblies. These systems have been in operation for the past year with an excellent reliability record (no loss of inspector data or maintenance downtime). New NDA systems are in the design phase to measure MOX holdup in glove boxes and waste barrels.

D. Spent Fuel Safeguards.

The goal of this subtask is to develop appropriate safeguards approaches for spent fuel at nuclear reactors, spent fuel reprocessing plants, and long-term fuel storage facilities, including nuclear materials measurement techniques and instrument calibration. We also evaluate alternative systems for measuring spent fuel in wet and dry storage and different physical storage configurations.

1. Detection of Partial Removal of Spent-Fuel Rods (G. E. Bosler and A. J. Nelson, N-1). We have made underwater measurements on a simulated spent-fuel assembly with a fork detector to examine the effects of missing or substituted pins. We used fresh MOX pins instead of actual spent-fuel pins. In the MOX assembly sources of gamma-rays and neutrons are different, compared to a typical spent-fuel assembly, and the neutron multiplication in the assembly is higher. However, the advantages of using the fresh MOX pins are:

- the presence of both plutonium and uranium isotopes as in spent-fuel materials,
- geometrically distributed neutron sources as in spent-fuel rods, and
- low radiation levels, so remote handling is not necessary.

Typically, water in the fuel storage pool of a PWR contains concentrations of boron. The boron is used as a poison for thermal neutrons in the reactor, which has a common water channel with the storage pool. For this reason, we made the pin removal studies both in water with no boron and in water with varying boron concentrations.

In the study, 12 pins (approximately 6% of the 204 pins in a 15 x 15 PWR assembly) were removed from various rows within the assembly. We also investigated the effects of substituting 12 low-enriched uranium (LEU) pins for 12 MOX pins.

Pin removal effects for different boron concentrations are shown in Fig. 45. In this figure, row one is the outside row closest to the fork detector, and row seven is in the middle of the assembly. For each boron concentration level, data are normalized to the count rate for the full assembly.

With no boron in the water, localized neutron multiplication effects compensate for the removal of pins. As a result, count rates for pins removed in all rows (except for the outside row nearest the fork) are reduced no more than 3%. In one case, the count rate with 6% of the pins missing is even greater than the count rate for the full assembly. With boron in the water, the situation is different. As the boron concentration increases, neutron multiplication is dampened, and the count rate always decreases when pins are removed. When boron concentrations are above 1000 ppm, count rates for all pin removal patterns are reduced 6% or more relative to the fully loaded assembly.

Substitution of LEU pins was investigated only for unborated water. Results of the LEU substitutions are shown in Fig. 46. Again the data are normalized to the count rate for a fully loaded MOX assembly. Because the LEU pins and MOX pins available for this exercise had identical reactivities, multiplication within the assembly does not change when the LEU pins are inserted. This can be seen in the coincidence (reals) count rate. The coincidence rate is, in part, a measure of the multiplication within the assembly. This rate basically does not change when LEU pins are substituted. The totals rate decreases when LEU pins are removed and the decrease is constant for all pin patterns. The decreased count rate is caused by the removal of neutron sources within the MOX pins. This source is not replaced when LEU pins are substituted.

For spent-fuel measurements, there are three implications. First, in storage pools with boron concentrations above 1000 ppm, missing pins are detectable with totals

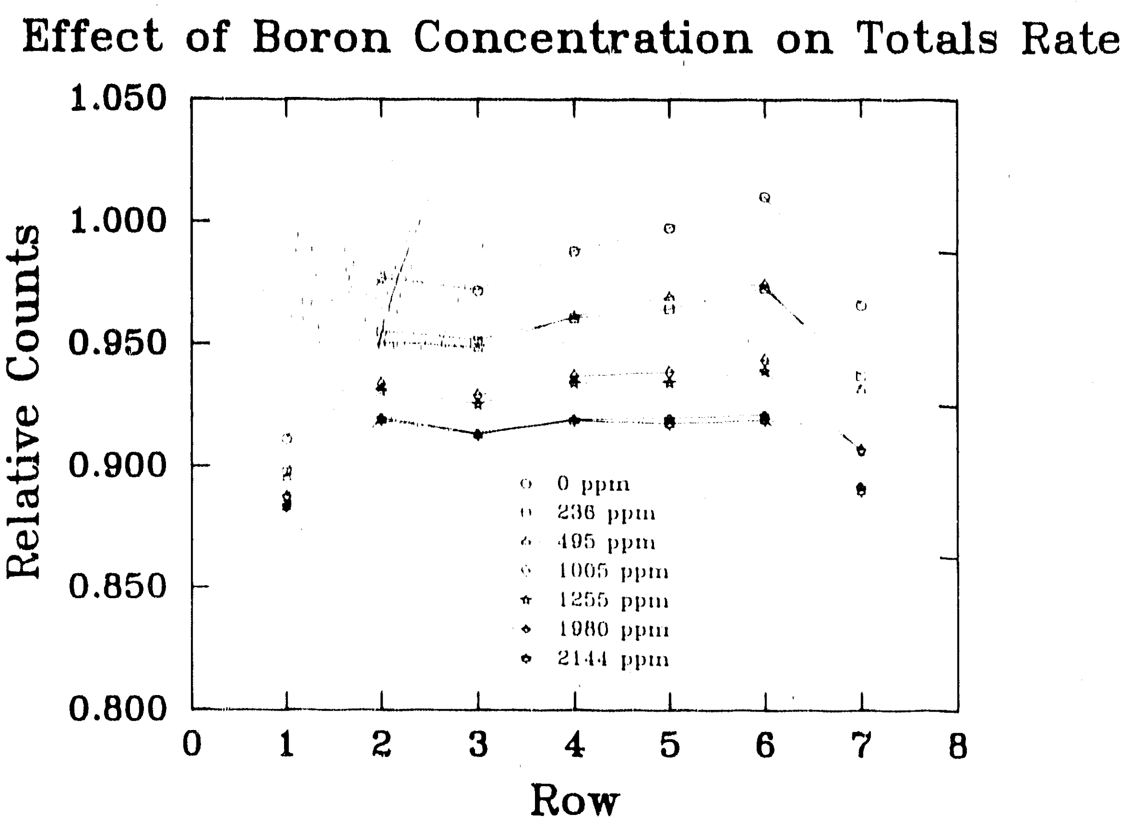


Fig. 45. The effect of boron concentration on the normalized totals rate for various configurations with 6% of the pins removed.

neutron counting techniques in a fork detector. Second, in unborated water, totals neutron counting can be ineffective. There are missing-pin patterns for which the totals count rate is actually higher than the corresponding count rate for the full assembly. Third, substitution of materials with comparable reactivities (if this is credible) may not be detectable.

Because most PWR storage pools have concentrations of boron in the water, totals counting techniques for spent-fuel assemblies should be adequate. In pools without boron, totals counting is probably inadequate if the ability to detect that 6% or less of the pins are missing is required. For the fresh MOX measurements, all missing-pin and pin-substitution patterns that were investigated could be detected with analysis techniques for multiplication-corrected coincidence counting. Such analysis assumes that the plutonium isotopics are known for the assembly. It is conceivable that coincidence counting techniques could be used for spent fuel, particularly in cases where assemblies are stored in unborated water. However, more research is needed to investigate this possibility.

2. International Safeguards for Spent Fuel in Geological Repositories (K. K. S. Pillay, N-4). Spent fuels from once-through fuel cycles placed in underground repositories have the potential to become attractive targets for diversion and/or theft because of their valuable material content and decreasing radioactivity. The first geologic repository in the US, as currently designed, will contain approximately 500 Mt of plutonium, 60 000 Mt of uranium, and a host of other fissile and strategically important elements. We have identified and highlighted several international safeguards issues relevant to the various proposed scenarios for disposing of the spent fuel.[77] In the context of the US program for geologic disposal of spent fuels, there are several issues that should be addressed in the near term by US industries, the DOE, and the Nuclear Regulatory Commission before the geologic repositories for spent fuels become a reality.[77,78]

US/IAEA agreements require the spent fuel disposal program to conform to IAEA safeguards requirements. However, because of the uncertainties in the geologic repository schedules, the US utilities are scrambling to incorporate alternative spent-fuel management strategies, including consolidation, dry storage, away-from-reactor storage, etc. To meet our obligations to international safeguards, it is necessary that a materials accountancy program be part of all spent-fuel management strategies.

Furthermore, in the international arena, the recent offer of geologic repository services for spent nuclear fuel by Canada, and earlier similar proposals by the Peoples Republic of China, are going to introduce new and unique safeguards problems. Because all of these issues need systematic examination and resolution, we organized another

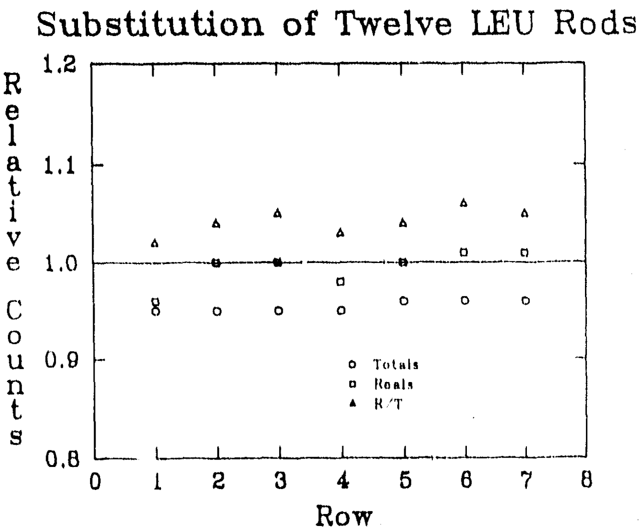


Fig. 46. Normalized responses for configurations with 6% of the MOX pins (12 pins) replaced by LEU pins.

technical session at the 1989 ANS winter meeting that had international participation, including representatives from the IAEA (see pages 240-248 of Ref. 78). In future safeguards systems studies at Los Alamos we hope to explore these issues in depth, to propose safeguards approaches to address international safeguards issues of spent nuclear fuel management.

II. BILATERAL TECHNICAL EXCHANGES

The purpose of this task is to assist the IAEA, EURATOM, and IAEA Member States in improving the effectiveness and efficiency of international safeguards.

A. IAEA -- Advisory Group on Nondestructive Assay (J. L. Parker and R. Augustson, N-1).

Two Los Alamos staff members participated in the IAEA Advisory Group meeting on NDA measurements. The final report of the Advisory Group will be a comprehensive report that reviews general principles and procedures for calibrating gamma-ray and neutron-based NDA equipment, with emphasis on the practical problems of the IAEA inspectorate. The Advisory Group also considered the problems of fabricating, certifying, and transporting physical standards with additional discussions of possibilities of minimizing the number of standards required through computational studies of appropriate physical models.

B. EURATOM

1. EURATOM Interactions (G. E. Bosler, A. J. Nelson, R. H. Augustson, and L. R. Cowder, N-1). In some reactor facilities around the world, but particularly in Europe, assemblies with MOX fuels are now being used. At most facilities, fresh MOX fuel assemblies are placed in the spent-fuel storage pool before refueling activities begin. During refueling, the assemblies are moved from the storage pool and loaded into the reactor core.

From a safeguards standpoint, fresh MOX assemblies are very important. Not only is there considerable plutonium (>10 kg) in each assembly, but unlike spent fuel, the material can be handled easily. Generally, fresh MOX assemblies are only in the storage pool for a short time (<3 months) before being loaded into the core. During this time, however, the amount of material in the assemblies must be verified.

We have developed a measurement system for determining the linear plutonium mass loading in a submerged MOX assembly. The system uses a detector that we developed previously for underwater spent-fuel measurements, called a fork detector.[79] A modified version of the fork detector is used for fresh MOX measurements. With this detector, we measure totals and coincidence count rates from a MOX assembly.

Multiplication-corrected coincidence count rates are used to determine the linear density. Measurements have been made in unborated water and in water with various boron concentrations. Thermal neutron absorption in the water does affect the measurement results, and data must be adjusted for boron concentration level.

In determining the effectiveness of the measurement and analysis techniques, we investigated the sensitivity to various missing-pin and pin-substitution patterns. Results of the measurements were quite good. Under controlled measurement conditions, differences in the predicted mass compared to the known mass were typically 2% or less.

A report on the measurements is being written and a field test is planned.

2. EURATOM Inspector Training Course on the Neutron Coincidence Collar (J. E. Stewart, N-1; G. P. D. Verrechia, EURATOM-Luxembourg; R. Carchon, SCK/CEN-Mol; P. Boermans, FBFC-Dessel). Developed by Los Alamos, the standard neutron coincidence collar[80] is an active interrogation device used routinely by EURATOM and IAEA inspectors to verify ^{235}U content in unirradiated LWR fuel assemblies. In fact, the collar is the only method currently available to completely verify these assemblies.

The EURATOM Safeguards Inspectorate, headquartered at the Luxembourg facilities of the CEC, has seven collars and is purchasing two or three more. During physical inventory verification exercises, the Inspectorate typically measures 60-80 assemblies in one week. During flow verifications, approximately 20 assemblies are measured in 2 days.

Because of (1) the heavy reliance on the neutron coincidence collar, (2) recent modifications in measurement procedures, and (3) the desire to acquaint inspectors more fully with the effects on collar measurements of the nuclear design (for example, burnable poisons and variable enrichments) of present-day LWR fuel in Europe, a specialized training course was organized and held for EURATOM inspectors during 6-13 December 1989 in Belgium. The course was given in two sessions of three days each; a total of 13 inspectors participated. Table XVII is the course schedule and outline.

The course was extremely valuable because it provided the unique combination of theoretical, laboratory, and facility sessions in three successive days. The EURATOM collars had been calibrated on the Los Alamos reference assemblies in October 1989. The cross-calibration procedures[85,86] that were used produced verification accuracies of approximately 4% with no significant bias. The accuracy quoted was for production assemblies measured at the Franco-Belge de Fabrication de Combustible (FBFC) facility near Dessel. These are 17 x 17-rod arrays compared with the reference 15 x 15-rod Los Alamos assembly. A correction was applied to account for the heavy-metal-loading difference between the reference and FBFC assemblies. This correction was successful in that it produced no significant bias. Figure 47 is a photograph of participants during verification measurements inside one of the assembly stores at FBFC.

At the Venus Critical Facility, there is a great deal of flexibility for configuring pin arrays, although their active length is only 50 cm. We made calibration measurements with three different types of poisoned rods (Ag-In-Cd, Pyrex, and steel) that showed poisoned-rod effects to be linear for the concentrations of interest. For example, the effect of combining three kinds of poisoned rods could be found by computing the product of the individual effects. There is a practical application of this result in that combinations of poisoned rods are found in production facilities.

The effort required to conduct this course was significant, but worthwhile. It was agreed by all participants that the course should be repeated, probably in May or June of 1990.

3. ESARDA Symposium on Safeguards and Nuclear Materials Management (J. E. Stewart, N-1; K. L. Coop, N-2; J. T. Markin and E. A. Hakkila, N-4). Four Los Alamos staff members participated in and presented technical papers at the Eleventh ESARDA Symposium on Safeguards and Nuclear Materials Management in Luxembourg (see the Publications List of this report).

C. People's Republic of China (E. A. Hakkila, N-4).

Mr. Qiao Shengzhong, Radiochemistry Group Leader at the Institute of Atomic Energy in Beijing arrived in Los Alamos in May to work as a research associate in the Safeguards Assay Group. He is involved in several NDA research and development projects in the group, particularly in the active neutron measurements of LWR fuel assemblies.

The visit is a follow-up to a previous visit by Mr. Zhu Rongbao to assist the People's Republic of China in developing safeguards expertise for their nuclear program.

D. Brazil

We continue to interact with the Comissao Nacional de Energia Nuclear (CNEN) in Brazil. A planned third set of spent-fuel measurements to be made by CNEN at the Angra-I reactor in 1989 was postponed for a variety of reasons. The measurements have been rescheduled for sometime in early 1990.

The measurements will be made on the same assemblies that were measured in the first two exercises. The measurements are being used to study the decay of neutron-producing isotopes in spent-fuel assemblies. Measured neutron count rate data are compared with calculated data. The calculated rates are determined from a burnup-depletion code that predicts the isotopic composition of the spent-fuel materials. Data from Angra-I are very important for verifying the accuracy of the calculational methods.

A paper discussing the results of the first two sets of measurements was presented at a reactor conference held in Recife, Brazil on April 26-28, 1989. Dr. Marco Marzo of CNEN presented the paper.

The third set of measurements will complete the cooperative effort under the current agreement between Los Alamos and CNEN. There is interest in continuing these types of measurements at Angra on other spent-fuel assemblies. Discussions on continuing the effort will be held in April, 1990 during the visit of a Los Alamos staff member to Brazil.

E. The UK (C. A. Coulter, N-4).

In October, 1989, a Los Alamos staff member visited British Nuclear Fuels plc (BNFL), Risley Warrington, England, and Cadarache Center, St. Paul lez Durance, France, to discuss methods of anomaly detection and identification in nuclear materials processing facilities. The host at BNFL was Dr. Barry Jones, who has done extensive work in developing joint Page's tests for detecting anomalies in material unaccounted for (MUF) values. On the first day, Dr. Jones arranged for a tour of the THORP facility, which is being constructed at Sellafield. THORP consists of two buildings: the Receipts and Storage facility, which is currently undergoing commissioning tests, and the Purex building proper, where equipment is still being installed. When THORP is fully operational, it is expected that the Receipts and Storage Facility will process about three shipping casks per day.

On the second day, we discussed Dr. Jones' work in sequential testing and anomaly identification and Los

Table XVII. Schedule of NCC Training for EURATOM Inspectors at Mol, Belgium (December 1989)

		Day 1 at SCK-Mol and VENUS		Day 2 at VENUS		Day 3 at FBFC	
		Topic	Person(s)	Topic	Person(s)	Topic	Person(s)
AM	9:00	I. Introduction	R. Carchon (Mol)	I. NCC Checkout	Carchon, Verrechia,	I. FBFC Overview	P. Boermans (FBFC)
		"	"	a) Background	Stewart	"	"
	9:30	II. Course Overview,	G. Verrechia (EUR-Lux)	b) AmLi, Cf-252	"	I. Set Up for FBFC Store	EUR Inspectors
		Objectives	"	"	"	"	"
	10:00	"	"	II. Calibration	"	II. NCC Checkout	"
		Coffee	all	Coffee	all	a) Background	"
	10:30	III. Facilities Descriptions	R. Carchon (Mol)	continue II. Calibration	Carchon, Verrechia,	b) AmLi, Cf-252	"
		(Venus, FBFC)	"	(without poison rods)	Stewart	III. Verify FBFC Store 1	EUR Inspectors
	11:00	IV. Basics of Neutron	J. Stewart (Los Alamos)	a) 3 assemblies	"	a) select 4 elements	"
		Coincidence Counting	"	b) Deming Fitting	"	(no poison)	"
	11:30	"	"	"	"	b) verify elements	"
		"	"	"	"	with NCC (X-Calib.)	"
PM	12:00	Lunch	all	Lunch	all	Lunch	"
		"	"	"	"	"	all
	12:30	"	"	"	"	"	"
		"	"	"	"	"	"
	1:00	V. Introduction to NCC	J. Stewart (Los Alamos)	III. Verification	Carchon, Verrechia,	continue with III.b)	"
		"	"	a) 1 ass'y (from AM)	Stewart	"	EUR Inspectors
	1:30	"	"	IV. Poison Calib.	"	III. Verify FBFC Store 2	"
		"	"	a) 5 assemblies -	"	a) select 4 elements	"
	2:00	VI. NCC Calibration	J. Stewart (Los Alamos)	variable Ag/In/Cd	"	(no poison)	"
		"	"	"	"	b) verify elements	"
	2:30	"	"	"	"	with NCC (X-Calib.)	"
		"	"	b) 4 assemblies -	"	"	"
	3:00	Coffee	all	variable steel	"	"	"
		VII. Move to Venus	G. Verrechia (EUR-Lux)	Coffee	all	Coffee	all
	3:30	a) NCC Setup	"	continue IV. b)	Carchon, Verrechia,	Wrap Up:	G. Smith (EUR-Lux)
		"	"	c) 4 assemblies -	Stewart	EUR-Lux Experience	and participants
	4:00	b) Demonstration of	"	variable pyrex	"	"	"
		software	"	"	"	"	"
	4:30	"	"	V. Poison Verification	"	"	"
		"	"	"	"	"	"
	5:00	Finish	all	Finish	all	Finish	all

Fig. 47. Participants at FBFC PWR fuel fabrication facility (Dessel, Belgium) during NCC training course for EURATOM inspectors.

Alamos work in modeling and simulation. Dr. Jones has developed a joint Page's test for MUF that combines two Page's tests, one with parameters selected for sensitivity early in the sequence and one with parameters selected for sensitivity late in the sequence. In addition, he has developed analytical methods for determining the appropriate parameters for this test. Dr. Jones also discussed some of his recent work in anomaly identification, in which he introduces a "diversion" in simulated materials balance values and then tries to determine what the most probable diversion is by testing all possible uniform diversions with varying beginning and ending periods within the sequential test run; adjusting the tota' mount of material assumed diverted to obtain the best to the balance sequence and then selecting as most probable, the diversion scenario with the smallest mean square error.

F. France

1. Facility Simulation (C. A. Coulter, N-4). Our principal contact for collaborative work with the Cadarache Center is Dr. Antonio Giacometti. During a recent visit to Cadarache, Dr. Giacometti described the current status of his simulation/anomaly-detection program Fil d'Ariane, and we summarized the status of our simulation program FacSim (see Part 2, Section B). Fil d'Ariane consists of four parts: a simulation program module, an optimal estimation module that applies Kalman filtering techniques to facility measurement information, an anomaly detection module for verifying parameters of spent fuel to be reprocessed, and a set of expert system modules for managing information and processing procedures in the other program elements. Fil d'Ariane is being applied to a section of the reprocessing plant test bed facility at Marcoule, France.

With Dr. Giacometti, we developed a work plan for a joint program to apply Fil d'Ariane and FacSim to different sections of a fuel-reprocessing/fuel-fabrication plant to test the effectiveness of several potential anomaly detection algorithms.

2. Technical Collaboration on Waste Assay (K. L. Coop and R. J. Estep, N-2). One Los Alamos staff member spent a day at the CEA Laboratory at Cadarache in early June to tour their waste assay facilities and develop a plan for joint experiments. Both parties agreed to plan a joint effort to develop the combined thermal/epithermal neutron (CTEN) method of waste assay for 55-gallon drums. This method of assay uses a pulsed neutron source to interrogate the waste container with both thermal and epithermal neutrons; it was originally developed at Los Alamos to assay small (1 gallon) cans of remotely handled waste.[83] The French investigators will visit Los Alamos early in 1990 to learn more about the technique, and then will configure their Promethee facility to perform joint experiments at Cadarache, tentatively beginning in the summer. Another Los Alamos staff member also visited the CEA Laboratory in November (in conjunction with a conference in Cadarache) to discuss additional aspects of the planned experiments.

3. Joint Spent-Fuel Detector Measurements with the French (P. M. Rinard, N-1). We took a Los Alamos "fork" spent fuel detector[84,85] and its GRAND-I electronics to the Cadarache Institute for Technological Research and Industrial Development for comparison with the French "Python" detector.[86] Both instruments use neutron and gross gamma-ray measurements to determine the burnups and cooling times of spent-fuel assemblies stored underwater, but have different final objectives and therefore different designs.

The fork was designed for safeguards; it is portable and takes data as quickly as is practicable. Python verifies the exposure of assemblies for criticality safety purposes before the casks are loaded for transportation. Python stays in a plant's storage pond and the assemblies are brought to it. Python is, therefore, large and heavy compared to the fork and does not need the portable electronics offered by the GRAND-I.

In the experiments, small capsules of ^{252}Cf and ^{137}Cs were placed in the center of a fuel pin and moved throughout a 17 x 17 array of fresh fuel pins. While the source pin was raised, axial profiles were measured to help determine the total responses of the instruments to complete assemblies. The measurements show the relative contributions, to the detectors' responses, of neutrons and gamma rays from different pin locations. Sums of these measurements simulate total responses and show how the instruments' responses are correlated. These measurements were repeated for different concentrations of boron in the water. In addition, neutron absorbing pins containing gadolinium were inserted into the assembly to measure the dampening of the neutron count rates.

The detector tubes in Python are much larger than those in the fork but are placed farther from an assembly. The net result is that the fork generates larger signals but is less sensitive to fuel pins at the corners of assemblies. A lead collimator in Python restricts its view of gamma rays to about a centimeter of an assembly's length, while the fork's ion chambers are uncollimated.

Examples of the measurements are shown in Figs. 48 and 49, where neutron count rates are given at different boron concentrations. (The fork has fission chambers with and without cadmium wrapping, while Python has a cadmium wrap only. Therefore, only data from cadmium-wrapped fission chambers were compared.) From such data it is concluded that the ratio of fork to Python neutron count rates is 2.74 for water with no boron and 3.16 for water with 2000 to 3000 ppm boron. All of the neutron correlations measured are shown in Fig. 50.

G. Other Bilateral Activities (E. A. Hakkila, N-4).

We provided technical input to the US government for bilateral safeguards interactions with Japan, France, the FRG, the UK, and EURATOM. A Los Alamos staff member participated in bilateral discussions with France and the UK.

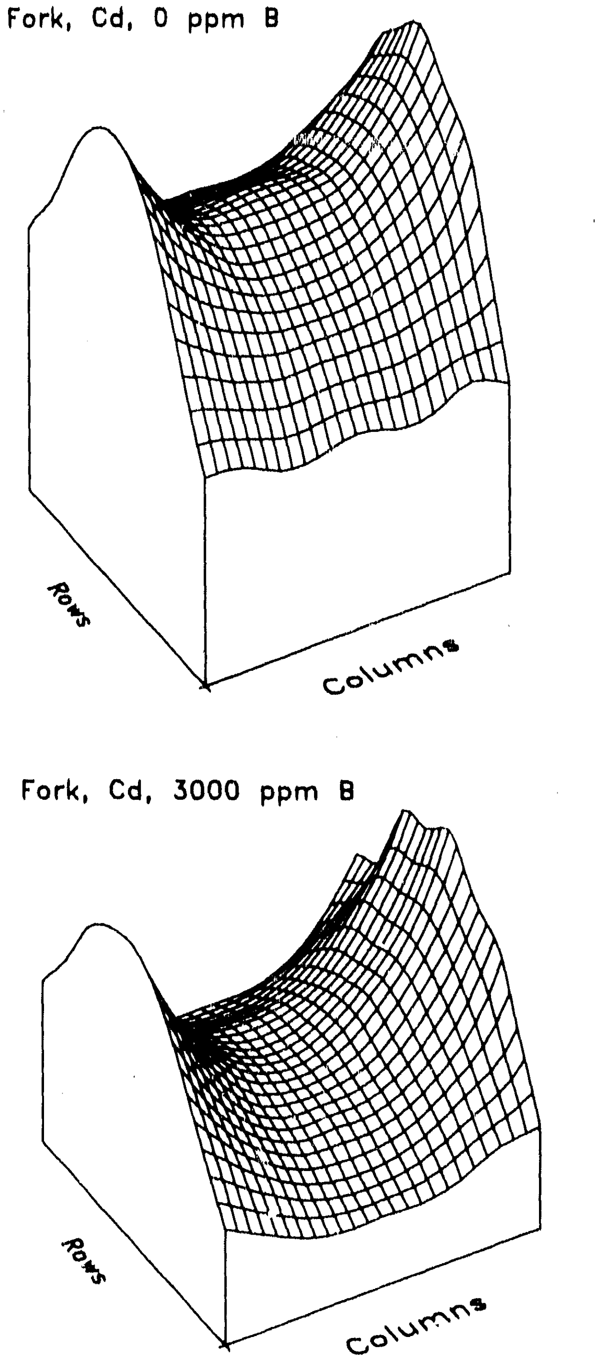
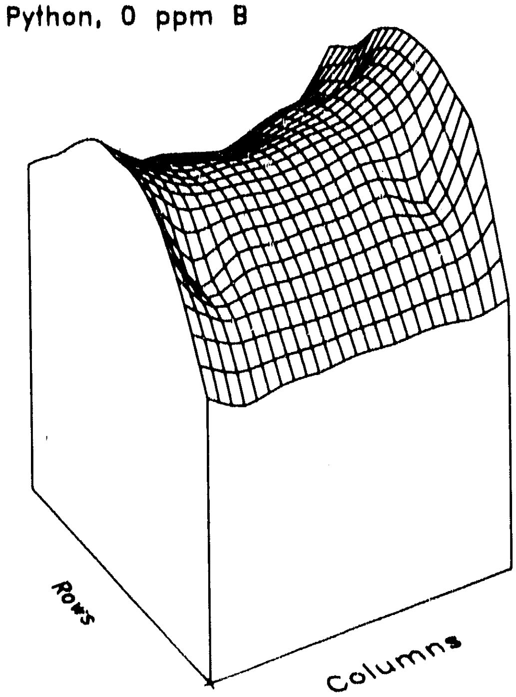


Fig. 48. Neutron count rates from the fork detector with a 17 x 17 array of fresh fuel pins vary with source-pin location as shown here for boron concentrations of 0 and 3000 ppm. One fission chamber in a tine of the fork was placed along the row axis and the other was placed along the opposite side of the assembly. The scales of the plots are identical, so the dampening effect of the boron is obvious. (The small-amplitude ripples on the boundary of the plot are artifacts of the surface-generation process.)



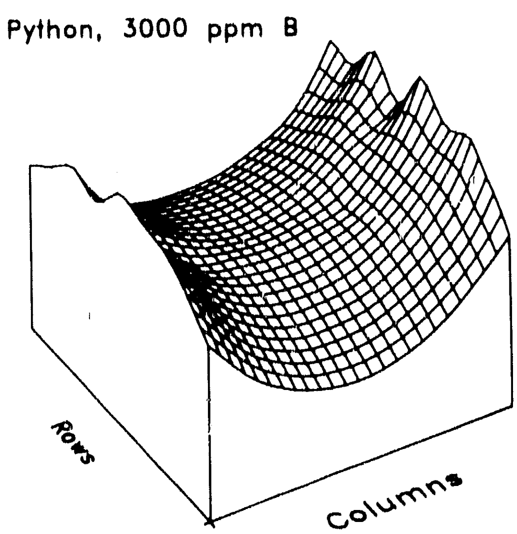


Fig. 49. These figures are similar to Fig. 45 but were taken with the Python detector under the same conditions stated for the fork.

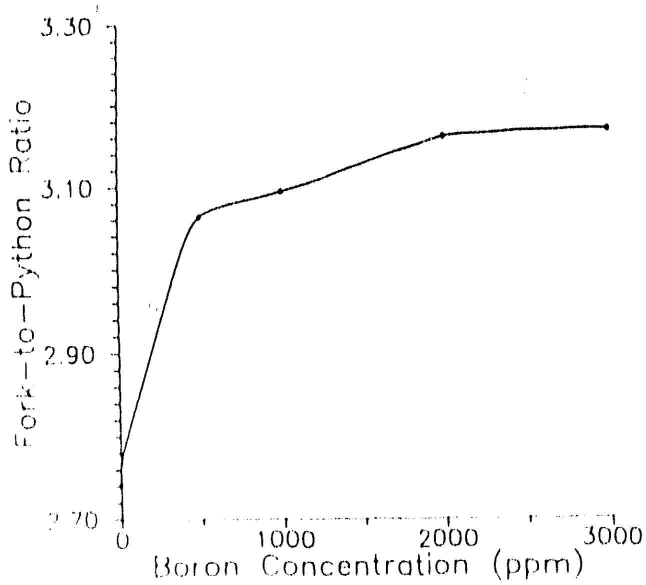


Fig. 50. The correlation between the fork and Python neutron count rates depends on the boron concentration as shown here.

III. NUCLEAR NONPROLIFERATION ACT TRAINING (T. D. Reilly, L. B. Robinson, and C. M. McCabe, N-1; E. A. Hakkila and K. K. S. Pillay, N-4).

The 7th International Training Course on Implementation of State Systems of Accounting for and Control of Nuclear Materials (SSAC) was presented 1-19 May 1989 in Santa Fe and Los Alamos, New Mexico, and Richland, Washington. This course is sponsored jointly by the IAEA and the US DOE and is mandated by a training provision in the Nuclear Nonproliferation Act of 1978. The course deals with the safeguards requirements of the Treaty on Nonproliferation of Nuclear Weapons and provides an opportunity for participants to discuss safeguards with senior IAEA personnel and senior personnel from States and facilities responsible for implementing international safeguards procedures.

The course staff was drawn from the US Departments of Energy and State; the Nuclear Regulatory Commission; Los Alamos and Sandia National Laboratories; the Advanced Nuclear Fuels Corporation (ANF); the IAEA; the EURATOM Safeguards Directorate; the Nuclear (Atomic) Energy Commissions of Argentina, Brazil, and Czechoslovakia; the ALKEM facility FRG; Ontario Hydro, Ltd. of Canada; and the Japan Nuclear Fuels Company, Ltd. Attendees came from Argentina, Brazil, Chile, Venezuela, Canada, Bulgaria, France, Sweden, Algeria, Morocco, South Africa, Zaire, Bangladesh, Indonesia, Iraq, Japan, the Peoples Republic of China, the Republic of Korea, Taiwan, and Thailand. The attendees hold important positions in their country's nuclear establishment and have primary responsibilities for nuclear materials control and IAEA interactions. Figure 51 is a photograph of the course participants in Santa Fe.

The SSAC course is divided into five major parts:

- Introduction to Nuclear Material Safeguards
- Nuclear Material Measurement Technology
- Nuclear Material Control and Accounting Theory/Practice
- Safeguards at a Model Fuel Fabrication Facility
- SSAC Design Workshop.

The participants visit Los Alamos for demonstrations and hands-on measurements of both fresh- and spent-fuel materials. They visit the Omega West Reactor and the Nuclear Safeguards Research Laboratory. Figure 52 shows one subgroup measuring a BWR fuel assembly in the old Los Alamos Safeguards Training Building. The week in Richland includes a general tour of the LWR fuel fabrication plant operated by ANF and an extensive tour of ANF's destructive and nondestructive measurement facilities.

For the SSAC Design Workshop, the attendees divide into four subgroups to design a safeguards system for a State that has a small LWR fuel fabrication plant. Significant improvements in this year's workshop included more extensive written materials and new subgroup sessions that provide a foundation for the final workshop.

The next SSAC course will be presented 6-24 May 1991 and again will emphasize accountability in bulk-handling facilities. The US course is presented every two years, alternating with an SSAC course on item-accounting facilities that is presented in Yalta, USSR. There also are several regional SSAC courses that complement the courses presented by the US and USSR. These regional courses are presented in Argentina, Australia, Brazil, and Japan.

The 1989 course was well received. In addition to promoting nuclear materials safeguards and nonproliferation goals, the course provides the attendees an opportunity to visit several parts of the United States and to establish contacts with safeguards professionals from the United States and many other countries. It provides an interesting opportunity to further communication and understanding between different parts of the world community.

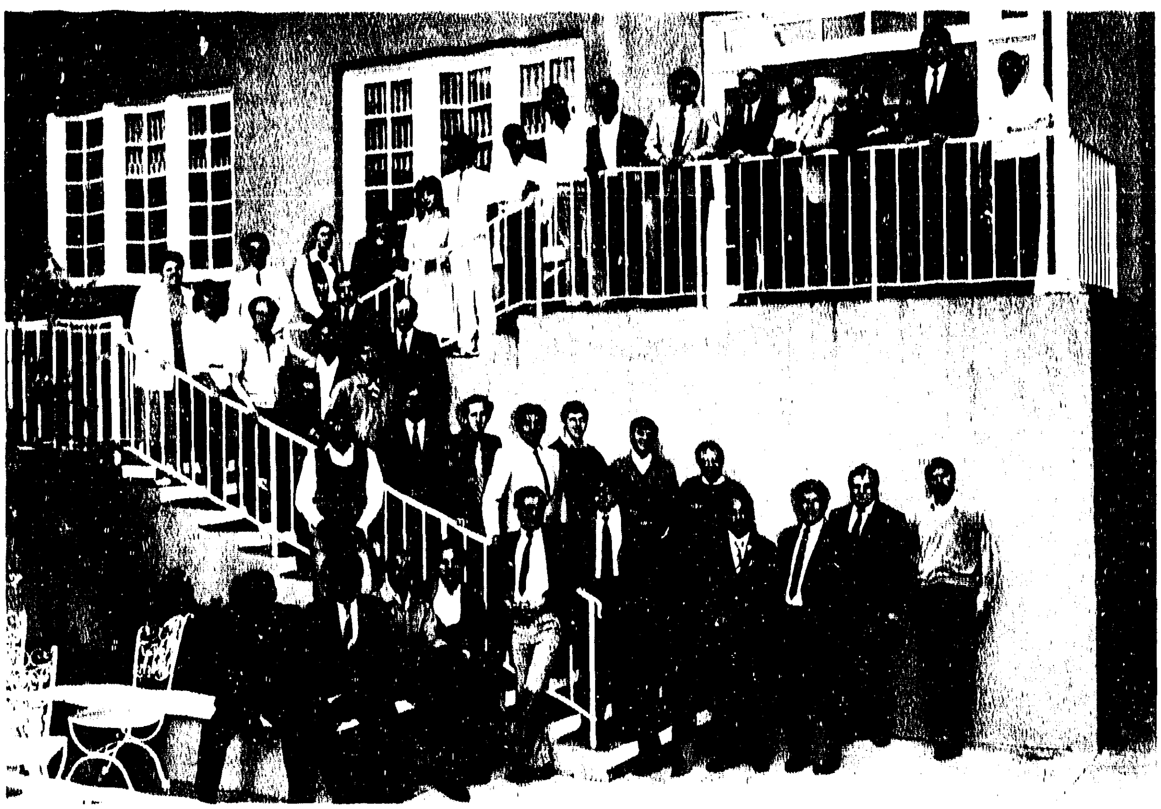


Fig. 51. SSAC course participants during 1989.

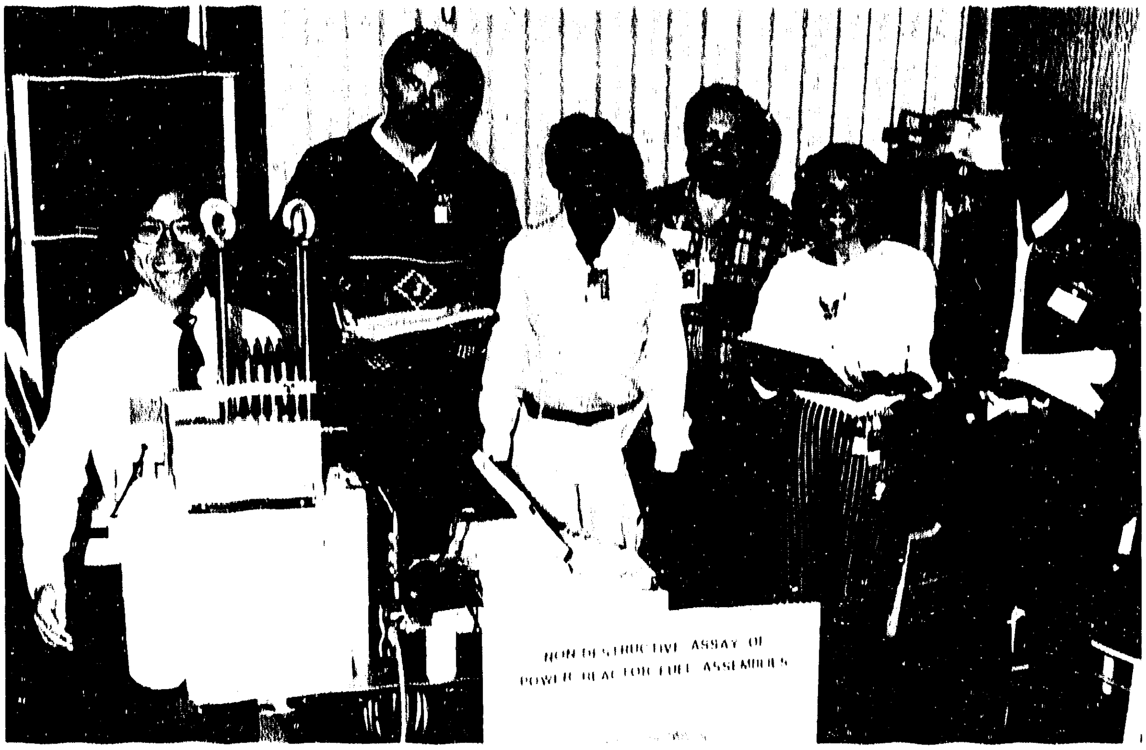


Fig. 52. One of the SSAC subgroups measures a BWR fuel assembly in the old Los Alamos Safeguards Training Building.

PART 5. RELATED SAFEGUARDS PROJECTS

In this part, we highlight R&D or implementation activities that are related to our DOE/OSS safeguards programs but are sponsored by others.

I. PROGRAM OF TECHNICAL ASSISTANCE FOR THE IAEA — POTAS

This was a busy year for POTAS tasks. A sampling of activities includes:

- Powder exercise measurement of moisture in MOX in Italy;
- Joint IAEA-USSR-US measurement of spent fuel at the LANL Omega West Reactor;
- Installation of software for the core discharge monitor at a Canadian reactor;
- Performance monitoring of continuous, unattended operation for plutonium measurements at a Fuel Fabrication Facility; and
- Development of a multiplication corrected neutron coincidence technique for measuring fresh MOX fuel assemblies underwater.

In addition, a training course for new inspectors was given and a Los Alamos staff member finished his second year in Vienna as a cost-free expert.

The IAEA has instituted a new system for support program administration designed to improve the connection between final end user and developer. We look forward to working with the IAEA in the coming year.

II. ARMS CONTROL VERIFICATION TECHNOLOGY

A. Chemical Weapons (CW) (K. E. Apt, CNSS, and J. T. Markin, N-4).

Development of a verification regime for CW agreements poses political, administrative, and technical challenges not encountered in verifying previous international treaties. With the possible exception of conventional forces, no other area of arms control (extant or proposed) has such a deeply intrusive verification framework. Although previous arms control agreements have tended to focus on weapons use and stockpiles, current concepts in CW verification extend far down into the hierarchy of preparation for use, including stockpiling, CW production, precursor production, and research.

We are developing a method for designing and evaluating a verification system that encompasses the diversity of facility types, chemicals, and activities involved in CW verification. Elements of this method are derived in part from the IAEA experience in verifying agreements with parties to the Nonproliferation Treaty, namely the systematic approach to accounting for materials in various forms. But the unique aspects of proposed CW agreements (e.g., challenge inspections and investigation of alleged usage) together with the potential resource intensiveness in monitoring large numbers of industrial facilities that process a wide variety of chemicals necessitate a new systems approach to the formulation of a CW verification regime.

Developing a system for verifying a CW agreement is essentially a problem of allocating limited resources among candidate technologies and activities to maximize the confidence that noncompliance with the agreement is detected. Our system design and evaluation method is a top-down hierarchical approach in which one begins with the general verification requirements (types of facilities, proscribed chemicals, or activities) and refines these into facility/site-specific descriptions, noncompliance scenarios, relevant verification activities and technologies, and verification criteria describing the system objectives in terms of timely detection of a material breach of the agreement.

Verification activities at each facility/site are organized in terms of key verification points, which are locations where activities or technologies are applied. Selection of these system elements at each key point is guided by the noncompliance scenarios and their component actions that may be detected by the verification activities or technologies.

Candidate verification systems are compared by evaluating the likelihood that they will detect the noncompliance scenarios. For cases in which the likelihood can be quantified as a detection probability, we choose the system providing the best overall detection of noncompliance. Alternatively, a qualitative comparison of candidate systems examines coverage of noncompliance scenarios to ensure that all scenarios are addressed by at least one verification activity and selects that system that optimizes the verification activities that are relevant to the noncompliance scenarios.

Application of a methodical approach to the definition of a verification system ensures an efficient allocation of resources among competing verification technologies and activities, provides a uniform rationale for selecting system components, and is a basis for communicating the reasons for technology decisions within the arms control community.

B. Fuel Cycle Simulation for Arms Control Verification Studies (C. A. Coulter, E. A. Hakkila, J. T. Markin, K. K. S. Pillay, W. D. Stanbro, and R. B. Strittmatter, N-4; J. W. Tape, N-DO/SG).

With support from internal Laboratory research funds and the DOE Office of Arms Control, we have been developing a simulation model of a national nuclear fuel cycle to study arms control agreements that limit military uses of fissile materials. Development of this model was continued in 1989, and we began converting the existing sections of the model to a superior new simulation language that has just become available. We presented a paper on the simulation work at the Arms Control Verification Technology Conference held at Los Alamos in August 1989. The current status of the simulation effort is summarized below.

- **Development of the model kernel.** Because the final simulation model and its associated data files will be quite large, we are taking great care in developing a core set of generic "housekeeping" procedures that will significantly reduce the effort required to develop each of the many sets of *specialized* modules that will ultimately be required in the model. An earlier version of this set of kernel routines is being reformulated in the new simulation language, and in the translation process, significant enhancements are being made.
- **Description of nuclear materials processing facilities.** We have previously developed a detailed, comprehensive computer model, FacSim, for simulating the operation of nuclear materials processing facilities. FacSim already has been used to describe portions of several DOE weapons complex facilities, including the projected SIS facility at INEL and portions of the Los Alamos Plutonium Facility, the Rocky Flats Plant, and the Savannah River Site. The FacSim model, in a separate effort, is now being converted to the same new simulation language that is being used for the arms control verification model. Many of the modules of FacSim will be incorporated into the arms control verification model with fairly minor changes–primarily in the form of *decreases* in the amount of detail needed for facility description.
- **Development of a nuclear reactor module.** We are formulating a completely new simulation module to describe the operation of nuclear reactors because this is our first involvement in reactor simulations. This module can describe arbitrary reactor types and can represent all phases of reactor operation, including shipping and receiving fresh and spent fuel, periodic or continuous fuel-change operations, and scheduled and unscheduled downtime.
- **Development of a nuclear weapons module.** We have constructed a nuclear weapons module that covers the manufacture of nuclear weapons and their associated delivery systems and platforms. The model simulates the weapon system's deployment, maintenance, and eventual retirement. The principal output of this module is the number of weapon systems of each type that are available for use.

A weapon system is considered to consist of two components: one or more *warheads* (or gravity bombs) and a *delivery system*. The weapon system is carried by a platform. As an example, a bomber (a platform) could carry several air-launched cruise missiles (delivery systems), each armed with one warhead. The bomber also might carry several gravity bombs (the delivery system in this case is notional) and a number of short-range attack missiles, each of which has its own warhead. The model has been structured to reflect the flexibility of deployment options that are available in building up a national nuclear strike system.

Production of each part of the national nuclear weapons system is controlled by an overall production manager who uses a prioritization scheme to order the manufacture. This prioritization scheme reflects the relative desirability of different weapon systems and platforms as well as the need to keep a balanced strike force. The supplies of U-235 and plutonium are constraints in the building of nuclear warheads. As the parts are produced, the commander of each platform type attempts to arm his units with the most efficient mix of available weapons. Such a choice is rather straightforward for a ballistic missile submarine or a missile silo, but could become rather complex for a bomber. The platform commander also orders unit maintenance at regular intervals depending on the requirements of the warheads, delivery systems, and platform. In addition, he orders the retirement of units with limited-lifetime components at appropriate times. At retirement, U-235 and plutonium used in the warheads are returned to a pool, after a delay, to be recycled into new warheads.

In the near future, we will enhance the model to increase its realism. Added features will include simulating competition among different warhead types for available factory capacity, recycling platforms after retiring their delivery systems, and rearming platforms during their active lives to increase their effectiveness as adequate numbers of high-priority weapons systems become available.

C. Warhead Dismantlement (E. A. Hakkila, M. F. Mullen, K. K. S. Pillay, and W. D. Stanbro, N-4).

We participated in preparing the Laboratory's position paper on warhead disposition. Los Alamos' section on Verification of Warhead Dismantlement considered problems associated with verifying numbers of warheads dismantled, recovered nuclear material, and material storage facilities.

III. SYSTEMS ANALYSIS STUDIES OF NRTA AT THE PLUTONIUM FUEL PRODUCTION FACILITY (PFPF) (K. K. S. Pillay, J. F. Hafer, and J. T. Markin, N-4; R. R. Picard, A-1)

As part of an agreement between the PNC of Japan and DOE for cooperation on research and development concerning nuclear materials control and accounting measures for safeguards, we have studied the safeguards systems for the PFPF. The objective of this project is to evaluate the NRTA system proposed for the PFPF MOX fuel fabrication facility at Tokai.

We completed Phase-I of this study during 1989 and submitted a detailed report on the project and its accomplishments to DOE (N-4/89-421) in July. The Phase-I study used design-basis data to illustrate good materials accountancy and applied a generic computer program developed earlier at Los Alamos. This computer program is known by the acronym MAWST, which stands for Materials Accountancy with Sequential Testing. We based the database for MAWST on the idealized material flow taken from design-basis data for the FBR line at PFPF and demonstrated the calculation of a materials balance using MAWST. Assuming uniform losses, we calculated covariances of sequential MUFs and demonstrated the value of sequential testing to detect protracted losses on a timely basis. We rewrote the MAWST manual for use at PFPF and transferred the MAWST program and input files to PFPF.

We began Phase-II of this study in July 1989. During this phase, we plan to consider materials accounting at PFPF based on operating data to incorporate non-steady-state material flows, more than one product specification, and other facility-specific material movements. Also, we have a parallel effort to develop a code to derive optimal sampling plans for IAEA inspection.

IV. LOS ALAMOS VULNERABILITY RISK ASSESSMENT (LAVA) (S. T. Smith, N-4)

The LAVA system is an original systematic approach to risk assessment that we developed to determine vulnerabilities and risks inherent in massive, complicated systems. Characteristics of such systems are huge bodies of imprecise data, indeterminate (and possibly undetected) events, large quantities of subjective information, and a dearth of objective information. LAVA was developed as a tool to help satisfy federal requirements for periodic vulnerability and risk assessments of a variety of systems and to satisfy the resulting need for an inexpensive, reusable, automated risk assessment tool firmly rooted in science. When the LAVA project began in 1983, there was no such tool; LAVA was designed to fill that gap.

LAVA is an alternative to existing quantitative methods; its approach is both objective and subjective and the results are both quantitative and qualitative. In addition, LAVA is used by some agencies as a self-testing aid in preparing for inspections, as a self-evaluating device in testing compliance with various orders and criteria, and as a certification device by an inspection team.

LAVA is a three-part systematic approach to risk assessment that can be used to model a variety of application systems such as computer security systems, communications security systems, information security systems, and others. The first part of LAVA is a mathematical model that is based on classical risk assessment, hierarchical multilevel system theory, decision theory, fuzzy possibility theory, expert system theory, utility theory, and cognitive science. The second part is the implementation of the mathematical risk model as a general software engine, written in a commercially available programming language for a large class of personal computers. The third part is the application data sets written for a specific application system. LAVA provides a framework for creating applications upon which the software engine operates; all application-specific information appears as data.

The user of a LAVA application is not required to have knowledge of formal risk assessment techniques. All the technical expertise and specialized knowledge are built into the software engine and the application system. LAVA applications (not all implemented as software systems) include the popular computer security application and applications for nuclear power plant control rooms, embedded systems, survivability systems, transborder data flow systems, property control systems, nuclear processing plant safeguards systems, and others. LAVA application systems have been in use by federal government agencies since 1984; the previous version of the computer- and information-security application--LAVA/CIS, Version 1.01--is used by over 100 agencies at more than 500 sites.

Recent Modifications to LAVA. Over the past three years since the release of LAVA/CIS Version 1.01, LAVA has progressed from a vulnerability assessment system (LAVA Version 1.01) to a full-blown risk assessment system (LAVA Version 2.0). Besides the major effort in developing and testing the general LAVA 2.0 software engine, four areas of applications were developed in 1989: LAVA/CIS, LAVA/FBI, LAVA/NSG, and LAVA/OPSEC.

The LAVA 2.0 General Software Engine. The LAVA 2.0 general software engine is compiled and fully self-contained and runs under MS- or PC-DOS (versions 2.0 and greater). No additional software other than DOS is required to run LAVA 2.0. It runs on the IBM-PC class of computers. Required hardware includes (1) 512 K of available random-access memory, (2) a hard disk with about 1 Mbyte of available space to store LAVA.EXE and the permanent application data sets, and

(3) a floppy disk drive for the diskette holding the volatile application data sets (the ones that are written to or altered during an assessment).

Instead of multiple code segments, LAVA 2.0 is integrated into a single menu-driven program; the menu items are selected with user-friendly light bars. Like previous versions, LAVA 2.0 applications are completely self-documented. Besides the many definition and instruction screens, the LAVA 2.0 software engine now can display specific definitions selected as needed by the user during the progress of a LAVA assessment.

Besides an updated, much-improved vulnerability assessment (VA) portion, the new version includes a consequence analysis (CA) portion, making LAVA 2.0 a full risk assessment software system. The CA portion comprises an asset-value estimation, a threat-strength estimation, and both monetary and nonmonetary (or intangible) impact analysis. Both the VA and CA sections have independent report generators. The interactive VA questionnaire segment has hotkeys for backing up in the questionnaire and for making a graceful emergency exit from the questionnaire if necessary. The VA interactive, scoring, and reporting segments can be executed without doing the CA section. The interactive portion of the CA can be executed before, after, or at the same time as the VA; however, the CA scoring and reporting segments cannot be run until after the VA has been completed. The CA format can be tailored by the user.

In addition, LAVA 2.0 includes a set of utility options that permits the user to print unanswered questionnaires, partially answered questionnaires as memory refreshers in mid-assessment, fully-answered questionnaires at the completion of the VA for documentation, and management worksheets for issue resolution. Finally, the LAVA 2.0 software engine now has color capabilities for those who have color monitors.

The data sets for computer- and information-security application (LAVA/CIS) Version 1.01 have been modified and expanded in Version 2.0. Some additional issues have been considered in the VA questionnaires, the security-requirement determination has been modified slightly, the underlying outcome set has been changed a little, and many of the VA questions have been clarified. The definition screens have been reorganized to show only one definition per screen. All data sets for the CA portion are new. All in all, the new LAVA 2.0 software engine is chock full of new features, all designed with the user in mind.

LAVA/CIS The first applications-development effort was to expand the general purpose LAVA/CIS data sets into a complete risk assessment system compatible with the new general software engine. The new version of LAVA, LAVA/CIS Version 2.0, will be ready for release on April 2, 1990. It will be released only to graduates of a LAVA training class taught by the LAVA staff. Classes will be held either at Los Alamos or at a site requested by a host agency.

LAVA/FBI The second applications-development effort was to tailor a version of the CIS of LAVA to meet the special purposes of the Federal Bureau of Investigation (LAVA/FBI). The FBI has a dual computing environment, with Headquarters and the regional computing centers in one environment and all the field offices in the other. LAVA/FBI has two parts to account for this duality. The LAVA/FBI software was delivered to the FBI in October 1989.

LAVA/NSG The third applications-development effort was to model a framework for the potential development of a nuclear safeguards application (LAVA/NSG). This consisted of developing the threat, asset, outcome, and safeguards function sets upon which an NSG application could be based. The specific software implementation of LAVA/NSG has not yet been funded.

LAVA/OPSEC The fourth applications-development effort, begun late in the year, was starting the development of an operations security application (LAVA/ OPSEC) in conjunction with (and funded by) the United States Secret Service and the National Security Agency. Progress to date on this application includes the specification of the threat, asset, and outcome sets, with a good start on defining the safeguards function sets. The software implementation of LAVA/OPSEC will continue through 1990.

V. ANOMALY DETECTION

A. Air Force (J. M. Prommel, N-4).

Because Los Alamos pioneered the theory and development of automated anomaly detection, the Air Force has requested that we take prototype software (developed elsewhere) and test it operationally and as an adversary. The Air Force also requested that Los Alamos provide the technical leadership for enhancements and make recommendations for future research.

This software, known as Haystack, was designed to detect intrusions into selected Air Force mainframes by downloading log-file information to a PC-248 for analysis. The end user is the system security officer. Haystack's goal is to reduce enormous quantities of obscure audit trail data to short summaries of user behaviors, anomalous events, and security incidents for further investigation of potential computer intrusions.

We began the project in October 1989 by preparing the code for beta testing, fixing bugs, and making minor enhancements. We now have installed the software at two beta test sites and are planning an April 1990 tiger team meeting of experts to evaluate the effectiveness of this software as well as that of our own internal effort, W&S. Haystack will undergo additional enhancement and will be released to the Air Force in September 1990.

This project benefits Los Alamos by allowing us to learn more about alternative approaches to anomaly detection and to develop criteria and methods for testing the effectiveness of this system, our own, and other systems. By so doing, we continue our leadership role in anomaly detection.

B. National Computer Security Center (NCSC) (H. S. Vaccaro, N-4).

Under a contract with the NCSC, we are testing our anomaly detection concepts for computer security and extending those concepts to mainframe and networked computer systems. This year, we configured our detection software, W&S, to analyze seven audit record formats generated by the NCSC's Dockmaster computer.

Dockmaster is used by NCSC staff and software vendors to evaluate computer security products. It also serves as a communication hub for many people involved in computer security research and development. The machine runs a B-1 rated Multics operating system, so its audit trails are of great interest. In fact, one of the earliest, and now most fully developed, expert systems for computer intrusion detection was written by NCSC staff for Dockmaster. Staff who developed this system, called MIDAS, are working with us on testing W&S and developing extensions to the W&S concepts.

In October, we delivered an IBM RT workstation and the specially configured W&S software to the NCSC for testing. After initial trials on audit data from a separate Multics testbed at the NCSC, it was decided to proceed with full testing on Dockmaster audit data.

At the present time, the Dockmaster audit data are being classified as SCI. To simplify data access, we ported W&S to one of the Sun workstations adjacent to the Dockmaster computer at NCSC. Data will be received by the Sun via an Ethernet connection. We will not have direct access either to the Sun or to the audit data; instead, NCSC staff will conduct the actual tests on Dockmaster data. This testing is expected to begin by March 1, 1990.

As part of the NCSC project, we also are testing W&S on controlled data sets. This has been carried out in collaboration with Dr. Gunar E. Liepins of the Oak Ridge National Laboratory. Thus far, tests have confirmed that W&S can detect anomalies in data at near the optimal nondetection and false alarm rates for simple data sets, although W&S was designed primarily for more complex data sets with many data fields and a sparse training set. We believe it performs as well for these design points; however, it has proven difficult to construct proper data sets and compute optimal nondetection and false alarm rates for more realistic data sets. These efforts will continue during 1990.

Major W&S software extensions are being completed under this project as well. These are being completed jointly with our DOE anomaly detection R&D project described in Part 1, Section II.F.

VI. TECHNICAL ASSISTANCE TO LOS ALAMOS

A. Closeout Activities for the Los Alamos Molecular Laser Isotope Separation (MLIS) Program (T. K. Li, P. Rinard, D. Siebelist, and C. Schneider, N-1).

The Los Alamos MLIS program for plutonium, which has been active at Los Alamos since 1982, was terminated in FY 1989. In this report period, we helped with closeout activities, including:

- the MLIS final report for the Nondestructive Assay Instruments section;
- documentation for N-1-supported NDA instruments;
- final holdup measurements for the batch tank, the collector, and traps;
- a paper on the entire set of NDA instruments that we developed for the MLIS program.

To provide important process-development and accountability information, we developed and installed the following five NDA instruments during the MLIS program:

- an at-line enrichment diagnostic system for SIS-I and -II projects;
- an in-line plutonium isotopic analysis system, consisting of two measurement stations, one for gaseous PuF_6 samples and another for solid and solution samples;
- an in-line fixed sodium iodide (NaI) monitoring system consisting of six 2 x 2-in., two 2 x 24-in., and one 2 x 22-in. NaI detectors at seven specified components in the flow loop;
- a high-resolution, portable, germanium gamma-ray system for plutonium isotopic analysis at various components and locations; and
- a portable NaI gamma-ray holdup monitor.

B. Assistance to the Los Alamos Nuclear Material Storage Facility (T. E. Sampson, S. M. Simmonds, C. M. Schneider, M. S. Krick, E. A. Kern, and H. O. Menlove, N-1; F. Hsue, OS-14).

We are continuing to work with the operators of Los Alamos' new Nuclear Material Storage Facility (NMSF) to assist in providing instrumentation for the count room.

In past years we designed and wrote procurement specifications for the commercial purchase of a newly designed passive NCC, the "flat-squared" counter.[87] We reviewed designs with the vendor in cooperation with OS Division, accepted delivery of the instrument, tested it, and installed new Los Alamos coincidence counting software.

The instrument was installed in the NMSF this past year; its use is awaiting facility startup.

In a similar fashion, we commercially procured a Los Alamos-designed AWCC, which also was installed in the NMSF this past year.

We wrote specifications for procuring two SGS instruments, then worked with the vendor to adapt his standard instrument to NMSF needs. These two instruments are currently undergoing testing in our laboratory at Los Alamos before the most current Los Alamos SGS software is installed. Delivery to the NMSF is scheduled for April 1990.

A FRAM plutonium isotopic assay system, completely designed and constructed in our safeguards laboratory, was delivered to the NMSF this past year. This system can measure two (expandable to four) samples simultaneously and will help facility operators interpret calorimetry and NCC measurements.

This project has been an excellent example of cooperation between Los Alamos groups and commercial vendors to provide needed NDA instruments and expertise in the most timely and cost effective manner.

PUBLICATIONS/PRESENTATIONS

E. L. Adams, "Destructive Testing of ^{3}He Proportional Counter Tubes," Los Alamos National Laboratory report LA-11705-MS (December 1989).

Helium-3 proportional counter tubes are useful in the nondestructive (active, passive, and delayed neutron) assay of many types of nuclear materials. It became apparent in the mid-1970s that intense neutron irradiation degraded the resolution and efficiency of the tubes. Experiments were performed to document this observation. When californium shufflers were introduced to assay fissile material, tube degradation became a concern to shuffler designers. Further testing was performed to quantify both the type of radiation and the dose necessary to damage ^{3}He tubes. This report presents the results of the study.

R. H. Augustson, "Role of IAEA Safeguards in Confidence Building," presented at Conference on Technology-Based Confidence Building, Santa Fe, New Mexico, July 9-14, 1989. Los Alamos National Laboratory document LA-UR-89-2713.

In this paper, I will examine some attributes of confidence building and connect them with how the International Atomic Energy Agency (IAEA) interacts with its member states in carrying out its safeguards function. These interactions and the structure set up to define them help maintain and strengthen confidence between the IAEA and the member states and among these states.

G. L. Barlich, "How to Write Application Code That Even a Security Auditor Could Love," presented at the 12th DOE Computer Security Group Conference, Amarillo, Texas, May 2-4, 1989.

In the past, the application programmer was frequently isolated from the computer security professional. The target machine might have various access controls and security plans, but when the programmer delivered a new application, it was rarely scrutinized from a security standpoint. Security reviews of application code are now being used to overcome this apparent oversight, but these reviews are often hampered by a lack of knowledge among programmers of techniques that make code secure and facilitate security analysis of the code. This paper informally describes 15 general principles for producing good code that is easily reviewed. This paper is not a formal guideline, but is intended as an inside view of how one reviewer looks at code from a security standpoint.

G. L. Barlich, "A PC-Based Analysis Package for Measurement Control," presented at the 30th Annual Meeting of the Institute of Nuclear Materials Management, Orlando, Florida, July 9-12, 1989; *Nucl. Mater. Manage.* **XVIII** (Proceedings Issue), 446-447 (1989).

Systematic measurements of standards for measurement-control purposes can generate large volumes of data for analysis. Hand charting is tedious and error-prone, so automated computer analysis is one solution. Commercial software packages are available for such tasks, but licenses for these packages can be expensive and some packages are not available for the personal computers used at most sites. The Safeguards Systems Group at Los Alamos developed two simple measurement-control packages to meet the needs described. Users can execute two versions of MCCAT (Measurement Control Charts and Tests) from source code and in the BASIC or dBase III languages. The analytical tools in this package are described and their use is explained.

G. L. Barlich and T. A. Kelly, "Software Design for the Instrument Control Computer at New Special Recovery," Los Alamos National Laboratory report LA-11449-MS (April 1989).

This report describes the software supplied by Los Alamos National Laboratory to accompany the Instrument Control Computer, a DEC VAX-11/750, which is one part of an integrated nondestructive assay system for the New Special Recovery Facility at the Savannah River Plant. A short history of the project is provided, and the system and the code structure are described.

R. C. Bearse and R. M. Tisinger, "SNM Accounting Systems - dBase Versus C," presented at the 30th Annual Meeting of the Institute of Nuclear Materials Management, Orlando, Florida, July 9-12, 1989; *Nucl. Mater. Manage.* **XVIII** (Proceedings Issue), 648-653 (1989).

The Fuel Manufacturing Facility (FMF) at Argonne National Laboratories-West (ANL-W) in Idaho Falls accomplishes its internal special nuclear material accounting with a PC-based DYnamic Material ACcounting (PC/DYMAC) system developed as a collaboration between FMF and Los Alamos National Laboratory staff members. This system comprises four computers communicating via floppy disks containing transfer information. The accounting software was written in dBASE and compiled under Clipper. The decision was made to network the computers and to speed the accounting process. Moreover, it was decided to extend the collaboration to Sandia National Laboratory staff and to incorporate their recently developed CAMUS and WATCH systems to automate data input and to provide a measure of material control. The current version of the code is being translated into the C language. The implications of such a change will be discussed.

G. E. Bosler and P. M. Rinard, "Safeguards Measurements for Long-Term Storage of Spent Fuel," presented at ANS 1989 Winter Meeting, San Francisco, California, November 26-30, 1989. Los Alamos National Laboratory document LA-UR-89-2238.

No abstract

G. E. Bosler, S. F. Klosterbuer, C. S. Johnson, W. R. Hale, R. D. Marsh, and R. J. Dickinson, "Unattended System for Monitoring Skip Movement at the Sellafield Facility in the United Kingdom," presented at INMM 30th Annual Meeting, Orlando, Florida, July 9-12, 1989. Los Alamos National Laboratory document LA-UR-89-2241.

An unattended system for monitoring spent-fuel movement in the storage area of a reprocessing facility has been developed and tested. The system uses radiation detectors to determine when fuel is being moved and a video system to record images of the container movement. In addition to the recorded image, other recorded data include the date and time of the movement and "fingerprint" information from the radiation detectors. The direction of motion either into or out of the storage pond is indicated on the video image and on the printed readout. This system was extensively tested at the Sellafield Facility in the United Kingdom. This paper gives the details of the system design and presents results of the field evaluation.

G. S. Brunson and G. J. Arnone, "A New System for Analyzing Neutron Multiplicities: Characterization and Some Specific Applications," Los Alamos National Laboratory report LA-11701-MS, Nov. 1989.

We use a RAM chip to perform the functions of 15 parallel shift registers. This provides separate pulse handling for up to 15 independent detector channels, which results in a significant reduction of dead time in the system as a whole. Our algorithm extracts parameters describing the statistics of individual pulse clusters (pulses from correlated neutrons). These parameters include several probability ratios and moments of various orders. The performance of the system has been examined with respect to the following factors: multiplication non-correlated background, counter sensitivity, time width of the window, and identity of the spontaneously fissioning isotopes (^{238}Pu, ^{240}Pu, ^{242}Pu, ^{244}Cm, and ^{252}Cf). The system gives the same chain characteristics for ^{252}Cf for count rates varying over three orders of magnitude, thereby demonstrating successful unfolding of overlapping chains.

"Computer Security Test Plan for the Westinghouse-Hanford Restricted Data Transmission Controller," Los Alamos National Laboratory, DOE Center for Computer Security report CCS-89-02 (March 1989).

No abstract.

K. L. Coop, "A Combined Thermal/Epithermal Neutron Interrogation Device to Assay Fissile Materials in Large Containers," Proceedings of the 11th Annual Symposium on Safeguards and Nuclear Material Management, European Safeguards Research and Development Association, European Centre Luxembourg, May 30 - June 1, 1989.

We used Monte Carlo techniques to investigate the neutronic properties of a device designed to interrogate large waste containers with both thermal and epithermal neutrons. The interrogating spectrum is obtained by gradually slowing down neutrons from a pulsed source. Shortly after the pulse, the neutrons are predominantly at epithermal energies, becoming completely thermalized at later times. We calculated the effects of using different moderating materials, cavity sizes, and container matrices on the interrogating flux. Such a device could detect "lumpy" or cadmium-shielded uranium in non-hydrogenous matrices and could provide assays of finely divided fissile materials, regardless of hydrogen content.

K. L. Coop, "Neutron Dieaway Methods for Criticality Safety Measurements of Fissile Waste," Proceedings for the International Topical Meeting on Safety Margins in Criticality Safety, American Nuclear Society, Nov. 26-30, 1989, pp. 143-152.

The differential dieaway technique (DDT), which uses a pulsed neutron source to interrogate containers of fissile materials with thermal neutrons, is reviewed. This method is widely used for certifying transuranic nuclear wastes for eventual emplacement at the Waste Isolation Pilot Plant. For purposes of criticality safety, an upper limit of 200 g of fissile material is permitted in a 55-gal waste drum. Problems involving waste-matrix effects and self-shielding may severely limit the accuracy of the DDT measurement. A dieaway method that uses both thermal and epithermal neutron interrogation, which has the potential for reducing these problems, is being developed. Recent experimental and calculational results for this development are described.

J. S. Dreicer and W. J. Hunteman, "Los Alamos Center for Computer Security Formal Computer Security Model," presented at the 30th Annual Meeting of the Institute of

Nuclear Materials Management, Orlando, Florida, July 9-12, 1989; *Nucl. Mater. Manage.* **XVIII** (Proceedings Issue), 236-238 (1989).

This paper provides a brief presentation of the formal computer security model currently being developed at the Los Alamos Department of Energy (DOE) Center for Computer Security (CCS). The need to test and verify DOE computer security policy implementation first motivated this effort. The actual analytical model was a result of the integration of current research in computer security and previous modeling and research experiences. The model is being developed to define a generic view of the computer and network security domains, to provide a theoretical basis for the design of a security model, and to address the limitations of present formal mathematical models for computer security. The fundamental objective of computer security is to prevent the unauthorized and unaccountable access to a system. The inherent vulnerabilities of computer systems result in various threats from unauthorized access. The foundation of the Los Alamos DOE CCS model is a series of functionally dependent probability equations, relations, and expressions. The model is undergoing continued discrimination and evolution. We expect to apply the model to the discipline of the Bell & LaPadula abstract sets of objects and subjects.

N. Ensslin, "Development of Neutron Multiplicity Counters for Safeguards Assay," presented at Institute of Nuclear Materials Management (INMM), INMM Annual Meeting, Orlando, Florida, July-5-9, 1989. Los Alamos National Laboratory document LA-UR-89-2066)

This paper reports on the development of a new generation of neutron multiplicity counters for assaying impure plutonium. The new counters will be able to obtain three measured parameters from the neutron multiplicity distribution and will be able to determine sample mass, multiplication, and (α,n) reaction rate, making it possible to obtain a more matrix-independent assay of moist or impure materials. This paper describes the existing prototype multiplicity counters and evaluates their performance using assay variance as a figure of merit. The best performance to date is obtained with a high-efficiency, low die-away-time thermal neutron counter with shift-register electronics.

R. J. Estep, "A Computer Study of Source-Detector Geometries Proposed for the Induced-Emission Tomography of Transuranic Elements," Los Alamos National Laboratory Program Technical Note LA-N2TN-89-477RE, June 6, 1989.

We describe two unusual source-detector geometries that are proposed for the tomographic imaging of transuranic elements using beam interrogation. Both geometries demonstrate a high inherent imaging power. A proposed bremsstrahlung beam imaging method proves to be feasible for imaging the important uranium isotopes. Open geometry imaging methods, which use uncollimated detectors, require high-quality data to produce faithful images.

R. J. Estep, K. L. Coop, T. M. Deane, and J. E. Lujan, "A Passive-Active Neutron Device for Assaying Remote-Handled Transuranic Waste," Topical Meeting on Nondestructive Assay of Radioactive Waste CEN, Cadarache, France, November 17-22, 1989.

A combined passive-active neutron assay device was constructed for assaying remote-handled transuranic waste. A study of matrix and source position effects in active assays showed that a knowledge of the source position alone is not sufficient to correct for position-related errors in highly moderating or absorbing matrices. An alternate function for the active assay of solid fuel pellets was derived, although the efficacy of this approach remains to be established.

P. E. Fehlau, "SNM-Detection Technology Update: A Review of Recently Developed Equipment," talk and handout for the Sandia National Laboratories Workshop on Entry-Control Technology, Albuquerque, NM, April 18-20, 1989 Los Alamos National Laboratory document LA-UR-89-1290.

Los Alamos has been a center for developing and evaluating radiation instruments for measuring and monitoring gamma-ray and neutron emissions from special nuclear materials (SNM) since 1972. During that period, we studied and developed different methods and types of equipment for detecting SNM radiation, evaluated much of the commercially available SNM monitoring equipment, and transferred to both manufacturers and users technology and information on how to best apply and maintain SNM detecting equipment. This review concentrates on our most recent experience in developing and evaluating new monitoring equipment.

P. E. Fehlau, W. S. Murray, K. B. Butterfield, and H. F. Atwater, "Hand-Held Verification Instruments for Intrinsic Radiation Detection," Poster paper presented at the Conference on DOE Technology Research and Development for Arms Control and Verification, Los Alamos, NM, August 29-31, 1989. Los Alamos National Laboratory Computer program LA-CP-89-332.

The Los Alamos Advanced Nuclear Technology Group has an extensive background in developing portable, radiation-detecting instruments for applications in Nuclear Safeguards and for the DOE Nuclear Emergency Search Team. Recently, the Group built on this experience to develop neutron and gamma-ray instruments for possible application to two forms of verification: before-launch, non-nuclear verification of test warheads and arms control treaty verification. Three recently developed instruments are a hand-held, stabilized, gamma-ray spectrometer; a hand-held, self-contained neutron scintillation detector; and a briefcase-size, self-contained ^{3}He neutron detector.

P. E. Fehlau, "Calibrating the Jomar JPM-22 Pedestrian SNM Monitor," Los Alamos National Laboratory report LA-11643-M (August 1989).

The Jomar JPM-22 is a commercial version of a portal monitor developed at Los Alamos. The monitor operates as a walk-through or wait-in SNM monitor. This manual describes how to calibrate the monitor's detection system and how to set its operating parameters for walk-through or wait-in operation.

R. G. Gutmacher, "Safeguards Approaches for an Industrial-Scale Reprocessing Plant," Los Alamos National Laboratory, Safeguards Systems Group, report N-4/89-566 (September 1989).

No abstract.

E. A. Hakkila, J. W. Barnes, and R. G. Gutmacher, "Near-Real-Time Verification Approaches for the Process Area of Reprocessing Plants," *Proceedings of the 11th ESARDA Annual Symposium on Safeguards and Nuclear Material Management* (Joint Research Centre, Ispra, Italy, 1989), ESARDA **22**, pp.71-76.

Adoption of near-real-time accountancy in large reprocessing plants will necessitate more timely verification. We discuss techniques and instruments that are suitable for on-site verification of input, output, waste streams, and in-process inventory estimation of tanks, solvent extraction contactors, and concentrators. Calculations show that estimates of solvent extraction contactor inventories may make an insignificant contribution to the total uncertainty of the material balance, relative to the contributions by transfer and process tank inventory measurements.

E. A. Hakkila, "New Trends in Safeguards Measurement Technology," presented at the 2nd Karlsruhe International Conference on Analytical Chemistry in Nuclear Technology, Karlsruhe, Federal Republic of Germany, June 5-9, 1989.

No abstract

E. A. Hakkila, J. W. Barnes, and R. G. Gutmacher, "Near-Real-Time Verification Approaches for the Process Area of Reprocessing Plants," Los Alamos National Laboratory report LA-11615-MS (June 1989).

Near-real-time accounting is being considered as a technique for improving accounting timeliness in reprocessing plants. A major criticism of near-real-time accounting is perceived disclosure of proprietary data for International Atomic Energy Agency verification, particularly in verifying the inventory of solvent extraction contactors. This study indicates that the contribution of uncertainties in estimating the inventory of pulsed columns or mixer settlers may be insignificant compared with uncertainties in measured throughput and measurable inventory for most reprocessing plants; thus, verification may not be a serious problem. Verification can become a problem for plants with low throughput and low inventory in process tanks if contactor inventory variations or uncertainties are greater than 25%. Each facility must be evaluated with respect to its specific inventory and throughput characteristics.

E. A. Hakkila, "Application of Process Monitoring to Verify Facility Design," presented at the 30th Annual Meeting of the Institute of Nuclear Materials Management, Orlando, Florida, July 9-12, 1989; *Nucl. Mater. Manage.* **XVIII** (Proceedings Issue), 572-575 (1989).

Process monitoring has been proposed as a safeguards measure to ensure that a facility is operating as designed, or as a surveillance measure to ensure that material is not removed from the facility in an undeclared manner. In a process-monitoring system, the facility operator monitors process operations such as tank levels, densities, and temperatures; process flows; and physical parameters such as valve positions to ensure that the operations performed are both desired and required. At many facilities (for example, Idaho), the process-monitoring system is also an important safety feature to prevent criticality.

Verifying facility design is necessary for application of safeguards in a reprocessing plant. Verifying all pipes and valves through comparison of blueprints with the as-built facility is an almost impossible task with the International Atomic Energy Agency's limited inspection resources. We propose applying process monitoring for international safeguards facility design verification. By carefully selecting process-operating variables, it may be possible to verify that plant flows are as described and that key measurement points are not bypassed.

L. M. Harris, "A Nitty Gritty, Structured Approach to Contingency Plan Testing," presented at the 12th DOE Computer Security Group Conference, Amarillo, Texas, May 2-4, 1989; Los Alamos National Laboratory document LA-UR-89-1003.

An important aspect of the contingency planning/disaster recovery process is the on-going test and evaluation of the contingency plan. The purpose of testing is to evaluate a facility's emergency response preparedness by examining the recovery procedures and activities as they are documented in the contingency plan to ensure that, when executed, they successfully recover computer operations.

Testing is proof of a contingency plan's effectiveness. This paper presents the Computer System Security Officer with a practical approach to testing a contingency plan by suggesting a structured approach over the life cycle of the contingency plan. Different methods of testing are presented for consideration, and the benefits of each method are discussed. The paper addresses the following as they relate to DOE Order 5637.1: the use of test plans and test scripts, frequency of testing, and documenting of test results.

S.-T. Hsue and R. Zhu, "Poor Man's Densitometry," presented at Institute of Nuclear Materials Management (INMM), INMM Annual Meeting, Orlando, Florida, July 9-12, 1989. Los Alamos National Laboratory document LA-UR-89-1862.

We have developed two novel methods of determining plutonium concentrations (and isotopic distribution) that require no external radioactive sources or x-ray generators but rely only on the natural radiation from the plutonium. The methods are ideally suited to assay reasonably pure plutonium solutions, such as product solutions of reprocessing plants and eluate solutions from anion exchange columns. The methods can be applied to aged or freshly separated plutonium and can be used to measure plutonium concentrations in pipes or tanks. Because these methods do not require expensive equipment, we call them "Poor Man's Densitometry."

S.-T. Hsue, D. G. Langner, V. L. Longmire, H. O. Menlove, P. A. Russo, and J. K. Sprinkle, Jr., "Measurements of Plutonium Residues from Recovery Processes," presented at Topical Meeting on Nondestructive Assay of Radioactive Waste, Cadarache, France, November 20-22, 1989. Los Alamos National Laboratory document LA-UR-89-3699.

Conventional methods of nondestructive assay (NDA) have accurately assayed the plutonium content of many forms of relatively pure and homogeneous bulk items. However, physical and chemical heterogeneities and the high and variable impurity levels of many categories of processing scrap bias the conventional NDA results. The materials also present a significant challenge to the assignment of reference values to process materials for purposes of evaluating the NDA methods.

A recent study using impure, heterogeneous, pyrochemical residues from americium molten salt extraction (MSE) has been aimed at evaluating NDA assay methods based on conventional gamma-ray and neutron measurement techniques and enhanced with analyses designed to address the problems of heterogeneities and impurities. The study included a significant effort to obtain reference values for the MSE spent salts used in the study. Two of the improved NDA techniques, suitable for in-line assay of plutonium in bulk, show promise for timely in-process assays for one of the most difficult pyrochemical residues generated as well as for other impure heterogeneous scrap categories.

W. J. Hunteman, "Lessons Learned in the DOE Computer Security Enhancement Review Program," Los Alamos National Laboratory report LA-11614-MS (July 1989).

During the last four years, DOE has made the most detailed and extensive computer security self-evaluation of any U.S. government organization. The breadth and depth of the examination have revealed some problems. Few of the problems are major; most are procedural, some administrative, a few technical, and almost none systemic. This report documents the lessons learned from one part of the evaluation process.

W. J. Hunteman, "Lessons Learned in the DOE Computer Security Enhancement Review Program," presented at the 12th DOE Computer Security Group Conference, Amarillo, Texas, May 2-4, 1989.

During the last four years, DOE has made the most detailed and extensive computer security self-evaluation of any US Government organization. The breadth and depth of the examination has revealed some problems. A few of the problems are major, most are procedural, some administrative, a few technical, and almost none systemic. The DOE facilities have received a thorough and systematic examination by some of the most knowledgeable people in the United States. The examinations were conducted in a nonadversarial manner and at minimal cost to the Government.

The reviews were conducted as part of the DOE Center for Computer Security (CCS) Computer Security Enhancement Review (CSER) program. Almost all of the computer security problems found during the reviews involved some form of lack of management or user awareness. Problems of this type do not admit to technical solutions. Improving management and user knowledge of the problems and providing access to expert information will help correct these problem areas.

DOE Order 5637.1 establishes policies for most of the computer security issues identified in this report. DOE and DOE contractor sites need additional information, e.g., guides, that outline policy implementation. Development of the guides will provide DOE contractors with suggested approaches for efficient implementation of DOE policies. These guides should contain suggested methods to implement the policy without becoming part of the policy documentation. Maintaining independent guides will allow rapid updating to reflect changes in technology and computer security policy implementation. If the guides are useful, they will be used by the sites as part of their everyday toolkit.

This paper is not an indictment of the DOE Classified Computer Security Program or any DOE site. Most systems in DOE or DOE contractor facilities do not contain any of the problem areas mentioned. The proper interpretation of these findings is that the DOE classified computer security program is very strong. The largest problems confronting the program are awareness at all levels and the dissemination of computer security information and solutions.

The program has been, and continues to be, a success!

W. J. Hunteman, "A Computer Security Advisor," presented at the Intelligence Community's 7th Annual AI Symposium, Langley, Virginia, October 4-6, 1989; Los Alamos National Laboratory document LA-UR-89-3242.

The rapid expansion of computer security information and technology has provided little support for the security officer to identify and implement the safeguards needed to secure a computing system. The Department of Energy Center for Computer Security is developing a knowledge-based computer security system to provide expert knowledge to the security officer. The system is designed to provide an integrated collection of policy requirements, safeguards, potential attack scenarios, and expert knowledge to the security officer. The design goals for the system include development of a comprehensive list of safeguards based on policy requirements; collection of a detailed description of the local computing environment; production of a list of safeguards that are applicable to the local computing environment with guidance on the required implementation approach for each safeguard; support "what-if" experimentation to allow the security officer to adjust the local computing environment or reject specific safeguards because of resource limitations; generate, on request, ADP Security Plans and Security Test Plans; and provide, on request, justification or explanation of each decision throughout the process.

S. F. Klosterbuer, E. A. Kern, J. A. Painter, S. Takahashi, "Unattended Mode Operation of Specialized NDA Systems," presented at INMM 30th Annual Meeting, Orlando, Florida, July 9-12, 1989. Los Alamos National Laboratory document LA-UR-89-2151.

Nondestructive assay systems have been developed to allow data acquisition equipment to operate unattended in an automated mixed oxide facility, reducing inspector time in a facility and giving them time for other activities. Fewer inspector visits mean less impact on plant operators. Neutron detectors are located at key measurement points in the facility. Near each detector is located an electronics cabinet, which contains two JSR-11 shift registers, two COMPAQ Portable III computers, and a printer. The signal from the detector is split and sent to each shift register for redundancy and reliability. The software for unattended operation consists primarily of two programs, COLLECT and REVIEW. The COLLECT program runs on the computers in unattended operation; shift-register data are acquired each 60 s. The COLLECT program distinguishes between a normal background and a disconnected signal, between material moving near the detector and material in the detector, and whether the material in the detector is a sample or a californium normalization source. Depending on the type of assay, different data are stored on the hard disk. During an inspection, the inspector stops the current measurement campaign, examines the data from both computers briefly at the electronics cabinet, copies the campaign data to floppy disk, and starts another measurement campaign. These data are examined later in another location using the REVIEW program running on high performance microcomputers: a COMPAQ DeskPro 386/20 or equivalent. The REVIEW program uses graphical displays to enable the inspector to quickly search through the massive amounts of accumulated data to learn when samples were measured. Data from the desired measurements are then transferred to the International Atomic Energy Agency high-level neutron coincidence program for further analysis.

M. S. Krick, L. Osborne, P. J. Polk, J. D. Atencio, and C. Bjork, "An In-Line Thermal-Neutron Coincidence

Counter for WIPP Certification Measurements," Los Alamos National Laboratory report LA-11674-M (October 1989).

A custom-designed, in-line, thermal-neutron coincidence counter has been constructed for the certification of plutonium waste intended for storage at the Waste Isolation Pilot Plant. The mechanical and electrical components of the system and its performance characteristics are described.

M. S. Krick, P. J. Polk, and J. D. Atencio, "Maintenance Neutron Coincidence Counter Manual," Los Alamos National Laboratory report LA-11659-M (ISPO-304) (September 1989).

A compact thermal-neutron coincidence counter has been constructed specifically for use by the International Atomic Energy Agency as a reference neutron detector for maintenance activities. The counter is designed for use only with ^{252}Cf sources in SR-CF-100 capsules. This manual describes the detector's mechanical and electrical components and its operating characteristics.

T. K. Li, P. M. Rinard, C. M. Schneider, J. D. Atencio, D. H. Hyman, K. E. Kroncke, J. A. Painter, R. Siebelist, and O. Holbrooks, "Development of NDA Instruments for the Los Alamos SIS Facility," presented at INMM 30th Annual Meeting, Orlando, Florida, July 9-12, 1989. Los Alamos National Laboratory document LA-UR-89-2259.

The Los Alamos Special Isotope Separation Facility produces special plutonium isotopes and converts plutonium scrap by using the molecular laser isotope separation (MLIS) process in a gaseous plutonium hexafluoride (PuF_6) phase. To provide important process-development and accountability information, we have developed and installed four nondestructive assay (NDA) instruments for that facility. These instruments are (1) an in-line plutonium isotopic analysis system to measure plutonium isotopes in gaseous, solid, and liquid phases, (2) an in-line sodium iodide (NaI) monitoring system consisting of six 2-in. by 2-in., two 2-in. by 24 in., and one 2-in. by 22-in. NaI detectors at specified components (a feed bottle, as feed-transfer cold trap, a compressor, a heat exchanger, a collector, a nozzle prefilter, and a tails cold trap) in the flow loop, (3) a portable high-resolution germanium gamma-ray system for plutonium isotopic analysis, and (4) a portable NaI gamma-ray holdup monitor.

This paper discusses the measurement principles, hardware and software designs, and performance associated with these NDA instruments.

G. E. Liepins and H. S. Vaccaro, "Anomaly Detection: Purpose and Framework," presented at the 12th National Computer Security Conference, Baltimore, October 10-13, 1989; *Proceedings 12th National Computer Security Conference* (National Computer Security Center, Ft. Meade, Maryland, 1989), pp. 295-504.

This paper places anomaly detection of computer use in the framework of overall computer security. A balance of physical security, access security, anomaly detection, misuse detection, and database management is proposed to provide the maximum practical security for computer systems. The fundamental concepts of the anomaly detection module Wisdom and Sense (W&S), including rule representation, rule generation, rule pruning, and evidence combining, are presented.

J. T. Markin, "Randomization of Inspections," presented at the 11th ESARDA Symposium on Safeguards and Nuclear Material Management *Proceedings 11th Annual Symposium on Safeguards and Nuclear Material Management* (Joint Research Center, Ispra, Italy, 1989), ESARDA **22**, pp. 253-258.

Assignment of inspection resources among facilities inspected by the International Atomic Energy Agency (IAEA) is a complex and important function affecting the quality of these inspections and the safeguards conclusions derived from them. Although the IAEA currently meets essentially all of its safeguards goals, future increases in the number of large bulk facilities combined with limited growth in inspection resources may reduce the attainment of those goals. This report examines the potential role of randomized inspections in improving the effectiveness and efficiency of inspection activities despite relative resource reductions. Randomization of the inspection activities at a single facility and randomization of inspection among groups of facilities are considered.

J. T. Markin, "Future Directions for Safeguards Design/Evaluation Tools," presented at the 1989 ANS Winter Meeting, San Francisco, November 26-30, 1989; *Trans. Am. Nucl. Soc.* **60**, 218 (1989).

No abstract.

D. P. Martinez, "VMS ALAP 1.0: An Automated Audit Trail Analysis Tool," presented at the 12th DOE Computer Security Group Conference, Amarillo, Texas, May 2-4, 1989; Los Alamos National Laboratory document LA-UR-89-1511.

Because multiuser computer systems typically record enormous quantities of information about user and system activities into system log files, auditing computer user/system activities is a formidable task. Recognizing that a manual audit of these log files is

difficult and usually ineffective, the DOE Center for Computer Security has developed an automated audit trail analysis tool, Audit Log Analysis Package (ALAP). ALAP employs methodology developed at Los Alamos National Laboratory for the detection and analysis of anomalous data in large databases. ALAP is capable of processing vast amounts of audit data for detection and analysis of anomalous computer user and system behavior. The first application tool is VMS ALAP 1.0, targeted for Digital Equipment Corporation (DEC) Virtual Memory Systems (VMS).

D. P. Martinez, K. Redle, C. Steverson, and K. Swartz, "VMS ALAP 1.2 for Computer Security Auditing," Los Alamos National Laboratory, DOE Center for Computer Security, manual LA-UR-89-3364.

No abstract

H. O. Menlove, R. H. Augustson, T. Ohtani, M. Seya, S. Takahashi, R. Abedin-Zadeh, B. Hassan, and S. Napoli, "Remote-Controlled NDA Systems for Feed and Storage at an Automated MOX Facility," presented at INMM 30th Annual Meeting, Orlando, Florida, July 9-12, 1989. (LA-UR-89-2152)

Nondestructive assay (NDA) systems have been developed for use in an automated mixed oxide (MOX) fabrication facility. Unique features have been developed for the NDA systems to accommodate robotic sample handling and remote operation. In addition, the systems have been designed to obtain International Atomic Energy Agency inspection data without the need for an inspector at the facility at the time of the measurements. The equipment is being designed to operate continuously in an unattended mode with data storage for periods of up to one month. The two systems described in this paper include a canister counter for the assay of MOX powder at the input to the facility and a capsule counter for the assay of complete liquid-metal fast breeder reactor fuel assemblies at the output of the plant. The design, performance characteristics, and authentication of the two systems will be described. The data related to reliability, precision, and stability will be presented.

H. O. Menlove, M. M. Fowler, E. Garcia, A. Mayer, M. C. Miller, R. R. Ryan, "The Measurement of Neutron Emission from Ti Plus D_2 Gas," presented at the Workshop on Cold Fusion Phenomena, Santa Fe, New Mexico, May 23-24, 1989. Los Alamos National Laboratory document LA-UR-89-1570.

We have measured neutron emissions from cylinders of pressurized D_2 gas mixed with various forms of Ti metal chips and sponge. For some of the cases, the Ti was coated with a surface layer of Pd. The gas pressure ranged from 20 atm to 50 atm, and the Ti loadings ranged from 30 g to 200 g.

The neutrons were measured using a high efficiency (34%) cavity-type detector containi g 18 ^{3}He tubes. Random neutron emissions were observed as well as time-correlated neutron bursts. The time spread in an individual burst was less than 200 μs.

The neutron emission was observed after the cylinder had cooled in liquid nitrogen temperature and was warming to room temperature. The bursts occurred about 40 minutes into the warm-up phase, and the random emission occurred for at least 12 hours after the sample reached room temperature. This cycle could only be repeated two or three times before neutron emission ceased.

The neutron emission rates were very low and the 12-hour random emission rate was 0.05-0.2 n/s. However, this yield was still 11σ above the background. The instantaneous neutron bursts were more dramatic with yields several orders of magnitude above the coincidence background rates.

H. O. Menlove and E. Garcia, "Update on the Measurement of Neutron Emission from Ti Samples in Pressurized D2 Gas," presented at the Cold Fusion Workshop, Washington, DC, October 18, 1989. Los Alamos National Laboratory document LA-UR-89-3633.

During the Workshop on Cold Fusion Phenomena, Santa Fe, New Mexico, May 2, 1989, we reported [Los Alamos National Laboratory report LA-11686-C (September 1989)] on the measurement of neutrons emitted during pressurized D_2 gas experiments using Ti and Pd samples. The experimental program has continued since the Santa Fe meeting, and our data base has more than doubled.

Our recent work has included detector upgrades, background investigations, acoustical emissions, and sample preparation and procedure investigations. This report will give a brief summary of our work in the above areas.

H. O. Menlove, R. Abedin-Zadeh, and R. Zhu, "The Analyses of Neutron Coincidence Data to Verify Both Spontaneous-Fission and Fissionable Isotopes," Los Alamos National Laboratory report LA-11639-MS (August 1989).

For neutron coincidence counter applications to the assay of plutonium samples, various calibration methods can be used to evaluate the data. The reals (R) and totals (T) rates can be used to predict the induced-fission rate. When this is done, the results give information related to both the spontaneous-fission isotopes and the fissionable isotopes (induced fission). The combination of both approaches is less sensitive to the plutonium isotopic uncertainties than the normal spontaneous-fission approach.

Five calibration methods are evaluated for a wide variety of sample types to emphasize the advantages and disadvantages of the methods.

For certain sample categories with well-defined geometries, such as light-water reactor and fast breeder reactor mixed-oxide fuel assemblies, a method is presented that allows the verification of the declared isotopics within given constraints.

Recommendations are given on analysis procedures to reduce the assay errors for coincidence counting of any type of sample.

H. O. Menlove, R. Palmer, G. W. Eccleston, and N. Ensslin, "Flat-Squared Counter Design and Operation Manual," Los Alamos National Laboratory report LA-11635-M (July 1989).

A well-type neutron coincidence counter has been designed for in-plant applications. The detector has external neutron shielding and special design features to give a uniform efficiency over the sample cavity. In addition, the detector design is relatively insensitive to sample matrix effects. This manual describes the detector design, performance characteristics, and calibration.

M. C. Miller, H. O. Menlove, R. H. Augustson, T. Ohtani, M. Seya, and S. Takahashi, "Remote-Controlled NDA Systems for Process Areas in a MOX Facility," presented at INMM 30th Annual Meeting, Orlando, Florida, July 9-12, 1989. Los Alamos National Laboratory document LA-UR-89-2178.

Nondestructive assay (NDA) systems have been designed and installed in the process area of an automated mixed-oxide (MOX) fuel fabrication facility. These instruments employ neutron coincidence counting methods to measure the spontaneous-fission rate of plutonium in the powders, pellets, and fuel pins in the process area. The spontaneous fission rate and the plutonium isotopic ratios determine the mass of plutonium in the sample. Measurements can be either attended or unattended. The fuel-pin assay system (FPAS) resides above the robotic conveyor system and measures the plutonium content in fuel-pin trays containing up to 24 pins (~1 kg of plutonium). The material accountancy glove-box (MAGB) counters consist of two slab detectors mounted on the sides of the glove box to measure samples of powder or pellets as they are brought to the load cell. Samples measured by the MAGB counters may contain up to 18 kg of MOX. This paper describes the design and performance of four systems: the fuel-pin assay system and three separate MAGB systems. The paper also discusses the role of Monte Carlo transport techniques in the detector design and subsequent instrument calibration.

J. L. Parker and N. Ensslin, "Nondestructive Assay Uncertainties - Present Status and Future Possibilities," presented at the American Nuclear Society, 1989 Annual Meeting, San Francisco, California, November 26-December 1, 1989. Los Alamos National Laboratory document LA-UR-89-2196.

No abstract

J. L. Parker, "Near-Optimum Procedures for Half-Life Measurement by High-Resolution Gamma-Ray Spectrometry," presented at International Committee for Radionuclide Metrology (ICRM), Braunschweig, Federal Republic of Germany, June 6-8, 1989. (Full paper, Los Alamos National Laboratory document LA-UR-89-1438).

A near-optimum procedure for using high-resolution γ-ray spectrometry to measure the half lives of appropriate γ-ray-emitting-nuclides is presented. It is appropriate for measuring half lives in the range from a few hours to perhaps a year. Among the important points of the procedure are the employment of the reference source method for implicit correction of pileup and deadtime losses; the use of full-energy peak-area ratios as the fundamental measured quantities; and continuous, high-rate data acquisition to obtain good results in a fraction of a half-life if desired. Equations are given for estimating the precision of the computed half-lives in terms of total measurement time, number of spectral acquisitions, and the precision of peak-area ratios. Results of ^{169}Yb half-life measurements are given as an example of the procedure's application.

R. R. Picard and K. K. S. Pillay, "International Safeguards for a MOX Facility--Verification to Detect Protracted Falsification," Los Alamos National Laboratory report LA-11609-MS (July 1989).

The theoretical underpinnings of sequential material unaccounted for minus the difference statistic [(MUF-D)] analysis are developed. Methodologically, procedures applicable to sequential MUF data can, in many cases, be adapted to the (MUF-D) problem. Detection of protracted falsification is illustrated in a system study of a modern, state-of-the-art mixed oxide fuel fabrication facility.

K. K. S. Pillay, Compiler, "Fundamentals of Materials Accounting for Nuclear Safeguards," Los Alamos National Laboratory report LA-11569-MS (April 1989).

Materials accounting is essential to providing the necessary assurance for verifying the effectiveness of a safeguards system. The use of measurements, analyses, records, and reports to maintain knowledge of the quantities of nuclear material inventories and materials balances to verify the presence of special nuclear materials are collectively known as materials accounting for safeguards. This manual, prepared as part of the resource materials for the Safeguards Technology Training Program of the U.S. Department of Energy, addresses fundamental aspects of materials accounting, enriching and complementing them with the first-hand experiences of authors from varied disciplines. The topics range from highly technical subjects to site-specific system designs and policy discussions. This collection of papers is prepared by more than 25 professionals from the nuclear safeguards field. Representing research institutions, industries, and regulatory agencies, the authors create a unique resource for the annual course titled "Materials Accounting for Nuclear Safeguards," which is offered at the Los Alamos National Laboratory.

K. K. S. Pillay, "Fuel Fabrication Plant Nuclear Material Accountancy System Elements," presented at the International Training Course on Implementation of State Systems of Accounting for and Control of Nuclear Materials, Santa Fe, New Mexico, May 1-19, 1989, Los Alamos National Laboratory document LA-UR-89-1185.

No abstract

K. K. S. Pillay, J. F. Hafer, and R. R. Picard, "Materials Accounting and International Safeguards for MOX Facilities," presented at the 30th Annual Meeting of the Institute of Nuclear Materials Management, Orlando, Florida, July 9-12, 1989; *Nucl. Mater. Manage.* **XVIII** (Proceedings Issue), 758-763 (1989).

Our experience with mixed oxide (MOX) fuel fabrication facilities leads us to conclude that there is inadequate guidance available to plant and process designers to make materials accounting systems timely, efficient, and minimally intrusive. A well-designed state system for accounting and control of nuclear materials would be beneficial to plant operations and verification by the International Atomic Energy Agency (IAEA) or state regulatory agencies. Among the difficult accounting problems that arise in a large-scale MOX facility are the following: (1) process steps (such as the blending and splitting of powders) that require the accounting system to track material flow, calculate quantities based on previous measurements, and propagate uncertainties as part of data analysis; (2) extensive buffer storage areas involving long residence times that necessitate frequent corrections for material loss from radioactive decay; and (3) facility accounting at one level (for example, fuel pins) that must be reconciled with verification measurements at another level (for example, pin trays or assemblies). Approaches to addressing these problems include designing a special facility, simulating material flow, developing software for near-real-time materials accounting, and establishing achievable verification goals. This paper elaborates on these problems and proposes approaches to a materials accounting system design that considers facility, state, and IAEA safeguards and verification objectives.

K. K. S. Pillay, "A Preliminary Report on Safeguards Aspects of Geologic Disposal of Spent Nuclear Fuels," Los Alamos National Laboratory, Safeguards Systems Group report N-4/89-425 (July 1989).

No abstract.

K. K. S. Pillay, R. R. Picard, J. F. Hafer, and J. T. Markin, "Studies of Near-Real-Time Accounting for Nuclear Materials at a Mixed Oxide Fuel Fabrication Facility: Phase-I," Los Alamos National Laboratory, Safeguards Systems Group report N-4/89-421 (PNC-SA0850-89-004) (July 1989).

No abstract.

P. M. Rinard, T. W. Crane, T. Van Lyssel, K. E. Kroncke, C. M. Schneider, and S. C. Bourret, "A Delayed-Neutron Monitor For A Liquid-Waste Stream With High Gamma-Ray Intensity," Invited paper, in *Proc. International Topical Meeting on Safety Margins in Criticality Safety*, San Francisco, California, November-26-30, 1989, (American Nuclear Society, Inc., La Grange Park, Illinois, 60525), pp. 158-160.

An instrument has been built to monitor the uranium concentration in a liquid-waste stream to avoid a criticality accident in a downstream holding tank. The measurement technique is based on the production and counting of delayed neutrons using the "shuffler" process because the waste contains enough fission products to produce a gamma-ray dose rate of 10 R/h on the surface of the assay tank.

P. M. Rinard, T. Van Lyssel, K. E. Kroncke, C. M. Schneider, and S. C. Bourret, "Monitoring a Liquid Waste Stream with a Delayed-Neutron Instrument," presented at Topical Meeting on Nondestructive Assay of Radioactive Waste, CEA-CENB Cadarache, France, November 20-22, 1989, Los Alamos National Laboratory document LA-UR-89-3171.

A flowing raffinate stream is to be continuously assayed by a delayed-neutron instrument to detect concentrations of ^{235}U that could cause a criticality problem in a holding tank. The instrument is to assay a concentration of 0.034 (g ^{235}U)/L in 100 s with a precision of 10% (1 σ) and to operate unattended for a few months at a time, so it can detect and adjust for changes in the neutron background, the flow rate, and for electronic drifts and malfunctions. In laboratory tests with conditions slightly different from what may be found in the plant, repeated assays on a solution with 0.034 (g ^{235}U)/L flowing at 80 L/h through the 2-L assay tank had relative precisions between 9 and 11%.

T. E. Sampson, G. W. Nelson, and T. A. Kelley, "FRAM: A Versatile Code for Analyzing the Isotopic Composition of Plutonium from Gamma-Ray Pulse Height Spectra," Los Alamos National Laboratory report LA-11720-MS (December 1989).

We describe the characteristics and features and demonstrate the performance of a new code for determining the isotopic composition of plutonium using gamma-ray spectroscopy. This versatile code can measure a wide range of isotopic compositions and is extremely easy to tailor to specialized measurement conditions. Measurement precision, accuracy, and throughput are significantly improved over previous Los Alamos codes.

S. T. Smith, "LAVA and Classical Risk Analysis," presented at the Second International Computer Security Risk Management Model Builders Workshop, Ottawa, Canada, June 20-22, 1989, Los Alamos National Laboratory document LA-UR-89-1558.

LAVA (the Los Alamos Vulnerability/Risk Assessment system) is a three-part systematic approach to risk assessment that can be used to model risk assessment for a variety of application systems such as computer security systems, communications security systems, information security systems, and others. The first part of LAVA is the mathematical methodology based on hierarchical systems theory, fuzzy systems theory, decision analysis, utility theory, and cognitive science; clear relationships exist between LAVA's approach and classical risk analysis. The second part, written for a large class of personal computers, is the general software engine that implements the mathematical risk model. The third part is the application data sets, each written for a specific application system; all application-specific information is data. Application models are knowledge-based expert systems to assess risks in application systems comprising sets of threats, assets, undesirable outcomes, and safeguards. The safeguards system model is in three segments: sets of safeguards functions for protecting the assets from the threats by preventing or ameliorating the undesirable outcomes, sets of safeguards subfunctions whose performance determines whether the function is adequate and complete, and sets of issues, appearing as interactive questionnaires, whose measures (in both monetary and linguistic terms) define both the weaknesses in the safeguards system and the potential costs of undesirable outcome occurrence. The user need have no knowledge of formal risk assessment techniques--all the technical expertise and specialized knowledge are built into the software engine and the application system itself. LAVA applications include our popular computer/information security application and applications for embedded systems, survivability systems, transborder data flow systems, property control systems, and others. LAVA application systems have been in use by federal government agencies since 1984.

S. T. Smith, "Modeling Risk Assessment Applications with LAVA," in *Proceedings of Fifth Annual Symposium and Technical Displays on Physical and Electronic Security* (Armed Forces Communication and Electronic Association, Philadelphia Chapter, 1989), pp. A4-12–A4-15.

LAVA (the Los Alamos Vulnerability/Risk Assessment system) is a three-part systematic approach to vulnerability and risk assessment that can be used to model a variety of application systems such as physical or operational security systems, communications security systems, information security systems, and others. Using LAVA, we build knowledge-based expert systems to assess risks in application systems comprising a subject system and a safeguards system. The methodology provides a framework for creating a variety of applications systems upon which the general software engine operates. All application-specific information is supplied as data and requires no code changes in the general software engine. This paper discusses the ingredients for creating a LAVA application system.

S. T. Smith, "Risk Assessment and LAVA's Dynamic Threat Analysis," in *Proceedings 12th National Computer Security Conference* (National Computer Security Center, Ft. Meade, Maryland, 1989), pp. 483-494.

LAVA (the Los Alamos Vulnerability/Risk Assessment system) is a three-part systematic approach to risk assessment that can be used to model risk assessment for a variety of application systems such as computer security systems, communications security systems, and information security systems. The first part of LAVA is the mathematical methodology based on such disciplines as hierarchical system theory, event-tree analysis, possibility theory, and cognitive science. The second part is the general software engine, written for a large class of personal computers, that implements the mathematical risk model. The third part is the application data sets written for a specific application system. The methodology provides a framework for creating applications for the software engine to operate upon; all application-specific information is data. Using LAVA, we build knowledge-based expert systems to assess risks in application systems comprising a subject system and a safeguards system. The subject system model comprises sets of threats, assets, and undesirable outcomes; because the threat to security systems is ever-changing, LAVA provides for an analysis of the dynamic aspects of the threat spectrum. The safeguards system model comprises sets of safeguards functions for protecting the assets from the threats by preventing or ameliorating the undesirable outcomes; sets of safeguards subfunctions whose performance determine whether the function is adequate and complete; and sets of issues that appear as interactive questionnaires, whose measures (in both monetary and linguistic terms) define both the weaknesses in the safeguards system and the potential costs of an undesirable outcome occurring. The user need have no knowledge of formal risk assessment techniques--all the technical expertise and specialized knowledge are built into the software engine and the application system itself. LAVA applications include our popular computer security application and other applications for embedded systems, survivability systems, transborder data-flow systems, and property control systems. LAVA application systems have been used by federal government agencies since 1984.

J. K. Sprinkle, Jr., and L. A. Stovall, "HEU Drum Monitor Manual (for Confirmatory Measurements)," Los Alamos National Laboratory report LA-11517-M (June 1989).

This manual describes the operation of the highly enriched uranium (HEU) drum monitor. The drum monitor measures the passive gamma-ray emissions from a sealed shipping container of HEU. These emissions are from ^{235}U and from daughters of ^{238}U and ^{232}U. These radiations span a wide range of energy; consequently, each is susceptible to attenuation and shielding to a different degree. The combination of these measured gamma-ray rates with a weight measurement provides a unique signature for each item. These unique signatures can be determined with similar instruments at both ends of a material transfer between Department of Energy facilities. A consistent result from the two instruments indicates material control has been achieved, specifically that no material was lost or diverted. An additional objective of this instrument is to separate the material control issue from the measurement control issue. This is achieved by not calibrating the instrument and by reporting count rates instead of masses. Consequently, the results do not include calibration uncertainties, and therefore they are more precise. In addition, there are no sampling errors. A signal unique to the special nuclear material (SNM) is obtained nondestructively from the entire item. This instrument complements traditional containment and surveillance techniques by providing a precise measurement of an attribute unique to the SNM in the sample.

J. K. Sprinkle, Jr., H. O. Menlove, N. Ensslin, and T. W. Crane, "Measurements of Uranium Waste Using A Californium Shuffler," presented at Topical Meeting on Nondestructive Assay of Radioactive Waste, Cadarache, France, November 20-22, 1989, Los Alamos National Laboratory document LA-UR-89-3740.

We describe a passive/active neutron counter (PAN) based on a ^{252}Cf shuffler for 208-L drums. It is a flexible instrument that can be used to measure the nuclear material content of large containers. This instrument is installed in the Portsmouth Gaseous Diffusion Plant in Piketon, Ohio. This paper describes the results of a calibration for an iron matrix. For 0 to 100 g ^{235}U, the PAN meets our accuracy goal of 10% and our precision goal of 1% for 100 g ^{235}U. With its passive and active capability, this shuffler addresses future needs for materials control and accountability, and health, safety, and environment. The hardware portion of the counter is a good candidate for transfer to the commercial sector. We plan to focus our future waste assay efforts on developing more sophisticated analysis techniques for this generic hardware, rather than developing customized hardware for each application.

J. K. Sprinkle, Jr., G. E. Bosler, S. -T. Hsue, M. P. Kellogg, M. C. Miller, S. M. Simmonds, and A. R. Smith, "Nondestructive Assay of Plutonium-Bearing Scrap and Waste," in *Proc. International Topical Meeting on Safety Margins in Criticality Safety*, San Francisco, California, November-26-30, 1989, (American Nuclear Society, Inc., La Grange Park, Illinois, 60525), pp. 153-157.

Assay of plutonium-bearing scrap and waste (S&W) for plutonium content is very difficult because of the heterogeneous nature of the assay items. Segregation of S&W into appropriate categories has made its assay somewhat easier. However, previous development efforts have been hampered by the lack of representative standards for the calibration and evaluation of measurement performance. We have characterized 25 S&W items to 2% or better, consisting of three distinct S&W categories. We used these items along with fabricated calibration standards to evaluate two state-of-the-art nondestructive assay instruments: a segmented gamma-ray scanner and a neutron coincidence counter. We show that some difficult-to-measure S&W samples can be assayed with less than 5% bias, but note that each category of S&W requires individual evaluation for measurement bias.

C. A. Steverson, "Detecting Change with Digital Imaging: An Application in Nuclear Safeguards," Los Alamos National Laboratory report LA-11632-MS (August 1989).

Recent advances in computer and imaging technology have provided a cost effective means for the application of image processing methods in a variety of disciplines. For security and safeguards applications, image subtraction and other methods of change detection have shown promise as a timely and efficient solution to some security problems. This report describes research done by the Safeguards Systems Group at Los Alamos National Laboratory involving the use of image subtraction and image processing techniques for security applications.

C. A. Steverson, "Image Processing Software Development for Materials Control and Verification," Los Alamos National Laboratory, Safeguards Systems Group, report N-4/89-530 (September 1989).

No abstract.

C. A. Steverson, "Investigating the Application of Infrared Sensors to Monitoring Plutonium Holdup in Processing Areas," Los Alamos National Laboratory, Safeguards Systems Group, report N-4/89-531 (September 1989).

No abstract.

J. E. Stewart, R. R. Ferran, S. M. Simmonds, and H. O. Menlove, "Calibration Parameters from Monte Carlo Simulations for Neutron Coincidence Assay of MOX Fuel Elements: A Substitute for Standards?" in *Proceedings of the 11th Symposium on Safeguards and Nuclear Material Management*, Luxembourg, May 30-June 1, 1989, pp. 135-141. Los Alamos National Laboratory document LA-UR-89-1772.

Results from application of a calculational model for the two-parameter (singles and doubles) passive neutron coincidence assay of finished Fast Breeder Reactor (FBR) subassemblies are compared with calibration measurements. Two assay instruments are considered; the Universal Fast Breeder Reactor Subassembly Counter (UFBC) and the Capsule Counter installed at the Japanese Plutonium Fuel Production Facility (PFPF). In the case of US Fast Flux Test Facility (FFTF) fuel, the absolute ratio of calculations to measurements for the multiplication-corrected coincidence calibration constant is +1.1 ± 1.0% (average of four subassemblies) for the UFBC and -1.3 ± 0.6% (average of five subassemblies) for the Capsule Counter. For initial measurements of Japanese fuel in the Capsule Counter, the absolute ratio is -1.0 ± 0.7% for three JOYO subassemblies and +0.8 ± 0.7% for one MONJU subassembly. Calculations of relative effects such as the change in coincidence response from, for example, subassembly can thickness or U enrichment are more accurate (better than 0.5%) than absolute calibration parameters. This very good accuracy offers more effective and less costly inspector verification of finished FBR fuel elements by reducing reliance on physical standards to expand the cross-calibration databases.

K. E. Thomas and J. W. Barnes, "A Preliminary Evaluation of the Uranium Solidification Facility Materials Accounting System," Los Alamos National Laboratory, Safeguards Systems Group, report N-4/89-473 (August 1989).

No abstract.

H. S. Vaccaro, "Detection of Anomalous Computer Session Activity," presented at the 1989 IEEE Symposium on Research in Security and Privacy, Oakland, California, May 1-3, 1989; *1989 IEEE Computer Society Symposium on Security and Privacy* (IEEE Computer Society Press, Washington, DC, 1989), pp. 280-289.

This paper briefly discusses Wisdom and Sense (W&S), a computer security anomaly detection system developed at Los Alamos National Laboratory. Anomaly detection provides another layer of defense against computer misuse after physical security and access security. W&S is statistically based. It automatically generates rules from historical data and, in terms of those rules, identifies computer transactions that are at variance with historically established usage patterns. Issues addressed in this paper include how W&S generates rules from a necessarily small sample of all possible transactions, how W&S deals with inherently categorical data, and how W&S assists system security officers in their review of audit logs.

REFERENCES

1. P. E. Fehlau, "An Applications Guide to Pedestrian SNM Monitoring," Los Alamos National Laboratory report LA-10633-MS (February 1986).

2. P. E. Fehlau, "An Applications Guide to Vehicle SNM Monitors," Los Alamos National Laboratory report LA-10912-MS (March 1987).

3. P. E. Fehlau, "A Low-Cost Safeguards Pedestrian Portal Monitor Using Chamber Neutron Detectors," in *Proceedings of the 9th ESARDA Annual Symposium on Safeguards and Nuclear Materials Management* (London, England, 1987), ESARDA **21**, pp. 77-80.

4. P. E. Fehlau, "Rugged, Lightweight, and Long-Operating Hand-Held Instruments for Neutron and Gamma-Ray Verification Measurements," in *Proceedings of the 22nd Midyear Topical Meeting of the Health Physics Society on Instrumentation*, December 4-8, 1988, San Antonio, Texas, pp. 87-95.

5. P. E. Fehlau, "SNM-Detection Technology Update: A Review of Recently Developed Equipment," a talk and handout for the Sandia National Laboratories Workshop on Entry-Control Technology, Albuquerque, New Mexico, April 18-20, 1989. Los Alamos National Laboratory document LA-UR-89-1290.

6. Paul E. Fehlau, "Perimeter Radiation Monitors for the Control and Physical Security of Special Nuclear Materials," Abstract of an invited paper for the ASTM Symposium on Access Security Screening: Challenges and Solutions for the 1990's, New Orleans, Louisiana, March 29-30, 1990. Los Alamos National Laboratory document LA-UR-89-2044.

7. P. E. Fehlau, "Calibrating the Jomar JPM-22 Pedestrian SNM Monitor," Los Alamos National Laboratory report LA-11643-M (August 1989).

8. P. E. Fehlau, "Calibrating the TSA Systems VM-250 SNM Portal Monitor," Los Alamos National Laboratory report LA-11871-M (July 1990).

9. P. E. Fehlau, "Evaluation of the Jomar Systems, Inc., JPM-22 Portal Monitor," Los Alamos National Laboratory Memorandum N2-89:324PEF to N. Nicholson (May 26, 1989).

10. P. E. Fehlau, "Evaluation of a National Nuclear Corp. DM60 Portal Monitor," Los Alamos National Laboratory Memorandum N2-89:347PEF to N. Nicholson (June 6, 1989).

11. P. E. Fehlau, "Evaluation of the TSA Systems, LTD., SPM-904 Portal Monitor," Los Alamos National Laboratory Memorandum N2-89:576PEF to N. Nicholson (September 29, 1989).

12. P. Dumesnil and J. L. Greco, "Optimization of the Processing of the Counts from a Doorway Monitor for Radioactive Substances," *Proceedings of the 7th ESARDA Annual Symposium on Safeguards and Nuclear Materials Management* (Liege, Belgium, 1985), L. Stanchi, Ed., Joint Research Centre, Ispra, Italy, ESARDA **19**.

13. G. S. Brunson and G. J. Arnone, "A New System for Analyzing Neutron Multiplicities: Characterization and Some Specific Applications," Los Alamos National Laboratory report LA-11701-MS (November 1989).

14. G. J. Arnone and G. S. Brunson, "A Programmable Multichannel Correlation Module for Analyzing Neutron Multiplicities," IEEE Nuclear Science Symposium, Oct. 23-27, 1989, San Francisco, California.

15. K. L. Coop and P. E. Fehlau, "Monitoring Packages and Containers," in "Safeguards and Security Progress Report, January—December 1988," D. B. Smith and G. R. Jaramillo, Comps., Los Alamos National Laboratory report LA-11709-PR (December 1989), p. 8.

16. C. A. Steverson, "Investigating the Application of Infrared Sensors to Monitoring Plutonium Holdup in Processing Areas," Los Alamos National Laboratory, Safeguards Systems Group report N-4/89-531.

17. H. O. Menlove, O. R. Holbrooks, and A. Ramalho, "Inventory Sample Coincidence Counter Manual," Los Alamos National Laboratory report LA-9544-M (ISPO-181), (November 1982).

18. J. F. Briesmeister, Ed., "MCNP-A General Monte Carlo Code for Neutron and Photon Transport, Ver. 3A," Los Alamos National Laboratory report LA-7396-M, Rev. 2 (September 1986).

19. J. Caldwell, K. E. Kunz, J. D. Atencio, G. C. Herrera, and J. C. Pratt, "Experimental Evaluation of the Differential Die-Away Pulsed Neutron Technique for the Fissile Assay of Hot Irradiated Fuel Waste," in *Proceedings of the American Nuclear Society Topical Meeting Treatment Handling Radioactive Wastes, Richland, Washington* (Battelle-Columbus Press, Columbus, Ohio, 1982), pp. 302-305.

20. M. S. Krick, G. E. Bosler, J. E. Swansen, and K. E. Kroncke, "Neutron Multiplicity Counter," in "Safeguards and Security Progress Report, January— December 1986," D. B. Smith, Comp., Los Alamos National Laboratory report LA-11120-PR (March 1988), p. 38.

21. T. E. Sampson, G. W. Nelson, and T. A. Kelley, "FRAM: A Versatile Code for Analyzing the Isotopic Composition of Plutonium from Gamma-Ray Pulse Height Spectra," Los Alamos National Laboratory report LA-11720-MS (December 1989).

22. S.-T. Hsue and R. Zhu, "Poor Man's Densitometry," *Nucl. Mater. Manage.* **XVIII** (Proc. Issue), 806-813 (1989).

23. J. G. Fleissner, "Nondestructive Assay of Plutonium Isotopically Heterogeneous Salt Residues," in *Proceedings of the Conference on Safeguards Technology: The Process-Safeguards Interface*, November 28 - December 2, 1983, Hilton Head Island, South Carolina (USDOE/New Brunswick Laboratory, August 1984), CONF-831106, pp. 275-285.

24. J. G. Fleissner and Merril W. Hume, "Comparison of Destructive and Nondestructive Assay of Heterogeneous Salt Residues," Rockwell International Rocky Flats Plant report RFP-3876 (March 29, 1986).

25. N. Ensslin, "A Simple Self-Multiplication Correction for In-Plant Use," in *Proceedings of the 7th ESARDA Annual Symposium on Safeguards and Nuclear Materials Management* (Liege, Belgium, 1985), ESARDA **19**, pp. 223-238.

26. J. E. Stewart, "A Hybrid Monte Carlo/Analytical Model of Neutron Coincidence Counting," *Trans. Am. Nucl. Soc.* **53**, 149-151 (1986).

27. S.-T. Hsue, D. G. Langner, V. L. Longmire, H. O. Menlove, P. A. Russo, and J. K. Sprinkle Jr., "Measurements of Plutonium Residues from Recovery Processes," presented at the Topical Meeting on Nondestructive Assay of Radioactive Waste, Cadarache, France, November 20-22, 1989, Los Alamos National Laboratory document LA-UR-89-3699.

28. S.-T. Hsue, J. L. Parker, and J. K. Sprinkle Jr., "The Advanced Segmented Gamma Scanning (SGS) Technique, in "Safeguards and Security Progress Report, January—December 1987," D. B. Smith and G. R. Jaramillo, Comps., Los Alamos National Laboratory report LA-11356-PR (September 1988), pp. 12-13.

29. J. K. Sprinkle, Jr. and S.-T. Hsue, "Recent Advances in SGS Analysis," *Proceedings of the Third International Conference on Facility Operations—Safeguards Interface* (American Nuclear Society, Inc., La Grange Park, Illinois 1988), ANS Order No. 700132, pp. 188-193.

30. E. R. Martin, D. F. Jones, and J. L. Parker, "Gamma-Ray Measurements with the SGS," Los Alamos Scientific Laboratory report LA-7059-M (December 1977).

31. J. L. Parker, "A Plutonium Solution Assay System Based on High Resolution Gamma-Ray Spectroscopy," Los Alamos Scientific Laboratory report LA-8146-MS (January 1980).

32. T. K. Li, T. Marks, and J. L. Parker, "Solution Assay Instrument Operations Manual," Los Alamos National Laboratory report LA-9820-M (September 1983).

33. R. Gunnink, "MGA2: A One-Detector Code for Rapid High-Precision Plutonium Isotopic Measurement," *Nucl. Mater. Manage.* **XVI** (Proc. Issue) 352-358 (1987).

34. J. E. Stewart and H. O. Menlove, "Moisture Corrections in Neutron Coincidence Counting of PuO_2," *Trans. Am. Nucl. Soc.* **55**, Supp. 1, 43-44 (1987).

35. J. K. Sprinkle, Jr., "Assay Methods for Other Nuclear Materials—^{237}Np," in "Safeguards and Security Progress Report, January—December 1988," D. B. Smith and G. R. Jaramillo, Comps., Los Alamos National Laboratory report LA-11709-PR (December 1989), p. 19.

36. Y. Kuno, S. Takeda, S. Sato, T. Akiyama, Y. Tsutaki, T. Suzuki, E. Kuhn, S. Deron, and K.

Sirisena, "Reprocessing Plant Input Accountability Measurements—A New and Simplified Spiking Technique," *ANS Transactions* **60**, 229 (1989).

37. T. K. Li, "Feasibility Study of Plutonium Isotopic Analysis of Resin Beads by Nondestructive Gamma-Ray Spectroscopy," in *Proceedings of the 7th ESARDA Symposium on Safeguards and Nuclear Materials Management* (Liege, Belgium, 1985), ESARDA **19**, pp. 245-248.

38. T. K. Li, Y. Kuno, K. Nakatsuka, and T. Akiyama, "Determination of Plutonium Concentration and Isotopic Composition by Isotope Dilution Gamma-Ray Spectroscopy on Resin Beads," Los Alamos National Laboratory report LA-11827-MS (June 1990).

39. J. K. Sprinkle, Jr., A. Goldman, P. A. Russo, L. Stovall (LANL), T. L. Brumfield, C.S. Gunn, D. R. Watson (ORNL), and R. Beedgen (KfK-FRG), "A Confirmatory Measurement Technique for HEU," *Proceedings of the Third International Conference on Facility Operations—Safeguards Interface* (American Nuclear Society, Inc., La Grange Park, Illinois, 1988), ANS Order No. 700132, pp. 116-122.

40. S. C. Bourret, D. L Garcia, J. E. Swansen, H. R. Dye, K. E. Kroncke, and P. A. Russo, "Compact NaI(Tl) Detector for Portable Holdup Measurements," in "Safeguards and Security Progress Report, January—December 1988," D. B. Smith and G. R. Jaramillo, Comps., Los Alamos National Laboratory report LA-11709-PR (December 1989), pp. 29-30.

41. P. A. Russo, G. W. Eccleston, J. E. Swansen, M. M. Martinez, and H. D. Ramsey, "Monitoring the Hydrofluorination Process by Neutron Counting," in "Safeguards and Security Progress Report, January—December 1986," D. B. Smith, Comp., Los Alamos National Laboratory report LA-11120-PR (March 1988), pp. 49-52.

42. P. A. Russo, J. E. Swansen, R. Siebelist, and J. K. Sprinkle, Jr., "Plutonium Holdup in 371 Tilt-Pour Furnaces," in "Safeguards and Security Progress Report, January—December 1986," D. B. Smith, Comp., Los Alamos National Laboratory report LA-11120-PR (March 1988), pp. 49-52.

43. P. A. Russo and H. O. Menlove, "Identification of an In-Line Test Application of the Swansen Compact, Self-Contained ^{3}He Neutron Discriminator," in "Safeguards and Security Progress Report, January—December 1988," D. B. Smith and G. R. Jaramillo, Comps., Los Alamos National Laboratory report LA-11709-PR (December 1989), pp. 6? 66.

44. J. Swansen, "Timer-Scaler Module Evaluation," in "Safeguards and Security Progress Report, January—December 1985," D. B. Smith, Comp., Los Alamos National Laboratory report LA-10787-PR (March 1987), p. 55.

45. P. A. Russo, R. Siebelist, J. A. Painter and J. E. Gilmer, "Evaluation of High-Resolution Gamma-Ray Methods for Determination of Solid Plutonium Holdup in High-Throughput Bulk Processing Equipment," Los Alamos National Laboratory report LA-11729-MS (January 1990).

46. D. C. Garcia, "Portable Gamma-Ray Holdup Instrument," Los Alamos Group N-1 mechanical drawing package SK-N1-296, D1–D12 (August 1989).

47. J. A. Painter and P. A. Russo, "Prototype Software for Automating Portable Gamma-Ray Holdup Measurements," in "Safeguards and Security Progress Report January—December 1988," D. B. Smith and G. R. Jaramillo, Comps., Los Alamos National Laboratory report LA-11709-PR (December 1989), pp. 59-63.

48. N. Ensslin, N. Dytlewski, and M. S. Krick, "Assay Variance as a Figure of Merit for Neutron Multiplicity Counters," to be published in *Nuclear Instruments and Methods in Physics Research* (1990).

49. M. S. Krick, G. E. Bosler, and J. E. Swansen, "A Neutron Multiplicity Counter for Neutron Multiplication Measurements," INMM Technical Workshop on Bias in Nondestructive Assay for Nuclear Materials Accountability, Boulder, Colorado (April 1987).

50. R. C. Bearse, J. W. Barnes, and R. M. Tisinger, "Integrated Safeguards for Argonne National Laboratory-West (ANL-W)," in "Safeguards and Security Progress Report, January—December, 1988," D. B. Smith and G. R. Jaramillo, Comps., Los Alamos National Laboratory report LA-11709-PR (December 1989), p. 46.

51. G. L. Barlich and P. Helman, "Distributed MC&A Database Design," in "Safeguards and Security Progress Report, January—December, 1988," D. B. Smith and G. R. Jaramillo, Comps., Los Alamos National Laboratory report LA-11709-PR (December 1989), p. 38.

52. J. T. Markin, J. E. Ballmann, and A. Zardecke, "RAOPS User's Manual," Los Alamos National Laboratory, Safeguards Systems Group, draft report N-4/90-422 (1990).

53. W. J. Hunteman, "Lessons Learned in the DOE Computer Security Enhancement Review Program," Los Alamos National Laboratory report LA-11614-MS (July 1989).

54. D. P. Martinez, "VMS ALAP 1.0: An Automated Audit Trail Analysis Tool," presented at the 12th DOE Computer Security Group Conference, Amarillo, Texas, May 2-4, 1989, Los Alamos National Laboratory document LA-UR-89-1511.

55. C. A. Coulter, R. S. Leonard, R. B. Strittmatter, and K. E. Thomas, "SIS Safeguards System Design," in "Safeguards and Security Progress Report, January—December 1988," D. B. Smith and G. R. Jaramillo, Comps., Los Alamos National Laboratory report LA-11709-PR (December 1989), p 44.

56. A. J. Grambihler, P. B. O'Callaghan, and R. L. Carlson, "Restricted Data Transmission Controller," Westinghouse Hanford Company document WHC-SP-0305.

57. P. B. O'Callaghan, A. J. Grambihler, and R. L. Carlson, "Restricted Data Transmission Controller Progress Report for 1989," Westinghouse Hanford Company document WHC-SA-0599-FP.

58. H. O. Menlove, R. Palmer, G. W. Eccleston, and N. Ensslin, "Flat-Squared Counter Design and Operation Manual," Los Alamos National Laboratory report LA-11635-M (July 1989).

59. H. O. Menlove and J. E. Swansen, "A High-Performance Neutron Time-Correlation Counter," *Nucl. Technol.* 71, 497-505 (November 1985).

60. J. K. Sprinkle, Jr., G. E. Bosler, S. -T. Hsue, M. P. Kellogg, M. C. Miller, S. M. Simmonds, "Nondestructive Assay of Plutonium-Bearing Scrap and Waste," presented at the Nuclear Criticality Safety Margins, American Nuclear Society Topical Meeting November 26-30, 1989, San Francisco, California, Los Alamos National Laboratory document LA-UR-89-2373,

61. J. K. Sprinkle, Jr., H. O. Menlove, L. R. Cowder, P. Polk, P. Collinsworth, and C. M. Schneider, "Shuffler for Measuring Gaseous Diffusion Wastes," in "Safeguards and Security Progress Report, January—December 1988," D. B. Smith and G. R. Jaramillo, Comps., Los Alamos National Laboratory report LA-11709-PR (December 1989), p. 53.

62. "High-Density Waste Shuffler," in "Safeguards and Security Progress Report, January—December 1985," D. B. Smith, Comp., Los Alamos National Laboratory report LA-10787-PR (March 1987), p. 32.

63. J. K. Sprinkle Jr., H.O. Menlove, N. Ensslin, and T.W. Crane, "Measurements of Uranium Waste using a Californium Shuffler," presented at the Topical Meeting on Nondestructive Assay of Radioactive Waste, Cadarache, France, November 20-22, 1989, Los Alamos National Laboratory document LA-UR-89-3740.

64. P. M. Rinard and T. W. Crane, "Uranium Monitor for Raffinate Lines," in "Safeguards and Security Progress Report, January—December 1988," D. B. Smith and G. R. Jaramillo, Comps., Los Alamos National Laboratory report LA-11709-PR (December 1989), pp. 51-53.

65. P. A. Russo, R. Siebelist, and J. A. Painter, "Evaluation of High-Resolution Gamma-Ray Methods for Determining Solid Plutonium Holdup in High-Throughput Bulk Processing Equipment," in "Safeguards and Security Progress Report, January—December 1988," D. B. Smith and G. R. Jaramillo, Comps., Los Alamos National Laboratory report LA-11709-PR (December 1989), pp. 54-58.

66. G. L. Barlich and T. A. Kelly, "Software Design for the Instrument Control Computer at New Special Recovery," Los Alamos National Laboratory report LA-11449-M (April 1989).

67. T. L. Brumfield, L. A. Johnson, and J. K. Sprinkle Jr., "A Confirmatory Measurement Technique for HEU," *Nucl. Mater. Manage.* **XVII** (Proc. Issue) 610-616 (1989).

68. J. E. Stewart, N. Ensslin, H. O. Menlove, L. R. Cowder, and P. J. Polk, "Confirmatory Measurements of UF_6 Using the Neutron Self-Interrogation Method," *Nucl. Mater. Manage.* **XIV** (Proc. Issue) 606-612 (1985).

69. C. A. Steverson, "Image Processing Software Development for Materials Control and Verification," Los Alamos National Laboratory, Safeguards Systems Group, report N-4/89-530 (September 1989).

70. K.K.S. Pillay, compiler, "Fundamentals of Materials Accounting for Nuclear Safeguards," Los Alamos National Laboratory report LA-11569-M (1989).

71. P. A. Russo, R. Siebelist, and J. A. Painter, "Current Technologies Applied to In-Plant Portable HPGe Measurements of Holdup," in "Safeguards and Security Progress Report January—December 1988," D. B. Smith and G. R. Jaramillo, Comps., Los Alamos National Laboratory report LA-11709-PR (December 1989), pp. 63-64.

72. "Inspection Activities Associated with Limited-Frequency Unannounced Access Model Applied to Gas Centrifuge Type Enrichment Plant," Hexapartite Safeguards Project, February 4, 1983.

73. M. C. Miller, T. K. Li, R. B. Strittmatter, and E. L. Adams, "Radiation Monitoring of AVLIS Separator Components for International Safeguards Inspection," Los Alamos National Laboratory report (to be published).

74. T. D. Reilly, R. B. Walton, and J. L. Parker, "The Enrichment Meter/A Simple Method for Measuring Isotopic Enrichment," in "Nuclear Safeguards Research and Development Program Status Report, September—December 1970," G. R. Keepin, Comp., Los Alamos Scientific Laboratory report LA-4605-MS (1971), p. 19.

75. S. F. Klosterbuer, "Portable Gamma-Ray and Neutron Detector Electronics User Manual," Los Alamos National Laboratory report (to be published).

76. E. A. Hakkila, J. W. Barnes, R. G. Gutmacher, and R. R. Picard, "Near-Real-Time Verification Approaches for the Process Area of Reprocessing Plants," Los Alamos National Laboratory report LA-11615-MS (June 1989).

77. K. K. S. Pillay, "A Preliminary Report on Safeguards Aspects of Geologic Disposal of Spent Nuclear Fuels," Los Alamos National Laboratory, Safeguards Systems Group report N-4/89-425 (July 1989).

78. K. K. S. Pillay, "Safeguards Issues Relevant to Geologic Disposal of Spent Nuclear Fuels," *Trans. Am. Nucl. Soc.*, **60**, 240-41 (November 1989).

79. J. R. Phillips, J. K. Halbig, H. O. Menlove, and S. F. Klosterbuer, "Apparatus for In Situ Determination of Burnup, Cooling Time, and Fissile Content of an Irradiated Nuclear Fuel Assembly in a Fuel Storage Pond," US Patent Number 4 510 117, April 9, 1985.

80. H. O. Menlove, "Description and Performance Characteristics for the Neutron Coincidence Collar for the Verification of Reactor Fuel Assemblies," Los Alamos National Laboratory report LA-8939-MS (ISPO-142) (August 1981).

81. H. O. Menlove and J. E. Pieper, "Neutron Collar Calibration for Assay of LWR Fuel Assemblies," Los Alamos National Laboratory report LA-10827-MS (ISPO-258) (March 1987).

82. H. O. Menlove and J. E. Stewart, "A New Method of Calibration and Normalization for Neutron Detector Families," Los Alamos National Laboratory report LA-11229-MS (ISPO-287) (April 1988).

83. K. L. Coop, J. T. Caldwell, and C. A. Goulding, "Assay of Fissile Material Using a Combined Thermal/ Epithermal Neutron Interrogation Technique" *Proceedings of the Third International Conference of Facility Operations--Safeguards Interface* (American Nuclear Society, Inc., La Grange Park, Illinois, 1988), ANS Order No. 700132, pp. 333-338.

84. P. M. Rinard and G. E. Bosler, "BWR Spent Fuel Measurements with the ION-1/Fork Detector and a Calorimeter," Los Alamos National Laboratory report LA-10758-MS (August 1986).

85. P. M. Rinard and G. E. Bosler, "Safeguarding LWR Spent Fuel with the Fork Detector," Los Alamos National Laboratory report LA-11096-MS (March 1988).

86. G. Bignan and J. Capsie, "Resultats des Essais de Qualification dans l'Installation Cobra du Dispositif de Controle du Taux de Combustion et de la Reactivite Residuelle des Assemblages Irradies en Piscines de Centrales EDF," CEA/DRP Technical Note 89-034 (November 15, 1989).

87. H. O. Menlove, G. W. Eccleston, N. Ensslin, and R. L. Palmer, "Design of the Flat Squared Counter," in "Safeguards and Security Progress Report, January— December 1988," D. B. Smith and G. R. Jaramillo, Comps., Los Alamos National Laboratory report LA-11709-PR (December 1989), p. 19.

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