LA-8620-C, Vol. II

INTERNATIONAL TRAINING COURSE ON NUCLEAR MATERIALS ACCOUNTABILITY FOR SAFEGUARDS PURPOSES

May 27 - June 6, 1980

PROCEEDINGS

Sponsored by U.S. Department of Energy
in cooperation
with International Atomic Energy Agency
presented by
Los Alamos Scientific Laboratory
Los Alamos, New Mexico

An Affirmative Action/Equal Opportunity Employer

UNITED STATES
DEPARTMENT OF ENERGY
CONTRACT W-7405-ENG. 36

LA-8620-C, Vol. II
UC-15
Issued: December 1980

INTERNATIONAL TRAINING COURSE ON NUCLEAR MATERIALS ACCOUNTABILITY FOR SAFEGUARDS PURPOSES

PROCEEDINGS

Lecture Text and Reference Material
Sessions 18—32

G. ROBERT KEEPIN, Course Director
B. PONTES, IAEA Scientific Advisor
CHARLES R. HATCHER, Course Coordinator
T. DOUGLAS REILLY, Course Coordinator
KAREN HUMPHREY, Admin. Assistant

Bishop's Lodge
Santa Fe, New Mexico

May 27 - June 6, 1980

INTERNATIONAL TRAINING COURSE ON NUCLEAR MATERIALS ACCOUNTABILITY FOR SAFEGUARDS PURPOSES

NNPA STEERING COMMITTEE

G. Hammond, DOE/OSS
G. Keepin, LASL
J. Partlow, NRC
L. Thorne, IAEA

NNPA WORKING GROUP

C. Buechler, IAEA
G. Hough, IAEA
D. Perricos, IAEA
B. Pontes, IAEA

J. Dailey, DOE/OSS
W. Hagis, DOE/OSS
J. Reilly, DOE/OSS

L. Wirfs, NRC
R. Jackson, NRC
M. Otin, NRC

J. Menzel, ACDA

R. Liimatainen, DOS

W. Higinbotham, BNL

T. Canada, LASL
A. Hakkila, LASL
C. Hatcher, LASL
N. Nicholson, LASL
D. Reilly, LASL
J. Shipley, LASL
H. Smith, LASL

INTERNATIONAL TRAINING COURSE ON NUCLEAR MATERIALS ACCOUNTABILITY FOR SAFEGUARDS PURPOSES

ABSTRACT

The two volumes of this report incorporate all lectures and presentations at the International Training Course on Nuclear Materials Accountability and Control for Safeguards Purposes, held May 27-June 6, 1980 at the Bishop's Lodge near Santa Fe, New Mexico. The course, authorized by the US Nuclear Non-Proliferation Act and sponsored by the US Department of Energy in cooperation with the International Atomic Energy Agency, was developed to provide practical training in the design, implementation, and operation of a National system of nuclear materials accountability and control that satisfies both National and IAEA International safeguards objectives.

Volume I, covering the first week of the course, presents the background, requirements, and general features of material accounting and control in modern safeguard systems. Volume II, covering the second week of the course, provides more detailed information on measurement methods and instruments, practical experience at power reactor and research reactor facilities, and examples of operating state systems of accountability and control.

INTERNATIONAL TRAINING COURSE ON NUCLEAR MATERIALS ACCOUNTABILITY FOR SAFEGUARDS PURPOSES

TABLE OF CONTENTS

VOLUME I

VOLUME II

INTERNATIONAL TRAINING COURSE ON NUCLEAR MATERIALS ACCOUNTABILITY AND CONTROL FOR SAFEGUARDS PURPOSES

MONDAY, MAY 26, 1980
1600-1800 - REGISTRATION
1700-1900 - SOCIAL HOUR

TIME	TUESDAY 27 MAY	WEDNESDAY 28 MAY	THURSDAY 29 MAY	FRIDAY 30 MAY		MONDAY 2 JUNE	TUESDAY 3 JUNE	WEDNESDAY 4 JUNE	THURSDAY 5 JUNE	FRIDAY 6 JUNE
0700	BREAKFAST	BREAKFAST	BREAKFAST	MEET AT STABLE TRAIL RIDE AND BREAKFAST		BREAKFAST	BREAKFAST	BREAKFAST	BREAKFAST	BREAKFAST
0730	REGISTRATION			(REGULAR BREAKFAST OPTIONAL)						
0800							DEPART BISHOP'S LODGE-TRAVEL TO LOS ALAMOS			
0830		6. DOMESTIC ACCOUNTABILITY AND CONTROL FEATURES (LUMB)	9. INTRODUCTION TO NUCLEAR FUEL CYCLES (KNIEF)			18. ELEMENTS OF CHEMICAL AND BULK MEASUREMENT TECHNOLOGY (BINGHAM)		23. AN LWR POWER REACTOR FACILITY (REED)	30. IMPLEMENTATION OF THE FACILITY SAFEGUARDS SYSTEM (POWERS)	31. WORKSHOP CONTINUED
0900	1. WELCOME (KERR, WEISZ, von BAECKMANN)						22. TOUR OF LASL SAFEGUARDS LABORATORIES			
0930				14. NUCLEAR MATERIALS ACCOUNTING AND CONTROL IN POWER REACTORS (FOLEY, HIGINBOTHAM)				BREAK		
0945	2. INTRODUCTION TO TRAINING COURSE (WEISZ, von BAECKMANN, KEEPIN)							24. A CANDU POWER REACTOR FACILITY (SINDEN)		
1000		BREAK	BREAK			BREAK	BREAK		BREAK	BREAK
1030	BREAK	7. EURATOM SAFEGUARDS (MIRANDA)	10. ELEMENTS OF NUCLEAR MATERIAL ACCOUNTING (LUMB)			19. ELEMENTS OF NONDESTRUCTIVE ASSAY (NDA) TECHNOLOGY (SMITH, CANADA)	22. TOUR CONTINUED		31. WORKSHOP IN FACILITY SAFEGUARDS SYSTEM DESIGN	32. PLENARY SESSION AND WRAP-UP OF SAFEGUARDS SYSTEM DESIGN WORKSHOP
1045				15. SAFEGUARDING OF NUCLEAR RESEARCH FACILITIES (JOHNSON)				BREAK		
1100	3. HISTORICAL AND POLITICAL FRAMEWORK OF SAFEGUARDS (TAPE)							25. A RESEARCH REACTOR FACILITY (TINGEY)		
1130										
1200	LUNCH AND FREE TIME	LUNCH AND FREE TIME	LUNCH AND FREE TIME	LUNCH AND FREE TIME		LUNCH AND FREE TIME		LUNCH AND FREE TIME	LUNCH AND FREE TIME	LUNCH AND FREE TIME
1230							BOX LUNCH AT BANDELIER			
1330								26. SAFEGUARDS SYSTEM DESIGN AND APPLICATION (SHIPLEY, COBB, HAKKILA, OLSON)		
1400		8. IAEA INTERNATIONAL SAFEGUARDS (BUECHLER)	11. NUCLEAR MATERIAL CONTROL (OLSON)	16. INSPECTION OF REACTOR AND SPENT FUEL STORAGE FACILITIES (THORNE)						33. GENERAL DISCUSSION, COMMENTS, AND CONCLUSIONS
1430								BREAK		
1445								27. DESIGN FEATURES RELEVANT TO IMPROVED SAFEGUARDS IMPLEMENTATION (GUPTA)		
1500		BREAK							31. WORKSHOP CONTINUED	
1515		IAEA INTERNATIONAL SAFEGUARDS (THORNE)								
1530			BREAK	BREAK						BREAK
1545								BREAK		
1600	4. DESCRIPTION OF A STATE SYSTEM AND ITS REQUIREMENTS (PARTLOW)		12. SURVEY OF STATISTICAL METHODS IN NUCLEAR MATERIAL ACCOUNTING AND CONTROL (JAECH)	17. PREWORKSHOP SESSION AND REVIEW (SHIPLEY; IAEA AND LASL STAFF)		20. NATIONAL SYSTEM OF MEASUREMENT STANDARDS (YOLKEN)	TOUR OF LASL SCIENCE MUSEUM	28. EXAMPLE OF AN OPERATING STATE SYSTEM – GDR (ROEHNSCH)		DOE WRAP UP OF COURSE
1615		BREAK								
1630		IAEA INTERNATIONAL SAFEGUARDS (HOUGH)								
1700										NO-HOST SOCIAL HOUR
1715							RECEPTION STUDY CENTER			
1730	NO-HOST GET-ACQUAINTED COCKTAIL PARTY	NO-HOST SOCIAL HOUR	NO-HOST SOCIAL HOUR	NO-HOST SOCIAL HOUR		NO-HOST SOCIAL HOUR		NO-HOST SOCIAL HOUR		
1800									NO-HOST SOCIAL HOUR	
1830	DINNER	MEXICAN DINNER	DINNER			DINNER	DINNER AT CASA del MIRADOR	DINNER		
1900				DINNER BAR-B-QUE						STEAK FRY
1930									BANQUET	
2000										
2030	5. DOMESTIC SAFEGUARDS: THREAT ANALYSIS AND RESPONSE CAPABILITIES (JENKINS)	TRAVELOG OF NEW MEXICO	13. ADVANCED SNM ACCOUNTING AND CONTROL SYSTEMS FOR BULK PROCESSING FACILITIES (HIGINBOTHAM, MALANIFY)			21. ASSAY/ VERIFICATION OF FRESH AND SPENT-FUEL ELEMENTS (LEE)		29. EXAMPLE OF AN OPERATING STATE SYSTEM - JAPAN (KURIHARA, OSABE)		

OPTIONAL TOUR OF NORTHERN NM — SATURDAY, 31 MAY 1980
FREE TIME — SUNDAY, 1 JUNE 1980

INTERNATIONAL TRAINING COURSE ON NUCLEAR MATERIALS ACCOUNTABILITY FOR SAFEGUARDS PURPOSES

SESSION #18: ELEMENTS OF CHEMICAL AND BULK MEASUREMENT TECHNOLOGY

SPEAKER: Dr. Carleton D. Bingham

US Department of Energy
New Brunswick Laboratory
Argonne, IL USA

Monday, June 2, 1980
8:30 a.m.

BIOGRAPHY

Education: A.B (with honors and distinction in chemistry) from San Diego State University in 1950 and Ph.D. (physical chemistry) from the University of California, Los Angeles (UCLA) in 1959.

Present Position: Director of the New Brunswick Laboratory, Chicago Operations and Regional Office of the US Department of Energy.

Past Positions: Radiological Engineer for the southern region of the University of California, 1953-1959. Research Chemist, Laboratory Manager, and Project Engineer for the Atomics International Division, Rockwell International, from 1959-1971.

Other Information: Has authored more than 40 technical papers covering the fields of nuclear and radiochemistry, analytical chemistry, sodium chemistry, and nuclear materials safeguards. Is a member of the American Chemical Society, American Nuclear Society, American Society for the Advancement of Science, American Society for Testing and Materials, Health Physics Society, Institute of Nuclear Materials Management, Scientific Research Society of America.

INTERNATIONAL TRAINING COURSE ON NUCLEAR MATERIALS ACCOUNTABILITY FOR SAFEGUARDS PURPOSES

Session Objectives

SESSION #18: ELEMENTS OF CHEMICAL AND BULK MEASUREMENT TECHNOLOGY

Conventional methods of determining concentration and isotopic composition of special nuclear materials encountered in the nuclear fuel cycle are surveyed. Problems of representative sampling, measurement uncertainties, and measurement traceability are discussed.

After this session, participants will be able to

1. Describe the current chemical and mass spectrometry methods typically used to characterize special nuclear materials.

2. Describe basic techniques of mass, volume, and flow measurement.

3. Identify the principle sources of error for each of the above methods.

TOPIC 18: ELEMENTS OF CHEMICAL AND BULK MEASUREMENT TECHNOLOGY

Carleton Bingham
U.S. Department of Energy
New Brunswick Laboratory

I. INTRODUCTION

Chemical and/or instrumental measurements are performed on materials in a variety of processes generically called the nuclear fuel cycle. The results of such measurements are used for a variety of purposes, some of which are:

a. to ascribe a financial value to a material;
   - What is the elemental content of this ore?
   - Based on a current market, what exchange of money is represented by an exchange of material between a supplier and a buyer?

b. to demonstrate that a given process is operating as specified;
   - Is the composition of this powder blend as specified?
   - Is the salt and acid content of this dissolver solution as specified such that proper separation of fissile material will occur?

c. to assure that regulatory requirements relating to health and safety are being met.
   - What is the elemental content of this waste stream?
   - How much material is stored in this criticality control area?

In almost all instances, the results of measurements given as examples above also are required for nuclear materials accountability as part of a country's safeguards program.

The measurement system must be defined to include sampling and subsampling processes as well as the measurement processes.

The results of the very best possible measurement can be invalidated by a sample that is not representative of the bulk from which it was taken (Fig. 1).

II. CHEMICAL ASSAY

Chemical assay of a material involves observing (i.e., measuring) a chemical or a physical property of that material which is directly related to the presence of the element of interest in the given material. Other elements which contribute to the observed response constitute interferences and are the source of biased measurements.

A. Gravimetry

A most useful physical property of a material of known composition is its mass. From the measured mass of material and a known (or assumed) concentration of the element of interest, the quantity of element present may be calculated. This is the principle underlying gravimetric methods of analysis. Gravimetric methods involve an initial measurement of the mass of a material and, if the composition of material is not known, it is treated to change its composition to one of known stoichiometry from which the elemental concentration may be calculated. Gravimetric assay is <u>not</u> a direct chemical measurement of the element of interest. Several examples from the fuel cycle will serve as illustrations.

a. $\underset{\text{mass x}}{UF_6} \xrightarrow{H_2O} UO_2F_2 \xrightarrow[\Delta]{H_2O} \underset{\text{mass y}}{U_3O_8}$

The uranium concentration, g U/g sample, is calculated:

$$\left(\frac{3U}{U_3O_8}\right) \cdot \frac{y}{x}$$, where U = atomic weight of uranium and U_3O_8 = molecular weight of urano-uranic oxide.

The ratio $\frac{3U}{U_3O_8}$ is called • <u>gravimetric factor</u>.

DESTRUCTIVE ANALYSIS OF NUCLEAR MATERIAL

SAMPLING ------ DISSOLUTION ---- ALIQUANTING ------ ASSAY ------- ISOTOPIC

CALCULATION

SAMPLING	DISSOLUTION	ALIQUANTING	ASSAY	ISOTOPIC
MASS	MASS	MASS	MASS	CURRENT
VOLUME	VOLUME	VOLUME	VOLUME	
			CURRENT	
			TIME	

Figure 1

b. $3UO_2 + O_2 \xrightarrow{\Delta} U_3O_8$

mass w mass y

The uranium concentration is calculated:

$$\left(\frac{3U}{U_3O_8}\right) \cdot \frac{y}{w}$$

c. $3UO_2(NO_3)_2 + 3H_2O \xrightarrow{\Delta} U_3O_8 + 6HNO_3$

mass s mass y

The uranium concentration is calculated:

$$\left(\frac{3U}{U_3O_8}\right) \cdot \frac{y}{s}$$

In each of the above examples, a correction to the observed mass (y) of U_3O_8 must be made to account for the presence of impurities. These impurity elements are usually determined by emission spectrography or by spark-source mass spectrography. When the total impurity content exceeds 0.05%, by weight, the uncertainty in the impurity correction becomes too large to permit accurate assay using the gravimetric method.

There can be several sources of systematic error in the gravimetric assay for uranium. One, the stoichiometry of the ignited product assumed to be U_3O_8 may vary, depending upon local variations in the ignition process. The partial pressure of oxygen (function of altitude above sea level) during the ignition process can be a source of site-to-site difference. For routine or production operations using gravimetric assay, it is essential to verify periodically the actual stoichiometry of the ignited product. Even under carefully controlled ignitions involving high purity material, the actual uranium concentration in "U_3O_8" may be 0.02-0.03% less than that calculated based on a stoichiometric compound.

A second source lies in the atomic weight selected for uranium. The gravimetric factor (g U/g U_3O_8) varies according to the atomic weight of uranium. The atomic weight is a function of

uranium enrichment and/or its irradiation history. The gravimetric factor differs by nearly 0.2% between low-enriched and high-enriched uranium. For accurate gravimetric assay measurements it is essential that the correct uranium atomic weight be used.

Gravimetric assay, especially where measurements of the same material is involved, is usually characterized by high precision but not always by high accuracy in the assay data (Fig. 2).

Gravimetric assay is less widely applied to plutonium materials. Ignition of PuO_2 in an air atmosphere at 1250°C is reported to yield stoichiometric PuO_2. Plutonium assay of nitrate product solutions or of oxide product streams could be performed as given in the above examples for uranyl nitrate and uranium dioxide.

The same considerations pertaining to systematic errors would apply in plutonium gravimetric assay - i.e., corrections for non-volatile impurities in the ignited product, assurance of stoichiometry, and assumption of atomic weight for plutonium.

B. Titrimetry

The chemical reactivity of a material is a measurable property which is most frequently used in titrimetric methods of analysis. Titrimetric methods involve the measured addition of known quantities of a substance which react in a specified manner with an element of interest. In the determination of fissile material, use is made most often of chemical reactions based on reduction-oxidation ("redox") properties of the element. A weighed sample is dissolved, the solution is treated to adjust the element to a known oxidation state, and a reactant is added until the end point of a specified reaction is reached. The end point may be indicated by the change in color of a reagent ("indicator") added for this purpose. The end point may be indicated by a rapid change in potential of the solution vs. a reference electrode (potentiometric detection) or by a change in the current conducted by the solution (amperometric detection)

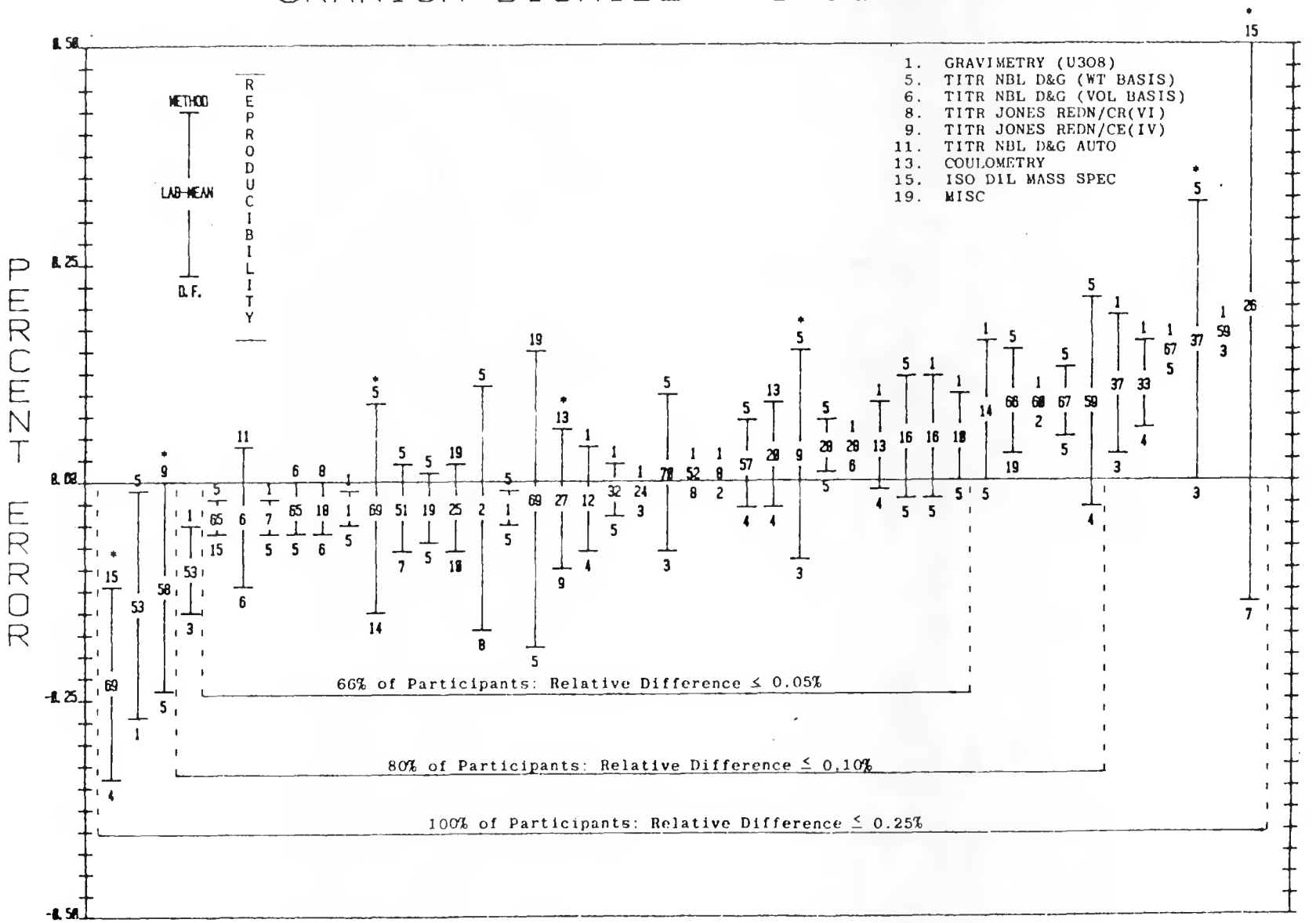

URANIUM DIOXIDE -- U-CONC (1978)
1. GRAVIMETRY (U3O8)
5. TITR NBL D&G (WT BASIS)
6. TITR NBL D&G (VOL BASIS)
8. TITR JONES REDN/CR(VI)
9. TITR JONES REDN/CE(IV)
11. TITR NBL D&G AUTO
13. COULOMETRY
15. ISO DIL MASS SPEC
19. MISC
METHOD
LAB MEAN
D.F.
REPRODUCIBILITY
66% of Participants: Relative Difference ≤ 0.05%
80% of Participants: Relative Difference ≤ 0.10%
100% of Participants: Relative Difference ≤ 0.25%

Figure 2

(Figs. 3 and 4). From the quantity of reactant added, the quantity of fissile material in the sample may be calculated.

Various procedures exist for solubilizing a sample of material taken from the fuel cycle. Complete dissolution is desirable but not essential if adequate methods exist for determining the fissile material content of any small residue. A summary of solvents used for various nuclear materials is given in Figs. 5 and 6.

Titrimetric measurements of uranium in solution are usually based upon redox reactions involving the U(IV) and U(VI) oxidation states.

Summary examples of frequently used methods are given below.

$U(soln) + reductant \longrightarrow U(IV)$; $U(IV) + oxidant \longrightarrow U(VI)$

a. Jones reductor - potassium dichromate

$$U(soln) \xrightarrow{Zn(Hg)} U(III) \xrightarrow{air} U(IV)$$

$$3U(IV) + 2Cr(VI) \longrightarrow 3U(IV) + 2Cr(III)$$

b. Jones reductor - ceric sulfate

$$U(soln) \xrightarrow{Zn(Hg)} U(III) \xrightarrow{air} U(IV)$$

$$U(IV) + 2Ce(IV) \longrightarrow U(VI) + 2Ce(III)$$

c. NBL-Modified Davies-Gray

$$U(soln) \xrightarrow[H_3PO_4]{Fe(II)} U(IV)$$

$$3U(IV) + 2Cr(VI) \xrightarrow{V(IV)} 3U(VI) + 2Cr(III)$$

Titrimetric measurements of plutonium are based upon redox reactions involving the Pu(III), Pu(IV) and Pu(VI) oxidation states.

Case 1.

$Pu(soln) + reductant \longrightarrow Pu(III)$; $Pu(III) + oxidant \longrightarrow Pu(IV)$

Typical Titration Curves Using Potentiometric End Point Detection

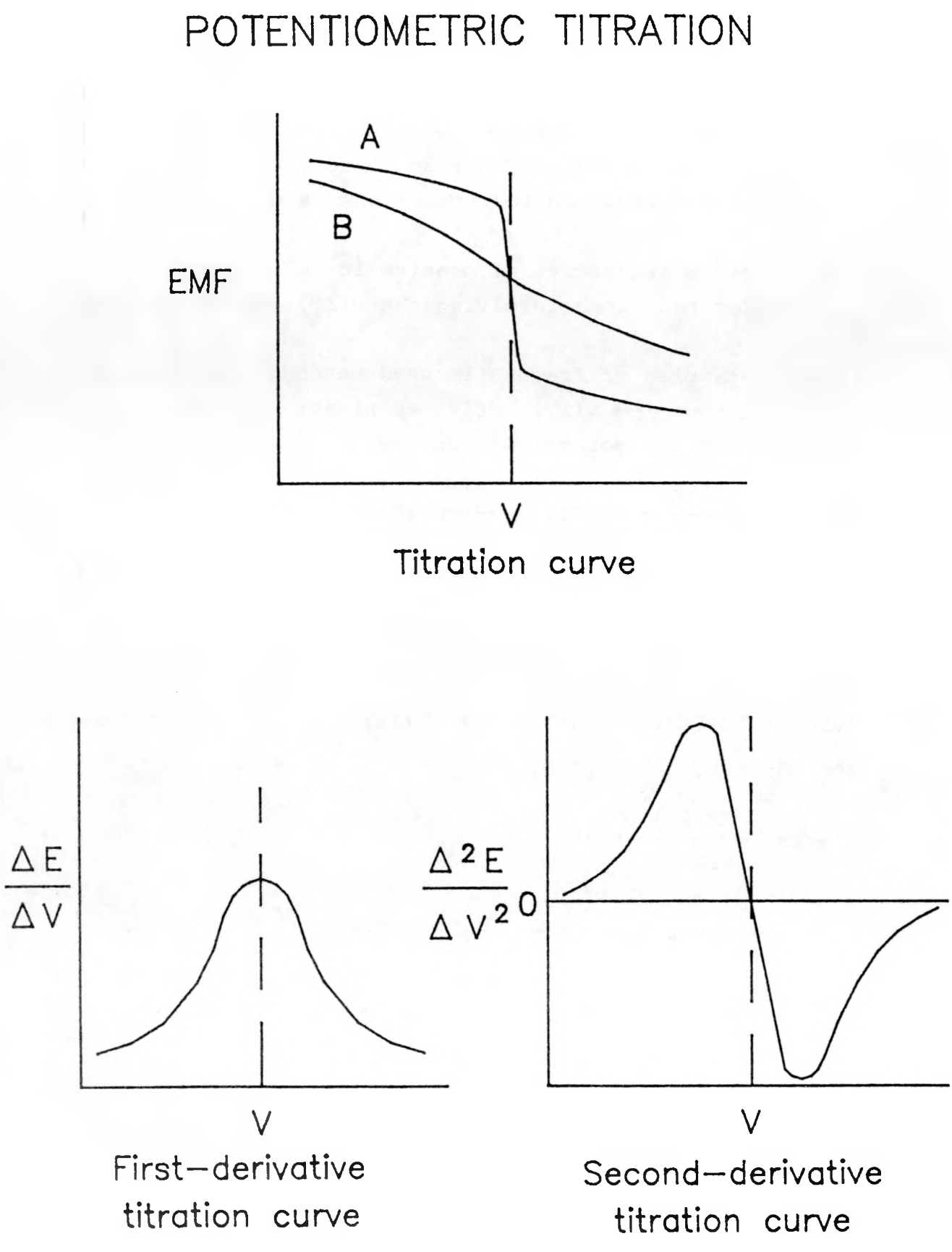


Figure 3

Typical Titration Curves Using Amperometric End Point Detection

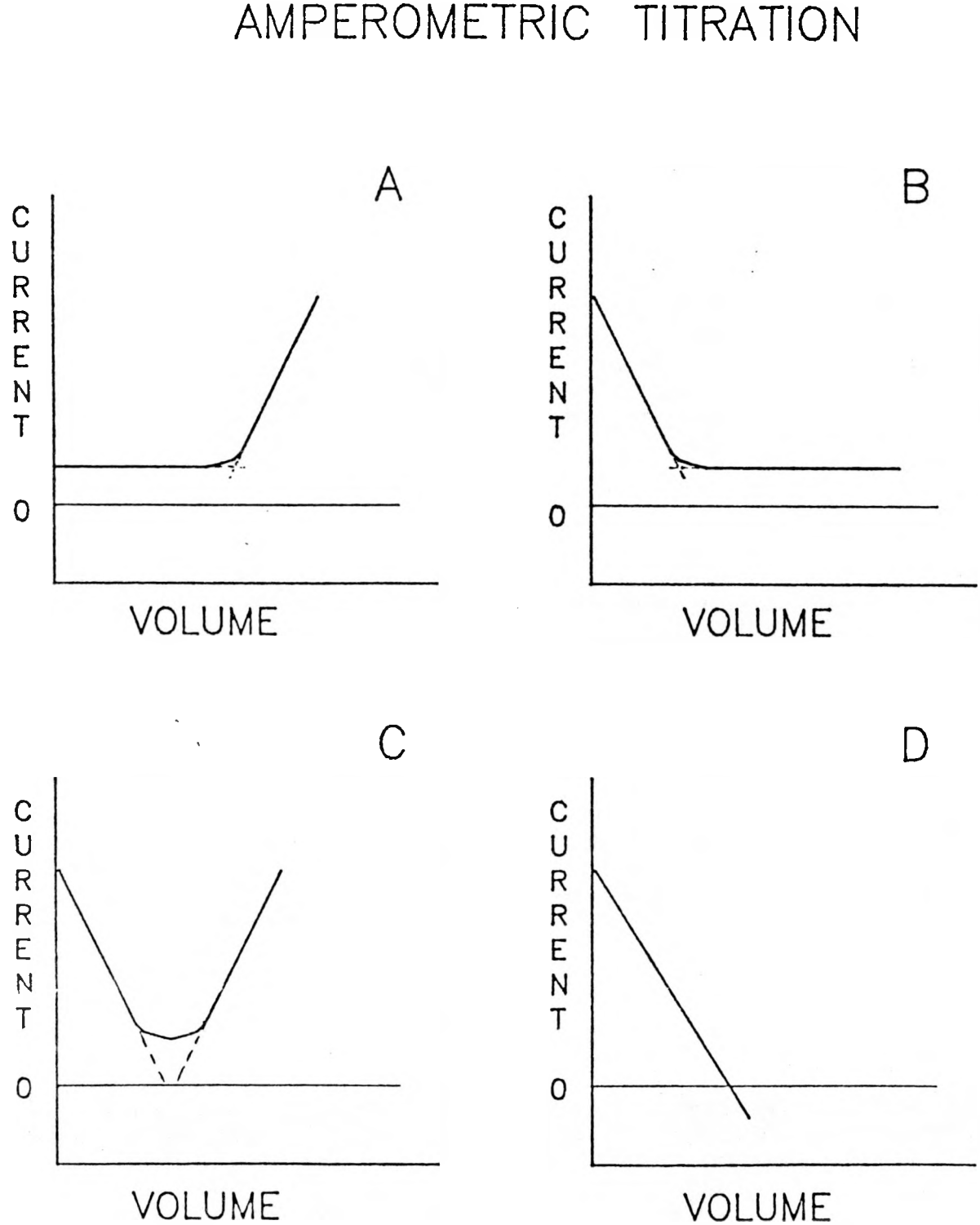


Figure 4

Solvents Used for Uranium-Containing Materials

Material	Treatment
U, UO_3, U_3O_8, UF	Dissolve in HNO_3. Fume sample aliquant with H_2SO_4.
UO_2 powders and pellets, ammonium diuranate, wastes, UO_2-ThO_2, etc.	Dissolve in HNO_3 and filter. Fuse residue in $NaHSO_4$ or Na_2CO_3, combine solutions. Fume sample aliquant in sulfuric acid.
Ore concentrates	Dissolve in nitric acid. Fume sample aliquants with H_2SO_4 and HF.
HTGR fuel beads	1. Ignite to remove carbon. Fuse with Na_2CO_3. Dissolve cake and fume with H_2SO_4 and HF to remove silica. Precipitate U with NH_4OH to remove excess fusion salts. Dissolve residue in HNO_3 and proceed. Fume sample aliquants in H_2SO_4.
	2. Ignite to remove carbon. Treat with Cl_2 at 900°C to decompose SiC, ignite as in 1, dissolve as with UO_2-ThO_2.
UC, UC_2	Ignite and dissolve in HNO_3. Fuse any residue. Fume sample aliquant in H_2SO_4.
U-Al, U-Si, UO_2-SS	Dissolve in HCl-HNO_3. Fume with perchloric acid, filter. Volatilize silica with HF, fuse remaining residue with Na_2CO_3. Fume sample aliquant in $HClO_4$.
Fissium alloy Fissium dross	Dissolve in HCl-HNO_3. Treat residue with NaOCl and NaOH and acidify with HCl. Combine solutions. Fume sample aliquant in $HClO_4$. (Residue from dross requires fusion with NaOH.)
Ash samples	Fuse with Na_2CO_3-$NaNO_3$ (5:1) and $NaHSO_4$, as necessary. Dissolve cake in HNO_3-$HClO_4$. Volatilize silica with HF and fume with $HClO_4$. Fume sample aliquant with H_2SO_4.
UO_2-ZrO_2-Nb-Zr, UO_2-BeO	Dissolve in HNO_3-HF. Fume sample aliquant in H_2SO_4 or $HClO_4$. (Large quantities of Nb may be removed by precipitation with SO_2 if desired.)
Dissolver solutions Organic solutions Wastes	Homogenize and reconstitute multiphase mixtures where necessary. Destroy organics with hot H_2SO_4-HNO_3. Fume sample aliquant in H_2SO_4.
U-Zr	Dissolve in HF. Fume sample aliquant in H_2SO_4 or $HClO_4$.

Figure 5

Solvents Used for Plutonium-Containing Materials

Material	Treatment
Pu, Pu-Al	H N HCl or 18 N H_2SO_4
U-Pu-Mo	3 N HCl - 8 N HNO_3 - 0.1 N HF
PuO_2, $(U\text{-}Pu)O_2$	8 N HNO_3 - 0.1 N HF; fuse in $NaHSO_4$
PuO_2, fired < 800°C	fuse in $NaHSO_4$; sealed tube - HCl+$HClO_3$
(U-Pu)C	ignite, 8 N HNO_3 - 0.1 N HF; fuse in $NaHSO_4$
"calcined ash"	leach in 8 N HNO_3 - 0.1 N HF, fuse in $NaHSO_4$; fuse in $NaHSO_4$
"brick residues" - (Al_2O_3, MgO, CaO, Fe_2O_3, SiO_2)	fuse in $NaHSO_4$
"grinder sludge" - (SiC)	fuse in $NaHSO_4$
Pu-fissium	sealed tube - HCl+$HClO_4$

Figure 6

a. Titanous reduction-ceric sulfate

$$Pu(soln) \xrightarrow{Ti(III)} Pu(III)$$

$$Pu(III) + Ce(IV) \longrightarrow Pu(IV) + Ce(III)$$

b. Lead or Jones reductor - ceric sulfate

$$Pu(soln) \xrightarrow[Pb]{Zn(Hg)\ or} Pu(III)$$

$$Pu(III) + Ce(IV) \longrightarrow Pu(IV) + Ce(III)$$

c. Jones reductor-potassium dichromate

$$Pu(soln) \xrightarrow{Zn(Hg)} Pu(III)$$

$$3Pu(III) + Cr(VI) \longrightarrow 3Pu(IV) + Cr(III)$$

Case 2.

Pu(soln) + oxidant → Pu(VI); Pu(VI) + reductant → Pu(IV)

a. Silver oxidation - ferrous sulfate

$$Pu(soln) \xrightarrow{AgO} Pu(VI)$$

$$Pu(VI) + 2Fe(II) \longrightarrow Pu(IV) + 2Fe(III)$$

b. Perchlorate oxidation - ferrous sulfate

$$Pu(soln) \xrightarrow[\Delta]{HClO_4} Pu(VI)$$

$$Pu(VI) + 2Fe(II) \longrightarrow Pu(IV) + 2Fe(III)$$

In addition to the use of chemical reagents to produce redox reactions, these reactions may be caused to occur using applications of electrochemical principles. Electrogenerated reductants or oxidants may be produced in a reaction cell at constant current or reactions driven at controlled potential. In each of these cases, measurement of the time-integrated current (i.e., coulombs) required to achieve the desired reaction is related through Faraday's Law to the quantity of element reacted.

$$U(IV) + 2V(V) \xrightarrow{\int idt} U(VI) + 2V(IV) \quad \text{at constant current}$$

$$U(VI) + 2e \xrightarrow{\int idt} U(IV) \quad \text{at controlled potential}$$

$$Pu(III) \xrightarrow{\int idt} Pu(IV) + e \quad \text{at controlled potential}$$

Redox measurements using chemical reagents are normally calibrated using reference material of the desired element (e.g., uranium metal, plutonium metal or their oxides, etc.) or reference material of an oxidant/reductant (e.g., $K_2Cr_2O_7$, As_2O_3, $Na_2C_2O_4$, etc.). Controlled-potential or constant-current coulometry may be calibrated chemically with reference materials of a given element, but they may also be calibrated by reference to fundamental constants derived from basic units (i.e., the Faraday).

Other elements which undergo redox reactions under the conditions of the above methods or which prevent the element of interest from quantitatively reacting constitute interferences and consequently give rise to biased measurement results. The effect of the presence of interfering elements can be eliminated by separation of the uranium/ plutonium from the interfering species. If the interferant reacts quantitatively with the "titrant", it can be determined by another method and a correction applied for its presence.

Systematic errors exceeding 0.1% can occur by failing to apply a correction for the appropriate atomic weight of uranium or plutonium in the sample measured (Figs. 7 and 8).

III. ISOTOPIC ASSAY

Measurements which are necessary to define the atomic weight of an element are performed using mass spectrometry. In thermal emission or thermal ionization mass spectrometry a sample (10^{-9} to 10^{-6} g), which has been deposited on a tungsten, tantalum, or rhenium filament, is vaporized under high vacuum and its elemental species ionized. Under high potential gradient, the

Systematic Measurement Error Resulting From Incorrect Uranium Atomic Weight
Measurement of High-Enriched Uranium with a System Calibrated for Low-Enriched Uranium

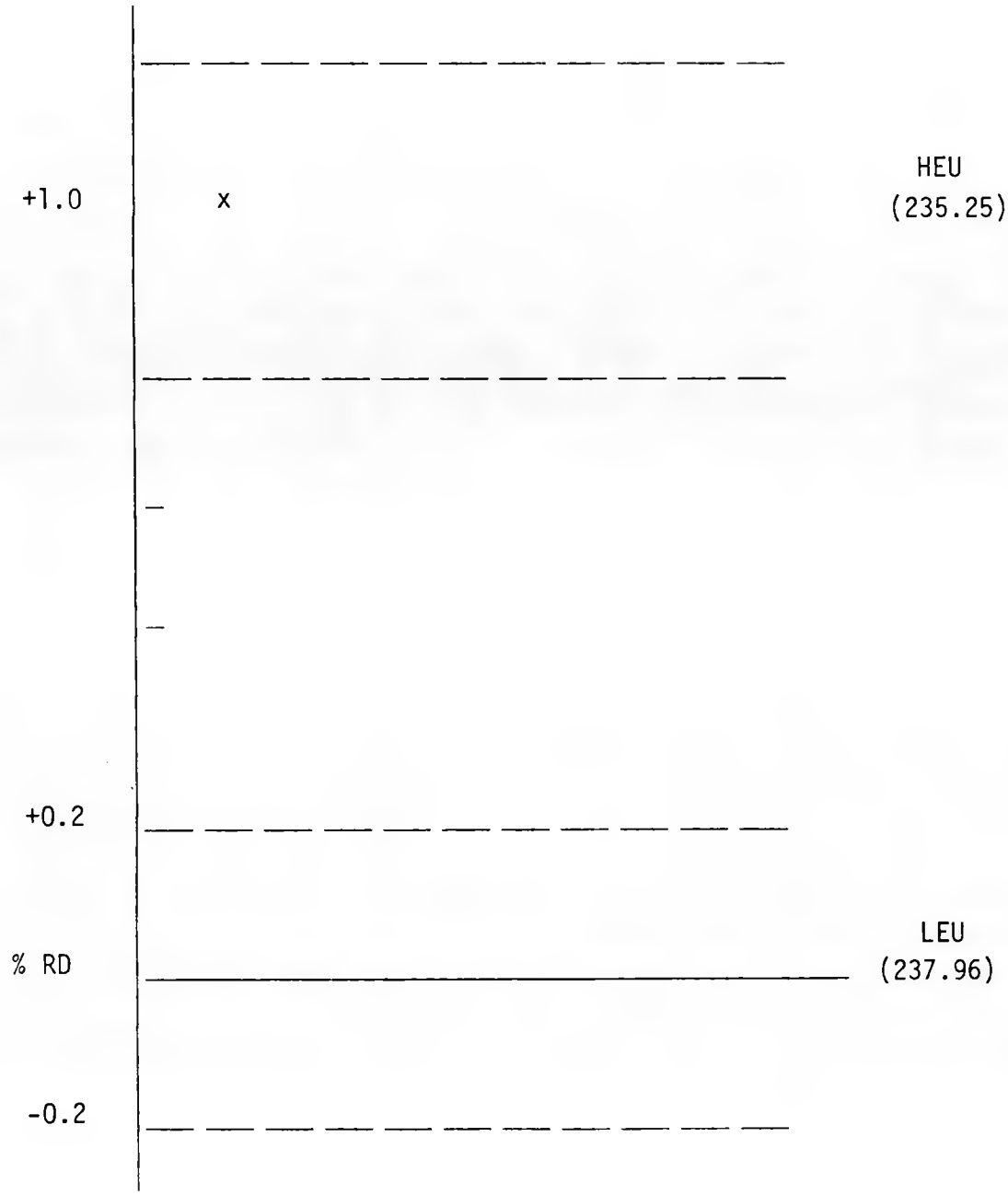


Figure 7

Systematic Measurement Error Resulting From Incorrect Plutonium Atomic Weight

Measurement of Reactor-Grade or Recycled Plutonium with a
System Calibrated for Weapons-Grade Plutonium

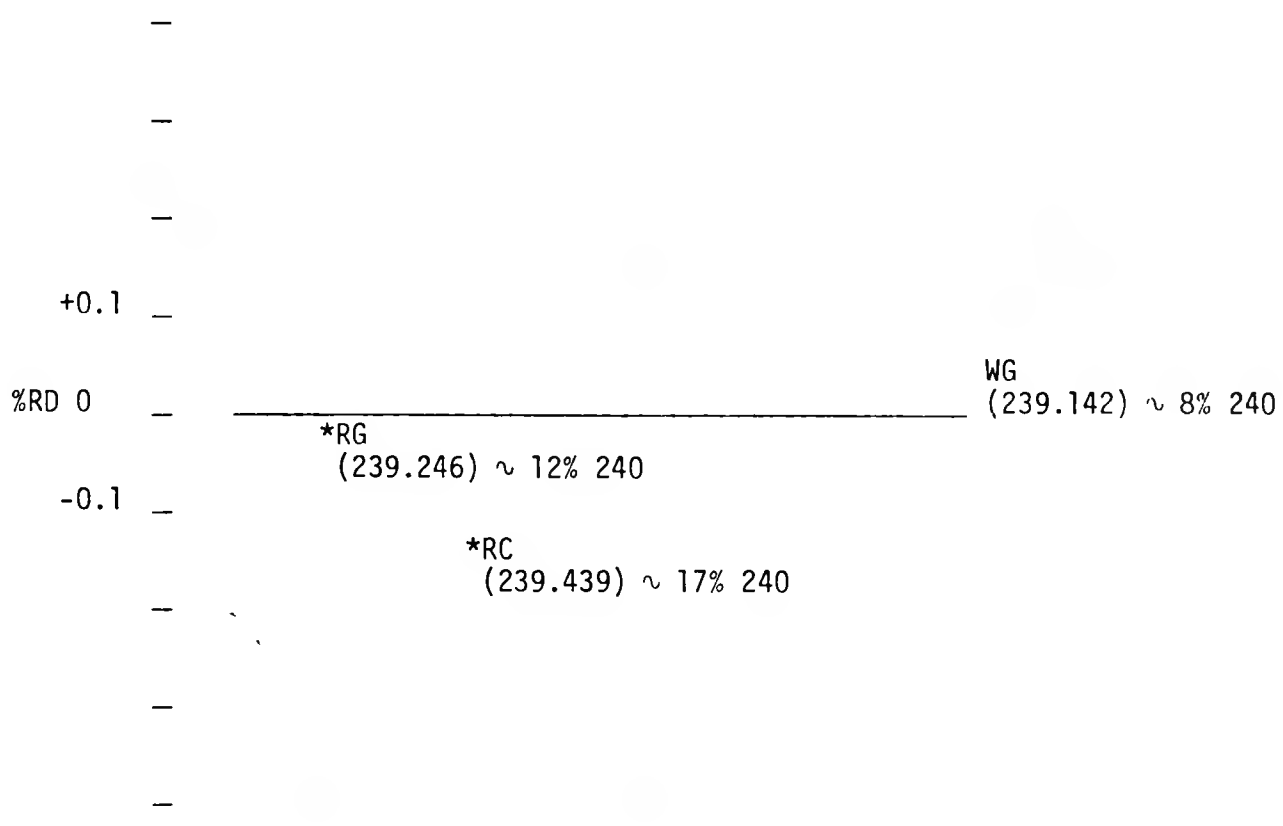


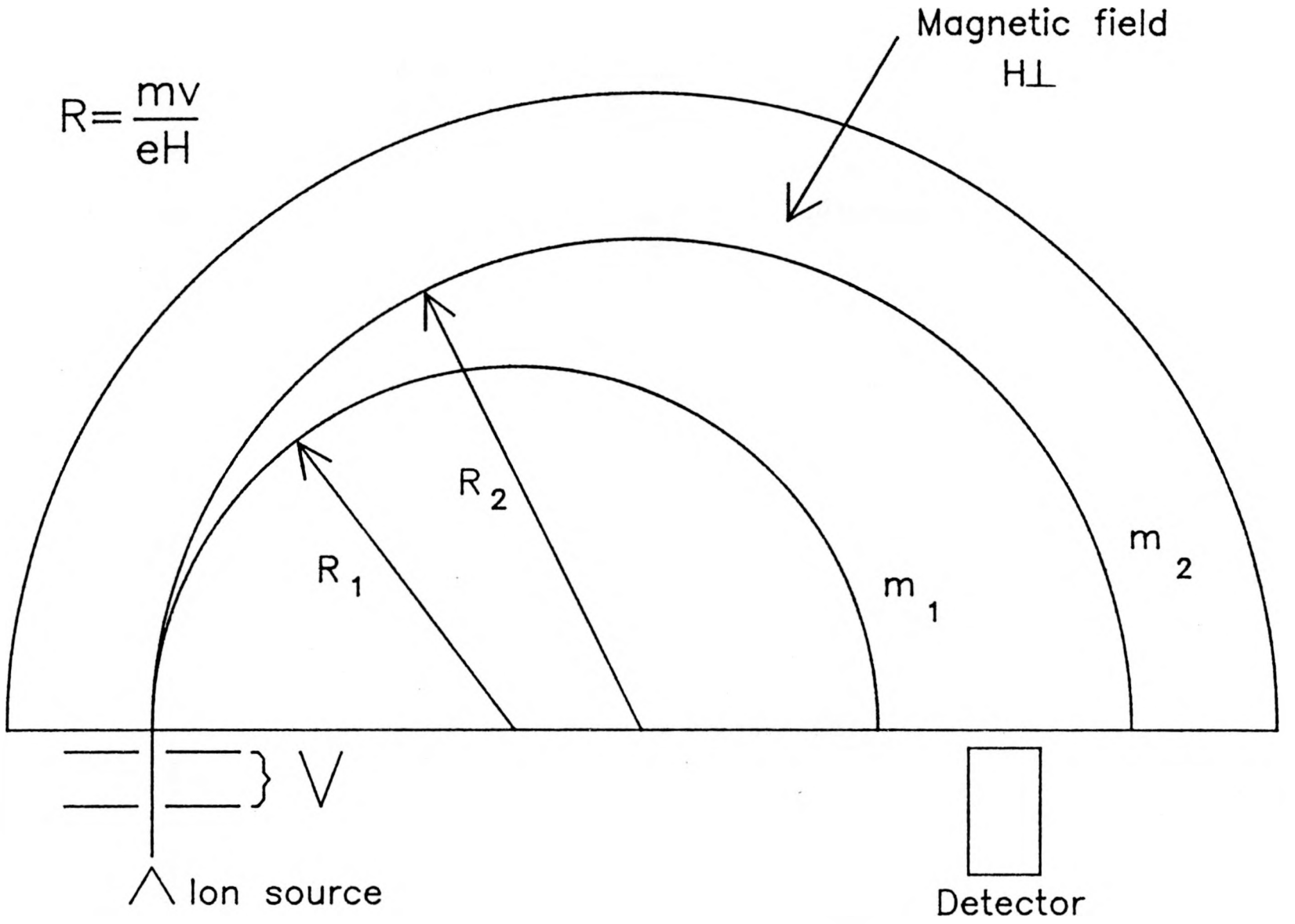
Figure 8

ionized species undergo acceleration into an evacuated tube (Fig. 9). A magnetic field is imposed which focuses the beam, i.e., defines the curvature of the path which a charged particle follows. The curvature, which is a function of the mass of the charged particle, is altered by the imposed field such that the ion mass (nuclide) of interest is focused upon the detector. The field may be changed to focus sequentially specific masses, e.g., 233, 234, 235, 236, 238 for uranium or 238, 239, 240, 241, 242 for plutonium. The measured ion current at each mass position is proportional to the isotopic abundances for a given element. Measurements are usually recorded as ratios of the ion current of a given mass to that of a more abundant mass, e.g. $^{234}U/^{235}U$, $^{240}Pu/^{239}Pu$. Calibration of a mass spectrometer against certified reference materials is required to relate a measured ratio (R_{obs}) to the certified ratio (R_{th}).

$$K = \frac{R_{th}}{R_{obs}}$$

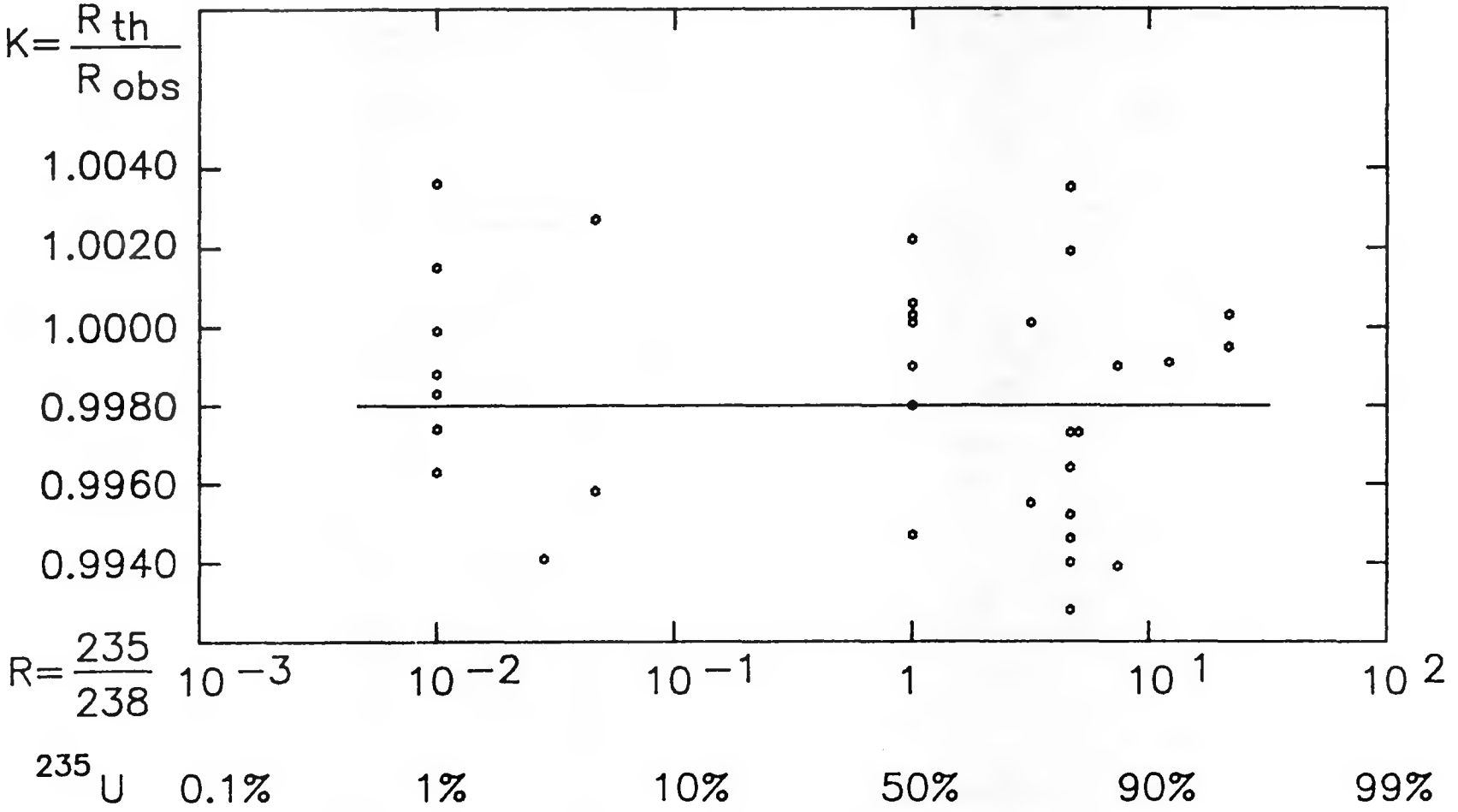
The accuracy and precision (i.e., the variations and uncertainties in K) with which isotope ratios can be measured are strongly dependent upon filament temperatures, sample size deposited on a filament, chemical species loaded onto a filament, and sample purity. If these operating parameters are not controlled, systematic errors of $\pm$ 0.5% will occur in the measurements of isotope ratios and hence in the reported isotopic abundances. Typical control procedures will produce precision of $\pm$ 0.2%. When careful control is maintained to reproduce all parameters $\pm$ 0.06% can be attained (Figs. 10, 11, 12).

Isotopic ratio measurements are performed at enrichment plants using mass spectrometers which ionize the gaseous UF_6 by electron bombardment. These so-called "gas mass spectrometers" are capable of measuring isotope ratios to a greater precision than normally practiced with thermal ionization instruments (Fig. 13). They require calibration using prepared mixtures of



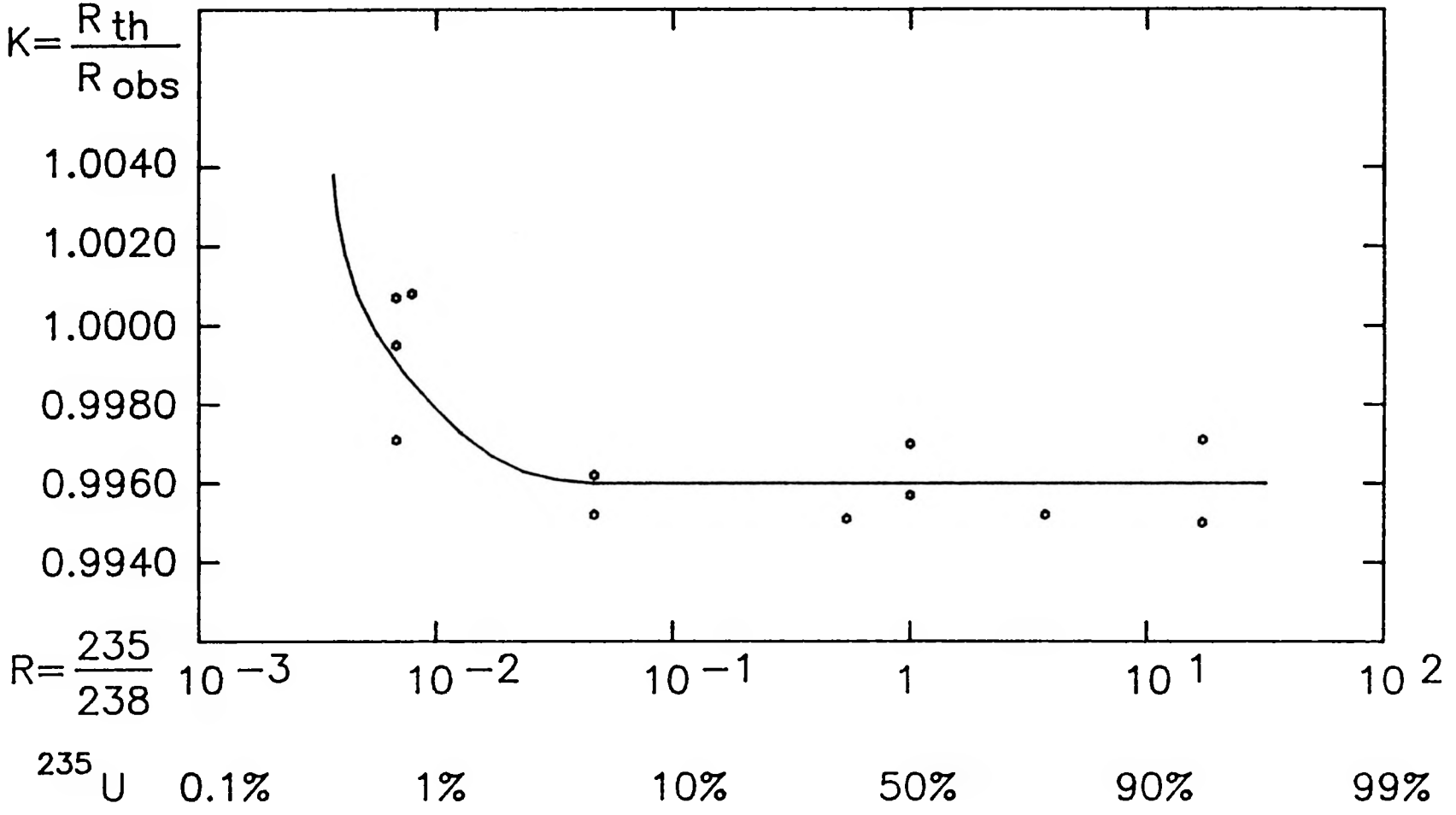
Principle of the Mass Spectrometer
Mass Spectrometric Measurement of Isotopic Abundance

Figure 9



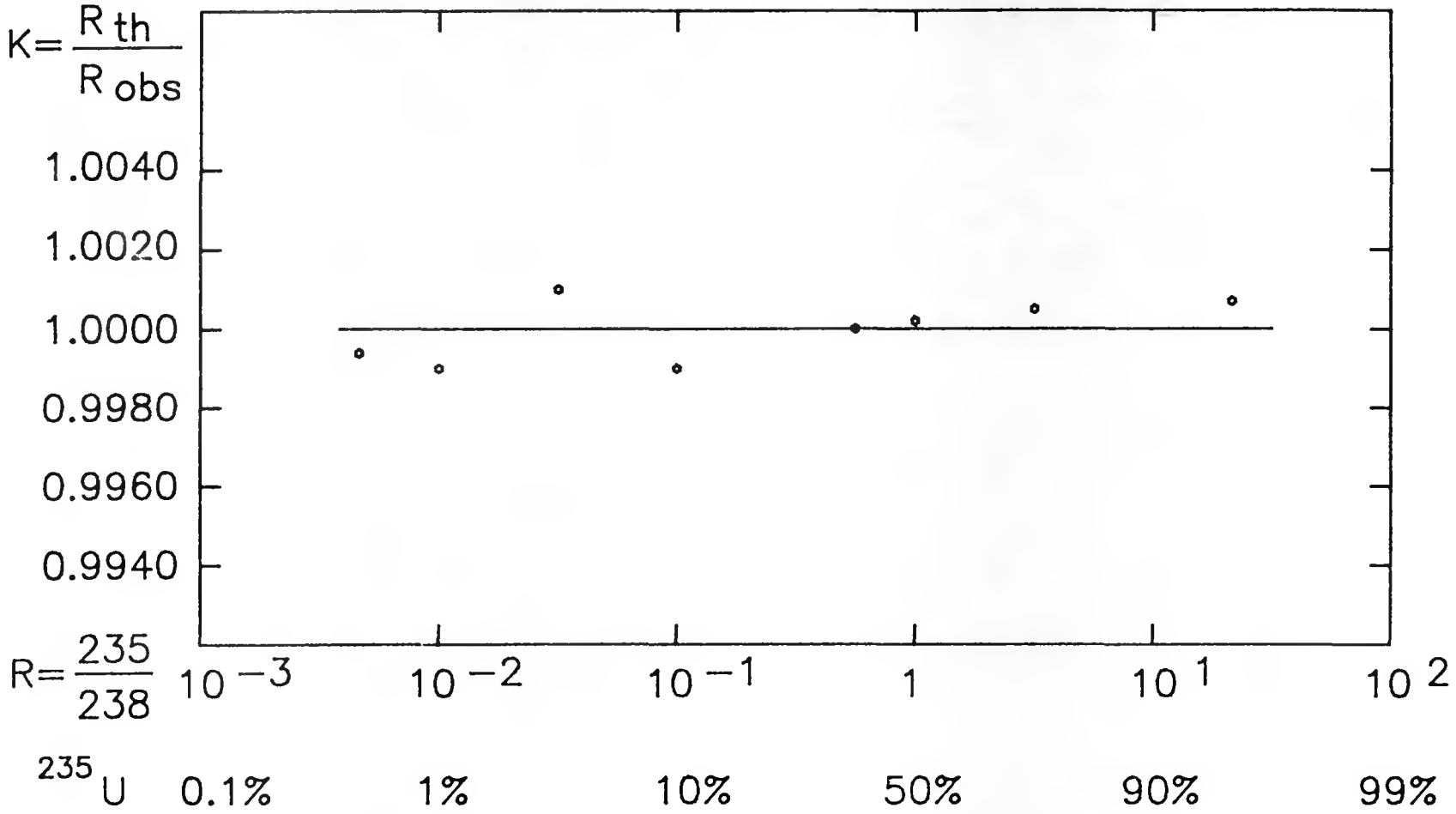
Calibration of large R range without control of thermal ionization parameters (K=0.998+0.005).

Figure 10



Calibration of large R range with some control of thermal ionization parameters ($K=0.997\pm0.002$ for $10^{-1} < R < 10^{+1}$ only).

Figure 11



Calibration of large R range with careful control of thermal ionization parameters (K=1.0000+0.0006).

Figure 12

Comparative ^{235}U Measurements in Uranium Dioxide

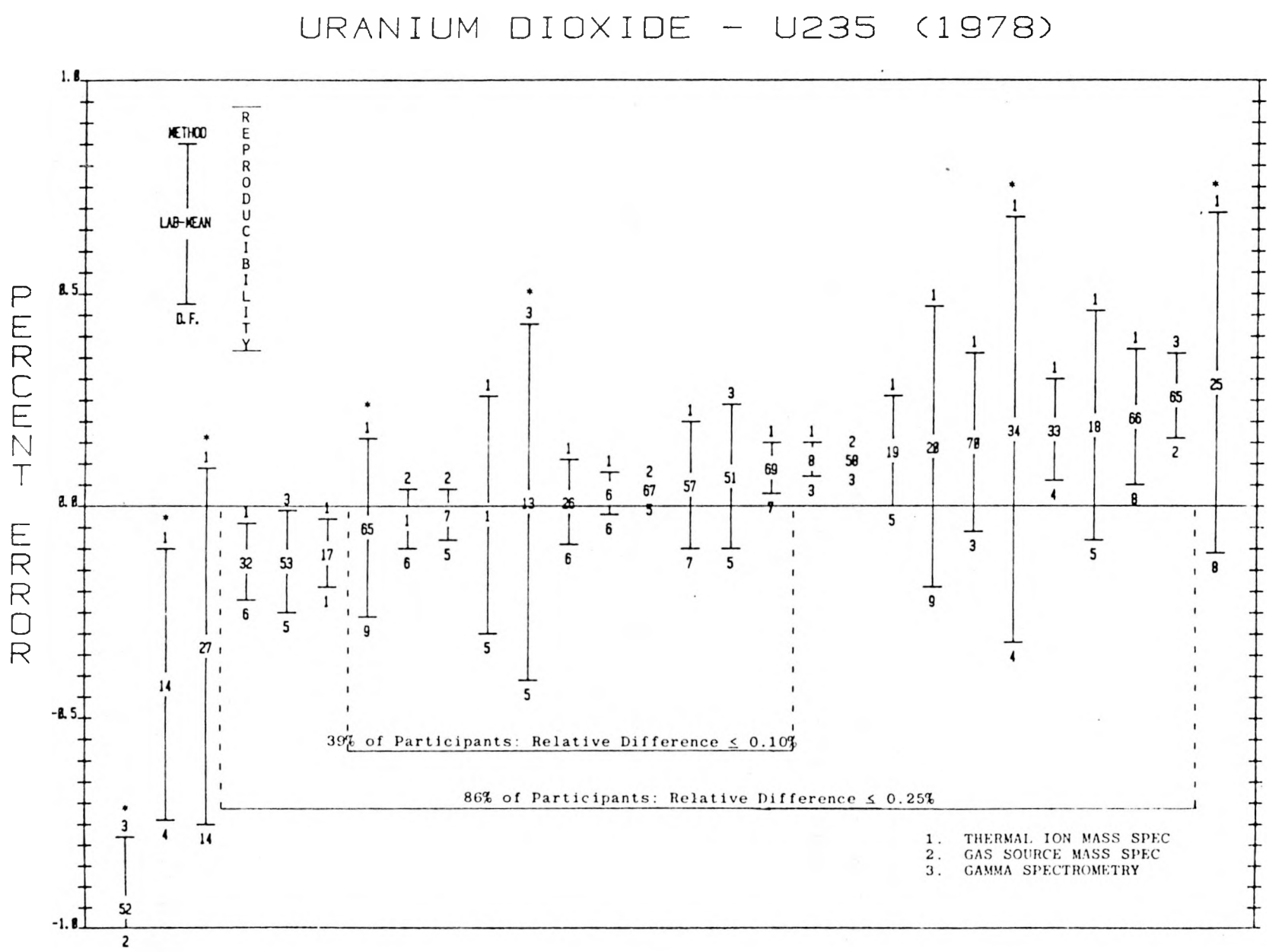


Figure 13

UF_6 of known isotopic composition. These instruments are subject to "memory effects" from previous measurements if care is not taken to preclude such an effect.

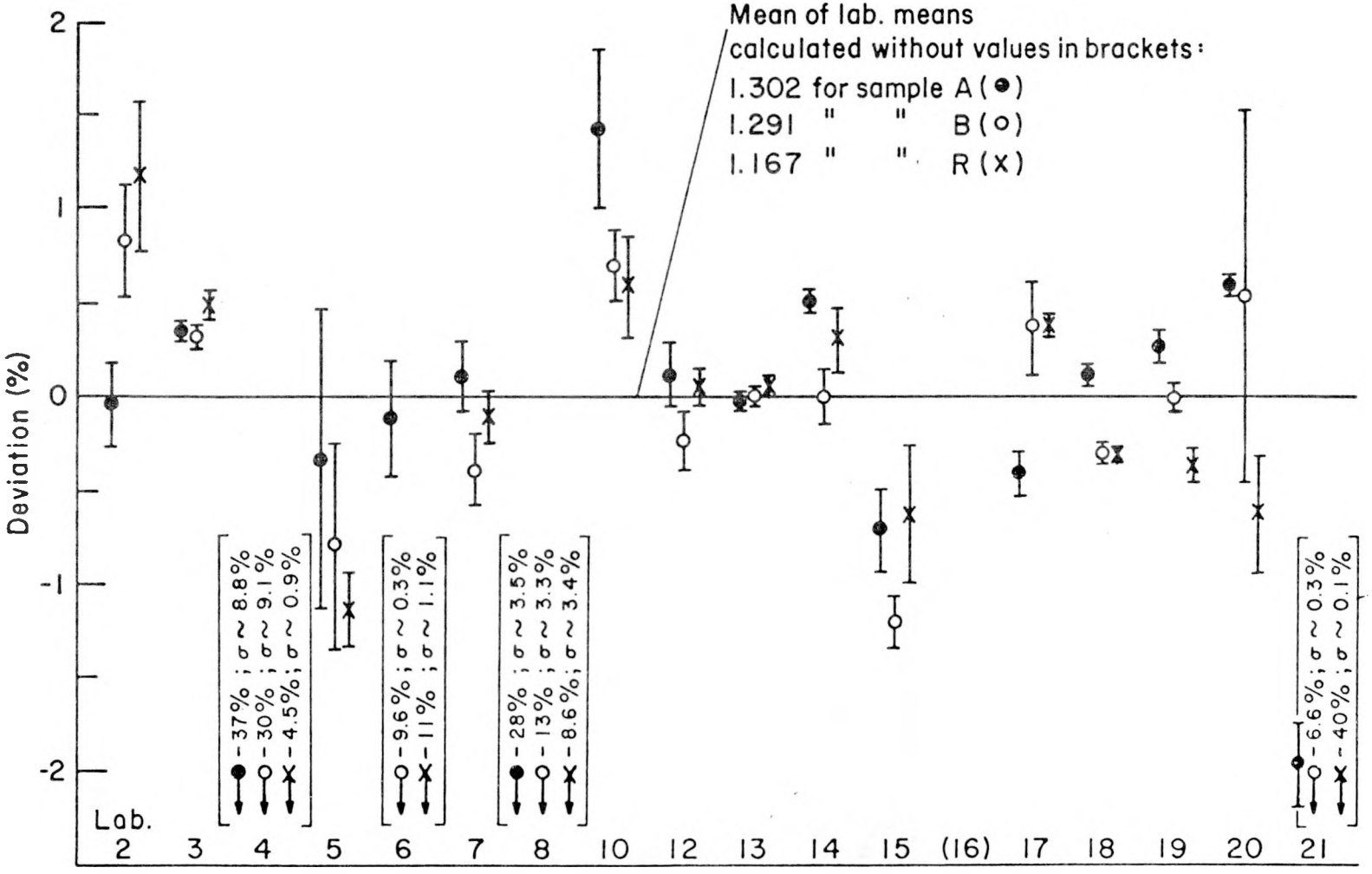
Mass spectrometric measurement of elemental concentration may be performed using a method called isotope dilution mass spectrometry (IDMS). In the fuel cycle this measurement method is most frequently applied to measurements of the input accountability tank of fuel reprocessing plants, but may be used elsewhere. IDMS involves adding a known quantity of an element of known isotopic composition (called the "spike") to a sample solution containing the same element of differing isotopic composition and, after chemical and isotopic equilibration, measuring isotopic ratios. From the change in the isotopic ratio of the sample caused by the spike, the elemental content of the sample may be calculated. IDMS requires complete isotopic mixing and equilibration between spike and sample before samples are withdrawn for measurement. Quantitative subsampling is not required. High measurement specificity is possible if subsequent chemical contamination does not occur. Small samples (10^{-5} to 10^{-10} g) can be measured - a great advantage where highly radioactive samples are involved.

Systematic errors may arise - beyond those mentioned for thermal ionization - due to incomplete isotopic equilibrium between sample and spike, because of sample contamination with the element to be measured, before or after spiking, and because of spike calibration errors (Fig. 14). Sample contamination is a major source of error as the quantity of the element of interest in the sample decreases. Uranium appears as a trace impurity in reagents or their containers in the 10^{-9} to 10^{-13} g/g range. Specially purified reagents in pre-cleaned containers are required where very small samples are to be loaded onto filaments.

IV. BULK PROPERTIES

At all processing points within the nuclear fuel cycle, measurement of bulk properties (i.e., mass, volume, density,

Comparative Measurements of Pu ^{242}/239 Ratios in Spiked Dissolver Solution



(Mean values per laboratory; error bars indicate ± 1σ-range of these means)

ISOTOPIC RATIO (242)/(239) IN SPIKED DISSOLVER SOLUTION

Figure 14

flow, etc.) provide essential information in determining the quantity of material present.

A. Mass

Measurements of mass range from gram and fractional gram samples taken from a process line to megagram (metric ton) containers of UF_6 at enrichment plants. Periodic calibration of scales and balances, as well as verification of performance, is essential to assure that accurate mass measurements are being performed. Mass artifacts, traceable to U.S. NBS and other national centers of metrology, are commercially available for laboratory balances. A Measurement Assurance Program (MAP) for weighing UF_6 cylinders has been established by NBS and replicate mass standard cylinders have been manufactured for verification of mass measurement at enrichment and conversion facilities.

A semi-portable device for weighing UF_6 cylinders in the field by IAEA inspectors has been developed by NBS. Operation is based on a load cell concept. Operational testing of a prototype has demonstrated better than $\pm$ 0.05% precision. Applications of load cell principles are being studied for direct measurements of the mass of solution delivered to or from input accountancy tanks at reprocessing plants.

The reliability of mass measurements using scales or balances is influenced by the environment in which the devices are used. Variations in temperature, humidity, and air currents all have an effect. Accurate measurements, especially those made at different geographical locations, may require applications of corrections for buoyancy which vary with the density of air (i.e., atmospheric pressure or altitude relative to sea level). Such corrections result from the differences in density of the object (substance) weighed and the density of the counterweights used on the balance.

B. Volume

Volume measurements of large or small volumes are performed by relating observations (measurements) of mass, density, differential pressure, etc., to volume via a calibration function.

Large processing, or accountability tank, volumes are calibrated by transferring known quantities (mass or volume) of a liquid, usually water, into the tank and measuring the height of liquid by differential pressure techniques such as a bubbler tube, pressure gauges or other transducers. Calibration equations are generated which relate the observation to the tank volume. The calibration function is subject to bias resulting from buildup of a "process heel" in the bottom of the tank and to in-plant temperature variations which affect tank volume.

Development and evaluation of improved volume measurement and calibration technology is being carried out, at (to name a few) NBS, the Savannah River Plant, and the Idaho National Engineering Laboratory in collaboration with the plant at Tokai Mura, Japan. A prototype automated tank volume calibrator is being evaluated at NBS. The calibrator dispenses water, which has passed through two flow meters in series, into the tank to be calibrated.

C. Flow

Measurements of time-integrated flow can be related to volume transfers within given processes and hence to near real-time accountability of at least the location of materials within a sequence of processes. Flow control and documentation logic with microprocessors is being evaluated for application to materials control and accountability in a reprocessing plant. A mobile flow standard, by which to compare in-place flow meters, has been constructed by NBS and demonstrated on waste water streams at the General Electric Fuel Fabrication Facility at Wilmington, NC.

Accuracy of flow meters depends, among other factors, on single-phase flow. To the extent that turbulence produces activation or that the presence of volatile solute/solvent produces substantial vapor components, accuracy of well-calibrated flow meters will be adversely affected.

Figures

INTERNATIONAL TRAINING COURSE ON NUCLEAR MATERIALS ACCOUNTABILITY FOR SAFEGUARDS PURPOSES

SESSION #19: ELEMENTS OF NONDESTRUCTIVE ASSAY (NDA) TECHNOLOGY

SPEAKER: Dr. Hastings Smith

Los Alamos Scientific Laboratory
Los Alamos, New Mexico USA

Monday, June 2, 1980
10:30 a.m.

BIOGRAPHY

Education: B.S. (Physics, 1965); M.S. (Nuclear Physics, 1967); Ph.D (Experimental Nuclear Physics, 1970), Purdue University

Present Position: Scientific Staff Member, LASL Nuclear Safeguards Research and Development (1978-Present)

Present Duties: Research and Development in Gamma-Ray-based Nondestructive SNM Assay Techniques; coordinator of LASL Safeguards Technology Training Program.

Past Positions: Assoc. Prof. of Physics, Indiana University 1972-1978, Post Doctoral Appointee LASL 1970-1972.

Have performed basic research in Nuclear Structure and Nuclear Reactions. Presently developing instrumentation for in-line nondestructive assay of Pu solutions for safeguards purposes.

INTERNATIONAL TRAINING COURSE ON NUCLEAR MATERIALS ACCOUNTABILITY FOR SAFEGUARDS PURPOSES

SESSION #19: ELEMENTS OF NONDESTRUCTIVE ASSAY (NDA) TECHNOLOGY

SPEAKER: Dr. Thomas R. Canada

Los Alamos Scientific Laboratory
Los Alamos, New Mexico USA

Monday, June 2, 1980
10:30 a.m.

BIOGRAPHY

Education: B.S., M.S., Ph.D. (1967) in Nuclear Physics from Indiana University

Present Position: Assistant Group Leader, Q-3, Los Alamos Scientific Laboratory

Present Duties: Research, development, and implementation of NDA techniques to the safeguarding of special nuclear material.

Past Positions: Post-doctoral appointments at Rice University, University of Pittsburgh, and Carnegie-Mellon University (1967-1971), where research activities involved the application of various spectroscopy techniques to nuclear structure problems. Associate Member, Sloan Kettering Institute for Cancer Research (1971-1974), studying the properties and applications of high LET radiation.

Memberships: APS

Session Objectives

SESSION #19: ELEMENTS OF NONDESTRUCTIVE ASSAY (NDA) TECHNOLOGY

Basic techniques will be described in the nondestructive assay of plutonium and uranium in various materials. A variety of assay situations will be considered, and the assay technique(s) best suited to each situation will be delineated.

After the session, the participants will be able to

1. Describe briefly the radioactive decay processes that, for uranium and plutonium, produce characteristic gamma rays and neutrons.

2. State the general principles of at least two gamma-ray and two neutron assay techniques and the method of calorimetry.

3. Cite some general assay situations under which neutron assay techniques would be preferable to gamma-ray measurements.

4. Point out some general assay situations under which gamma-ray assay techniques would be more appropriate than neutron measurements.

5. Identify the NDA techniques that are best suited for the assay requirements in a power reactor/spent-fuel storage facility.

SESSION 19: BASIC NONDESTRUCTIVE ASSAY INSTRUMENTATION

Hastings Smith, Jr.
Los Alamos Scientific Laboratory

I. INTRODUCTION

Most techniques for the nondestructive assay (NDA) of special nuclear material (SNM) take advantage of the fact that these materials emit penetrating radiation in the form of gamma rays and neutrons. (These materials also emit alpha particles, which are not very penetrating, but which contribute to the heat produced by samples of SNM. This heat is used in the calorimetric assay of SNM, but no calorimetry instrumentation will be discussed here.) As we have described earlier, some of this radiation is emitted spontaneously, while in other cases the SNM can be induced to emit gamma rays and neutrons. In this section, we will discuss some of the instruments that have been developed to determine the amount of SNM in samples by the measurement of this radiation. We will look at instruments that use the spontaneously emitted radiation as the basis for their assay. These devices are passive measurement systems because they do not interact with the sample being assayed. We will also discuss active assay instruments that interact with the sample material, stimulate it to emit neutrons and/or gamma rays, then perform measurements on the induced radiation.

This chapter will not deal with the complete details of the assay methods used by the instruments to be discussed. Where possible, reference has been made to other articles where this detail can be found. A very useful general discussion of NDA

techniques can be found in Ref. 1. Further detail on gamma-ray assay techniques is available in Ref. 2. In many cases, the instruments discussed below are described in operations manuals, which will be referenced where the instrument descriptions are given.

II. RADIATION DETECTORS

The most essential element in all NDA instruments described here is the radiation detector. Whether detecting gamma rays or neutrons, the detector employs basically the same principles; only engineering details vary, depending on the type of radiation being detected. The detector is essentially a sensitive volume of material (solid, liquid, or gas). When radiation passes through this material, it collides with the electrons in the molecules of the material, ionizing some of these molecules and producing electric charge that is free to move. This material is ultimately made part of an electronic circuit that is normally dormant when no radiation is incident on the detector volume. However, when radiation ionizes some of the detector material, the charge created begins to move in the circuit causing an electronic "pulse" that signals the presence of radiation in the detector volume. The amplitude of this pulse is designed to be proportional to the amount of charge produced in the detector volume, which in turn is proportional to the energy lost by the radiation in the detector.

The detector, therefore, delivers a number of pulses of varying magnitude when exposed to a radiation field. Attached to the detector and part of the electronic circuit is the pulse-processing electronics that reads these detector pulses and analyzes them. This equipment both counts the pulses and usually also sorts them according to their magnitude. The result of this analysis is a record of the amount of radiation detected and the relative intensities of the various radiation energies.

In gamma-ray detectors for NDA measurements, the sensitive volume of the detector is usually a solid crystal of germanium (denoted Ge or Ge(Li)) or sodium iodide (NaI). In Ge detectors, the charge produced is converted directly into an electronic pulse. In NaI detectors, the charge created by the radiation in the crystal produces light that is picked up by a photomultiplier (PM) tube attached to the crystal. The PM tube then produces electronic pulses in response to the light it detects.

In neutron detectors, gas-filled tubes usually serve as the sensitive volume, although scintillator crystals (such as NaI or some plastics) can be used as well. The incident neutrons interact with the materials in the sensitive volume and usually induce nuclear reactions of some kind. These reactions in turn cause ionization in the detector volume, which then produces electronic pulses as described before. The neutron-based instruments to be described here use gas-filled tubes, with gases such as ^{3}He, ^{4}He, or BF_3.

III. GAMMA-RAY INSTRUMENTS

A. Passive Measurement Devices--General

The simplest gamma-ray assay device uses the passive measurement approach and consists of a gamma-ray detector and pulse-processing electronics. A schematic arrangement of this type of measurement is shown in Figs. 1 and 2. This type of instrumentation is typically quite portable and therefore very useful in field inspection exercises and in-plant surveys of SNM inventory. A given quantity of a specific type of SNM emits a characteristic amount of gamma radiation. Thus, by counting the gamma radiation coming from the sample, a passive assay instrument can provide a direct measurement of the amount of SNM in the sample.

Each isotope of special nuclear material emits its own characteristic spectrum of gamma rays. A spectrum of uranium

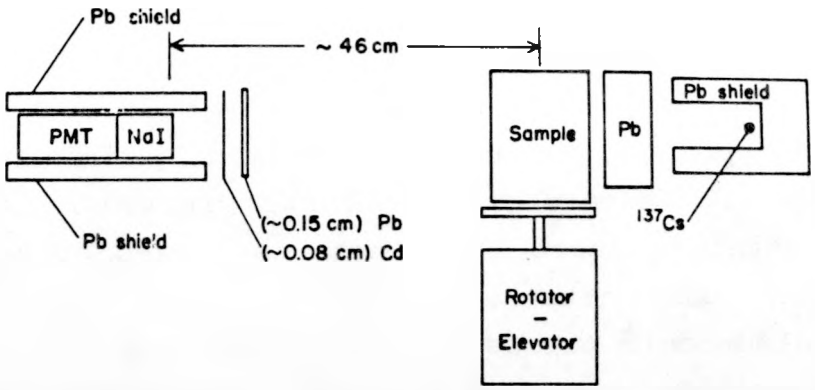


Fig. 1.
Illustration of a passive gamma-ray assay measurement apparatus. The sample is placed on a rotatable sample table, and the detector counts the gamma radiation coming from the sample. (Detector shown at left). Correction for the attenuation of the SNM gamma rays by the sample is determined by measurement of the gamma-ray intensity passing through the sample from an external radioactive source (in this case, Cs-137) shown at the right.

(enriched in ^{235}U) is shown in Figs. 3a (high resolution) and 3b (low resolution). In Fig. 4 are shown both types of spectra for a plutonium sample enriched to 93% in ^{239}Pu. Thus, a general sample of SNM emits not only a specific quantity of gamma rays but also gamma rays of specific identity (ie., energy), depending on which isotopes of uranium and plutonium are present in the sample. Thus, measurement of the relative intensities of these gamma rays (ie., the relative areas under the various gamma-ray peaks in the spectrum) gives a direct indication of the relative isotopic composition of the sample.

The actual SNM assay measurement with the type of passive system shown in Figs. 1 and 2 involves three components, 1) the raw measurements with the unknown sample, 2) applications of measured corrections to the raw data, and 3) measurements with known standards for purposes of calibration. The raw gamma-ray data reflect only a small sample of the total radiation emitted

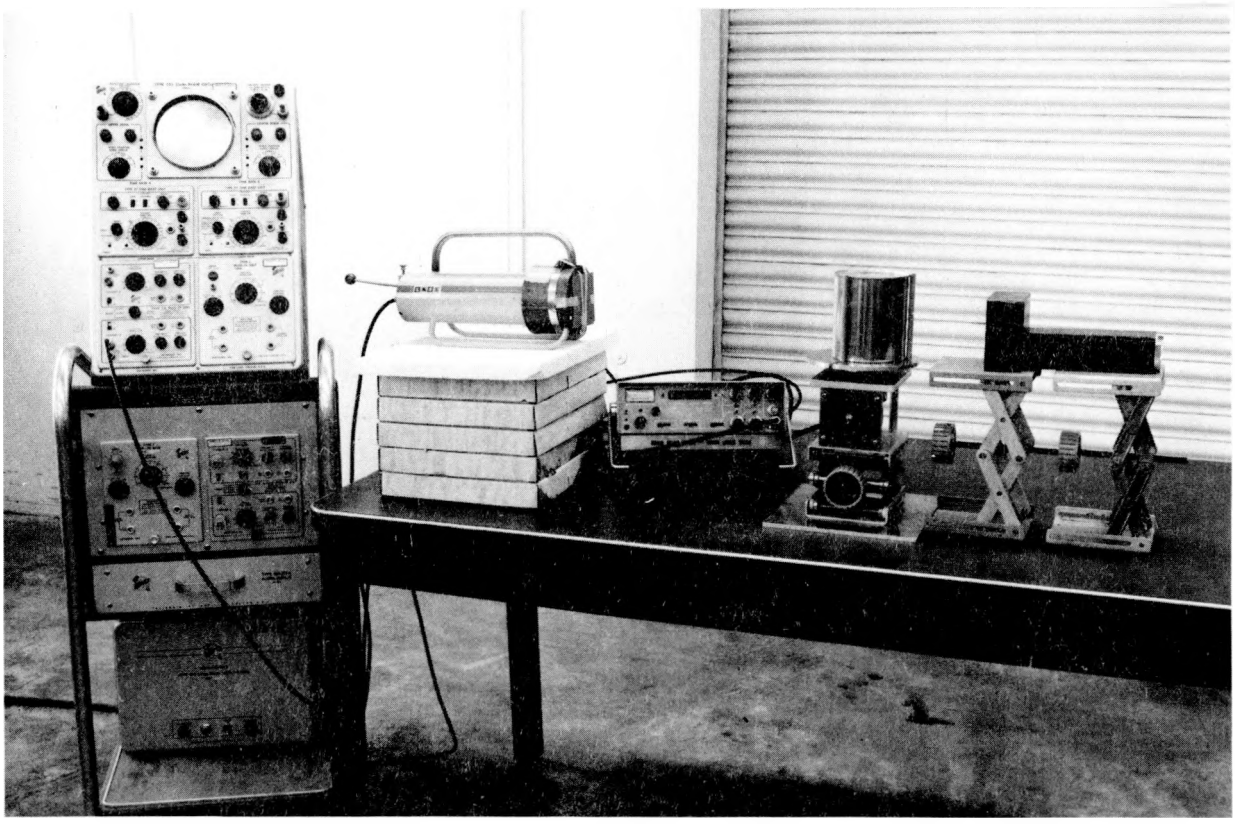


Fig. 2.
Photograph of a simple passive gamma-ray assay setup. At the left is the gamma-ray detector (a NaI crystal attached to a photomultiplier tube). The sample is shown mounted on the rotating table, and the transmission source is positioned to the right of the sample. (The arrangement is the same as that shown in Fig. 1.) The pulse-processing electronics is an Eberline SAM-2 shown on the table below the detector.

by the assayed material; some of the gamma rays produced are absorbed by the sample material itself and are never detected by the instrument. In addition, the gamma radiation that <u>does</u> emerge from the sample is emitted in all directions, and the gamma-ray detector intercepts only a small fraction of this total emission. Finally of those gamma rays that intercept the detector, only a fraction are actually detected and register as counts in the spectrum peaks used in the analysis of the assay data. Thus, the <u>measured</u> gamma-ray peak intensities must be

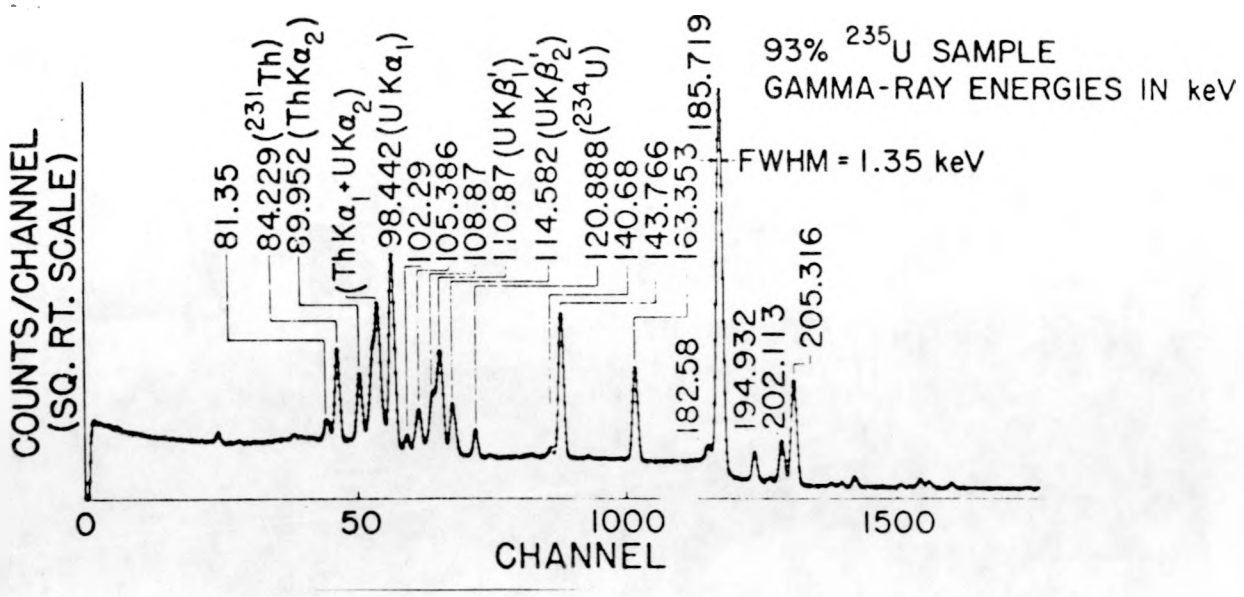


Fig. 3a.
Characteristic gamma-ray spectrum emitted by a sample of highly enriched uranium (93% U-235). The spectrum was taken with a high-resolution Ge(Li) gamma-ray detector. Of particular interest for passive gamma-ray assay of U-235 is the gamma ray that produces a peak in the spectrum at an energy of 185.7 keV.

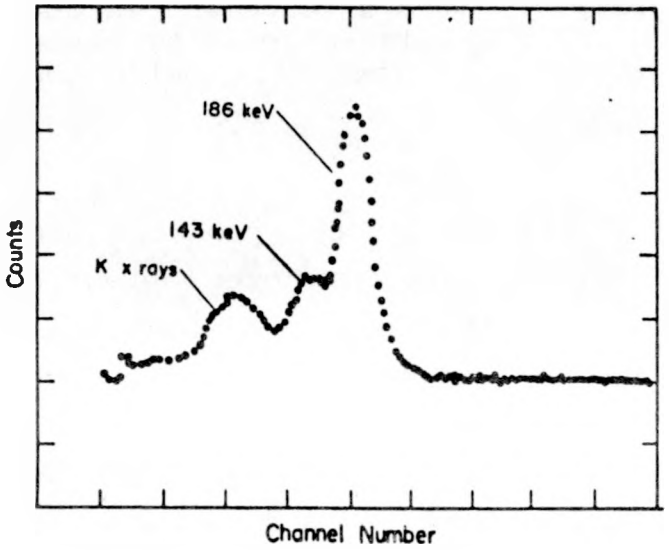


Fig. 3b.
Characteristic gamma-ray spectrum emitted by the same type of uranium sample as in Fig. 3a, but the spectrum has been taken with a lower resolution NaI detector. The spectrum in the region of the 185.7-keV U-235 gamma-ray peak is shown. This 185.7-keV peak is now less well resolved from the adjacent peaks, but it can still be counted reliably for assay of U-235.

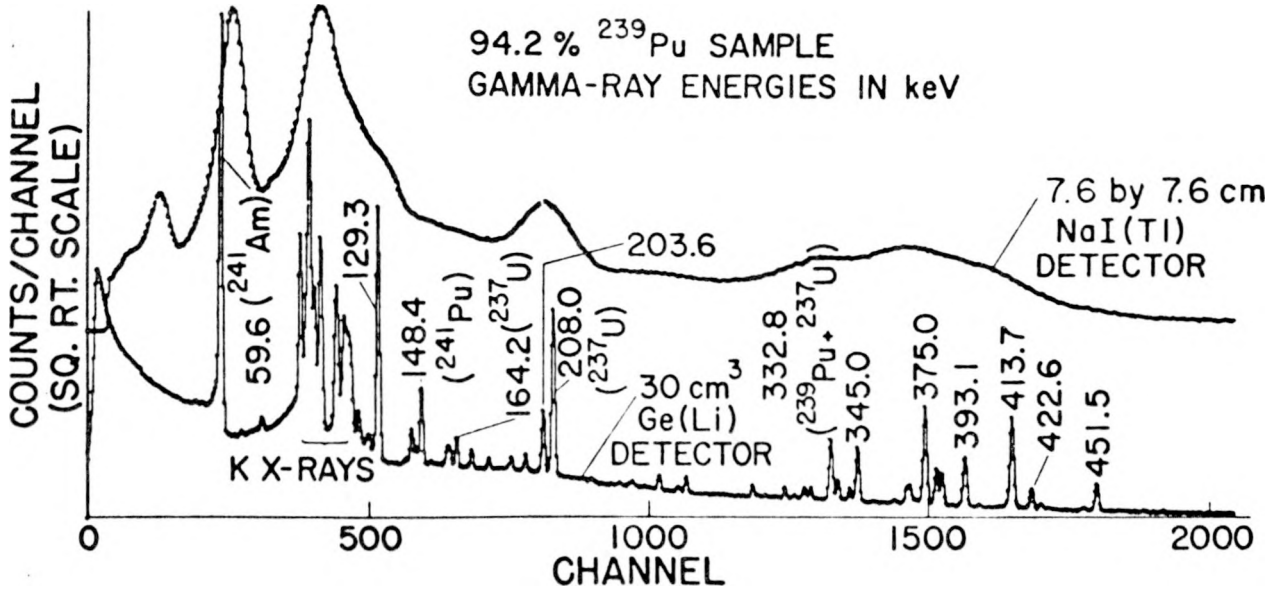


Fig. 4.
Characteristic gamma-ray spectrum emitted by a sample of plutonium, enriched to 93% in ^{239}Pu. Of particular interest for passive gamma-ray assay are the gamma-ray peaks at 129 keV and 414 keV, both of which are emitted by ^{239}Pu. Gamma rays from other plutonium isotopes are pointed out. The high-resolution spectrum was taken with a Ge(Li) detector, giving the detailed spectrum in the lower portion of the plot. The spectrum in the upper portion of the figure was taken over the same gamma-ray energy range but with a lower resolution NaI detector. With NaI detectors, the passive gamma-ray assay usually concentrates on the energy region containing the 414-keV peak.

corrected upwards to account for these sources of lost gamma-ray intensity in the detector.

To correct for the sample self-absorption, one usually measures the degree to which the sample material absorbs gamma rays from an external radioactive source. This absorption measurement can then be transformed into a correction factor for the effective absorption by the sample of gamma rays originating from within the sample. The correction factor thus obtained is then an empirical measurement of the absorption properties of that particular sample. If the correction factor

is measured for each sample (as it usually is), then variations in the sample composition are taken into account and do not contribute to unwanted fluctuations in the accuracy of the assay results.

To determine what fraction of emitted gamma rays actually reach the detector and are detected, one must determine two quantities, 1) the fraction of total gamma-ray intensity intercepted by the detector (the so-called "solid angle" subtended by the detector) and 2) the fraction of the gamma rays incident on the detector that are actually registered by the detector and its electronics (the so-called "detection efficiency"). Both items can be calculated, with some effort; but a much more efficient and reliable way to get the same information is to calibrate the measurement system using standard (known) amounts of SNM as samples. To accomplish this, the assayist places a standard sample in the measurement position in the same geometrical arrangement as will be used in the assay of unknown samples. (It is also best if the SNM standard is prepared in the same type of container and with the same type of other materials as will accompany the unknown samples. A more thorough discussion of the concept of measurement standards will be presented in the next section (Session 20) on National Systems of Measurement Standards.) Once the standard is ready to be measured, the complete assay procedure is performed, including the correction for self-absorption in the sample as described above. The assay results is then compared to the known value associated with the standard sample. The difference between the two result arises from the detector solid angle and detection efficiency described above, and a correction factor is applied to the measured data to bring it into agreement with the standard value. This correction factor is then applied to all subsequent measurements on unknown samples and thereby results in a calibrated measurement apparatus. This calibration will remain

valid as long as the measurement geometry and detector used remain the same. If any of these measurement features changes, then a new calibration will be necessary.

The passive assay measurement technique is very well suited for use in situations where portability of equipment is required. Gamma-ray assay of the SNM in piping or other relatively inaccessible locations can often be carried out quite accurately by placing a detector and transmission source in a convenient geometry on either side of the vessel containing the SNM. One arrangement for looking at SNM holdup in a duct is shown in Fig. 5.

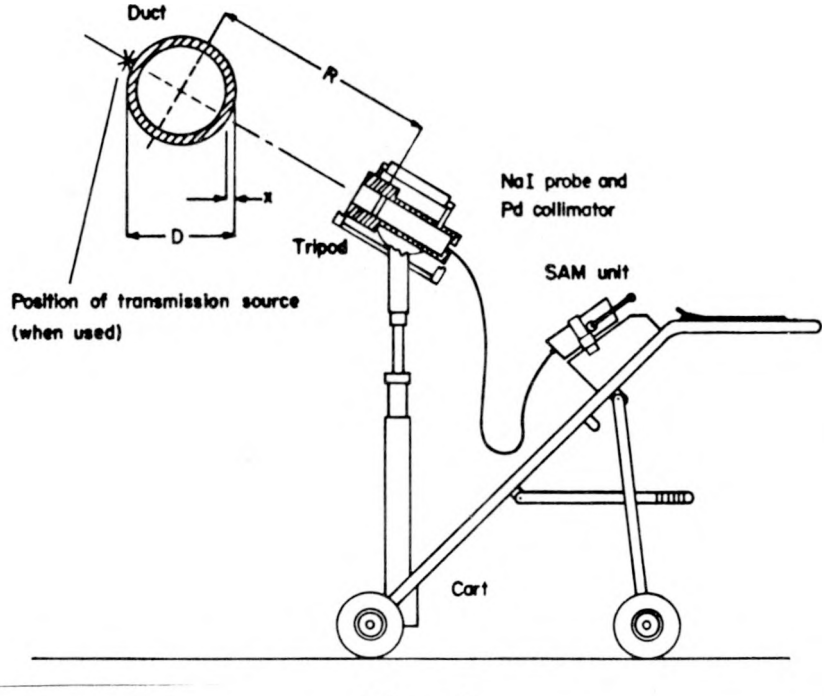


Fig. 5.
Schematic of a passive gamma-ray assay setup for a holdup measurement. The detector (mounted on a cart) looks at the duct of interest, on the opposite side of which is a transmission source. The pulse-processing electronics is also mounted on the cart. (Compare with Figs. 1 and 2.)

B. Uranium Enrichment Measurements

The principles of the determination of the degree of enrichment in ^{235}U of a sample of uranium are very similar to those employed in the general passive gamma-ray assays described above. The mathematical details of the enrichment measurement are outlined in Ref. 1, and only the general idea will be described here. The geometry of the measurement is depicted in Fig. 6. The detector views a fixed area of uranium through an appropriate collimator and counts the 185-keV gamma ray from ^{235}U (see Fig. 3). If the sample material is infinitely thick with regard to the 186-keV gamma ray (i.e., if essentially no 186-keV gamma rays can pass through the entire sample), then the count rate of the 185-keV gamma ray will be proportional to the per cent enrichment of the sample in ^{235}U. The instrumentation is calibrated by first measuring some samples with known enrichment (i.e., enrichment standards) and obtaining the appropriate correction factors, which are then used to correct the measured count rate to ^{235}U enrichment. This measurement is a passive assay procedure and also uses highly portable equipment, such as the NaI detector and SAM-2 electronics indicated in Fig. 5.

C. Spent Fuel Assay Measurements

The techniques for spent fuel assay will be covered more thoroughly in Session 21 to follow. However, in this chapter

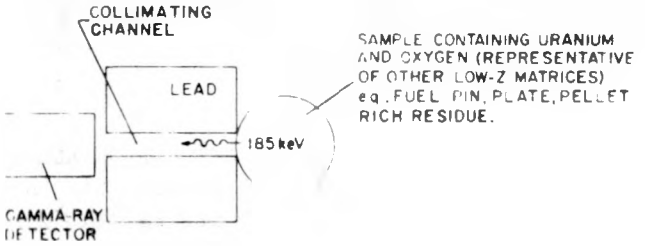


Fig. 6.
Schematic illustration of an enrichment measurement. The detector and collimator are shown, but the pulse-processing electronics is not.

we simply show one application of passive gamma-ray counting in this area. In Fig. 7 below is shown a Ge(Li) detector viewing a fuel element in a spent fuel storage pond. The collimator used assures that the detector views primarily the fuel element of interest, with minimum interference from other elements in the same storage area. The gamma-ray spectra acquired by the detector will be dominated by the radiation from the many fission products in the fuel element. However, this radiation reveals very nicely the irradiation history of the fuel element

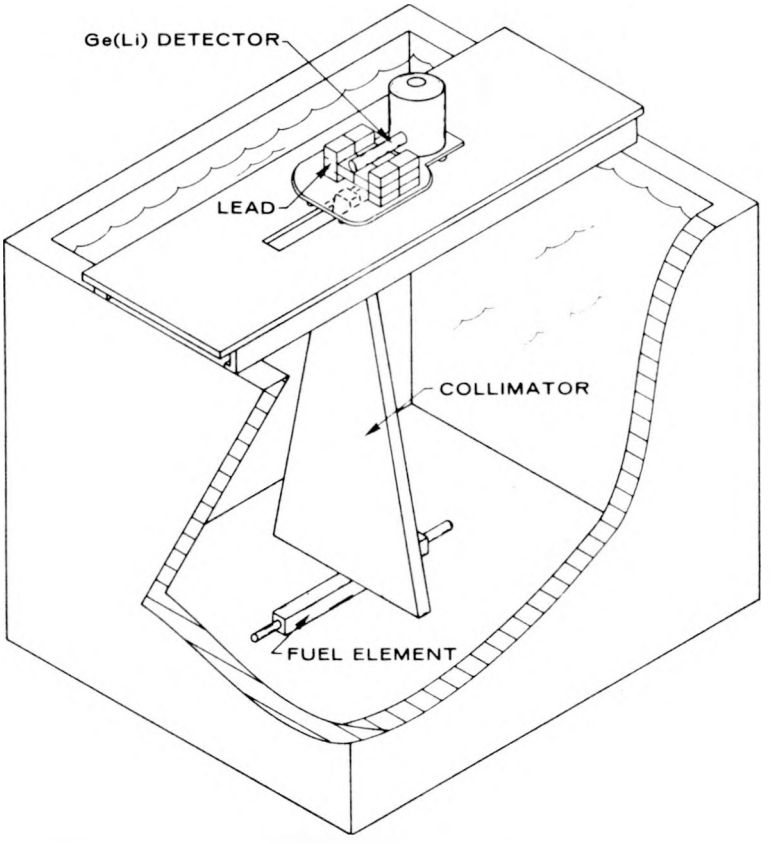


Fig. 7.
Schematic of a passive gamma-ray assay of spent fuel elements in storage. The Ge(Li) detector is placed on top of the storage pond and heavily shielded with lead. It then views a restricted region of the storage area through a collimator and acquires gamma-ray spectra for analysis.

and can be used to infer fuel element burnup and fissile content. For additional detail, see Refs. 3-5.

D. Passive Gamma-Ray Measurement Devices--The Segmented Gamma Scan

In the passive SNM assay technique described in the previous sections, the assumption is made that the sample container is uniformly filled with sample material. As illustrated in Fig. 8, variations in the degree to which the sample container is filled will effectively cause variations in the counting geometry. This means that the instrument calibration will not be valid for all possible cases, and so the assays can exhibit greater inaccuracies than desired.

A passive assay technique has been developed that removes this problem to a great extent, the segmented gamma scan. The basic idea is to divide the sample into a series of horizontal segments and assay each segment (one at a time) using the

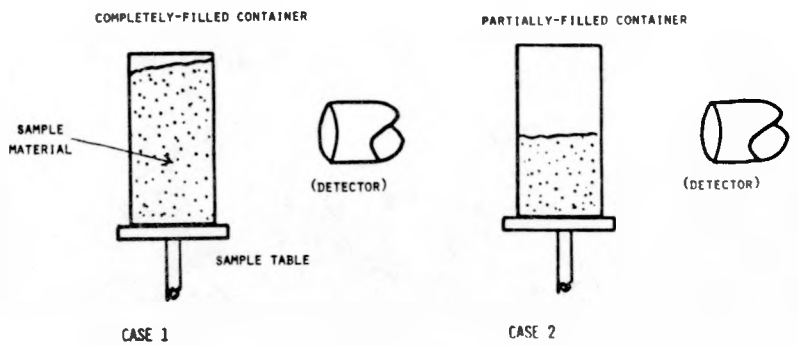


Fig. 8.
Illustration of the variation in counting geometry caused by differences in the amount of filling of the sample containers to be assayed. These variations mean that the instrument calibration (which is based on the material arrangement in the calibration standards) will not be valid for all of the samples assayed, and inaccuracies will be present in the final results.

conventional passive gamma-ray assay technique described above, with self-absorption correction determined for each segment. Then when all segments have been measured, the results are summed to give the total assay result for the sample. The sequential assay of sample segments is illustrated in Fig. 9.

This assay technique is especially valuable in assaying samples in which one does not know the distribution of the SNM and other matrix material (i.e., everything else) in the container. For example, containers of low-level waste (paper, gloves, coveralls, etc.) may contain traces of SNM on some items and not others. The segmented gamma scan procedure will assay the SNM properly, regardless of where in the waste container it is located on any given measurement.

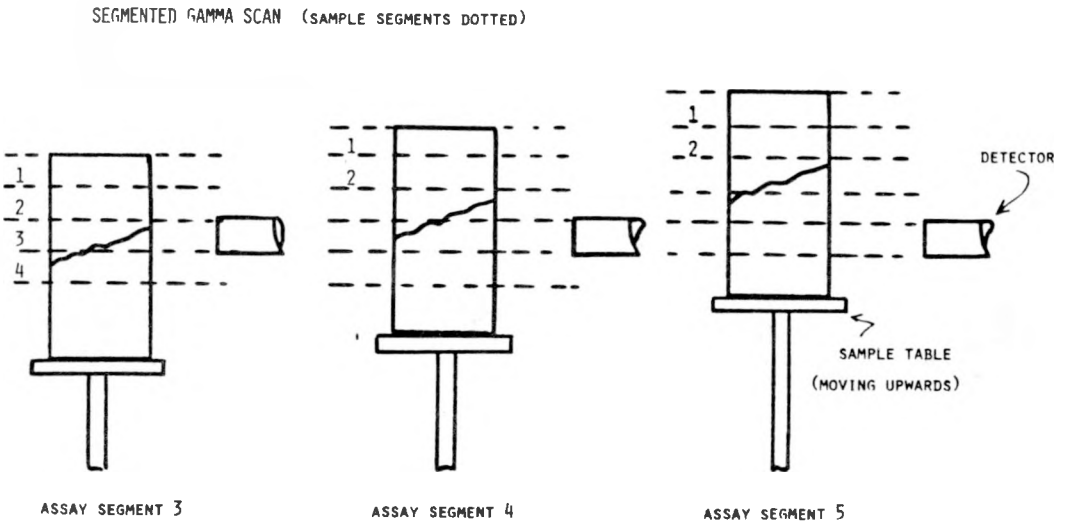


Fig. 9.
Schematic of the segmented gamma-ray assay of a sample of SNM. The detector and transmission source are fixed, and the sample is moved vertically, exposing different segments of the sample to the assay instrument. When all segments have been assayed, the individual results are summed to give the SNM content of the total sample. Individual segment data give the SNM profile within the sample.

In Fig. 10 we show a photograph of a segmented gamma scanner (SGS) that is designed for assays of small samples. The sample table is located in front of the operator, and it both rotates and moves vertically. The Ge(Li) gamma-ray detector sits to

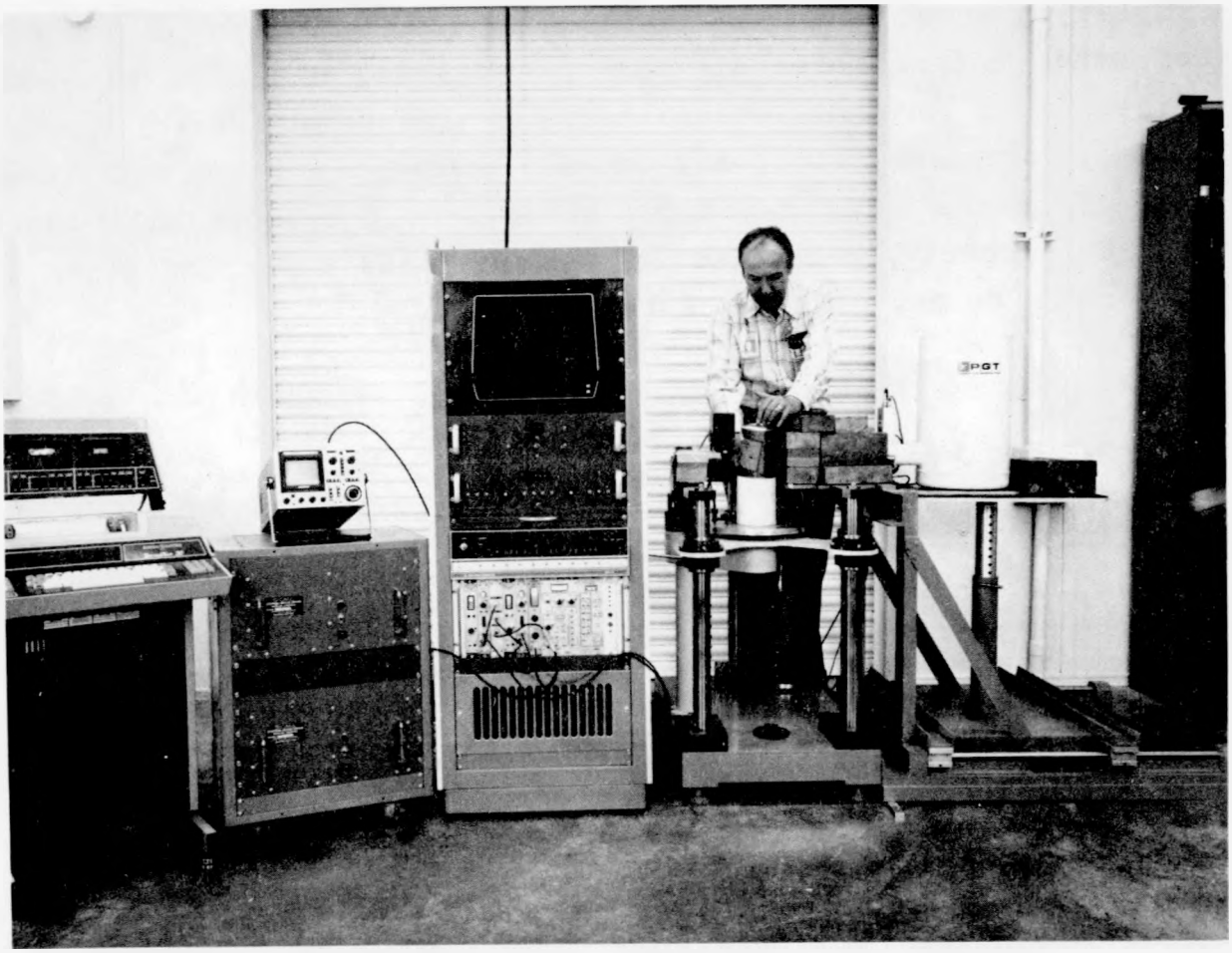


Fig. 10.
Photo of the segmented gamma scanner (SGS) for the assay of the SNM in small samples. The sample table is in front of the operator, and the detector is shown to the right of the sample table. The pulse-processing electronics is to the left of the operator, and at the far left of the picture is the typewriter output unit on which the final assay results are printed and at which the initial assay request is generated.

the right of the sample, behind a collimator and lead shielding. The transmission source (for the self-absorption correction) is to the left of the sample.

In a typical assay sequence with the SGS, the table is positioned so the top of the sample container is just below the detector axis, and the assay begins. The sample is rotated continually (to average out any sample nonuniformities within a given segment), and the vertical position of the sample table is varied in steps. For each table height, the passive assay is performed, followed by movement of the table upwards to the next vertical position. This procedure continues until the entire vertical dimension of the sample container has been scanned. The individual segment assays are then tabulated and summed to give the total SNM result as well as the SNM vertical profile in the sample. The instrument is computer controlled, so data acquisition, data analysis, and mangement of the hardware are all done automatically. In Fig. 11 is shown a closeup of the small-sample scan table, detector, shielding, and

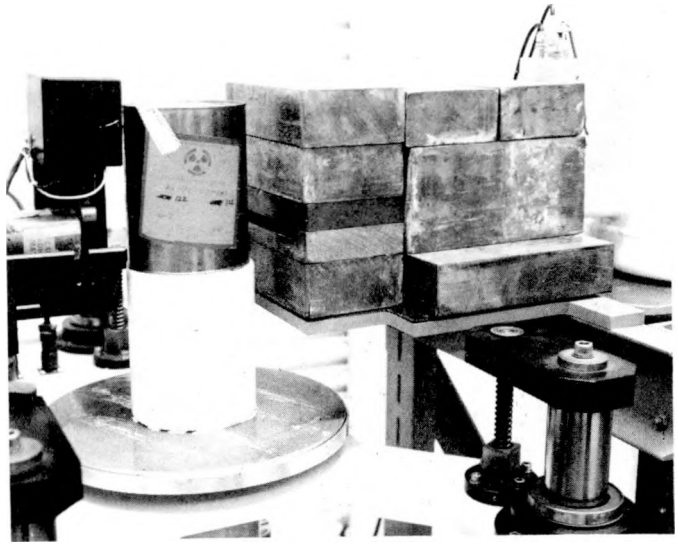


Fig. 11.
Closeup of the small-sample scan table. A sample is shown in position such that one of its lower segments is being assayed. The transmission source is to the left of the sample and is presently shielded from the detector by a tungsten "shutter" suspended in front of it. The detector views the sample and source from the right of the picture and is shielded from background radiation by the lead shown in the picture.

transmission source. Extensive detailed description of this instrument is available in its operation manual.[6]

In Fig. 12 is shown a similar SGS instrument for use with large containers such as the 55-gal drum shown on the sample table. In this figure, the detector is on the left and the transmission source is on the right. The pulse-processing and control electronics are at the right of the picture, and the printing typewriter terminal is to the left of the electronics rack.

E. Active Gamma-Ray Assay Devices

If one employs an assay technique in which the sample is induced to emit gamma rays, then the technique is regarded as an active assay technique. All materials can be stimulated to

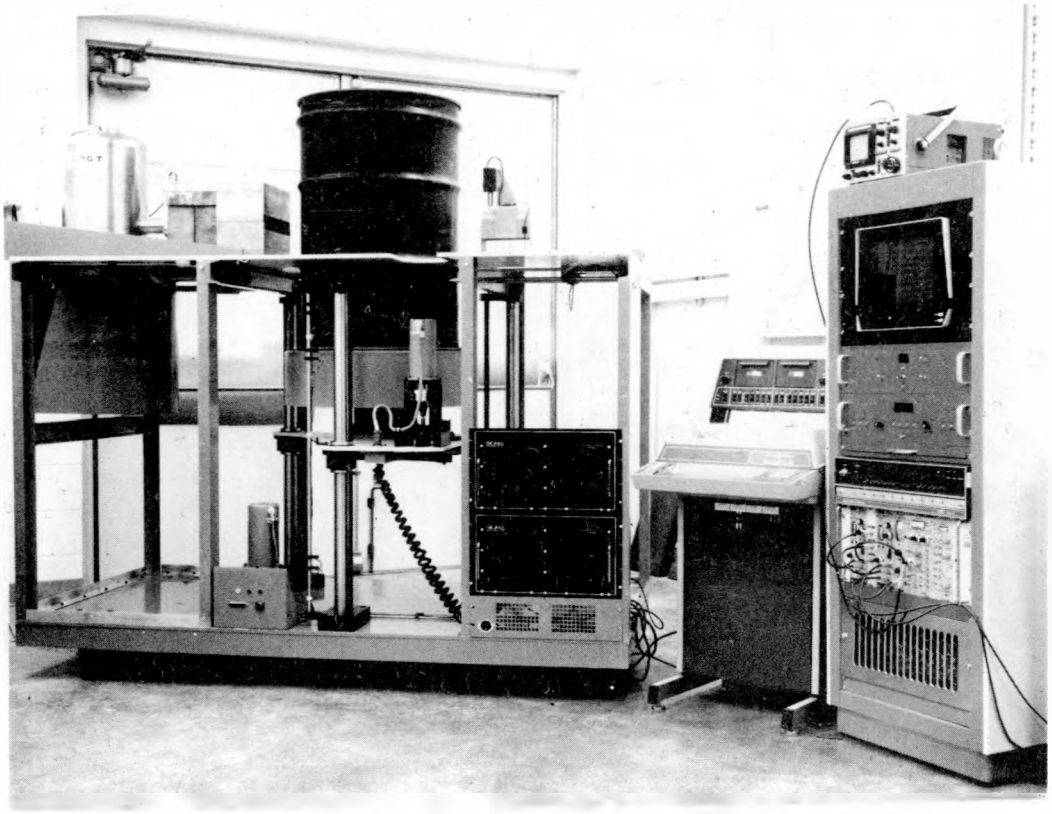


Fig. 12.
The SGS for large samples.

give off x rays (which are relatively low-energy gamma radiation originating from the atomic electron shells rather than the atomic nucleus), and the energies of these x rays are characteristic of the chemical element that has been excited. Uranium and plutonium x rays occur in the energy range up to approximately 120 keV and so are relatively easy to detect with conventional gamma-ray detectors. One can stimulate samples of SNM to emit x rays by irradiating them with gamma rays whose energies are slightly greater than the x-ray energies themselves. A schematic of a simple assay device that employs this principle is shown in Fig. 13 below. The technique is called x-ray resonance fluorescence (XRF). The sample is placed in the path of exciting gamma radiation (for example, the radiation could be the 122-keV gamma ray from ^{57}Co). This radiation interacts with the SNM sample and induces some atoms of the sample to emit characteristic uranium and/or plutonium x rays. The detector picks up the x rays and thereby registers the presence of SNM in the sample. The x-ray intensity must be corrected for the self-absorption of the sample, just as in the passive assays described above. This is done with a transmission source as before; the transmission measurement is made when the shielding

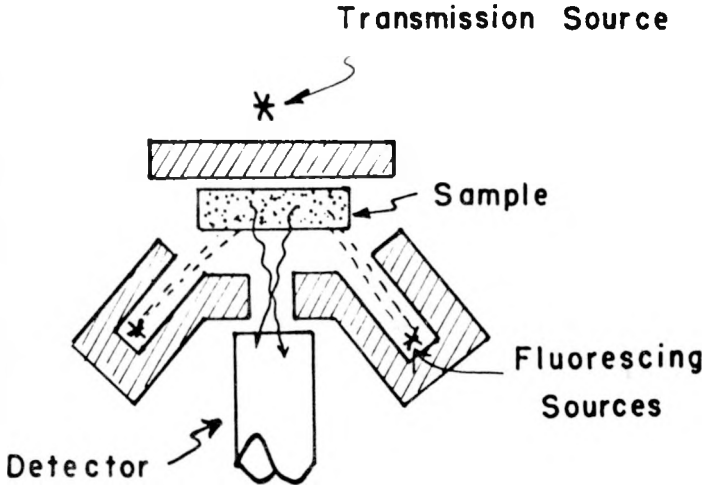


Fig. 13.
A schematic of an XRF assay apparatus. The exciting sources induce x-ray emission from the SNM in the sample, and the detector registers the x rays and the presence of SNM in the sample. Sample self-absorption is corrected for with a transmission source in much the same manner as in the passive assays.

between the source and the detector is temporarily removed. The absorption-corrected x-ray intensity is then proportional to the amount of SNM in the sample. The constants that allow the conversion of the measured x-ray intensity to amount of SNM are determined by calibration with known standard SNM samples. This technique can be applied to assays with highly portable equipment, and the high resolution possible with Ge detectors permits the simultaneous assay of several chemical elements in a single measurement.

IV. NEUTRON-BASED INSTRUMENTS

Neutron counting has several advantages over gamma-ray assay techniques. Fast neutrons, in the range from 10 KeV to several MeV, have relatively high penetrability in high-atomic-number nuclear materials. In such samples, the attenuation of gamma rays is very critical, and, if the sample is large enough, gamma-ray assay measurements become impossible. In such cases, neutron measurements are still possible and can often be done to accuracies of better than one per cent. Gamma-ray measurements are also difficult in the presence of moderate neutron backgrounds because of the radiation damage produced in Ge(Li) and NaI crystals by the neutron flux. Active neutron assays (which involve neutron irradiation of the sample followed by counting of prompt fission or delayed neutrons from the activated SNM) often permit the fissile content of a sample to be measured directly; by contrast, in a gamma-ray assay, the fissile content may sometimes only be inferred, such as in a spent fuel measurement. In addition, as we have seen above, gamma-ray assays often require sophisticated corrections to the raw data for attenuation and also decay of the gamma ray being measured. On the other hand, no such corrections are required for neutron assays because of the high penetrability of the

neutron radiation and the long half-lives of the neutron-emitting SNM isotopes. Neutron measurements generally employ integral counting (i.e., counting of all neutrons above a certain predetermined energy), and the measurement equipment is relatively simple and much less expensive than high-resolution gamma-ray equipment. In addition, neutron systems are generally easier to maintain and much less susceptible to damage through misuse than the more delicate gamma-ray systems.

Assay measurements based upon neutron counting also have a number of disadvantages. The simple equipment used in integral counting does not allow a determination of the particular isotope that is emitting the neutrons. This is because the neutron spectra themselves do not carry such detailed energy information and neutron detector systems have intrinsically poor energy resolution. Problems are also encountered in neutron counting by the presence of moderators and neutron poisons (i.e., substances that are especially effective at absorbing neutrons) in the samples to be measured. Often these samples can be handled with appropriate calibration standards coupled with active interrogation using fast neutrons. However, the active neutron assay systems are heavily shielded, bulky, and not easily portable.

In short, gamma-ray assays when they are possible generally yield more detailed information about the sample, such as isotopic composition, than do neutron techniques. However, there exists an abundance of assay tasks that are not possible with gamma-ray methods but can still be accurately achieved using neutron techniques. In addition, some applications are best accomplished with a combination of both gamma-ray and neutron measurements. In this section, we will present examples of both passive and active neutron assay systems. In addition, both singles and coincidence neutron counting will be illustrated.

A. Passive Measurement Devices--Singles Counting

The basic principle in passive singles neutron assay is to count the number of neutrons being produced by the sample and to determine from that number the amount of SNM in the sample. The most convenient portable passive neutron assay instrument

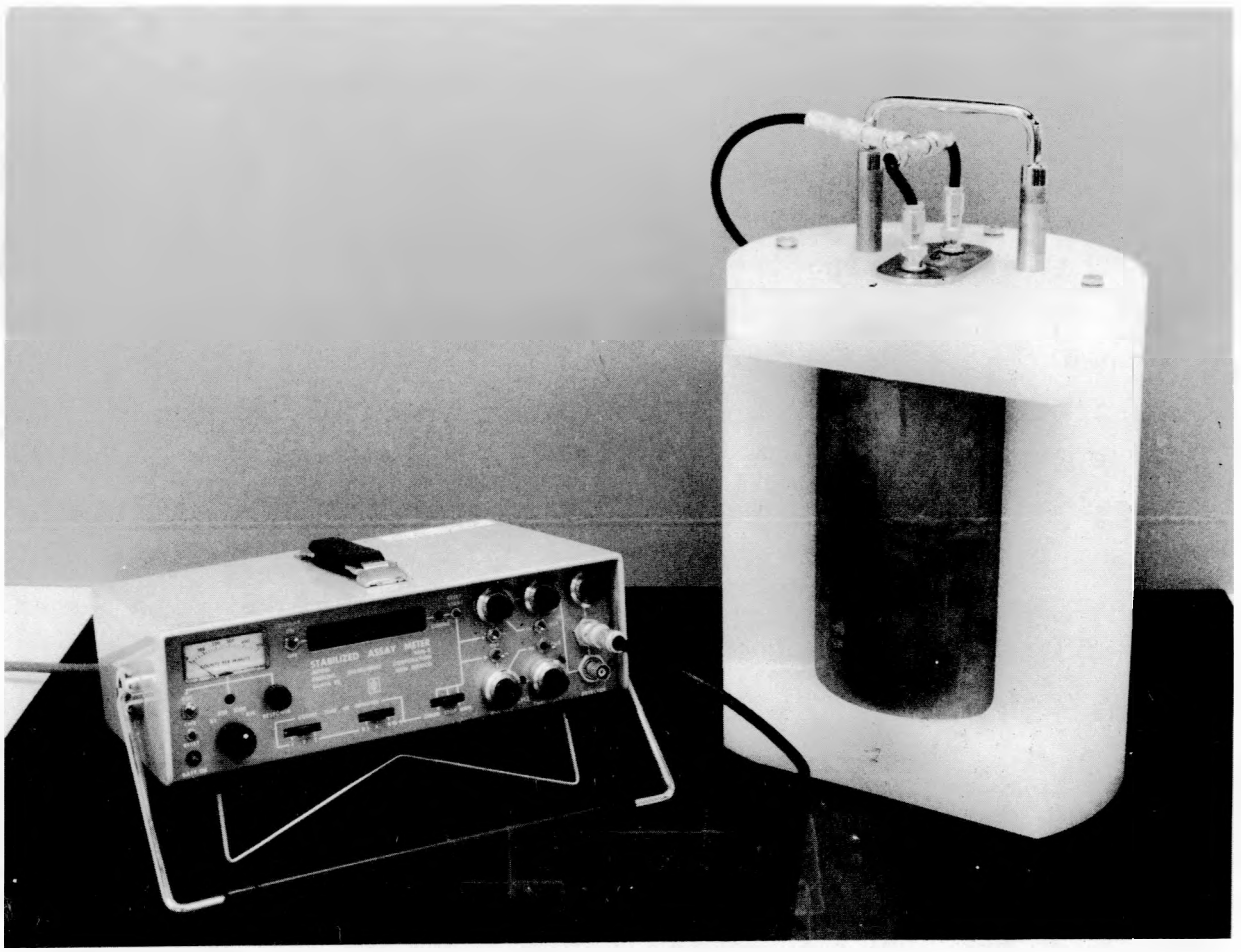


Fig. 14.
The Shielded Neutron Assay Probe (SNAP) system. The neutron detectors are tubes of ^{3}He (gas-filled) proportional counters of the type described in the earlier section on radiation detectors. The pulse-processing electronics consists of an Eberline SAM-2, which receives the detector pulses and counts them if they exceed a predetermined energy threshold. For further description of the instrument, see text.

developed for this purpose is the Shielded Neutron Assay Probe (SNAP) detector. The SNAP system operates as an integral counter, collecting all neutron events above a certain threshold energy setting, which is selected in the pulse-processing electronics. A photo of the SNAP and its electronics package (an Eberline SAM-2) is shown in Fig. 14. The SNAP detector consists of a cylindrical polyethylene core (127 mm diameter by 305 mm length), which contains two ^{3}He-filled tubes as neutron detectors. Small samples, such as light water reactor (LWR) fuel rods, can be counted internally by placing them in the 19-mm-diameter hole through the center of the core. To achieve directional sensitivity and good background discrimination, a removable 240^{o} annular polyethylene shield (241 cm diameter) surrounds the core. A simple turntable mechanism is usually added to rotate samples in front of the detector during data collection. The detector pulses are processed and sorted by an Eberline SAM-2 assay meter.

To count singles neutrons, as is done with a SNAP system, one must thoroughly understand all the possible origins of neutrons in the region of the sample material. These sources are listed below:

1. Alpha-particle-induced neutrons. The isotopes of uranium and plutonium emit alpha particles as one of their modes of decay. When these alpha particles interact with the matrix materials found in nuclear fuels (N, O, F, Mg, C, Si, etc.), neutrons are produced. Thus, the neutron yield from a given sample of SNM depends not only on the amount of SNM in the sample but also on the chemical composition of the sample material. As a result, singles neutron count rates can only be understood clearly if the sample materials are well characterized before assay. Neutrons produced from alpha-particle interactions are produced randomly, or one at a time.

2. Prompt fission neutrons. The even isotopes of uranium and plutonium decay spontaneously by fission, releasing

several neutrons at a time. Usually, a given decay by fission produces at least two coincident neutrons. This coincidence "signature" is unique to the fission process and so serves as an unambiguous indication of the presence of fissile material in a sample.

3. Induced fission neutrons (multiplication). The fissile isotopes of uranium ($^{233,235}U$) and plutonium (mainly ^{239}Pu) can be induced to fission if they capture a neutron. In large samples of SNM, the neutrons produced by the spontaneous fission of the even SNM isotopes can induce fissions in the fissile isotopes. As a result, the number of fission neutrons per gram of SNM will appear to increase because of the enhancement of neutron production by this multiplication process. Details of this multiplication process and the corrections for it may be found in Ref. 7.

4. Delayed fission neutrons. Some of the nuclear fragments left behind after a fission are unstable enough to decay by neutron emission. As these nuclei do so, they contribute to a neutron flux for which the neutrons are again produced one at a time (randomly) and at times long after the fission has occurred (sometimes many seconds later).

In using the SNAP system for passive assay, one relies on the principle that the observed singles neutron count rate will be proportional to the sum of spontaneous fission neutrons (generally ^{240}Pu in plutonium samples) and neutrons from alpha-particle-induced reactions. This detector is used primarily for plutonium assay, for which the spontaneous fission decay rates are significant. These rates in uranium are very low, making passive neutron assay of uranium impractical in general. For high-purity plutonium metal, the alpha-particle-induced yield is usually negligible, since very little matrix material is present. However, a correction factor for neutron multiplication is required for large samples (greater than 1 kg). Knowledge of the isotopic composition of the sample (from other data) will allow the determination of the various plutonium isotopes. For oxides or other nonmetallic compounds, the sample must be well characterized with respect to the light

element impurities. This is because such impurities will contribute significantly to the neutron yield through the alpha-induced production mechanisms. Thus the nature of the impurities must be well known to generate accurate corrections for these extra neutrons.

This instrument is useful for measuring neutron yields when portability is important. Because of the integral character of the counting and the fact that singles neutrons are counted (rather than concentration on coincidence events), one must be careful to assay well-characterized samples with this instrument.

B. Passive Measurement Devices--Coincidence Counting

In coincidence neutron assay instruments, emphasis is placed by the detection system on the correlated prompt fission neutrons produced by the fissioning isotopes of the SNM. Once again, it should be emphasized that the spontaneous fission rate in the plutonium isotopes is adequate for use in passive coincidence neutron counters. However, the uranium isotopes decay by fission much more slowly and as a result are less appropriate for passive coincidence neutron assay.

A highly portable neutron coincidence counter developed for field assays of plutonium by coincidence measurements is shown below in Fig. 15. This neutron coincidence counter assays plutonium-bearing material by detecting spontaneous fission neutrons from the plutonium, in the presence of large random neutron backgrounds, originating primarily from alpha-particle-induced interactions. The pulse-processing electronics for this instrument separates the time-correlated (i.e., coincident) neutrons from the randomly generated (singles) neutrons. The neutron detectors are positioned in the system so as to surround the sample, thereby maximizing the detection efficiency (which is important when one must detect two neutrons in order to have

Fig. 15.
The portable neutron coincidence counter. On the right is the cylindrical cavity, which is surrounded by neutron detectors and in which the sample are placed. At the left is the processing electronics, consisting of coincidence circuitry (center) and calculator/printer (left).

a valid event). The arrangement of detectors and shielding is shown in Fig. 16.

Since the primary source of coincident fission neutrons is the isotope ^{240}Pu, this instrument measures effective ^{240}Pu in metal, oxide, or mixed oxides. The presence of other matrix material will cause the production of alpha-particle-induced neutrons; however, these neutrons will not be time correlated and will be essentially rejected by the pulse-processing

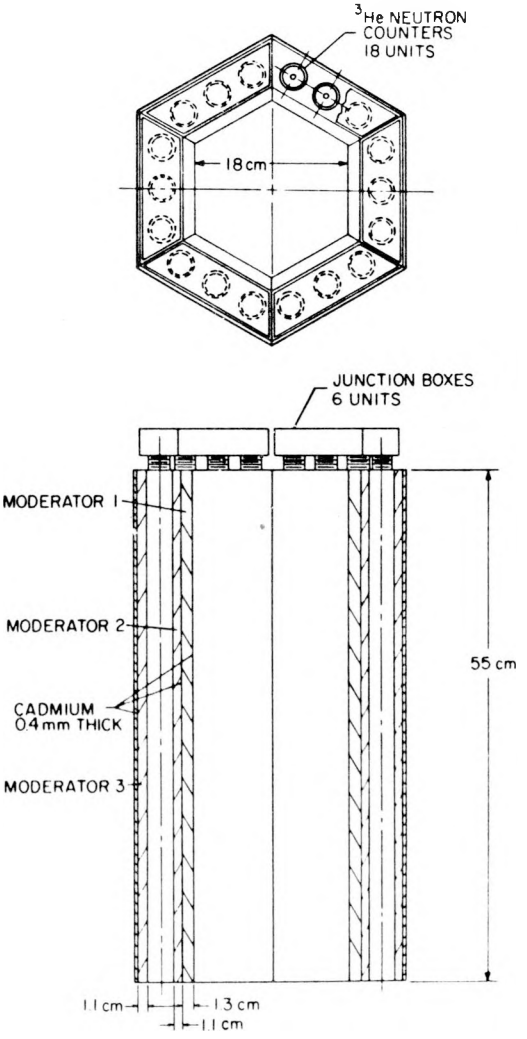


Fig. 16.
Schematic of the neutron coincidence counter, showing the arrangement of the ^{3}He-filled neutron detectors and the shielding. The shielding (moderator) is used to slow down the neutrons produced by the SNM so that the detectors (which have better detection efficiency at lower neutron energies) will register more useful events in a given period of time.

electronics, which is looking for coincident neutron events. This instrument is also relatively insensitive to gamma-ray background. If the plutonium isotopic composition of the sample is known from other sources (e.g., a high-resolution gamma-ray measurement), then the effective ^{240}Pu content can be translated into a complete plutonium assay of the sample. Under these conditions, accurate results are possible over a wide range of loadings (from 1 to 1000 grams of plutonium @ 20% ^{240}Pu). For further details on this instrument, see its operation manual (Ref. 8).

C. Active Measurement Devices--Coincidence Counting

The passive well coincidence counter described above has proven to be a very useful NDA instrument for plutonium assay. IAEA inspectors have found the portable high-level neutron coincidence counter (HLNCC) unit particularly useful for field applications. However, the instrument has not been applicable to the assay of ^{233}U or ^{235}U because of their extremely low spontaneous fission yields. To make this type of instrument applicable to these uranium isotopes, one must induce the fissions in the sample material with an external neutron source and then count the coincident neutrons from the induced fissions. An instrument now exists that uses this principle of active neutron interrogation of the sample followed by coincident neutron counts. This instrument is called the active well coincidence counter (AWCC) and is shown in Figs. 17-21. The term "active well" refers to the fact that the sample well contains a neutron source that actively stimulates the fissions in the sample material. The source used is composed of ^{241}Am (an alpha-particle-emitting isotope) and lithium. The alpha

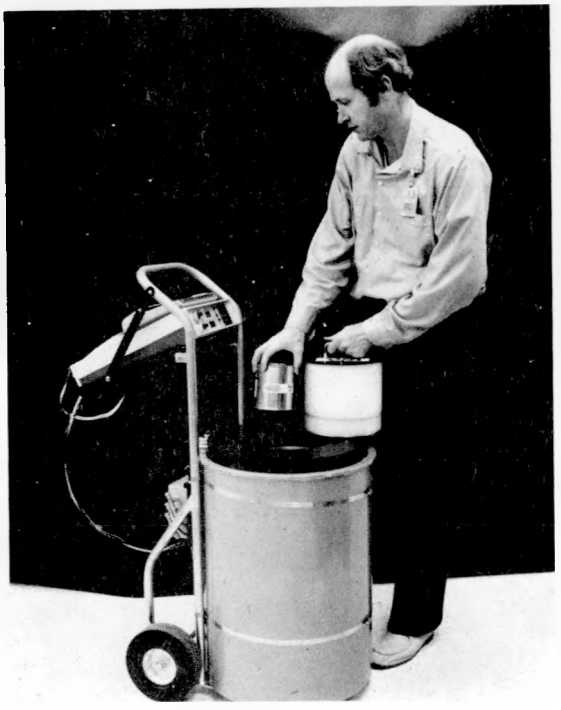


Fig. 17.
Complete view of the active well coincidence counter (AWCC). The assayist is shown lowering a sample into the well of the counter, after which he will replace the plug he is holding in his left hand. The counter is mounted on a cart to facilitate portability, and the pulse-processing electronics (similar to the passive coincidence counter electronics) is also attached to the cart for convenient readout of the raw data.

particles from the americium interact with the lithium and produce neutrons. These neutrons are in turn captured by the fissile isotopes of uranium and the fissions take place.

The AWCC can also be used in a passive mode to assay plutonium samples. All that is necessary to achieve this is to remove the AmLi neutron sources. Since the electronics package for the AWCC is essentially the same as for the passive coincidence counter, no other changes are necessary to perform plutonium assays with this device.

The AWCC can also be used to assay extended samples such as fuel rods. This is accomplished by setting the counter on its side as shown in Fig. 20. A bottom plug is also removed from the counter, which then allows the extended sample to pass through the well from top to bottom. For further details on this instrument, see Ref. 9.

Fig. 18.
Front view of the AWCC. The coincidence electronics and calculator/printout unit are at the top of the figure; the camera is looking down toward the interior of the sample well. A sample container typical of those assayed in this instrument is on the rim of the well. Within the annulus around the sample well are a number of ^{3}He detectors arranged in a manner similar to that used in the passive coincidence counter. See also Fig. 19.

Fig. 19.
Photograph of the ^{3}He detector ring in the AWCC. The detector ring has been partially removed from the polyethylene shielding in which it is housed so it would be more easily seen in the picture.

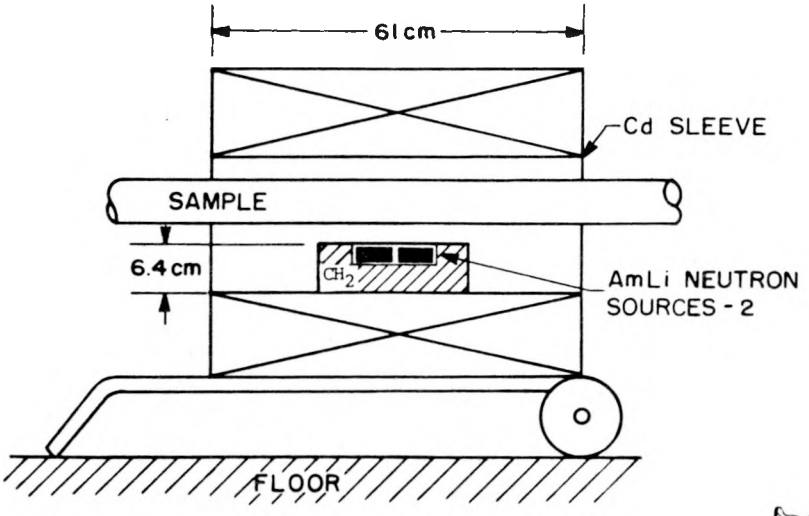


Fig. 20.
Schematic diagram of the AWCC showing side operation to accommodate long samples such as fuel rods and trays.

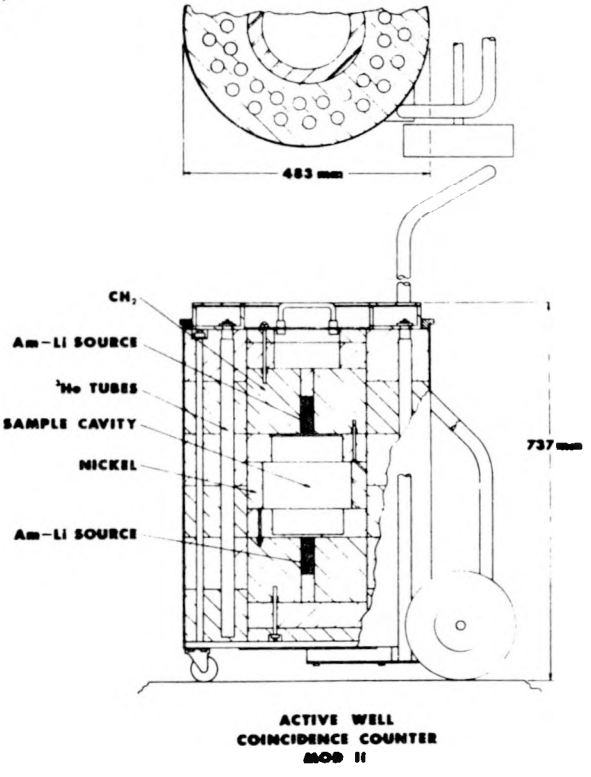


Fig. 21.
Schematic of AWCC, showing placement of ^{3}He tubes and AmLi neutron sources.

D. Active Measurement Devices--Singles Counting

One very popular nondestructive assay technique uses the delayed neutrons produced following induced fissions in an SNM sample. The instrument in which this technique is applied is called a "^{252}Cf Shuffler" and is described in greater detail in Ref. 10.

A ^{252}Cf shuffler measures fissile isotopes (^{233}U, ^{235}U, ^{239}Pu, and ^{241}Pu) by an active NDA technique employing neutron interrogation followed by delayed neutron counting. The neutrons from a spontaneously fissioning ^{252}Cf source are used to irradiate the item being assayed. The neutrons from the ^{252}Cf source, after some moderation, initiate fissions in the sample. A fraction of these fissions result in fission fragments that are neutron unstable and emit "delayed neutrons" with decay half-lives between 0.23 and 55 seconds. The delayed neutrons are detected once the ^{252}Cf source has been withdrawn to a shielded storage position. The irradiation of the sample is performed inside a high-efficiency neutron well counter used to count the delayed neutrons. For samples containing plutonium, this well counter can also be used in the passive coincidence mode to count the spontaneous fission neutrons primarily from ^{240}Pu. The neutron counter consists of a polyethylene matrix filled with 25 ^{3}He tubes.

A motor is used to transfer cyclically the ^{252}Cf source between the shielded storage position and the irradiation position. The source is coupled to the motor by a cable, and the transfer (typically between 1.2 and 1.5 m) is accomplished in approximately 1/2 second. The irradiation and delayed neutron counting times are about 10 seconds each. The active assay cycle (transfer source to irradiation position, irradiate sample, transfer source to storage position, count delayed neutrons) is repeated until the desired assay precision is achieved or a predetermined number of cycles has been completed.

Shown in Fig. 22 is a schematic of an existing ^{252}Cf shuffler system currently under test and evaluation at the Savannah River Plant (SRP). This instrument has been designed to measure the ^{235}U content of recycle scrap and waste at the SRP (SRP) fuel fabrication facility. Shufflers have also been developed that are intended for the assay of fuel rods,

S.R.P. SHUFFLER

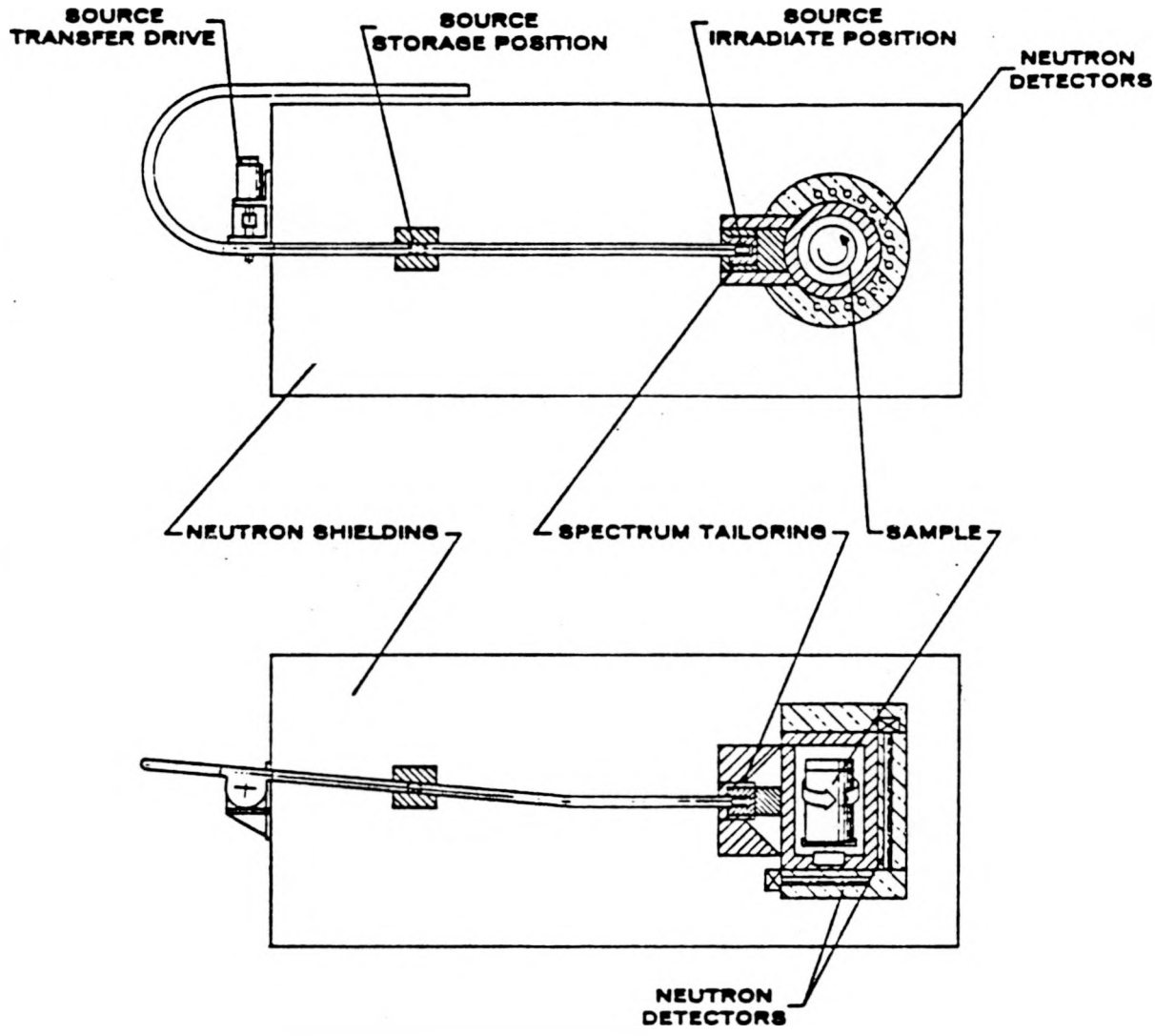


Fig. 22.
A ^{252}Cf shuffler system currently in operation at the Savannah River Plant (SRP). The californium source is "shuttled" in and out of the irradiation position along the U-shaped channel evident in the top figure. The sample area is at the right and contains a rotatable sample table and an array of ^{3}He neutron detectors. The approximate size of the main body of the shuffler is 2.5 m long by 1 m high by 1 m wide.

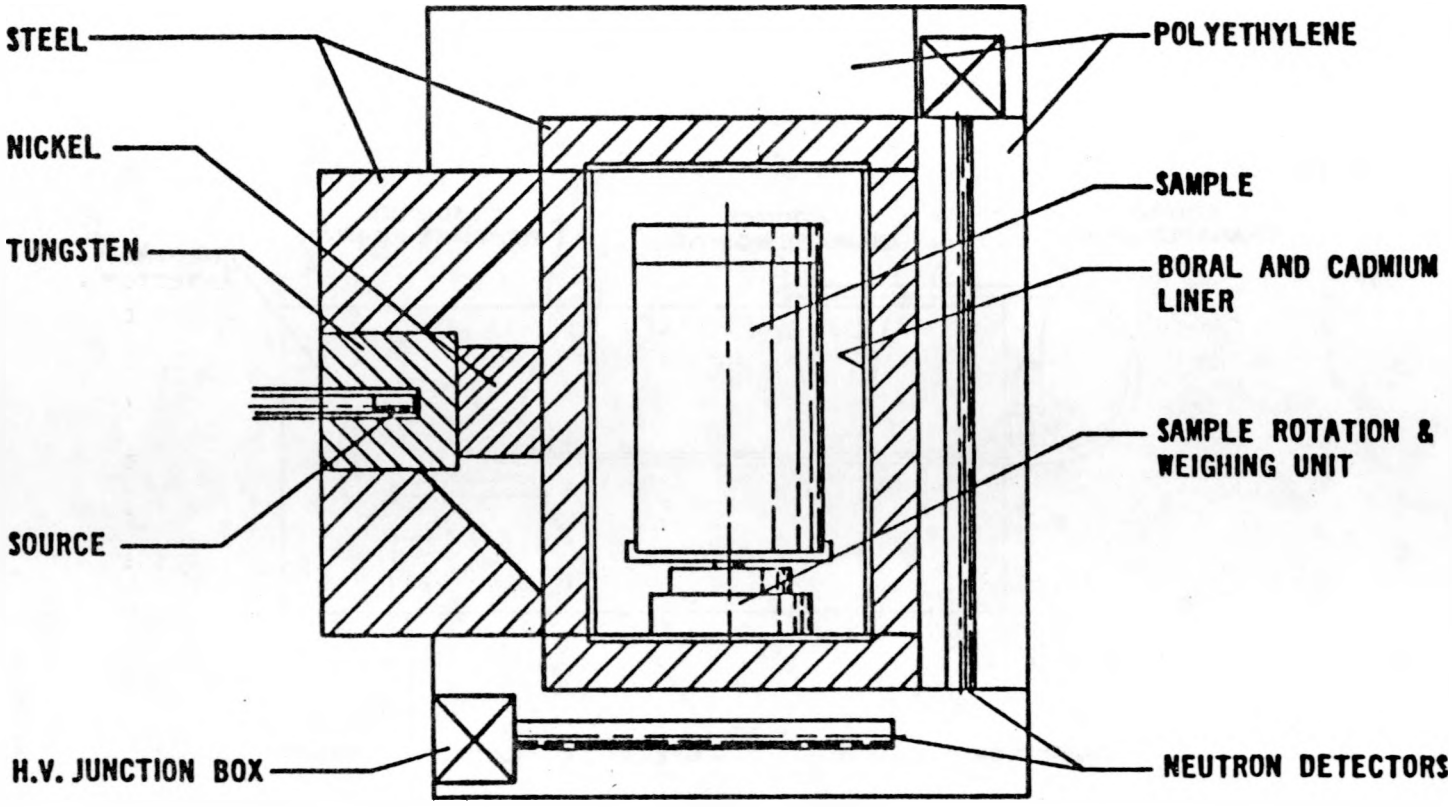


Fig. 23.
Closeup of the sample interrogating/counting well of the SRP shuffler, showing the placement of neutron detectors, shielding, and the source irradiation position.

inventory samples, uranium ore, plutonium mixed-oxide fuel, and irradiated fuel.

The ^{252}Cf shuffler technique can be applied to a fissile loading as low as about 0.1 mg and to loadings as high as criticality safety will permit. For metallic samples, an assay of 10 to 20 kg of ^{235}U is possible. Sample sizes have varied from 1-dram vials to 55-gallon barrels. However, it is generally not practical for a single instrument to measure the entire range of sizes or loadings. To achieve accuracy and minimum assay time, a shuffler should be optimized for the sample size, fissile loading ranges expected, and enrichment of the items to

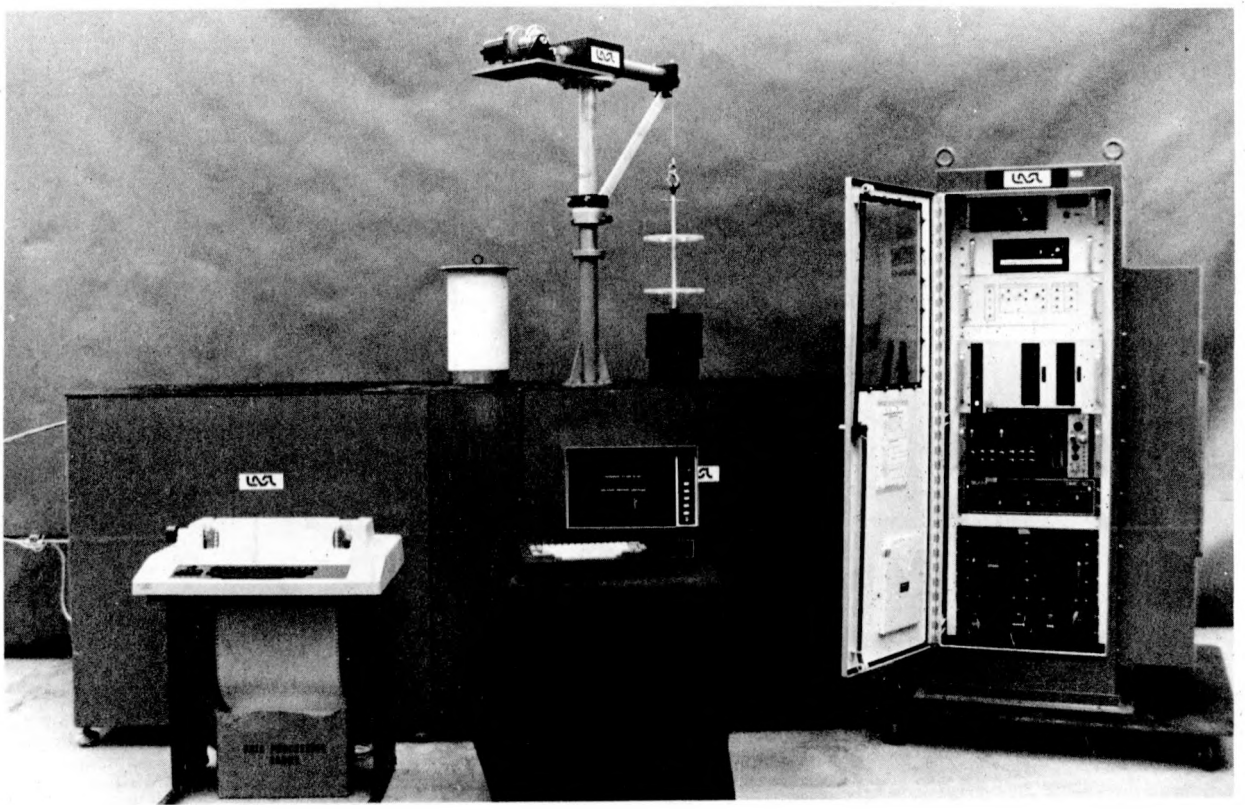


Fig. 24.
Photograph of complete SRP shuffler system. The large body in the background houses the sample chamber and interrogation source, along with all of the requisite shielding. At the far right is the control electronics for the shuffler hardware as well as the pulse-processing electronics and assay computer. In the foreground are the TV readout/communications terminal and the typewriter terminal for hard copy of the assay results. Samples are lowered into the sample chamber by means of the mechanized hoist shown in the approximate center of the main shuffler body.

be assayed. Ultimately the accuracy of the shuffler system depends on the accuracy of the calibration standards and how well they match the materials being assayed. The uniformity and composition of the sample materials are also important considerations. In addition to assay precision, the shuffler system can also be very effective at detecting small quantities

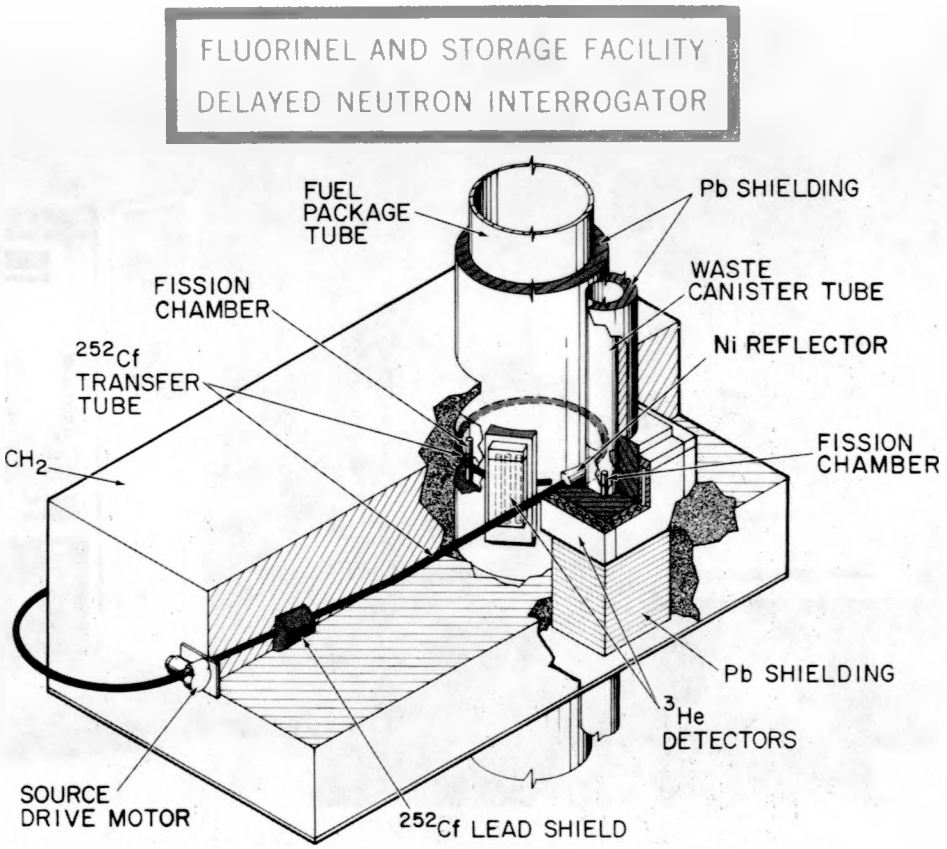


Fig. 25.
Cutaway view of combination waste canister and fuel element shuffler assay system. Fuel elements will be lowered through the fuel package tube at a rate of approximately 50 cm/min while the assay cycles are being carried out. The waste canisters will be lowered through the waste canister tube at a rate of about 13 cm/min during their assay cycles. The two assay tubes have their own, tailored detector arrays and shielding for optimum performance of the particular assay desired. In addition, the source irradiation position is different for the two assay situations in the design.

of fissile material. For large dense samples such as 55-gal barrels, the ^{252}Cf shuffler presents the most sensitive technique that has been demonstrated to be practical. Sensitivities in the milligram range to uranium or plutonium in barrels make the shuffler a valuable tool in waste management tasks.

Another shuffler system being developed is an example of a slightly more versatile design. The shuffler, shown in Figs. 25 on the preceding page and 26 on this page, is designed to assay the ^{235}U content in bulk waste solids and high enriched spent fuel assemblies. The instrument will be used for materials accountability and criticality control by the facility operator. A range of fissile content varies from 0 to 400 g in the waste and from about 8 to 12 kg in the spent fuel assemblies. Measurements must be obtained in the presence of large neutron and gamma-ray backgrounds.

E. Active Measurement Devices--Fuel Assembly Assay

As mentioned above, spent fuel assay techniques will be discussed in some detail in a following lecture (Session 21).

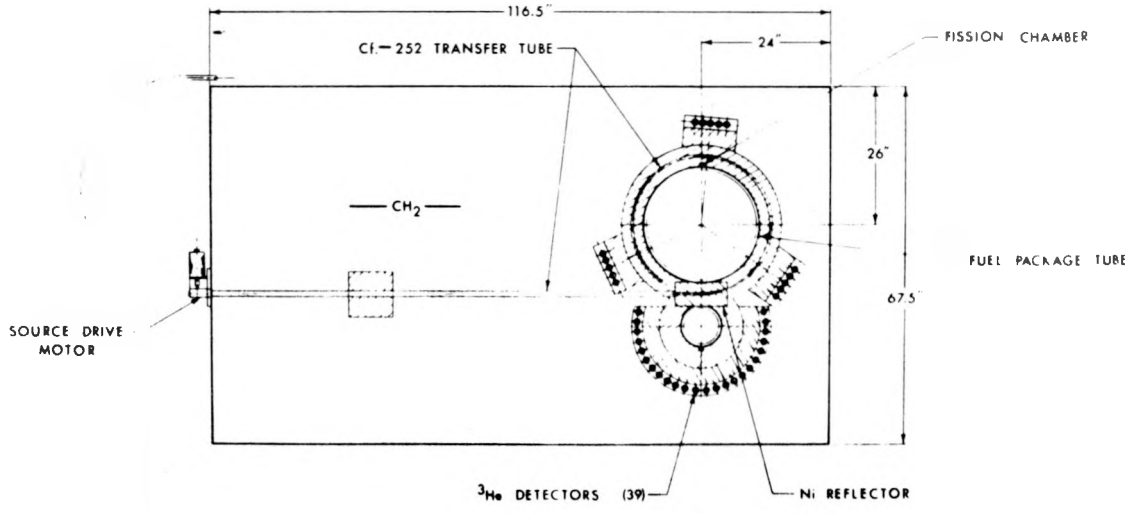


Fig. 26.
Top plan view of combination shuffler, showing more clearly the two detector arrays and sample chambers.

However, we show here very briefly a portable active neutron assay system suitable for an assay of an LWR fuel assembly for fissile content. A schematic of this instrument is shown in Figs. 27-29, and the instrument is discussed in detail in

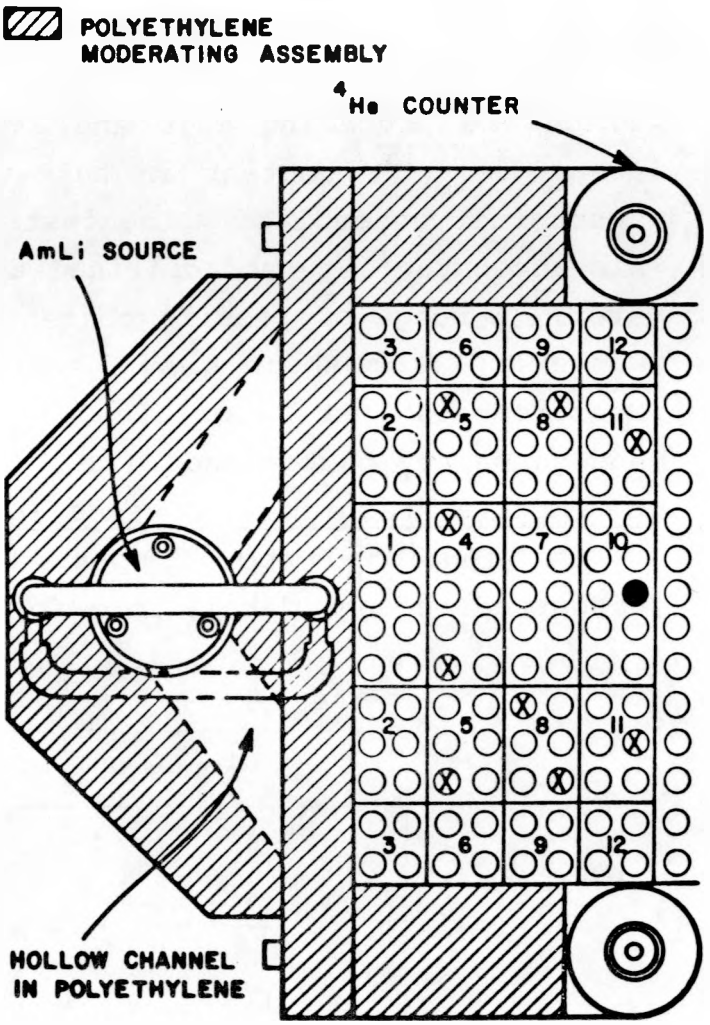


Fig. 27.
Active neutron assay system for LWR fuel assemblies. Two ^{4}He detectors flank the fuel assembly and are stimulated by an AmLi neutron source. The small circles represent individual fuel rods in part of a 15-by-15-rod assembly. The AmLi source is shown in its irradiation position.

Ref. 11. The model shown in these two figures relies on the counting of singles neutrons from the AmLi neutron source interrogation of the fuel. A more recent model uses the coincidence counter philosophy and counts the correlated prompt fission neutrons.

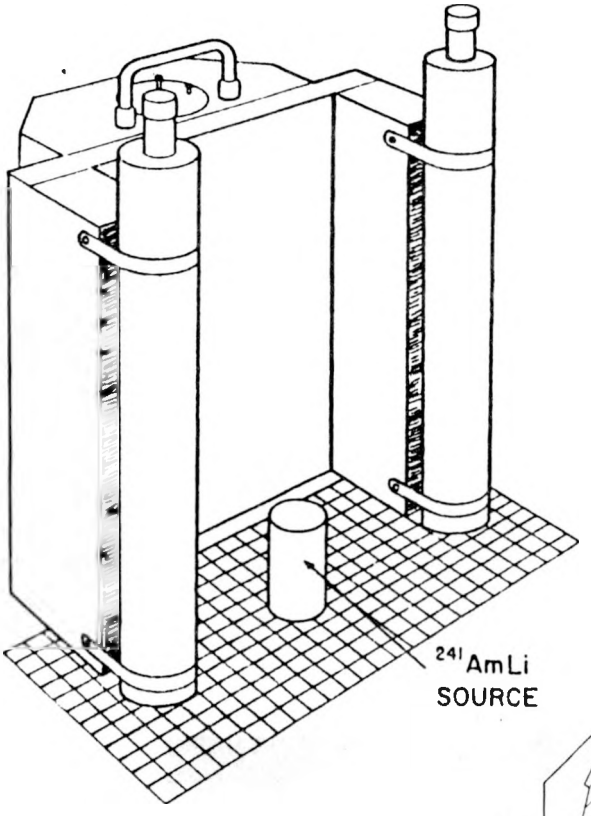


Fig. 28.
LWR active neutron assay system, shown in field calibration configuration, with AmLi source out between the ^{4}He detectors.

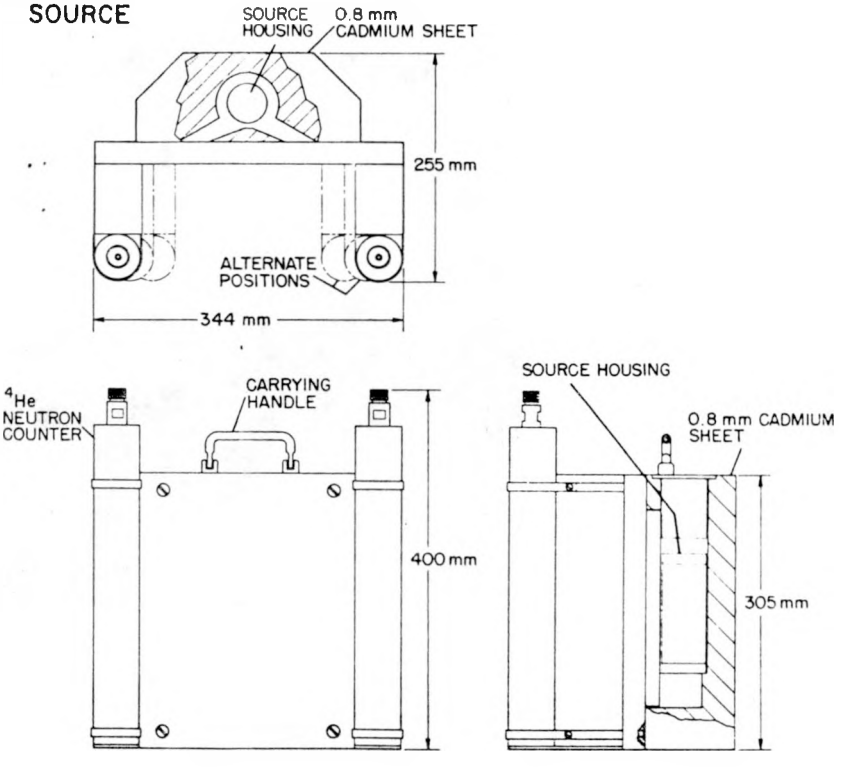


Fig. 29.
Drawing of the LWR active neutron assay system showing dimensions, assembly details, and the source holder.

V. COMBINATION NEUTRON-BASED AND GAMMA-RAY-BASED INSTRUMENTS

It was mentioned earlier that the information obtained from neutron-based instruments is in some ways complementary to that routinely obtainable from high-resolution gamma-ray-based systems. In the former case, very precise measures of fissile content are obtainable from neutron-based systems, even in high gamma-ray and possibly high neutron backgrounds. When combined with the detailed isotopic information available from high-resolution gamma-ray measurements, these data can be used to give a very complete SNM assay of a sample. An example of such an application is shown in Fig. 30. In this figure we see a passive neutron coincidence counter in combination with a high-resolution Ge detector for assaying a fast critical assembly fuel drawer. The same approach can be used to assay fuel rods and fuel rod assemblies.

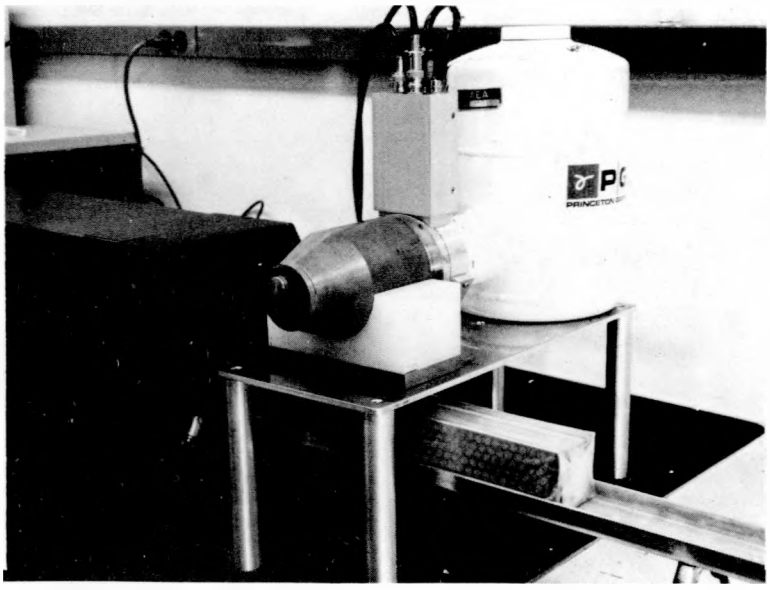


Fig. 30.
Photo of a measurement setup for fissile content and isotopic assay of fast critical assembly fuel drawers. The fuel drawer is lying in the well of a passive coincidence counter (HLNCC), which is shown on the left. Adjacent to this is a germanium detector that has been collimated so that it views a specific region on the fuel drawer, just below the detector shielding in the picture.

VI. REFERENCES

1. R. Augustson and T. Reilly, "Fundamentals of Nondestructive Assay of Fissionable Material," Los Alamos Scientific Laboratory report LA-5651-M (1974).

2. T. Reilly and J. Parker, "A Guide to Gamma-Ray Assay for Nuclear Material Accountability," Los Alamos Scientific Laboratory report LA-5794-M (1975).

3. S. T. Hsue, T. W. Crane, W. L. Talber, and J. C. Lee, "Nondestructive Assay Methods for Irradiated Nuclear Fuels," Los Alamos Scientific Laboratory report LA-6923 (1978).

4. S. T. Hsue, "Methods of Nondestructive Assay of Irradiated Nuclear Fuels for Safeguards," Atomic Energy Review 16, 1, p. 89 (1978).

5. J. R. Phillips, T. R. Bement, K. Kaieda, and E. C. Medina, "Nondestructive Verification of Relative Burnup Values and Cooling Times of Irradiated MTR Fuel Elements," Los Alamos Scientific Laboratory report LA-7949-MS (1979).

6. E. R. Martin, D. F. Jones, and J. L. Parker, "Gamma-Ray Measurements with the Segmented Gamma Scan," Los Alamos Scientific Laboratory report LA-7059-M (1977).

7. N. Ensslin, J. Stewart, and J. Sapir, "Self-Multiplication Correction for Neutron Coincidence Counting," Nucl. Mat. Mgmnt. VIII, 2, p. 60 (Summer 1979).

8. H. O. Menlove and M. S. Krick, "The High-Level Neutron Coincidence Counter (HLNCC): Users' Manual," Los Alamos Scientific Laboratory report LA-7779-M (1979).

9. H. O. Menlove, "Description and Operation Manual for the Active Well Coincidence Counter," Los Alamos Scientific Laboratory report LA-7823-M (1979).

10. H. O. Menlove and T. W. Crane, "A ^{252}Cf-Based Nondestructive Assay System for Fissile Material," Nucl. Instr. Meth. 152, 549 (1978).

11. J. D. Brandenberg, H. O. Menlove, and E. Medina, "Portable Active Neutron Interrogation System for Light-Water-Reactor Fuel Assemblies," Los Alamos Scientific Laboratory report LA-7528-M (1978).

INTERNATIONAL TRAINING COURSE ON NUCLEAR MATERIALS ACCOUNTABILITY FOR SAFEGUARDS PURPOSES

SESSION #20: NATIONAL SYSTEM OF MEASUREMENT STANDARDS

SPEAKER: Dr. H. Thomas Yolken

National Bureau of Standards
Washington, DC USA

Monday, June 2, 1980
4:00 p.m.

BIOGRAPHY

Education: University of Maryland - B.S. in Metallurgy - 1960
University of Maryland - Ph.D. in Materials Science - 1970

Present Position: Chief, Office of Measurements for Nuclear Technology, National Bureau of Standards, Washington, DC 20234.

Past Positions: 1971-1975 Deputy Chief, Office of Standard Reference Materials, NBS; 1966-1970 Special Scientific Assistant to the Director, Institute for Materials Research, NBS; 1960-1966 Corrosion Research at NBS.

Session Objectives

SESSION #20: NATIONAL SYSTEM OF MEASUREMENT STANDARDS

The concept and purpose of measurement standards and their role in the assay of special nuclear material (SNM) will be reviewed. A National Standards System will be described that would establish standardized assay procedures with prescribed methods of defining limits of error, and would provide primary standard reference materials against which all SNM assays can be compared. Methods of setting up local (secondary) standards and establishing clear paths of traceability back to the primary reference material will also be described.

After the session, participants will be able to

1. Define what is meant by a "primary" and a "secondary" reference standard.

2. Describe briefly the role of a reference material in the calibration of a measurement instrument.

3. Discuss the importance of traceability of the calibration of an assay instrument back to primary standard reference materials.

4. Cite and describe briefly examples of existing National Systems of Measurement Standards.

INTERNATIONAL TRAINING COURSE ON NUCLEAR MATERIALS ACCOUNTABILITY FOR SAFEGUARDS PURPOSES

SESSION 20: NATIONAL SYSTEM OF MEASUREMENT STANDARDS*

H. Thomas Yolken
Office of Measurements for Nuclear Technology
National Bureau of Standards

I. INTRODUCTION

Most measurements are made to communicate information on properties of material things in a purposeful way to accomplish useful goals. Communications can be effective or ineffective. In parallel, measurements can be meaningful or not meaningful. Meaningful measurements allow us to make decisions on a solid, objective basis. For example, a transaction involving 10 kilograms of sucrose of 99 percent purity is likely to cause little controversy between buyer and seller, because analytical instruments and scales are available to determine 99 percent purity and 10 kilograms to within some specified degree of tolerance. On the other hand, measurements of fragrance and taste on a scale of desirability are still highly subjective, and decisions based on results of "experts" are frequently subject to controversy.

The principal difference in the two situations is that the first involves numbers associated with distinct, well-defined properties, while the second does not. To see what constitutes meaningful measurements, and how they may be propagated throughout the industrial and technological communities, the measurement process must be examined.

II. THE MEASUREMENT PROCESS

Measurement in science and technology is that process whereby a numerical value is associated with a distinct, specific, and unique property of a material. The magnitude of the number is related to the amount or degree of that property in a particular material or similar class of materials. The word "material" is taken in its broadest sense to include all those things considered to constitute the physical objects of the observable universe.

---

*Much of this lecture is taken from "Standard Reference Materials: The Role of SRM's in Measurement Systems," NBS Monograph 148, January 1975.

In 1939, Shewhart[1] pointed out two aspects of the measurement process that he described as quantitative and qualitative. The former aspect concerns numbers associated with a scale, pointer reading, counter, or the like. In this lecture, the quantitative consideration will be associated with: a well-characterized material, called a Reference Material (RM); reference data (e.g., the half-life of Pu-239); transfer artifacts such as a meter stick or standard weights; or an instrument calibration service. In reference materials, one or more properties will have several numbers assigned in a manner analogous to the numbers associated with a meter stick, although in many instances the RM will have one unique value, rather than a series of incremental values. Thus, while a meter stick will have numerous divisions along its length, a uranium RM may have just one number associated with its uranium content and with each isotopic ratio. In any case, the RM represents the quantitative aspect of measurement, especially useful where chemical or isotopic composition is the property being measured. The qualitative aspects of measurement are included in what is often called the procedure or the method. Included in this factor are such things as apparatus, reagents, indeed all those things that are used or can affect the course of the measurement. Obviously, the instrumentation and method, the sequence of operations, control of the ambient conditions, etc., must be stated in the written procedure used by the operator to make the actual measurements.

Experience over many centuries has taught man that, if he can agree on one universal set of coherent scales, he can more effectively communicate with his fellow man across time and geographical boundaries. In principle, there is no logical reason why many different sets could not be utilized, as historically they have, but the economic, political, and social benefits of one universal set are so apparent, that most of the world's nations have now agreed to use the set of measurement scales called the International System of Units (Système International d'Unités) and abbreviated as SI. This rational, self-consistent system of units of measurement includes the base units (mass, length, time, electric current,

thermodynamic temperature, luminous intensity, and the mole), and the derived units (area, density, energy, etc.), together with rules for their use.[2,3]

Access to the units is provided through highly refined measurement processes. In some cases one can reconstruct the unit, in others one relies on artifacts such as sets of weights or gauge blocks, whose magnitudes in terms of the unit have been carefully established. Use of artifacts or reference standards is especially important when the magnitudes met in local measurement practice are far removed from (i.e., are large multiples or small fractions of) the base unit. The uncertainty in the use of reference standards is a function of both the method and the process precision. The uncertainty of the assigned value of the reference standards becomes a systematic error of the process in which the artifact is used.

III. COMPATIBILITY IN MEASUREMENT

If measurements within a nation, between nations (or international organizations), between industries, between buyer and seller, indeed between any two or more parties are to be useful, the measurements must be compatible. Assume two different laboratories measure the same specific property on samples taken from the same lot of a stable material. If the two independently determined values agree, the two measurements, and hence, the two laboratories are said to be compatible. The critical question is, "Agree within what limits?" In practical measurement situations, these limits should be defined in terms of the useful end requirements. Having established the limits, one is concerned with verifying that the results of the measurement are compatible with the limits.[4]

IV. MEANINGFUL MEASUREMENT

By definition, a measurement system produces a numerical value for a well-defined property of a material. The technique by which such a numerical value is obtained is called a measurement method.

The practical measurement processes of industry are varied and complex. In some instances, such as the measurement of the disintegration of nuclear particles using a radiation detector, the process is essentially a counting operation and therefore conceptually simple. However, in the majority of situations, the process is far more complex and consists of a sequence of operations, each of which may be a process of some complexity. At the end of this sequence of operations, a numerical result emerges together with an estimate of uncertainty. In most cases this result can be expressed in units belonging to an accepted system; for science and technology, that system is the SI. While every one could start with a fresh realization of the base units, the accumulative systematic error as one moves through the process can be large. Such an approach, in addition to being costly, may not produce results that are within the desired limits. It is here that the role of measurement standardization or karminization emerges.

If a measurement process is to be meaningful, the numerical values obtained should be specific, precise, and free of systematic error (or bias) within the agreed on or practical limits required for the end use. When these goals have been achieved, the measurement results may be said to be accurate. Thus, by this definition, a meaningful measurement is termed an accurate measurement. A detailed discussion of concepts of accuracy is not appropriate for this lecture. Detailed discussions may be found in Refs. 5 and 6.

Figure 1 has been used with some success as an analogy to explain the difference between accuracy and precision. Three imaginary marksmen fire a rifle at a target. In the top target, the marksman is both imprecise and inaccurate. The marksman is quite precise but inaccurate in the middle target, and in the bottom target, the marksman is both accurate and precise. Of course, the analogy in all cases is that the bull's eye corresponds to the target value or "true" value.

A. Specificity

During the measurement process, only the property under test must be measured, and not some combinations of properties that may give the false impression of singularity. Non-specificity can be considered a

special case of systematic error, and could be included in that discussion. However, especially in the measurement of chemical composition, its insidiousness as a special source of error is so striking (when found) that special emphasis is warranted.

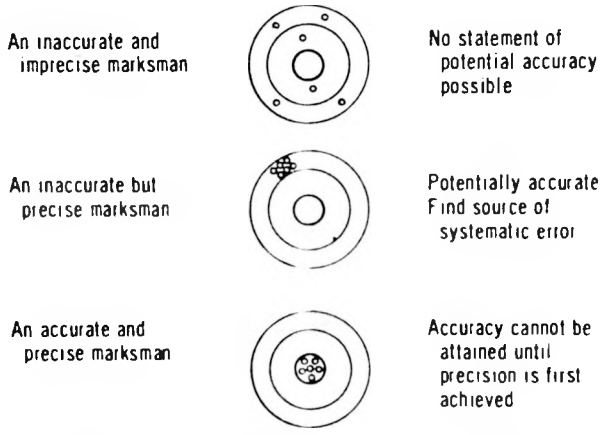


Fig. 1. Accuracy and Precision

The determination of strontium in granite done at NBS several years ago is an interesting example of the meaning of specificity. The results obtained by two different methods are shown in Fig. 2: 1.0 ppm Sr

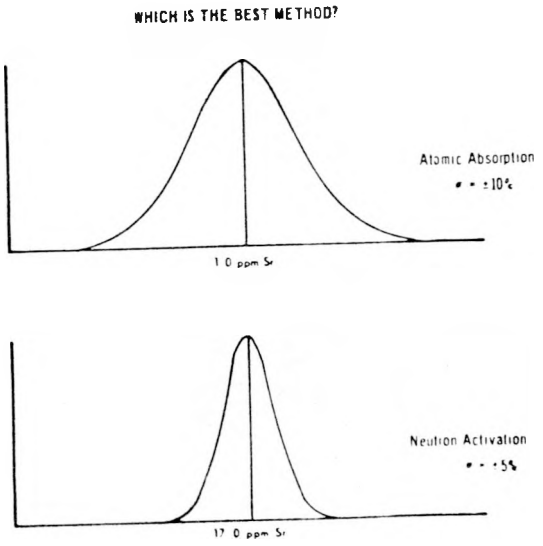


Fig. 2. Strontium in Granite

obtained by atomic absorption with a standard deviation of 20 percent; and 17.0 ppm Sr obtained by neutron activation analysis with a standard deviation of 5 percent. Which result is closer to the bull's eye? In this case, the atomic absorption results were more accurate even though they had a standard deviation twice that of the neutron activation analysis measurements.

The reason for the larger error in the neutron activation analysis results was non-specificity of the technique. The granite not only contained Sr but also uranium-235, which fissioned on irradiation with neutrons to give an additional apparent 16 ppm Sr, as shown by Fig. 3.

SYSTEMATIC ERROR - THE VILLAIN

The Strontium contained in the granite is made radioactive by irradiation with neutrons and the radioactive Sr is then counted and the original Sr content calculated: Thus:

$$^{88}Sr + n \longrightarrow {}^{89}Sr \xrightarrow[54d]{\beta} {}^{89}Y \text{ (stable)}$$

BUT (the villain) granite contains uranium and

$$^{235}U + n \xrightarrow[\text{products}]{\text{fission}} {}^{89}Sr \xrightarrow[54d]{\beta} {}^{89}Y \text{ (stable)}$$

And there is no physical way of differentiating the ^{89}Sr formed from the uranium from that formed from the natural occurring strontium

1 ppm U gives an apparent 16 ppm Sr

Fig. 3 Systematic Error

B. Precision

A high degree of precision in a measurement process is demonstrated when essentially the same numerical value is repeatedly obtained. In some measurement circles, the measure of precision within the same

laboratory is called repeatability; between different laboratories, reproducibility. (Alternatively, the terms, "intra-" and "inter-laboratory" precision, respectively, are often used.) The interplay and complications of varying degrees of inaccuracy with varying degrees of imprecision in a measurement process are discussed in depth by Eisenhart.[6] In practice, as opposed to theoretical considerations, a high degree of accuracy is usually positively correlated with a high degree of precision. But, also in practice, highly precise systems are sometimes found to be highly inaccurate. This is a real danger and must be carefully considered.

C. Systematic error

The third requirement for a measurement to be meaningful is that it be free of systematic error. When systematic errors are present the numerical result differs from the "true value." From a practical point of view for the large majority of measurements made in industry and technology, the "true value" can be considered in an operational sense. This requires that a careful assessment be made of the systematic errors in each step of the measurement process. When the systematic errors have been identified and eliminated, the resulting numerical value can be equated to the "true value." Furthermore, the value obtained by this process should be essentially the same as that obtained by any other acceptable process used to measure the same property of the same material.

D. Other Desirable Characteristics

There are, of course, other desirable attributes of measurement--sensitivity of detection, a large dynamic range, ease of operation, speed, low cost, and several others. These, however, are pragmatic considerations by and large, while specificity, precision, and freedom from systematic error are absolute essentials to the attainment of meaningful measurement.

V. A NATIONAL AND INTERNATIONAL STANDARDS SYSTEMS APPROACH TO MEANINGFUL MEASUREMENT

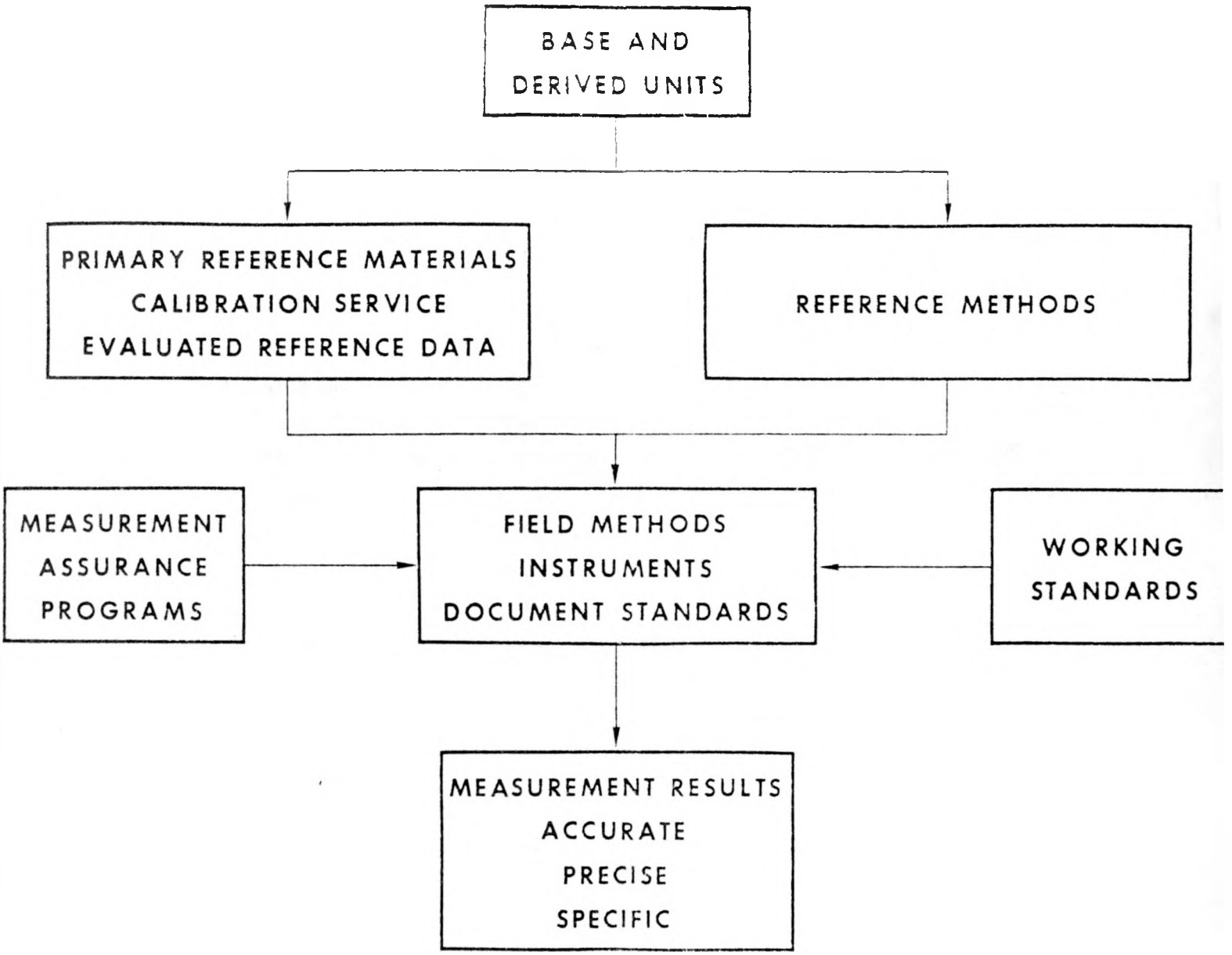
There are several ways in which a meaningful national/international measurement system can be built, maintained, or expanded. Many of

these modes are now in operation in various sciences, industries, and technologies. Principal among these are: calibration services, especially for instruments calibrated at a central, competent source and returned to the user; publication of standard reference data, which if critically evaluated and given together with the detailed measurement procedure, allows others to use the data directly or to reproduce the original measurements; the provision of measurement signals (time interval, frequency, etc.) via a central source to users; transfer through methodology dependent upon locally produced materials of realizable purity and stability (e.g., specification of the purity of platinum used to realize the candela); manufactured devices and/or materials made available to produce compatibility in a narrow field on a relative rather than absolute (or accuracy) basis.

In this discussion emphasis will be placed on a meaningful measurement system based on RM's, artifacts (e.g., reference weights, length scales), and reference methodology. Together, these provide a mechanism whereby compatibility can be transferred with speed and modest cost into practical measurement fields. There are five major components of this system that will be described, some in more detail than others. The relationships of these components are shown in Fig. 4.

A. Component 1--a Rational, Self-Consistent System of Units of Measurement

By international agreement, this system of units is the SI, now widely used in scientific measurement areas (metrology, physics, chemistry, etc.). In addition, full implementation is almost complete in areas of industrial technology. The most obvious exception is that of the U.S. where many engineering and technical measurements are still made and reported in non-SI units. For well over 90 percent of the RM's issued in the U.S. through NBS, the properties are given in SI units, although the corresponding non-SI units may also be reported. For some engineering oriented RM's, arbitrary, non-coherent units are used where the RM is made part of a test recipe (e.g., the "flame spread index" of the Surface Flammability RM).


BASE AND
DERIVED UNITS
PRIMARY REFERENCE MATERIALS
CALIBRATION SERVICE
EVALUATED REFERENCE DATA
REFERENCE METHODS
MEASUREMENT
ASSURANCE
PROGRAMS
FIELD METHODS
INSTRUMENTS
DOCUMENT STANDARDS
WORKING
STANDARDS
MEASUREMENT RESULTS
ACCURATE
PRECISE
SPECIFIC

B. Component 2--The Materials to Realize in Practice the SI Units and their Derivatives

To realize (or determine) the SI unit candela, platinum of a specified and known purity is necessary because the candela is defined in terms of the radiation of a black body at the freezing temperature of platinum. A rigorous procedure is specified for this determination. The eventual accuracy will depend on both the purity of the platinum and the adequacy of the method. Such methodology is called a reference method. Similarly, if the mass and isotopic composition of uranium in nuclear fuel is to be determined with known accuracy, so that compatibility throughout the nuclear safeguards community may be propagated, materials with known isotopic and chemical content and a reference method of analysis to specify the actual steps in their determination must be available.

In much of the world, these well-characterized materials are called Reference Materials (RM's) and are prepared, measured, and certified, in most instances, by national standards laboratories. A formal definition of an RM, which includes these primary uses, is given in Fig. 5. The key characteristic of an RM is that the properties of interest be measured and certified on the basis of accuracy.

A FORMAL DEFINITION OF REFERENCE MATERIALS (RM's)

RM's are well-characterized and certified materials, produced in quantity:

(1) To help develop reference methods of analysis or test; i.e., methods proven to be accurate.

and/or

(2) To calibrate a measurement system in order to:

(a) Facilitate the exchange of goods
(b) Institute quality control
(c) Determine performance characteristics
(d) Characterize at scientific frontiers

and/or

(3) To assure the long-term adequacy and integrity of the quality control process.

thus

ENSURING THE COMPATIBILITY AND MEANINGFULNESS OF MEASUREMENT IN THE NATION

for

...Science and Technology
...Production and Distribution of Goods and Services
...Government

Fig. 5

A list of reference materials currently available on a worldwide basis for nuclear safeguards applications has been published.[7] The RM's fall into three broad categories for use in: (1) chemical assay of Pu and U by destructive methods; (2) isotopic assay of Pu and U by destructive methods; and (3) non-destructive isotopic and chemical assay by various methods, e.g., gamma spectrometry, calorimetry, and active and passive neutron spectroscopy.

All methods for the determination of uranium and plutonium require reference materials for calibration of the procedure and for measurement control on a daily basis. Even "absolute" techniques, such as controlled-potential coulometry, require a single standard solution, prepared from a reference material, to verify the accuracy and precision of the method. The performance of a gravimetric technique should be regularly checked with a suitable reference material. Surface ionization mass spectrometry, an "absolute" technique in which isotopic ratios are measured, nevertheless requires a standard reference material for the determination of the mass discrimination bias factor and other reference materials for the measurement control program.

In most analytical and plant control laboratories, two categories of reference materials are employed. The first category consists of standard or primary reference materials. These are stable materials characterized, certified, and distributed by a national or international standards body. In the U.S., primary reference materials are offered for sale by the National Bureau of Standards (NBS)[8] and the New Brunswick Laboratory (NBL)[9] of the U.S. Department of Energy. Plutonium and uranium reference materials available from these two organizations are listed in Tables I and II. The proper use of reference materials in the nuclear fuel cycle has been described.[10] In addition to the materials listed, NBS also certifies several high-purity chemicals that are used as oxidation-reduction standards in the titrimetric determination of uranium and plutonium. These chemicals are:

SRM 40	Sodium oxalate
SRM 83	Arsenic trioxide
SRM 136	Potassium dichromate.

TABLE I

NUCLEAR REFERENCE MATERIAL AVAILABLE FROM THE NATIONAL BUREAU OF STANDARDS[a]

Special Nuclear Materials

These SRM's are available to DOE contractors, NRC or State Licensees, and foreign governments that have entered an Agreement for Cooperation with the U.S. Government concerning the Civil Uses of Atomic Energy. The purchase request for these SRM's must be made on special forms obtainable from the Office of Standard Reference Materials, Room B311, Chemistry Building, National Bureau of Standards, Washington, D.C. 20234.

Plutonium Assay Standards

SRM	Type	Certified for	Wt/Unit (grams)	Purity (%)
944	Plutonium sulfate tetrahydrate	Plutonium Content	0.5	47.50*
945	Plutonium metal, standard matrix	Impurities	5	(99.9)
949e	Plutonium metal assay	Plutonium Content	0.5†	99.996
955	Plutonium—244 Spike	IN PREP		

*Stoichiometric †Nominal weight (Values in parentheses are not certified, but are given for information only.)

Plutonium Isotopic Standards

SRM	Type	Wt. Units (grams)	Atom Percent ^{238}Pu	^{239}Pu	^{240}Pu	^{241}Pu	^{242}Pu
946	Plutonium Sulfate Tetrahydrate	0.25	0.247	83.128	12.069	3.991	0.565
947	Plutonium Sulfate Tetrahydrate	.25	.296	75.696	18.288	4.540	1.180
948	Plutonium Sulfate Tetrahydrate	.25	.011	91.574	7.914	0.468	0.0330

Uranium Assay Standards

SRM	Type	Certified For	Wt Unit (grams)	Purity (%)
950b	Uranium Oxide	Uranium Oxide	25	99.968 (U_3O_8)
960	Uranium Metal	Uranium	26	99.975 (U)
993	Uranium—235 Spike (solution)	Uranium	15	99.8195 (U-235)

Uranium Isotopic Standards

SRM	Uranium Oxide (U_3O_8)	Wt (grams)	Atom Percent ^{234}U	^{235}U	^{236}U	^{238}U
U-0002	Depleted	1.0	0.00016	0.01755	<0.00001	99.9823
U-005	Depleted	1.0	.00218	.4895	.0046	99.504
U-010	Enriched	1.0	.00541	1.0037	.00681	98.984
U-015	Enriched	1.0	.00850	1.5323	.0164	98.443
U-020	Enriched	1.0	.0125	2.038	.0165	97.933
U-030	Enriched	1.0	.0190	3.046	.0204	96.915
U-050	Enriched	1.0	.0279	5.010	.0480	94.915
U-100	Enriched	1.0	.0676	10.190	.0379	89.704
U-150	Enriched	1.0	.0993	15.307	.0660	84.528
U-200	Enriched	1.0	.1246	20.013	.2116	79.651
U-350	Enriched	1.0	.2498	35.190	.1673	64.393
U-500	Enriched	1.0	.5181	49.696	.0755	49.711
U-750	Enriched	1.0	.5923	75.357	.2499	23.801
U-800	Enriched	1.0	.6563	80.279	.2445	18.820
U-850	Enriched	1.0	.6437	85.137	.3704	13.848
U-900	Enriched	1.0	.7777	90.196	.3327	8.693
U-930	Enriched	1.0	1.0812	93.336	.2027	5.380
U-970	Enriched	1.0	1.6653	97.663	.1491	0.5229

[a]Taken from Ref. 8.

TABLE II

REFERENCE MATERIALS FOR CHEMICAL ANALYSIS[a]

Uranium Assay Materials

RM No.	Type	Certified For	Wt/Unit (grams)	Content Uranium	(Wt%) U-235
17B	Uranium tetrafluoride	Uranium content, U(IV), UO_2, some impurities	200	75.87	---
18	Uranium oxide (UO_3)	Uranium content, some impurities	500	82.10	---
97[b]	Uranium oxide (UO_2), enriched	Uranium, ^{235}U	100	87.75	2.380
112	Uranium metal, chips	Uranium, impurities	50	99.909	
113[b]	Uranium hexafluoride (UF_6), enriched	Uranium, ^{235}U	5-10	67.580	1.7126
114	Uranium oxide (U_3O_8)	Uranium, impurities	50	84.73_9	---
115	Uranium metal, rod, depleted	Uranium, ^{235}U	75	99.97_7	0.2008
116[b]	Uranium metal, chunk, enriched	Uranium, ^{235}U	1.5	99.96_7	93.120
118[b]	Uranium-thorium carbide, BISO bead form, enriched	Uranium, thorium, ^{235}U	10	13.38_8	93.095
119[b]	Uranium-thorium carbide, TRISO bead form, enriched	Uranium, thorium, ^{235}U	15	7.12_8	93.095
120[b]	Uranium oxide (UO_2), enriched	Uranium, ^{235}U	50	87.35_5	1.349

Uranium Isotopic Materials

RM No.	Type	Certified For	Wt/Unit (grams)	$^{233}U/^{235}U$	$^{238}U/^{235}U$
117[b]	Uranium isotope mixture, solution	Atom ratio, 3/5 and 8/5	0.3	1.0076	1.0119

Trace Element Materials

98(1-7)	Uranium oxide (U_3O_8)	30 impurity elements	7 x 25

[a]Taken from Ref. 9.

[b]Special nuclear material. License required for purchase.

The reference materials listed in the tables are for chemical and isotopic assay by destructive methods. Reference materials for non-destructive methods of analysis (NDA) are being prepared and evaluated. The NDA Working Group of ESARDA initiated a U_3O_8 NDA standard reference material project that has been undertaken by the Central Bureau of Nuclear Measurements (CBNM) at Geel, the U. S. National Bureau of Standards, and the various member laboratories of ESARDA. The project will produce internationally certified reference materials for gamma-ray spectrometric measurement of uranium enrichment in light-water reactor fuel.[11]

The New Brunswick Laboratory has developed three prototype NDA reference material matrices for gamma-ray spectrometric analysis of scrap and waste. The matrices are ion-exhange resin, cellulose fiber, and synthetic calcined ash. A total of 19 reference materials containing varying amounts of 93% enriched uranium have been prepared.[12] These materials are now being evaluated by laboratories in the U.S. and will also be sent to laboratories in the United Kingdom, the Netherlands, and France for analysis. The reference materials will be certified for total uranium content as well as for isotopic composition.

The second category of reference materials consists of working reference materials (or working standards). They are materials that have been derived from primary reference materials or have been characterized against them. Working reference materials are used to monitor measurement methods, to calibrate and test methods and equipment, and to train and test personnel. A working reference material frequently is a product material of the plant that has been thoroughly homogenized and well characterized against a primary reference material.

Primary reference materials are relatively costly and are in limited supply. In addition, their composition may be quite different from that of material encountered in the plant. It is therefore desirable, if not necessary, for each laboratory to prepare and characterize working standards for daily use in the measurement control program. Guides for preparation and evaluation of working reference materials are available.[13-16]

Clark and Jackson[17] have described the preparation, handling, characterization, and packaging of uranyl nitrate solution for use as a working reference material.

A reference material should be analyzed daily or by each shift to ensure that the analytical method is under control. Process samples should not be analyzed until satisfactory results have been obtained on reference materials.

In addition to RM's, in many cases, national standards laboratories provide artifacts to transfer measurement accuracy and compatibility. For example, standard weights, volumes, and length scales are provided.

National standards laboratories also often provide instrument calibration services, to assure accurate and compatible measurements coupled to the SI units. In this approach, the user ships their instrument to a central facility that calibrates the instrument and returns it to the user.

As mentioned earlier, reference data, such as the freezing point of water, can be provided by a standards laboratory to enable the user to calibrate his own temperature measuring device.

C. Component 3--Reference Methods of Measurement Used with or Based on SRM's

A reference method is defined as "a method of proven and demonstrated accuracy." Such methods have been variously called: umpire methods, referee methods and standard methods. In any case, the operational definition just given is the crux of the matter, although international agreement on a descriptor would help to avoid future misunderstanding. Absolute accuracy, implying methods with no biases, is an unattainable goal, not achievable by mere mortals. It is important to realize that the cost of obtaining greater accuracy increases exponentially. Therefore, only that degree of accuracy required should be sought, making allowance for further advances in the state-of-the-art. A good guideline is to strive for a reference method whose accuracy is three times better than that currently required by the end use.

The development of reference methods is a time-consuming, expensive, and complex process, involving these steps (although permutations are possible).

1. A group of experts surveys the literature to choose a candidate method--one expected to have small biases. They also decide what the accuracy goal should be for the reference method, considering the required end use.

2. A central laboratory is chosen to coordinate the work; develop the statistical design; prepare and distribute samples that have been previously measured by the central laboratory using an independent method of known accuracy, but one not usually available to the field in question; and distribute the RM. A precondition is the availability of the appropriate RM.

3. The group of experts, in conjunction with the central laboratory, writes the first version of a detailed procedure (protocol), and helps select a group of measurement laboratories (usually 6 to 10) willing to perform the work.

4. The central laboratory distributes the protocol, sample, RM, and instructions to the cooperating laboratories. The cooperating laboratories perform the work according to a schedule. The analytical data plus other pertinent information are returned to the central laboratory.

5. The group of experts, plus qualified personnel from the central laboratory, analyze the data, identify sources of error and then revise the protocol to eliminate them.

6. Steps 4 and 5 are repeated as often as necessary to achieve the desired accuracy.

7. The protocol is written in final form and published in a journal, a collection of reference methods, or another appropriate publication.

The number of reference methods, world-wide, is discouragingly small, and in view of the needs, an interim solution to this question of accuracy may have to suffice for now. Some scientists have proposed accuracy by edict, a scheme whereby experts declare a particular method to be the accurate method against which all other alternative methods will

be assessed. If the method chosen by edict is carefully selected after some modest interlaboratory testing is done, then this interim solution may meet present pressures and at least assure compatibility.

In the long run no substitute exists for the hard, scientific work that establishes the accuracy of the analytical method in the laboratory.

D. Component 4--Establishment of Compatibility into a Wider Area of Technology via RM, Reference Data Artifacts, and Reference Method

Components 1, 2, and 3 are sufficient in themselves to bring about accurate measurements in a few well-qualified laboratories. The real problem is, however, to improve the quality of and make compatible the measurement in the average laboratory on a routine basis. There are two aspects to this problem, one involving the field (routine) methods per se, the other concerned with commercially produced (or in-house) working standards.

1. Assessment of field methods.

As reference methods, RM's, reference data, and artifacts become available, responsible groups should begin the assessment of the various field or everyday methods currently in use. When the test materials to be used in the assessment process are characterized on an absolute (accuracy) basis via the reference method and RM, the inaccuracies of the tested field methods will become more readily apparent. Alternatively, in some cases, an RM can be used alone for this purpose. As the testing data accumulate and become widely disseminated, a selection process will occur, and highly inaccurate methods will tend to fall into disuse and eventually disappear. For those field methods having desirable characteristics (speed, low-cost, portability, etc.) it should be possible to correct or eliminate any biases found, thereby placing them, in turn, on an accurate basis.

Many reference methods will not be suitable (because of complexity, cost or lack of speed) for use in daily routine practice. Furthermore, not every laboratory will have the facilities or instruments required by the reference method, while other laboratories may prefer to determine

some constituents through use of simpler methods. Indeed, it is not necessary nor even desirable to do away with present field methods, as long as they are tested against the reference method.

2. Upgrading the quality of working standards.

Given RM's, reference data, artifacts, and reference methods, the manufacturers of working or secondary standard materials (including reagents) and instruments will be able to test those products for accuracy. In the U.S., some manufacturers already are using RM's, where available, to test the quality of their reagents and secondary standards. Without reference methodology, this testing must necessarily be on a relative rather than an absolute basis.

The implementation of these assessment activities is more complex and difficult than those of Components 1, 2, and 3. Modes of implementation, including legal requirements, differ from country to country. In the U.S., standardization in most fields of technology is strictly a voluntary process, as opposed to the practice in many nations where standardization procedures are legally imposed. However, the U.S. does impose regulations for standardization in the nuclear safeguards area.

E. Component 5--Assuring the Long-Term Integrity of the Measurement Process

Measurement systems are notorious for getting out of control unless carefully monitored. Loss of precision is usually the first indication that the measurement process is not in control. In most measurement laboratories, the question of control is one of almost daily concern and one that has been extensively studied and addressed. Although each individual laboratory must ultimately be responsible for assuring its own quality control, professional societies and governmental agencies can, and often do, provide a mechanism that helps to assure, to a degree, long-term quality control.

If RM's and reference methods are available, the mechanism for assuring the long-term integrity of the measurement process in a large number of measurement laboratories is quite straightforward.

1. The sponsoring or testing agency prepares a series of test samples (in this case nuclear materials) incorporated in a suitable matrix that cover the range of values likely to be encountered in real life.

2. The properties are determined by the sponsors' laboratory (or laboratories) using the reference method to obtain values of known accuracy.

3. The test samples, as unknowns, are distributed with suitable instructions and reporting forms to the laboratories under test who perform the work as instructed. In true blind studies, these samples will not be differentiable from daily, routine samples.

4. Results are returned to the sponsoring agency and statistically analyzed. In a well-designed and controlled program, each laboratory should receive back the following information for each property tested: its day-to-day precision within the laboratory; the accuracy of the method used; its rank compared to other laboratories using the same methodology; the accuracy of its method compared to alternative methods; a statement of acceptability of the results (if norms for that technology have been established).

VI. ECONOMIC AND SOCIAL COSTS OF MAKING "BAD" MEASUREMENTS

In today's highly technological society, the costs of making "bad" measurements can be monumental. On the other hand, the benefits that can accrue from "good" measurements, both economically and socially, can be equally large. A measurement system that is non-compatible is obviously a wasteful system. In such systems, the transfer of useful measurement data across different technological or geographic boundaries becomes difficult or impossible, and certainly wasteful. In measurement systems that are not continually in control, the expenditure of a significant portion of the available measurement time in redoing measurements obtained during the out-of-control period is not uncommon. The examination of two large U.S. industries illustrates the economic side of measurements. The first is the U.S. steel industry, whose measurement

system is long established and well under control. The second is the U.S. "health" industry, especially clinical chemistry, which has areas in which lack of agreement between laboratories has received publicity.

VII. CONCLUSION

To have an effective national measurement standards system such as the one described, the system must be developed and maintained domestically. In addition, a similar system is needed internationally to assure compatibility and refined accuracy of measurements made between countries shipping nuclear materials and between countries and the IAEA. This international measurement standards system for nuclear safeguards does not yet exist; however, cooperative efforts between member states and the IAEA have been proposed[18] and are underway.

REFERENCES

1. W. A. Shewhart, "Statistical Method from the Viewpoint of Quality Control," (U.S. Dept. of Agriculture, Washington, DC, 1939).

2. "Standard for Metric Practice," ASTM E 380-79 (American Society for Testing and Materials, Philadelphia, 1979).

3. "SI Units and Rcommendations for the Use of Their Multiples and of Certain Other Units" ISO/1000-1973 (International Organization for Standardization, Geneva, Switzerland, 1973. Available in the U.S. from American National Standards Institute, 1430 Broadway, New York, NY 10018).

4. R. D. Huntoon, "Standard Reference Materials and Meaningful Measurements--An Overview," Proc. Sixth Materials Research Symposium, Gaithersburg, Maryland, October 29, 1973.

5. H. H. Ku, Ed., Precision Measurement and Calibration Vol 1, Statistical Concepts and Procedures. NBS Spec. Pub. 300-I (Superintendent of Documents, U.S. Government Printing Office, Washington, DC, 1969).

6. C. Eisenhart, "Realistic Evaluation of the Precision and Accuracy of Instrument Calibration Systems," J. Res. Nat. Bur. Stand. (U.S.) 67C (2), 161-187 (1963). Also available as Paper 1.2 in Ref. 5.

7. Y. LeDuigou, compiler, "Catalogue of Chemical and Isotopic Reference Materials, 1980," Joint Research Center, Central Bureau for Nuclear Measurements report CBNM/AS/77/80, Geel, Belgium (1980).

8. National Bureau of Standards, "NBS Standard Reference Materials Catalog," NBS Special Publ. 260 (1979).

9. New Brunswick Laboratory, "NBL Reference Materials 1978-1979 Price List."

10. K. F. Lauer and Y. LeDuigou, "Proper Use of Reference Materials for the Accurate Determination of Uranium, Plutonium, and Thorium in the Nuclear Fuel Cycle," in Analytical Methods in the Nuclear Fuel Cycle, Proc. Symp., Vienna, November 29-December 3, 1971 (International Atomic Energy Agency, Vienna, 1972), pp. 145-152.

11. R. J. S. Harry and H. T. Yolken, "Development of Uranium Oxide Reference Materials for Gamma-Ray Measurements of the Enrichment," in Proceedings of the 20th Annual Meeting, Institute of Nuclear Materials Management, Albuquerque, New Mexico, July 16-18, 1979. Nucl. Mat. Manage. VIII, Proceedings Issue, 54-64 (1979).

12. A. M. Voecks and N. M. Trahey, "Preparation of Prototype NDA Reference Materials: A Progress Report," Trans. American Nucl. Soc. Conf. on Measurement Technology for Safeguards and Materials Control, Kiawah Island, Charleston, SC, November 27-30, 1979, pp 2-3.

13. J. E. Rein, S. F. March, G. C. Swanson, G. R. Waterbury, G. L. Tietjen, and R. K. Zeigler, "Preparation of Working Calibration and Test Materials: Plutonium Nitrate Solution," Los Alamos Scientific Laboratory and U.S. Nuclear Regulatory Commission report LA-NUREG-6348 (January 1977).

14. J. E. Rein, G. L. Tietjen, R. K. Zeigler, G. R. Waterbury, and G. C. Swanson, "Preparation of Working Calibration and Test Materials: Plutonium Oxide," Los Alamos Scientific Laboratory and U.S. Nuclear Regulatory Commission report LA-7260, NUREG/CR-0061 (August 1978).

15. J. E. Rein G. L. Tietjen, R. K. Zeigler, and G. R. Waterbury, "Preparation of Working Calibration and Test Materials: Mixed Oxide," Los Alamos Scientific Laboratory and U. S. Nulcear Regulatory Commission report LA-7322, NUREG/CR-0139 (September 1978).

16. S. S. Yamamura, F. W. Spraktes, J. M. Baldwin, R. L. Hand, B. P. Lash, and J. P. Clark, "Preparation of Working Calibration and Test Materials: Uranyl Nitrate Solution," Allied Chemical Corporation, Idaho Chemical Programs and U.S. Nuclear Regulatory Commission report NUREG-0253 (May 1977).

17. J. P. Clark and C. H. Jackson, "Establishing the Traceability of a Uranyl Nitrate Solution to a Standard Reference Material," in Analytical Methods in Safeguards and Accountability Measurements of Special Nuclear Material, H. T. Yolken, J. E. Bullard, Eds., NBS Special Publication 528, (Superintendent of Documents, Washington, DC, November 1978), pp. 133-148.

18. H. T. Yolken, "Possibilities for International Cooperation in Standardizing Measurement Methods for Nuclear Safeguards," in Nuclear Safeguards Technology 1978, Proc. Symp., Vienna, October 2-6, 1978 (International Atomic Energy Agency, Vienna, 1979), IAEA-SM-231/96, Vol. II, pp. 243-250.

INTERNATIONAL TRAINING COURSE ON NUCLEAR MATERIALS ACCOUNTABILITY FOR SAFEGUARDS PURPOSES

SESSION #21: ASSAY/VERIFICATION OF FRESH AND SPENT-FUEL ELEMENTS

SPEAKER: Dr. David M. Lee

Los Alamos Scientific Laboratory
Los Alamos, New Mexico USA

Monday, June 2, 1980
8:30 p.m.

BIOGRAPHY

Education: B.S. physics, Manhattan College, 1966; Ph.D., nuclear physics, Univ. of Virginia, 1971.

Present Position: Staff member, Q Division, International Safeguards.

Present Duties: Design and implement NDA techniques for nuclear safeguards.

Past Positions: Staff member, MP Division, medium energy physics, instrumentation and nuclear physics research and development.

Session Objectives

SESSION #21: ASSAY/VERIFICATION OF FRESH AND SPENT-FUEL ELEMENTS

The basic nature of unirradiated and irradiated fuel elements and their radiological measurement signatures are considered. Emphasis is placed on measurement accuracy and the use of item identification and surveillance techniques.

After the session, participants will be able to

1. Describe the use of gamma-ray enrichment measurements and the Neutron Collar for verifying fresh fuel elements.

2. Describe the various NDA measurement techniques for irradiated fuel assemblies, including gross gamma and neutron profiles and high-resolution gamma spectroscopy.

3. Discuss the basic measurement principles for relating NDA data to fuel quantities.

4. Discuss applicable surveillance techniques.

SESSION 21: ASSAY AND VERIFICATION OF FRESH AND SPENT FUEL

D. M. Lee
Los Alamos Scientific Laboratory

PREFACE

The information contained in this report is a compilation of the information contained in many more detailed reports on the assay and verification of fresh and spent fuel. Many parts of the section on spent fuel verification can be found in the reports LA-6923 by S. T. Hsue, et al., LA-8076-MS by J. R. Phillips, et al., and reference 35 by D. M. Lee, et al. The entire section on containment and surveillance is derived from LA-7730-MS by D. D. Cobb, et al. These reports and the many reports cited in them should be consulted for more detailed work. This report is intended to provide an overview of the measurement problems and techniques that exist.

I. INTRODUCTION

In the nuclear fuel cycle, the verification of the fissile content of fresh and spent fuel has been identified as an important international inspection problem by the IAEA.

Assay or verification of fresh fuel requires the verification of ^{235}U inventory of all rods in an assembly and since the fuel assembly is an integral unit and is not disassembled for the measurement, this implies that the NDA measurement technique is sensitive to all rods in the assembly.

Spent fuel verification is similar in concept to the verification of fresh fuel in that the total fissile inventory (^{235}U and plutonium) must be verified, but it is complicated by the fact that the assembly is now highly radioactive (10^4-10^5 R/h) and submerged in 30-40 feet of water.

The assay or verification of fresh and spent fuel utilizes different NDA techniques. Generally, an assay condition is considered when the absolute fissile inventory is determined directly through NDA measurements. By verification we mean that the fissile inventory is determined indirectly through NDA measurements. Generally, an assay measurement is more precise than a verification measurement. In the following, we will describe various NDA measurement techniques for both fresh and spent fuel and outline the underlying basis for such measurements.

II. FRESH FUEL VERIFICATION

The verification of fresh fuel can be accomplished by measuring one or more of its physical or radiological characteristics. Since fresh fuel is normally stored in accessible locations, the physical characteristics such as size, weight, and identification number are easily verified. Typical physical characteristics of LWR fuel assemblies are shown in Table I.[1]

The single most important radiological signature from fresh fuel is the 186-keV gamma ray from the alpha decay of ^{235}U. This gamma ray is the basis for all enrichment measurements of uranium. The ^{235}U enrichment of the exterior fuel rods can be measured using passive gamma-ray techniques; however, because of absorption problems, the interior rods cannot be measured in this way.

Autoradiographs of x-ray film placed between the fuel pins have been used by the IAEA and Brumback and Perry[2] at ANL (Argonne National Laboratory) to give semiquantitative verification of the enrichment of the interior fuel pins. However,

TABLE I

CHARACTERISTICS OF FRESH LWR FUEL ASSEMBLIES

	BWR	PWR
Overall assembly length, m	4.470	4.059
Cross section, cm	13.9 x 13.9	21.4 x 21.4
Fuel-element length, m	4.064	3.851
Active fuel height, m	3.759	3.658
Fuel-element OD, cm	1.252	1.07
Fuel-element array	8 x 8	15 x 15
Assembly total weight, kg	275.7	657.9
Uranium/assembly, kg	183.3	461.4
UO_2/assembly, kg	208.0	523.4
Zircaloy/assembly, kg	57.9[a]	108.4[b]
Hardware/assembly, kg	9.77[c]	26.1[d]
Total metal/assembly, kg	67.7	134.5
Nominal volume/assembly, m^3	0.0864[e]	0.186[e]

[a]Includes Zircaloy fuel-element spacers.
[b]Includes Zircaloy control-rod guide thimbles.
[c]Includes stainless steel tie-plates and Inconel springs.
[d]Includes 10 kg stainless steel nozzles and Inconel-718 grids.
[e]Based on overall outside dimension.

the exposure times for the measurements are 3-4 h for BWR and PWR fuel assemblies and the enrichment sensitivity was 3±1%. Since LWR fuel assemblies have 2-4% enrichment, the enrichment cannot at present be reliably detected by autoradiography.

During the past few years, an active neutron technique (the neutron collar)[3] to verify ^{235}U content by neutron interrogation and fast neutron counting using ^{4}He detectors has been developed.

The neutron collar is shown in Fig. 1. The principal components of the assay system are the polyethylene main frame, the interrogating neutron source, the ^{4}He neutron detectors, and electronic counting system (not shown). The interrogating source is an AmLi (α,n) source at a source strength of 5×10^5 n/s. Neutrons from the source are thermalized in the polyethylene and induce fissions in the fuel assembly. The neutrons from the induced fissions are detected in the ^{4}He neutron detectors. This technique is termed subthreshold interrogation because the interrogating thermal neutrons are below the detection threshold of the neutron detectors so that the detectors are insensitive to the interrogating source.

This approach has limited sensitivity in the interior regions of PWR (pressurized water reactor) fuel assemblies because of thermal neutron penetrability problems and geometric considerations. Recent advances with the AWCC[4] Active Well Coincidence Counter) have made it possible to apply this same technical approach to the verification of full LWR fuel assemblies. The method involves neutron interrogation with an AmLi neutron source and coincidence counting the induced fission reaction neutrons from the ^{235}U. The coincidence counting separates the fission neutrons, which originate from ^{235}U, from the random neutrons used in the interrogation. This "coincidence collar" approach has the following advantages over the previously developed neutron collar.

1. The AmLi neutron source strength requirement is 10 times smaller, reducing transportation and handling problems. The neutron collar uses 5×10^4 n/s sources.
2. The sensitivity to the removal of interior fuel pins in an assembly is at least 2 times better with the coincidence collar. The sensitivity at a 95% confidence level for rod removal in the center of the fuel assembly for the neutron collar is 7.7 rods or 3.4 percent of the rods in a 15x15-rod fuel assembly, while for the coincidence collar it is better than 4 rods or 1.7 percent.

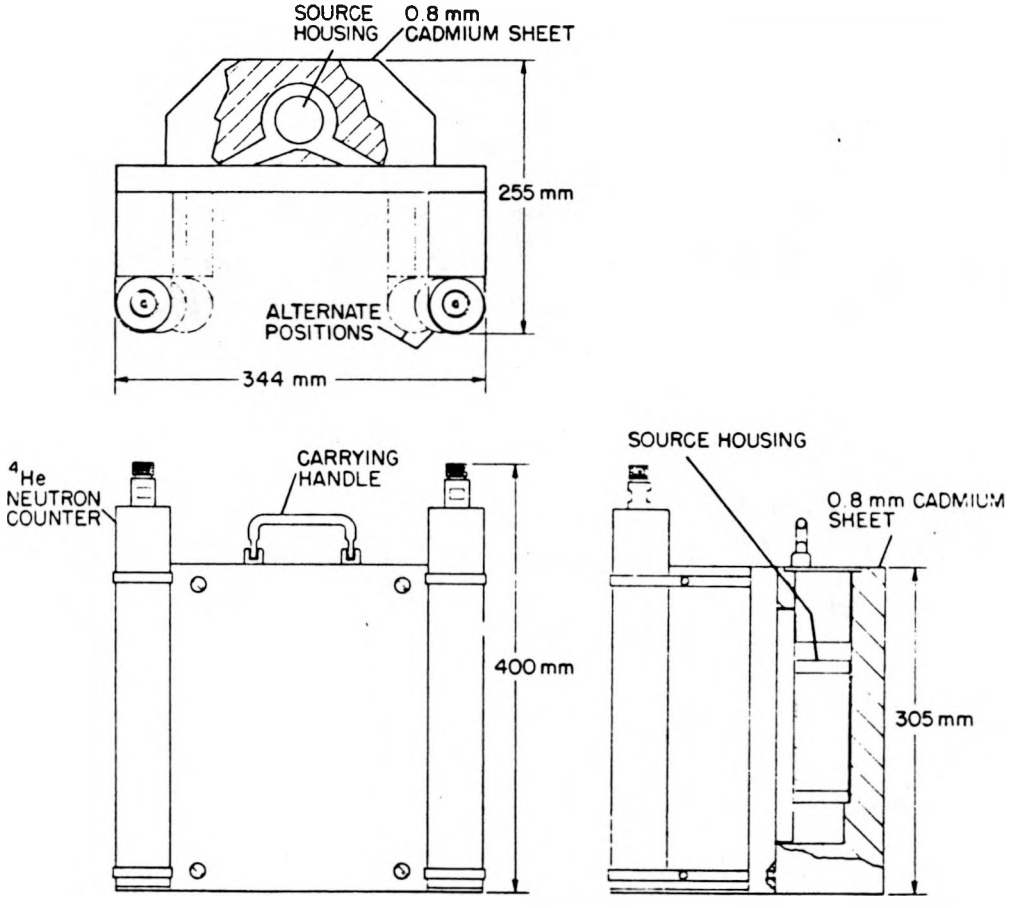


Fig. 1.
Drawing of the neutron collar showing the assembly details and source.

3. In addition to verifying the ^{235}U in the active interrogation mode, the coincidence collar can verify ^{238}U and ^{234}U in the passive mode.

III. SPENT FUEL VERIFICATION

The principal goal of assaying spent fuel is to determine the burnup and/or fissile content. Passive and active techniques have been employed in the past, with the passive methods the most predominantly used, although active neutron interrogation methods are the only possible means of assaying the fissile content at the present time. Passive methods do not yield the fissile content directly, but it may be inferred from a burnup calculation or from an empirically determined correlation. Two common definitions of burnup are:[5] (1) Burnup is the number of fissions per 100 heavy nuclides initially present in fuel, and (2) burnup is the integrated energy released from the fission of heavy nuclides initially present in fuel.

The first definition is used for dissolved irradiated fuel, and concentrations of a selected fission-product burnup monitor and the heavy nuclide atoms are determined.

The burnup computational relationship is then

$$BU\ (\%) = 100\ \frac{P/Y}{P/Y + H}\ , \qquad (1)$$

in which

BU(%) = percent fission

P = atom concentration of fission-produce burnup monitor,

Y = effective fractional fission-yield value, and

H = final atom concentration of heavy nuclides (mass 232).

The second definition is used mainly for power reactors, where burnup is expressed in MWD/MTU (megawatt days per metric ton uranium initial). In NDA of spent fuel, this definition is used because heavy nuclides are not determined.

The burnup relationship is

$$\text{Number of fission/MTU} = N/Y \ . \tag{2}$$

$$\text{BU (MWd/MTU)} = 1.8563 \times 10^{-24} \times E \times N/Y \ , \tag{3}$$

in which

N = number of atoms of burnup monitor formed during irradiation per metric ton of initial heavy metal,

Y = effective fractional fission-yield value, and

E = effective energy released per fission in MeV.

$$Y = \frac{\sum_f^{235} \ Y^{235} + \sum_f^{238} \ Y^{238} + \sum_f^{239} \ Y^{239} + \sum_f^{241} \ Y^{241}}{\sum_f^{235} + \sum_f^{238} + \sum_f^{239} + \sum_f^{241}} \ ,$$

and

$$E = \frac{\sum_f^{235} \ E^{235} + \sum_f^{238} \ E^{238} + \sum_f^{239} \ E^{239} + \sum_f^{241} \ E^{241}}{\sum_f^{235} + \sum_f^{238} + \sum_f^{239} + \sum_f^{241}} \ ,$$

in which

$\sum_f^{(I)}$ = time-averaged macroscopic fission cross section of I = ^{235}U, ^{238}U, ^{239}Pu, and ^{241}Pu,

$Y^{(I)}$ = fission yield of isotope I, and

$E^{(I)}$ = energy released per fission in MeV of isotope I.

The two burnup expressions are related by the conversion

$$\text{BU (MWd/MTU)} = 46.977 \times E \times \text{BU(\%)} \ . \tag{4}$$

Both the effective fission yield and the energy released per fission depend on the relative fission contribution from plutonium and uranium.

Assuming an effective energy release of 202 MeV per fission, a burnup of 1% corresponds to a burnup of 9489 MWd/MTU.

The passive NDA of spent fuel has basically involved the measurement of one or more radioactive signatures from the spent fuel and correlating these with burnup and then through the above relationships relating the burnup to fissile content.

A. Measureable Characteristics of Spent Fuel

The verification of a spent fuel assembly requires the measurement or observation of one or more of the characteristics of the spent fuel. These characteristics can be divided into (1) physical characteristics such as the serial number, weight, and physical appearance, and (2) nuclear characteristics such as neutron emission, gamma emission, and axial activity profile.

1. Physical Characteristics

a) Serial Number. Spent fuel assemblies are discrete units, suitable to item counting, and usually have a unique identification number permanently attached to the top supporting structure. An example of an identification number is shown in Fig. 2 for a PWR assembly and Fig. 3 for a BWR assembly.

b. Weight. All PWR fuel assemblies of the same manufacturer should weight the same. An 8x8 ASEA-ATOM BWR 75 fuel assembly weights 305 kg for example and a 15x15 Babcock and Wilcox BWR fuel assembly weights 704.5 kg.

c. Physical Appearance. Irradiation exposure changes the appearance and color of the assembly. The surface of an individual fuel rod is often covered by a thin layer of corrosion deposits and rub marks where the deposits have been removed. The colors of rods in the assembly may vary from shiny gold and brown to reddish brown or dull gray.

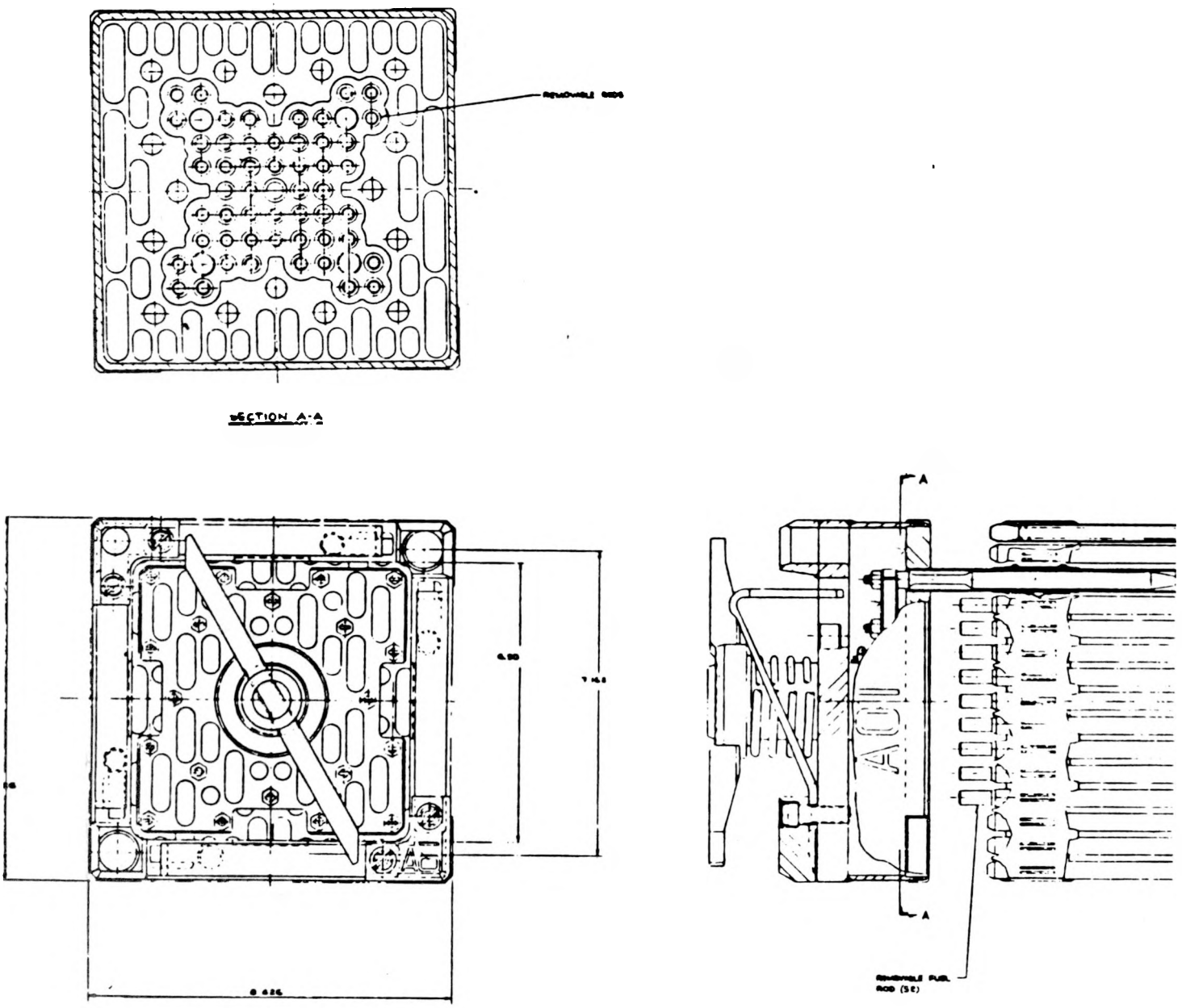


Fig. 2.
A PWR fuel assembly schematic showing a typical identification number.

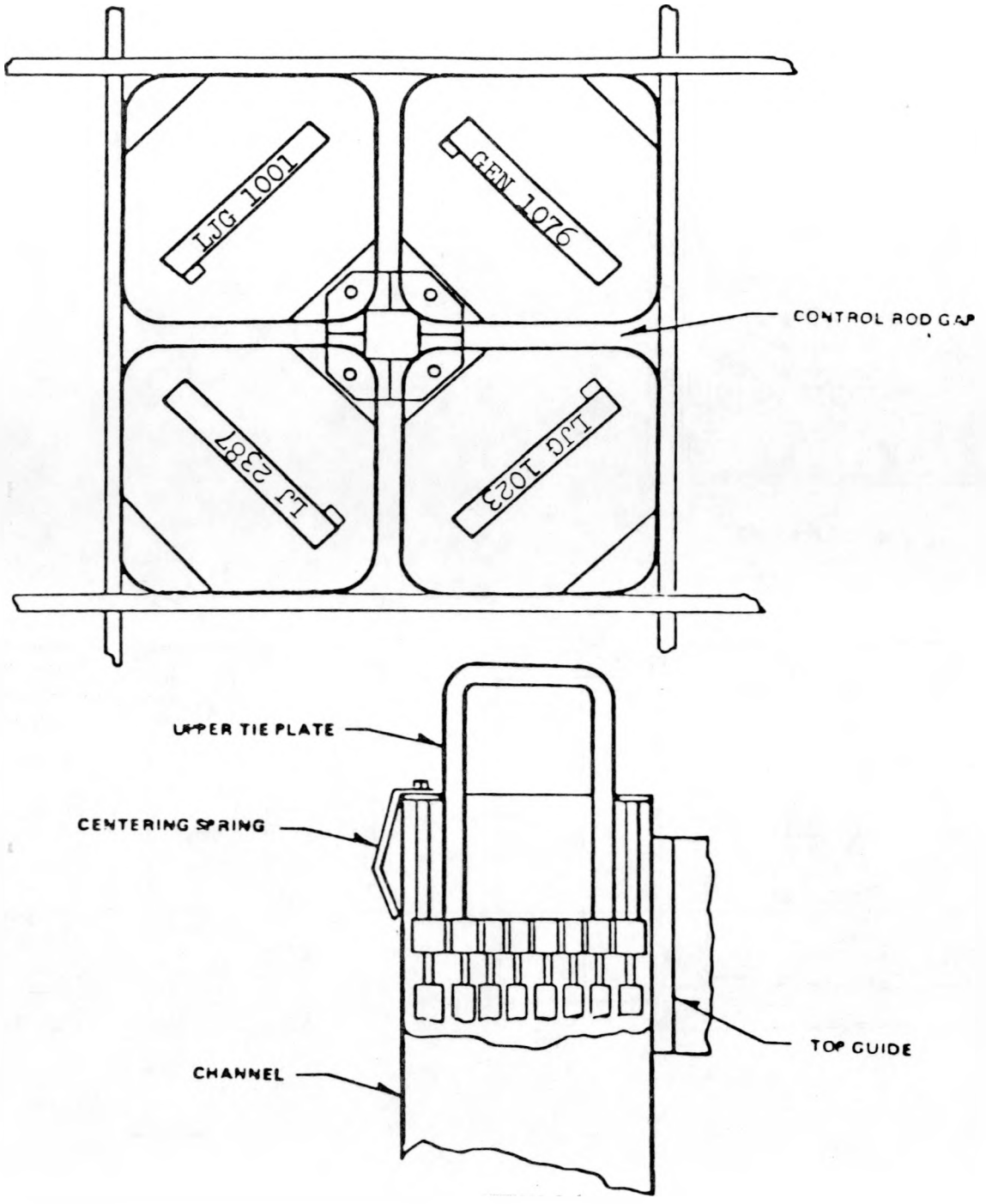


Fig. 3.
A BWR fuel assembly schematic showing a typical identification number.

A visual inspection of these physical characteristics can provide a low level of verification even though a quantitative measure of burnup has not been done. For a more quantitative measure of burnup one or more of the nuclear characteristics must be measured.

2. Nuclear Characteristics

a. Neutron Emissions. Neutrons from spent fuel assemblies arise from either spontaneous fission or from (α,n) reactions. The even isotopes of plutonium and curium undergo spontaneous fission. The (α,n) neutrons result from reactions of alpha particles from the radioactive decay of plutonium, americium, and curium with oxygen in the matrix. The neutron yield is a function of the alpha particle energy, the (α,n) cross sections of the matrix elements, and the matrix configuration. In a spent fuel assembly, the neutron emission rate depends strongly on the quantity of curium present. The quantity of ^{242}Cm (162.8 day half-life) is particularly important for relatively short cooling times. A list of the principal sources of neutrons from irradiated fuel is given in Table I.[6] Typical neutron emission rates from spent fuel assemblies range from 10^3-10^5 $n/cm^2/s$ depending on the burnup. It is not possible to identify the transuranic isotopes from the neutron emissions.

b. Gamma-Ray Emissions. The gamma-ray emission from spent fuel is primarily from the radioactivity decay of fission products. The gamma dose rate at the surface of an LWR assembly with 20 000-30 000 MWd/MTU burnup can reach levels of 10^4-10^5 R/h. Of the more than 800 fission products contained in spent fuel assemblies, only a few produce radioactive signatures which can be used to characterize spent fuel. Table II lists those measureable fission product isotopes that have half-lives greater than 30 days.[6]

TABLE I

PRINCIPAL SOURCES OF NEUTRONS IN IRRADIATED UO_2 MATERIALS[6]

Isotope	Half-lives (yr)	Neutrons Produced per gram-second: (α,n) Reaction	Spontaneous Fission	Total
^{235}U	$7.038 \pm 0.005 \times 10^{8}$	$7.21 \pm 0.72 \times 10^{-4}$	$3.86 \pm 0.99 \times 10^{-4}$	$1.11 \pm 0.12 \times 10^{-3}$
^{238}U	$4.4683 \pm 0.0024 \times 10^{9}$	$8.43 \pm 0.84 \times 10^{-5}$	$1.36 \pm 0.02 \times 10^{-2}$	$1.36 \pm 0.02 \times 10^{-2}$
^{238}Pu	87.71 ± 0.03	$1.56 \pm 0.16 \times 10^{4}$	$2.60 \pm 0.11 \times 10^{3}$	$1.82 \pm 0.16 \times 10^{4}$
^{239}Pu	$2.4131 \pm 0.0016 \times 10^{4}$	$4.25 \pm 0.43 \times 10^{1}$		$4.25 \pm 0.43 \times 10^{1}$
^{240}Pu	$6.570 \pm 0.006 \times 10^{3}$	$1.56 \pm 0.16 \times 10^{2}$	$8.85 \pm 0.10 \times 10^{2}$	$1.04 \pm 0.19 \times 10^{3}$
^{242}Pu	$3.763 \pm 0.009 \times 10^{5}$	2.27 ± 0.23	$1.743 \pm 0.015 \times 10^{3}$	$1.743 \pm 0.015 \times 10^{3}$
^{241}Am	432.0 ± 0.2	$3.17 \pm 0.32 \times 10^{3}$		$3.17 \pm 0.32 \times 10^{3}$
^{242}Cm	0.4456 ± 0.0001	$4.48 \pm 0.45 \times 10^{6}$	$2.25 \pm 0.08 \times 10^{7}$	$2.70 \pm 0.09 \times 10^{7}$
^{244}Cm	18.099 ± 0.015	$8.82 \pm 0.88 \times 10^{4}$	$1.081 \pm 0.007 \times 10^{7}$	$1.090 \pm 0.007 \times 10^{7}$

$\bar{\nu}_{sp} = 3.756$ for ^{252}Cf.

TABLE II

MEASURABLE FISSION PRODUCTS IN LWR FUEL ASSEMBLIES

Isotope	Half-life	Principal Gamma Rays (keV)
^{95}Nb	34.97 ± 0.03 days	765.8 (99.82%)[a]
^{103}Ru	39.35 ± 0.05 days	49.71 (86.4%), 610.3 (5.4%)
^{95}Zr	63.98 ± 0.06 days	724.2 (43.1%), 756.7 (54.6%)
^{144}Ce	284.5 ± 1.0 days	696.4 (1.34%), 1489.2 (0.26%), 2185.6 (0.66%) gamma rays from ^{144}Pr ($t_{\frac{1}{2}}$ = 17.3m) daughter
^{106}Ru	366.4 days	622.2 (9.8%), 1050.5 (1.6%), 1562.2 (0.17%) gamma rays from ^{106}Rh ($t_{\frac{1}{2}}$ = 29.8s) daughter
^{134}Cs	2.062 ± 0.005 yr	604.7 (97.6%), 795.8 (85.4%), 801.8 (8.7%), 1038.5 (1.00%), 1167.9 (1.81%), 1365.1 (3.04%)
^{154}Eu	8.5 ± 0.5 yr	996.3 (10.3%), 1004.8 (17.4%), 1274.4 (35.5%)
^{137}Cs	30.17 ± 0.03 yr	661.6 (89.9%) gamma ray from ^{137m}Ba ($t_{\frac{1}{2}}$ = 2.55m) daughter

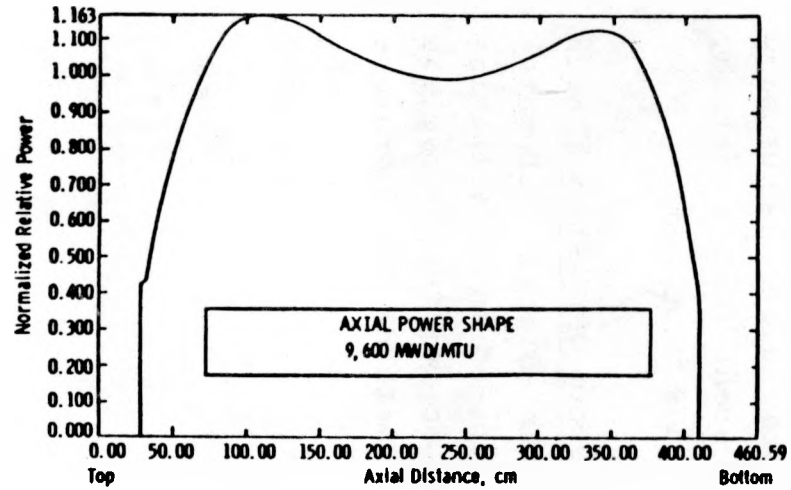
[a] Values in parenthesis are the branching ratio of the specific gamma ray.

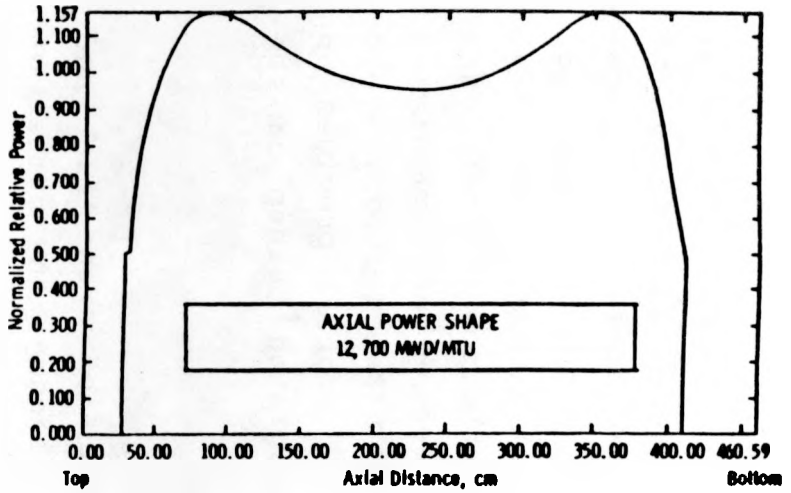
c. Axial Activity Profile. The activity profile of a spent fuel assembly depends on the type of reactor, i.e., BWR, PWR, or MTR, burnup, and the reactor operating conditions. Some calculated PWR profiles are shown in Fig. 4 where it can be seen that the shape of the power profile changes with burnup. The activity profile will reflect these changes in the power profile and therefore the activity profile can be used as an identifying characteristic. In addition, since burnup calculations yield values for the entire assembly, the axial profile provides an integrating function from which the integrated burnup can be obtained.

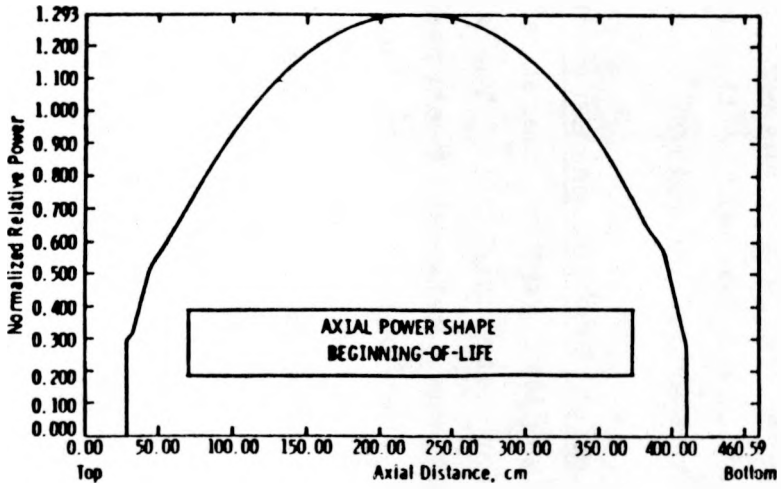
B. NDA Measurement Techniques

To measure the various characteristics of spent fuel and achieve a high level of verification confidence, any number of NDA techniques can be employed. These techniques include the use of high resolution gamma-ray spectroscopy, ion chambers, neutron detectors, Be(γ,n) or gamma-specific detectors, profile monitors, Cerenkov glow detection system, and active interrogation systems. All of these detection systems have certain advantages and disadvantages and one system alone is usually not sufficient to verify the integrity of spent fuel. Instead, a combination of detection systems is normally required.

1. High Resolution Gamma Ray Spectroscopy (HRGS) has been widely applied to the characterization of irradiated fuel materials for safeguards.[1,7] The technique is based on establishing correlations between measured isotopic activities (absolute activity method) or ratios (activity ratio method) and operator-declared values. These correlations are then used to verify the operator-declared values. Values that have been verified include cooling times, burnup, and Pu/U ratio.[8]







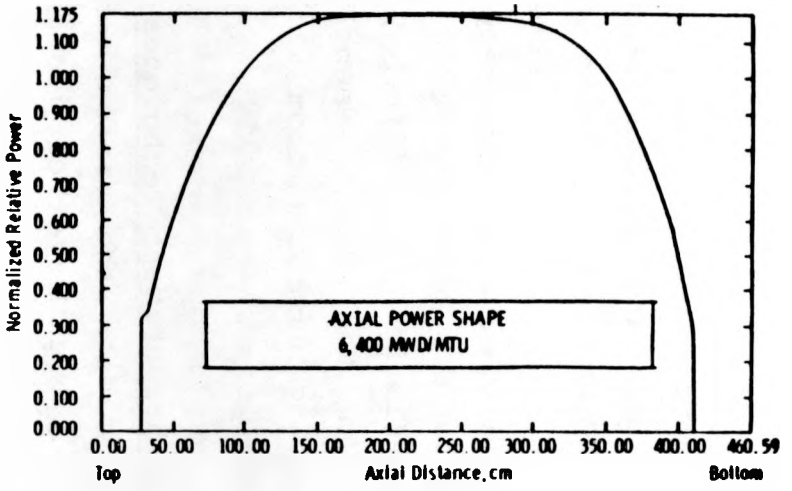


Fig. 4.
Calculated axial power profiles for different burnups.

Correct interpretation of HRGS results depends on understanding several limitations of the technique. In an LWR fuel assembly, only the outer regions can be examined because the inner rods are self-shielded. For example, for the relatively high-energy gamma ray of ^{140}La (1596 keV), the intensities of the centermost rods in a 15 x 15 PWR fuel assembly are attenuated by a factor of nearly 15 compared to the outer row of rods.[9] Similar effects have been calculated for other power reactor fuels.[10]

Fission products may not be uniformly distributed axially and radially within a fuel assembly. The local (pin-to-pin) power distribution can vary because of the location of burnable poison rods and control rods, their location in the core, and the initial enrichment and configuration of the assembly. Another factor is the possible axial and radial migration of fission products within the individual fuel pins.[11] Axial migration can adversely affect the results if the entire fuel assembly is not scanned. The radial migration can affect the relative intrinsic efficiency corrections because different isotopes migrate to different radial locations. If an intrinsic efficiency calibration is used, then all the isotopes are assumed to be distributed identically within the assembly. These limitations must be recognized in evaluating the accuracy and the usefulness of the spectral data obtained from HRGS.

The two NDA methods of gamma-ray assay are the absolute gamma activity measurement and the activity ratio measurement.

2. Absolute Activity Measurement.[5] When the number N of burnup monitor atoms formed during irradiation is determined, the burnup can be calculated by Eq. (3). N is related to the gamma-ray intensities by

$$N = \frac{X_i}{k_i \varepsilon_i S_i} \quad \frac{e^{\lambda T_c}}{\lambda} \quad , \tag{5}$$

in which

X_1 = number in the ith gamma-ray peak observed per unit time,
k_1 = number of the ith gamma ray per disintegration,
ε_1 = absolute detector efficiency at the energy of the gamma-ray peak,
S_1 = the effective attenuation at the energy of the gamma-ray peak,
λ = decay constant, and
T_c = cooling time.

In Eq. (5), the k_1 and λ depend on the status of the nuclear data; and for these radioactive fission-product monitors, they are generally known to approximately 1 to 2%. The X_1 can be measured to 1% or better. The attenuation factor S_1 may be determined fairly accurately for a single rod if diametral rod scan is performed; the uncertainty in S_1 is considerably larger for a whole assembly. The most crucial and difficult factor to determine is the absolute detector efficiency ε_1, which depends on the measurement geometry, the collimator, and the intrinsic efficiency of the detector. The absolute efficiency may be determined by a calibration source of known total activity. The geometries for assay and efficiency calibration must be identical.

3. Activity Ratio Measurement.[5] Activity ratios have been suggested[12,23] for use as burnup monitors. Several experiments since 1971 have explored this possibility and development is continuing. If we assume that the flux is constant during irradiation, the activity from a direct fission product N_D formed (such as ^{137}Cs), and that from neutron capture of fission products N_1 (such as ^{134}Cs) are respectively proportional to

$$N_D \propto \Sigma_f \cdot (\phi T) \text{ and} \tag{6}$$

$$N_1 \propto \Sigma_f \cdot \sigma(n,\gamma) \cdot (\phi T)^2 \tag{7}$$

in which

ϕ = spectrum and time-averaged neutron flux,

Σ_f = spectrum and time-averaged fission cross section,

$\sigma(n,\gamma)$ = spectrum and time-averaged neutron-capture cross section of the fission product, and

T = irradiation time.

Equations (6) and (7) show that the N_1/N_D ratio is also proportional to (ϕT) and, in principle, can be used as a burnup monitor. The activity ratios that have been explored up to now are $^{134}Cs/^{137}Cs$ and $^{154}Eu/^{137}Cs$.

For an absolute activity measurement, absolute detector efficiency must be known, and the measurement must be performed under strictly controlled geometry. The activity ratio measurement, however, is less sensitive to the geometrical arrangement and requires only that the relative detector efficiency be known. This distinct advantage makes the activity ratio measurement much more suitable for use in field inspection. However, a disadvantage is that effective fission yields of the activity ratios are not well known. Thus, to deduce burnup, correlations between burnup and activity ratios must be determined empirically.

A recent improvement in the activity ratio measurement[14-16] is to introduce an intrinsic calibration, where measured intensity ratios for gamma rays of a given isotope are compared with established branching ratios. Using these data, an overall relative efficiency curve that includes mass attenuation and detector efficiency can be determined. This curve can be used to determine the activity ratios of two different isotopes. The advantage of this method is its simplicity; all necessary information to determine the activity ratio (relative efficiency and cooling time) is contained in a single gamma-spectrum measurement. However, this method relies on a fundamental assumption that may or may not be valid in an actual situation. It is accurate only if the measured isotopes (^{134}Cs, ^{137}Cs, ^{154}Eu) have the same spatial distribution within the assembly.

Figure 5 shows a typical HRGS system in which the fuel assembly is moved vertically past the collimator, and complete spectra (300-2200 keV) are being recorded at specified axial positions. The spectral data are compared with the operator-declared values to establish correlations that can be used to predict burnup from measured parameters. If only one HRGS measurement is obtained, then an integrating function is required to relate the measurement to the entire fuel assembly. Several applicable techniques for obtaining the axial profile will be discussed later.

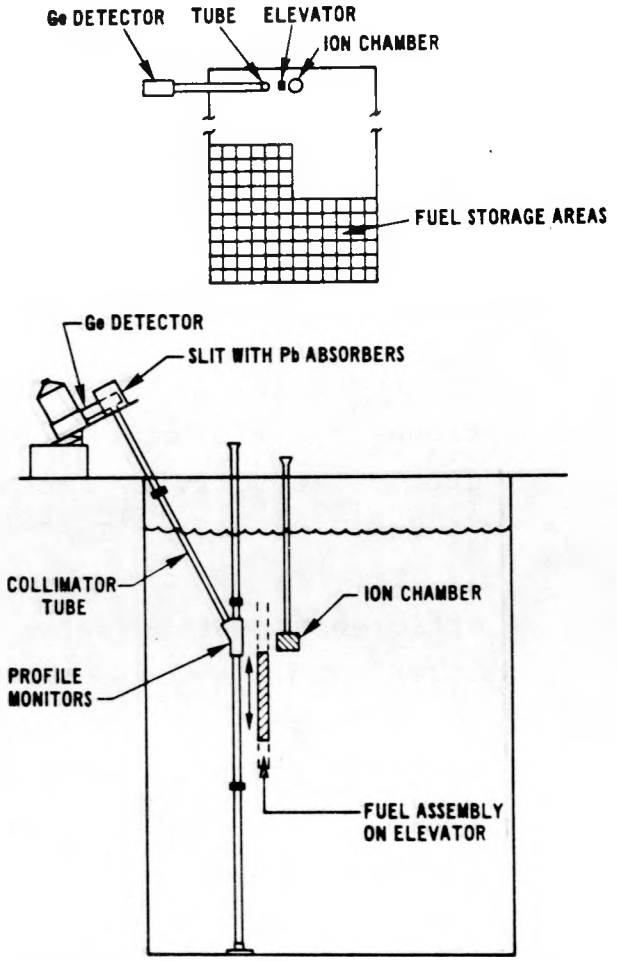


Fig. 5.
Typical spent-fuel examination system in a reactor storage pool.

The kinds of correlations obtainable using HRGS are shown in Figs. 6-8, where the measured parameters ^{137}Cs, $^{134}Cs/^{137}Cs$, and $^{154}Eu/^{137}Cs$ are plotted with operator-declared burnup values for a set of 14 PWR assemblies. The data have been corrected for decay since discharge using the operator-declared discharge date. Each plot has the 95% confidence bounds plotted. For the ^{137}Cs isotope, the average deviation between the regression line and the given burnup values was 5.1%. For the isotopic ratios $^{134}Cs/^{137}Cs$ and $^{154}Eu/^{137}Cs$, the average deviations were 6.5% and 7.8%, respectively. These results were based on single measurements at the centers of the fuel assemblies. Although a linear relationship between burnup and the measured parameters was assumed, there is some evidence that the relationship can be nonlinear.[17,18]

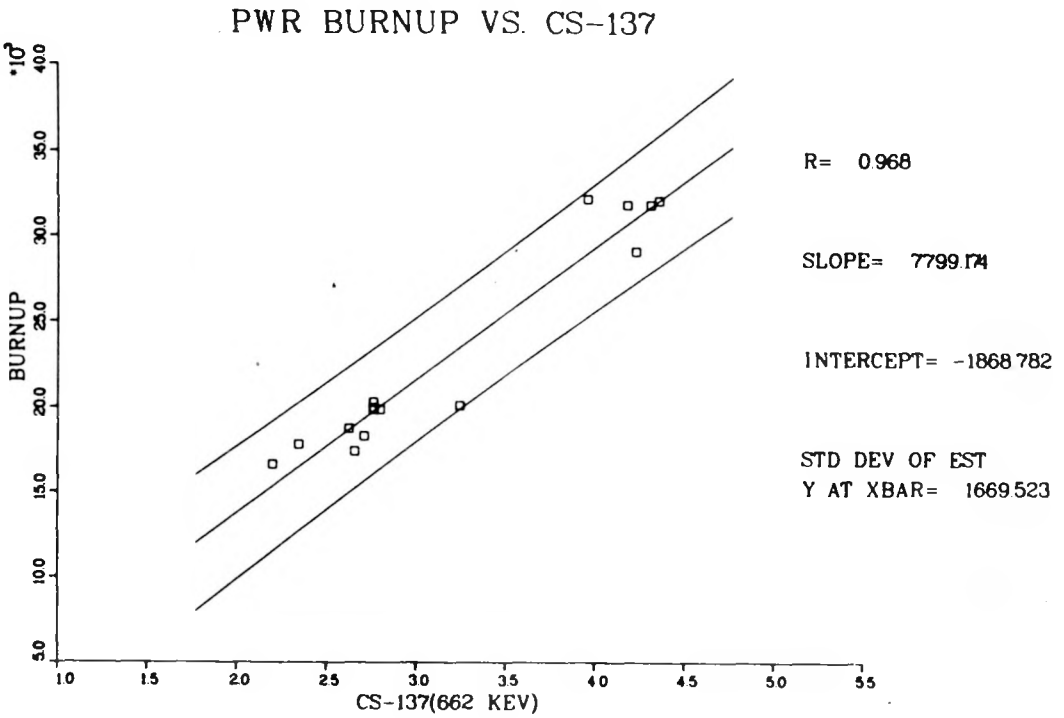


Fig. 6.
^{137}Cs activity with respect to operator-declared burnup values with the 95% confidence bounds and the best fit regression line.

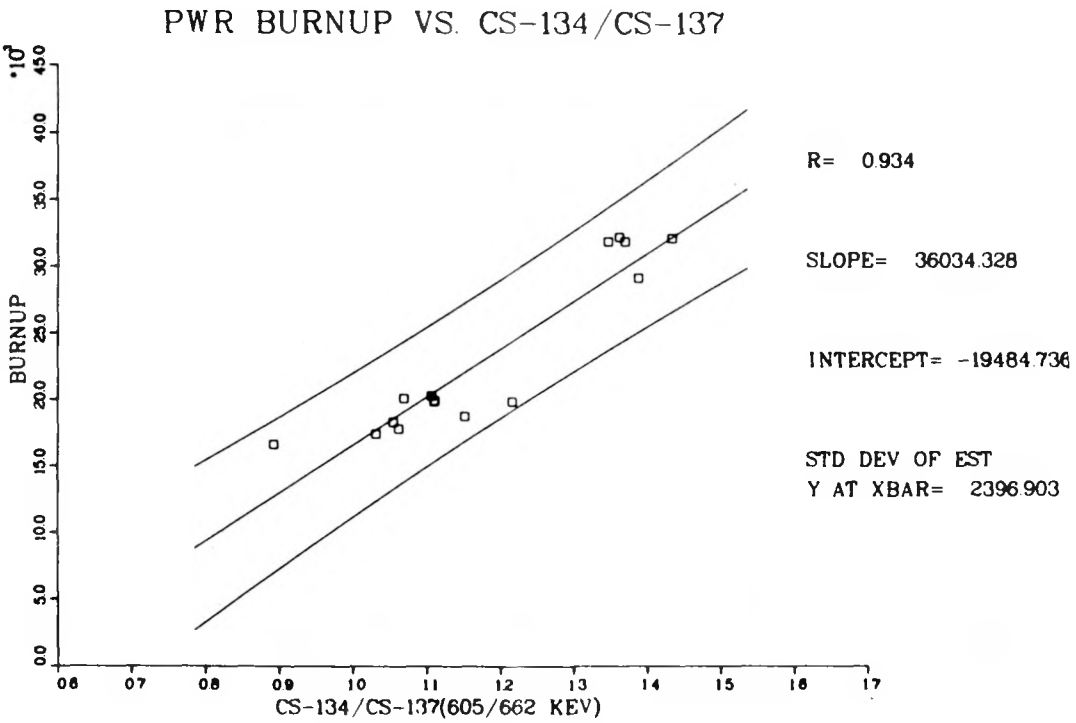


Fig. 7.
$^{134}Cs/^{137}Cs$ isotopic ratios with respect to operator-declared burnup values with the 95% confidence bounds and the best fit regression line.

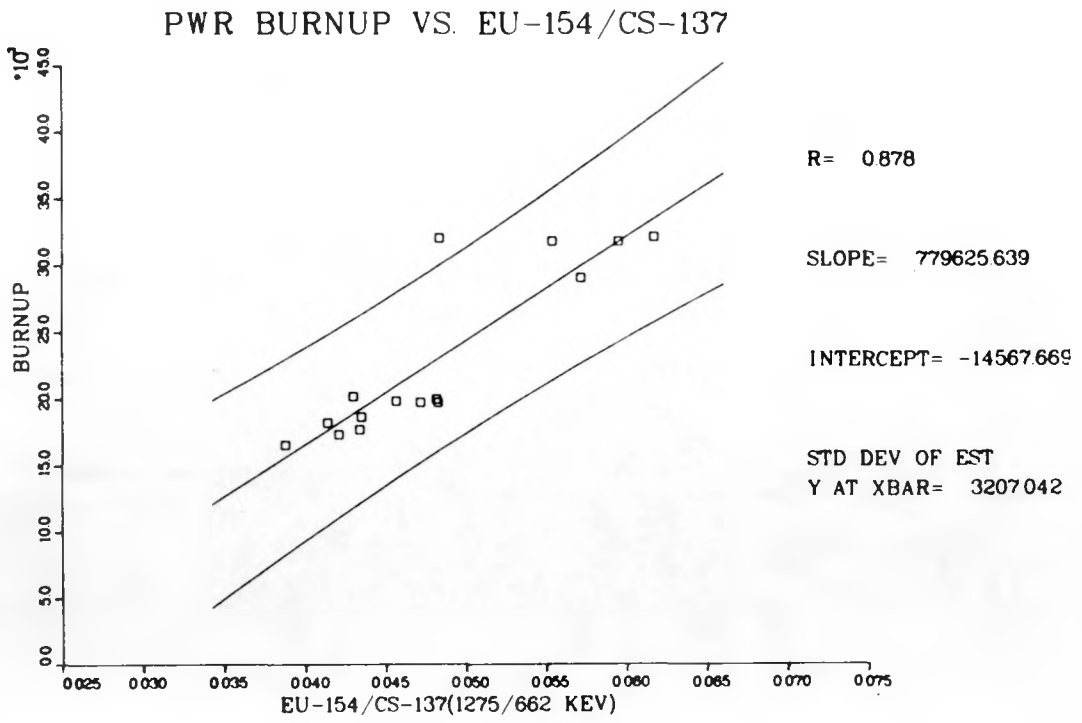


Fig. 8.
$^{154}Eu/^{137}Cs$ isotopic ratios with respect to operator-declared burnup values with the 95% confidence bounds and the best fit regression line.

In typical LWR fuel assemblies, precisions in the range of 5-10% can be obtained for relative burnup values using HRGS.

IV. GROSS NEUTRON AND ION CHAMBERS

A. Quantitative Measurements

HRGS normally requires that the spent fuel assembly to be examined be isolated from adjacent fuel assemblies. Such isolation is not only time consuming, but not always acceptable to the facility operators or safeguards inspectors. In addition, the collimator assembly, scanning system (either mechanical or multielement), and locating system are expensive, cumbersome and not easily and quickly assembled at each reactor site.

A simpler and perhaps less detailed measurement can be made by measuring the gross passive neutron and gross gamma-ray yields.

Passive neutron assay has been identified as a potentially useful inspection assay method of spent fuel in a recent review[5,19] and in the IAEA Advisory Group Meeting.[20] Two aspects make the passive neutron measurement technique particularly attractive for the measurement of spent fuel assemblies. First, neutrons are less subject to self-absorption in the fuel assembly than are gamma rays. Monte Carlo calculations have shown the interior rods of PWR assemblies contribute nearly the same amount to the total neutron emission rate as the exterior rods.[21] Therefore the neutron measurement is more sensitive to all the interior pins in the fuel assembly than is the gamma-ray spectrometry measurement. Secondly, the passive neutron measurement requires very simple electronics and a neutron detector and this simplicity can be a distinct advantage in the hostile environment of a spent fuel storage facility.

The dominant sources of neutrons from the irradiated fuel assemblies are the spontaneous fissioning of the actinide isotopes and the (α,n) reactions on light materials.

The principal isotopes that contribute to the neutron yield of irradiated UO_2 are listed in Table II. The relative contributions of specific isotopes as a function of cooling time for different burnups are shown in Figs. 9 and 10 for a BWR assembly,[9] and in Figs. 11 and 12 for a PWR assembly. The data for these plots were calculated by applying the decay factor to results from a destructive analysis.[22]

All four plots show that for cooling times up to approximately 3 yr, ^{242}Cm (half-life = 163 days) contributes significantly to the total neutron emission rate. For longer cooling times, ^{244}Cm and ^{240}Pu become the dominant contributors. The amount of ^{242}Cm in the irradiated fuel is a sensitive function of the irradiation history, with the ^{241}Pu isotope being

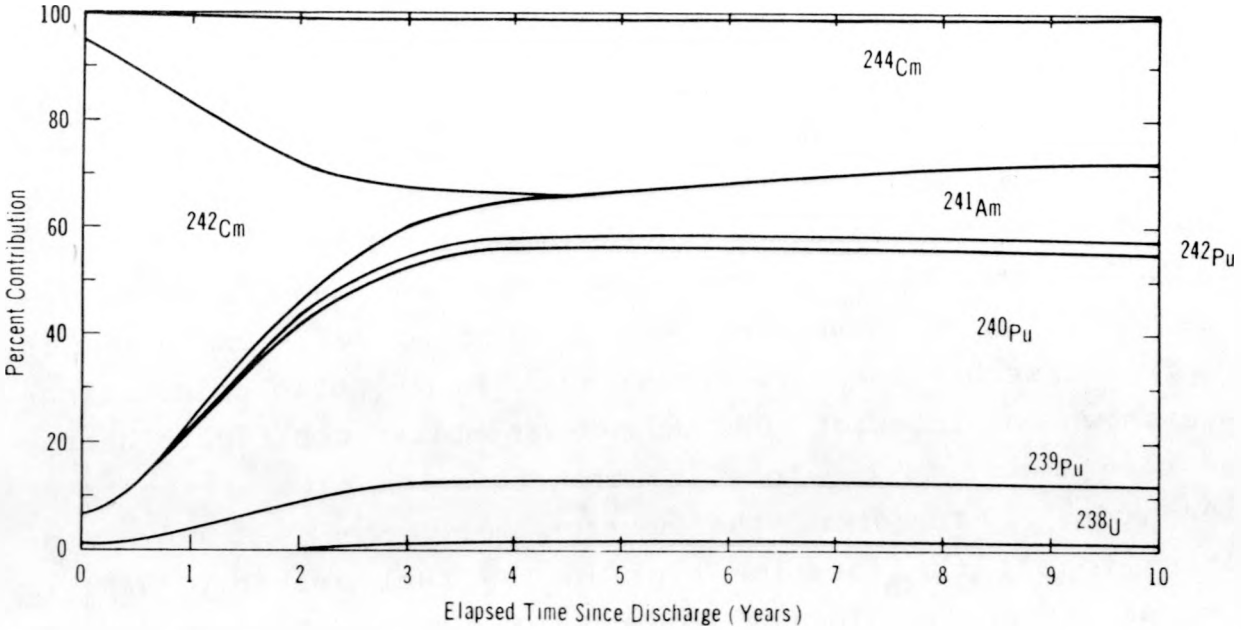


Fig. 9.
Relative percent contribution to the total neutron rate by specific actinide isotopes for a BWR assembly with 7400 MWd/MTU burnup.

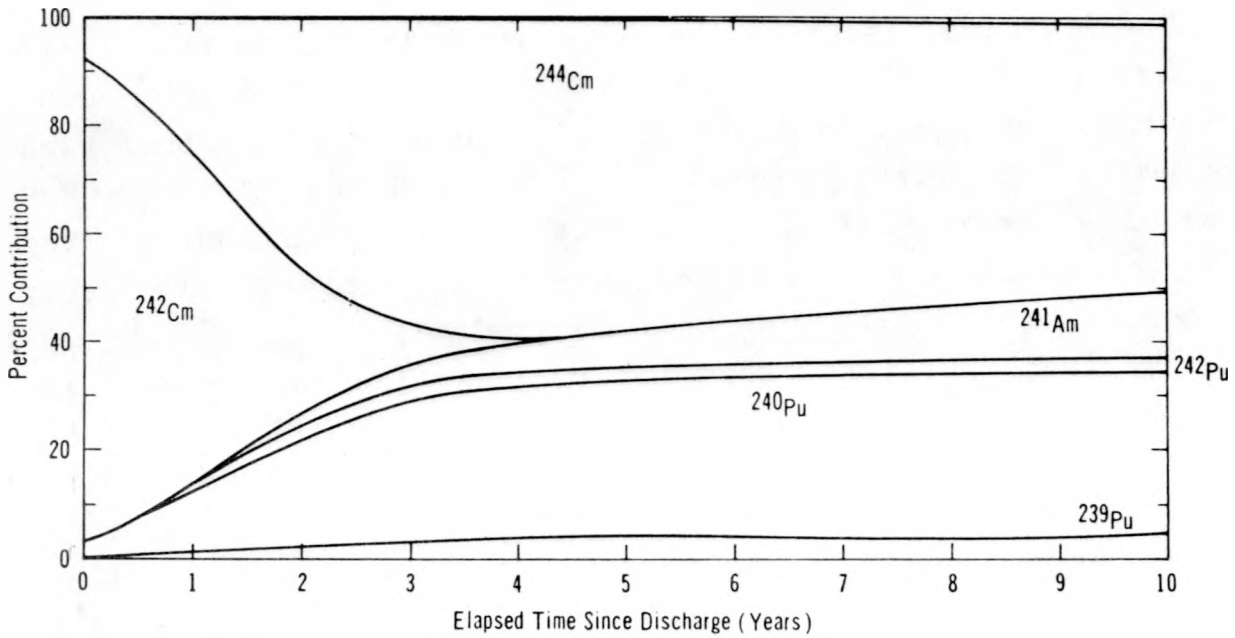


Fig. 10.
Relative percent contribution to the total neutron rate by specific actinite isotopes for a BWR assembly with 11 450 MWd/MTU burnup.

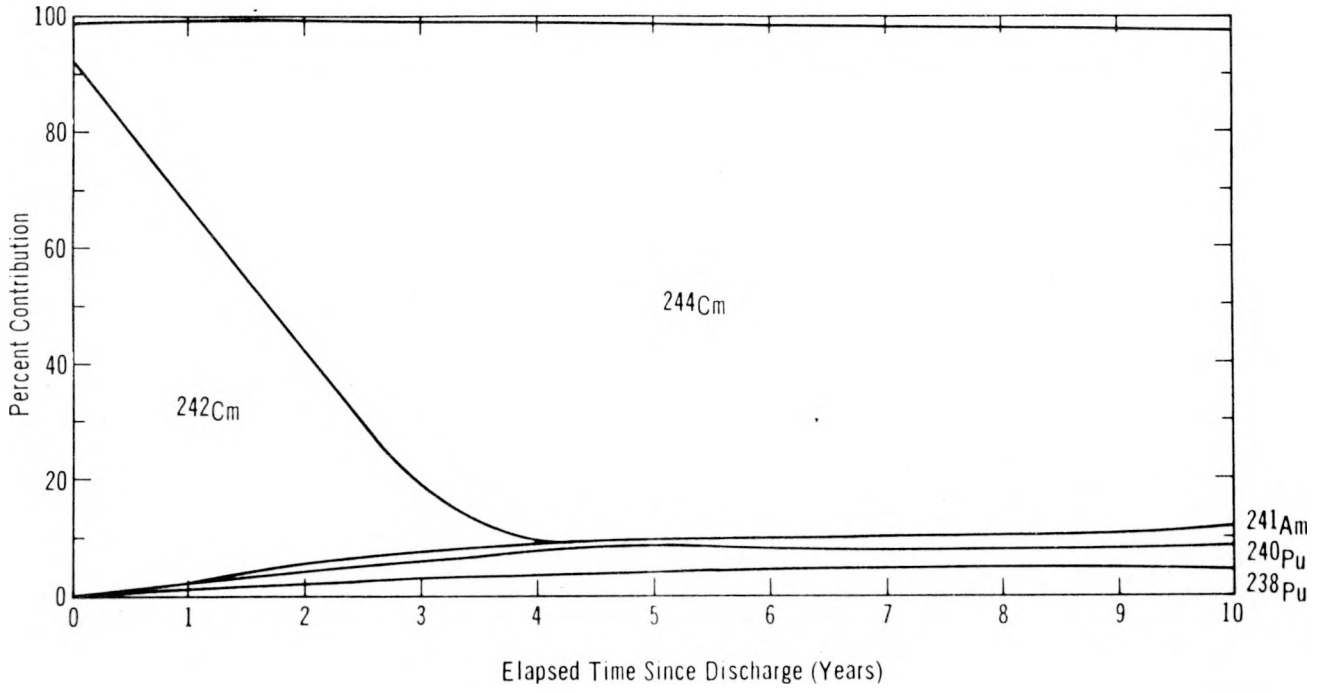


Fig. 11.
Relative percent contribution to the total neutron rate by specific actinide isotopes for a PWR assembly with 20 060 MWd/MTU burnup.

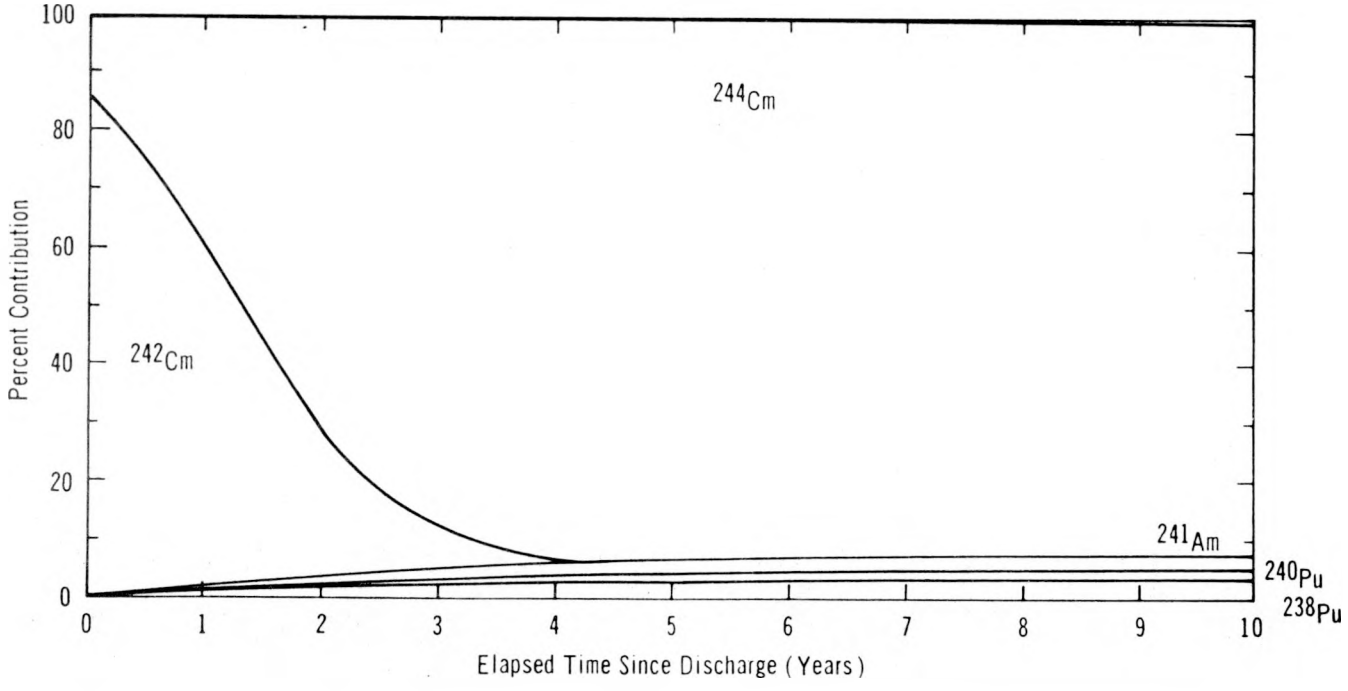


Fig. 12.
Relative percent contribution to the total neutron rate by specific actinide isotopes for a PWR assembly with 25 095 MWd/MTU burnup.

the critical precursor. Direct measurement of the fissile content of fuel assemblies is not possible using passive neutron techniques. One must infer the fissile content from experimental correlations of passive neutron measurements or calculational techniques.[6]

The quantitative gross-gamma measurement of spent fuel can be accomplished either through integral pulse counting techniques, NAI detectors, cadmium telluride and germanium detectors or through current measuring techniques. All measurements are basically sensitive to the same quantity, the dose level of the spent fuel assembly and by far the simplest and most reliable method of measuring the dose level is through the use of ionization chambers operating in the current mode. Although the dose measurement is not able to differentiate the presence of specific fission products, it does make use of the high radiation field that must be present in a spent fuel assembly.

Ionization chambers can be constructed in many configurations with wide ranges of sensitivity, and the use of ion chambers in radiation dosimetry is well understood and extremely reliable. Operating the ion chamber in the current mode, the measurement times are short, typically, a few seconds.

Recently, an annular detector was constructed that incorporates both a gross gamma measurement and a passive neutron measurement. This detector is shown in Fig. 13. It was divided into quadrants with a neutron detector and an ion chamber in each quadrant. This detector was designed so that the fuel assembly is positioned inside the annulus. The annular design minimizes variations in signal response due to changes in the fuel assembly to detector separation.

The ring detector was tested on 36 spent fuel assemblies at a commercial PWR reactor storage facility. In the test, two different types of neutron detectors were examined, a B-10 lined and a fission chamber. Each type was placed in opposite quadrants. Both detectors were 2.54 cm diameter x 12.5 cm long.

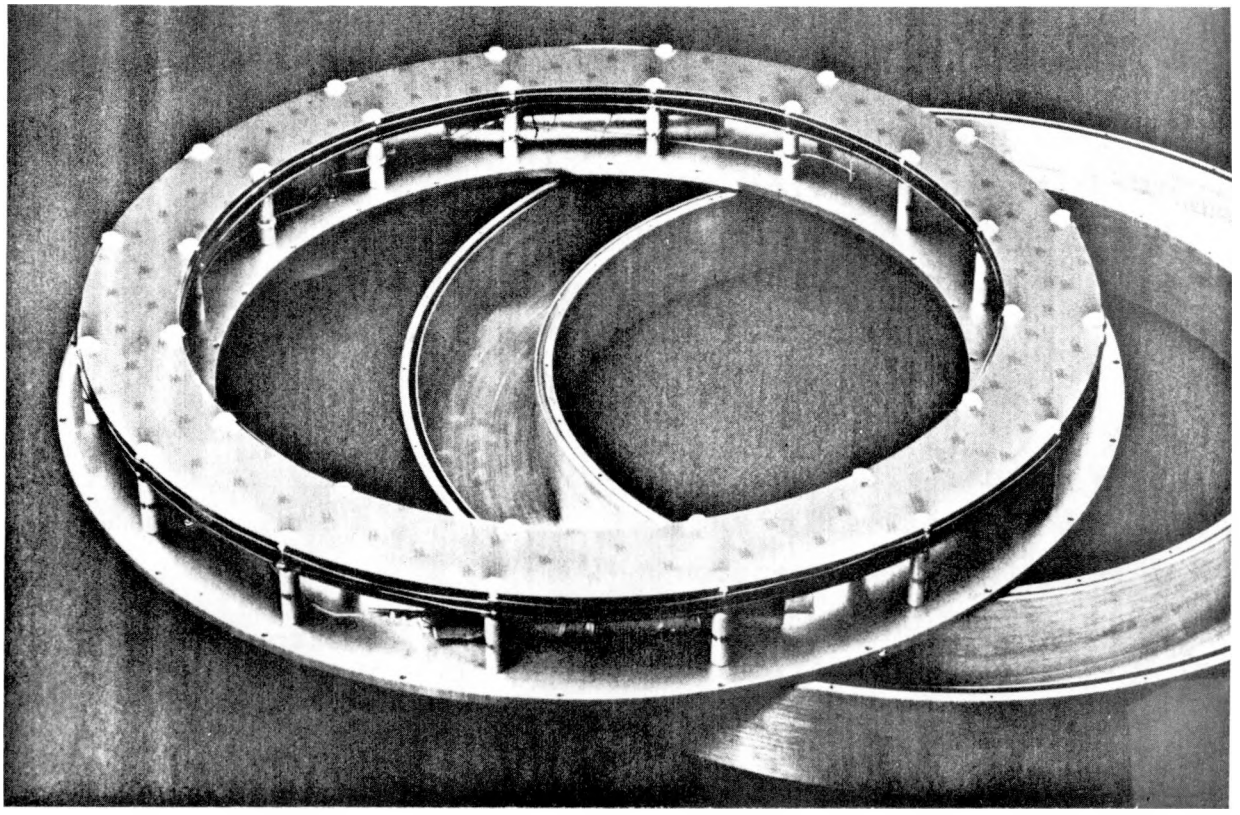


Fig. 13.
Annular "ring" detector for gross-gamma and passive neutron measurements of spent fuel.

During the examinations the B-10 lined neutron detectors lost sensitivity, presumably because of the high-radiation fields present (10^3-10^5 R/h). Therefore, the neutron results were limited to the data from fission chambers. The results for this test are shown in Fig. 14 for the neutron measurements and Fig. 15 for the gross-gamma measurements. Plotted are the sum of the 2 neutron detectors in Fig. 14 and the sum of the 4 ion chambers in Fig. 15. The data was obtained from a single measurement at the center of the assemblies.

The neutron data fall into basically six groups according to cooling times ranging from 4 months to 40 months. The burnup values within each group are similar with the values for the different groups ranging from 18 000-38 000 MWd/MTU. The uncertainties in the measurements (15-20%) were greater than the statistical uncertainty and is probably due to the difficulty

in accurately positioning the detector with respect to the fuel assembly. In a water media a 1-cm positioning error can translate to an error of 10% in the relative neutron counting rate. The errors would be reduced by having a neutron detector in each quadrant.

It is interesting to note that the data plotted in Fig. 14 appear (with the exception of the 4 month cooling time data) to follow the following empirical relationship:

$$\text{Count rate} = \alpha * (\text{Burnup})^{\beta}.$$

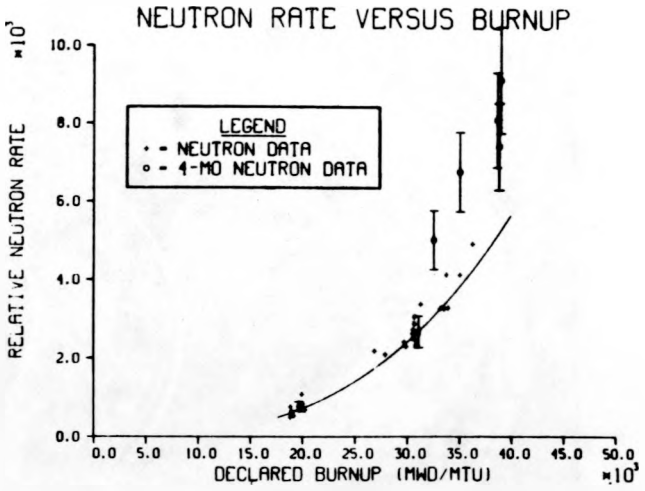


Fig. 14.
Measured neutron rate as a function of burnup from the ring detector.

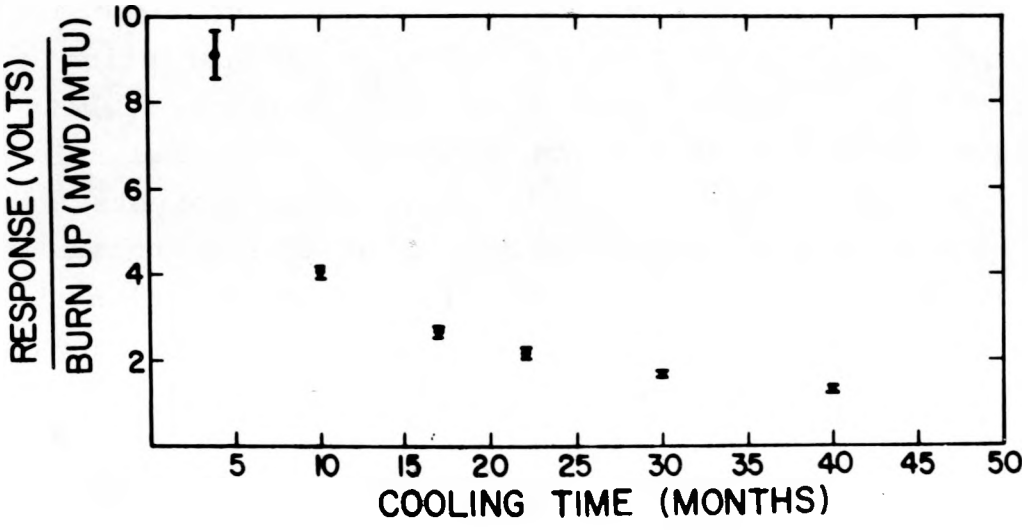


Fig. 15.
Gross-gamma results as a function of burnup from the ring detector.

The data have not been corrected for cooling time. The fact that the short cooling time data do not follow this relationship can be explained by noting that the ^{242}Cm isotope has a short half-life (163 days) and is an important neutron emitter for short cooling times but decreases in importances for cooling times $\geq$ 1 year. A functional relationship similar to the above has been observed several times before.[21,23]

The gross-gamma ion chamber results also fall into 6 groups. The average for each group is plotted with the error bars indicating the standard deviation for that group average. The results exhibit a smooth functional relationship similar to that found for the decay power of irradiated fuel after discharge.

The results of the gross neutron and gamma-ray measurements might have important significance for safeguards verification. For generic types of reactor fuel, i.e., fuel assemblies of the same manufacture, one might be able to establish calibration curves similar to those observed in Figs. 14 and 15 from which the burnup and cooling time information can be obtained. Although these empirical relationships depend on similar operating conditions for the reactor, no reliance has been placed on a detailed knowledge of the operating history of the reactors only the assumption that reactors of the same manufacture are generally operated in similar ways.

B. Qualitative Techniques

A very useful technique has recently been developed that allows an inspector to observe the Cerenkov glow resulting from the interaction of the radiation from a spent-fuel assembly with the cooling water.[36] The device in its simplest form is shown in Fig. 16. It is a hand-held night-vision camera operated by an inspector above the surface of the storage pool. Although overhead light sources must be eliminated, the ability for making a measurement without entering the water is a distinct advantage. An example of the Cerenkov light is shown in Fig. 17 where 4 PWR fuel assemblies can be seen. Efforts are now underway to electronically digitize the light signal so that more quantitative results can be obtained.

Fig. 16.
Hand-held night-vision camera for observing the Cerenkov glow from spent fuel.

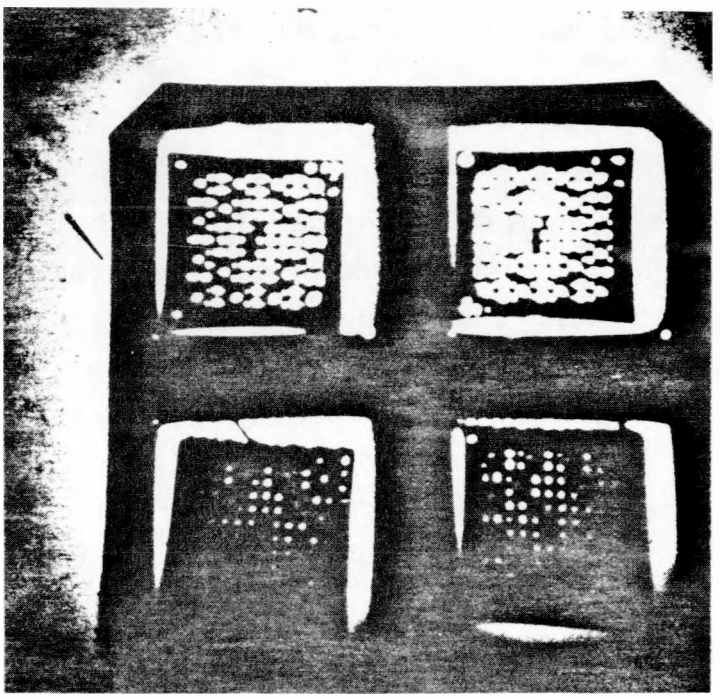


Fig. 17.
Cerenkov light from 4 PWR fuel assemblies. This picture was not taken with the hand-held night-vision camera.

V. Be(γ,n) - A GAMMA SPECIFIC DETECTION TECHNIQUE

A detector that is sensitive to a specific fission product gamma ray can be quite useful if more detailed information is required than is available from a gross-gamma measurement but the complexity of an HRGS system is undesirable. A detector of this type has been developed at LASL that has the capability of operating in the high radiation fiels of the fuel assembly but is still sensitive to a direct fission product.

The Be(γ,n) detector (Fig. 18) is a small fission chamber surrounded by a polyethylene annulus and a beryllium sleeve. It detects fission product gamma rays of energies greater than the 1660-keV threshold through the photoneutron reaction,

$$\gamma + {}^{9}\mathrm{Be} \rightarrow {}^{8}\mathrm{Be} + n - 1660\ \mathrm{keV} \longrightarrow 2\,{}^{4}\mathrm{He}$$

The neutrons released in this reaction are first moderated in the polyethylene and then detected in the fission chamber. The primary gamma ray above the 1660-keV threshold is the 2186-keV gamma ray of the ^{144}Pr (half-life = 17.3 min), which is in secular equilibrium with its fission product parent, ^{144}Ce (half-life = 284.5 days). The Be(γ,n) measurement reflects the more recent irradiation exposure (3 yr or less) of the fuel assembly. Because this detector relies on pulse counting, the measurement time depends mostly on the statistical precision desired. To obtain 2% statistics for a typical fuel assembly with approximately 25 000 MWd/MTU burnup and 2- to 3-yr cooling time, count times of approximately 5 min would be required for the detector shown in Fig. 18. The principal advantage of this detector is that it is only sensitive to a direct fission product. The Be(γ,n) detector can operate in the high radiation fields of the fuel assembly because the fission chamber is insensitive to high dose rates $< 10^{6}$ R/h.

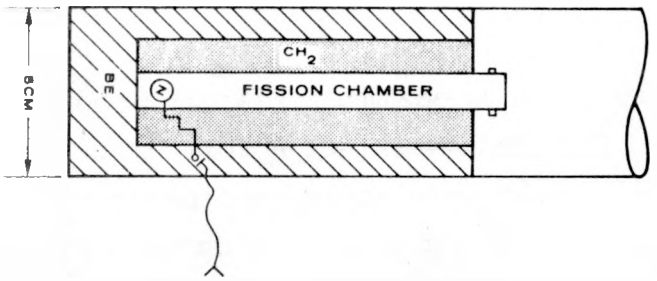


Fig. 18.
Be(γ,n) detector for measuring gamma rays with energies more than 1660 keV.

VI. AXIAL ACTIVITY PROFILE MEASUREMENTS

Measurement of the axial activity profile of spent fuel assemblies can verify the integrity of the fuel assembly by determining that the assembly is radioactive for its entire length, that the activity profile is typical for that type of assembly, and that the absolute activity is similar to that of other assemblies of the same type, cooling time, burnup, and irradiation history. In addition, the axial activity profile can provide an integrating function that can be combined with HRGS to provide a more accurate measure of burnup.

Among the many ways to measure the axial activity profile are HRGS, fission chamber, Be(γ,n), ion chamber, and Cerenkov measurements. The HRGS and Be(γ,n) measurements can be related to specific fission product gamma rays, whereas the ion chamber and Cerenkov measurements are nonspecific and can only measure the gross gamma-ray activity profile. The ^{137}Cs activity profile is often assumed to represent the actual burnup profile for the BWR and PWR assemblies. Profile measurements with the Be(γ,n) detector and ion chambers have been shown to be in good agreement with ^{137}Cs activity profiles.[24] Examples of BWR and PWR profiles are shown in Figs. 19 and 20.

Axial profiles measured with neutron detectors are in good agreement with the ^{137}Cs profiles measured with HRGS.[21] The neutron rate appears to be a nonlinear function of burnup with an empirical power relationship being selected. Unlike the axial gamma measurements that relate the fission product profile to the burnup profile, the neutron measurements relate

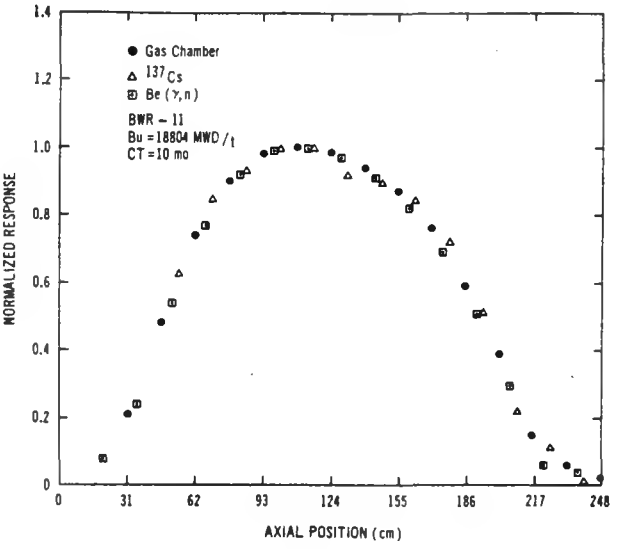


Fig. 19.
Profile measurements of a BWR spent fuel assembly.

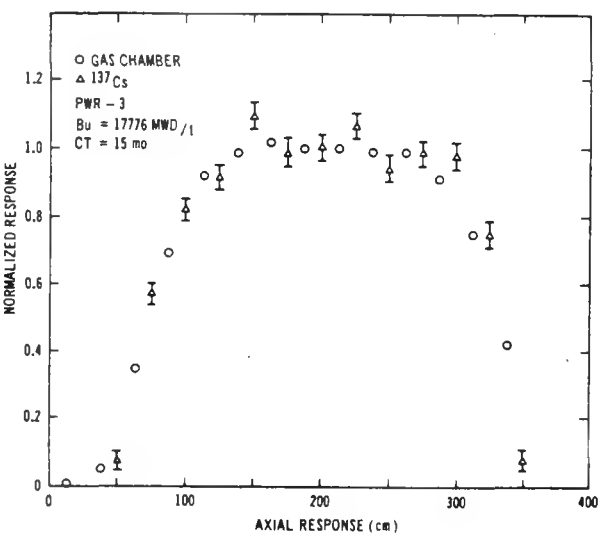


Fig. 20.
Profile measurements of a PWR spent fuel assembly.

the axial profile of the transuranic isotopes to the burnup profile. The results of a neutron activity profile of a BWR fuel assembly is shown in Fig. 21. In this plot the neutron response is raised to a negative root because of the nonlinear dependence with burnup.

All spent fuel profile measuring systems in use rely on mechanical scanning systems. These systems either move the fuel assembly past the detector or move the detector past the fuel assembly. In general, it takes 30-90 min to measure one assembly by mechanical scanning in conjunction with pulse counting methods HRGS, fission, Be(γ,n) . Although mechanical scanning systems have been successful, the requirement of fuel assembly or detector movement is a major drawback to axial profile measurements as applied to safeguards and plant control.

Recently, a multielement system was designed at LASL to eliminate the mechanical scanning system, and it may reduce the measurement times, including gamma-ray spectra and neutron measurements, to less than 10 min. This system uses multiple ion chamber elements in one long detector, so that all measurement points on the profile are taken simultaneously rather than sequentially, as is the case with mechanical scanning.

The multielement profile detector can have 15 to 64 identical ion chamber elements equally spaced along the axis of the fuel assembly.[25] They are in a waterproof enclosure and the signal from each element is transmitted through a waterproof cable to multiplexing electronics located away from the pool side. A prototype constructed for evaluation of Materials Testing Reactor (MTR) fuel is shown in Fig. 22. This device can record the gamma profile on a cassette, teletype, or other similar recording device in less than 10 s.

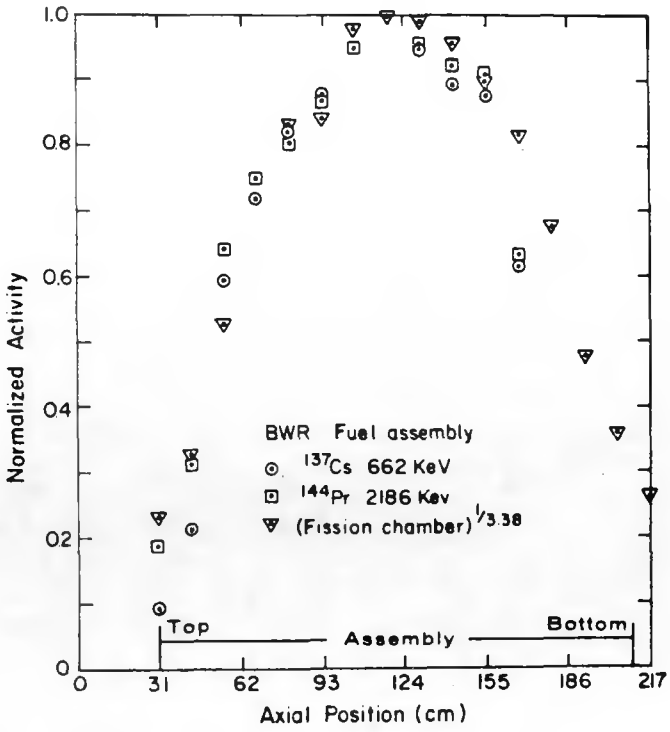


Fig. 21.
Neutron activity profile of a BWR spent fuel assembly.

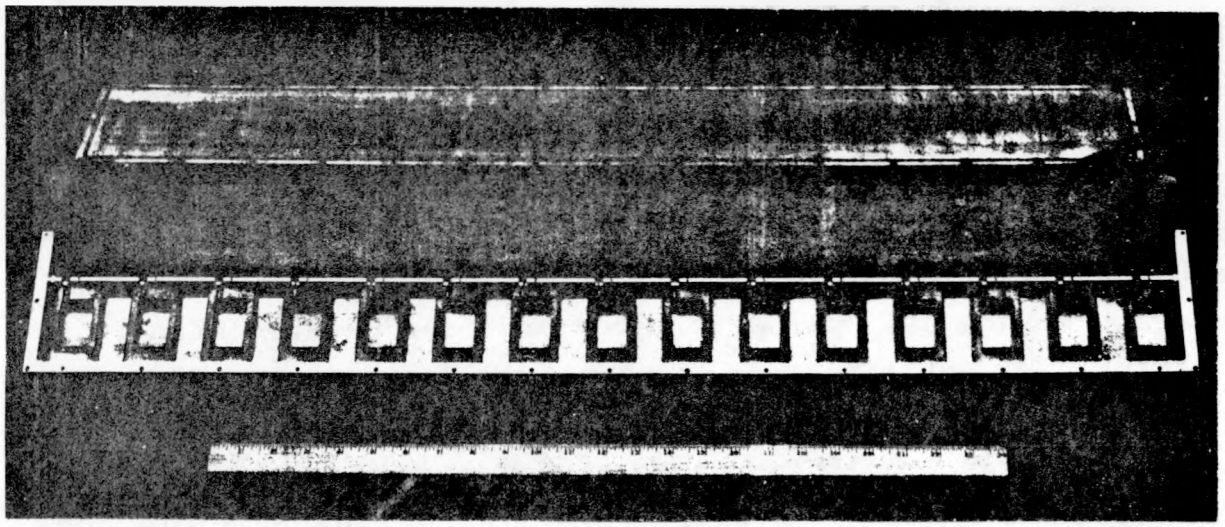


Fig. 22.
Multielement ion chamber for obtaining the axial gross-gamma profiles of irradiated fuel elements.

VII. ACTIVE INTERROGATION TECHNIQUES

Recent advances in neutron generators and radioactive sources have opened the possibility of using active neutron interrogation techniques to measure directly the fissile content of the spent fuel assemblies. Such neutron interrogation systems have been proposed by Czubek[26] and are in routine use by the well-logging industry with transportable field instrumentation. Both pulsed neutron generators and ^{252}Cf neutron sources have been used for this application.

Past work at LASL on spent fuel verification using neutron interrogation[27-29] has been restricted to high-enrichment fuel. Thus, the spent fuel has ^{235}U as the only significant fissile component. However, the IAEA is most interested in verification of LWR fuel assemblies that contain both uranium and plutonium fissile components. The straightforward assay (neutron interrogation and counting the induced prompt or delayed fission neutrons) cannot determine the plutonium content. During burnup in the reactor, the ^{235}U decreases as the plutonium increses, so the sum of the two components does not give the plutonium content directly.

VIII. CONTAINMENT AND SURVEILLANCE TECHNIQUES[30]

A. Introduction

The purpose of containment and surveillance (C-S) is to detect the undeclared movement or alteration of nuclear material. Through a combination of seals, surveillance monitors, and inspections, the C-S system can monitor areas containing measured material, such as a PuO_2 storage vault, thus preserving the materials accounting data, or areas containing unmeasured material, such as scrap storage. In any case, C-S systems are ideally suited to areas where the material form does not change and where transfer control is based on identification and piece count of items. Therefore, spent-fuel storage pools are amenable to the application of C-S techniques.

B. Containment and Surveillance Techniques for Spent Fuel Storage

A conceptual C-S system that would detect the movement of fuel in spent- LWR-fuel storage pools has been proposed.[31] The system would rely on the collection of data from C-S instrumentation with local or remote data analysis and only occasional inspection. A combination of radiation, crane, acoustic, portal, electric power, and closed-circuit television monitors would be used to detect movements of the fuel assemblies. The C-S hardware would be equipped with tamper-indicating devices.

1. Ultrasonic Seals. Ultrasonic identification and integrity devices ("seals") have been under development at the Ispra Laboratories* since 1970.[32-34] They are currently being evaluated for possible use in a fuel assembly identification device (FAID) system.[31] An item is identified by ultrasonic signals reflected from inclusions or from randomly dispersed natural defects, such as welds. Integrity is maintained by rendering the device unusable when it is removed from the item to which it is attached, although the inclusions can still be read to identify the device after is it removed. A seal identity pattern should include at least eight amplitude peaks. At least one million seals with random inclusions can have unique signatures.[34]

Ultrasonic cap seals were in experimental use on BWR fuel bundles in the Lingen VWL Gundremmingen URB reactors.[32] The cap seal was developed for spent CANDU fuel in tests at Douglas Point, Canada, and will be manufactured for use with the 600 MW CANDU reactors.[33] In spent-fuel storage pools, ultrasonic cap seals could be used to identify spent fuel assemblies and to preserve the integrity of any measurements that are made. If a transducer is permanently attached or is an integral part of the seal, then continuous identification and integrity could be maintained.

---

*Commission of the European Communities - Joint Research Centre at Ispra, Italy.

The long-term objective is to develop a FAID system for the lifetime of LWR fuel assemblies. The continuous integrity of any such system during reactor irradiation remains to be demonstrated.

2. Surveillance Monitors. The proposed radiation monitor consists of an array of Geiger-Mueller tubes. Unfolding techniques can be used to estimate the strength, position, and direction of travel of the source.

The crane monitor reports position, load, direction of travel, and activity. The sensors for these four functions are strain gauges.

Acoustic monitors provide an intrusion alert whenever acoustic signals within the pool are characteristic of fuel assembly movements. Methods are being developed to distinguish between fuel movements and expected background signals.

Portal monitors indicate door openings, and electric power monitors indicate the use of electric motors.

The closed-circuit television system records a TV picture at intervals determined by the inspector or when an anomalous condition is detected by the other sensors.

3. Data Collection and Analysis. Data is transmitted from each sensor through a tamper-indicating fiber optic system to a data collection and analysis computer. The computer provides on-site analysis and transmittal of data on command to a remote monitoring station.

REFERENCES

1. C. W. Alexander, C. W. Kee, A. G. Croff, and J. O. Blomeke, "Projections of Spent Fuel to be Discharged by the US Nuclear Power Industry," Oak Ridge National Laboratory report ORNL/TM-6008 (October 1977).

2. S. S. Brumback and R. B. Perry, "Autoradiography as a Safeguards Inspection Technique for Unirradiated LWR Fuel Assemblies," Argonne National Laboratory report ANL-78-27 (May 1978).

3. J. D. Brandenberger, H. O. Menlove, and E. Medina, "Portable Active Neutron Interrogation System for Light-Water Reactor Fuel Assemblies," Los Alamos Scientific Laboratory report LA-7528-M (October 1978).

4. H. O. Menlove, "Description and Operation Manual for the Active Well Coin- cidence Counter," Los Alamos Scientific Laboratory report LA-7823-M (May 1979).

5. S. T. Hsue, T. W. Crane, W. L. Talbert, Jr., and J. C. Lee, "Nondestructive Assay Methods for Irradiated Nuclear Fuels," Los Alamos Scientific Laboratory report LA-6923 (January 1978).

6. J. R. Phillips, G. E. Bosler, J. K. Halbig, and D. M. Lee, "Nondestructive Measurement of Spent Fuel Assemblies at the Tokai Reprocessing and Storage Facility," Los Alamos Scientific Laboratory report LA-8076-MS (December 1979).

7. T. Dragnev, "Nondestructive Assay Techniques for Nuclear Safeguards Measurements," in Atomic Energy Review, Vol. 11, No. 2, 1973.

8. Advisory Group Meeting on the Nondestructive Analysis of Irradiated Power Reactor Fuel, IAEA report AG-11, Vienna, Austria, April 4-7, 1977.

9. G. R. Parkos and G. F. Valby, "Gamma Scan Measurements at Zion Station Unit 2 Following Cycle 1," Electric Power Research Institute report EPRI NP-509 (October 1977).

10. M. B. Cutrone and G. F. Valby, "Gamma Scan Measurements at Quad Cities Nuclear Power Station Unit 1 Following cycle 2," Electric Power Research Institute report EPRI NP-214 (July 1976).

11. N. Beyer, M. deCarolis, E. Dermendjiev, A. Keddar, and D. Rundquist, "IAEA NDA Measurements as Applied to Irradiated Fuel Assemblies," IAEA-SM 231/106 Int. Symp. Nucl. Mat. Safeguards, Vienna, October 2-6, 1978.

12. H. Hick and M. Lammer, "Interpretation of Gamma Spectrometric Measurements on Burnt Fuel Elements," in Safeguards Techniques, Proc. Symp., Karlsruhe, July 6-10, 1970 (International Atomic Energy Agency, Vienna, 1970), Paper IAEA-SM-133/5, Vol. I, pp. 533-537.

13. R. L. Heath, "The Potential of High-Resolution Gamma-Ray Spectrometry for the Assay of Irradiated Reactor Fuel," Safeguards Res. Devel., Proc. Symp. Argonne National Laboratory, June 26, 1967, report WASH-1076, pp. 115-129 (1967).

14. H. G. Mehner, "Bestimmung der Relativen Effektivatat in Gamma-Spectrometrischen Abbrandmesseinrichtungen," Kernenergie 19, No. 1, 3-5 (1976).

15. C. Beets, P. Bemelmans, and R. Pirard, "Gamma Measurements on SENA Spent Fuel Assemblies and Dissolver Solutions," Contributions to the Joint Safeguards Experiments Mol IV at the Eurochemic Reprocessing Plant, Mol, Belgium, report BLG-486 (September 1973).

16. T. N. Dragnev, "Intrinsic Self-Calibration of Nondestructive Gamma Spectrometric Measurements (Determination of U, Pu, and Am-241 Isotopic Ratios)," International Atomic Energy Agency report IAEA/STR-60 (August 1976).

17. J. R. Phillips, D. M. Lee, T. R. Bement, J. K. Halbig, S. T. Hsue, and K. Kaieda, "Nondestructive Examination of Irradiated LWR Fuel Assemblies," Proc. INMM 20th Annual meeting, Albuquerque, New Mexico, July 16-19, 1979.

18. T. Suzaki, H. Tsuruta, and S. Matsuura, "Nondestructive Gamma-Ray Spectrometry and Analysis on Spent Fuel Assemblies of the HPDR-1 Core," Japan Atomic Energy Research Institute report JAERI Memo 8164 (March 1979).

19. S. T. Hsue, "Methods for the Nondestructive Assay of Irradiated Fuels for Safeguards," Atomic energy Review 16, No. 1, 89-128 (1978).

20. "Advisory Group Meeting on the Nondestructive Analysis of Irradiated Power Reactor Fuel," International Atomic Energy Agency report AG-11 (April 1977).

21. S. T. Hsue, J. E. Stewart, K. Kaieda, J. K. Halbig, J. R. Phillips, D. M. Lee, and C. R. Hatcher, "Passive Neutron Assay of Irradiated Nuclear Fuels," Los Alamos Scientific Laboratory report LA-7645-MS *ISPO 43) (February 1979).

22. P. Barbera, G. Bidoglio, M. Bresesti, R. Chevalier, et al., "Post-Irradiation Examination of the Fuel Discharged from the Trino Vercellese Reactor after the 2nd Irradiation Cycle," Commission of European Communities report EUR 5605.e (1976).

23. E. Dermendjiev, private communication, International Atomic Energy Agency (1978).

24. D. M. Lee, J. R. Phillips, S. T. Hsue, K. Kaieda, J. K. Halbig, E. G. Medina, and C. R. Hatcher, "A New Approach to the Examination of LWR Irradiated Fuel Assemblies Using Simple Gas Chamber Techniques," Los Alamos Scientific Laboratory report LA-7655-MS (ISPO 48) (March 1979).

25. D. M. Lee, J. R. Phillips, S. T. Hsue, K. Kaieda, J. K. Halbig, E. G. Medina, and C. R. Hatcher, "A Quick and Accurate Profile Monitor for Irradiated Fuel," Proc. 1st ESARDA Symp. on Safeguards and Nucl. Mat. Management, pp. 265-269, Mol, Belgium, April 25 and 26, 1978.

26. Jan A. Czubek, "Pulsed Neutron Method for Uranium Well Logging," Geophysics 37, No. 1 (Febeuary 1972), pp. 160-173.

27. G. R. Keepin, "Nuclear Safeguards Research and Development, Program Status Report, May-August 1970," Los Alamos Scientific Laboratory report LA-4523-MS (1970).

28. R. H. Augustson, H. O. Menlove, D. B. Smith, A. L. Bond, D. C. Durrill, W. P. Hollowell, and C. P. Bromley, "An Automated Nondestructive Assay System for the Measurement of Irradiated Rover Fuel," Nucl. Mater. Manage. (Fall 1975), pp. 152-166.

29. H. O. Menlove, G. W. Eccleston, D. A. Close, and L. G. Speir, "Neutron-Interrogator Assay System for the Idaho Chemical Processing Plant Waste Canisters and Spent fuel, Preliminary Description and Operating Procedures Manual," Los Alamos Scientific Laboratory report LA-7250-M (May 1978).

30. D. D. Cobb, H. A. Dayem, and R. J. Dietz, "Preliminary Concepts: Safeguards for Spent Light-Water reactor Fuels," Los Alamos Scientific Laboratory report LA-7730-MS (1979).

31. J. P. Holmes, "Conceptual Design of a System for Detecting National Diversion of LWR Spent Fuel," Sandia Laboratories, Albuquerque report SAND78-0192 (September 1978).

32. S. J. Crutzen, C. J. Vinche, W. H. Burgers, and M. R. Combet, "Remote Controlled and Long Distance Unique Identification of Reactor Fuel Elements or Assemblies," IAEA/SM/231-21 (October 1978).

33. S. J. Crutzen and R. G. Dennys, "Use of Ultrasonically Identified Security Seals in the 600 MW CANDU Safeguard System," IAEA/SM/231-124 (October 1978).

34. W. L. Zijp and S. J. Crutzen, "Statistical Aspects of Ultrasonic Signatures," IAEA/SM/231-22 (October 1978).

35. D. M. Lee, J. R. Phillips, J. K. Halbig, S.-T. Hsue, L. O. Lindquist, E. M. Ortega, J. C. Caine, J. Swansen, K. Kaieda, and E. Dermendjiev, "New Developments in Nondestructive Measurement and Verification of Irradiated LWR Fuels," presented at the American Nuclear Society Topical Conference, Measurement Technology for Safeguards and Materials Control, at Kiawah Island, SC, November 26-29, 1979, LA-UR-79-3223.

36. E. J. Dowdy and J. T. Caldwell, Eds., "Irradiated Fuel Monitors. Preliminary Feasibility Study," Los Alamos Scientific Laboratory report LA-7699 (ISPO 51) (May 1979).

INTERNATIONAL TRAINING COURSE ON NUCLEAR MATERIALS ACCOUNTABILITY FOR SAFEGUARDS PURPOSES

SESSION #22: LECTURE/TOUR OF SAFEGUARDS RESEARCH FACILITIES AND DEMONSTRATION OF INSTRUMENTATION

SPEAKER: Dr. Nicholas Nicholson

Los Alamos Scientific Laboratory
Los Alamos, New Mexico USA

Tuesday, June 3, 1980
9:00 a.m.

BIOGRAPHY

Education: Ph.D. in Physics from West Virginia University, 1965.

Present Position: Assistant Group Leader in the Detection, Surveillance, Verification, and Recovery Group (Q-2) at Los Alamos Scientific Laboratory.

Present Duties: Developing techniques and instrumentation for several safeguards programs.

Past Positions: Program monitor for Atomic Energy Commission. Experience in weapons diagnostics at LASL.

INTERNATIONAL TRAINING COURSE ON NUCLEAR MATERIALS ACCOUNTABILITY FOR SAFEGUARDS PURPOSES

SESSION #22: LECTURE/TOUR OF SAFEGUARDS RESEARCH FACILITIES AND DEMONSTRATION OF INSTRUMENTATION

SPEAKER: Dr. T. Douglas Reilly

Los Alamos Scientific Laboratory
Los Alamos, New Mexico USA

Tuesday, June 3, 1980
9:00 a.m.

BIOGRAPHY

Education: Amherst College, Amherst, Mass.; physics, A.B. cum laude 1964; Case Institute of Technology, Cleveland, Ohio; physics, Ph.D. 1970.

Present Position: Staff member in International Safeguards group at Los Alamos Scientific Laboratory.

Present Duties: Develop and implement nondestructive analysis techniques for use by operators and safeguards inspectors. Develop and present training courses in the use of NDA techniques.

Past Positions: Physics Instructor at Case Institute of Technology. Deep mine experiments to measure solar neutrinos and very high energy cosmic rays. Visiting Staff Member at Joint Research Centre of the Commission of European Communities, Ispra, Italy (1976-1978).

INTERNATIONAL TRAINING COURSE ON NUCLEAR MATERIALS ACCOUNTABILITY FOR SAFEGUARDS PURPOSES

SESSION #22: LECTURE/TOUR OF SAFEGUARDS RESEARCH FACILITIES AND DEMONSTRATION OF INSTRUMENTATION

SPEAKER: Dr. Hastings Smith

Los Alamos Scientific Laboratory
Los Alamos, New Mexico USA

Tuesday, June 3, 1980
9:00 a.m.

BIOGRAPHY

See Session # 19

Session Objectives

SESSION #22: LECTURE/TOUR OF SAFEGUARDS RESEARCH FACILITIES AND DEMONSTRATION OF INSTRUMENTATION

The tour will cover the facilities at the Los Alamos Scientific Laboratory (LASL) where research and development of nondestructive assay (NDA) instruments are carried out. This work ranges from development of the simplest, highly portable assay devices for field use to complex computer-based instruments for use in high-precision assays of special nuclear materials in a variety of forms. The selection of assay instruments on display will attest to the wide range of NDA problems being addressed and will afford Course participants a first-hand acquaintance with state-of-the-art NDA technology. A dynamic materials accounting system (DYMAC), which makes possible near-real-time accountability of in-process special nuclear materials (SNM) in a variety of physical and chemical forms is in an advanced stage of development and in-plant evaluation. It will be shown how such dynamic materials accounting capability can greatly enhance both the timeliness and sensitivity for detection of nuclear material diversion, while also contributing to improved process and quality control.

SESSION 22: LECTURE/TOUR OF SAFEGUARDS RESEARCH FACILITIES AND DEMONSTRATION OF INSTRUMENTATION

H. Smith
Los Alamos Scientific Laboratory, Group Q-1

I. INTRODUCTION

Instrumentation and measurement systems for safeguarding special nuclear materials (SNM) have been under development at the Los Alamos Scientific Laboratory for over a decade. Many of these systems have been implemented and have become standard components in the nuclear measurement industry. Several are fully developed and are being commercially produced. In addition, there is growing interest in the integration of much of the established instrumentation into dynamic materials accounting systems. Such systems would be invaluable in a complex SNM environment where detailed materials accounting information is needed on a timely basis. LASL's development efforts in this area have been outlined in Session #13.

There continue to be many unsolved SNM measurement problem areas that are being addressed by research and development. The motivation for these research activities arises from several sources:

(a) The increasing variety of nuclear materials and environments for which adequate safeguards are required.

(b) The need to improve measurement accuracy.

(c) Operational and process requirements, identified by both safeguards systems analyses and direct requests from nuclear facility operators.

(d) A growing interest by the international safeguards community in applying nondestructive assay (NDA) techniques.

The instrumentation that you will be shown on this tour will be a selection from the wide variety of NDA research projects that have been and are being carried out at the Los Alamos Scientific Laboratory. You should refer to the text material from Sessions #19a and #19b for further discussion of specific instruments.

II. TECHNICAL DATA SHEETS ON SELECTED NDA INSTRUMENTATION

Contained in this section are some technical details on NDA instruments developed at LASL. Further details may be obtained from the course material for Session #19.

A. Segmented Gamma Scanner

1. Problem. Scrap and waste streams contain strategic quantities of fissile materials mixed with a great variety of process residues that cannot be analyzed by conventional sampling and analysis techniques. Typical nuclear fuel cycle facilities may routinely recycle 25% of their throughput as solid waste or scrap during normal operations and as much as 100% during startup or process setups. The need exists for methods to assay the fissile content of these potential diversion paths. Currently the fissile content of mixed scrap and waste matrices can only be measured by NDA techniques. The Segmented Gamma Scanner (SGS) offers a method for obtaining reasonable assays of the fairly large fraction of waste (and some scrap) that is amenable to its assay methods.

2. Accuracy. For reasonably uniform and homogeneous materials $\sim\pm 5\%$.

3. Sensitivity. In the usual configuration the useful sensitivity is a few grams of ^{235}U or ^{239}Pu. In modified configurations with modified procedures the sensitivity can be as low as $\sim$0.01 g of ^{239}Pu or $\sim$0.01 g ^{235}U for some material types and packages.

4. Precision. Often as good as $\pm 1\%$ (1σ) but strongly dependent on system configuration, particular procedures, and, most of all, on the amount of SNM being measured and the nature of the sample containing it.

5. Measurement Time. For a rather wide range of sample size, SNM mass, and matrix materials, 10 to 15 minutes provides a reasonable assay.

6. Matrix Effects. The worst effects are due to nonuniform and inhomogeneous samples. If samples meet the necessary minimum requirements of uniformity and homogeneity, the assays are not sensitive to the chemical composition of the sample.

7. Status. Now commercially available. Continued development directed at improved performance and technology transfer and incorporation of plutonium isotopic assay capability.

NOTE** See Figs. 10-12 in Session #19 text.

B. In-Line Pu K-Edge Densitometer

1. Problem. Reprocessing plant and nitrate-to-oxide conversion plant systems studies have identified in-line assay of Pu solutions as highly desirable for near real-time accounting systems. Rapid, accurate, and nonobtrusive assay instruments for in-line measurement need to be developed and tested in the field.

2. Principle of Measurement. Determination of differential transmission of gamma rays across the energy region of the 121.8-keV Pu K-absorption edge. Two radioactive sources are used: Se-75 (gamma-ray energy = 121.1 keV) and Co-57 (gamma-ray energy = 122.1 keV). The solution Pu concentration is related logarithmically to the ratio of these two transitions. The same transmission data is combined with the passive gamma-ray spectrum from the sample to infer isotopic composition.

3. Description. The Pu solution to be assayed is pumped through a bypass solution loop out of the process holding tank to an assay cell that is positioned in an extension of the process cabinet containment barrier. Once the cell is full, solution flow is stopped, and the Se-75 and Co-57 gamma-ray spectra are taken sequentially by rotation of the necessary sources into the radiation position. These spectra are compared with their empty-cell counterparts, taken earlier during a measurement control run, and transmissions are computed. If isotopics is desired, the Se-75 and Co-57 sources are rotated out of position and behind shielding, and the radiation from the sample solution is counted through an enlarged collimator. Once the solution spectrum is acquired, the appropriate peak areas are determined, transmission corrections are applied to the raw data, and the isotopic composition of the sample is computed.

The management of the instrument hardware, data acquisition, and data reduction are all carried out by an LSI-11/2

minicomputer. This highly automated capability permits extensive self-consistency checks by the instrument on its performance and on the operator's running of the assay.

4. Performance and Accuracy. This instrument is designed for assay of Pu solutions in the 30 grams Pu/liter concentration range, but it can be optimized for higher concentration ranges by reduction of the sample cell size. Measurement precisions on the order of 0.5% are routinely possible with a one-hour assay and transmission source strengths in the 10-50 mCi range.

5. Status. Prototype instrument under development for test and evaluation in the product line of the Savannah River Plant (SRP) reprocessing facility. The instrument is expected to provide accountability data at a key measurement point identified by safeguards systems studies. Future developments will extend the applications to U solutions and to fission product contaminated solutions for assay at other key measurement points within reprocessing facilities, e.g. HEF.

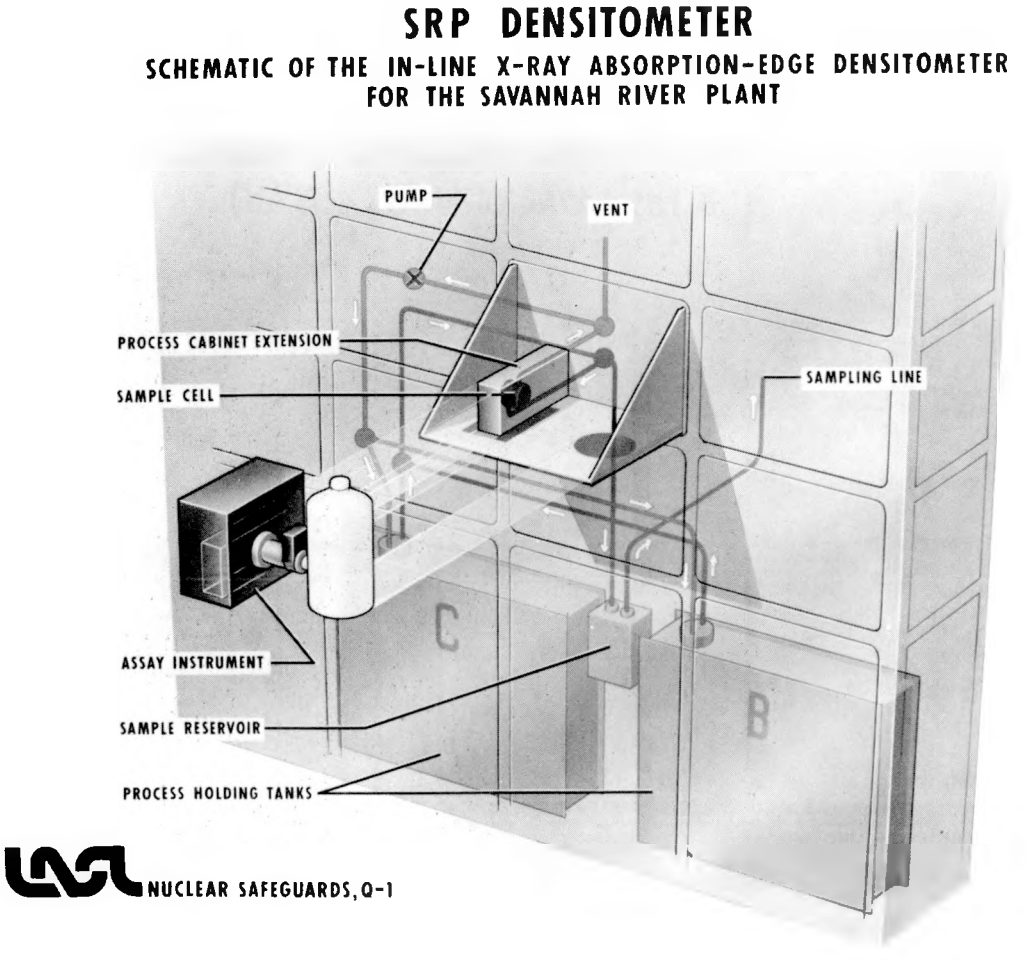


Fig. 1.
Schematic of in-line installation of K-edge Pu solution densitometer. The measurement station (see Fig. 2) is positioned 6.5 feet above the floor level, on a shelf as shown. The Pu solution is pumped from one of the process holding tanks to a measurement cell, which is located inside an extension of the process cabinet containment barrier. The measurement station resides outside the containment but surrounds the sample cell to provide for the proper transmission geometry for the measurement.

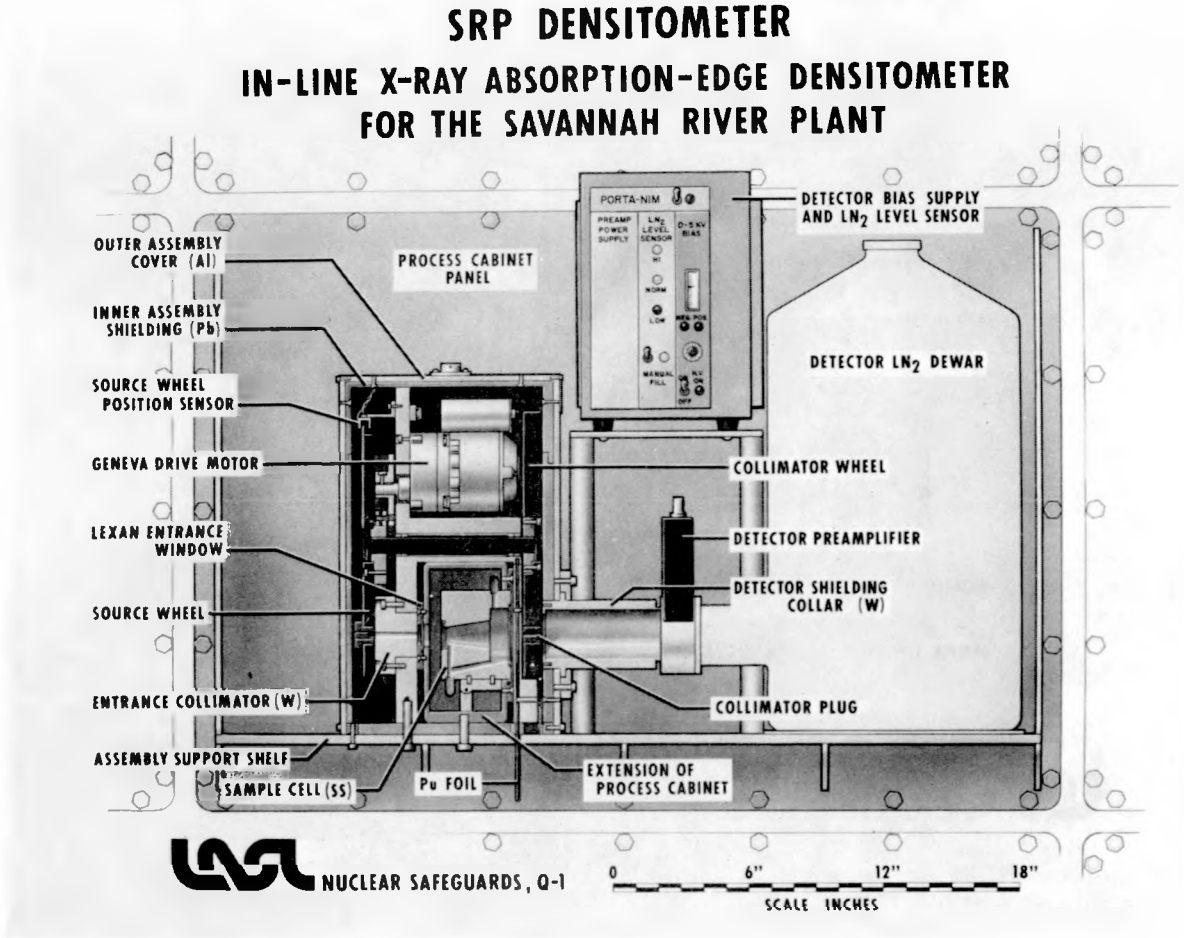


Fig. 2.
Detail of measurement station. The source and collimator wheels sit on either side of the containment cabinet extension, so that the transmission measurement through the sample cell is permitted. The gamma detector sits to the right of the cell, and its electronics is shown on a platform between the source positioning mechanisms and the detector LN dewar.

C. High-Level Neutron Coincidence Counter

1. Problem. One of the most important measurement problems in safeguarding strategic quantities of SNM is high-mass plutonium samples. This material is encountered in the form of powder oxides, pellets, pins, fuel assemblies, metal plates, and liquids. In many cases, NDA methods are required for timeliness and to avoid the destruction of the finished product. The High-Level Neutron Coincidence Counter (HLNCC) was developed to give IAEA inspectors portable instrumentation that can be used to verify the plutonium content for the large variety of sample categories. Present and projected capabilities are given below.

2. Principle of Measurement. Coincidence neutrons from the spontaneous fission of ^{240}Pu are counted in the passive mode. Coincidence counting allows this technique to be used even in the presence of large backgrounds produced by (α,n) reactions in the matrix material.

3. Description. The portable instrument consists of ^{3}He detectors in a polyethylene moderator that completely surrounds the sample measurement cavity. The compact coincidence logic unit is an improved design shift register that can handle very high count rates and is directly interfaced to an HP-97 programmable calculator for automated data reduction and rapid printout of assay results.

4. Performance and Accuracy. The HLNCC can be used for plutonium samples in the range of a few grams up to several kilograms. For measurement times of 1000 s, a precision of 1% or better can be obtained for plutonium samples above several hundred grams. If well-matched samples are available, a similar accuracy can be obtained.

5. Deployment. The unit has received extensive field application by IAEA inspectors and a commercial version of the unit is available.

NOTE** See Figs. 15-16 in Session #19 text.

D. Dual Range Neutron Well Coincidence Counter

1. Problem. Safeguards accountability measurements are required for a wide range of plutonium-containing materials including scrap and waste and bulk product materials. Reliable sampling and chemical assay are frequently not possible or are not rapid enough for timely accounting measurements. A simple assay method that can be applied to a broad range of problems will be useful for a variety of nuclear facilities.

2. Principle of Measurement. Plutonium metal, oxide, and scrap can be assayed with neutrons emitted during the spontaneous fission of the even plutonium isotopes. By detecting neutrons in coincidence, the effects of background neutrons and neutrons produced in (α,n) reactions can be largely eliminated.

3. Description. The Dual Range Coincidence Counter is designed for in-plant applications. ^{3}He tubes are used to detect neutrons emitted from samples placed in the 6"-i.d. well. A 4" polyethylene shield surrounding the detectors minimizes the effects of room background. The device can be operated in a low-efficiency mode (12%) for very hot samples or in a high-efficiency mode (25%) for normal samples (dual range).

4. Range. The lower limit of sample size for quantitative assay by coincidence counting is between 1 and 10 g Pu. The 3σ detectability limit for low-level waste is roughly 5 mg. The upper limit is determined by criticality safety considerations: about 4 kg for Pu metal, 2 kg for oxide, and 1 kg for scrap.

5. Precision. Typically 0.25 to 1.0%, depending on sample size.

6. Accuracy. 1-2% for Pu oxides, 2-4% for metals, and >5% for scrap and waste. For uniform samples assay accuracy can approach the statistical precision.

7. Timeliness. Assay times are typically 5-20 minutes.

8. Matrix Effects. The Dual Range Coincidence Counter is designed to be insensitive to small amounts of moisture (1-2% by weight) in the sample. Coincidence counting is insensitive to single neutrons produced by (α,n) reactions in the matrix; however, self-multiplication of spontaneous fission or (α,n) neutrons within the sample does affect the coincidence response and is at present the limiting factor for assay accuracy.

9. Status. Commercially available. Current laboratory developments are directed toward improving the accuracy and extending the range of applications, such as shipper-receiver verifications. Continuing research is directed at reducing sensitivity to multiplication effects.

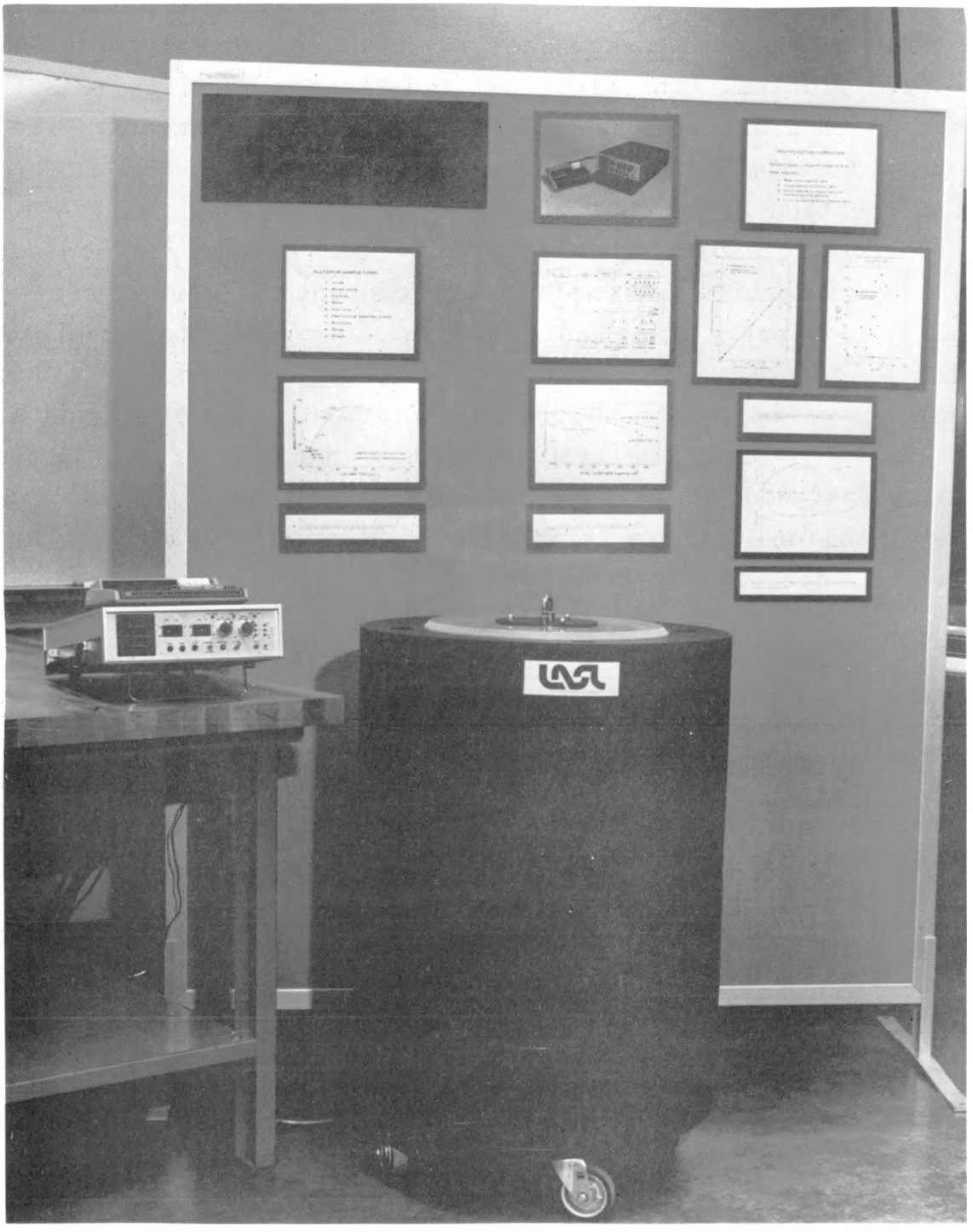


Fig. 3.
The Dual-Range Coincidence Counter. At the left, on the table, are the data analysis and pulse processing electronics.

E. Fluorinel and Storage Facility Delayed Neutron Interrogator

1. Problem. The Fluorinel and Storage Facility (FAST) will be used to dissolve high-enriched spent fuel elements. The dissolved fuel will be sent for reprocessing to the ICPP after clarification by removal of the waste solids.

Before dissolution of a spent fuel element, a nondestructive verification of the fissile content is needed for criticality control. The waste solids must also be examined for fissile content to provide accountability data. Chemical analysis of the waste solids is difficult, expensive, and slow.

A ^{252}Cf-based delayed neutron interrogator is currently being designed to measure both the spent fuel assemblies and waste solids. Characteristics of the FAST Facility Delayed Neutron Interrogator are listed below.

	Fuel Element	Waste Canister
Californium Source	5 mg = 1.23×10^{10} n/s	
^{235}U Quantity	- 10 kg	0 to 400 g
Background Radiation	$\leq$ 30 000 R/h n = 1.7×10^{7} n/s	$\leq$ 30 000 R/h n = 1.7×10^{7} n/s
Sample Dimensions	19" diameter 120" long	5.5" diameter 24" long
Sample Scan Rates	20"/min	5"/min
Measurement and Handling Time	<30 min	<30 min
Accuracy Requirement	$\pm$5% (2σ)	$\pm$30 g (2σ)

2. Status. Under development, scheduled for completion of testing at LASL and for installation in Idaho. System development has been coordinated with the facility architectural engineer since early stages of facility design. Will be used as an experimental test bed to address other high-level radioactive measurement problems, such as the assay of spent light water reactor (LWR) fuels.

NOTE** See Figs. 25-26 in Session #19 text.

F. Californium-252 Shuffler

1. Problem. The assay of feed and product materials as well as scrap and waste streams for fissile uranium and fissile and fertile plutonium. The technique has been successfully employed for the assay of closed containers of sizes ranging from vials to 55-gallon barrels. Depending on the design of the instrument, fissile contents between 0.1 mg and >10 kg can be assayed.

2. Principle of Measurement. Passive neutron counting followed by a cyclical ^{252}Cf neutron irradiation and delayed neutron counting with the source withdrawn.

3. Accuracy. 1-3%, depending on standards. For well-characterized samples 0.5% is possible.

4. Precision. Better than 0.1%. Both short term (repeat assays) and long term, >6 months.

5. Time. Typically 8 minutes, although longer times are possible to improve either the detectability or precision.

6. Matrix Effects. Can be included in the calibration or by correction factor data obtained concurrently with the delayed neutron data.

7. Status. A Shuffler system has been installed and is being evaluated at the Savannah River Plant for assay of HEU scrap and waste generated during the fabrication of production reactor fuels. The range of application is being extended to low-level waste assay, and systems are being developed to provide assay capability for irradiated materials.

NOTE** See Figs. 22-24 in Session #19 text.

G. Californium-252 Barrel Shuffler

1. Problem. Reprocessing plant safeguards and waste management studies have identified the need for an instrument that can both screen for plutonium content at the 10 nCi/g fiducial for permanent burial and measure fissile U and Pu at higher levels for safeguards accounting and surveillance. The assay system should tolerate fission product radiation fields up to 1000 R/h and be compatible with automated materials handling equipment.

2. Principle of Management. Passive neutron counting followed by cyclical ^{252}Cf neutron irradiation and delayed neutron counting with the source withdrawn. Tolerance to high radiation is achieved with a lead liner for the sample cavity and use of radiation resistant detectors.

3. Accuracy. 5-30%, depending on standards and material characteristics.

4. Precision. 3-30% for Pu loadings in the range of 100 g-10 mg.

5. Sensitivity. Approximately 5 mg Pu in a 55-gallon drum.

6. Time. 8-16 minutes.

7. Status. Proof-of-principle measurements in the laboratory are partially completed. More laboratory tests are needed followed by design of a prototype and evaluation at a host facility.

NOTE** See Figs. 22-24 in Session #19 text.

H. Active Well Coincidence Counter

1. Problem. The measurement of high-mass, highly enriched uranium samples is an important safeguards problem. Because of the high self-absorption of ^{235}U gamma rays in uranium, gamma-ray assay techniques can only measure the surface characteristics of high-mass samples. Passive neutron coincidence counting techniques are also useless because of the extremely low spontaneous fission rates of the uranium isotopes. The Active Well Coincidence Counter (AWCC) was developed to measure directly the fissile content of large uranium samples.

2. Principle of Measurement. Random neutrons (uncorrelated in time) from an AmLi (α,n) source irradiate the sample inducing fissions in the fissile isotopes, mainly ^{235}U. These fission events are detected with coincidence neutron counting techniques. The induced neutron signal is separated from the interrogating neutron signal by measuring only the time-correlated component of the neutron flux on the detector.

3. Description. The instrument consists of a polyethylene annulus filled with ^{3}He neutron detectors. The sample measurement cavity (the well of the annulus) is plugged with two end caps, which contain the AmLi interrogation sources and appropriate reflector and moderator material. The sources can be configured to provide either a fast or a thermal neutron irradiation. The compact coincidence logic unit is the same shift register circuit used with the High-Level Neutron Coincidence Counter (HLNCC). Data analysis is done directly in the HP-97 programmable calculator interfaced to the coincidence circuit.

4. Performance and Accuracy. The AWCC can measure highly enriched uranium samples in the range of approximately 50g ^{235}U up to several kilograms. With a measurement time of 1000 s, a sample containing 200 g of ^{235}U can be measured with a precision of 3%. With a thermal interrogation low-enriched

uranium samples containing several grams up to 50-100 g ^{235}U can be measured provided that adequate reference materials and calculations are available to handle the large neutron absorption effects. The system can also be used to measure plutonium samples using the active mode to assay the fissile isotope content and the passive mode (spontaneous fission counting with AmLi sources removed) to measure fertile isotope content.

5. Deployment. The AWCC has been delivered to the IAEA for measurement of highly enriched uranium metal materials associated with MTR reactors and fuel fabrication facilities. Several special end caps with AmLi sources have also been supplied for use with the HLNCC in an active assay.

NOTE** See Figs. 17-21 in Session #19 text.

I. Coincidence Neutron Collar

1. Problem. The IAEA has identified the verification of fresh LWR fuel assemblies at fabrication facilities and power reactors as a high-priority safeguards problem. The major goal is to verify the assembly enrichment and check for the possible substitution of dummy fuel rods within the assembly. Passive gamma-ray techniques are only sensitive to rods in the outer rows of the assembly and are, therefore, inadequate to check for rod substitution.

2. Principle of Measurement. Random neutrons from an AmLi (,n) source irradiate the fuel assembly inducing fissions in ^{235}U. The fission events are detected by measuring the coincidence component of the neutron flux on the detector.

3. Description. The instrument is made of four slab-like sections (approximately 30 cm by 40 cm by 5 cm) arranged as shown in Fig. 4. One side contains the AmLi source and the remaining three sides contain ^{3}He neutron detectors. The standard shift register coincidence unit with the HP-97 calculator is used to power the detector and analyze the data.

4. Performance and Accuracy. The coincidence collar can be used with either PWR or BWR fuel assemblies. In a 1000-s measurement it is capable of detecting the substitution of approximately 3 rods in a 15 by 15 PWR fuel element enriched to 3.19% ^{235}U.

5. Deployment. The coincidence collar is presently undergoing final testing at Los Alamos and will then be used in joint experiments by LASL and the IAEA at a U.S. Fuel Fabrication Facility.

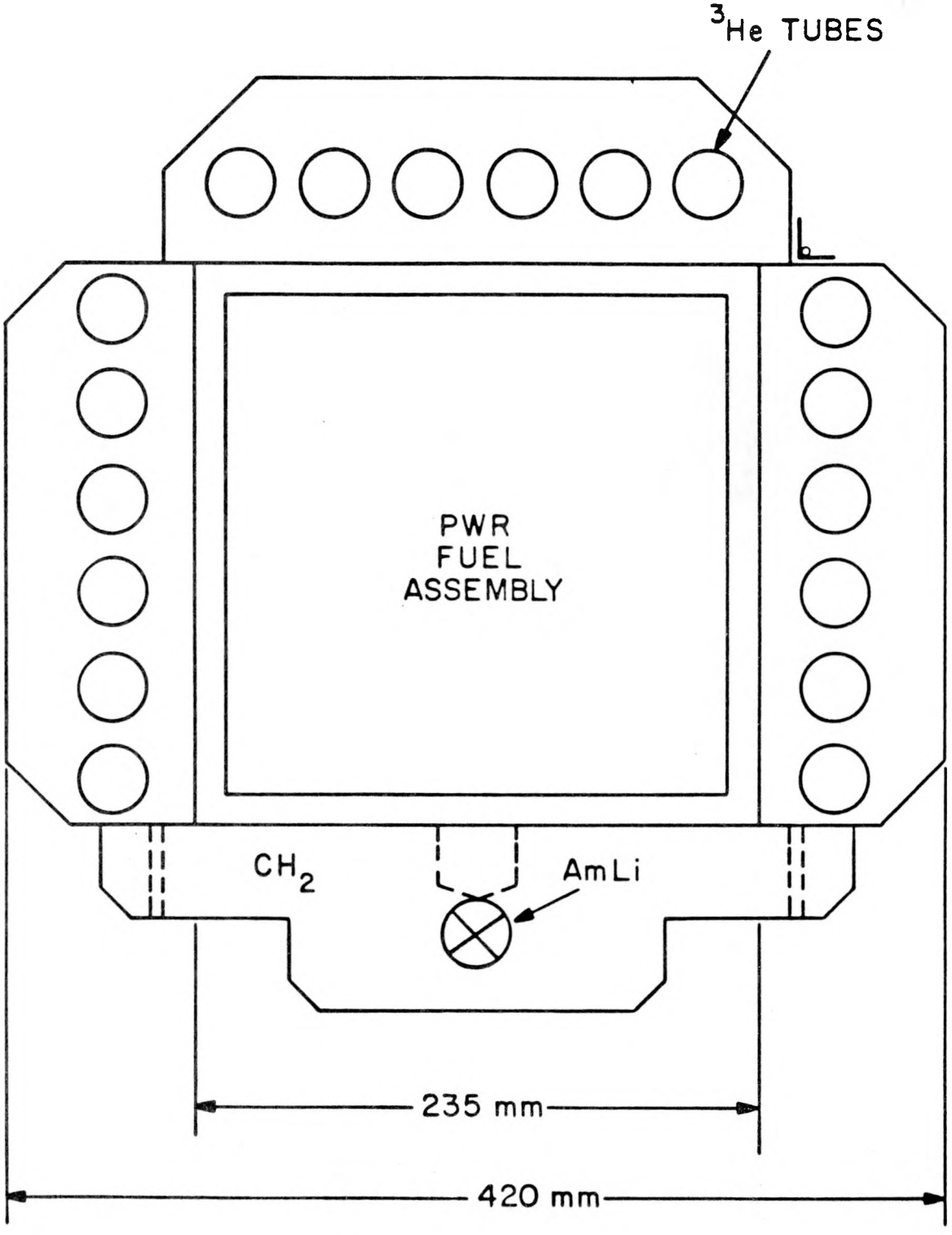


Fig. 4.

Horizontal cross section of coincidence neutron collar. Vertical dimension is 400 mm.

INTERNATIONAL TRAINING COURSE ON NUCLEAR MATERIALS ACCOUNTABILITY FOR SAFEGUARDS PURPOSES

SESSION #23: AN LWR POWER REACTOR FACILITY

SPEAKER: Cordell Reed

Vice President for Nuclear Activities
Commonwealth Edison Co.
Chicago, IL USA

Wednesday, June 4, 1980
8:30 a.m.

BIOGRAPHY

Education: Bachelor of Science Degree in Mechanical Engineering from the University of Illinois.

Present Position: Vice president (12/79 to present) Commonwealth Edison Company

Present Duties: Responsible for all of the Company's nuclear activities including the operation of three nuclear stations, the Nuclear Engineering Department, as well as nuclear licensing, nuclear fuels, and environmental activities.

Past Positions: Assistant Vice-President, Manager of Nuclear Operations. Started working at Edison in 1960. Assignments included coal-fired and nuclear power plants' design & operation. Station Nuclear Engineering Department Manager from 3/75-12/77 and Assistant Vice President for Nuclear Licensing, Nuclear Fuel Services, & Environmental Affairs Departments from 12/77-12/79.

Other Information: American Nuclear Society, Western Society of Engineers, National Technical Association, Economic Club of Chicago, American Association of Blacks in Energy.

Director of Independence Bank of Chicago and Project Management Corporation featured in the US Department of Energy's booklet, "Black Contributors to Science and Energy Technology."

INTERNATIONAL TRAINING COURSE ON NUCLEAR MATERIALS ACCOUNTABILITY FOR SAFEGUARDS PURPOSES

Session Objectives

SESSION #23: AN LWR POWER REACTOR FACILITY
SESSION #24: A CANDU POWER REACTOR FACILITY
SESSION #25: A RESEARCH REACTOR FACILITY

The basic features of existing safeguards systems in specific operating facilities are considered. Emphasis is placed on detailed examples and practical experience in actual operating facilities rather than the basic features and general principles described in earlier sessions (14 & 15).

After the sessions, participants will be able to

1. Compare actual facility safeguards system characteristics and operational performance with the generalized principles and conceptual systems described in earlier sessions.

2. Discuss impact of the various safeguards requirements on facility operations.

3. Have an appreciation of basic safeguards costs and resource requirements in power reactor facilities.

SESSION 23: SAFEGUARDS IN AN LWR POWER REACTOR FACILITY

C. Reed
Commonwealth Edison Company

I. INTRODUCTION

A. The Company

The Commonwealth Edison Company is principally engaged in the production, purchase, transmission, distribution, and sale of electricity. Its electric service territory is 11,525 square miles in the northern part of the State of Illinois (Fig. 1), including the city of Chicago. It provides electric service to 8,000,000 people.

The Company is committed to nuclear energy as an economic and necessary source of electric generation. In 1979, 40% of its electric production was from nuclear generation (remainder: 45% coal, 12% oil, 3% gas). The Company's owned net summer generating capability is about 18,000 megawatts. Of this, over 5,000 megawatts is nuclear. It has three operating nuclear plants with a total of seven reactors. Additional nuclear generating units with a total capacity of over 6,600 megawatts are being constructed (Table I). The Company also has contracted for some equipment for two additional 1,120 megawatt nuclear units at a proposed site in western Illinois.

The Company anticipates that it will maintain adequate quantities of uranium concentrate for the operation of its nuclear generating units through at least 1990. In addition to the Company's present inventory, significant quantities of uranium concentrate are expected from suppliers such as Mary Kathleen Uranium, Ltd., and two wholly-owned subsidiaries of the Company, Cotter Corporation, and Edison Development Company.

Edison's commitments for segments of the nuclear fuel supply cycle, other than for reprocessing spent fuel, have been obtained at least through the years shown on Table II.

Because no plants for reprocessing of spent nuclear fuel are in operation in the United States, the Company has plans, including storage pool modifications, to meet spent nuclear fuel storage requirements at its operating nuclear stations through the early 1990's. Storage requirements beyond this time could require a separate storage facility.

B. Nuclear Unit Refueling Outage Schedules

Edison's dominant load peaks occur during the summer and winter seasons. The load valleys between these peak seasons are used for generating unit maintenance including reactor refueling outages. These plans are developed for the on-going 5 year period. Nuclear unit outages are scheduled first because of their large size. Another restraint is the need to avoid simultaneously refueling outages for both units in a two-unit station. Figure 2 shows Edison's nuclear unit refueling schedules for 1979-1985. Cycle lengths vary because of specific maintenance needs or plant refit projects. Actual cycle lengths may vary from the plans for several reasons. Examples are: lower than anticipated operating capacity factor which encourages delaying an outage to improve fuel utilization, and time shifts to accommodate unexpected outage conflicts with other units on the power system. This chart is used as a concise reference for the timing and status of fuel deliveries and for planning required fuel quantities.

C. Edison's Nuclear Fuel Experience

Commonwealth Edison has had experience with nuclear power plant fuel accountability for over 20 years including a short period when nuclear fuel reprocessing was available. Since the first nuclear plant was placed in service in 1960, the Company

has accounted for 10,354 fuel assemblies. Of these, 1,653 discharged assemblies have been shipped off-site, 4,655 are in disfuel pools, 3,282 are in reactors, and 764 are in the plant fuel vaults. Of those in the pools, 3,749 are spent and 908 are awaiting reactor loading at the LaSalle County plant. These totals are as of May, 1980 (Table III). With reprocessing, the total inventory at the plants would be reduced by several thousand assemblies. Edison's studies of the loss of full core discharge capacity resulted in decisions to increase pool capacities by closer spacing and absorber type racks.

D. Reprocessing Results

The number and type of fuels shipped to reprocessors, reprocessed, and in storage is exhibited in Table IV. No fuel shipments have been made since 1974 due to the suspension of shipping cask licenses and the moratorium on reprocessing. The four Dresden #1 fuel assembly batches which have been reprocessed (Table V) were found to be in reasonable agreement with General Electric's calculated values; within 2% for total plutonium content and within 7% and 5% for residual enrichment and fissile plutonium, respectively. These results were obtained from less sophisticated calculational methods than those available today; however, no new reprocessing results have been available since 1969.

II. NUCLEAR POWER STATION SECURITY

A. Introduction

Commonwealth Edison has a Nuclear Security Administrator responsible for Edison's nuclear power station security programs, contingency plans and their implementation.

The Nuclear Regulatory Commission established, through regulation in Title 10, part 2.790, that the specifics of nuclear station security plans are not to be disclosed to the public, but only to those who have an established need for the information. For example, specifics such as the number of guards,

their available weapons, routes and frequency of patrols, and planned response measures, if made available to the public, could have a deleterious effect on security.

B. Security Program History/Requirements

Edison's original nuclear power station security program was based on 10 CFR 50.34 and placed into effect in 1974. It generally followed guidance presented in ANSI N 18 17, "Industrial Security for Nuclear Power Plants," and was accepted by the Nuclear Regulatory Commission. In May 1977, a new security plan was drafted in response to the promulgation of 10 CFR 73.55.* This plan was approved by the NRC on March 14, 1979. The general requirements of 10 CFR 73.55 are:

> The licensee shall establish and maintain an on-site physical protection system and security organization which will provide protection with high assurance against successful industrial sabotage by both of the following:
>
> (1) A determined violent external assault, attack by stealth, or deceptive actions of several persons with the following attributes, assistance and equipment: (i) well-trained (including military training and skills) and dedicated individuals, (ii) insider assistance which may include a knowledgeable individual who attempts to participate in both a passive role (e.g., provide information) and an active role (e.g., facilitate entrance and exit, disable alarms and communications, participate in violent attack), (iii) suitable weapons, up to and including hand-held automatic weapons, equipped with silencers and having effective long-range accuracy, (iv) hand-carried equipment, including incapacitating agents and explosives for use as tools of entry or otherwise destroying the reactor integrity, and
>
> (2) An internal threat of an insider, including an employee (in any position).

*The upgraded security program mandated by 10 CFR 73.55 has been implemented in stages. Initial steps following promulgation of the regulation in 1976 included increased numbers of armed guards, a reduction in public tours, and expanded guard training. Finally, the Commission has deferred until August 1, 1979 implementation of certain additional security measures relating to the threat of insider sabotage postulated in 10 CFR 73.55 (44 Fed. Reg. 11201, February 28, 1979).

Edison has met these general performance requirements through a station security program that includes:

Security Organization (10 CFR 73.55 (b))
Physical Barriers (10 CFR 73.55 (c))
Access Requirements (10 CFR 73.55 (d))
Communication Requirements (10 CFR 73.55 (f))
Response Requirements (10 CFR 73.55 (H)).

C. Security Organization

The security organization includes armed guards. One full-time individual has the authority and training to direct the activities of these guards and is on site at all times. All guards have been screened by polygraph examination, undergone psychological evaluation, and have met standards established for physical ability, hearing and vision. Each guard has been trained in accordance with State requirements, as well as the Security Rlan requirements, which are based on NRC Regulatory Guide 5.20 and are being changed to conform with Appendix B, 10CFR73. Pursuant to this training program each guard has received 80 hours of instruction including range qualification with a firearm prior to assignment to a station. Upon station assignment each new guard has received on-the-job training in the specific requirements of his security assignments. Annual requalification is required for initial training topics and for physical standards. Firearms requalification is accomplished semiannually.

D. Physical Barriers

10 CFR 73.55 (c) requires that plant areas where vital equipment is located by designated "Vital Areas." Vital equipment is defined as, "any equipment, system, device or material, the failure, destruction, or release of which could directly or indirectly endanger the public health and safety by exposure to radiation. Equipment or systems which would be required to function to protect public health following such failure, destruction or release are also considered to be vital."

The regulation also requires that a Vital Area be located within a "Protected Area" which is defined as, "an area encompassed by physical barriers and to which access is controlled." The plant barrier consists primarily of two 8-foot-high parallel chain-link fences with barbed wire topping. This fencing is monitored continuously by an alarm system and closed circuit television. The area around the Protected Area is lighted to a minimum of 0.2 footcandles, measured horizontally at ground level during non-daylight hours. The Protected Area is routinely patrolled by members of the guard force. The purpose of the fencing, alarm system, closed circuit television, lighting, and the guard patrols is to assure that anyone attempting to gain unauthorized access to the Protected Area is detected.

E. Access Requirements

Access to the Protected Area without an escort is permitted only for persons requiring such access for the performance of work within the plant and only after meeting established personnel screening requirements which include:

- Psychological testing and a background check, or;
- At least one year of employment and a background check, or;
- Verification of trustworthiness and reliability through at least three years of employment.

Within the Protected Area, vehicles are escorted by a guard unless designated for normal use on site. A vehicle designated for normal use on site may be operated only by authorized persons, and the keys are removed from the vehicle when it is not in use. All personnel and vehicles entering the Protected Area are searched for weapons, explosives, and incendiary devices. Metal and explosive detection equipment and physical searches are used.

The individuals responsible for controlling access to the Protected Area are protected by a bullet resisting structure within the main access control building. Each person entering

the Protected Area is provided with a badge for identification, which must be displayed while within the Protected or Vital Area.

Only authorized persons may enter a Vital Area. Vital Area access is controlled either by a lock and key system, a guard or a computerized electronic access control system operable by inserting a key card into a card reader located at an access control point. All doors which would provide access to a Vital Area are alarmed to detect any unauthorized entry. In accordance with 10 CFR 73.55 (e), these alarms are tamper-indicating and self-checking.

The access control measures and screening requirements described above apply not only to regular plant activities and plant employees, but also to any additional activities and individuals on site. For example, any vehicle delivering equipment or material will be searched and escorted by a guard within the Protected Area. All contractors' employees will be accompanied by an escort at all times during the job or will be screened in a manner equivalent to the program described above for company employees.

Table VI summarizes the plant access control system. The many forms and logs routinely used by the guards in the performance of their functions are reviewed and audited by plant management. Guards have and are trained in the use of equipment such as shown in Table VII.

F. Communication Requirements/Security Control Centers

All members of the security force at the station are capable of two-way voice communications between each other and with two Security Control Center operators using portable radio tranceivers. Additionally, each Security Control Center operator has the capability of communicating with the nearest Police Department using a direct radio link.

Two Security Control Centers are located within the plant Protected Area. These centers house consoles for monitoring

security communication, the alarm systems, closed circuit television, guard activities, and access controls. The centers are constructed to bullet resisting standards and each is continuously manned.

G. Response Requirements

Safeguards contingency plans, which supplement the station security plans were submitted to the NRC. The purpose, scope, and contents of these contingency plans reflect the requirements of Appendix C to 10 CFR 73. These contingency plans are presently under review by the NRC. Among other things, they identify events which could present a threat to the station such as Protected or Vital Area intrusion. They describe a predetermined set of response actions for each event and identify the individuals responsible for completing the appropriate actions.

H. Spent Fuel Security

10 CFR Part 73 does not specifically require that licensees design their security programs to prevent theft of spent fuel. This is because the weight and highly radioactive state of spent fuel assemblies would make removal of such assemblies from the station extremely difficult. Even if such removal took place, reprocessing would be necessary to retrieve special nuclear material in usable form from the stolen spent fuel. Nevertheless, the features of the security program would also be effective against theft.

The weight of a single fuel assembly varies from over 250 pounds for Dresden 1 fuel to over 1,000 pounds for Zion fuel (Table VIII). A diverter would need to use the plant's service crane (1/2 to 1 ton capacity) to lift and insert a fuel assembly into the heavily shielded shipping cask. Any movement of this cask would require the use of the plant's equipment crane (capacity 50 tons). All such activity, however, must be accomplished under water since the radioactivity of any given assembly would be in the range of 200,000 roentgens per hour even after significant decay periods.

III. NUCLEAR MATERIAL SAFEGUARDS

A. Introduction

Essentially all nuclear materials in the possession of utility companies are contained in full-size nuclear fuel assemblies. Each fuel assembly is uniquely identified with an embossed serial number which it retains throughout its lifetime until reprocessing. Fuel assemblies are valuable assets kept under rigid physical control.

The movements of fuel assemblies in the possession of a utility include the following: the receipt of new fuel assemblies from a fuel fabricator, transfers to fuel vaults, transfers to fuel pools in preparation for insertion into a reactor, transfers to reactors, movements made between reactors and fuel pools during refueling, and storage of spent fuel in fuel pools awaiting reprocessing or other disposition. Each movement is preplanned and the actual movements are controlled and documented by designated station personnel.

B. Nuclear Material Control System - General Features

1. Introduction. Edison's nuclear material control system is set forth in written nuclear procedures authorized by executives of the Company. The procedures comply with Nuclear Regulatory Commission regulations and the American National Standard Nuclear Material Control System. An essential requirement of an adequate material control system is the organization and delegation of responsibilities. To accommodate this, the various responsibilities involved are assigned among different areas of the Company so that there is a built-in internal checking of all activities. The areas involved include accounting and finance, station personnel, internal audit, nuclear fuel data bank, quality assurance, security, nuclear licensing and fuel purchasing. The primary duties of those areas most important to fuel accountability will be discussed.

2. Organization/Responsibilities. A list of individuals responsible for nuclear materials safeguards is given in Table IX. The functions of the principal people are explained below. A Nuclear Materials Safeguards Manager is appointed by the President and reports to the Comptroller of the Company. His major responsibilities are to implement the control system as prescribed by the Company's written nuclear procedures, to perform the necessary accounting for nuclear fuel, and to prepare and file nuclear material transaction reports and status reports as required by the NRC.

Nuclear generating Station Superintendents and their assistants are given the responsibility for the overall physical control of nuclear material at their respective stations and the implementation, administration, and compliance with the Company's nuclear procedures, security plan activities, and NRC license and other applicable regulations.

A Nuclear Materials Custodian for each generating station is designated by the Station Superintendent. The custodian is responsible for all control functions at the plant site. This includes the responsibility for all movements of fuel assemblies and physical inventories of assemblies within the Item Control Areas assigned to him. He prepares all necessary reports to verify receipts of fuel assemblies within the station site. He promptly reports the results of all physical inventories and all fuel movements to the Nuclear Materials Safeguards Manager.

Internal auditors are required to be present and observe physical inventories of nuclear fuel assemblies. They also perform regular audits of the nuclear material control system to ensure compliance with the Company's nuclear procedures.

Quality assurance personnel perform periodic audits to insure compliance with the security plan, licenses, and all related procedures including nuclear plant operating and fuel handling procedures.

The Nuclear Security Administrator has the responsibility to develop security plans for each location and to assist in the implementation of procedures.

3. Inventories. Physical inventories of irradiated fuel assemblies by serial numbers are performed by several sets of observers including internal auditors. Fuel assemblies in reactors are always verified with binoculars or periscope devices immediately prior to replacing the reactor head cover. Inventories of fuel assemblies in fuel pools are made on a regular basis by making piece counts and by verification of serial numbers. Physical inventories of new fuel assemblies in fuel valuts are made at regular intervals.

4. Physical Security. The station security system is described in detail in section II. Unirradiated nuclear fuel is stored in fuel vaults. A typical fuel vault is located below floor level on the refueling floor and is secured by concrete covers over metal grates. These concrete covers are extremely heavy and can only be lifted by use of an overhead crane. If the fuel contains plutonium, additional protection is provided by an intrusion alarm or patrol by a guard or watchman. Access to the refueling area is limited to the specifically authorized personnel. If unirradiated fuel cannot be immediately placed in a fuel vault, the fuel is temporarily stored in a location that has closed circuit surveillance and is regularly patrolled by a guard or watchman.

5. Nuclear Fuel Data Bank. The Company's Nuclear Fuel Data Bank is a computerized system which keeps a complete record of historical and current data for each fuel assembly. This includes information for each fuel assembly on the original and current element and isotopic composition, the computation of nuclear fuel burnup, the heat production, the physical location, the original cost, charges to fuel expense based on actual heat production, and the accumulated amortization to date.

The Nuclear Fuel Services Department computes the burnup and residual uranium and plutonium in irradiated fuel assemblies by use of complex computer codes and correlates such information

with process computer data. The data bank generates information required for material status reports. The data bank is used to correlate accounting and financial information with the nuclear material data and statistics. The Nuclear Materials Safeguards Manager's staff checks all physical movements of fuel assemblies and physical inventories with data bank printouts and reviews all burnup data, amortization of fuel costs and other relevant data.

C. Safeguarding Fuel Vaults

The safeguards protection required for fuel vaults varies depending on their construction and location.

The Dresden 1 fuel vault is locked with a three-position manipulation resistant, dial-type, combination padlock at all times except when authorized work is in progress.

The fuel vaults at other reactors are secured by concrete block covers or locked metal grates at all times when fuel is present except when authorized work is in progress. The covers are put back in place at the end of each shift if no authorized work is planned for the following shift.

Vaults are protected by intrusion alarms to summon guards or watchmen. The intrusion alarm is inspected and tested at intervals not exceeding seven days. Records of such tests and inspections are kept by the NM Custodian.

Where an intrusion alarm is not operable, the vault will be protected by a guard or watchman. The guard or watchman records the time and results of his inspection of the fuel vault in an appropriate log. The patrol log and the record of intrusion alarm tests are made available for examination by representative of the Company's Auditor, the Manager of Quality Assurance, and the Nuclear Security Administrator.

D. Nuclear Fuel Assembly Records and the Nuclear Fuel Data Bank

1. Introduction. The Nuclear Material Custodian of each nuclear generating station maintains records of all fuel inventories, receipts, shipments, transfers, and all other records deemed necessary by the Nuclear Material Safeguards Manager. These pertain to nuclear fuel safeguards required by the NRC and/or the International Atomic Energy Agency (IAEA).

The Nuclear Fuel Data Bank (NFDB) includes records for all nuclear fuel assemblies in the possession of the Company. Such records shall include: dates of receipts, transfers and shipments, quantities of nuclear materials received, fissioned, produced, transferred and shipped, exposure inspection results, and the audit dates of annual and reactor inventories. The NFDB history file contains records of all fuel transactions. In addition, the NFDB maintains a real-time data base consisting of current information for all of the fuel at the nuclear stations. These data files are utilized by various departments within the company to prepare internal fuel audit reports, internal accounting reports from the Comptroller's office and official fuel accountability documents delivered to the Nuclear Regulatory Commission (NRC).

The information contained in the NFDB can be divided into three categories. These are fuel location information, burnup data including isotopic inventories, and fuel cost accounting information. The procedures used to verify the information in each of these categories are summarized below.

2. Fuel Location Information. The verification procedures listed here are used to validate NFDB information pertaining to the location of fuel. The fuel transfer lists prepared at the station, verified by the Nuclear Materials Custodian, and approved by the Station Superintendent are sent to the NFDB. The Company audit staff and outside auditors provide audit checks of the fuel locations approximately twice each year for

the pools and vaults. Each reactor is inventoried at the conclusion of its reload outage. Audit records are sent to the NFDB. Fuel location lists from the NFDB and from station audits are checked by the Audit Staff and by the NFDB Staff to ensure that the station and the NFDB are in agreement. The Comptroller's Staff verifies the piece count of these audits and of monthly pool and vault inventories.

3. Burnup Data. Burnup information contained in the NFDB is composed of fuel assembly exposure data and the concentrations of specified isotopes by assembly. This information is obtained from the process computers for the large Boiling Water Reactors. The Dresden 1 and Zion reactors utilize off-line computers to generate this information.

Each month the Process Computer Dump Tapes containing burnup data for Quad Cities Units 1 and 2 and Dresden Units 2 and 3 are processed directly by NFDB programs. At 6-month intervals this information is verified by Nuclear Fuel Services (NFS) using a three-dimensional power exposure and isotopics program. Verification is also made by the fuel supplier. Each month the burnup data for the Dresden 1 reactor is computed by NFS using similar computer programs. Each month the burnup data for the Zion reactors are computed using programs developed by Westinghouse. The calculated discharge assembly exposures are verified by NFS. This flow of data is described in Fig. 3.

4. Fuel Cost Accounting Information. Fuel accounting information includes the initial cost and estimated salvage value of the fuel assemblies and current and cumulative amortization. The capitalized allowance for expenditures during construction, use taxes, investment tax credits, and rates per megawatt day for each write-off are also included. The fuel accounting information is prepared and coded by the Comptroller's Staff, verified by the NFDB Staff, and utilized by an NFDB processing program to update the NFDB data files. After fuel accounting

updates are completed, the accounting reports are reviewed by the Comptroller's Staff to insure that the proper information was processed by the NFDB Staff.

E. Inventory and Discrepancy Reports

The Comptroller's Staff prepares the NRC Material Status Reports and Nuclear Material Transaction Reports required, obtains the approval of the NMS Manager or other corporate officer, and submits the reports and schedules to the Nuclear Regulatory commission. Typical NFDB reports and their users are identified on Table X. Figure 4 is an example of a 742 semi-annual report to the NRC.

If any discrepancy is discovered between station records and physical inventories or if fuel is observed to be in an unauthorized or unprotected location or condition, the Nuclear Material Custodian, other station personnel, or company auditor must report this immediately to the Station Superintendent and to the Nuclear Material Safeguards (NMS) Manager. The Station Superintendent must investigate each reported incident immediately and advise the NMS Manager of the progress of the investigation. If a reported incident is not resolved within 24 hours or if the investigation shows that fuel is missing, the NMS Manager must immediately notify the appropriate company executives and the U.S. Nuclear Regulatory Commission.

F. New Fuel Assembly Supply

Fuel design fabrication and shipment is the responsibility of the vendor. This includes specification as to quantities of uranium and its enrichment and number of assemblies required to meet the utility's requirements. Edison makes all arrangements for conversion, enrichment, and UF_6 delivery to the fuel fabricator. The utility must forecast its needs at least 1.5 years prior to a scheduled refueling to assure flexible scheduling of conversion, enrichment, and fabrication to meet a fixed

delivery schedule. A considerable amount of time and analytical effort is required prior to fixing the delivery dates and the quantities of fuel material required. Much of this is in the preparation of the reload analyses required for license submittals to the NRC which confirm the safety operation of the reactor during the related cycle.

Edison's Quality Assurance Department personnel periodically visit the vendor's fabrication facilities to observe and audit quality control and assurance procedures including uranium material balance records. Edison's Nuclear Fuel Services Department personnel meet with the vendor's reload design group during the latter stages of the reload design safety analysis studies and license submittal preparation to assure its adequacy for transmittal to the NRC, and to participate in the decision making relative to alternative reload design before the final design alternative is selected.

G. Fuel Assembly Identification

Fuel assembly identification is provided by a six-character numbering system consisting of a prefix of two alphabetic characters, which identify the individual fabrication facility, followed by a serial number consisting of 4 alpha-numeric characters. The NRC assigns a three-character code to each fabrication facility. The last two characters of this code are those used in the prefix to ensure that no two fuel assemblies manufactures in the United States have the same number. The following four-character alpha-numeric characters must be assigned without repetition to individual assemblies by the fuel fabricator who is responsible for maintaining records of assignments. This identification number is cast, machined or engraved on the end fitting which is fastened to the assembly by a mechanical means (Fig. 5). It provides a readily accessible and legible identification suitable for underwater viewing via use of above-water binoculars, underwater borescopes, and TVs with video tape recording.

H. Fuel Research and Development

Over the years since Dresden #1 has been in operation, both its initial and subsequent fuel vendor and various National Laboratories have asked for irradiated fuel assemblies or portions thereof for hot lab studies of irradiation effects. Metallurgical properties of cladding, pellet densification, fission gas release, and fission products and isotope distribution were among the properties of interest. These test programs are summarized in Fig. 6.

New fuel assemblies which are damaged during shipment or other handling or do not pass inspection are repaired by the vendor on-site or returned to the factory for repairs with accountability documentation.

Some irradiated fuels which failed in service during Edison's early experience were reconstituted on-site by interchanging damaged fuel rods with equivalent partially-used sound rods. Fuel rods, segments or fragments which could not be installed in the scavenged assembly were specially packed in separate containers awaiting resolution of disposal. The total history of each assembly subject to such activity is available and easily retrievable through use of Nuclear Power Station records. Each rod has its own serial number stamped on its end plug. Reconstitution of assemblies, though of minimum interest currently, may be required in the future if fuel supply becomes unstable.

I. Fuel Assembly Receipt and Verification

Vendors are required to furnish Edison with a "RELOAD LICENSING SUBMITTAL" and assistance in obtaining necessary amendments to the NRC Operating License allowing use of the vendor's fuel. They are also required to attach to each Nuclear Material Transaction Report (Form NRC/DOE-741) a certified statement of the special nuclear materials contained in each fuel assembly.

A license to receive fuel at the particular station must be obtained prior to shipment of the fuel by the vendor.

In the event a particular nuclear fuel shipment fails to arrive within 24 hours of the estimated time of arrival, the Nuclear Materials Custodian must immediately notify the Station Superintendent, Purchasing, the NMS Manager, and the vendor.

The nuclear fuel shipment is kept under tight security until it is unloaded, uncrated, and verified. Unloading routines have been developed to ensure that all nuclear fuel assemblies are verified and receiving reports forwarded to the NMS Manager. The shipment is inspected against the shipping papers presented by the carrier's driver. Discrepancies (damaged or missing containers) on the shipping papers are reported to the NM Custodian, Station Superintendent, NMS Manager, and Purchasing-Fuel.

Unloading, uncrating, verification of serial numbers, and transfer of the fuel assemblies to the fuel vault or fuel pool is conducted by Fuel Handlers. As each shipping container is opened, they verify the serial number of each fuel assembly and enter a check mark on a copy of the packing list. Tags are made for all fuel assemblies included on the packing list. The Nuclear Material Transaction Report (Form NRC/DOE-741) contains a statement showing the enrichment and weight of the nuclear materials contained in each fuel assembly.

J. Fuel Inventories

A physical inventory by serial number and location of all nuclear fuel rods, assemblies, or other special nuclear material in each fuel vault (unirradiated fuel) is taken at least once every month. A physical inventory need not be taken of vaults that have remained sealed since a previous inventory provided not more than 12 months have elapsed and the integrity of the seal is confirmed.

A physical inventory of nuclear fuel rods or assemblies in each fuel pool (irradiated fuel) is taken by serial number and

location at least once every six months and after each refueling of the related reactor. A piece-count inventory of the nuclear fuel rods or assemblies in each fuel pool is made at least once each month.

A physical inventory of all nuclear fuel assemblies in the reactor is made immediately prior to installing the reactor head upon completion of the initial fuel loading and immediately prior to each subsequent replacement of the reactor head. Whenever the head of a reactor vessel is removed for more than one month, a concurrent piece-count inventory of the fuel assemblies in the reactor and contiguous fuel pool(s) is made at least once in each month the head is removed.

K. Movements of Fuel Assemblies

1. Transfers to or from Reactors. A tentative loading and refueling plan is prepared by the core management engineers along with a description of the nuclear fuel related events which will take place during the outage. Before removing a reactor vessel head for refueling or core alteration, a description is obtained of the core unloading, sipping, and core reloading. The execution of the steps of the Nuclear Fuel Transfer (Form C) is performed in accordance with Station Operating Procedures. A computer program to develop an optimized fuel movement sequence with printout instructions for the engineers and fuel handlers has been developed at Edison. Figure 7 is a sample printout.

2. Transfers Between or Within Fuel Vaults and Pools. The execution of the steps listed in the Nuclear Fuel Transfer List is verified and the Tag Board is updated. A physical inventory of each vault and pool is periodically taken while fuel movements are in progress by filling in a blank copy of the appropriate inventory maps. The changes in location of fuel assemblies, by serial number, are entered on a working copy of the fuel vault and/or fuel pool maps, as appropriate.

IV. MANPOWER, TRAINING, EQUIPMENT, AND COSTS

A. Manpower and Training

The total manpower in each station is shown in Table XI. Only a few of these are involved even part-time in nuclear materials safeguards and accountability. The primary responsibility for fuel accountability at the stations is given to the nuclear engineers assigned to the reactor and specifically to the Nuclear Material Custodian. Similarly, Table XII identifies the General Office involvement which is also a part-time effort for all but the Nuclear Fuel Data Bank personnel.

The education received by accountants, nuclear engineers, and fuel handlers related to material safeguards has primarily been obtained on-the-job. This is supplemented by vendor, station, and general office personnel lectures related to fuel handling, inspection, and verification of identity and location. Major formal documents used in training are identified in Table XIII along with the group responsible for their updating.

Techniques of reactor analysis reload planning and fuel management have been provided under contract with vendors for nuclear engineers involved in such activities. Retraining and requalification of reactor analysts is described in Edison's Quality Assurance Manual.

B. Equipment

Equipment for the direct verification of initial uranium or isotopic content of any new fuel assembly for the residual uranium and isotopic content of an irradiated fuel assembly is not readily available for use in an operating plant. The initial uranium and isotopic content is based on vendor certification and the utility's knowledge of the vendor's design and quality assurance techniques. Its in-reactor performance is directly related to its isotopic and fixed poison distribution which must be carefully controlled. Also, its discrete position within the core and relationship to other fuels in its vicinity is

important. Its residual content is calculated based on both vendor and Edison's off-line computer programs and analytical techniques. The accuracy of these programs has been improved as information from minimum critical testing and reprocessing becomes available. Nuclear power unit operation data which is evolved via use of on-line process computers provides the variational input needed for use of the general purpose off-line computers (Table XIV).

Visual and photographic means such as hand-held binoculars, underwater borescopes, and underwater TV video-taping machines are used at the plant site for verification of fuel assembly position, orientation and audits. The resulting records consist of lists, maps, tag boards, and video-tape records.

C. Costs

Since 1977, the capital cost and annual expenses assignable to plant security is about 6 million dollars for equipment and 3 million dollars for manpower and training, for each of Commonwealth Edison's Nuclear Power Stations. That portion of the above costs specifically assignable to Nuclear Material Safeguards is impossible to assess. Each of Edison's fuel assemblies is considered a unique capital asset and is accounted for in essentially the same manner as if it were a discrete piece of equipment. The Company's procedures, accounting methods, and controls assure that it doesn't disappear or become lost.

Most accountability expenses would occur even though no material safeguard regulations were in force. The knowledge of each fuel assembly's history in terms of location, orientation, type, initial enrichment, and usage throughout its residence in a given reactor is necessary for safe and economic reactor operation. That fuel which is designated spent and in the pool awaiting disposal or reprocessing is of economic interest in terms of residual fuel value, and recycle potential. Fuel shipped for disposal in a high-level waste facility or reprocessed must therefore be completely identifiable to allow shipment and/or reprocessing. The data obtained by reprocessing in

the future will be used to further assess the adequacy of computer programs and to provide benchmarks for their further improvement.

V. CONCLUSION

Nuclear fuel assemblies and their uranium and isotopic content are documented starting with the power company's quality assurance inspector visiting the fuel fabrication facility. As each fuel assembly is received at the nuclear power plant, it is carefully inspected along with its detailed shipping papers and records. While at the power plant, its various location in the reactor, vault, and pool are frequently observed, recorded, and audited. Its changing isotopic content is computed and careful records maintained. At specified intervals, these records are reviewed and audited and status reports prepared for internal Company use and for submission to the Nuclear Regulatory Commission.

Practically all of the procedures and surveillance actions to accomplish the above are necessary for nuclear fuel accountability and cost accounting as a valuable and depletable asset and also to ensure the operational safety of the reactor and plant. Specific security actions at the plant necessitated by the presence of the fuel alone are not required. Provisions for overall plant security also encompass these needs for fuel. Special reporting requirements of the Nuclear Regulatory Commission for nuclear fuel are essentially the only additional commitment which must be met.

TABLE I

COMMONWEALTH EDISON COMPANY

NUCLEAR POWER REACTORS

NUCLEAR POWER STATION	UNIT	CUMULATIVE NUMBER UNITS	YEAR OPERATIONAL	RATING* MWe	URANIUM* CORE WEIGHT TONS	CORE ASSEMBLIES NUMBER
Dresden	#1	1	1960	200	50 STU	464
Dresden	#2	2	1970	800	150 STU	724
Dresden	#3	3	1971	800	150 STU	724
Quad Cities	#1	4	1972	800	150 STU	724
Quad Cities	#2	5	1972	800	150 STU	724
Zion	#1	6	1973	1100	89 MTU	193
Zion	#2	7	1974	1100	89 MTU	193
LaSalle	#1	8	1981	1100	164 STU	764
LaSalle	#2	9	1982	1100	164 STU	764
Byron	#1	10	1982	1100	89 MTU	193
Byron	#2	11	1983	1100	89 MTU	193
Braidwood	#1	12	1983	1100	89 MTU	193
Braidwood	#2	13	1984	1100	89 MTU	193

* NOMINAL

TABLE II

COMMONWEALTH EDISON COMPANY

NUCLEAR FUEL SUPPLY COMMITMENTS

UNIT	NET CAPABILITY (MEGAWATTS)	URANIUM CONCENTRATE	CONVERSION	ENRICHMENT	FABRICATION
Dresden 1	207	1990	1981	2008	1984
Dresden 2	794	1990	1981	1998	1986
Dresden 3	794	1990	1981	1998	1986
Quad Cities 1	591	1990	1981	1998	1987
Quad Cities 2	592	1990	1981	1998	1987
Zion 1	1,040	1990	1981	1999	1982
Zion 2	1,040	1990	1981	1999	1982
LaSalle County 1	1,078	1990	1981	2000	1986
LaSalle County 2	1,078	1990	1981	2000	1986
Byron 1	1,120	1990	-	2000	1991
Byron 2	1,120	1990	-	2000	1992
Braidwood 1	1,120	1990	-	2010	1992
Braidwood 2	1,120	1990	-	2011	1993

TABLE III

COMMONWEALTH EDISON COMPANY

NUCLEAR FUEL SERVICES INVENTORY

5-1-80

	DRESDEN	QUAD CITIES	ZION	LASALLE	TOTALS
Vaults	76	0	68	620	764
Reactors	1448	1448	386	0	3282
Pools	2111	1268	368	908*	4655
Shipped					
NFS N.Y.	892	-	-	-	892
S.R. S.C.	8	-	-	-	8
G.E. Morris	753	-	-	-	753
	5288	2716	822	1528	10354

* New Fuel Assemblies
All other fuels in pools are reactor discharged fuels.

NFS N.Y. - Nuclear Fuel Services, New York

S.R. S.C. - Savannah River, South Carolina

TABLE IV

COMMONWEALTH EDISON COMPANY

REPROCESSING BATCH HISTORY

DRESDEN #1 ASSEMBLIES

	BATCHES REPROCESSED[a]					
ASPECT	1	2	3	4	5	6
Ownership	AEC	AEC	AEC	AEC	AEC	AEC
No. of Assemblies	184	8[b]	188	109	(b) (c)	202
Where Shipped	NFS	SR	NFS	NFS	SR	NFS
Year Shipped	1965 1966	1969	1966 1967	1966	1969	1968 1969
Cladding	Zr-2	S.S.	Zr-2	S.S.	S.S.	Zr-2
Fuel	UO_2	UO_2 ThO_2	UO_2	UO_2	ThO_2	UO_2

(a) LEASED

(b) STATUS UNKNOWN

(c) THORIA RODS REMOVED FROM ASSEMBLIES INCLUDED IN REPROCESSING BATCH 4. (952 WHOLE ELEMENTS AND 17.861 ELEMENT EQUIVALENTS).

SYMBOLS

AEC - Atomic Energy Commission
NFS - Nuclear Fuel Services, New York
S.R.- Savannah River, South Carolina

TABLE V

COMMONWEALTH EDISON COMPANY

DRESDEN #1

REPROCESSING BATCHES

CALCULATED VS MEASURED

ISOTOPICS

REPROCESSING BATCH NUMBER	CONTENT	TOTAL URANIUM GRAMS	FISSILE CONTENT U-235 GRAMS	FISSILE CONTENT Pu(f) GRAMS	TOTAL PLUTONIUM GRAMS
1	Calculated	20,398,421	187,562	55,082	68,461
	Measured	20,399,999	192,222	55,855	67,696
	Deviation,%*	+0.0077	+2.424	+1.384%	-1.130
3	Calculated	20,714,413	146,791	67,485	92,099
	Measured	20,680,841	152,671	70,940	92,952
	Deviation,%*	-0.1623	+3.851	+4.870	+0.9177
4	Calculated	8,881,104	154,348	24,937	29,848
	Measured	8,936,552	158,542	25,899	30,368
	Deviation,%*	+0.6205	+2.634	+3.707	+1.712
6	Calculated	21,488,213	140,957	74,377	105,827
	Measured	21,543,965	151,647	77,757	105,165
	Deviation,%*	+0.2587	+7.049	+4.347	-0.629

* Deviation of Measured Values from Calculated Values in Percent of Measured Values.

TABLE VI

NUCLEAR POWER STATION ACCESS CONTROL SYSTEM

Controlled Area Facility Protection

Guard
Padlocks
Magnetic Key Card Locks
Interlocked Doors

Automatic Computerized Logging System

Ingress/Egress Requirements

Approval of shift engineer logged into computer.

Use of magnetic keycard to open specified doors.

Immediate door closure upon entry or exit.

Alarm set off if above procedure is not adhered to.

Monitored by master and backup computers, many local process monitors, and intrusion alarms.

Access to Controlled Areas for Operations/Work

Requires: Approval by shift engineer
Rad key logged out by shift engineer
Work permit signed by shift engineer.
Adherence to radiation protection specifications.

Radiation limits time vs levels
Radiation monitors/badges/pencils
Protective clothing/masks.

TABLE VII

TYPICAL SECURITY EQUIPMENT

Guard House

Screening Equipment	Control
Metal detectors	Duress buttons
Nitrate sniffers	Gate house doors interlocks
X-rays	Truck gate lock

Guard

Guard Communication Equipment

Telephones
Intercoms
Walkie talkies (radio)
Alarms

TABLE VIII

TYPICAL FUEL ASSEMBLY CHARACTERISTICS

RATING MWe	REACTOR TYPE	ASSEMBLY Kg	ASSEMBLY LENGTH* INCHES	ASSEMBLY WIDTH AND DEPTH INCHES	RODS ARRAY	RODS URANIUM
200	BWR	103	108/134	4.3	6x6	35
800	BWR	186	145/171	5.4	8x8	62
1100	BWR	183	150/176	5.4	8x8	62
1100	PWR	461	144/156	8.4	15x15	225

* URANIUM ACTIVE LENGTH/OVER ASSEMBLY LENGTH

TABLE IX

SAFEGUARDS DEPARTMENT: GROUPS AND INDIVIDUALS INVOLVED IN NUCLEAR MATERIAL ACCOUNTABILITY AND SAFEGUARDS

GENERAL OFFICE	PERSONNEL
Vice President	Licensing Administrator Nuclear Security Administrator
Purchasing	Manager of Fuel and Budgets
Accounting	Comptrollers Staff & Auditors Nuclear Materials Safeguard Manager Independent Public Accountants
Quality Assurance	Manager of Quality Assurance Nuclear Materials Safeguard Engineer
Nuclear Fuel Services/ Nuclear Fuel Data Bank Reactor Analysis	Director & Staff Nuclear Engineers

NUCLEAR POWER STATION	PERSONNEL
Management	Station Superintendent
Operations	Shift Engineer
Safeguards	Nuclear Material Custodian
Fuel Handling	Shift Foreman
Core Management	Nuclear Engineers
Radiation Chemistry	Supervisor/Foreman
On-Site Review Function	On-Site Review Personnel

TABLE X

NUCLEAR FUEL DATA BANK

REPORTS/LISTS/MAPS

REPORTS	PRIMARY SOURCE & USERS						
	Sta.	NFS	PUR	ACC	AUD	NRC	IAEA
FUEL ASSEMBLY							
Receipts NRC ERDA Form 741	S	-	U	U	-	U	U
Inspections	S	-	U	-	-	-	-
Locations	S	U	-	U	-	-	-
Status NRC ERDA Form 742	S	U	-	U	-	U	U
Shipments NRC ERCA Form 741	S	-	U	U	-	U	U
REACTOR CORE/ASSEMBLY							
Exposure & Isotopics							
Beginning of Cycle	S	U	-	-	-	-	-
Monthly	S	U	-	U	-	-	-
Semi-Annually	-	U	-	U	-	U	U
End-of-Cycle	-	U	-	-	-	-	-
Discharge	-	U	U	U	-	-	-
MAPS							
Vault	S	-	-	U	U	U	U
Pool	S	U	-	U	U	U	U
Reactor	S	U	-	U	U	U	U
Shipping Cask	S	-	U	-	U	U	U

SYMBOLS

S - Data Source
U - Data User

SOURCE & USER IDENTITY

STA - STATION
NFS - NUCLEAR FUEL SERVICES
PUR - PURCHASING
ACC - ACCOUNTING
AUD - AUDITORS
NRC - NUCLEAR REGULATORY COMMISSION
IAEA - INTERNATIONAL ATOMIC ENERGY AGENCY

TABLE XI

NUCLEAR POWER STATION MANPOWER

NUMBER OF PERSONNEL

PERSONNEL	STATIONS IN OPERATION			STATIONS UNDER CONSTRUCTION		
	DRESDEN	QUAD CITIES	ZION	LASALLE	BYRON	BRAIDWOOD
Guards(1)	Many	Many	Many	Some(2)	Some(2)	Some(2)
General Plant	433	231	303	242	151	123
Licensed Personnel & Operators	73	57	50	(58)*	-	-
Radiation Protection & Chemistry	40	38	38	22	18	-
Fuel Handlers	15	11	8	5	-	-
Nuclear Engineers	6	5	4	-	4	4
Nuclear Material Custodian	1	1	1	-	1	1
TOTAL STATION	568	343	404	327	169	128

(1) Not included in Total Station Complement

(2) Construction Security

* only 9 of the 58 men currently on site licensed on other reactors.

TABLE XII

NUCLEAR MATERIAL SAFEGUARDS

GENERAL OFFICE MANPOWER

PERSONNEL	CURRENT MANPOWER		
	BOILING WATER REACTORS	PRESSURIZED WATER REACTORS	COMBINED REACTORS
PURCHASING	1	1	-
QUALITY ASSURANCE	1	1	-
NUCLEAR FUEL SERVICES			
REACTOR ANALYSIS	7	3	-
NUCLEAR FUEL DATA BANK	-	-	2
METALLURGICAL	-	-	1
SAFETY	2	2	1
COMPUTER PROG. DEV.	-	1	-
LICENSE ADMINISTRATORS	1	1	-
ASSISTANTS	2	2	-
ACCOUNTING PERSONNEL & AUDITORS	2	2	1
NUCLEAR MATERIALS ADMINISTRATOR	-	-	1
NUCLEAR MATERIALS SAFEGUARD MGR.	-	-	1
NUCLEAR MATERIALS SAFEGUARD ENG.	-	-	1

TABLE XIII

EDISON'S DOCUMENTS USED IN THE TRAINING OF NULCEAR POWER STATION PERSONNEL

	UPDATE RESPONSIBILITY					
DOCUMENTS	NPSS	NFS	PD	LIC	QA	ACC
EQUIPMENT MANUALS	X	---	--	---	--	---
STATION PROCEDURES	X	---	X	---	X	---
TECHNICAL SPECIFICATIONS	X	X	X	X	--	---
QUALITY ASSURANCE MANUAL	----	---	--	---	X	---
NUCLEAR PROCEDURES	----	X	--	---	--	X

SYMBOLS

NPSS - NUCLEAR POWER STATION STAFF
NFS - NUCLEAR FUEL SERVICES DEPT.
PD - PRODUCTION DEPT.
LIC - LICENSING
QA - QUALITY ASSURANCE DEPT.
ACC - ACCOUNTING DEPT.

TABLE XIV

GENERAL OFFICE EQUIPMENT

NUCLEAR FUEL SERVICES	NOW	ADDITION SOON
Time Sharing Scopes	4	3
Microfiche Readers	3	8
Word Processors	1	
COMPUTER SYSTEMS		
General Purpose Computers		
IBM 3033's	2	
Disk Packs 3330 & 3350	Many	
Tape Drives* 7 & 9 Track	*	
Key Punchers	Many*	
Card Readers*	Few	
PROCESS COMPUTER RESEARCH & DEVELOPMENT		
GE (Honeywell) 4010	1	
Prime Corp. Prime 750	1	

* Extensive Peripheral Equipment

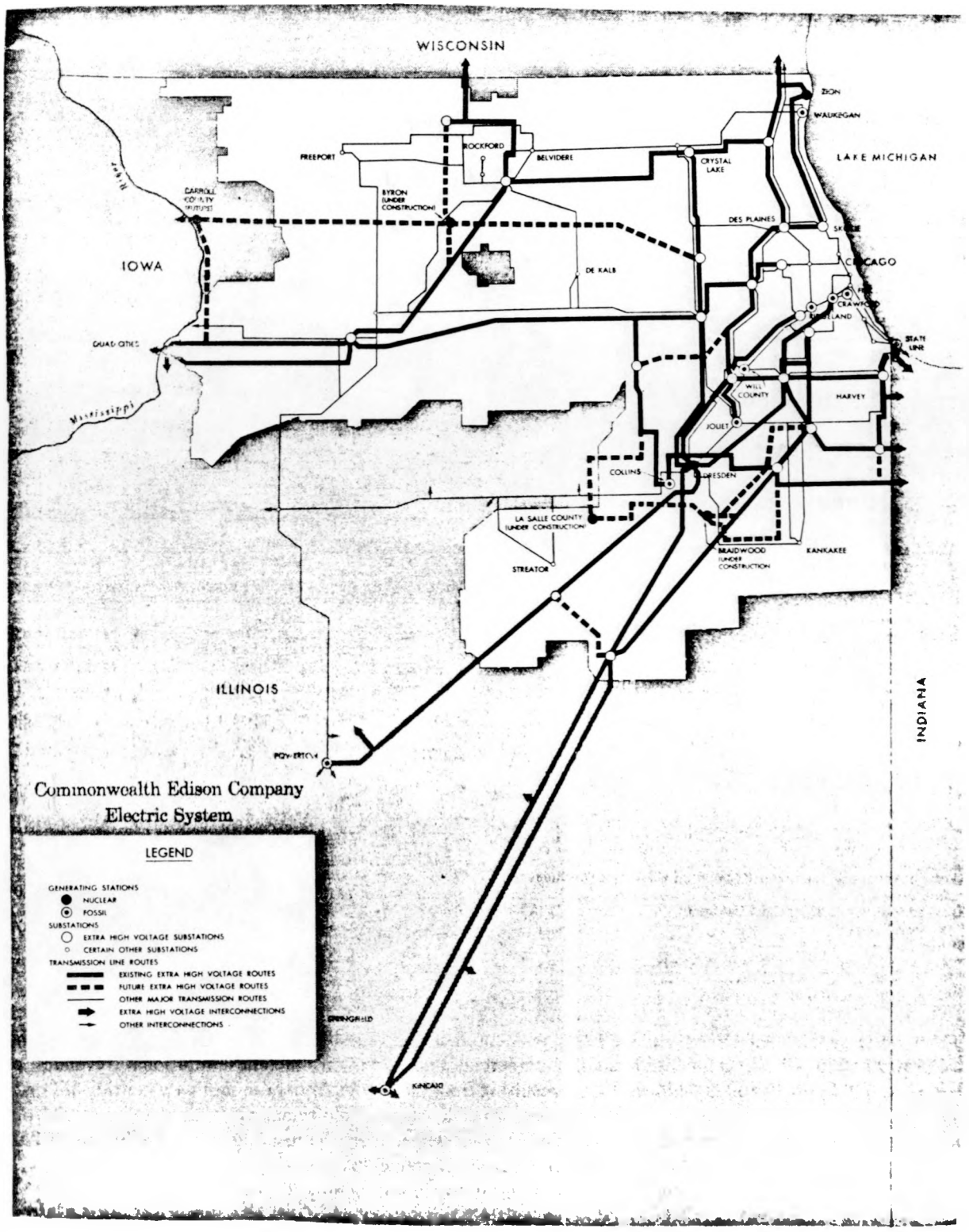


Fig. 1.
Service area of the Commonwealth Edison Company.

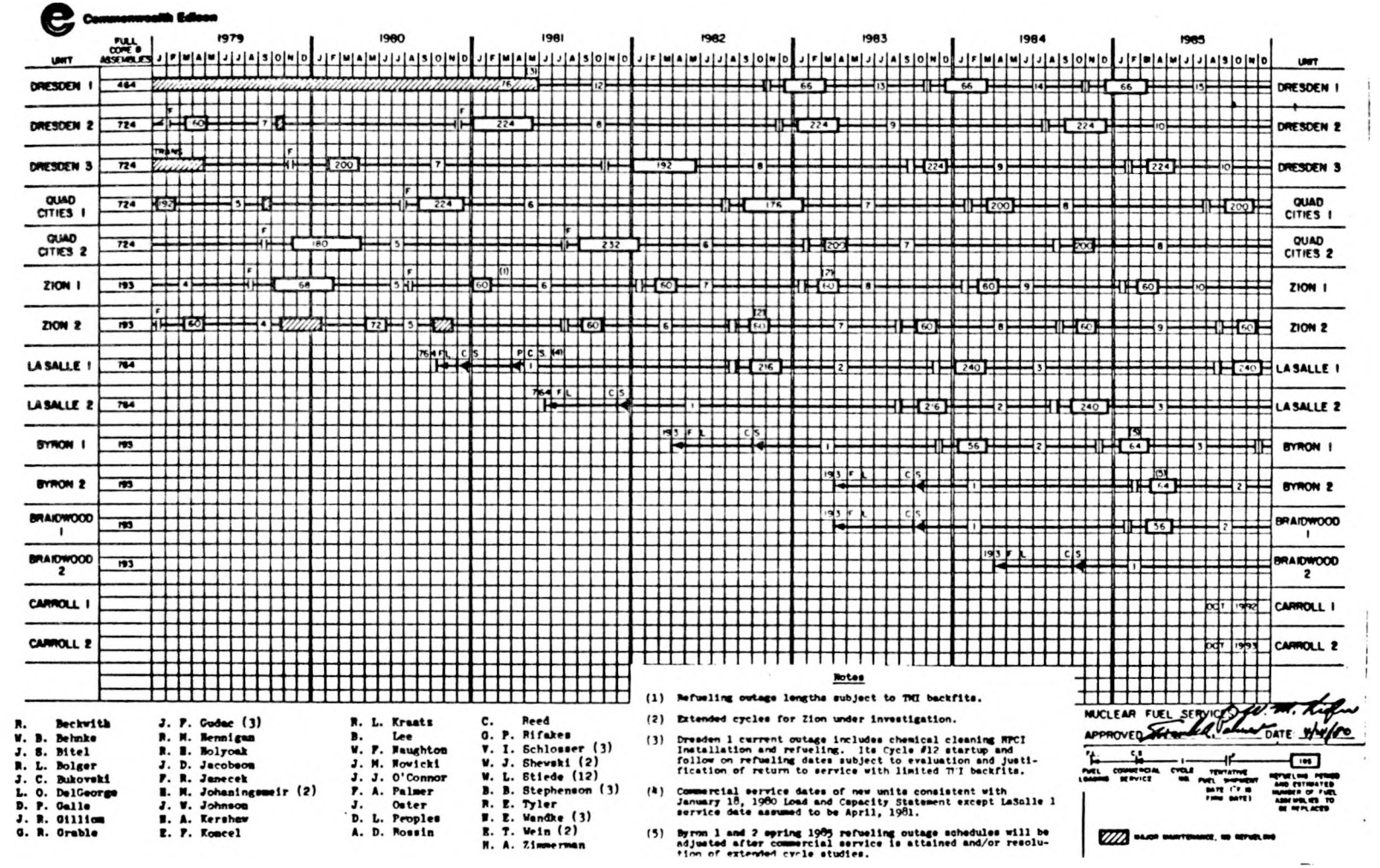


Fig. 2.
Summary of nuclear station refueling schedules.

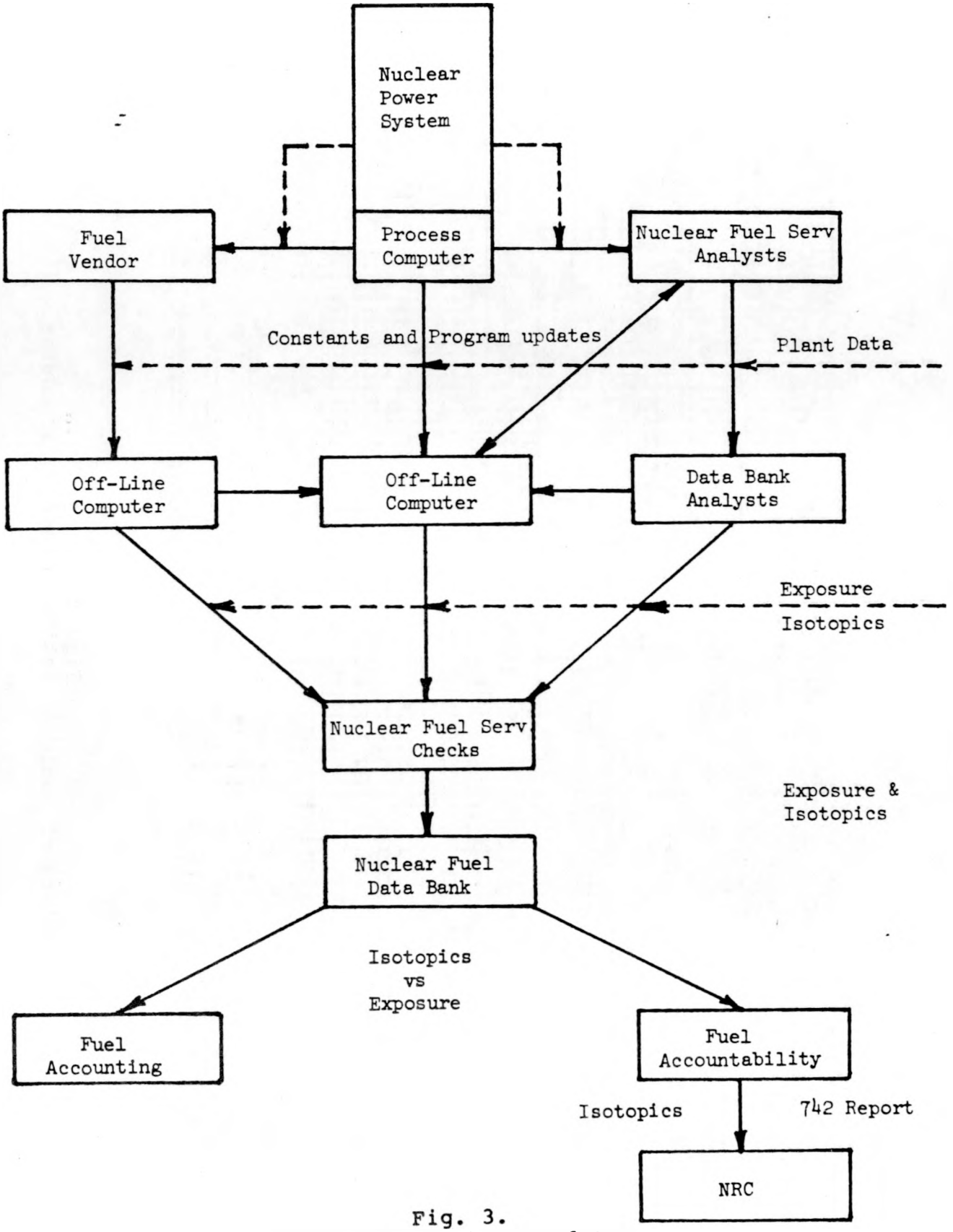


Fig. 3.
Flow of fuel burnup data.

NUCLEAR FUEL DATA BANK SYSTEM

742 REPORT DETAILS
PLUTONIUM REPORT 3/31/80
FOR PERIOD 10/01/79 - 3/31/80
RIS NO. YVE

	ELEMENT WT.	ISOTOPE WT.		ELEMENT WT.	ISOTOPE WT.
BEGINNING INVENTORY -	192,247,026.4	2,482,052.5			
RECEIPTS - FROM -			SHIPMENTS TO -		
TOTAL ON HAND-	192,247,026.4	2,482,052.5			
			TO DEPLETED URANIUM -	4,865,250.5	32,809.2
			FISSION -	507,692.7	252,824.3
			ADJUSTMENTS -	0.0	0.0
			IN REACTORS	-113,431,396.8	1,479,710.1
			IN POOLS	- 73,442,686.5	716,708.8
			IN VAULTS -	0.0	0.0
			ON-SITE -	0.0	0.0
			ENDING INVENTORY	-186,874,083.2	2,196,418.9

BREAKDOWN IN POOL WEIGHTS

	ELEMENT WT.	ISOTOPE WT.
UNIRRADIATED FUEL -	0.0	0.0
IRRADIATED FUEL -	73,442,686.5	716,708.8

ALL WEIGHTS ARE IN GRAMS.

Fig. 4.
Example of semi-annual report to NRC (form 742).

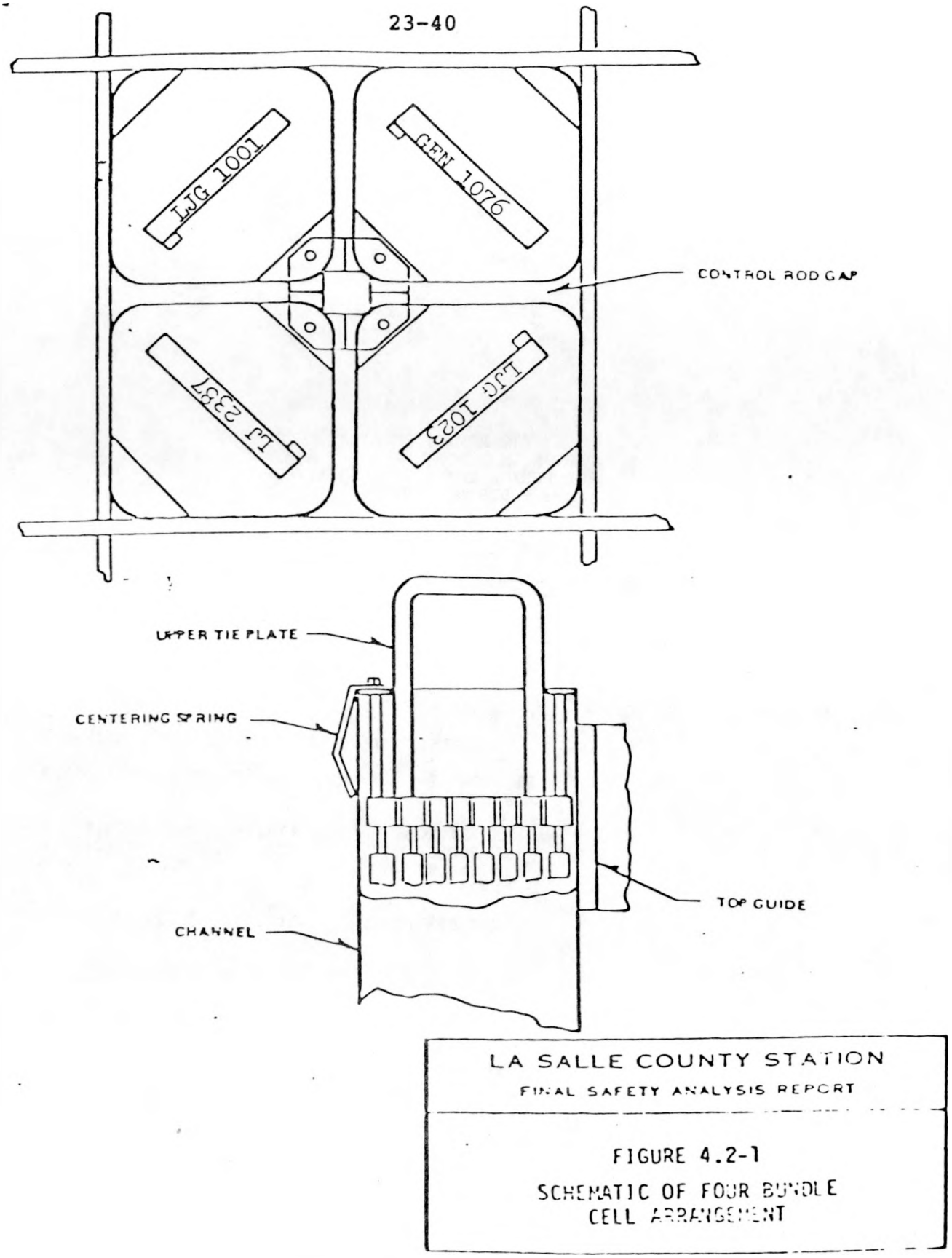


Fig. 5.
Typical fuel assembly identification.

NUCLEAR FUEL RESEARCH

PROGRAMS

FUEL MATERIAL	DRESDEN 1	DRESDEN 2	DRESDEN 3	QUAD CITIES 1	QUAD CITIES 2	ZION 1	ZION 2
Test Assemblies	SA-1 (1*)	-	-	STR (5)	-	2 (8)	-
Prototype Assemblies	11 (1*)	-	-	-	-	-	-
Mixed Oxide Rods	4 (1*)	-	-	-	-	-	-
Mixed Oxide Assemblies	11 (2*)	-	-	5 (6)	-	-	-
High Burnup Assemblies	-	-	-	-	-	-	4 (9)
Barrier Fuel Assemblies	-	-	-	4 (7)	-	-	-
Standard Assembly Rods	10 (3*)	-	-	-	-	-	-
Standard Assemblies	2 (4*)	-	-	-	-	-	-

* - Removed

Note	Fuel Vendor	Financial Support	Metallurgical Lab	Fuel Material Type	Fuel Material Number	Purpose
(1)	GE	GE	Vallecitos	segments	many	Standard design
(2)	UN/GGA	ERDA	B&W	rods	6	Recycle design
(3)	GE	EPRI	Battelle Col.	rods	10	Metallurgical
(4)	GE	DOE	Idaho	Assemblies	2	Reprocessing
(5)	GE	GE	-	Assemblies	1	Standard design
(6)	GE	EPRI	-	"	5	Recycle design
(7)	GE	DOE	-	"	4	Standard design
(8)	W	EPRI	-	"	2	" "
(9)	W	EPRI	-	"	4	High Burnup

Fig. 6.
Edison's cooperation in nuclear fuel research programs.

CENTER RUN ABC
4/06/79

NUCLEAR MATERIALS TRANSFER CHECKLIST
NUCLEAR FUEL TRANSFER REPORT (FORM C)

* IF *GRAPPLE LATCHED* LIGHT IS INOPERABLE, INDEPENDENT (I.E. VISUAL) VERIFIER INITIALS HERE (D2/3 ONLY)

STEP NO.	COMPONENT SERIAL NO	MOVE FROM RIS=YVL ICA	MOVE FROM RIS=YVL LOCATION/ ORIENT	MOVE TO RIS=YVL ICA	MOVE TO RIS=YVL LOCATION/ ORIENT		CHECKS TYPE	CHECKS LOCATION	GRAPPLE* LATCHING VERIFIED	NOTES	COMPLETED DATE/INIT
865	BN1152	PD2	B-26-2	RD2	59-36	SW					
866	BG	RD2	57-36/59-34	PD2	POOL						
867	LJU045	PD2	E-18-6	RD2	57-36	SE					
868	BN1037	PD2	D-06-4	RD2	59-34	NW					
869	LJU024	PD2	E-18-5	RD2	57-32	SE					
870	LJU122	PD2	C-21-7	RD2	59-30	NW					
871	BG	RD2	59-32/57-30	PD2	POOL						
872	LJU104	PD2	L-22-8	RD2	57-30	NE					
873	LJU050	PD2	B-23-2	RD2	59-32	SW					
874	LJU115	PD2	E-22-4	RD2	57-26	NE					
875	BNU794	PD2	B-22-3	RD2	59-28	SW					
876	BG	RD2	57-28/59-26	PD2	POOL						
877	LJU043	PD2	B-01-7	RD2	57-28	SE					
878	BN1040	PD2	D-06-9	RD2	59-26	NW					
879	LJ7370	PD2	F-23-0	RD2	57-24	SE					
880	CYU512	PD2	D-08-0	RD2	59-22	NW					

COMPONENT:
BG=BLADE GUIDE
C=CHANNEL
DA=DUMMY ASSEMBLY

CHECKS:
F=FUNCTIONAL
SM=SHUTDOWN MARGIN
SRM=SOURCE RANGE MONITOR

ICA:
P=POOL
R=REACTOR
V=VAULT
I=INSPECTION

STATION APPROVAL

DATE COMPLETED _____

BY _____

NRC _____

Fig. 7.
Nuclear materials transfer checklist nuclear fuel transfer report (Form C).

INTERNATIONAL TRAINING COURSE ON
NUCLEAR MATERIALS ACCOUNTABILITY
FOR
SAFEGUARDS PURPOSES

SESSION #24: A CANDU POWER REACTOR FACILITY

SPEAKER: Dr. David B. Sinden

Atomic Energy Control Board
Ottawa, Canada

Wednesday, June 4, 1980
9:45 a.m.

BIOGRAPHY

Education: Royal Military College, Bachelor of Engineering (Mech.)

Present Position: Manager, Safeguards and Security Division, Atomic Energy Control Board of Canada (AECB)

Past Positions: Military Service; Reactor Supervisor, AECL; Isotope Power Development - Atomic Energy of Canada Limited (AECL); Assistant and Associate Scientific Advisors (AECB); Chief of Security Section (AECB)

INTERNATIONAL TRAINING COURSE ON NUCLEAR MATERIALS ACCOUNTABILITY FOR SAFEGUARDS PURPOSES

Session Objectives

SESSION #23: AN LWR POWER REACTOR FACILITY
SESSION #24: A CANDU POWER REACTOR FACILITY
SESSION #25: A RESEARCH REACTOR FACILITY

The basic features of existing safeguards systems in specific operating facilities are considered. Emphasis is placed on detailed examples and practical experience in actual operating facilities rather than the basic features and general principles described in earlier sessions (14 & 15).

After the sessions, participants will be able to

1. Compare actual facility safeguards system characteristics and operational performance with the generalized principles and conceptual systems described in earlier sessions.

2. Discuss impact of the various safeguards requirements on facility operations.

3. Have an appreciation of basic safeguards costs and resource requirements in power reactor facilities.

SESSION 24: SAFEGUARDS FOR A CANDU REACTOR

D. B. Sinden
AECB - Canada

I. INTRODUCTION

In this presentation I would like to examine a number of subjects related to the safeguarding of a CANDU reactor. First we will examine why the on-load fuelled reactor requires special consideration in the application of international safeguards. We will touch briefly on safeguards objectives and how these objectives can be met in the light of typical diversion scenarios special to the CANDU reactor. We will also consider containment and surveillance techniques which have been developed for application in this field. We will cover briefly a history of the development of a variety of safeguard techniques for the on-load fuelled reactor and examine in some detail the system which has been developed and will be applied to the standard 600 NW CANDU reactor design. Reactors of this type are now under construction at two Canadian sites as well as at two off-shore locations.

II. NATURE OF THE CHALLENGE

The unique feature of the on-load fuelled reactor and specifically the CANDU reactor, from a safeguards point of view, is the nature of the fuel flow through the system. While fuel

flow through an off-load fuelled reactor is handled in batches at infrequent intervals while the reactor is shutdown, the CANDU reactor is fuelled and de-fuelled in a more or less continuous fashion while the reactor operates at full power. In order to effectively apply safeguards to such a system it is necessary not only that the IAEA inspector be able to verify the spent fuel inventory but he must also be able to confirm that undeclared flows of material have not found their way through the reactor. He must, therefore, be able to perform two functions, (1) a verification of safeguarded inventory, and (2) the verification of material flow. While these functions are the same as for any other nuclear facility, it is necessary to provide special techniques for verifying the continuous flow of material through the reactor.

When considering ways in which the safeguards objectives are to be achieved, it is necessary to consider the several diversion scenarios which have been suggested to be unique to the on-load fuelled reactor. The first scenario is the clandestine introduction and irradiation of undeclared fuel inventories. This fuel once irradiated, is removed from the site in the absence of an IAEA inspector and is not detectable by an inventory verification procedure. A second scenario is that spent fuel is removed from storage while not under the observation of an IAEA inspector and in its place are substituted facsimile or dummy fuel bundles which will subsequently be counted as real by the inspector. All suggested diversion scenarios are variations on these two general themes, both of

which confirm the need for dependable flow verification since neither diversion route is detectable through inventory verification procedures.

III. THE DEVELOPMENT OF CANDU SAFEGUARDS TECHNIQUES

Cooperation between Canada and the IAEA in the development of safeguards techniques for on-load fuelled reactors goes back to 1964 when the IAEA Division of Safeguards was examining the analysis of tritiated heavy water as a measure of integrated power production. Work in this field proceeded with the cooperation of Atomic Energy of Canada Limited, the Ontario Hydro Electric Power Commission and the appropriate Canadian government agencies. The development of instrumented techniques began in earnest in November 1968 when it was agreed between Canadian and U.S. agencies that a joint program to develop and test instrumented safeguards techniques would be of great assistance to the IAEA. (Figure 1) Thus the TRUST program was born. This a program to examine Tamper Resistant Unattended Safeguards Techniques and was largely a result of the efforts made by the staff of the U.S. Arms Control and Disarmament Agency and the technical wizards of Sandia Laboratory, General Electric Vallecitos Laboratory, and the U.S. National Bureau of Standards. The program had two objectives. The first was to develop and evaluate prototype instrumentation which could be applicable to the safeguarding of power reactors and other nuclear facilities. The second objective was to test tamper resistant and tamper indicating techniques and devices in the severe environment of an

operating power reactor. During the nearly five years that this program was conducted at the Ontario Hydro operated power reactor at Rolphton, Ontario a number of valuable lessons were learned and a great deal of pioneering work was done in the area of safeguards instrumentation. The program included testing of a variety of fuel flow monitors, a reactor power monitor, television and film surveillance techniques as well as a variety of tamper indicating and tamper resistant techniques for protecting both equipment and safeguards data from operator or state interference. Since the equipment was designed specifically for the Rolphton reactor, it did not have a broad application. The most important conclusion to be drawn from this program was that in order for a complex unattended instrumented safeguard system to be cost effective it was necessary to give very close attention to reliability of components and the maintainability of the system. It soon became apparent that the cost advantages of the unattended system were very quickly eliminated if frequent maintenance, especially that requiring highly qualified technical personnel, was necessary.

A second phase of the TRUST program was established at the Pickering Generating Station at Pickering, Ontario just outside Toronto. This station has four 500 MW reactors and is typical of a large scale CANDU installation. Here a number of pieces of equipment were tested. Spent fuel flow monitors or bundle counters were developed and a prototype was installed at the station to monitor the flow of spent fuel to the spent fuel storage bay. In addition, work was carried out on simple and

inexpensive neutron monitoring techniques to follow both fuel flow and reactor power. The bundle counter development was very successful and led directly to the inclusion of this concept into safeguard system designs ultimately proposed to the IAEA. The neutron detection techniques while initially very promising were incumbered with a number of problems which eventually led to the discontinuation of work in this field.

When the Canadian support program to the IAEA was officially launched, resources were made available to develop comprehensive safeguards schemes and to provide the IAEA with specific equipment required by these schemes. It was under this program that the first comprehensive safeguards scheme for an on-load fuelled reactor and indeed perhaps the most complete for any type of reactor, was put forward to the IAEA. The safeguards scheme developed for the standard AECL designed 600 MW CANDU reactor system was based on an extensive diversion analysis undertaken by Canadian contractors and a subsequent system analysis to examine the inter-relationship of all equipment in the scheme to ensure adequate effectiveness. To be effective the safeguards scheme was designed to satisfy the following criteria: (Figure 2)

1) It should compliment both the state and the IAEA safeguards system.
2) The design should cover all foreseeable diversion routes which could result in the removal of a significant quantity of nuclear material.
3) It should be sufficiently sensitive to detect a diversion of a significant quantity.

4) It should be able to detect diversion in a timely fashion.
5) The system should have sufficient redundancy to ensure that equipment failure did not significantly reduce the possibility of diversion detection.
6) The system should be tamper indicating.
7) The system should not significantly interfere with the operation of the facility.
8) The data obtained from the system should be limited to that which the inspector would require to quickly assess whether or not diversion had occurred.

IV. THE 600 MW REACTOR SAFEGUARDS SCHEME

The 600 MW CANDU is a standardized unit offered by AECL to both the domestic and export market. Figure 2a shows the major features of the design. The safeguards system proposed for the 600 MW CANDU reactor relies on both instrumented safeguards techniques and traditional material accounting procedures. It relies heavily on surveillance and containment techniques as well as discreet item accounting. Quantitative NDA techniques are not part of this scheme and considerable discussion has surrounded the merits of item accounting versus quantitative material analysis. This question will be discussed later.

The CANDU 600 MW reactor is fuelled with bundles each containing approximately 21 kg of natural uranium oxide (see Fig.3). The complete bundle assemblies are easily handled manually (See Fig. 4). New fuel is loaded into the reactor,

irradiated to approximately 7,000 MWd/Te and discharged to trays in the spent fuel storage bay. Through the use of appropriate visual surveillance systems such as television and film cameras and yes/no radiation monitors placed on potential fuel diversion paths, the inspector can be confident that fuel which has been discharged from the reactor is conveyed through its normal path to the spent fuel bay. A spent fuel bundle counter located between the reactor and the spent fuel bay is able to record the number of bundles moved and also confirm that bundles have not moved in a reverse direction through the fuel handling system. Fuel in the spent fuel storage bay is under visual surveillance by T.V. and film cameras to ensure that fuel bundles are not removed from the bay. Yes/no radiation monitors are applied to fuel paths leading out of the bay which are not easily monitored visually. Finally, the scheme provides for the verification of spent fuel inventory in the bay and the subsequent placing of verified fuel assemblies under a high integrity seal. Thus the continuity of safeguards control is maintained on spent fuel inventory which has already been verified, regardless of the operation of the visual surveillance system.

The present television surveillance system is a highly reliable recording system which includes effective motion sensing devices, redundant video recording systems, micro-processor controlled testing and switching systems and a high degree of equipment redundancy. The film cameras have been designed to be activated both by a periodic and random timer as well as by motion detection and the detection of high radiation fields. The

high level of redundancy makes it very improbable that visual surveillance will be broken.

The spent fuel inventory verification techniques provide for an attribute test of the spent fuel to confirm that the observed bundles are not dummies but are in fact spent fuel. A bundle verifier which is able to look at a tray of 24 bundles simultaneously is under development. It will respond to the gamma activity of each bundle. A dual channel gamma spectrometer is available to the IAEA to do a more conclusive verification of a randomly selected population of fuel bundles to confirm their authenticity.

The physical inventory and transfer of verified fuel bundles to sealed storage would take place during the normal IAEA visit to the facility which is anticipated about 6 times annually. At each visit the inspector would be able to analyze television records or carry back the pictures taken by the surveillance equipment since the last inspection. He would also check the count provided on the bundle counter to establish the number of bundles leaving the reactor and randomly check the yes/no radiation monitors. Finally, the inspector would confirm the number of fuel bundles under seal and do a random confirmation that the seals in question had not been compromised.

V. OTHER STATIONS

There are large CANDU generating stations now in operation at the Pickering Generating Station and at the Bruce Nuclear Power Development on the shores of Lake Huron in western Ontario.

Both of these stations currently have 4 reactors operating and four additonal units under construction. While the safeguards objectives at these stations are essentially the same as those for the 600 MW stations, there are problems arising from the fact that they are operating and certain backfitting is required. Additional surveillance equipment is required because spent fuel first transferred to the station's spent fuel bay is eventually transferred to an auxiliary storage bay. The flow path of the fuel is somewhat longer than that found in the 600 MW system. While camera surveillance systems have been in place at both these stations for a long period of time, the IAEA has recently indicated that it would want to establish a more reliable mechanism for monitoring fuel flow. Also, it is desirable to simplify the spent fuel inventory verification by placing verified spent fuel in sealed storage in a manner similar to that proposed in the 600 MW system. Both attribute monitoring equipment and spent fuel storage and sealing equipment is now being constructed for installation in these facilities. As an interim measure the Canadian government has agreed to allow the IAEA full access to the reactor facilities without regard to the man-day limitation imposed by the respective facility attachments. The ability of the inspector to observe at will any fuel transfer procedures and to verify movements of fuel within the plant provides a high degree of assurance that any attempt at diversion would be detected.

VI. SPECIAL ISSUES

The above has dealt exclusively with the application of IAEA safeguards to the CANDU reactor. There has been no attempt to discuss the additional provisions of the state system for ensuring the prudent management of nuclear materials. In Canada, for example, the state safeguards authority is the Atomic Energy Control Board and that agency maintains a full time staff at each of the major nuclear power facilities in the country. These offices are staffed by professionals with a thorough knowledge of plant operating procedures and design. These personnel also have full access to the facility. A combination of the AECB presence on site and the IAEA inspection gives the state authority a high degree of confidence that nuclear material is being used only in an authorized way.

An issue mentioned earlier which is the basis of continuing discussion is the use of item accounting techniques as opposed to traditional quantitative material balances. The long standing use of the material balance equation by the IAEA and the analysis of the associated propagated errors has led to somewhat slow and less than enthusiastic acceptance of item accounting techniques. It has been our view for some time that accounting for nuclear material contained in spent reactor fuel need not necessarily depend on accurate quantitative analysis of the fuel until such time as the fuel has been dissolved in a reprocessing facility. It is sufficient in our view to know that all nuclear material which has been irradiated in the reactor is present and accounted for in the spent fuel storage area. Whether plutonium production

estimates are accurate within a percent or within tens of percent would not in our view affect the effectiveness of safeguards provided the safeguards techniques employed could confirm that the items, in this case, fuel elements or bundles, which contain the nuclear material had not been diverted.

A final issue concerns the proportion of safeguarding resources that the IAEA brings to bear on nuclear reactors of all kinds. The in-core production of plutonium at any power reactor is of far less consequence to the interests of non-proliferation if there are no facilities available for the extraction of that plutonium from the fuel. The consequence is further reduced in states which have concluded INFCIRC/153 type agreements with the IAEA. By far the facilities of greatest safeguards concern are those in which material appears in a weapon useable form. To my knowledge the only facilities meeting this criterion are enrichment plants and reprocessing plants. It is essential in my view that more resources be brought to bear on providing high confidence levels that nuclear materials diverted from these facilities will be detected. It is this area where the highest proportion of safeguards resources must be expended. Without effective and reliable safeguards techniques in these facilities, efforts made in on-load fuelled reactors or reactors of any kind will be lost at such time as spent fuel from these facilities begins to be reprocessed.

VII. CONCLUSION

We have proposed and largely developed a safeguards approach for on-load fuelled reactors which will provide a very high degree of reliability and assurance that nuclear material diversion will be detected. Although the development of the equipment discussed has been a rather expensive proposition, it is worth noting that the bulk of the expense has been incurred in the development of a reliable optical surveillance system with complete and automatic redundancy. The inclusion of a system of this reliability in the IAEA arsenal of safeguards tools will be of great value in applying agency safeguards to facilities of all kinds. In fact unreliable surveillance systems have proved to be the weak link in many safeguards systems now relying on containment and surveillance techniques.

This brief overview of CANDU safeguards techniques has shown that although the challenge is somewhat different from many other reactor types, it can be met by a comprehensive system of containment and surveillance. The IAEA, of course, will augment the power of this system with traditional accounting techniques in new fuel inventory verification, and the audit of fuel transfer and nuclear production records.

TAMPER RESISTANT UNATTENDED SAFEGUARDS TECHNIQUES

FIG.1

SYSTEM CRITERIA

1) COMPLIMENT IAEA AND STATE SAFEGUARDS SYSTEM

2) COVER ALL FORESEEABLE DIVERSION PATHS

3) SENSITIVE TO A SIGNIFICANT QUANTITY

4) TIMELY DETECTION

5) REDUNDANCY

6) TAMPER INDICATING

7) LIMITED INTERFERENCE WITH FACILITY

8) RECORD RELEVANT DATA ONLY

FIG.2

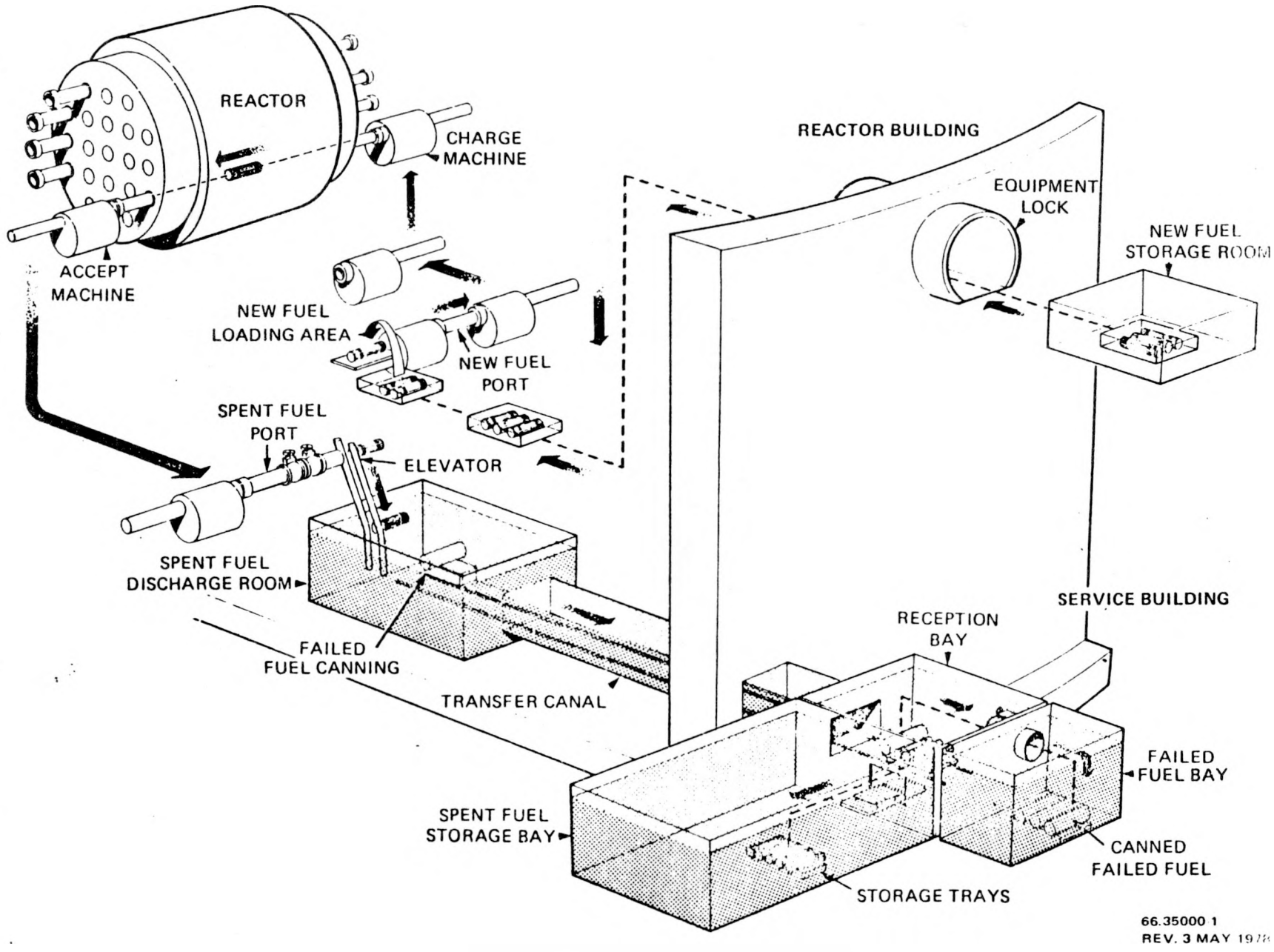
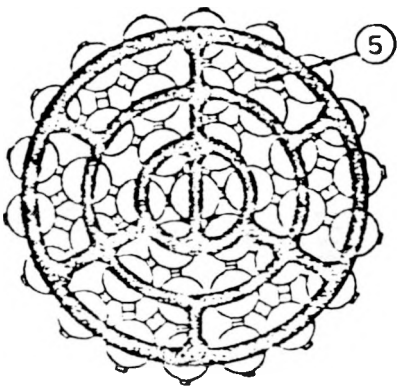
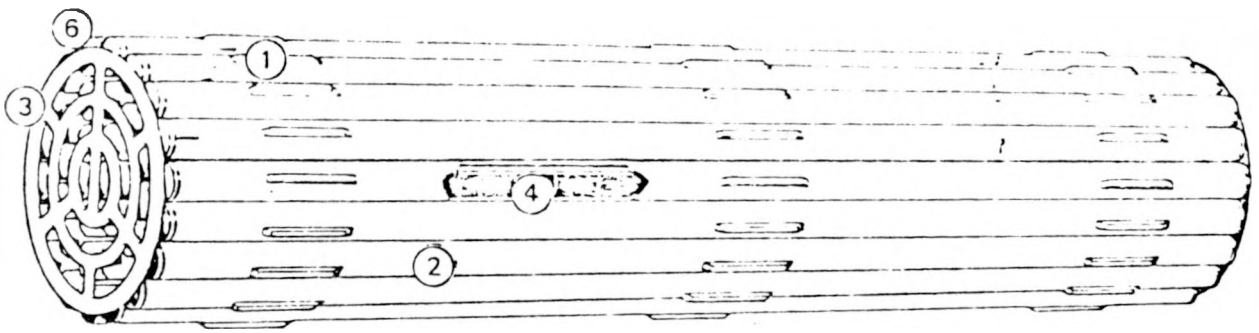


FIGURE 17 FUEL HANDLING SEQUENCE

FIG.2A

1. Zircaloy bearing pads
2. Zircaloy fuel sheath
3. Zircaloy end support plate
4. Uranium dioxide pellets
5. Inter-element spacers
6. End caps





FIG,3

FUEL BUNDLE (37 ELEMENTS)

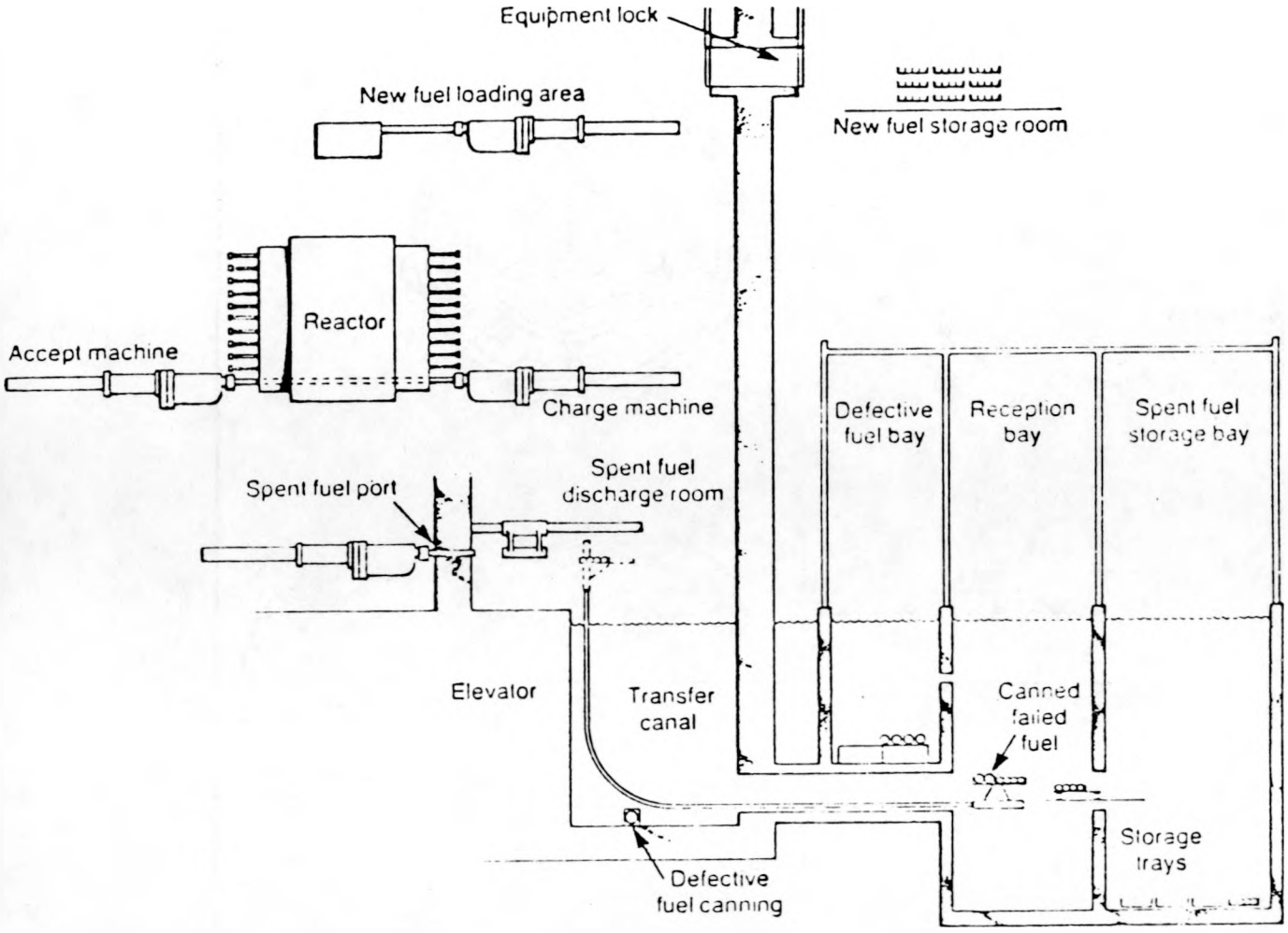


FIG.4

FUEL HANDLING SEQUENCE

INTERNATIONAL TRAINING COURSE ON NUCLEAR MATERIALS ACCOUNTABILITY FOR SAFEGUARDS PURPOSES

SESSION #25: A RESEARCH REACTOR FACILITY

SPEAKER: Dr. Fred H. Tingey

University of Idaho
Idaho Falls, Idaho USA

Wednesday, June 4, 1980
11:00 a.m.

BIOGRAPHY

Education: Ph.D., Mathematics and Mathematical Statistics, University of Washington.

Present Position: Director, Idaho Falls Center for Higher Education, University of Idaho.

Present Duties: Provide a graduate level continuing education program for Idaho National Engineering Laboratory employees and the surrounding communities.

Past Positions: Assistant General Manager, Programs, responsible for the management of eight major programs at the Idaho National Engineering Laboratory. Program areas include Water Reactor Safety, Liquid Metal Fast Breeder experimentation, Geothermal energy, Conservation, Radioactive Waste Management, and Advanced Energy Systems.

Director, Field Experimentation, Research Operations Experimentation Center for the US Army at Fort Ord, California. The work involved designing and conducting field experiments to evaluate new weapon systems, organizations, tactics, communications systems, and devices.

Other Information: Institute of Nuclear Materials Management, American Nuclear Society (local chapter), Institute of Mathematicl Statistics, Certified Nuclear Materials Manager

INTERNATIONAL TRAINING COURSE ON NUCLEAR MATERIALS ACCOUNTABILITY FOR SAFEGUARDS PURPOSES

Session Objectives

SESSION #23: AN LWR POWER REACTOR FACILITY
SESSION #24: A CANDU POWER REACTOR FACILITY
SESSION #25: A RESEARCH REACTOR FACILITY

The basic features of existing safeguards systems in specific operating facilities are considered. Emphasis is placed on detailed examples and practical experience in actual operating facilities rather than the basic features and general principles described in earlier sessions (14 & 15).

After the sessions, participants will be able to

1. Compare actual facility safeguards system characteristics and operational performance with the generalized principles and conceptual systems described in earlier sessions.

2. Discuss impact of the various safeguards requirements on facility operations.

3. Have an appreciation of basic safeguards costs and resource requirements in power reactor facilities.

SESSION 25: SAFEGUARDS IN A RESEARCH REACTOR FACILITY

Dr. F. H. Tingey
University of Idaho

I. INTRODUCTION

Research reactors are designed to accommodate large numbers and varieties of experiments subjected to high neutron fluxes over extended periods of time, or they may consist of special purpose reactors such as a power burst facility or a small scale power reactor designed to conduct highly specialized tests on fuel rods, assemblies or cores subjected to programmed power-coolant mismatches or power excursions leading to fuel melt down. In addition, the usual test reactor complex may have associated with it critical facilities for mocking up reactor cores, gamma irradiation facilities which in the simplest form are water filled canals containing an array of spent reactor fuel elements where experiments requiring high gamma fluxes can be performed, hot cells, laboratories and waste disposal facilities.

Special nuclear material on inventory at any given time at a test reactor site consists mainly of that contained in fuel elements and experiments. Fuel elements dominate the inventory and thus considerable effort is justified in devising and implementing measurement and control procedures for these items. In contrast, not all experiments contain special nuclear material

and most that do have very small amounts. Furthermore, these items as received are frequently encapsulated, canned, or fabricated into elaborately instrumented pilot scale devices essentially ready for insertion. Rarely is it possible to verify the content of such items. Accordingly, their receipt is acknowledged upon recognition of an item identification code or a signature analysis. Following irradiation these items are usually returned to the sender at the same value as received; no report is made from the reactor site regarding estimated results of nuclear processes, e.g., fission loss.

Experimental item accountability and control is the primary responsibility of those technically responsible for the experiments. Such groups serve as the intermediary between experiment sponsors and reactor operations. These individuals estimate the effects of nuclear processes such as heat developed, and duration of irradiation to accomplish the results for which the experiment is being conducted. This information serves as a guide to those in authority who approve the experiment insertion. The technological values of an experiment by the time it is fully prepared for insertion are usually many times the nominal dollar value of the material content. Hence, observations of the precautions necessary to safeguard the experimental data are usually adequate for protection of the material content.

Reactor fuel accountability requires continual monitoring of fuel assembly movements and the establishment of procedures and instrumentation that are responsive to abnormal conditions. This is effected through the combined efforts of Nuclear Materials

Control, Operations, and Technical personnel. An effective organizational structure is given in Figure 1 and will be assumed in subsequent discussions. It is to be noted that by reporting directly to Management, the Nuclear Material Control group is able to expeditiously cross organizational lines to implement it's procedures. It should also be divorced from any responsibility for the physical assignment and handling of nuclear materials at the reactor site. This responsibility is assigned by material balance area to Operations and Technical personnel who assume physical custody where applicable. Fuel transfers between areas are accompanied by inter-area transfer forms generated by the various custodians, a copy of which is forwarded to the Material Control Office. Periodic summation of these transfers provides the necessary data for material balance summary inventories. Data generated are scrutinized for qualitative and quantitative reliability and formulated into periodic reports as required.

The nuclear materials control system for research and test reactors can best be described in terms of the operational control required (material flow), material control necessary (flow of written records), and measurement technology needed to effectively implement the procedures. The degree to which these can be coordinated and integrated will in turn define the degree of effectiveness of the nuclear materials control system. A discussion of each of these aspects of control with emphasis on their integration to provide an efficient overall system follows.

II. SYSTEM DESCRIPTION

A. Introduction

The term "fuel elements" will be used here to include the typical flat plates used in test reactor fuel, arrays of plates, fuel pellets, fuel rods, and arrays of rods. The term "fuel assemblies" will be used in reference to both plate and rod type assemblies.

Fuel elements must be properly identified, inspected, tested, measured, and stored. Identification (by fuel number and prefix, when feasible) is compared with original shipping documents and the fabricator's specification sheet. Inspection procedures are stringent to reveal dimensional inconsistencies and non-compliance with specifications. Certain physical defects in assemblies are tested by subjecting them to water flow pressure tests. These tests reduce the possibility of charging fuel to the reactor which may collapse under pressure. When the assemblies have passed the inspection and testing stages, they are non-destructively assayed to verify U-235 content. A more detailed description of the assay measurement device is given later in this paper. Proposed core loadings of acceptable test reactor fuel assemblies are then transferred to the Reactor Critical Test Facility for core mock-up tests. Here the elements are arranged in a critical array and certain measurements performed. Finally, the elements must be stored in a critically safe, security vault. The vault must be designed to provide adequate storage under Health and Safety and Security Regulations.

B. Process Control (Material Flows)

1. Introduction. Process control of fuel element movements is designed to maintain constant surveillance of fuel inventories by providing proper accountability, materials control, and physical security. The control mechanism functions more satisfactorily if the research reactor installation is subdivided into defined geographic boundaries. The material flow (fuel movements) in a typical research reactor is outlined in Figure 2. A discussion of the principal process control points with reference to Figure 2 is given in subsequent paragraphs of this section.

2. Vault Storage. Fuel vault storage facilities are under the responsibility of Reactor Operations. Special nuclear materials in fuel elements become the responsibility of Reactor Operations, while experiment items are under the jurisdiction of Project Engineering. Traffic is responsible for informing Operations of impending deliveries of fuel elements. Advance notice must be received so that unloading and transfer arrangements can be made. Theoretical Physics is responsible for the nuclear safety of the fuel storage configurations provided in the vault. They confirm in writing the safety of proposed configurations and approve any contemplated changes. Upon notification that a shipment of fuel is to be delivered, Operations requests Traffic to unload the shipment directly from the truck to the vault. Temporary storage of fuel elements on the reactor floor is not allowed since it invites security and safeguard hazards. A vault custodian is assigned for each vault

to cover the normal work week. Additional custodians are authorized to enter the vault on weekends and shift hours. These persons maintain the same control over vault fuel traffic as that required of the daily custodian. A written procedure manual on vault storage control is an integral part of the system.

3. Material Issued From Vault. The vault custodian is responsible for issuing items for inspection, testing, assaying, etc. This responsibility includes complying with criticality safeguards and is accomplished through geometrical considerations by controlling the total number of fuel element assemblies absent from the vault at one particular time.

Material is received into the vault on the basis of values measured by the fuel vendor as witnessed by a resident inspector, or on the basis of NDA measurements performed as a receiver measurement. Photographs are also taken of end box weld patterns on fuel elements for material control purposes.

Removals from the vault are on the basis of item identification and are documented by Internal Transfer Forms (Exhibit 1). Signatures of both the sender and the recipient are recorded, thereby transferring custodial responsibility.

4. Criticality Control. Storage configurations within the vault are in strict accordance with written procedures supplied by Theoretical Physics. The vault custodian is responsible for storing fuel within the recommended configurations and consulting Theoretical Physics before changing the approved storage matrix.

5. Critical Facility Area. The critical facility test core is designed to operate at zero power to provide neutron flux

and reactivity data for proposed reactor cores and experimental loadings.

Inventory control for fuel assemblies can be perpetually recorded either in log books or on inventory control panels. The latter arrangement provides a picture of the fuel assemblies actually in the critical facility and those in the attached canal storage area. One system consists of a board showing the test facility core with the attached canal storage locations. A tag system is used for core arrangements and storage identification.

6. Reactor Charges. Fuel transferred to the reactor is accompanied by the Internal Transfer Form. Predetermined core arrangements are listed on a charger loading schedule. Upon signed receipt, the fuel assemblies become the custodial responsibility of Operations. Previously irradiated fuel assemblies returning to the reactor core and reactor discharges to the cooling basin are documented by a note indicating a change in location. No Internal Transfer Form is needed since the reactor and the cooling basin are one material balance area.

7. Cooling Basin Traffic. The cooling basin is considered to be part of the reactor for material accountability purposes. Fuel assemblies or bundles discharged from the reactor are transferred to fuel storage grids located in the cooling basins. The storage grids are designed to accommodate an entire reactor core. They are also identified by a numerical or alphabetical system which can be read under a minumum of twenty feet of water. The Reactor Charging Grids are usually located at the foot of the reactor discharge chute. These grids are loaded with fuel

assemblies for future reactor loadings. Inventory control is accomplished by the combined use of the Internal Transfer Form and the Perpetual Inventory Control Panel. The control panel may be constructed of plexiglass with outline drawings of canal grids and grid positions. Marking pencils can be used to enter fuel element numbers. Fuel traffic in and out of the various grids can be recorded on the control board. This system is recommended over a tag control board because of the large volume of fuel traffic frequently associated with this area. Off-site shipments are recorded on the Internal Transfer Form.

8. Gamma Facility. Transfers to Gamma Facilities are documented by the Internal Transfer Form. The storage problems are identical to those associated with the cooling basin. The fuel traffic through this area may be less than that expected in the cooling basin. In this case, a tag board system will suffice. Shipments to off-site areas are also recorded on the External Transfer Form.

9. Hot Cells. Hot cells contain irradiated material of various enrichments and forms. Receipts and shipments are documented by the Internal and External Transfer Forms, respectively. Shipments are normally to burial or to recovery facilities at another area.

10. Laboratories. Analytical and Research Laboratories contain only a small amount of special nuclear material at any time. Receipts and shipments are infrequent and variable as required to support research or to determine content of a material (principally pellets) received or being shipped.

Transfers in and out are documented by the Internal Transfer Form.

11. Waste Disposal. Low level transuranic waste not economically or strategically recoverable is stored on site in specially constructed drums. These wastes are for disposal, i.e., the facility is an interim repository, and consequently no accountability is maintained except in the sense of a perpetual inventory record.

12. Experiment Control. Process control of experimental items must be rigid for several reasons. The majority of items are small in nature and are usually capsulated and identified by outside markings. The majority of these experiments are difficult to identify once they have been removed from the reactor. Identification markings are smeared or become corroded in the process of irradiation. The control process by which this difficulty can be eliminated will be described in a subsequent paragraph. The traffic of experimental items used in research reactor installations is described in Figure 3.

13. Experiment Receipts. All experimental items are received from the sponsor by a receiving agent. This agent is under direct supervision of Project Engineering. Each item which is small or capsulated is photographed by the receiving agent. Experiment identification is placed on a card directly below the sample and is photographed with the experiment. If the sample is misplaced after irradiation, the photographs can be consulted for proper identification. Experiments to be inserted in out-of-pile loops are not photographed. These experiments are usually large

and can be readily identified. They are transferred to the preparation room for assembly or modification.

After the experiments have been inspected and photographed, they are transferred to a vault or secured storage area. The vault is independent of the fuel storage area. Custodial responsibility is with the receiving agent for the vault.

Experiments discharged from the reactor can be stored in the cooling basin in wall racks or inventory buckets. In either case, the storage space must be identified by markers large enough to be seen from the basin rail. The experiment number and location bins can be marked on the inventory control panel discussed under the section, "Cooling Basin."

C. Material Control (Material Balance Areas)

1. Fuel Element Movements. The basic unit of a materials control system is the material balance area, a subdivision of the process in which all inputs, outputs and inventories can be measured accurately. An acceptable breakdown for material control purposes is illustrated in Figure 4. The paper flow accompanying experiment and fuel movement is covered in a subsequent section of this paper. Internal Transfer Forms required for each fuel movement are also shown in Figure 4. Further subdivision by material balance area could be made if better definition is required. For instance, the Cold Storage Balance Area may be subdivided into five material balance areas. This action would tend to control movements of fuel before insertion into the reactor. Balance Area No. 2 could be

subdivided into two major areas, namely Reactor and Cooling Basin. However, reactor and canal inventories are available from Internal Transfer Forms and the charger loading records. To include the canal and reactor as separate inventory accounts would necessitate the recording of additional internal transfer forms covering movements from the reactor to the canal and vice versa. Separate inventory summations can be made available for both areas regardless of account segregation.

Inventory summation for each material balance area is derived from consolidation of Within Area Transfer Form series; Cold Storage to Reactor A, Reactor A to Gamma Facility, etc. This is performed in the Material Balance Summary Ledgers discussed in a subsequent section.

2. Experiment Accounting. Experimental item accounting is designed to perform much the same as a city library. Geographic locations are ignored to simplify the recording mechanisms. Experiment locations are readily identified by consulting the operation control data. Nuclear Materials control and Accountability (MC & A) assigns only custodian responsibility.

It is practically impossible to segregate the experimental items by geographic location. The use of the internal form in this case would be so burdensome that efficient accountability would not be served. Each custodian (usually a Project Engineer) maintains geographic control of experiments under his assignment. As indicated previously, inventory locations are also available. MC & A can, therefore, maintain proper materials control by simply assigning experiments to the Project Engineers

(by signature). The engineer is not relieved of physical responsibility until the material has been shipped or transferred to another custodian. In each case, he must issue a new Internal Transfer Form, forwarding a copy to MC & A. The original data are entered on the form by a member of MC & A. Again, as in the case of fuel receipts, the information is transcribed from the transfer form. The flow of the Health Physics monitoring forms (on shipments) and the courier receipts (both in and out) is coordinated and dispersed by the experiment receiving agent assigned to Engineering. By maintaining close contact with this individual, MC & A is advised through either verbal or written means of all experiments entering or leaving the reactor project installation.

In summary, it must be noted that MC & A does not become responsible for physical assignment but serves only as an original informative source of experimental quantities and identification.

III. FLOW OF WRITTEN RECORDS

A. Introduction

The following is a discussion of the Nuclear Material Control records systems utilized by MC & A and Operations. The systems are not mechanisms for process control or operational safeguards, but accounting practices for maintaining nuclear material control.

The source data is recorded into journals, summarized, and later transferred to final ledger balances. It consists of

reactor fuel, fuel components and experiment transfer data pertinent to the research reactor operation. The records are maintained on an item control basis. Ledgers are designed to reflect process operations. Resultant inventory balance summations can be transferred to inventory composition schedules without compiling consolidated working papers.

In order to implement the system it is desirable to have an established procedures manual which describes the Nuclear Materials Control System with special emphasis on the need for cooperation between members of the Nuclear Material Control Office and the Operational and Technical Groups.

It is to be noted that source data are compiled by members of the Operational and Technical Groups and not Nuclear Materials Control Accountants.

B. Source Data

The proper control of special nuclear materials at a research and test reactor installation is dependent on accurate source information. The source data consists of written documents (transfers) covering all information pertinent to nuclear material identity and quantity in case of a change of custodianship. The written records or forms necessary for both internal and external transfers are discussed below.

1. External Source Data. External transfers should be documented by a standardized form (see Exhibit 2). This form serves as a written notification to charge and relieve the facility account with uranium quantities as identified. Both shipper and receiver should independently measure the transferred

nuclear material if possible. When the receiver cannot measure the nuclear material content of the shipment within 10 days, but intends to do so at a later date, receipt of the material should be acknowledged. This can be done through issuance of a form similar to Exhibit 3. The distribution of the External Transfer Forms is shown in Figure 5.

2. Internal Source Data. Internal source data originates from Internal Transfer Forms (exhibit 1) which serve to control material movements within the reactor project. A research type reactor accounting system must be designed to accommodate control of both fuel elements and sponsor-owned experiments. The transfer form, therefore, should be flexible enough to record information on fuel transfers and at the same time assign proper custodial responsibility for both fuel elements and sponsor-owned experiments. The flow of the internal form for fuel assembly traffic is shown in Figure 4.

The Nuclear Materials Control Group initiates the Internal Transfer Form upon receipt of the fuel elements. Data are entered from information on the External Transfer Form (Exhibit 2). The custodian in charge of the Cold Fuel Storage Balance Area signs the transfer form upon receipt of the assemblies and forwards a copy to the Nuclear Materials Control office. Subsequent transfers between material balance areas are recorded on Internal Transfer Forms one copy of which is forwarded to the Nuclear Material Control office. When the fuel is ready to be shipped to an off-site installation, the custodian issues an Internal Transfer Form to MC & A. From the information

contained on this form the Nuclear Materials Control Office prepares an External Transfer Form (Exhibit 2) to cover the fuel shipment.

The use of Internal Transfer Forms to assign custodianship of reactor experiments is described by Figure 6. The control of experimental items by custodian and material type is discussed under the section on "Journals and Ledgers."

3. Supplementary Data and Forms.

a. Health Physics. The health physics monitoring form, Exhibit 4, is used primarily for health and safety purposes, however, it is a very useful source of information to MC & A. All shipments leaving the reactor site are accompanied by a monitoring form. MC & A, by being on the distribution list for the forms, can use the information contained thereon to insure that transfer forms were prepared properly on all shipments.

b. Courier Receipts. Courier receipts are issued on all receipts and shipments at the reactor site. A copy of the receipt is forwarded to MC & A.

c. Expended In Research. Exhibit 5 is designed to record the expenditure of nuclear material consumed or to be consumed in research. It may also be used to authorize expenditure of material by burial. MC & A approval must be obtained before disposing of any nuclear material.

d. Journal Entry. Exhibit 2 is used to document changes in the nuclear materials inventory not covered by other external or internal source documents. Reasons for journal entries

include such items as material expended in research, losses, writeoffs, and project number changes.

C. Journals and Ledgers

The journals and ledgers to be used in the material control system can be designed to be simple yet adequate for compiling and maintaining the information necessary for good accounting practice. Proper design will eliminate the necessity for keeping the additional records usually classified as "Working Papers." A discussion of a practical system of journal and ledger control will follow.

1. Journals. Two journals are used to control uranium accountability for both fuel element and reactor experiments. These journals are classed into two categories; namely, External Transfer Journal and Internal Transfer Journal. They will be discussed in that order.

a. External Transfer Journal. The External Transfer Journal is designed to permit the recording of external receipts, removals and adjustments of nuclear materials and inventories for all special nuclear material at the facility. A sample journal which has proved valuable is shown below.

Facility_____ External Transfer Journal Material Type_____

Material Received				Material Shipped			
Date	Series	Material Description	SN Net U-235	Date	Series	Material Description	SN Net U-235

Monthly and year-to-date totals are recorded for each Material Control Facility. The recording of shipments and receipts on a single page eliminates the need for an additional journal. Furthermore, the journal facilitates detection of any unreturned items for a particular facility without requiring reference elsewhere for associated information. The External Transfer Form is the source document for all entries to the External Transfer Journal.

b. Internal Transfer Journal. The Internal Transfer Journal is a listing of all internal and external receipts, removals and adjustments of nuclear material at the facility. Rather than record individual transfers to this journal, Internal Transfer Forms are summarized and bound at the end of each month to form the journal. The totals are transferred to the Material Balance Summary Ledger. The Internal Transfer Form (Exhibit 1) is used as the source document for all entries to the Internal Transfer Journal. A journal which records each individual fuel

assembly transaction is described later under "Subsidiary Control Ledger."

2. Ledgers.

a. Material Balance Summary Ledger. The Material Balance Summary Ledger is designed to accommodate journal postings (monthly) without referring to more than one ledger page. This system eliminates posting errors which are so commonplace where postings are usually entered in two different ledgers. Balance summarization is more efficient where debits and credits can be viewed on the same page. The system also fulfills the the double entry bookkeeping requirement of most regulatory agencies. A simplified example of a ledger which would be sufficient for journal postings in a research reactor project is shown in Exhibit 6. Monthly entries are taken from both the External Transfer Journal and the Internal Transfer Journal. Credits (shipments) are entered in red and debits (receipts) are entered in black. The Internal Facility postings are entered in the appropriate Balance Area columns. The External Transfer postings (monthly summaries) are entered in the Facility Control column and in the appropriate balance area accounts. Entries are made by indicating the transfer series identification in the Description column. Monthly ending inventory summations for the Balance Area Accounts can be derived without reference to other ledgers. The total sum of Balance Area inventories should agree with the Facility Control Ledger. Other sources can also provide data which will affect the Facility Control Account and certain balance area accounts. Such transactions include burnup or

fission loss write-off and losses of material expended in research.

Experimental items received and shipped enter the Facility Control and Experiment columns. The Experiment column is included as a separate ledger account to distinguish experimental quantities from the fuel element accounts. This information is very desirable if composition inventory requirements are to be met. In addition, the segregation also permits proper subsidiary control accounting; that is, subsidiary records can be classed into two major categories; namely, fuel and experiments. The fuel subsidiary accounts are further subdivided into geographical classifications which coincide with the Balance Areas shown in the Material Balance Summary Ledger.

b. Subsidiary Control Ledgers. The use of subsidiary control accounting depends largely upon the complexity of the research reactor operations. Small research reactors operating at low power levels and utilizing the same fuel elements for a considerable length of time can be controlled without the use of subsidiary accounting. This discussion concerns research reactor operations where large fuel quantities are used to maintain high reactor flux levels.

(1) Fuel Element Subsidiary Control Ledgers. The fuel element subsidiary controls are facilitated through a card system amenable to automatic data processing (ADP). Fuel identification and quantity are entered on the cards. Data are entered from forms received with fuel shipments. One card is used for each fuel element. The cards are filed by Material Balance Area as

shown in the Balance Summary Ledger. These records are maintained by daily transactions involving fuel movements. Ledger changes corresponding to fuel element transfers are made manually by moving ledger cards from one inventory account to another. The source data required for these changes are the Internal Transfer Forms. If several fuels are moved between balance areas simultaneously, and ADP is available, summaries may be printed from the fuel cards and attached to the Internal Transfer Form. This form is used by the accountant at the end of the reporting period to compile a formal Internal Transfer Journal. The monthly posting of this journal to the Material Balance Summary Ledger was discussed earlier.

Burnup and fission loss values are also recorded on ADP data cards. These values are obtained from cycle burnup computation. The ADP data card representing the fission loss and burnup of a fuel element is then matched with the corresponding Subsidiary Control Ledger card and the indicated adjustment is made. Total cycle burnup quantities are posted to the Material Balance Summary Ledger showing a decrease in the Facility Control column and the affected reactor account. The source for this entry can be obtained from either an ADP printed list of burnup cards or the original burnup computation sheet.

At the end of the reporting period, the ADP fuel card adjustments are summed. Residual fuel totals by material balance area are checked against the Material Balance Summary Inventories. The advantages of using ADP procedures in this activity are apparent.

(2) Experiment Subsidiary Control Ledger. This Subsidiary Control Ledger is composed of data cards including all data pertinent to the experiment. The information is transcribed from the External Transfer Forms directly to data sheets. If ADP is used, cards are later punched from the raw data sheets. The information contained on each card includes experiment number, external transfer series number, date received, content and custodian. The ledger is maintained by use of the Internal Transfer Forms. Additions and removals of ledger cards are made upon signed receipt by the custodian. Listings of reactor experiments can be made in a number of ways, for example, by experiment number, sponsor segregation, material type and custodian. Printed lists can be issued periodically to the various project engineers. In this way inventory discrepancies can be minimized.

D. Consolidation and Trial Balances

As indicated, all source data are consolidated into two journals, namely, the External Transfer Journal and the Internal Transfer Journal. The journals are likewise consolidated in the Material Balance Summary Ledger. A summary of the beginning inventory plus receipts minus shipments and burnup will reveal the ending inventory balances. The sum of the Balance Area Inventory Accounts will agree with the Facility Control Account, and Subsidiary Ledger inventory listings must agree with the Material Balance Area Accounts before reports can be issued.

E. Reports

Material Status and Material Balance Reports are issued periodically. Material Status Reports contain the beginning inventory total, receipts (month and year-to-date), shipments, burnup, losses, ending inventory and the Book Physical Inventory Difference. They include Composition of Ending Inventory Reports which are a summary of the nuclear material inventory arranged by material type and identification number.

Material Balance Reports are prepared from the Material Balance Summary Ledger. These may consist of several separate reports. Following periodic physical inventories, a monthly inventory report is submitted to the regulatory agency. Work papers and other supporting material are included to document reconciliations, inventory differences and the new ending inventory by material balance area. It is sound accounting practice to design the ledger control accounts to coincide with expected inventory composition requirements. This eliminates the need for consolidating work papers.

IV. MEASUREMENT TECHNOLOGY

A. Introduction

The success of any research reactor material control system depends upon the generation of sufficient data to facilitate the degree of control required. Because the nuclear material inventory must be continuously corrected for loss by burnup, certain reactor and reactor fuel measurements are necessary. These are the measurements of initial nuclear material content of

the fuel, the total heat (integrated power) generated by the fuel in the reactor, and the distribution of the heat generation among the fuel elements.

B. Nondestructive Assay of Highly Enriched Fuel Elements

The cold content of the fuel elements can be measured most accurately during their fabrication. Individual plates or rods are gamma-scanned using a dual-channel gamma analyzer with a sodium iodide crystal and associated electronics. Measurements are made by a resident inspector by comparing measured gamma activity to that of known standards. When a fuel assembly is received, it is checked with an Isotopic Source Assay System (ISAS) which uses Californium-252 to induce fission of U-235 atoms in the assembly. Four scintillation detectors surround the assembly and measure gamma rays and neutrons. A coincidence circuit is used to separate the induced fission events from other background radiations. Responses from fuel assemblies are compared to responses from known standards to determine U-235 content.

No nondestructive measurements are made on irradiated fuel elements. The lack of a suitable technique for making these measurements necessitates the calculation of uranium burnup values for material accounting from reactor power and other fuel element exposure information.

C. Gross Power Calculation

1. Temperature-Differential x Flow Calculation. For research and test reactors which remove heat by circulating gas

or liquid through the reactor, the gross power can be calculated by using an expression of the form

$$\text{Power} = K(\Delta T)F$$

where

ΔT is the temperature differential of the coolant across the reactor,

F is the coolant flow rate and

K is a factor dependent on certain physical properties of the coolant at the existing conditions and energy conversion factors.

For purposes of burnup computation, reactor power is computed continuously from data generated by resistance thermometers located in the upstream and downstream sections of the main flow lines, and by a Gentile pressure differential flowmeter across the primary flow tube.

There are three primary sources of uncertainty or error associated with the power calculation as determined in this manner. These are:

(1) The measurement error in the ΔT factor.

(2) The measurement error in the flow determination. This error results from:

  (a) the measurement uncertainty in the differential pressure determination across the flow tube, and

  (b) the interpretation of differential pressure as flow.

(3) The error identified with the mechanism by which the differential temperature and flow are converted to power.

Usually the interpretation of differential pressure as flow is accomplished by means of a calibration. The error associated with this calibration is derived from data obtained under highly controlled conditions prior to installation of the flow tube. This should be a full scale calibration rather than one which would result from the application of dimensional analysis to scale up from the smaller flow tubes. The conversion of the differential pressure to a flow is through instrumentation which is approximately, but not precisely, equivalent to the original calibration. Thus the component (2)(b) above is in fact a composite of:

(i) The uncertainty to be associated with the original calibration.

(ii) The bias or deviation between the flow indicated on the instrument and the calibration value for any given pressure differential.

Experience has shown that the uncertainty in the original flow tube calibration will dominate the uncertainty in the gross power calculation unless particular care is taken prior to installation to obtain an accurate primary flow tube calibration. Furthermore, the error in this particular component is systematic and could be extremely serious.

The major problem in flowmeter calibration for high power reactors using water as a coolant is associated with the high flow rates involved. This condition requires a pump sump with a capacity sufficient for several minutes flow. Test equipment must be installed on the sump tank to enable observers to read

the drop in water level and the corresponding time intervals accurately. The tank must be carefully calibrated to insure an accurate value for volume delivered between level readings. During each run the flow tube pressure differential is measured with a manometer. The data (flow rate versus pressure differential) are fitted by regression analysis techniques to the functional form,

$$F = a(\Delta P)^P$$

or

$$\log F = A + b \log \Delta P$$

The error associated with the fitting procedure can be obtained from the regression analysis. Regression theory would suggest that the individual calibration runs be concentrated at the end points of the range of anticipated application. Since the instrument response may not be linear, intermediate values should also be obtained to verify the form of the regression equation.

In the reactor, the temperature differential is measured by a multiplicity of resistance thermometers located in the main flow lines. These thermometers are located sufficiently downstream from the reactor to allow thorough mixing of the water. Temperature differential measurements contribute negligibly to the overall error in gross power measurement, particularly since accuracy control is easily accomplished at periodic intervals by submerging the upstream and downstream thermometer bulbs in controlled temperature baths, applying a

predetermined temperature differential between the two baths, and recording the ΔT chart reading.

2. Pool-type Reactor Calibration Power determination in pool-type reactors presents particular problems because the conventional flow times ΔT procedure may not be applicable. The following are some alternatives:

(1) Fission Rate Method. One calibration method applicable to all reactors of this type involves calculating the fission rate from the known fuel distribution in the core and an experimentally determined neutron flux distribution. This method is particularly appropriate for reactors operating at power levels below one MW. At higher powers the flux measurements are principally used for determining the detailed distribution of power within the core and not the absolute value.

(2) Pool Calorimeter Method. In this method, the reactor is operated at constant power for a period long enough to obtain an accurately measurable increase in pool temperature. The power is then calculated from the temperature rise and volume of pool water. This method directly measures all of the available fission energy since it includes the gamma-ray energy absorbed by the pool water. The power level at which this method is feasible depends on the pool volume. It is desirable to have some means of providing complete mixing of the pool water so that the pool temperature can be measured accurately. Care must be taken, if this method is used,

to prevent harmful thermal stresses in the pool walls due to a rapid temperature rise. In some cases pool heat losses will be important and must be measured.

(3) Heat Balance Method. A heat balance method is applicable to reactors with forced cooling and a secondary coolant system. The power output of the reactor can be calculated directly from the measured flow rate and the temperature in either the primary or secondary circuit, if thermal equilibrium exists in the system. If the system is not in equilibrium, corrections can be made from observed changes in pool temperature. Nonequilibrium conditions are common. Pool volumes and operating power levels are such that small changes in pool temperature represent a significant fraction of output power, so it is necessary to observe temperatures over a period of several hours to obtain an accurate power determination. Consequently, this method does not accurately measure the instantaneous power level. As in the pool calorimeter method, gamma energy dissipated in the pool is included in the measurement. The application of this procedure to the calibration of the Battelle Research Reactor is described in Figure 10. Flow indicators were installed in both the primary and secondary coolant circuits and temperature indicators were installed at the inlet and outlet of the primary and secondary legs of the U-tube heat exchanger. In addition, two

indicators were located in the pool water at different vertical locations (one near the core and the other near the pool surface) to monitor bulk pool temperature. Reactor operators recorded the temperatures every half hour.

D. Calculation of U-235 Depletion

1. Total Burnup Computation. Total burnup of fissionable material includes losses by fission and by conversion to other isotopes through other nuclear reactions. If U-235 is the fuel material, the burnup includes conversion of U-235 to U-236 by the (n,γ) reaction. The ratio of grams U-236 to grams U-235 fissioned is essentially equal to the capture-to-fission cross section ratio,

$$\alpha = \frac{(n, \gamma)}{(n, f)} \quad .$$

The gross loss by fission can be computed from the relationship

$$\frac{\text{grams fissioned}}{\text{MWD}} = \left(\frac{\text{grams fissioned}}{\text{fission}}\right)\left(\frac{1}{\text{Mev/fission}}\right)\left(\frac{\text{Mev}}{\text{MWD}}\right)$$

Substituting appropriate values for these factors a simple relationship relating fission loss to burnup expressed in megawatt-days (MWD) can be obtained.

$$\frac{\text{grams fissioned}}{\text{fission}} = 3.90 \times 10^{-22}$$

$$\text{MeV/fission} \approx 200$$

$$\text{Mev/MWD} = 5.39 \times 10^{23}$$

Therefore, the fission loss in grams U-235 equals 1.05 times the burnup in megawatt days. For a thermal reactor:

$$\text{Burnup} = (1 + \alpha)(\text{fission loss})$$

$$= (1 + \alpha)(1.05)(\text{MWD})$$

The total loss of uranium is calculated from the relationship:

$$\text{grams U lost} = \text{grams U-235 lost} - \text{grams U-236 generated}$$

$$= 1.05(1 + \alpha)\ \text{MWD} - \frac{236}{235}\,\alpha\ (1.05)\text{MWD}$$

$$= 1.05(\text{MWD})(1-.004\alpha)$$

The value for α depends upon the reactor and the associated fuel assembly loadings. Values in use range from 0.175 to 0.235. When a core loading contains different types of elements, a weighted average value of α is used to make the burnup calculation.

In the subsequent chemical processing of the spent fuel fission stoichiometry can also be utilized to provide burnup estimates from chemical data. Using the notation

U = quantity of total uranium

E = enrichment on U-235

Z = U-236 content

α = capture-to-fission cross section ratio

and using the subscripts "0" and "1" to denote pre- and post-irradiation respectively, we have by definition

$$\text{U-235 burned up} = U_0E_0 - U_1E_1 \quad .$$

Utilizing α and allowing for possible changes in U-238 content as well as U-235 and U-236, one can derive the stoichiometric relationship

$$\text{U-235 burned up} = \frac{(Z_1/E_1) - (Z_0/E_0)}{(\alpha/(1+\alpha)) + (Z_1/E_1)} \quad U_0E_0 \quad .$$

When the necessary isotopic ratios are known for large segments of processed fuel, the stoichiometric relationship can be extremely valuable for checking the burnup computation based on reactor power. Errors in the variables involved in the calculation can be propagated to determine the error associated with the burnup. This, along with the error in the burnup computation from total power, can be used as a criterion for judging the significance of differences between the methods of calculation.

2. Burnup Apportionment by Position. From the total energy generated per reactor cycle, each individual fuel element must be assigned its proportionate share of the U-235 burnup. This would be no particular problem if entire reactor cores were removed at one time and the subsequent chemical reprocessing were on a core basis. Since this is not usually the case, problems of apportionment must be considered.

Variations in fission rate within a test reactor core are usually large and have a major effect upon the reliability of any estimate of burnup in a given lattice position. This is aggravated by the usual practice of cycling fuel elements in the

reactor from the periphery of the core to the center. Also, because many experiments are tied into the reactor control circuitry, occasional scrams occur. If at these times the reactor is caught by xenon, fresh fuel elements are inserted to replace a few which are only partially spent in order to avoid unnecessary delays. After a week or so these partially spent elements may be returned to the reactor resulting in spent reactor elements with a heterogeneous irradiation history.

It is extremely difficult to collect sufficient data for an accurate estimate of fission rate or integrated fission by lattice location. The fuel lattice burnup apportionment factors are computed from neutron flux measurement data collected at essentially zero power. Flux measurements are of particular importance when core configurations involve previously irradiated elements because the U-235 content of the fuel assemblies must be known within small limits to minimize reactor hazards and provide valid nuclear material control data.

The flux intensity at a particular lattice position in a research reactor can be determined by the following procedure. A cobalt wire is inserted into each fuel element which extends the full length of the element. After neutron exposure, the wire is gamma scanned by increments. The resulting count data show the accumulated flux intensity along the wire. The flux curve is then integrated and the average height for each core position determined and added to that of other core positions to arrive at an average flux value for the total core. The average flux at each position is reported relative to the average core flux.

A slightly different technique can be used to determine flux patterns if a critical test facility is available. Gold activation foils are inserted into the center of each fuel element in a core mock-up at the critical facility. The relative counting rates of the foils give flux intensity factors for the fuel position. Gold is used because it is more sensitive than cobalt to the low neutron fluxes normally found in zero power test facilities. The apportionment factors are slightly influenced by irradiation time and total exposure; hence, factor corrections may be required in some cases.

The flux patterns near control rods are also measured in the critical facility. Some control rods have fuel sections which are drawn into the core as the poison is withdrawn. The burnup apportionment to this fuel is dependent of the rod intime as well as its core position.

A simplified computation for each core position is made in the following manner:

$$\text{Burnup} = \frac{(X_i)(Y_i)}{\Sigma X_i Y_i} (1+\alpha)(1.05)\text{MWD}$$

where

X_i = average flux intensity for i^{th} position

Y_i = U-235 content of fuel in i^{th} position and

$\frac{(X_i)(Y_i)}{\Sigma(X_i)(Y_i)}$ = weighting factor for i^{th} position.

The weighting factors can also be estimated by applying perturbation theory to the flux measurements taken at the start of a cycle and taking into consideration the effects of the

varying experimental loads. An alternative procedure is a statistical analysis of reactor burnup computations obtained by perturbation theory. A set of empirical relationships can be derived to prorate the total core burnup to the various lattice positions. The simplified procedure assumes that percentage burnup is a linear function of both lattice position and uranium content at the beginning of the cycle. Several lattice positions yield essentially the same burnup and can be compared for significant differences. In cases where no significant differences exist a single equation is sufficient. Since several equations are involved over the entire reactor core, it is usually necessary to normalize the weighting factors such that their sum is unity. Statistical uncertainties are extremely difficult to calculate for the weighting factors computed in this manner. Fortunately, the uncertainties associated with the lattice position weighting tend to cancel for an entire cycle.

E. Power Recording Instrumentation

The power generation is calculated using appropriate instrumentation and recorded by the Foxboro Power Calculator. This measures the coolant flow rate and temperature rise and applies an appropriate conversion factor. The reactor power is measured continuously.

1. Flow Rate Measurement. The flow measurement is by a flow transmitter which measures and indicates differential pressure (ΔP) across a Gentile Flow Tube. The Gentile flow tube is a short spool piece, the inner circumference of which is equipped with two groups of nozzles, one pointing upstream to the

main flow and the other pointing downstream. The nozzles are connected together by pressure rings attached to the high and low pressure sides of the flow transmitter. The signal is electrically cabled to the calculator unit.

2. Dynatherm Bridge. The Dynatherm Bridge is an electrical circuit containing eight resistance bulbs. The bulbs are installed in the inlet and outlet streams. They are positioned at intervals around the main flow pipe. Each bulb is inserted approximately six inches into the main flow duct. The bridge receives the output signal of the flow transmitter and produces an output voltage which is directly proportional to the product of the flow rate and ΔT.

3. Power Calculating Unit. The power calculating unit uses the above signals to calculate reactor power.

A friction compensating unit makes a correction to the calculated power. The result is sent to the power recorder unit which is calibrated directly in megawatts.

4. Calibration. This electronic equipment is calibrated with standards housed within the calculator unit. The flow rate and ΔT signals can be switched out of the circuit and standard signals injected. This substitution permits calibration of the flow recorder, dynatherm bridge and the power recorders. These calibration checks are performed during each shutdown. More extensive calibration checks on all electronic equipment measuring ΔT and water flow are performed annually.

F. Other Measurements

1. Mass Measurements. A Mettler HP15 portable electronic balance (accurate to 0.1 grams over the range 0 to 15 kilograms) is used for all receiver, inventory and shipper mass measurements performed by MC & A.

2. Non-Destructive Assay. The SAM II portable dual channel gamma-ray spectrometer with sodium iodide crystal is used to measure containers of pellets and bulk uranium metal. Measurements are made by comparing measured gamma activity to that of known standard.

3. Uranium and U-235 Chemical Assay. Uranium and U-235 in dissolved metal or pellet samples are determined by the Davies-Gray method and mass spectroscopy, respectively.

V. ANALYSIS AND CONTROL

A. Introduction

The nuclear material control system must be continuously evaluated in order to determine whether the system provides the information and control for which it was designed. This involves technical evaluation of the process and the control system, periodic audit and inspection and recommendations for corrective action where required.

It is to be noted that the nuclear materials control function for research and test reactors as outlined above is highly integrated through the Operation, Technical, and Nuclear Material Control groups. The Statistical group supports this function by providing the necessary procedures to insure the

quality of the data generated and to interpret the measurements inherent to the system. For such a system to be effective the usual communication problems of large organizations must be solved so that all individuals involved in nuclear material control are aware of their responsibilities and contributions to the overall effort. The location of the Nuclear Materials Control group in the organizational structure can also influence the effectiveness of the control system.

B. Analysis

Since measurement data are a fundamental part of the nuclear material control system of a research and test reactor, the measurement methods must be continuously evaluated to insure that they provide accurate information. The measurement methods of primary concern are those used for the nondestructive assay of incoming fuels, for the calculation of burnup and for the apportionment of burnup among the fuel elements.

1. Fuel Content Measurements. The principal components of error in nondestructive fuel measurements are:

(a) the systematic and random errors associated with the calibration.

(b) the systematic and random errors associated with gamma counting of a fuel plate.

The systematic error associated with the U and U-235 in the standards is due to the weighing and the bias of the chemical analysis. The random errors are those associated with the wet chemistry and mass spectrometry determination of U and U-235 in the standards.

The systematic and random errors associated with scanning a fuel plate include equipment bias and gamma-ray counting statistics. Estimates of these errors are given below.

To be Associated with	Relative Precision (2σ) of Random	Relative Precision (2σ) of Systematic
Standards	.09%	.06%
Scanning	1.2 %	.05%

In a given application the U-235 in a plate is determined by the equation:

$$U_p = U_\ell + \frac{\left(\frac{C_p - C_\ell}{T_p - T_\ell}\right)}{\left(\frac{C_h - C_\ell}{T_h - T_\ell}\right)} (U_h - U_\ell)$$

where

U_p = U-235 productive plate value

U_ℓ = U-235 low standard plate value

U_h = U-235 High standard plate value

C_p = Production plate count

C_h = High standard plate count

C_ℓ = Low standard plate count

T_p = Production plate time scanned

T_h = High standard plate time scanned

T_ℓ = Low standard plate time scanned

The error associated with a given measurement or group of measurements can be determined by standard error propagation techniques.

2. Burnup. Using the equations:

$$\text{Burnup} = B_{U\text{-}235} = 1.05(1+\alpha)\text{MWD}$$

and

$$\text{Grams U lost} = B_U = 1.05(1-.004\alpha)\text{MWD}$$

and standard error propagation procedures the following expressions for the variances of burnup and uranium loss are found.

$$\sigma^2 BU\text{-}235 = B^2_{U\text{-}235}\left[\frac{\sigma^2 1.05}{(1.05)^2} \quad \frac{\sigma^2_\alpha}{(1+\alpha)^2} \quad \frac{\sigma^2 MWD}{MWD^2}\right]$$

and

$$\sigma^2 BU = B^2_U\left[\frac{\sigma^2 1.05}{(1.05)^2} + \frac{0.004^2\sigma^2_\alpha}{(1\text{-}0004\alpha)^2} + \frac{\sigma^2 MWD}{MWD^2}\right]$$

The following are typical estimates for the component standard deviations

$$\sigma_\alpha \cong 0.05$$

$$\frac{\sigma 1.05}{1.05} \cong 0.05$$

$$\frac{\sigma MWD}{MWD} \cong 0.05$$

It should be noted when calculating uncertainties associated with several cores and cycles σ_α and $\sigma_{1.05}$ should be treated as systematic errors whereas σMWD can be considered a random error.

C. Errors of Apportionment

No formal analysis of apportionment errors has been made since there is no method of validating the computation short of dissolving individual fuel elements.

D. Inventory Sampling Plans

Sampling plans are devised to remeasure approximately 10% of the item inventory annually and to insure that all the items are inventoried over a given period of time.

The assessment of the probability of detecting specified losses within specified time periods depends on the nature of the hypothesized adversary action. A divertor may attempt to gain a certain amount of material by either taking whole items, taking items and substituting fakes, or taking parts of items. The level of protection provided by material accounting against each of these threats is different.

In the case of simple whole item removal the 100% piece count taken at inventory time guarantees detection within an inventory period (two months for SNM storage/use).

If substitution is involved, a system's ability to detect the diversion relies on the verification measurements taken during inventories. With five substituted items the probability of detection within a year with 10% bimonthly sampling is high. Furthermore, the detection probability increases as more items are diverted.

The situation which is most difficult to analyze is the case of small removals or partial substitutions for many items. The detection probability relies heavily on sampling for removals of amounts just less than the amount the verification measurement system is certain to detect, since relatively few such removals would be needed to obtain a specified amount of material. As the amount diverted per item decreases and the number of marred items

increases, the detection probability dependence shifts to whether the verification measurement systems can detect the resulting small bias. A major computer simulation study similar to the one described by D. D. Cobb, et al. in "Concepts for Inventory Verification in Critical Facilities" (Los Alamos Scientific Laboratory Report LA-7315, December, 1978) would be required to assess the many possibilities.

In summary, the material control and accounting system is designed to provide 100% confidence of detecting whole item removals of SNM within 2 months and 98% confidence of detecting the loss of 5 effective kilograms of U-235 in whole item substitutions in one year.

E. Statistical Techniques Related to Calibrations

All calibrations of scales, balances and assay meters are carried out under statistically designed programs and the data are analyzed by statisticians.

F. Statistical Analysis of Inventory Differences

Since most of the inventory material in a Research Reactor Facility is item accountable, most of the material balance areas do not have an inventory difference. However, whenever measurements are made and balances are constructed on those measurements, limits of error are estimated and the inventory difference is evaluated on the basis of limits of error.

G. Analysis of Shipper/Receiver Difference

1. Receipts that can be Destructively Assayed. Samples of pellets or uranium metal scrap which are amenable to wet chemical analysis are assayed for U-235 by net weight times U

concentration times U-235 assay. Error propagation shows that the relative variance of such a product is the sum of the relative variances of the individual factors.

2. Receipts that cannot be sampled. Receipts of items such as rods, plates or fuel assemblies which cannot be sampled without altering the item are analyzed using nondestructive analysis equipment. Error analysis of plates is described in a previous section.

3. Shipment of Spent Fuel. Shipment of spent fuel to the reprocessing facility is based on burnup calculations. No attempt is made to directly assay the fuel elements.

H. Dissolution Data

Consideration should be given to collecting burnup and U-235 depletion information from the analysis of fuel batch solutions in the subsequent chemical reprocessing of spent fuel. These data, along with their uncertainties, can provide indications of bias in the burnup computation made from reactor heat values. The uncertainties involved in this comparison will be primarily those associated with the apportionment of burnup to groups of fuel elements in the dissolver batch unless large amounts of fuel processing data are accumulated. Hence, means of improving the apportionment factors are very important. Future development of better measurement methods which provide a more exact knowledge of flux distributions within research and test reactor cores may increase the value of such comparisons.

I. Control Tamper Proof Devices

1. Introduction. Tamper proof devices are used extensively in Research Reactor Material Control because of the high dependence on item control. Containers of scrap, pellets, fuel plates and even fuel elements that are in long term storage are sealed for materials control purposes.

The reliability of a tamper indicating seal (TIS) program is directly related to the control of the seals used as tamper indicating devices (TID). This control is facilitated by the following:

- Identification of a seal custodian.
- Secured seal storage.
- Logging and identification of seal use.
- Accountability of seals.
- Documented and witnessed records.
- Used seal disposal.
- Auditing of procedure.

2. Type of Seals Used. Although there are several types of seals which might have application in a Research and Test Reactor Facility, the two most commonly used will be described.

a. Pressure-Sensitive Seal. Label seals are constructed of sheet vinyl or paper with pressure sensitive backing. The seal material will tear if attempts are made to peel it. Attempts to remove the seal by means of solvents will cause the background ink to run or will destroy the seal. The seals are consecutively numbered with printed or perforated dots.

b. Cup/Wire Seals. The Cup/Wire seal is commonly referred to as a Type E seal. It consists of three sheet metal stampings, two of which are fastened together to form the bottom of the seal. The third stamping forms a solid top piece. The seal is installed by threading wire through the item to be sealed and then through the holes in the seal bottom and fastening the two wire ends together with a crimp-type sleeve or other device. The top is snapped into the bottom, thereby capturing the wire juncture within the metal cup enclosure.

J. External Audit

Periodically an independent agency must survey the nuclear material control records and their supporting documents. The frequency of an audit may vary but generally it should be performed at least once a year. The audit survey includes an examination of the journals, ledgers, and source documents to determine the accuracy of the record keeping. The ending inventories are verified by physical identification where possible and inventory listings are compared with subsidiary supporting records. The listings of all unirradiated fuel elements in the inventory are verified by element identification while a random sample of the irradiated fuel elements is physically identified by reading the identification numbers of the elements in the cooling basins. Reactor inventories are verified by referring to reactor charge records.

A review of measurement and statistical procedures includes an examination of the fuel element assay techniques and a technical appraisal of the uncertainties associated with reported

values on assay results. Consideration is given to the burnup calculation including a review of uncertainties associated with water flow calibration data and heat loss calculations.

A sampling of the experimental items is checked against subsidiary control records. Custodian responsibility is checked by requiring the responsible project engineer to locate and identify an experiment selected by the survey team under a sampling plan. Monthly inventory reports are checked against ledger balances to confirm proper reporting procedures.

K. Internal Audits

Periodic audits are conducted by members of the internal audit staff using auditing procedures somewhat similar to those used in the external audit. The survey period is dictated by management and is usually conducted on an impromptu basis.

INTERNATIONAL TRAINING COURSE ON NUCLEAR MATERIALS ACCOUNTABILITY FOR SAFEGUARDS PURPOSES

SESSION #26: SAFEGUARDS SYSTEM DESIGN AND APPLICATION

SPEAKER: Dr. Donald D. Cobb

Los Alamos Scientific Laboratory
Los Alamos, New Mexico USA

Wednesday, June 4, 1980
1:30 p.m.

BIOGRAPHY

Present Position: Staff member in Safeguards Systems Analysis Group, Q-4, of Los Alamos Scientific Laboratory.

INTERNATIONAL TRAINING COURSE ON NUCLEAR MATERIALS ACCOUNTABILITY FOR SAFEGUARDS PURPOSES

Session Objectives

SESSION #26: SAFEGUARDS SYSTEM DESIGN AND APPLICATION

The general structural features of a national system of accountability and control are considered. Techniques for carrying out the design of such systems, including modeling and simulation, are discussed. Measures of system performance and methods for evaluating those measures are described. Examples of the safeguards design process for selected fuel-cycle facilities will be presented.

After the session, participants will be able to

1. Identify the major components of an effective national system of accountability for nuclear materials.

2. Describe qualitatively methods for designing an accountability system.

3. Describe suitable performance measures for an effective accountability system.

4. Identify special safeguards design considerations and applications to the selected fuel-cycle facilities.

SESSION 26: SAFEGUARDS SYSTEM DESIGN AND APPLICATIONS

Donald D. Cobb
Los Alamos Scientific Laboratory

I. INTRODUCTION

The nuclear fuel cycle consists of a series of operations beginning with the mining of uranium ore and ending with the interment of radioactive waste (Fig. 1). As much of the world moves toward large-scale utilization of nuclear energy during the last decades of this century, more stringent controls are required on the nuclear materials used by the nuclear power industry. There are several reasons for this--the increased incidence of organized, overt terrorism; the potential widespread use of plutonium and highly enriched uranium as nuclear fuels; publicity about the fabrication of crude nuclear bombs; the hazards of malevolent dispersal of radioactive material; and worldwide concern over the proliferation of nuclear weapons.

The problem of maintaining strict accounting and control over all nuclear material will be exacerbated by the nuclear power demands of the future, which will require high-throughput facilities possibly supporting any of several alternative fuel cycles. Spent-fuel reprocessing facilities having the capability to process over 100 kg of plutonium per day have been built, and even larger ones are being designed. The scale of these operations has forced a reassessment not only of facility design, construction, and process operation, but also of the safeguards methods employed to prevent unauthorized use of the nuclear materials contained therein.

This lecture includes principles that can serve as guidelines for the design of effective nuclear materials control and accounting systems. These guiding principles should be of particular

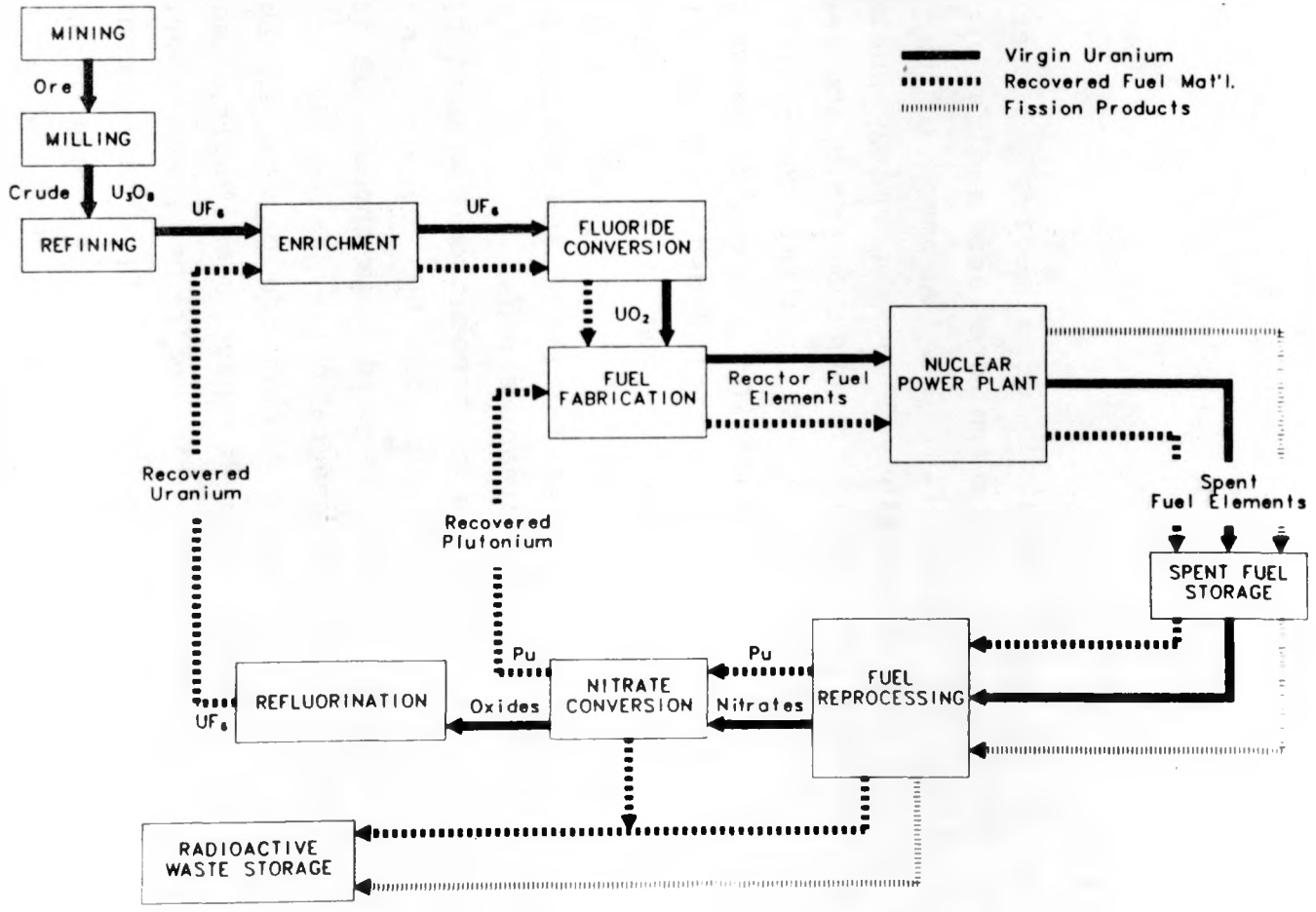


Fig. 1.
The power reactor nuclear-fuel cycle.

value to those comtemplating future nuclear processes and facilities that must meet stringent safeguards criteria. After a brief review of the objectives and the structures of national and international safeguards systems, features of advanced materials control and accounting systems are described.

II. THE STATE'S SAFEGUARDS SYSTEM

The essential purpose of any nuclear fuel cycle plant is to produce, process, or consume nuclear materials safely and economically. Coordination between plant and safeguards designers at the earliest design stages is the most efficient and effective means of achieving both plant and safeguards goals. A comprehensive safeguards strategy includes four principal functions:

1. Excluding all unauthorized persons from the facility and selectively excluding others from sensitive areas within the plant;
2. Monitoring all activities involving nuclear material to determine whether each such activity is consistent with safeguards requirements and with normal expected facility operation;
3. Accounting for all nuclear material in the facility to determine whether the correct amounts of all materials are present in their proper locations;
4. Responding to the safeguards status of the facility and reporting to the regulatory authority.

These functions are accomplished by several subsystems, including the physical protection system (PPS), the process monitoring system (PMS), and the materials measurement and accounting system (MMAS).

Figure 2 shows a safeguards system structure that has been developed through numerous interactions with the U.S. nuclear

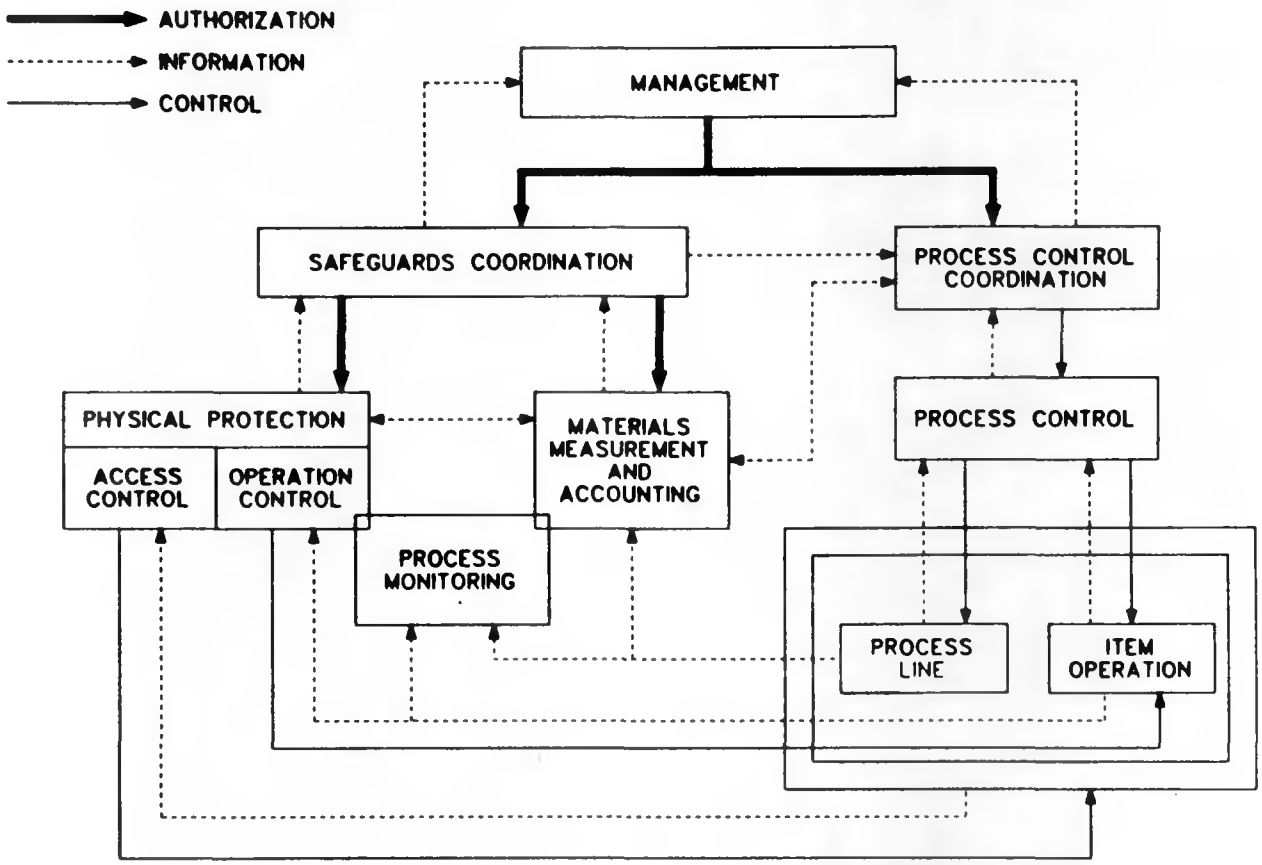


Fig. 2.
Structure of the safeguards system.

industry and the safeguards community. The safeguards coordination unit (SCU) supervises nuclear material safeguarding in the plant. As the focal point for safeguards decisions, the unit interacts with management and process control coordination to ensure effective safeguards while minimizing process disruptions. The SCU has three primary functions: (1) data collection and processing, which is required for (2) safeguards condition assessment, which in turn is the basis for (3) the response decision.

The physical protection system controls personnel entry and exit for the plant and for restricted areas inside. The system emphasizes the use of automated equipment and sufficient guard forces to provide the initial response in an emergency. The PPS expands the conventional security functions, such as access control, to include control of item-handling operations. Item operations control is applied to those portions of the facility, such as feed and product storage areas, that are outside the closely coupled process line and in which uninterrupted material flow is not critical to process operation.

The process monitoring system combines elements of both physical protection and materials accounting and provides supplementary information regarding compliance of actual process operating modes with approved procedures. The concept may be regarded both as an extension of physical-protection monitoring and surveillance functions into the process line, and as an upgrading of process-control monitoring devices (or appropriate placement of them) to allow gross materials accounting. The PMS collects timely information to detect process abnormalities. The system uses plant-grade instrumentation wherever possible to assess materials balances on transfers of process materials.

The materials measurement and accounting system combines conventional chemical analysis, weighing, and volume measurements with the timely measurement capability of on-line non-destructive

assay (NDA) instrumentation to provide rapid and accurate assessment of the locations and amounts of material.

III. THE INTERNATIONAL SAFEGUARDS SYSTEM

In the early 1960s, as more and more countries acquired nuclear power plants, there was increasing concern worldwide over the possible misappropriation of nuclear material, facilities, and technology for use in weapons. As a result, safeguarding of nuclear material became important internationally. The basis for most current international safeguards arrangements is the Treaty on Non-Proliferation of Nuclear Weapons (NPT), which has been agreed to by more than 100 nations. The detailed terms and conditions under which specific facilities are safeguarded are negotiated with the IAEA, in accord with the general conditions of Article III of the NPT, as set forth in IAEA document INFCIRC/153. The objective of international safeguards, as declared in these documents, is the "...timely detection of diversion of significant quantities of nuclear material from peaceful nuclear activities..., and the deterrence of such diversion by the risk of early detection." The details of compliance are negotiated between the IAEA and the host nation on a facility-by-facility basis and are documented in so-called "Subsidiary Arrangements" and "Facility Attachments."

Agreements conforming to INFCIRC/153 require that "...the State shall establish and maintain a system of accounting for and control of all nuclear material subject to safeguards..., and that such safeguards shall be applied in such a manner as to enable the Agency (the IAEA) to verify, in ascertaining that there has been no diversion of nuclear material from peaceful uses to nuclear weapons or other nuclear explosive devices, findings of the State's system." Furthermore, the IAEA "shall make full use

of the State's system of accounting for and control of all nuclear material subject to safeguards under the Agreement, and shall avoid unnecessary duplication of the State's accounting and control activities."

Thus, a major role of the international safeguards system is the independent verification of the validity and integrity of facility-generated materials accounting data as a means of confirming that the State's undertakings to limit nuclear activities to peaceful purposes are being fulfilled. Figure 3 indicates the relationships between the State and the IAEA safeguards systems. Clearly, the effectiveness of the international safeguards system depends on the quality of the State's safeguards system that supplies the input data. The Agency must make full use of the State's safeguards system and avoid unnecessary duplication. The inspector's verification activities consist of independent, confirmatory measurements of materials and audits of facility records, as well as independent observations of the integrity of the containment. The result of the Agency's verification activities is "a statement, in respect of each materials balance area, of the amount of material unaccounted for over a specific period, giving the limits of accuracy of the amounts stated."

Effectiveness criteria for international safeguards are negotiated between the IAEA (Agency) and the State (operator) on a case-by-case basis and are not quantifiably documented. Values of "goal quantities" for the detection of diversion have been proposed by the IAEA, but have not been generally accepted by Member States. These "goals" are derived from estimates of the quantities of nuclear materials required to produce an explosive device and the times necessary to convert these materials to that purpose. The goals include the detection of the diversion of:

- 8 kg of plutonium in irradiated fuel in 1-3 months.
- 8 kg of plutonium in unirradiated material in 1-3 weeks ("abrupt diversion").

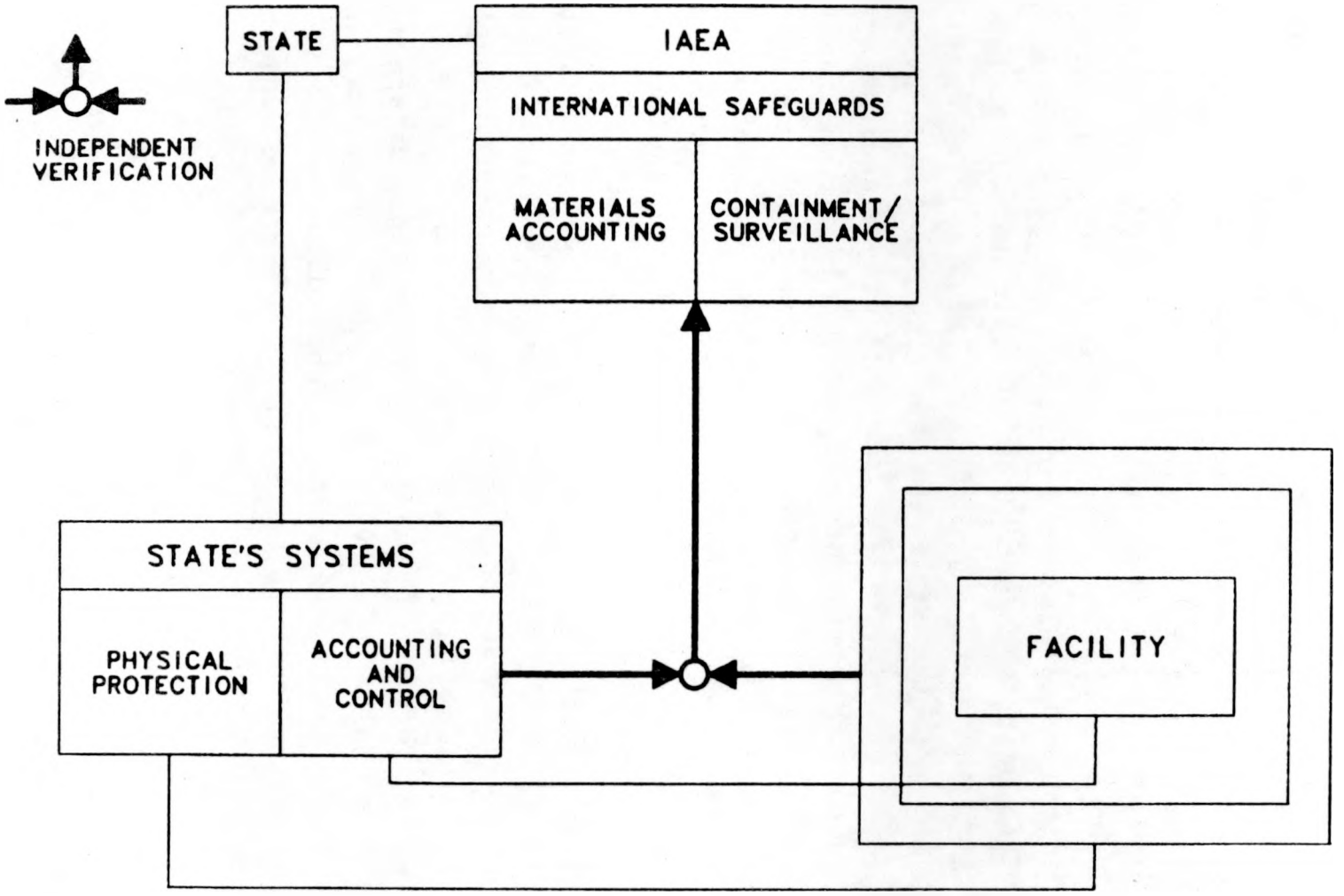


Fig. 3.
Relationship between the State and IAEA safeguards systems.

- 8 kg of plutonium over a period of 1 year ("protracted diversion").
- 75 kg of uranium-235 contained in low-enriched (<20%) uranium over a period of 1 year.
- 25 kg of uranium-235 contained in high-enriched (>20%) uranium in 1-3 weeks.
- 25 kg of uranium-235 contained in high-enriched uranium in 1 year.

IV. THE MATERIALS MEASUREMENT AND ACCOUNTING SYSTEM (MMAS)

At present, materials accounting primarily relies on forming materials balances following process shutdown, cleanout, and physical inventory. The materials balance area (MBA) often contains the entire plant or a major portion of the process. The classical materials balance is formed by adding all measured receipts to the initial measured inventory and subtracting all measured removals and the final measured inventory. During routine production, materials control is vested largely in administrative and process controls augmented by secure storage of discrete items.

Although current accounting practices are essential to safeguards control of nuclear materials, they have inherent limitations in sensitivity and timeliness. The sensitivity limit results from measurement uncertainties that may conceal losses of significant quantities of nuclear material in large plants. The timeliness of materials accounting is limited by the frequency of physical inventories. There are practical limits on how often a facility can be shut down for inventory and still remain productive.

Improvements in materials measurement and accounting can be obtained through implementation of dynamic (or near-real-time)

materials accounting concepts. This approach combines conventional chemical analysis, weighing, and volume measurements with the on-line measurement capability of NDA instrumentation to provide rapid and accurate assessment of the locations and amounts of nuclear materials in a facility.

To implement dynamic materials accounting, the facility may be partitioned into discrete accounting envelopes, called unit-process accounting areas (UPAAs). A UPAA can be one or more chemical or physical processes and is chosen on the basis of process logic and the ability to draw a materials balance, rather than on geography, custodianship, or regulatory requirements. By dividing a facility into unit processes and measuring all significant materials flows and in-process inventories, quantities of material much smaller than the total plant inventory can be controlled on a timely basis. Also, any discrepancies are localized to that portion of the process contained in the UPAA.

Materials balances drawn around UPAAs during plant operation are referred to as dynamic materials balances to distinguish them from materials balances drawn around MBAs after cleanout and physical inventory. Ideally, dynamic materials balances would be zero unless losses of nuclear material have occurred. In practice, they never are zero for two reasons. First, measured values are never exact because of the errors inherent in any measuring procedure. Second, constraints on cost or effects on processing operations may dictate that not all components of a dynamic materials balance be measured equally often; therefore, even if the measurements were exact, the dynamic materials balances would not be exactly zero until all sidestreams and holdup residuals are measured. In the interim, historical data can be used to estimate unmeasured material, and then the estimates can be updated when additional measurements become available.

The mechanism within the facility safeguards system that ensures the quality of the measurement data is an organized measurement-control program designed to measure and monitor the accuracy and precision of each instrument in the system and to verify application of NDA techniques in compliance with accepted practices. This measurement-control program is an integral part of the safeguards system in that the computer system records and monitors both measurement and calibration data. A performance history thus is maintained for each instrument.

Dynamic materials accounting must be applied flexibly to be useful. Proper application must take account of specific process design and operating features and should be graded according to the strategic value and vulnerability of process materials.

Computer-generated control charts derived from measurements and process operating characteristics can be used to indicate thefts, losses, or excessive holdup. This detailed control forces a potential divertor to remove material in sufficiently small quantities that his individual removals will be masked by measurement uncertainties. Thus, to obtain a usable quantity of material, the divertor must commit many diversions with the concomitant high risk of detection by the accounting system, surveillance instruments, and the physical protection system.

V. MMAS DESIGN METHODOLOGY

A. Modeling and Simulation

Because large fuel-cycle plants are not yet in operation, computerized modeling and simulation of each process and measurement system are used in developing preliminary MMAS designs. The modeling and simulation approach requires a detailed dynamic model of the process based on actual process design data. Design concepts are evolved by identifying key measurement points and appropriate measurement techniques, comparing possible materials

accounting strategies, developing and testing appropriate data-analysis algorithms, and quantitatively evaluating the proposed MMAS's capability to detect losses. By using modeling and simulation techniques, the effects of process and measurement variations over long operating periods and for various operating modes can be studied in a short time.

Computer codes can be used to simulate the operation of the reference process using standard Monte Carlo techniques. Input data include initial values for all process variables and values of statistical parameters that describe each independent, stochastic process variable. These data are best estimates obtained from process designers and operators. Each unit process is modeled separately. When a process event occurs in a particular unit process, the values of nuclear material flows and in-process inventories associated with that unit process are computed and stored in a data matrix. These data are available for further processing and as input to computer codes that simulate accounting measurements and materials balances.

The nuclear materials flow and inventory quantities from a process model are converted to measured values by applying simulated measurements. Each measurement type is modeled separately; measurement errors are assumed to be normally distributed, and provisions are made for both additive (absolute) and multiplicative (relative) errors. Significant measurement correlations are included explicitly. The measurement models are based on the performance of similar instrumentation characterized in both laboratory and field applications to similar materials. Simulated measurements are combined to form materials balances under various strategies for materials accounting.

B. Measurement Error Models

Because the sensitivity of any MMAS is limited by intrinsic measurement errors, measurement models and error estimates for

various types of instrumentation are used to predict MMAS performance. A simple measurement model is given by

$$m = M(1 + \varepsilon + \eta) \quad , \tag{1}$$

where m is the measured value of a true quantity M. The measurement errors, ε and η, are discussed below. This model applies when error standard deviations are expressed on a relative basis and is appropriate for measurement situations in which the associated error tends to be proportional to the quantity being measured.

The measurement errors have been grouped in two categories, instrument precision ε and calibration η, and both are regarded as observations on random variables. The instrument precision, ε, represents the deviation of the measured value from the true quantity caused by the scatter or dispersion in a set of individual measurement results (for example, the uncertainty caused by counting statistics in NDA measurements). The calibration error, η, represents those errors that persist, unchanged, throughout a limited set of measurements as a result of the uncertainty in converting raw measurement results into the quantity of interest (for example, converting counts to plutonium mass for NDA measurements). The latter errors are the most difficult to estimate because they include uncertainties in standards, calibration parameters, instrument environment, and measurement control procedures. There may be several independent η-error components, each arising from a different error source that correlates a different set of measurements. A major function of measurement control and quality assurance is to identify the sources of measurement error and to control them through appropriate calibration procedures.

The error random variables (ε and η) are assumed to have means of zero and variances σ_ε^2 and σ_η^2 respectively. This implies that all significant measurement biases have been identified and corrected for in the measurement control program. The variance σ_m^2 of the measured value m is given by

$$\sigma_m^2 = M^2\left(\sigma_\varepsilon^2 + \sigma_\eta^2\right) \quad . \tag{2}$$

To simulate a series of measurements from a given instrument, one value of ε is sampled from the appropriate ε-error distribution for each measurement, whereas a new value of η is sampled from the appropriate η-error distribution only when a calibration is performed. All measurements from the same instrument having the same η error are correlated. These correlations may dominate the materials balance uncertainty. The covariance between the i^{th} and j^{th} measurements is given by

$$\sigma_{ij} = M_i M_j \sigma_\eta^2 \quad . \tag{3}$$

C. Ideal Process Example

A simple example will illustrate materials accounting concepts and principles. Figure 4 represents an ideal process having a daily throughput of 50 kg of nuclear material consisting of twenty-five 2-kg batches and no process losses. The in-process inventory of nuclear material is 25 kg, and the residual holdup is 5 kg after shutdown and cleanout, which is postulated to occur once each month. The entire process is contained in a single MBA (Fig. 4a), whereas storage areas for feed and product are in separate MBAs and are not shown.

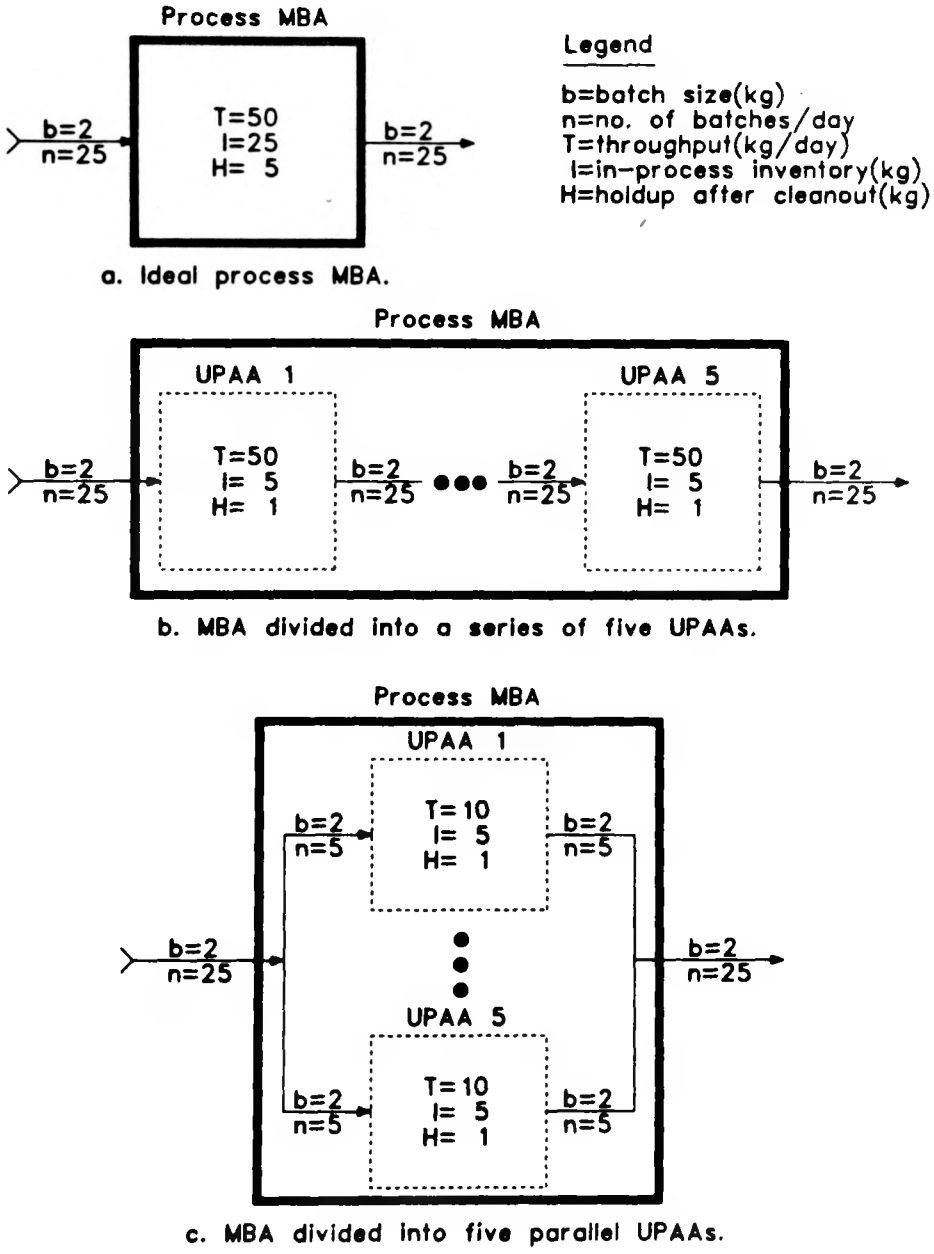


Fig. 4.
Ideal process block diagram.

Figures 4b and 4c show two possible divisions of the process MBA into UPAAs for dynamic accounting purposes. In Fig. 4b the MBA is divided into a series of five UPAAs. To accomplish this division, transfers of nuclear material between adjacent UPAAs and the in-process inventory in each UPAA must be measured. In Fig. 4c, the MBA is divided into five parallel UPAAs. In this case the input, output, and inventory of each UPAA must be measured. In practice, the division of the MBA depends on the process configuration.

Measurement errors in dynamic materials balances applied to the ideal process can be calculated using the measurement model described in the previous section [Eqs. (1-3)]. For a given accounting period during which N batches are processed, the dynamic materials balance, MB_N, for one UPAA is given by

$$MB_N = \Delta I_N + T_N \quad , \tag{4}$$

where ΔI_N is the net change in nuclear material inventory and T_N is the net transfer of nuclear material (inputs minus outputs) across the UPAA. If there were no measurement errors, MB_N would be exactly zero and, if the process were operated at steady state, ΔI_N and T_N would also be zero.

Measurement errors produce an uncertainty in MB_N having a variance σ^2_{MB} (assuming no correlation between transfer and inventory measurements) given by

$$\sigma^2_{MB} = \sigma^2_{\Delta I} + \sigma^2_T \quad . \tag{5}$$

Understanding the behavior of the inventory-change and net-transfer variances, $\sigma^2_{\Delta I}$ and σ^2_T, is basic to effective MMAS design.

1. Inventory-Change Variance. If the initial and final inventories, I_0 and I_N, are measured during the same calibration period (i.e., have the same η error), the variance, $\sigma^2_{\Delta I}$, of the net inventory change, ΔI, is given by

$$\sigma^2_{\Delta I} = \left(I_0^2 + I_N^2\right)\sigma^2_{\varepsilon I} + \left(I_0 - I_N\right)^2\sigma^2_{\eta I} \quad , \tag{6}$$

where $\sigma^2_{\varepsilon I}$ and $\sigma^2_{\eta I}$ are the ε- and η-error variances of the inventory measurements. Note that if the initial and final inventories are equal, $I_0 = I_N$, then $\sigma^2_{\Delta I}$ has the minimum value

$$\sigma^2_{\Delta I} = 2I_0^2\sigma^2_{\varepsilon I} \quad . \tag{7}$$

For a large class of process equipment, efficiency and economy dictate that the in-process inventory be held nearly constant during normal operation. Such near-steady-state operation benefits materials accounting by reducing the materials balance uncertainty. Furthermore, the condition $I_0 \simeq I_N$ implies that the dependence of σ_{MB} on $\sigma_{\eta I}$ is weak [Eq. (6)]; hence, a well-known value for $\sigma_{\eta I}$ is not required. This result is important because standardization of in-process inventory measurements may be difficult, especially for process equipment located in high radiation fields behind heavy shielding. The ideal process is assumed to satisfy the steady-state condition so that Eq. (7) holds. The inventory measurement error ($\sigma_{\varepsilon I}$ = 10% in this example) limits the dynamic accounting sensitivity over short accounting periods.

2. Net-Transfer Variance. The variance σ^2_T of the net material transfer T is given by

$$\sigma_T^2 = 2Nb^2\left(\sigma_{\varepsilon b}^2 + \sigma_{\eta b}^2\right) + 2N(N-1)b^2\sigma_{\eta b}^2 \quad , \tag{8}$$

where b is the input and output batch size, and $\sigma_{\varepsilon b}^2$ and $\sigma_{\eta b}^2$ are the ε- and η-error variances of the batch transfer measurements. For simplicity of presentation, the error variances of input and output batch measurements have been set equal in value (hence the factor of 2), but the two measurements are independent (i.e., uncorrelated).

The first term in Eq. (8) occurs whenever N input and N output batches are measured during the accounting period and is present even if the transfer measurements are uncorrelated. The second term in Eq. (8) accounts for pair-wise correlations among the transfer measurements [Eq. (3)]. The transfer measurements are correlated primarily because the instruments are not recalibrated during the accounting period. Note that the number of pair-wise correlations increases approximately as N^2; if N is sufficiently large, correlations make the dominant contribution to σ_T^2. The second term in Eq. (8) is equal to the first term after N_o batches have been processed, where N_o is given by

$$N_o = \frac{\sigma_{\varepsilon b}^2 + 2\sigma_{\eta b}^2}{\sigma_{\eta b}^2} \quad . \tag{9}$$

3. Effect of Calibration. The effect of correlations is reduced by recalibrating the transfer-measuring instruments. If the instruments are calibrated K times during the accounting period, and if n_k is the number of batches processed between the k^{th} and $(k + 1)^{th}$ calibrations, then σ_T^2 is given by

$$\sigma_T^2 = 2Nb^2\left(\sigma_{\varepsilon b}^2 + \sigma_{\eta b}^2\right) + 2b^2\sigma_{\eta b}^2 \sum_{k=1}^{K} n_k(n_k - 1) \quad , \tag{10}$$

where

$$N = \sum_{k=1}^{K} n_k \quad .$$

The number of correlation terms in this case increases approximately as Σn_k^2 rather than as N^2.

The effect on σ_T of daily versus monthly recalibration of the transfer-measuring instruments is shown in Fig. 5. The relative standard deviation (RSD), σ_T divided by the throughput Nb, is plotted as a function of the number N of processed batches. Values of $\sigma_{\varepsilon b}$ and $\sigma_{\eta b}$ have been taken to be 2% and 0.5%, respectively; these values correspond to $N_o = 18$ [Eq. (9)]. The net-transfer RSD varies as $[(\sigma_{\varepsilon b}^2 + \sigma_{\eta b}^2)/N]^{1/2}$ for small N and as $(\sigma_{\eta b}^2/K)^{1/2}$ for large N; that is, when the transfer correlations are dominant.

Correlations between transfer measurements limit the sensitivity of materials balances over sufficiently long accounting periods. Therefore, the parameters $\sigma_{\eta b}$ and K are especially important. The value of $\sigma_{\eta b}$ depends primarily on the measurement control procedures and on the quality of available calibration standards, whereas the value of K depends on how often the transfer-measuring instruments are recalibrated. Adequate measurement controls must include well-characterized standards for the transfer measurements and must provide for recalibration of the transfer-measuring instruments.

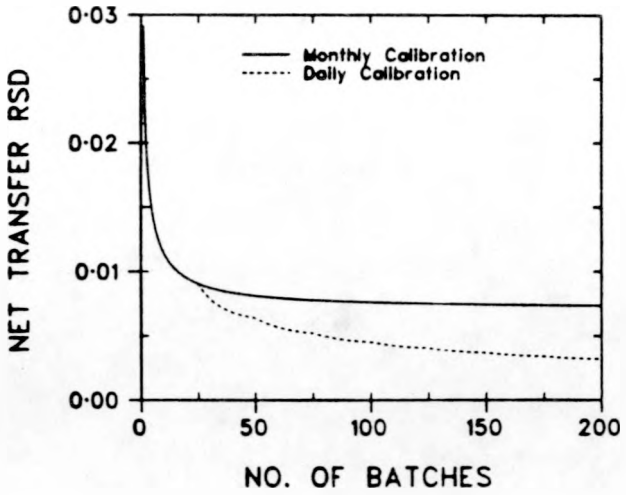


Fig. 5.
Effect of calibration on transfer measurement errors.

4. Results. Table I contains values of the standard deviation σ_{MB} of materials balances calculated for the ideal process. Results are given for four accounting periods: one batch, one day, one week, and one month (30 days), and for two transfer calibration periods, one day and one month. The inventory-change and net-transfer components of σ_{MB} are given separately. Calculated values are shown for one UPAA in a series arrangement, one UPAA in a parallel arrangement, and for the entire process MBA (see Fig. 4). Note that the data for the process MBA are a synthesis of the UPAA data. In practical application the capability of combining the same accounting data in different ways to form materials balances for various accounting envelopes provides obvious safeguards advantages that can be exploited by the MMAS software.

Examination of the data in Table I supports the following conclusions. For relatively short accounting periods the materials balance standard deviation (σ_{MB}) is determined primarily

TABLE I

MATERIALS ACCOUNTING IN AN IDEAL PROCESS

Accounting Period	Standard Deviation (kg)					
	Monthly Recalibration			Daily Recalibration		
	Series UPAA	Parallel UPAA	Process MBA	Series UPAA	Parallel UPAA	Process MBA
Batch						
Inventory change	0.71	0.71	1.58	0.71	0.71	1.58
Net transfer	0.06	0.06	0.06	0.06	0.06	0.06
Materials balance	0.71	0.71	1.58	0.71	0.71	1.58
Day						
Inventory change	0.71	0.71	1.58	0.71	0.71	1.58
Net transfer	0.45	0.14	0.45	0.45	0.14	0.45
Materials balance	0.84	0.72	1.64	0.84	0.72	1.64
Week						
Inventory change	0.71	0.71	1.58	0.71	0.71	1.58
Net transfer	2.59	0.60	2.59	1.20	0.38	1.20
Materials balance	2.68	0.93	3.03	1.39	0.80	1.98
Month						
Inventory change	0.14	0.14	0.32	0.14	0.14	0.32
Net transfer	10.72	2.23	10.72	2.48	0.79	2.48
Materials balance	10.72	2.24	10.72	2.48	0.81	2.50

by the size of the inventory (I) and the inventory instrument-precision RSD ($\sigma_{\varepsilon I}$). For longer accounting periods, σ_{MB} is determined by the number (N) and the size (b) of the transfers, the transfer calibration-error RSD ($\sigma_{\eta b}$), and the number (K) of transfer-instrument recalibrations.

The use of parallel process lines having reduced throughput and inventory for the same total plant throughput can markedly improve materials accounting sensitivity. Reduction of in-process inventory and accessibility of process equipment for inventory measurements are important design considerations. In this regard, large-capacity tanks present special accounting problems, and strict surveillance (process monitoring) measures should be considered in addition to materials accounting measures. Processing of relatively small batches and operation of the process near steady state generally enhance the capability of materials accounting.

From the point of view of materials measurements, rapid in-line or at-line assay techniques that provide precise inventory measurements and accurate transfer measurements, with provision for frequent recalibration of the transfer-measuring instruments, are generally favored. The period between physical inventories should be coupled to the buildup of transfer-measurement correlations; that is, after the materials-balance error standard deviation for the MBA becomes unacceptably large, a physical inventory is necessary to "rezero" the accounting system.

VI. DECISION ANALYSIS

The most promising measurement and accounting strategies are combined with statistical techniques in comparative studies of loss-detection sensitivities. Analysis of materials accounting data for indications of possible nuclear material diversion is one of the major functions of the MMAS. Diversion may occur in

two basic patterns: abrupt diversion (the single theft of a relatively large amount of nuclear material), and protracted diversion (repeated thefts of nuclear material on a scale too small to be detected in a single materials balance because of measurement uncertainties).

The use of unit-process accounting and dynamic materials balances enhances the ability to detect losses, but it also means that the operator of the safeguards system will be inundated with materials accounting data. Furthermore, the significance of any isolated set of measurements is seldom readily apparent and may change from day to day, depending on plant operating conditions. Clearly, it is imperative that the safeguards system operator be assisted by a coherent, logical framework of analysis tools.

Decision analysis, which combines techniques from estimation theory, decision theory, and systems analysis, is such a framework, and is well suited for statistical treatment of the dynamic materials accounting data that become available sequentially in time. Its primary goals are detection of nuclear material losses, estimation of the amount(s), and determination of the significance of the estimates.

The detection and estimation functions of decision analysis are based on classical hypothesis testing and modern state-variable estimation techniques. The systems analysis portion attempts to set thresholds for the hypothesis tests in a rational fashion, for example, by using utility theory to determine acceptable false-alarm and detection probabilites.

The detection function is based on acceptance of the hypothesis (H_1) that some (initially unknown) amount of nuclear material is missing versus the hypothesis (H_0) that all nuclear material is present. One useful kind of decision test compares a likelihood ratio to a threshold. The likelihood ratio is defined roughly as the ratio of the probability that nuclear material is missing to the probability that it is not, with the threshold determined by the desired false-alarm and detection probabilities.

A. Sequential Decision Tests

A typical sequential decision test is illustrated by Fig 6. The curves represent possible values of a test statistic that is derived from accounting measurements in the two cases of no missing nuclear material and missing nuclear material. These two cases are represented by the curves centered at 0 and at 3, respectively. The uncertainty in the statistic is represented by the widths of the curves. Clearly, if the amount of missing material is large, the two curves will not overlap significantly, and the decision is straightforward. However, if the amount of missing material is small, the two curves overlap and the possibility arises of making incorrect decisions. To make decisions that have the desired characteristics, two boundaries, ZU and ZL, are selected. If the statistic falls to the left of ZL, one concludes that there probably is no missing material. If the value falls to the right of ZU, one concludes that material may be missing. If the value falls between ZL and ZU, no decision is made until more data are gathered.

Note that two incorrect decisions can be made. One can conclude that there is nuclear material missing when there is none, denoted by the shaded area in Fig. 6 labeled FAP for false-alarm probability, or one can conclude that there is no missing nuclear material when in fact there is, denoted by the shaded area in Fig. 6 labeled MP for miss probability. The basic problem in detection is to minimize the probabilities of these two incorrect decisions.

B. Test Statistics

A variety of test statistics can be formed from the materials accounting data and tested sequentially for indications of diversion. Each statistic is based on a different assumption concerning the state of prior knowledge of the measurement errors and of the diversion strategy. Three of the most useful test statistics are the Shewhart, Cusum, and Uniform Diversion statistics.

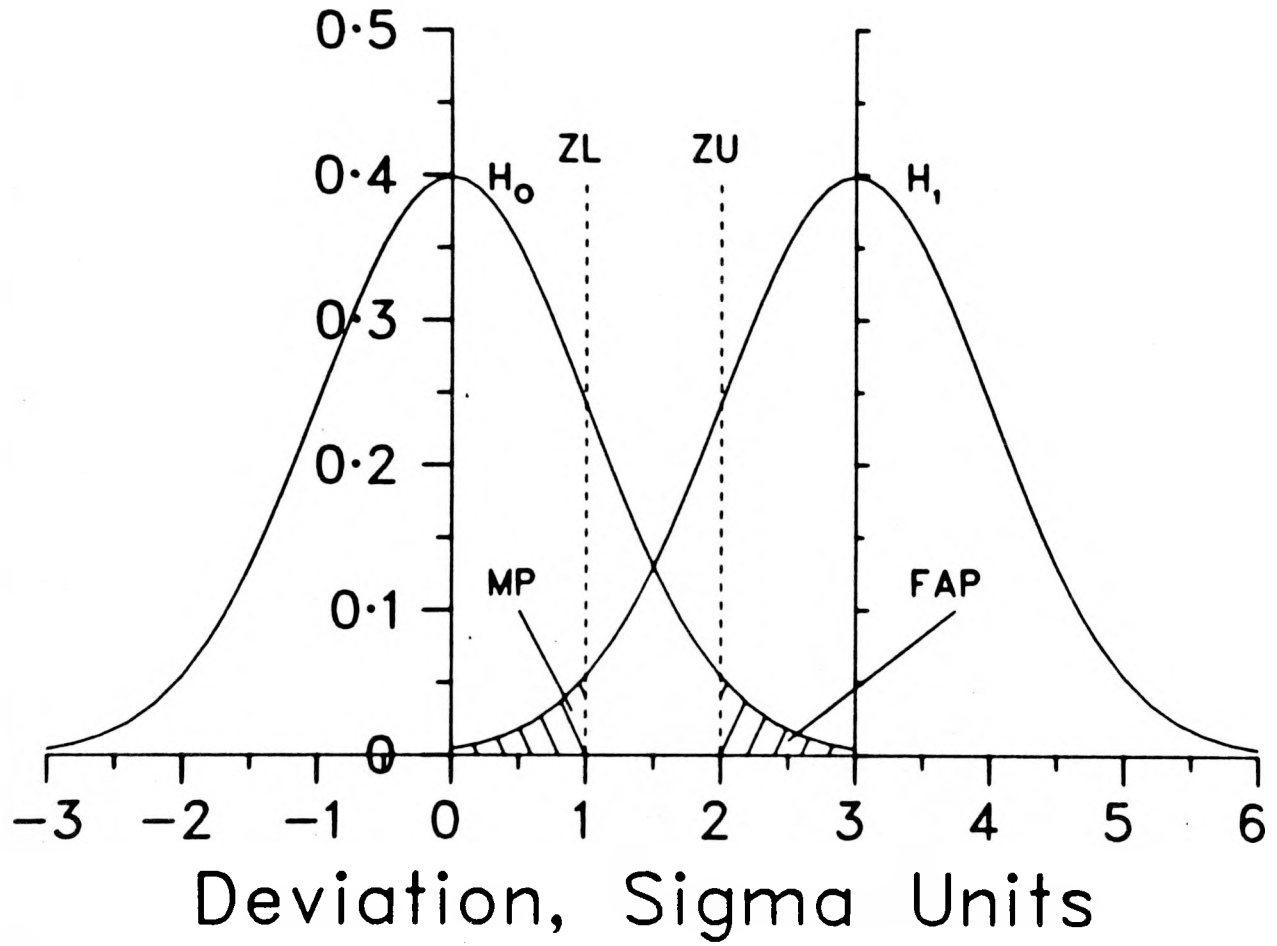


Fig. 6.
Probability density functions representing no missing nuclear material and missing nuclear material.

1. Shewhart. The Shewhart chart is the oldest graphical-display tool to be widely used by industry for process control. In the chart's standard form, measured data are plotted sequentially on a chart where 2σ and 3σ levels are indicated. In safeguards applications, the Shewhart chart is a sequential plot of the materials balance data with 1σ error bars. This chart is most sensitive to large, abrupt shifts in the materials balance data.

2. Cusum. The Cusum statistic is computed after each materials balance period. It is the sum of all materials balances since the beginning of the accounting interval. Cusum charts are sequentially plotted values of the Cusum statistic that are used to indicate small shifts in the materials balance data. The Cusum variance is a complex combination of the variances of individual materials balances, because these balances usually are not independent. Correlation between materials balances has two principle sources. The first is the correlation, discussed previously, between measurement results obtained by using a common instrument calibration. The magnitudes of the associated covariance terms depend on the magnitude of the calibration error and the frequency of each instrument recalibration; omission of these terms can cause gross underestimation of the Cusum variance. The second source of correlation between materials balances is the occurrence, with opposite signs, of each measured value of in-process inventory in two adjacent materials balances. As a result, only the first and last measurements of in-process inventory appear in the Cusum, and only the corresponding variances appear in the Cusum variance.

3. Uniform Diversion Test. The Kalman filter is applied widely to communications and control systems for signal processing in stochastic environments. It is a powerful tool for extracting

weak signals embedded in noise. It has been applied recently to safeguards, because dynamic materials accounting systems rapidly generate large quantities of data that may contain weak signals caused by repeated, small diversions embedded in the noise produced by measurement errors.

The uniform diversion test (UDT) is designed to detect a small, constant diversion during each materials balance period. Minimum-variance, unbiased estimates of the average diversion and the inventory at each time are obtained using the Kalman filter.

The Cusum and the UDT are complementary in several respects. The Cusum estimates the total amount of missing nuclear material at each time step, and its standard deviation is the 1σ error in the estimate of the total. The UDT, on the other hand, estimates the average amount of nuclear material missing from each materials balance, and its standard deviation estimate is taken as the 1σ error in the estimate of the average. Thus, both the Cusum and the UDT search for a persistent, positive shift of the materials balance data--the Cusum by estimating the total, the UDT by estimating the average.

C. Data Analysis Graphic Aids

1. Alarm Charts. The decision tests examine all possible sequences of the available materials balance data because, in practice, the time at which a sequence of diversions begins is never known beforehand. Furthermore, to ensure uniform application and interpretation, each test is performed at several levels of significance (false-alarm probability). Thus, it is useful to have a graphic display that indicates those alarm-causing sequences, specifying each by its length, time of occurrence, and significance. One such tool is the alarm-sequence chart, which has proven useful in summarizing the results of the various tests and in identifying trends of the materials accounting data.

To generate the alarm-sequence chart, each sequence that causes an alarm is assigned a descriptor that classifies the alarm according to its significance (false-alarm probability), and a pair of integers (r_1,r_2) that are, respectively, the indexes of the initial and final materials balances in the alarm sequence. The alarm-sequence chart is a point plot of r_1 vs r_2 for each sequence that caused an alarm, with the significance range of each point indicated by the plotting symbol. One possible correspondence of plotting symbol to significance is given in Table II. The symbol T denotes sequences of such low significance that it would be fruitless to examine extensions of those sequences; the position of the symbol T on the chart indicates the termination point.

For example, consider a sequence of materials balance data beginning at balance number 12, and suppose that one of the tests gives an alarm with a false-alarm probability of 2×10^{-4} at

TABLE II

ALARM CLASSIFICATION FOR THE ALARM-SEQUENCE CHART

Classification (Plotting Symbol)	False-Alarm Probability
A	10^{-2} to 5×10^{-3}
B	5×10^{-3} to 10^{-3}
C	10^{-3} to 5×10^{-4}
D	5×10^{-4} to 10^{-4}
E	10^{-4} to 10^{-5}
F	$<10^{-5}$
T	~ 0.5

balance number 19. Then on the alarm-sequence chart for that test, the letter D would appear at the point (12,19). This procedure continues for all possible sequences of the available materials balances. It is always true that $r_1 \leq r_2$, so that all symbols lie to the right of the line $r_1 = r_2$ through the origin. Persistent data trends (repeated diversions) cause long alarm sequences ($r_1 << r_2$), and the associated symbols on the alarm chart extend far to the right of the line $r_1 = r_2$.

2. Examples. Simulated results of diversion detection for 1 week of process operation are given in Figs. 7-9. Each figure shows results obtained with one of the decision analysis tests described above, the Shewhart, Cusum, and UDT. Each figure shows plots of the test statistic and the corresponding alarm chart for the case of no diversion (upper) and for the case of diversion (lower). In each case a strategy of low-level uniform diversion is simulated during the 51-125th materials balances. The diversion occurs during the third, fourth, and fifth days of the week. Note that significant alarms are given by the Cusum and UDT during the fourth day (the second day in the diversion scenario).

D. Systems Performance Analysis

Essential to the design of nuclear materials accounting systems is an analysis of their expected performance in detecting losses of nuclear material. Systems performance analysis, in turn, implies the definition of suitable performance measures that can be easily related to externally established criteria. Thus, there are two aspects of the performance analysis problem: first, defining performance measures, and second, relating those measures to established, quantitative performance criteria.

Performance measures for any nuclear materials accounting system embody the concepts of loss-detection sensitivity and loss-detection time. Because of the statistical nature of materials

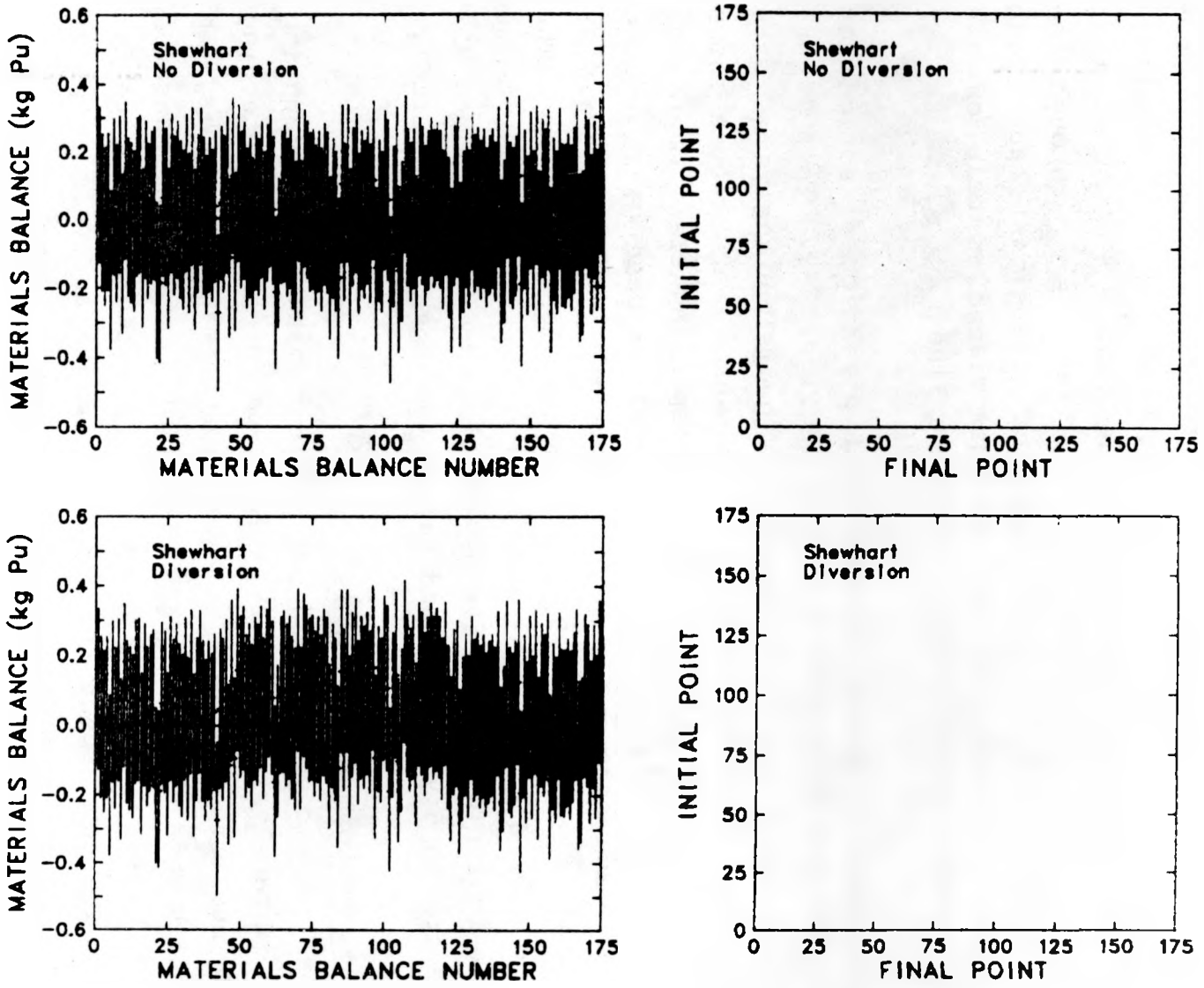


Fig. 7.
Shewhart and alarm charts.

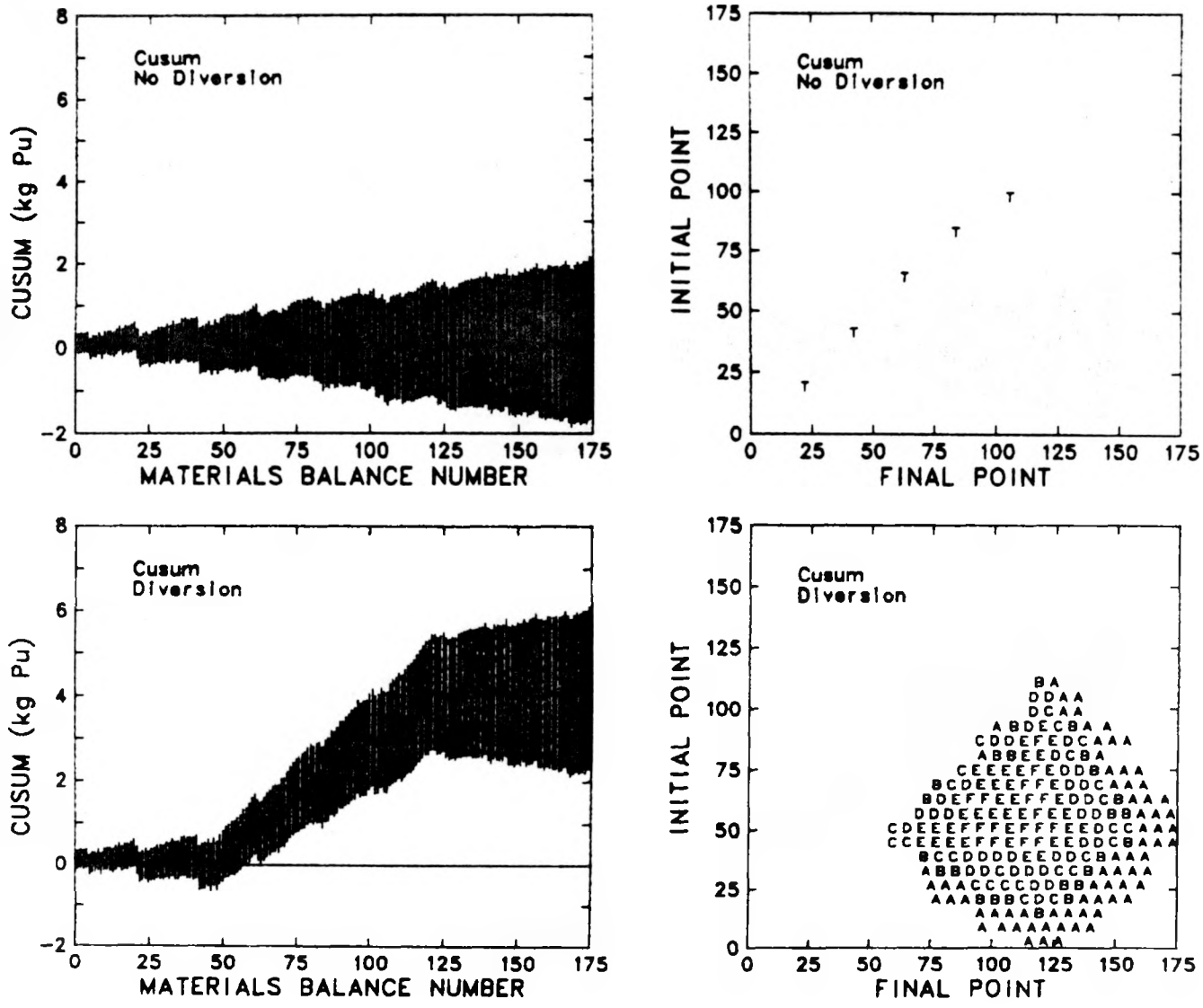


Fig. 8.
Cusum and alarm charts.

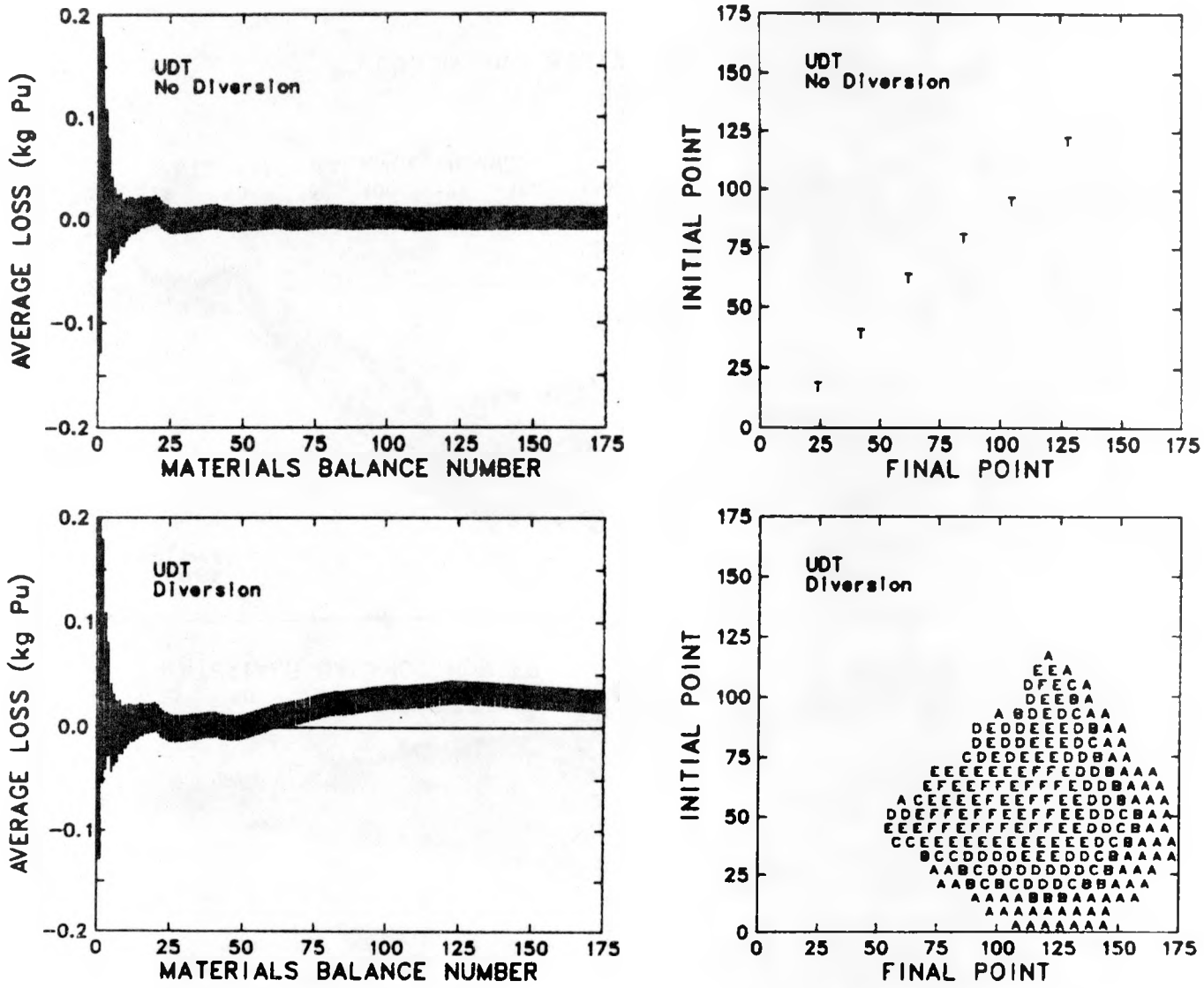


Fig. 9.
UDT and alarm charts.

accounting, loss-detection sensitivity can be described in terms of the probability of detecting some amount of loss while accepting some probability of a false alarm. Loss-detection time is the time required by the accounting system to reach some specified level of loss-detection sensitivity. Note that the loss scenario is not specified; that is, whether the loss is abrupt or protracted, the total loss is the measure of performance. Note also that loss-detection time refers only to the internal response time of the accounting system.

1. Performance Surfaces. Intuitively, the performance of any accounting system is describable by some function

$$P[L,N,\alpha] \quad ,$$

where P is the accounting system's probability of loss detection, L is the total loss over a period of N balances, and α is the false-alarm probability. Thus, a convenient way of displaying system performance is a three-dimensional graph of the surface P vs L and N for some specified value of α. We call such graphic displays performance surfaces. They are plotted in the three-dimensional space (N, L, P) illustrated in Fig. 10. They portray (correctly) the expected performance of an accounting system as a function of the three performance measures, loss, time, and detection probability, rather than as a single point.

2. Cusum Performance Surfaces. Because systems performance may depend on the details of a particular diversion strategy as well as on details of the accounting system, the overall performance is difficult to quantify. Fortunately, however, the Cusum statistic does not depend on how the material was lost, but responds only to the total loss L during any time interval N.

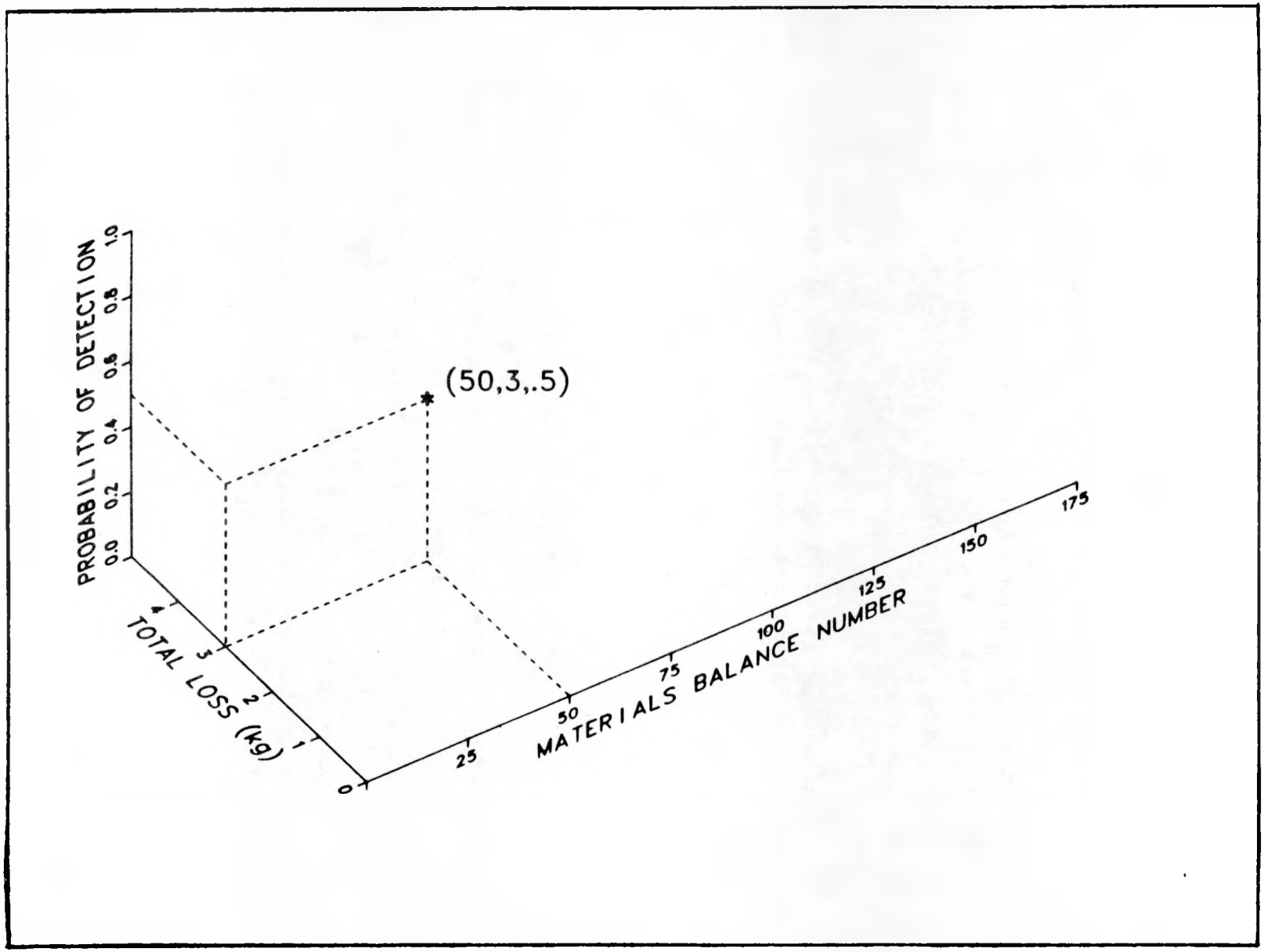


Fig. 10.
Three-dimensional space of performance surfaces.

Moreover, the Cusum test detects any loss relatively well, even though it is seldom the best test for any particular scenario.

If the Cusum test is always among the tests applied to the accounting data, the performance of the accounting system will always be at least as good as the loss-detection power of the Cusum test. Thus, the Cusum test provides a conservative, scenario-independent measure of systems performance.

Performance surfaces generated using the Cusum test (only) are referred to as Cusum performance surfaces because they are approximations to the expected performance of the system. The performance of more powerful tests for specific loss scenarios, such as the UDT, should be compared with the Cusum test performance to ensure that the Cusum approximation is not unduly pessimistic.

3. Examples. Figure 11 shows two examples of Cusum performance surfaces produced using a commercially available computer graphics program (DISSPLA) that plots isometric contours of total loss L and materials balance number N. Note that contours of fixed loss-detection probability are also plotted on the Cusum performance surfaces in probability increments of 0.1.

Figure 11 illustrates the use of Cusum performance surfaces in accounting systems design and analysis. The expected performance of "worst-case" and "best-case" accounting systems are shown. The improvement in sensitivity primarily obtained by periodically recalibrating feed and product measuring devices is obvious by comparing the figures.

VII. DISCUSSION

The materials accounting systems discussed above enhance materials control and accounting by providing better information on the locations and amounts of nuclear material than is currently

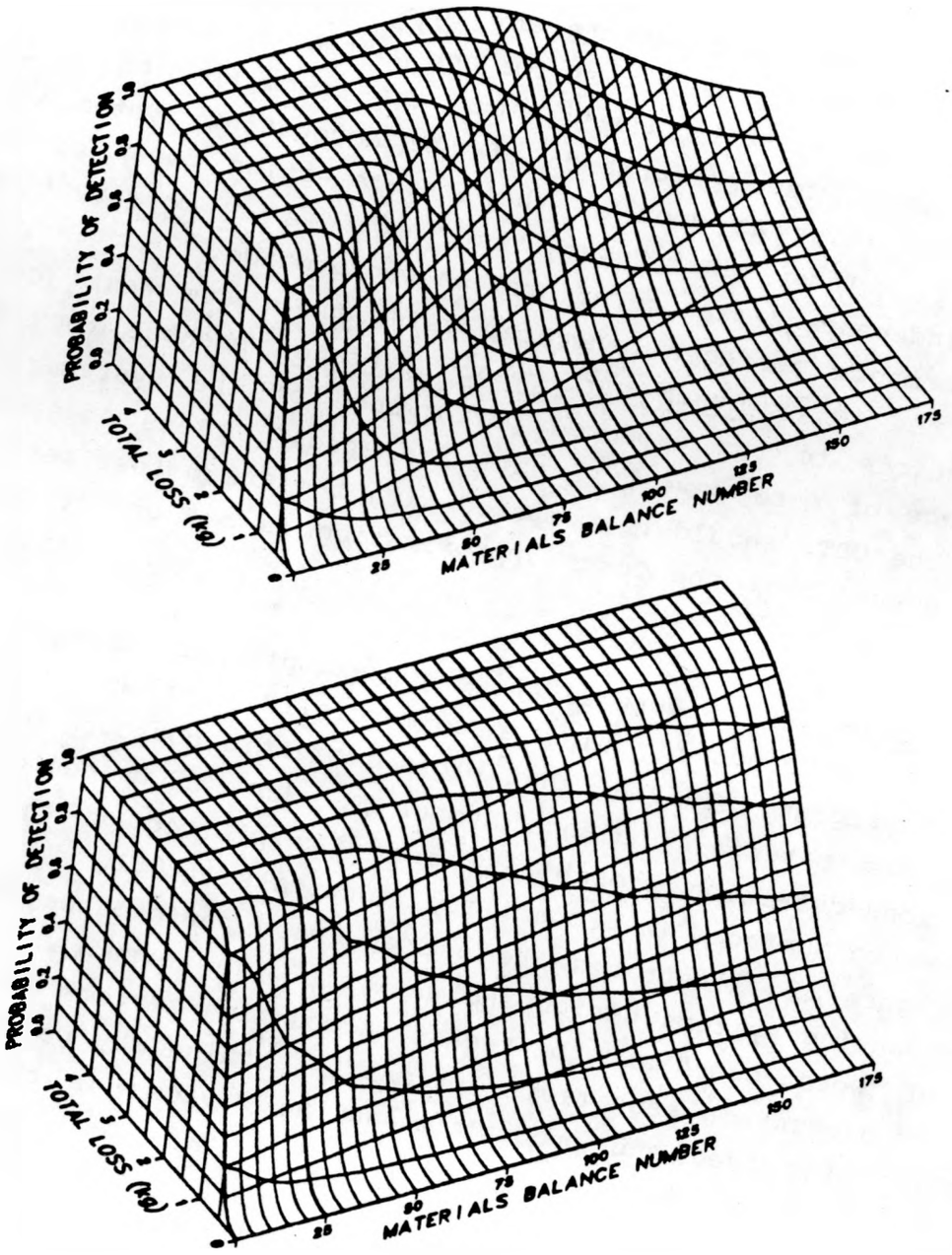


Fig. 11.
Cusum performance surfaces for two accounting cases; worst (upper), best (lower).

available by conventional methods. Advanced accounting systems must be integrated into the process and therefore should be incorporated early in the design of fuel-cycle facilities.

Dynamic accounting systems have many features in common with advanced process control systems. Improved measurements and automated data handling techniques benefit both systems. Such systems must be tailored for each process, and instrumentation must be evaluated in terms of sensitivity, reliability, and operational acceptability.

Particular process design features can have important materials accounting consequences that should be considered during process design. Based on experience, it should be expected that design alternatives can be identified that are beneficial to safeguards and benevolent to the process.

BIBLIOGRAPHY

"The Structure and Content of Agreements Between the Agency and States Required in Connection with the Treaty on the Non-Proliferation of Nuclear Weapons," International Atomic Energy Agency document INFCIRC/153 (corrected) (June 1972).

IAEA Safeguards Technical Manual, Part A, "Safeguards Objectives and Criteria Requirements," International Atomic Energy Agency technical document IAEA-174 (Vienna, 1976).

IAEA Safeguards Technical Manual, Part F, "Statistical Concepts and Techniques," International Atomic Energy Agency technical document IAEA-174 (Vienna, 1977).

R. H. Augustson, "Dynamic Materials Control Development and Demonstration Program," Nucl. Mater. Manage. VII(3), 305-318 (1978).

D. D. Cobb and J. P. Shipley, "Performance Analysis of Nuclear Materials Accounting Systems," Nucl. Mater. Manage. VIII(2), 81-92 (Summer 1979).

H. A. Dayem, D. D. Cobb, R. J. Dietz, E. A. Hakkila, J. P. Shipley, and D. B. Smith, "Dynamic Materials Accounting in the Back End of the LWR Fuel Cycle," Nucl. Technol. 43, 222-243 (1979).

J. L. Jaech, "Statistical Methods in Nuclear Material Control," TID-26298, Technical Information Center, Oak Ridge, Tennessee (1973).

G. R. Keepin and W. J. Maraman, "Nondestructive Assay Technology and In-Plant Dynamic Materials Control--DYMAC," in Safeguarding Nuclear Materials, Proc. Symp., Vienna, 1975 (International Atomic Energy Agency, Vienna, 1976), IAEA-SM-201/32, pp. 305-320.

T. I. McSweeney, J. W. Johnston, R. A. Schneider, and D. P. Granquist, "Improved Material Accounting for Plutonium Processing Facilities and a 235-U-HTGR Fuel Fabrication Facility," Battelle-Pacific Northwest Laboratories report BNWL-2098 (October 1975).

T. D. Reilly and M. L. Evans, "Measurement Reliability for Nuclear Material Assay," Los Alamos Scientific Laboratory report LA-6574 (January 1977).

J. P. Shipley, "Decision Analysis and Nuclear Safeguards," in Nuclear Safeguards Analysis - Nondestructive and Analytical Chemical Techniques, E. A. Hakkila, Ed. (Am. Chem. Soc., Washington, DC, 1978).

J. P. Shipley, "Efficient Analysis of Materials Accounting Data," Nucl. Mater. Manage. VII(3), 355-366 (1978).

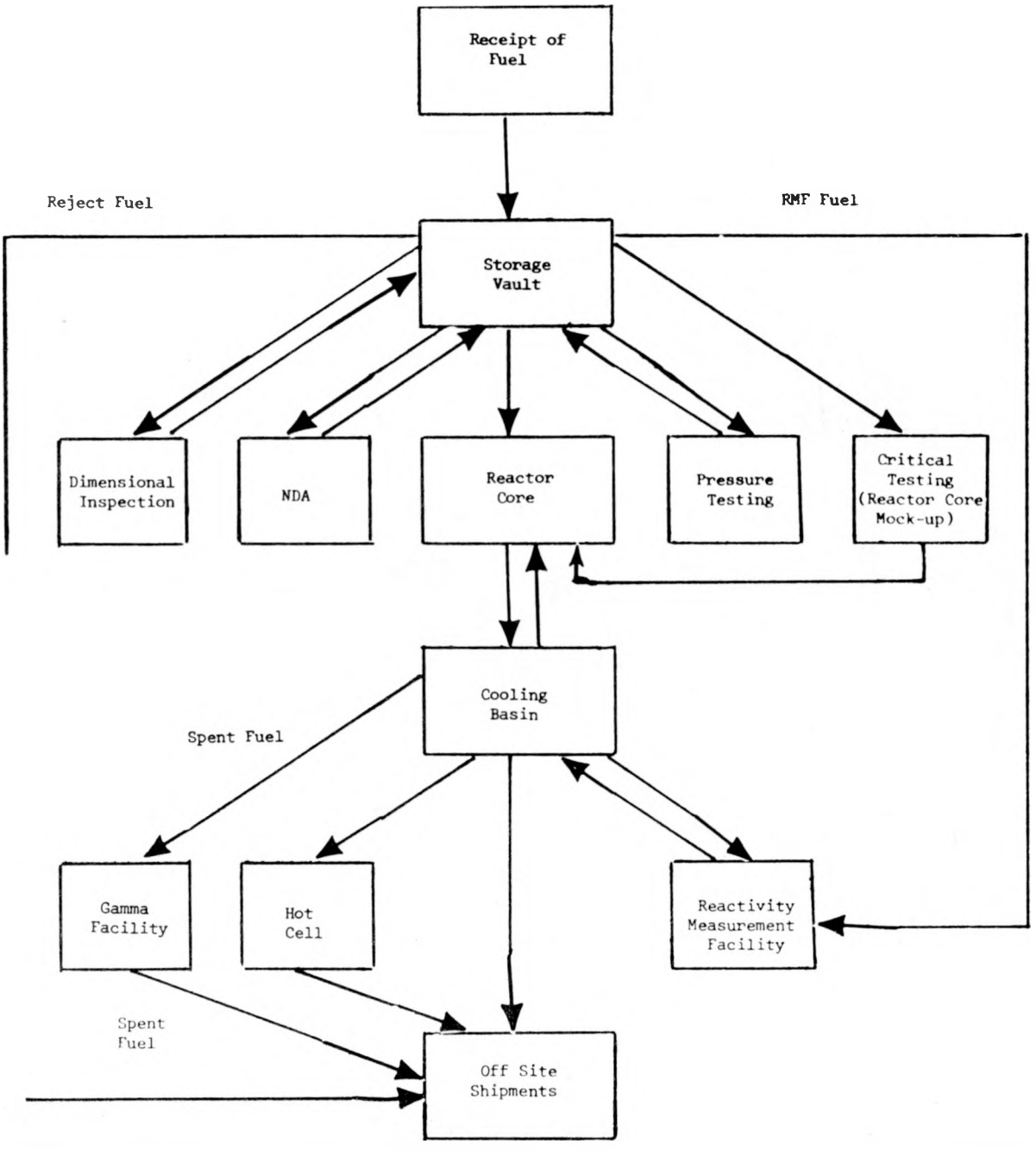


FIGURE 2

Fuel Element Flow in a Research Reactor Facility

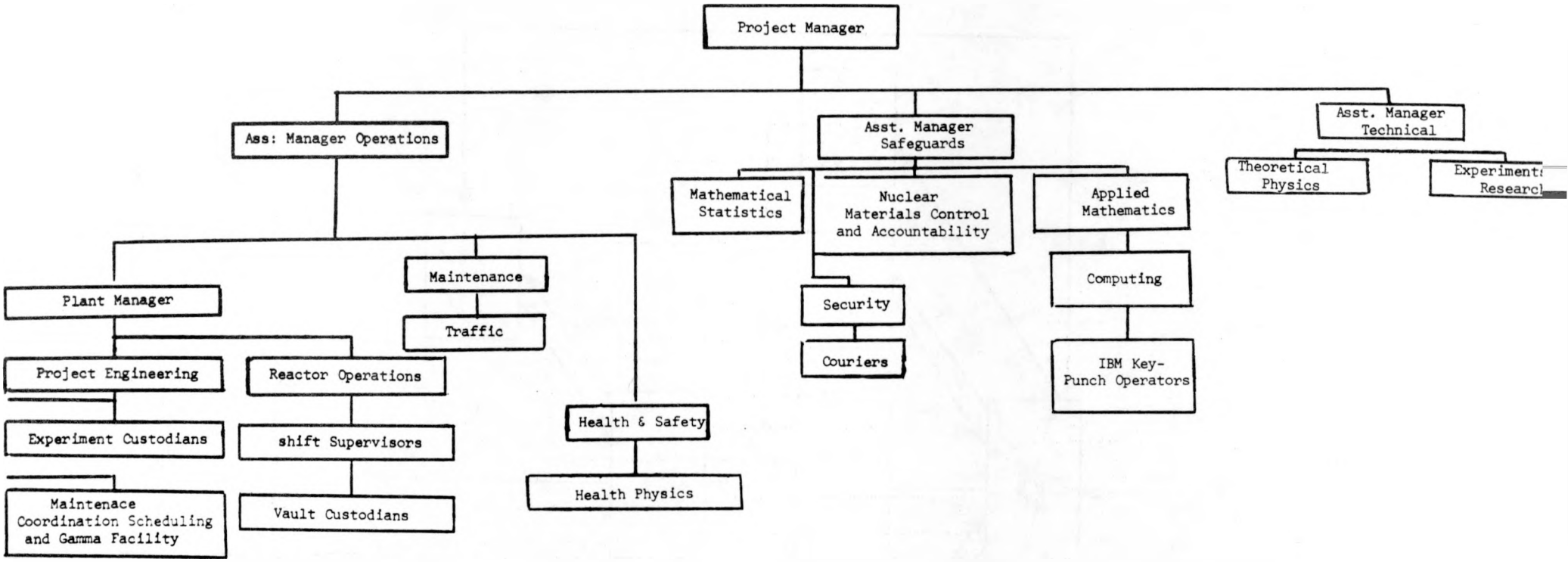


FIGURE 1

Proposed Organizational Chart For Effecting Nuclear Materials Control

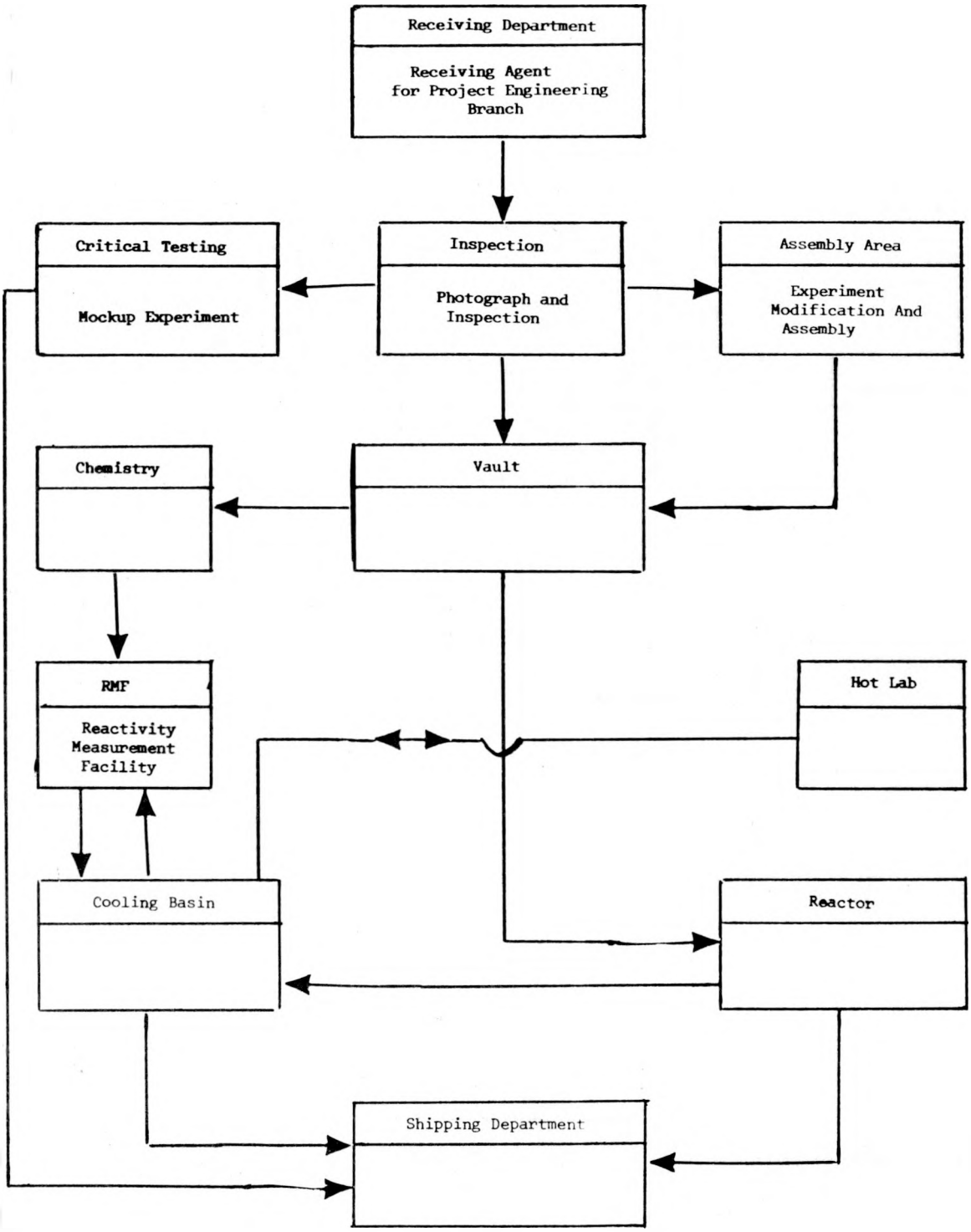
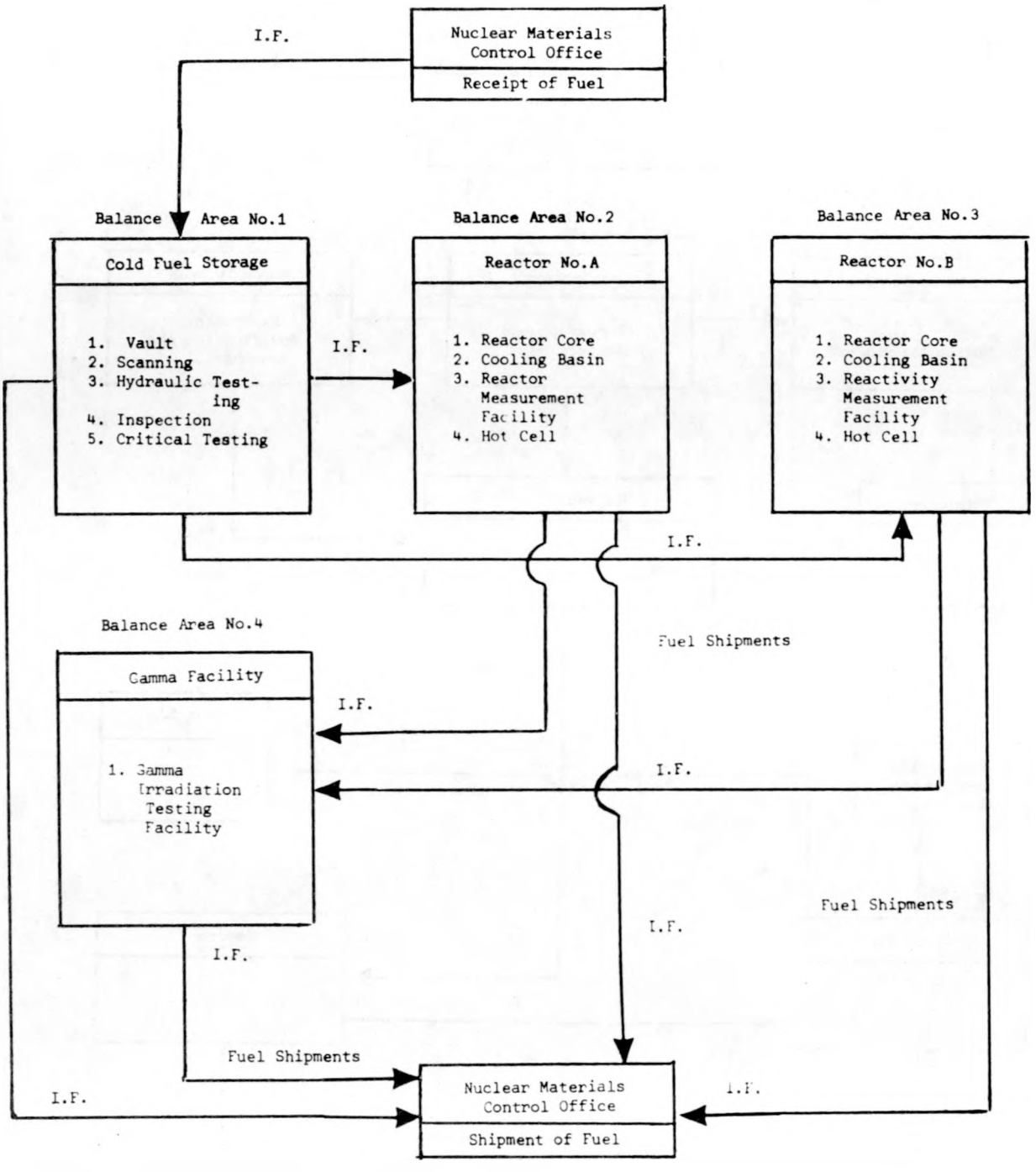


FIGURE 3

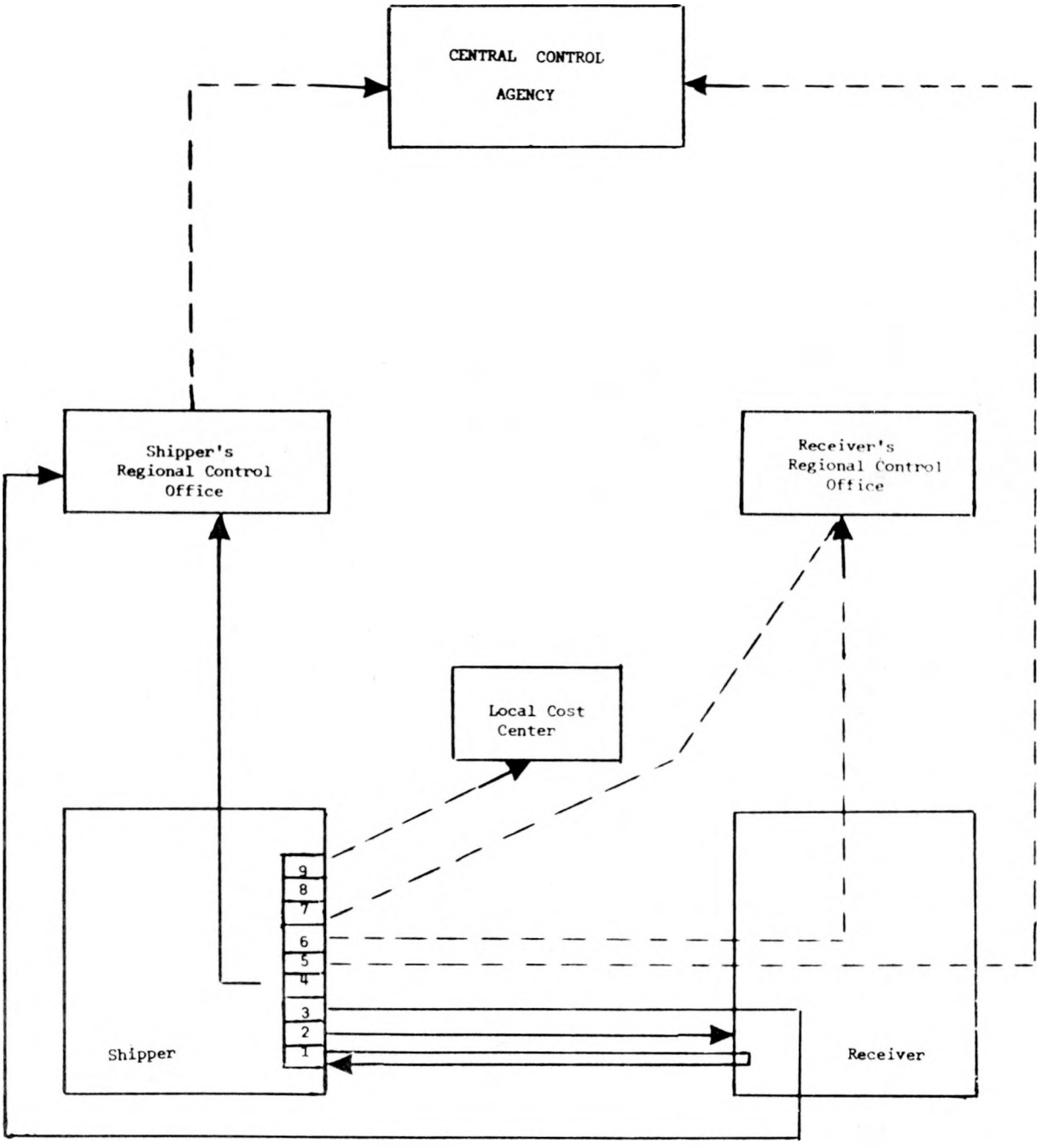
Experimental Item Flow in a Reactor Facility



I.F. = Internal Transfer Form

FIGURE 4

Nuclear Material Control Balance Areas



Dotted lines denote additional distribution when transfers are between two different Regional Offices.

FIGURE 5

Distribution of External Transfer Forms

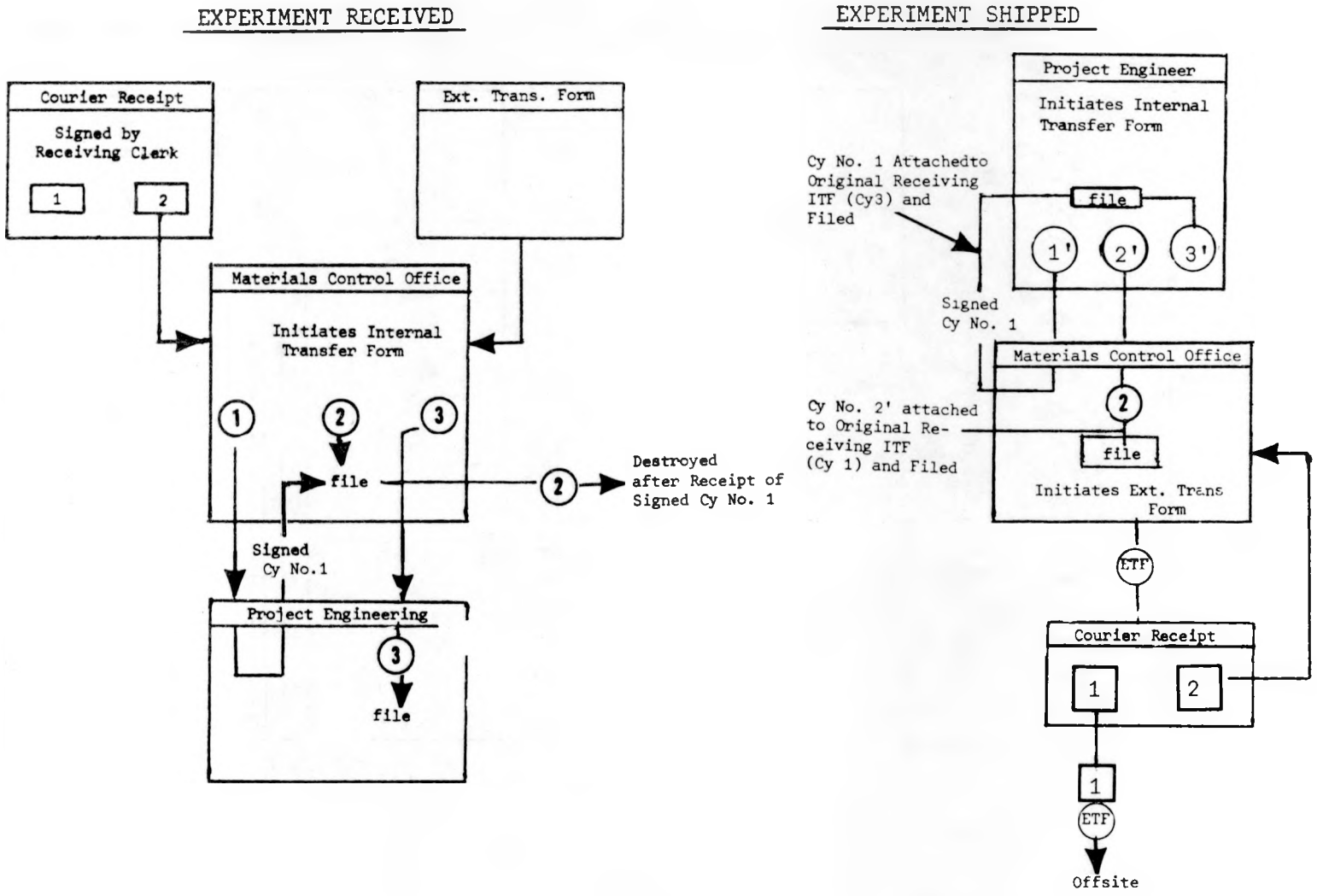


FIGURE 6

Internal Transfer Form Issuance and Disposition for Experimental Item Accounting

TRANSFER NUMBER

NUCLEAR MATERIAL TRANSFER (INTERNAL)

INTERNAL TRANSFER JOURNAL	FROM	TO

DATE MATERIALS SHIPPED _____ DATE MATERIAL RECEIVED _____ TRANSFER DOCUMENT

MATERIAL TYPE	DESCRIPTION LOG, ITEM, EXPERIMENT NUMBERS	PHYSICAL NET WT.	ELEMENT WT.	ISOTOPE WT.

ACCOUNTABLE MATERIAL TYPES: ENRICHED URANIUM, PLUTONIUM, DEPLETED URANIUM, NORMAL URANIUM U-233, PU-242, AM-241, AM-243, CURIUM, BERKELIUM, CALIFORNIUM, LITHIUM, NEPTUNIUM, PU-238 HELIUM-3, TRITIUM, THORIUM.

SIGNATURE OF SENDER _____ SIGNATURE OF RECEIVER _____

FOR NM REP USE ONLY

DATE	FROM	TO	NUMBER	MT	ELEMENT	ISOTOPE
1 2 3 4 5 6	7 8 9 10	11 12 13 14	15 16 17 18 19	20 21 22 23	24 25 26 27 28 29 30 31 32	33 34 35 36 37 38 39 40 41

WHITE - Receiver sign and forward to Nuclear Material Representive PINK - Receiver sign and forward to sender YELLOW - Receiver's file copy
GREEN - Senders suspense copy BLUE - Sender forward to Nuclear Material Control

NUCLEAR MATERIAL TRANSACTION REPORT

(EXTERNAL)

1. SHIPPER'S Ident.	2. RECEIVER'S Ident.	3. TRANSACTION NO. (9-14)	4. CORRECTION NO. (15)	5. PROCESSING CODE (16) SHIPPER	5. PROCESSING CODE (16) RECEIVER	6. RESERVED (17)	7. ACTION CODE (18) SHIPPER	7. ACTION CODE (18) RECEIVER	8. DATA CODE (19)	DOCUMENTATION (Only if document is classified SECRET) PAGE	OF	PAGES	COPY	OF	COPIES SERIES

9. A. NAME AND ADDRESS OF SHIPPER	B. LICENSE NO	10.A. NAME AND ADDRESS OF RECEIVER	B. LICENSE NO.	11. NO OF DATA LINES (20-21)	12. NATURE OF TRANSACTION (22)
C. ATTENTION D. TELEPHONE		C. ATTENTION D. TELEPHONE		13. A. SHIPPED FOR ACCOUNT OF	B. Ident. (23-26)
				14. A. SHIPPED TO ACCOUNT OF	B. Ident. (27-30)

15. TRANSFER AUTHORITY CONTRACT, NM DRAFT, OR ORDER NUMBER (34-60)	16. EXPORT OR IMPORT TRANSFERS A. LICENSE NO (32-33)
	B. PORT EXIT/ENTRY (32-35)
17. MATERIAL TYPE AND DESCRIPTION	18. TRANSPORTATION PROFILE
21. MISCELLANEOUS	

No	DISTRIBUTION OF COPIES
1	
2	
3	
4	
5	
6	
7	
8	
9	

No	B. CARRIER IDENTIFICATION	C. TRANSFER POINT	A. MODEL ID	B. NUMBER
1	(36-39)	(40-44)	(22-25)	(26-39)
2	(45-48)	(49-53)	(29-32)	(33-39)
3	(54-57)	(58-62)	(36-39)	(40-42)
4	(63-66)	(67-71)	(43-46)	(47-49)
5	(72-75)		(50-53)	(54-58)

19. PACKAGE IDENTIFICATION	20. ACTION DATE	MONTH (71-72)	DAY (72-73)	YEAR (74-75)
	A. SHIPMENT			
	B. SHIPPER'S CORRECTION			
	C. RECEIPT			
	D. RECEIVER'S MEASUREMENT			
	E. RECEIVER'S CORRECTION			

22. TOTAL GROSS WEIGHT (57-66)	23. TOTAL VOLUME (Waste Transfers Only) (67-75)

24. SHIPPER'S DATA

A. SIGNATURE OF AUTHORIZED OFFICIAL AND DATE SIGNED

LINE NO (20-21) B	PROJECT NUMBER AND IDENTIFICATION (24-53) C	MATERIAL TYPE (54-55) D	COMPOSITION CODE (56-60) E	OWNER CODE (61) F	USE CODE (22-23) G	NUMBER OF ITEMS (40-43) H	COUNTRY CONTROL NUMBER (62-69) I	GROSS WEIGHT J	NET WEIGHT K	ELEMENT WEIGHT (43-53) L	WEIGHT % ISOTOPE (58-66) M	ISOTOPE WEIGHT (65-75) N	LIMITS OF ERROR ELEMENT (54-57) O	LIMITS OF ERROR ISOTOPE (76-80) P

25. RECEIVER'S DATA

A. SIGNATURE OF AUTHORIZED OFFICIAL AND DATE SIGNED

LINE NO (20-21) B	PROJECT NUMBER AND IDENTIFICATION (24-53) C	MATERIAL TYPE (54-55) D	COMPOSITION CODE (56-60) E	OWNER CODE (61) F	USE CODE (22-23) G	NUMBER OF ITEMS (40-43) H	COUNTRY CONTROL NUMBER (62-69) I	GROSS WEIGHT J	NET WEIGHT K	ELEMENT WEIGHT (43-53) L	WEIGHT % ISOTOPE (58-66) M	ISOTOPE WEIGHT (65-75) N	LIMITS OF ERROR ELEMENT (54-57) O	LIMITS OF ERROR ISOTOPE (76-80) P

26. REMARKS

EXHIBIT 2

SS MATERIAL TRANSFER RECEIPT

SS MATERIAL ON TRANSFER	FROM (Station Symbol)	TO (Station Symbol)	NO.
SHIPPING DATE		DATE RECEIVED	

Material has been verified as follows (check One):

Piece Count ☐ Container Count ☐ Gross Weight Check ☐

Other: _____

Shipper's weights and SS contents are accepted pending final verification measurements

By. _____

EXHIBIT 3

OFF-SITE RADIOACTIVE MATERIAL SHIPMENT RECORD

Log No. _____

To _____

_____

_____

From _____

_____

_____

_____

Charge No. _____ Classified ☐

Collect ☐ Prepaid ☐

Carrier(s) _____

Regular Comm. ☐ Sole Use ☐ Other ☐

Type: Air ☐ Truck ☐ Rail ☐ Water ☐

Physical Form: Solid ☐ Liquid ☐ Gas ☐ Special ☐ Normal ☐

Chemical Form: _____

Material(s) Shipped: _____

_____

Principal	Transport Group	Curies(Ci)	Quantity
			Limited Quantity (or less) ☐
			Man. Devices ☐
			Type A ☐
			Type B ☐
			Large ☐
			LSA ☐

Total Activity _____ Curies

Container Used (describe): _____

Size _____ Weight _____

Type: Dot Spec. _____ Type A ☐ Type B ☐

Certificate of Compliance No. _____

FISSILE MATERIALS

Not Applicable ☐ Pu _____

Class I ☐ U _____

Class II ☐ Transp. Index _____

Class III ☐ Transp. Controls _____

_____

_____

Nuc. Safety Approval _____

Date _____

ACCOUNTABLE NUCLEAR MATERIALS
(Any Quantity)

Not Applicable ☐

Shipment Contains Accountable Material ☐

Trans. Form No

Safeguards Rep

Date

LOADING CONTENTS INTO CONTAINER

Loading Inst. _____ Loaded: Dry ☐ Wet ☐ Container Drained ☐ Cleaned ☐

Container Insp. and Maint. Current ☐ Closures Secured ☐ Remarks _____

Loader _____ Date _____ Organization _____

Certification: Shipment Prepared in Accordance with regulations ☐ Consignee is Authorized to Receive Shipment ☐

Remarks: _____

_____

Originator: _____ Date _____ Organization _____

_____

No.(s) _____ Affixed By _____ Organization _____ Date _____

Tie-down Equipment Adequate ☐

Safety Inspector _____ Date _____

Radiation:(surface) _____ MREM/HR (3 feet) _____ MREM/HR

Contamination: (Averaged over any 300 cm^2 Package Surface)

Beta-Gamma _____ dis/min/100 cm^2

Alpha _____ dis/min/100 cm^2

Remarks: _____

_____

Package(s) Surveyed By: _____ Date

Additional Surveys: Vehicle ☐ Driver ☐ Other _____

Surveyed By: _____ Date

LABELS

None Required ☐ White I ☐

Yellow II ☐

Yellow III ☐

Transport Index _____

Empty ☐

Other(s) _____

VEHICLE PLACARD(s)

None Required ☐

Radioactive Material ☐

Other(s) _____

CERTIFICATION AND AREA RELEASE

1. This is to certify that the above-named materials are properly classified, described, packaged, marked, and labeled, and in proper condition for transportation according to the applicable regulations of the Department of Transportation ☐
2. For shipments on passenger carrying aircraft this shipment is within the limitation prescribed for passenger-carrying aircraft ☐

Area Supervisor _____ Area _____ Date _____

Traffic Agent _____ Date Released _____

NUCLEAR MATERIALS EXPENDED IN RESEARCH

SECTION I: Completed by originator.

This is to certify that the following nuclear materials (were) (are to be) expended in research during the period

_____ 19____ to _____ 19____

(Give details as to identity, composition, enrichment, quantity, etc.)

Description	Element	Isotope

The materials itemized above (were) (are to be) expended in research under the following conditions. Provide adequate justification for non-recovery.

_____

_____

_____

_____

_____

_____

_____

_____

_____ Signature _____

Date NM Custodian

SECTION II: Completed by Safeguards Section

Total Estimated Value of Material $ _____

Material Evaluated and Found (to be) (not) Economical to Recover.

Date _____ Remarks: _____

_____

_____

Facility NM Representative Approval: _____ _____

Signature Date

NM Management Approval: _____ _____

Signature Date

SECTION III: Completed by Agency

Approved: _____ _____

Signature Date

Authorization Number _____

Page 1 - Completed copy to Safeguards Section Page 2 - Completed copy to originator Page 3 - Retained by Agency Page 4 - Retained by Safeguards Section Page 5 - Retained by originato

EXHIBIT 5

Material Type_____

Month of_____19_____ Weight Units_____

Description	Station Control		Cold Fuel Storage		Hot Storage				Experiments		Gamma	
					Reactor A		Reactor B					
	Nat U	U-235	Nat U	U-235	Nat U	U-235	Nat U	U-235	Nat U	U-235	Nat U	U-235

Balance Areas

Material Balance Summary Ledger

EXHIBIT 6

INTERNATIONAL TRAINING COURSE ON NUCLEAR MATERIALS ACCOUNTABILITY FOR SAFEGUARDS PURPOSES

SESSION #27: DESIGN FEATURES RELEVANT TO IMPROVED SAFEGUARDS IMPLEMENTATION

SPEAKER: Dr. Dipak Gupta

Gesellschaft Fuer Kernforschung GmbH
Karlsruhe, Federal Republic of Germany

Wednesday, June 4, 1980
2:45 p.m.

BIOGRAPHY

Education: B.Ch.E. - University of Jadavpur; D.Sc. - University of Aachen, chemical engineering, nuclear chemical engineering, fuel cycle optimization for fast reactor systems, international safeguards.

Present Position: Head, Division of Nuclear Research R&D, Nuclear Research Center Karlsruhe.

Present Duties: Coordination and guidance of R&D activities in the field of international safeguards and physical protection since 1973.

Other Information: Member of the SAGSI, (IAEA Standing Advisory Group on Safeguards Implementation); chairman of ESARDA (European Safeguards Research and Development Association) 1979/80; member scientific council Nuclear Research Center Karlsruhe; more than 60 scientific publications.

INTERNATIONAL TRAINING COURSE ON NUCLEAR MATERIALS ACCOUNTABILITY FOR SAFEGUARDS PURPOSES

Session Objectives

SESSION #27: DESIGN FEATURES RELEVANT TO IMPROVED SAFEGUARDS IMPLEMENTATION

This session will show how the interrelationship of the State system of accountancy and control, facility design features, and safeguards measures available to the IAEA can influence the design and implementation of effective IAEA safeguards.

SESSION 27: DESIGN FEATURES RELEVANT TO IMPROVED SAFEGUARDS IMPLEMENTATION IN EXISTING NUCLEAR FACILITIES

D. Gupta
Nuclear Research Center Karlsruhe
F. R. Germany

I. INTRODUCTION

Following a series of discussions at the SAGSI (Standing Advisory Group on Safeguards Implementation) IAEA, a structure containing the various elements relevant to the design and implementation of the IAEA safeguards systems was established. This structure is reproduced in Fig. 1. It is seen that after the design goals of the Agency systems have been established, the following elements influence directly the design and the implementation of IAEA safeguards at the level of a facility:

1. Activities and elements of the State's Systems of Accountancy and Control (SSAC)
2. Design features and practices in a facility
3. Limitations and capabilities of the safeguards measures, accountancy, containment and surveillance
4. Capabilities and resources of the IAEA.

These elements are all interrelated and may influence the design and implementation of Agency safeguards in a fairly complicated manner. The present paper discusses briefly the role and influence of these elements with special emphasis on improved implementation possibilities in existing facilities.

Note: This paper forms the basis of a presentation in the frame of the DOE/IAEA sponsored International Training Course on Nuclear Material Accountability. It is an extended and modified version of the paper "Design Features Relevant to Improved IAEA Safeguards" by D. Gupta and J. Heil, IAEA-SM-231/10 presented at the IAEA Symposium on Nuclear Safeguards Technology in 1978.

In a generic sense, the contribution of these features or elements to an improved implementation of the IAEA safeguards may be expected in the following areas:

1. In limiting the uncertainty of the IAEA knowledge with regard to a diversion in the amount-location-time plane in such a manner that the Agency can achieve its design safeguards objectives
2. In keeping the safeguards burdens and efforts for the facility operators and the Agency within acceptable limits
3. To reduce delay and complications in obtaining a safeguards relevant statement or in executing a safeguards measure.

II. STATE'S SYSTEM OF ACCOUNTANCY AND CONTROL (SSAC)

The SSAC in a state is a necessary prerequisite for the proper implementation of Agency safeguards. The mere existence of the SSAC is, however, not sufficient for the Agency to attain its safeguards goals. The SSAC lays down the legal, administrative, and technical framework to enable the Agency to execute its safeguards functions in territories under the control of a state. It also ensures that organizational and functional responsibilities and the required safeguards infrastructure are defined or laid down (and maintained) at the level of a nuclear facility under safeguards in such a way that the Agency can carry out its activities in order to achieve its safeguards goals. It is also the responsibility of a SSAC to provide or make available all the relevant information and data so that the Agency can verify them with a view to ascertain that there has been no diversion of nuclear materials from peaceful uses to nuclear weapons or other nuclear explosive devices.

Some of the technical elements of the SSAC relevant to Agency safeguards have been indicated in (1). The Agency has worked out detailed guidelines for establishing and maintaining a State's System of Accountancy and Control (2). In

the frame of the present training course, the different elements of the SSAC will be discussed in detail. Therefore, no elaborate analysis will be presented here. However, for the sake of completeness, the major elements of an SSAC as discussed in (2) are presented in Table I.

The extent and detail that the different elements of a SSAC in a state should have, will depend on the extent and type of nuclear activities in that state. A complete absence of the different elements of a SSAC in a state might force the Agency to make increasing use of subjective judgement for meeting its goal. Partial absence might severely hamper its activities. The main influence of the absence or an incompleteness of SSAC on the design and implementation of the Agency safeguards would be to provide for intensive and additional Agency activities including additional inspections and to make use of subjective judgement factors in achieving its goals.

Some examples in the framework of an SSAC that might contribute to an improved implementation of the IAEA safeguards are indicated below.

1. Inclusion of IAEA safeguards requirements in the licensing procedures for a nuclear facility in a state: The facility operators will then be required to foresee such requirements during the design phase of a facility. Inclusion of such requirements in the area of health physics, safety, and physical protection, is at present normal practice for the licensing procedures of a facility.
2. Provision of standardized methods for MUF evaluation: This can enable the Agency to utilize the results obtained by the SSAC and thereby reduce its own efforts required in making such evaluations.
3. Simplification of transport regulations for export of analytical samples containing uranium and plutonium: This can simplify and reduce delay in obtaining results of the accountancy data for Agency verification purposes.

III. DESIGN FEATURES AND PRACTICES IN A FACILITY

Both the design and the implementation of Agency safeguards are most profoundly influenced by the facility characteristics. They set practical limits to the safeguards performance and determine the level of uncertainty and extent of the Agency knowledge in the time-location-amount plane in connection with safeguards (3). The extent of knowledge required by the facility operators for material management practices may not always be sufficient for achieving safeguards goals by the Agency. In any nuclear facility, the layout, the process and operational conditions determining the flow and inventory characteristics, and the information system used for nuclear material management are among the more important facility features influencing Agency safeguards. The possibility of an adaptation of the safeguards measures to these features, and vice versa, might be the most important single factor influencing the design and implementation of Agency safeguards.

Most of the existing facilities that have come under Agency safeguards were not designed with international safeguards in view. Therefore, safeguards measures are to be adapted to the relevant features and practices in these facilities. In some cases the existing practices can also be modified to the requirements of the Agency safeguards. To illustrate this particular point, mention may be made of a number of such features and practices in existing facilities of a given type, in which adaptation of the safeguards measures or modification and/or adaptation of the existing practices to the Agency requirements could lead to an improvement or simplification in the implementation of the Agency safeguards.

A. LWR Systems (4)

1. Organization of movements of irradiated fuel elements in the fuel-storage bay in such a way that significant movements of the fuel elements can be identified positively against a background of unimportant movements, by, for example, a camera system

2. Simple sealing systems for clear and unique identification of fuel elements or containers with fuel elements
3. Arrangements of the irradiated fuel elements in clearly identifiable areas in the fuel-storage bay so that they remain accessible for verification (either individually or in groups in containers)

B. Critical Assemblies (6,7)

Some implementation problems were recognized by the Agency in connection with the safeguarding of large fast critical assemblies with plutonium or high-enriched uranium (4). The more important difficulties expected were:

1. The frequent movement of the fuel items between the storage, intermediate fuel loading, and reactor core areas and vice versa, and
2. The necessity of frequent inventory verification inside the core.

The safeguards system designed (and now being implemented) for a fixed core SNEAK-type critical assembly (6,7), is an eloquent example of how mutual adaptation of safeguards and plant practices can vastly improve safeguards implementation.

The problems are basically solved by providing continuous presence of inspection and specially designed sealing systems for the bird cages, fuel elements, and the reactor core. The physical inventory verification can be carried out virtually at any time by checking the appropriate sealing systems.

C. LEU Fabrication Facility

The uncertainties associated with material balances in existing LEU fabrication facilities appear to be quite adequate for the purposes of safeguards. Besides, the importance of this material from the point of view of international safeguards is relatively low. However, solution of two generic types of problems could cause an improvement in the safeguards implementation in such facilities.

1. Process inventory of LEU material

   Possible solutions could be achieved, for example, by keeping the process inventory materials in identifiable (and where possible) sealable containers when not in process use; by keeping an updated record system for the book inventory, and by taking the physical inventory (once a year) at a time when the process inventory of LEU in unmeasurable parts of the process is a practical minimum.

   In most of the existing facilities such adaptations have already taken place.

2. Verification of nuclear materials in finished unirradiated fuel elements:

   A possible solution could be obtained by ensuring through proper containment and surveillance measures (or through measurement), that no substitution of nuclear materials can take place in the fuel elements. Some development work may be required to achieve this.

D. Reprocessing Facilities (4,5)

The present day reprocessing facilities under Agency safeguards are relatively small (~200 t U/yr throughput). The uncertainties in a yearly material balance as well as the unmeasured process inventories in such facilities range around 5-8 kg of plutonium. Safeguards problems, if at all, may arise out of the fact that some of the design features and practices relating to the nuclear materials may increase the uncertainty in knowledge of the Agency on the amount and location of these materials or cause complications and delay in the execution of safeguards measures. Typical examples may be (4,5):

1. Accountability tank area:

   It may not be possible to ensure that

   a. all the chopped fuel elements have entered the dissolver tank.

b. the dissolver solution does not bypass the accountability tank.

c. samples taken are representative of a tank solution.

d. some valves or T-lines, which could be used for a possible diversion, are not used for such a purpose since no seals can be applied or since they are not accessible.

Such problems can be normally eliminated (and are eliminated in practice) by having Agency inspectors present during a dissolution period and using different types of corroborating and correlating information (e.g., level indicator correlation between the dissolver and the accountability tank indicators, tracer techniques for calibration of accountability tank, and ensuring representativeness of samples etc.).

2. Product storage area:

Some of the storage vessels and pipelines may not be accessible to the IAEA inspectors for verification. This problem is also solved or eliminated in practice by ensuring the continuous presence of IAEA inspectors and by the use of extensive C/S measures.

3. Preparation and transport of analytical samples:

Transport of such samples to Agency Headquarters according to (4,5) has always caused some difficulties and are considered to be one of the more difficult problems to solve in connection with the proper implementation of IAEA safeguards. Sufficient manpower and laboratory facilities may not be available at site and the internal regulations in a country may cause delay in the transport of such samples.

Two possible solutions have been considered.

a. provision for creation of adequate laboratory facilities for the purpose of preparation and analysis of Agency samples at site, or

b. reducing the sample size for transport below the free limit. R and D activities have been initiated by the Agency in the area of the reduction of sample size.

IV. CAPABILITIES AND LIMITATIONS OF SAFEGUARDS MEASURES

A. Design Features influencing the Capabilities and Limitations

The Agency can use material accountancy, containment, and surveillance measures to fulfill its goals. The extent and relation amongst these measures are laid down in the respective facility attachments for a nuclear facility. One of the most common features particularly for bulk facilities is the fact that in some parts of such a facility, the nuclear materials in question may not be accessible or available in readily measurable form for Agency verification purposes (4), so that Agency knowledge on the location and amount for this part of the material might be zero or associated with a very high degree of uncertainty. The facility operator might have additional process information or estimates at his disposal (to which the Agency might not have access) and/or may not require such materials to be available in accessible or measurable form for the purposes of plant operation. The Agency safeguards measures have to be adapted under such conditions to the facility features in such a manner that it can still meet its design goals. This may be possible by making use of a number of "operation indicators" or correlations (8) and combining a number of safeguards measures (9). A Typical example of such a case is the nuclear material content in the piping network of a process area between two successive tanks in a reprocessing facility. Although the content of the

pipeline cannot be directly verified by the Agency, it can ensure and verify that this amount can be discharged only into the second process tank. This assurance can be obtained by a combination and correlation of the amounts and flow rates for the two process tanks and the use of sealing and surveillance measures around the piping network in question. Although with such a combination of safeguards measures, the uncertainty in knowledge associated with the amount of material in the pipeline network still remains, the uncertainty in knowledge for the Agency with regard to the use of this particular amount is eliminated.

Another design feature or practice in the bulk facility in general is the high-measurement uncertainty of waste streams containing nuclear materials. If the rest of the process streams and inventory amounts are verified with the desired degree of high accuracy, the Agency inspectors can verify the measurement uncertainty of the waste streams and ensure through observation or other surveillance along with sealing measures that the discarded waste streams would no longer be accessible to plant operation without the knowledge of the Agency inspectors. Again with such a combination of measures, the Agency can fulfill its goals of ensuring that nuclear material from this particular stream has not been diverted for nuclear weapon or other nuclear explosive devices, even though the uncertainty in knowledge with regard to the exact amount in these streams may be fairly high and still exists.

B. Capabilities and Limitations of Measuring Instruments and C/S Measures for Present Day Safeguards Implementation

Some of the measurement instruments and containment and surveillance systems, which are under routine use at present for IAEA safeguards, are presented in Table 2. The fact that a

relatively small number of such instruments and systems are in use indicates that these techniques have to fulfill some stringent conditions before being accepted as worthy of routine use by the Agency (4). Such instruments should remain reliable under extreme operating conditions in a facility. Nonspecialists must be in a position to assemble and operate such instruments or systems. They should be simple, rugged, and should require little maintenance at site. The data generation and registration system associated with such instruments or system should provide reliable and verifiable data. A number of countries are cooperating at present with the IAEA in providing such instruments or C/S systems for improving the implementation capability of the Agency safeguards.

V. CAPABILITIES AND RESOURCES OF THE IAEA

According to (4), the safeguards operations of the Agency during the period of 1971 to 1977 have increased at an extremely high rate. To illustrate this point, some of the relevant figures are reproduced in Table 3 from (4). The number of bulk facilities that have come under IAEA safeguards has increased from 10 to 44 during this period. Corresponding figures for power reactors are 0-100. The amount of the safeguarded plutonium has increased from 1.7 tons to 18 tons. Although such a high rate may not be maintained constantly in the future, such a high increase in safeguards demand during a fairly short period of six years can itself bring some problems associated with the implementation of the Agency safeguards. Some of the more important problems may be:

1. Availability of an adequate number of capable and trained inspectors
2. Availability of a sufficient number of field-tested reliable measurement instruments, seals and cameras, or TV systems
3. Adequate planning, management, and data-evaluation capability at the Agency Headquarters.

The Agency has launched a fairly exhaustive program to meet the increased demands and provide adequate solutions to these problems.

ACKNOWLEDGEMENT

The author would like to thank Mr. Rundquist, IAEA, for providing the list of measuring instruments and C/S techniques in routine use by the Agency.

REFERENCES

(1) International Atomic Energy Agency, "The Structure and Content of Agreements between the Agency and States Required in Connection with the Treaty on the Non-Proliferation of Nuclear Weapons," INFCIRC/153 (corrected), IAEA (1972).

(2) "States System of Accountancy and Control of Nuclear Materials," (SSAC); AG-43/12 (Rev. 6), IAEA (1979).

(3) GUPTA, D., "Various Aspects of Systems Analysis, Safeguards Techniques," Proc. Symp. Karlsruhe, 1970, 2, IAEA, Vienna (1970).

(4) HAGINOYA, T. et al., "Experience in the Application of IAEA Inspections Practices," IAEA-SM-231/105, Proc. Nuclear Safeguards Technology, Vienna (1978).

(5) "Some of the Practical Difficulties of Safeguarding Reprocessing Plants," The Department of Safeguards, IAEA, Paper Presented at the IWG-RPS Meeting, January (1980).

(6) BRUECKNER, Chr. et al., "International Safeguards in the Large Inventory Fast Critical Assembly SNEAK," Proc. 1st Annual Symp. on Safeguards and Nuclear Material Management, ESARDA, pp. 67-71, Brussels (1979).

(7) STANNERS, W.; SCHENKEL, R., "The Implementation of Safeguards in a Fast Critical Assembly," Proc. 2nd Annual Symp. on Safeguards and Nuclear Material Management, ESARDA, pp. 42-46, Edinburgh (1980)

(8) GUPTA, D.; HEIL, J., "International Safeguards in Large-Scale Nuclear Facilties," Nuclear Power and Its Fuel Cycle (Proc. Conf. Salzburg, 1977) 7, IAEA, Vienna (1977)

(9) ROMETSCH, R.; HOUGH, G., "The Position of IAEA Safeguards Relative to Nuclear Material Control Accountancy by States," ibid pp. 441-454

Table 1: Elements of a States System of Accountancy and Control (from (2))

1. Organization and functional elements at the level of a state
   a. Authority, responsibility
   b. Laws, regulations, others
   c. Information system
   d. Requirements of NMAC
   e. Ensuring compliance
   f. Technical support
2. Organization and operation at the level of a facility
   a. Elements of a general character, e.g., categorization, MBA, flow, inventory
   b. Requirements for bulk facilities
   c. Requirements for reactors, critical assemblies, etc.

Table 2: Instruments in Routine Use by the Agency[+)]

TYPE	NAME AND DESCRIPTION	WHERE USED	REMARKS
NDA	SAM II/BSAM single channel analyzer	fuel fab. enrichment reactors	quantitative measurement of enrichment
NDA	NIS 322 (SCA)	all	qualitative determination
NDA	high purity GE detector system	Pu isotopes in spent fuel	requires experienced inspector
NDA	high-level n-coincidence counting	MOX fuel fab. critical facilities	determines Pu passively and HEU actively
Surveillance	film cameras	reactor storage ponds	
Surveillance	CCTV	reactor storage ponds	
Seals	metal type E	all	
Seals	paper seals	all	temporary sealing

[+)] Information provided by D. Rundquist, IAEA

Table 3: Safeguards Operations of the Agency during the Period 1971-1977 (Reproduced from (4))

	1971	1974	1977
No. of inspections	234	474	704
No. of bulk facilities	10	26	44
Power reactors	9	36	100
Research + training reactors	66	110	185
Nuclear materials			
Pu (t)	1.7	6.3	18
Enriched U (t) with	523	2305	7860
contained U-235 (t)	11	55	194
Source material U-nat, thorium (t)	595	3910	12230

INTERNATIONAL TRAINING COURSE ON
NUCLEAR MATERIALS ACCOUNTABILITY
FOR
SAFEGUARDS PURPOSES

SESSION #28: EXAMPLE OF AN OPERATING STATE SYSTEM
(GERMAN DEMOCRATIC REPUBLIC)

SPEAKER: Dr. W. Roehnsch

National Board for Atomic Safety and Radiation Protection
Berlin, German Democratic Republic

Wednesday, June 4, 1980
4:00 p.m.

BIOGRAPHY

Present Position: Vice President, National Board for Atomic Safety and Radiation Protection.

INTERNATIONAL TRAINING COURSE ON
NUCLEAR MATERIALS ACCOUNTABILITY
FOR
SAFEGUARDS PURPOSES

Session Objectives

SESSION #28: EXAMPLE OF AN OPERATING STATE SYSTEM
(German Democratic Republic)

The establishment and operation of an advanced state system of accountability and control is presented, including its interface with facility operators, the IAEA, and the public. Experience with national and international safeguards inspection will be described.

Session 28: EXAMPLE OF AN OPERATING STATE SYSTEM (GERMAN DEMOCRATIC REPUBLIC)

W. Roehnsch
National Board for Nuclear Safety
and Radiation Protection of the GDR

I. CONDITIONS FOR THE PEACEFUL USES OF NUCLEAR ENERGY IN THE GDR

The GDR is a small, densely populated, intensively utilized country. It is one of the industrially developed countries with centrally planned economy and mainly national property. This applies particularly to the entire nuclear field, for possession, application, transportation, trade, etc., related to nuclear material and facilities is restricted to national institutions. Accordingly high is the degree of national and economic organization. Thus conditions for a uniform strict enforcement of a State system are favorable.

The use of nuclear energy in the GDR started in cooperation with the USSR in the mid-50s. Today we are operating five power reactors (of the PWR type) with an electric capacity of 1,830 Mwg and further NPP units are under construction. A number of research reactors are used for isotope production, scientific investigations, and training. Isotopes and radiation are extensively used in all fields of industry, agriculture, medicine, and in research.

The basic conditions for the use of nuclear energy in the GDR were laid down in the Atomic Energy Act of 1962. All efforts have been aimed at the exclusively peaceful use of nuclear energy for the welfare of the population. Therefore the universal protection of life and health of radiation workers and the general public is an integral part of its application. For this purpose the Atomic Energy Act provides that import and export, production, ownership, processing, transfer, distribution, transport, storage, disposal, and all other aspects of handling nuclear material are liable to licensing and accounting and are subject to State control.

On this basis, a system for the universal protection against the dangers of nuclear energy has been created in the GDR. The central component of this system is the National Board for Nuclear Safety and Radiation Protection (from now on referred to as "the Authority") that is directly subordinate to the Council of Ministers of the GDR and has been given total State authority to enforce the observance of the requirements of nuclear safety and radiation protection throughout the country. In this field, it has the governmental responsibility for legislation, licensing, surveillance of people, facility and environment, training measures, and information.

Partners of the State control authority are the enterprises and institutions in any way concerned with the application of nuclear energy. Here the managers are responsible for the direct guarantee of all protection aspects. In their work they are advised and supported by intra-plant control officers who, in close cooperation with the Authority, attend to the observance of legal regulations in their fields of control.

II. DEVELOPMENT OF THE STATE SYSTEM OF ACCOUNTING FOR AND CONTROL OF NUCLEAR MATERIAL

Owing to its properties, nuclear material in general necessitates the observance of various safety aspects. It requires

- radiation-protection measures as a radioactive substance
- measures of nuclear safety as a special fissionable material
- safety measures in an international sense as a potential component or source material for nuclear weapons

In all cases the presupposition of effective safety measures is physical protection combined with registration, accounting for, and control of nuclear material.

In the beginning, the regulations made and measures taken on the basis of the Atomic Energy Act were, according to the national and international conditions of that time, mainly concerned with the first-mentioned aspects of protection: licensing for use and accounting for and control of nuclear material

were performed largely according to the valid regulations for the handling of radioactive material and operation of nuclear facilities.

It was only when the NPT came into force as a conspicuous expression of the expanded international safety aspect that the safeguarding of nuclear material associated with the subsequent conclusion of the Safeguards Agreement between the GDR and the IAEA in 1972 resulted in the following tasks:

1. Supplementation and specification of the legal regulations according to international control obligations;
2. Extension of the organizational State system of licensing and control for the use of nuclear material according to Item 7 of the Safeguards Agreement.

These and further tasks have been entrusted to the National Board as the already existing control authority.

By components of the GDR's State System of Accounting for and Control of Nuclear Material (SSAC) we understand

- the legal bases and other obligatory regulations,
- State and intra-plant organization and responsibilities,
- central and intra-plant records for facilities and nuclear materials,
- reports and information within the State and to the IAEA,
- inspections by the State,

including their interaction as well as all measures carried out on this basis with the aim of a complete control of all nuclear material for which the GDR is responsible.

III. NUCLEAR MATERIAL IN THE GDR

For the definition of nuclear material we proceed from Article XX of the IAEA Statute in connection with INFCIRC/153.

In the GDR there are neither facilities for enriching uranium nor for producing or reprocessing nuclear fuel. The fuel supply for power and research reactors as well as its retransfer has been regulated with the USSR on a contractual basis. Thus the flow of nuclear material in the GDR is simplified to the three types represented in Figure 1, which are at the same time characteristic of the types of nuclear material and its changes.

In Figure 1, type I represents the flow of material in nuclear power plants. During use, considerable changes in composition and properties of material occur but there is no violation of the integrity of fuel assemblies as accounting units.

Type II shows the use of nuclear material in research reactors. The fuel flow of larger research reactors is more similar to type I, whereas in critical assemblies working at zero power, the fuel practically remains "fresh" and the production of plutonium can be neglected. To a small extent, there is irradiation and processing of nuclear material for isotope production.

Type III comprises the use of nuclear material for various research purposes and for practical applications to both nuclear and non-nuclear uses. Accordingly diverse are the types and batches of material including uranium of different degrees of enrichment, thorium and plutonium, bulk material, as well as certain individual items.

Allowing for these peculiarities of nuclear materials and facilities in the GDR, and based on information submitted to the IAEA in the "Design Information Questionnaires" for the various facilities, so far nine Material Balance Areas (MBAs) have been established, and the respective Facility Attachments to the Safeguards Agreement have been concluded with the Agency.

These Facility Attachments are the result of intensive discussions between operators, Authority, and IAEA. The establishment of Facility Attachments and Material Balance Areas is always a problem of optimization from both the national and the Agency's view: under given marginal conditions maximum safeguards effectiveness should be aimed at. Table I presents a survey of fields of application of nuclear material according to viewpoints of nuclear material safeguards in the GDR.

At present, our MBAs comprise a total of more than 3,000 batches of nuclear material, ca. 95% of which are fuel assemblies for nuclear power plants.

IV. LEGAL BASES AND OTHER OBLIGATORY REGULATIONS

Proceeding from the Atomic Energy Act and its subsidiary regulations as a basis for the registration, licensing, and control of nuclear material in the interest of radiation protection and nuclear safety, the necessity of special legal regulations for nuclear safeguards arose due to increasing amounts of nuclear material on the one hand and to the increasing importance of international viewpoints on the other. As early as 1970 the nuclear material of the GDR was excluded from the general accounting for radioactive substances and registered in a separate central accounting file, and the system of records and reports was largely developed. After the Safeguards Agreement with the IAEA, including the Subsidiary Arrangements with their Facility Attachments, came into force as valid legal bases, and in the light of the experience gained, the "Nuclear Material Control Order" was established in 1973. It regulates the function of SSAC in detail and contains provisions of responsibilities, licensing and control, records, reports, transfers, exemption inspections, etc.

At enterprise level, the requirements of nuclear material control have been laid down in detailed industrial standards and instructions by facility managers according to central regulations and Subsidiary Arrangements. These documents are coordinated with the Authority and allow for the special aspects in the various fields of use of nuclear material at all stages of planning and operation. They also contain provisions of intra-plant responsibility, operating instruction, bookkeeping, paths and dates of information, dates and procedures of inventory taking, etc.

V. ORGANIZATION AND RESPONSIBILITIES IN SSAC

Organization and responsibilities have been fixed as legally binding by the above-mentioned regulations. As the national regulatory body, the Authority is responsible for all central international and national tasks (Table II).

To meet its tasks, the Authority has set up a Nuclear Material Inspectorate, to which an analytical laboratory has been attached. The rights and duties of this inspectorate go as far as the authorization to demand (in the event of serious violations of legal regulations by the operator) the stoppage of work with nuclear material and to seal such nuclear material.

At enterprise level, the head of an institution that possesses, uses, or trades in nuclear material is responsible for the observance of legal regulations, accounting for nuclear material and for its physical protection. For his own support as an internal control authority, the manager of such an establishment has to appoint a Nuclear Material Officer, who is the direct partner of the Authority's Inspectorate in all questions of nuclear material.

In his field of work, it is the duty of the Nuclear Material Officer to perform control functions, to take steps against violations, to keep facility records and to make reports to the Authority, to cooperate in planning relevant work projects, and so forth. In large facilities several Officers may be necessary and, for intra-plant reasons, a division of MBAs into subareas may be appropriate.

The described organizational relations between State Authority and establishments are represented in Figure 2, which illustrates the embedding of nuclear material safeguards in the total protection system for the peaceful uses of nuclear energy and its parallel with other control relations. In our experience, this embedding makes it posible to have close cooperation of the various inspectorates and officers in the broader interest of all-around safety, and guarantees a valuable exchange of cross information.

IV. SYSTEM OF RECORDS

The type, content, and dates of central and facility records within the SSAC meet, besides national tasks, the requirements of international safeguards.

Accordingly, in addition to all legal regulations, the central records contain:

- design information for "nuclear material facilities" and any change of these data,
- list of material balance areas and, if necessary, subareas,
- accounting records including procedures for inventory taking,
- accounting and other reports to the IAEA,
- total balances of the material subject to and exempted from safeguards,
- sealing records,
- records for national and international inspections including approval of IAEA inspectors and instructions for inspection statements,
- directives made for nuclear material institutions and notes on their fulfillment,
- national and international correspondence relevant for control of nuclear material.

The records to be kept by the operators to account for their nuclear material comprise accounting and operating records. For this purpose, records that are necessary anyway for operational reasons can in many cases be used to meet the requirements of safeguards. A list of the accounting and operating records required in a nuclear power plant can be seen in Table III.

In general, accountancy data for nuclear material can be taken from:

- producers' or shippers' certificates,
- results of identification and accounting for completeness,
- results of analyses,
- calculations and measurements of material changes due to use.

The respective accounting methods applied have to be stated by the operator.

VII. INFORMATION, REQUESTS, REPORTS

The reports to be made and other information to be given to the Authority by the operators have to meet (besides national requirements) the stipulations of the Safeguards Agreement with the IAEA and the Subsidiary Arrangements and have to allow for the necessary period of processing in the Authority and for transmission to the IAEA within given dates through the channels agreed. For this national reporting, special forms or codes are not prescribed.

Requests for exemption of nuclear material from international safeguards according to Article 36 or 37 as well as requests for termination of safeguards according to Article 35 of the Safeguards Agreement are made by the Authority on application of an operator or owing to central considerations.

Material exempted from safeguards by the IAEA still remains subject to national control. Consumption in non-nuclear activities has to be documented. For the discharge of nuclear material into radioactive wastes special criteria concerning amount, procedure, and control have been provided.

Reports and information for the IAEA are prepared and communicated exclusively by the Authority. Copies of these reports are included in the central records and in the respective facility records. Figure 3 represents a survey of the information flow within the reports system of SSAC.

VIII. DATA PROCESSING

The central accounting of physical inventory and its changes and the preparation of reports require the processing of a large amount of data and considerable paper work. Moreover, quick retrieval of data, error-proof processing and reproducible playback are necessary.

On the other hand, the constant nature of tasks and requirements, in terms of form and content, creates favorable conditions for computer methods of data processing. Therefore a universal computer program for control of nuclear material has been worked out in the Authority to rationalize accounting

and reporting, beginning with the large number of fuel assemblies in nuclear power plants with their identical kind of labeling and use and gradually extended to all inventories of nuclear material.

This universal ADP system allows for the respective IAEA requirements on reporting (Code 10 of Subsidiary Arrangements as well as IAEA/STR-42 and IAEA/STR-59). As data carriers, both printouts and magnetic tapes (according to IAEA/STR-53) are used; the data can be read directly into the processing equipment of the IAEA, and thus manual work decreased.

Besides the notes and reports (ICR, PIL, MBR) to be given to the IAEA, the individual programs supply data within the national tasks of control.

Figure 4 presents a survey of the ADP system for nuclear safeguards in the GDR. Its central basis is the nuclear material data bank, section I of which stores all reports made so far, while section II contains the latest book inventory. A key function is held by the batch names of uniform structure. To adapt batch names of foreign suppliers, a "renaming" procedure is followed that works in the same way as the Code 10 rebatching procedure.

The "ICR" program serves to update the book inventory (File II) and to prepare reports on inventory changes to the IAEA according to Code 10. It makes corrections possible and allows for material change due to burnup and element transmutations. To register material that (according to Articles 35 to 37 of the Safeguards Agreement) is not subject to IAEA control but still remains under national control, "fictitious MBAs" were established.

The "PIL MBR" program produces Physical Inventory Listings (PIL; on the basis of File II) and Material Balance Reports (MBR; by means of File 1).

As an example of an ICR output a printout is compared with the respective standard form of IAEA R.01.1/C in Figure 5. Reporting forms R.02/C and R.03 have been drawn up accordingly.

The "POL" program calculates relevant nuclear material concentrations in spent fuel assemblies based on their dependence on burnup.

In addition to the mentioned main programs there are a number of subsidiary programs e.g., "LISTE" to print out any part of the file (e.g., to prepare inspections, for purposes of comparison, etc.) and "DELE" for aimed access to stored data. The "SIP" program lists all seals in the MBAs and their data.

IX. INSPECTIONS

Inspections within SSAC are made by the members of the Nuclear Material Inspectorate of the Authority in the presence of the head of the institution inspected (or a deputy appointed by him) and the Nuclear Material Officer. They are preferably made when working out control concepts, taking the inventory of nuclear material, transferring nuclear material into or out of the MBA, during reactor loadings, unusual occurrences etc., but also randomly to check inventory and records. Thus they also follow the priorities of international inspections. In a similar way, inspection effort is related to type and amount of nuclear material.

Tasks of inspections are to check design information, data in reports, operating records, inventories of nuclear material directly, as well as measures of containment and surveillance. Inspections may include measurement and sampling.

Following an inspection, a protocol is drawn up and signed by the partners. It contains the object and result of inspection and possible directives. As a rule, such inspections at the same time serve to discuss and clear up special control problems of the institutions inspected.

In general, international inspections are prepared by the Nuclear Material Inspectorate of the Authority. This holds especially for coordination of dates allowing for the concrete operational situation of the operator. As IAEA Inspectors are accompanied by members of the Nuclear Material Inspectorate of the Authority in international inspections, these inspections

are performed at the same time as inspections within SSAC. The results of international inspections are included in the analysis.

X. CONTAINMENT AND SURVEILLANCE MEASURES

On a national scale as well, containment and surveillance measures play an important role. They are also applied to a great extent by the operator to meet his duties of good management, physical protection, and other safety requirements. Accordingly, they consist of a coordinated system of barriers, automatic safety devices, surveillance equipment, regulations, and controls.

On the part of the State, C and S activity is generally limited to the assessment and control of measures taken by the operator.

The C and S measures applied in connection with international control considerably reduce the control effort by fixing the "status quo" of material or facilities. Therefore the operators have never offered any resistance to these measures but always cooperatively pooled ideas with the Authority. With respect to their importance and the consequences of violation associated with them, we distinguish between "strategic" and "convenient" seals. Surveillance cameras are (in many cases redundantly) used at strategic points or accesses to these points.

XI. EXPERIENCE IN THE COOPERATION OF DIFFERENT LEVELS OF CONTROL

Authority - operator

The main basis of any constructive cooperation with operators is the insight that, on a national scale, control of nuclear materials is necessary for the security of personnel and population, for safe operation of facilities, and for the protection and optimum utilization of valuable materials. Also on an international scale, control of nuclear material has to be considered a necessary part of protective measures in using nuclear energy. It would be a wrong conclusion to assume that

today only national aspects could be taken into account: no State can, in view of the high internationalization of nuclear energy, neglect the security interests of its neighbors or other States.

It should be principally stressed that the relationship of Authority and operators is not at all limited to exchanging reports and to inspections. From a national view, we do not regard the observance of formal duties as the primary task. It should be considered that in any case the Authority can apply the complete spectrum of State regulations and also means of power. It can exploit the entire complex of licensing and control functions including respective information. In this connection also, the primary interest of the State in prophylaxis should be pointed out i.e., interest in safe management, physical protection, and careful handling of nuclear material (as well as economic reasons). Apart from international control requirements, under our conditions it appears to be the main task to educate the operator (by advising, training, and control) to handle nuclear material accordingly. For this purpose sufficient confidence and close unbureaucratic cooperation are imperative. This does not mean that tasks and responsibilities are blurred.

Thus motivated, operators are generally ready and willing to cooperate and eagerly try to demonstrate the exact accounting for material as the sign of a well-managed enterprise, particularly for international inspectors. In this respect, "practical workers" e.g., operators of NPP, are in no way inferior to "research workers" e.g., in institutions. Detected incorrectness may lead operators to diminish underestimations and to review the internal regime - an example of an additional effect of international inspections.

On the basis of convincing motivation, also the problem of effort and interference can be cleared up reasonably. Of course these questions play a role; in the case of producing

facilities, the temporal and economic effort is a major concern; in the case of research institutions the ensurance of confidentiality of work is generally at the fore. Internal requirements have to be taken seriously and brought into agreement with the aims and methods of inspection, in careful discusion with the Authority and IAEA. In this respect, especially those inspections that lie on the "critical path" of an operation (e.g., the forthcoming start-up of a loaded reactor after refueling) should be considered and limited to the inevitable minimum.

Supplier - receiver

In the cooperation between supplier and receiver of nuclear material (especially in international transfers), the punctual and exact notification of the shipment to the receiver and the "control-related" issuance of certificates are of particular importance so that IAEA notification periods can be observed, exact material data given, and ambiguities (e.g., with respect to shipper-receiver differences) avoided. Figure 6 shows a useful path of information parallel to the direct trade channel that considerably facilitates control and should be agreed upon between State authorities.

Certain problems of different batch names used by supplier and receiver can easily be solved by means of ADP.

SSAC - IAEA

At present, there have been more than eight years of experience in performing international safeguards of nuclear material in my country. Within this period about 670 reports were communicated to the IAEA. The Agency's inspectors made more than 100 inspections in the GDR. This experience enables us to state that the international controls by the IAEA

- respect the sovereign rights of the State,
- do not hamper the GDR's development and international cooperation in the peaceful uses of nuclear energy,
- do not hamper or endanger the operation of our nuclear facilities or other institutions,
- do not require undue expenditure.

On the other hand, inspectors and statements by the IAEA confirm that the GDR observes the obligations undertaken in the Safeguards Agreement. Thus the SSAC has also stood the test.

These statements can be made only because both parties, convinced of the sense of their efforts, try their best to fulfill the "spirit of the Agreement." On this basis there always has been and still is a continuous exchange of questions, ideas, and experience in an endeavor to adhere to the "letter of the Agreement" i.e., for reasonable interpretation and limits in special cases, for appropriate forms, effective procedures etc., that also includes the users of nuclear material in the GDR. This cooperation begins as early as the working out of design information and continues with the determination of Material Balance Areas, the elaboration of Facility Attachments and the planning of strategic points for control of flows and inventory, seals, surveillance equipment, etc. It has proved a success also in necessary concrete arrangements in the many cases in which Safeguards Agreement and Subsidiary Arrangements cannot supply explicit information.

It is also of importance in this respect that, in addition to all its technical tasks, the IAEA also continuously pays the necessary attention to the motivation of its inspectors and to the development of a uniform control philosophy. This is not only an internal requirement of every control organization but also necessary for a convincing and unambiguous conduct towards the institutions controlled.

That my country supports the IAEA control system not only in moral but also in practical terms by considerable efforts (e.g., in the development of measuring methods for spent-fuel assemblies, by cooperation in the training of inspectors and in study tours within the framework of technical assistance) should be only briefly mentioned here.

Today we may state with satisfaction that this cooperation has proved fruitful for all parties and has served further development. In these constructive relations we do not see any contradiction to the Safeguards Agreement but a presupposition for its fulfillment.

TABLE I, MBA Types in the GDR

Facility Type	Material	Flow Scheme (Fig.1)	KMP Inventory +)
Nuclear Power Plants (PWR)	Fuel assemblies (LEU)	I	. Fresh fuel storage . Reactor core . Spent fuel storage . Other locations (if needed)
Research reactors with attached a.o. laboratories	Fuel assemblies, rods, plates, pellets (LEU - HEU);	II	. Fresh fuel storage . Reactor core . Spent fuel storage
	Research material (different elements, enrichment, forms)	III	. Special lab.s . Other locations
Miscellaneous locations	Research materials (different categories);	III	. Trade organization . Locations of main users
	Neutron sources with Pu;		. SAAS (Compilation of all small users)
	Others		

+) Flow KMPs in all cases: . Increase (e.g. receipt, nuclear production, a.o.)
. Decrease (e.g. shipment, nuclear lose, exemption, a.o.)
. Accidental loss; Measured discards

TABLE II, Tasks of the State Control Authority in the SSAC

International tasks:	- maintenance of contacts with the IAEA and settlement of all organizational and technical questions with the IAEA in implementing the Agreement
	- reporting and passing on of necessary information to the IAEA and checking of communications by the IAEA
	- co-ordination with the IAEA in control procedures and developments
	- preparation and approval of Subsidiary Arrangements, beginning with Design Information
National tasks:	- checking and, if necessary, revision of legal regulations
	- licensing and control of all projects associated with nuclear material
	- keeping of central records for control of nuclear material
	- preparation, accompanying and evaluation of IAEA inspections and conduct of inspections by the Authority
	- national co-ordination on questions of safeguards with other State authorities
	- supply of information and provision of training for users of nuclear material with the aim of motivation and guaranteeing the necessary level of accounting
	- approval of intra-plant regulations for the use of nuclear material
	- performance or stimulation of necessary systematic or experimental development work, as far as required for meeting national demands.

TABLE III, Accounting and Operating Records for Nuclear Material Control in an NPP

Accounting records:	- general ledger for each category of the available nuclear material
	- fuel assembly certificates (supply and retransfer, respectively)
	- certificates, measurement protocols and other records of bulk material and neutron sources as far as available in an NPP
	- shipper/receiver protocols
	- copies of information to SAAS
	- copies of information of SAAS to the IAEA
	- copies of information of the IAEA to SAAS
	- cards containing data on location of each fuel assembly as well as on burn-up upon discharge, including content of residual uranium and of plutonium of spent fuel assemblies, and other nuclear material data
	- nuclear material journal accounting for material which is not available as fuel assemblies
Operating records:	- diagram showing the integrated thermal reactor power
	- fuel diagram for reactor core, fresh and spent fuel storages
	- log-book of the operations of the refuelling machine crane
	- book containing information on all receipts and shipments of nuclear material
	- protocols of handling nuclear material in laboratories as far as such a work is carried out in an NPP
	- records of the reactor hall lighting
	- seals records

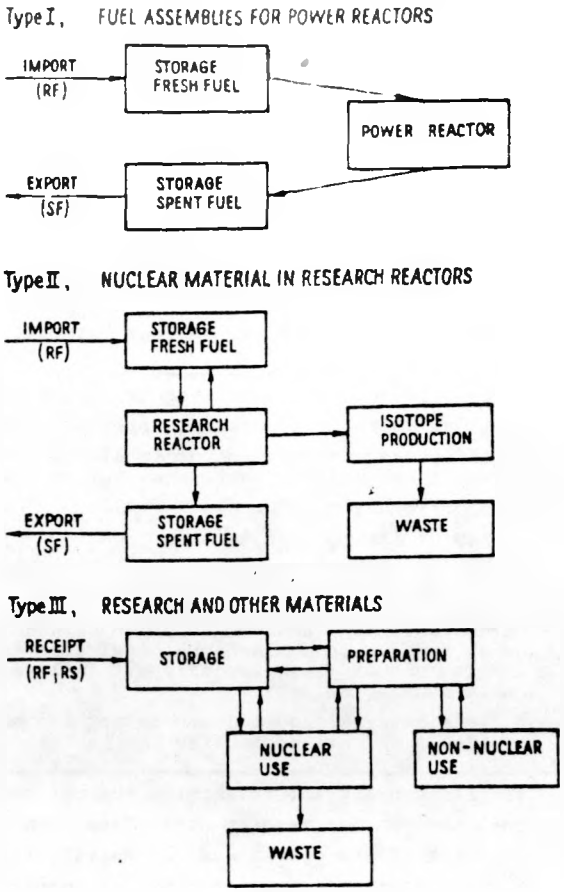


FIG. 1. Flow of nuclear material in the GDR.

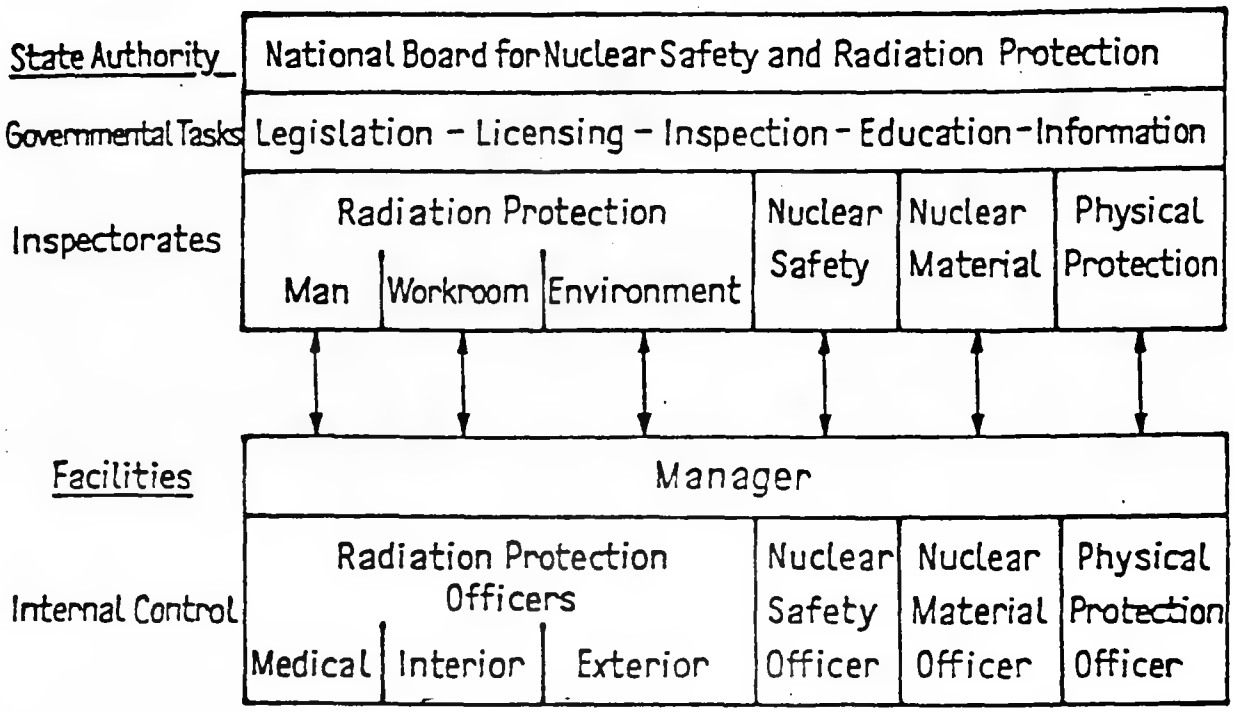


Fig. 2 Nuclear Material Control as a Part of the Overall Protection System in Nuclear Energy in the GDR

Fig. 3 Information Flow
Material Balance Area ⇄ SAAS ⇄ IAEA

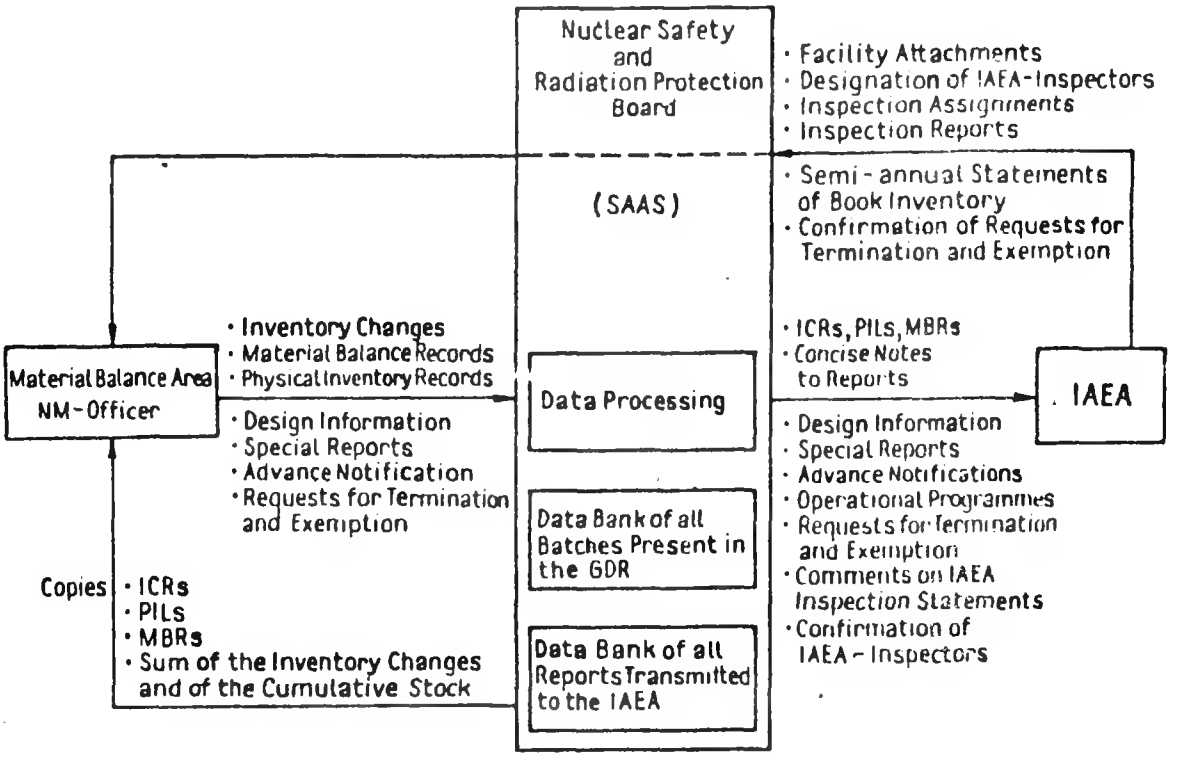
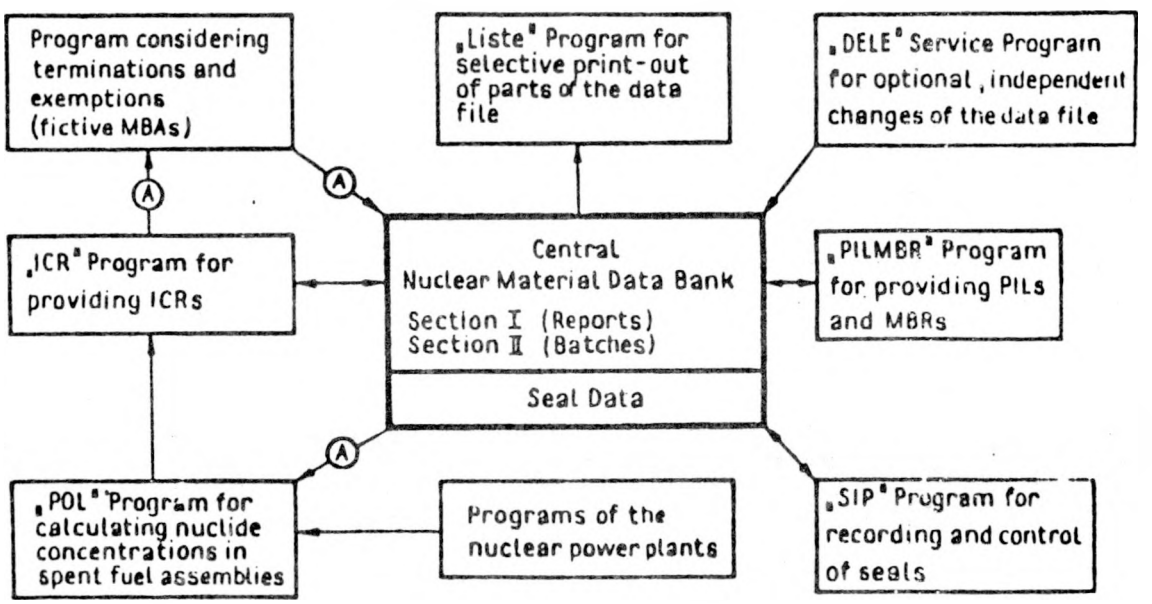


Fig. 4 Compunents of the Universal ADP System of the GDR for Recording, Control and Reporting of Nuclear Material



Ⓐ: To avoid intermediate working processes these partial systems are automatically linked

INVENTORY CHANGE REPORT (ICR) FORM R.01.1/c

COUNTRY DD
FACILITY DDG-
MATERIAL BALANCE AREA DD-G

PERIOD COVERED BY REPORT FROM 77-01-01 TO 77-01-01
REPORT No. 1
PAGE No. 1 OF 1 PAGES SIGNATURE

0 0 1 DDG-DD-G 7 7 0 1 0 1 7 7 0 1 0 1 2 8 1

ENTRY No.	CORRECTION	DATE OF INVENTORY CHANGE	MBA/COUNTRY FROM	MBA/COUNTRY TO	TYPE OF INVENTORY CHANGE	EMP CODE	NAME OR NUMBER OF BATCH	NUMBER OF ITEMS IN BATCH	MATERIAL DESCRIPTION	ACCOUNTANCY DATA ELEMENT	WEIGHT OF ELEMENT	UNIT kg/g	WEIGHT OF FISSILE ISOTOPES (URANIUM ONLY) (g)	UNIT kg/g	MEASURE BASIS	CONCISE NOTE	CORRECTION TO REPORT No.	CORRECTION TO BY TRY No.
01		770101	Z	DD-G	RF	1	3-125	1	EASF	E	180000	G	2500	G	N			
02		770101	DD-G	DD-G	RM	2	3-125	1	EASF	E	180000	G	2500	G	L			
03		770101	DD-G	DD-G	RP	1	GEGK3125	1	EASF	E	180000	G	2600	G	L			

N 69A/Rev 3 (Sept 77)

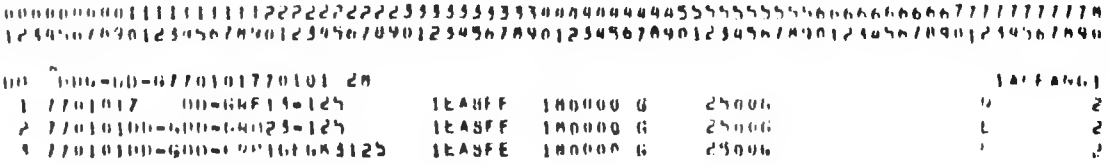


Fig. 5, Example of an ICR in the standard form and as a computer print-out

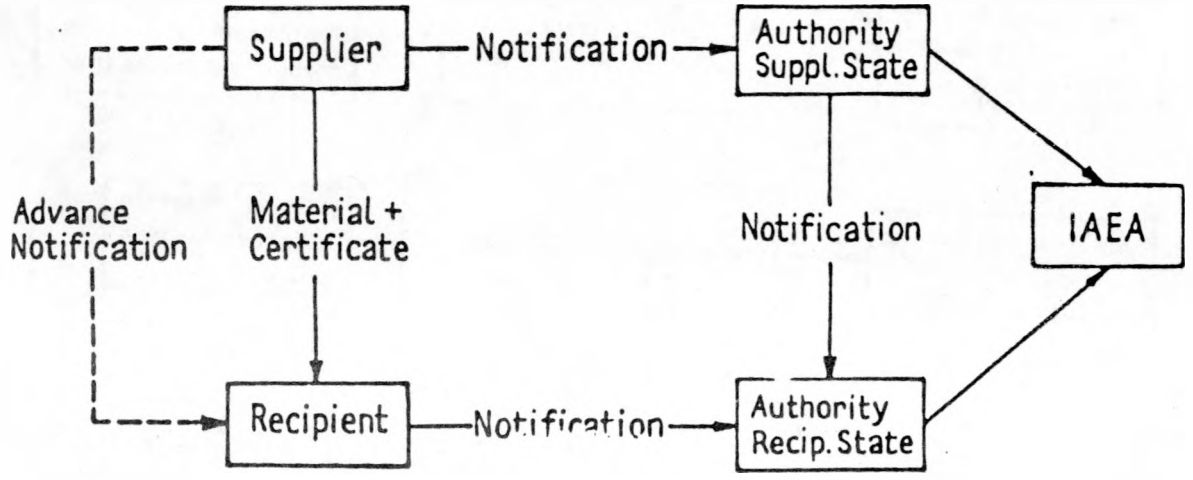


Fig. 6 Flow Scheme SF - RF

INTERNATIONAL TRAINING COURSE ON NUCLEAR MATERIALS ACCOUNTABILITY FOR SAFEGUARDS PURPOSES

SESSION #29: EXAMPLE OF AN OPERATING STATE SYSTEM (JAPAN)

SPEAKER: Dr. Hiroyoshi Kurihara

Embassy of Japan
Washington, DC USA

Wednesday, June 4, 1980
8:30 p.m.

BIOGRAPHY

Education: Chiba University (Japan), Agricultural Chemistry, B.A.; London University, Radiation Biology & Radiation Physics.

Present Position: Science Counselor, Embassy of Japan in Washington, DC.

Present Duties: Diplomatic Services in connection with Science & Technology including Nuclear Affairs.

Past Positions: Director, Safeguards Division, Science & Technology Agency of Japan; First Officer, Division of Development, IAEA.

INTERNATIONAL TRAINING COURSE ON NUCLEAR MATERIALS ACCOUNTABILITY FOR SAFEGUARDS PURPOSES

SESSION #29: EXAMPLE OF AN OPERATING STATE SYSTEM (JAPAN)

SPEAKER: Takeshi Osabe

Japan Nuclear Fuel Company, Limited
Kanagawa-Ken, Japan

Wednesday, June 4, 1980
8:30 p.m.

BIOGRAPHY

Education: Graduate from Mechanical Engineer Course at Rakuyo Technical Highschool in Kyoto-city, Japan.

Present Position: Manager, Nuclear Materials Management, Quality Assurance Department of Japan Nuclear Fuel Company.

Present Duties: Establish safeguards and accountability system for the facility and supervise personnel to perform SNM accounting and control.

Past Positions: Accounting and property control. Quality control in construction of nuclear reactor.

INTERNATIONAL TRAINING COURSE ON NUCLEAR MATERIALS ACCOUNTABILITY FOR SAFEGUARDS PURPOSES

Session Objectives

SESSION #29: EXAMPLE OF AN OPERATING STATE SYSTEM (JAPAN)

The establishment and operation of an advanced state system of accountability and control is presented, and its interface with facility operators, with the IAEA international safeguards system, and with the public sector is discussed.

The implementation of a plant accountability system, and experience with application of safeguards inspection (national and international) to a low-enriched uranium fuel-fabrication facility in Japan will be described in detail.

SESSION 29a: EXAMPLE OF OPERATING STATE SYSTEM

Hiroyoshi Kurihara
Embassy of Japan

I. INTRODUCTION

The purpose of my lecture is to give the participants the experiences of the establishment and operation of a national system of control, using as an example Japan's case, and give some advice to the people who are associated with the national system. The operating experiences of a fuel fabrication plant will be explained by my co-lecturer Mr. Osabe.

II. STATE SYSTEM OF MATERIAL CONTROL

Japan ratified the Treaty on the Non-Proliferation of Nuclear Weapons (NPT) in June 1976. According to Article 3 - 4 of the NPT, Japan had to negotiate a Safeguards Agreement with the IAEA. The Japanese Government had already signed the NPT in 1970, immediately before the opening of the ratification discussions; therefore, the Japanese Government had started preliminary negotiations with the IAEA in June, 1972. In 1975 the draft Agreement was initialed, although Japan had not ratified the NPT at that time. Because the approval of the Japanese Parliament was needed for entry into force of the Safeguards Agreement, the actual entry into force of the

Agreement was delayed until December 2nd, 1977. When I assumed the responsibility for safeguards in the Atomic Energy Bureau (AEB) of the Government after coming back from the IAEA in December of 1975, the organization in the AEB was called the Safeguards Office. The Safeguards Agreement between the IAEA and Japan was already concluded, but there were no discussions on the Subsidiary Arrangements or on the facility Attachments. I had to face very difficult but challenging jobs; on one hand I had to negotiate matters with the IAEA, on the other hand I had to construct our national safeguards system to conform with the IAEA system.

III. MEASURES TAKEN FOR IMPROVING OUR NATIONAL SYSTEM

As you already learned through other topics in this course, a country which ratified the NPT and concluded a Safeguards Agreement with the IAEA has a responsibility to establish and maintain a national system of material control. There are two alternative national systems, i.e. a system with or without an independent verification capability. Japan had chosen the former, therefore she must maintain the independent verification capability through the national inspection activities. I understand that so far EURATOM and Japan have chosen this alternative of having independent verification capability. It is rather easy to say that we can establish and maintain the effective national system of material control, but doing so actually is a very difficult and complicated job. I would like to describe the kinds of requirements that will be requested when a state intends

to establish a national system. Also, I would like to discuss my own experiences.

A. Promulgation of Regulations

First, you should decide and establish the framework of regulations, taking into account the State's responsibility stemming from the Agreement and also from the NPT itself. (Refer to Article 2 of the NPT. "Each non-nuclear weapon state ..... undertakes not to manufacture or otherwise acquire nuclear weapons ....."). Normally the regulations to be promulgated would incorporate the rights of a central control body of the Government. Those rights are as follows: (i) to receive the information on the design and material management system of the facilities: (ii) to require the facility people to record and to maintain the necessary information for the material accountancy within the facility; (iii) to ask delivery of the reports to the central body; and (iv) to accept the IAEA and the national inspections, if there are such inspections. It is important that you coordinate the contents of your State's regulations with the requirements of the IAEA. In our case, we changed some of our domestic regulations on the control of nuclear energy in Japan after completing the negotiation of the Safeguards Agreement with the Agency. It had to be submitted to the Parliament for approval.

B. Japan's Legal System on Nuclear Energy

Now, I would like to explain our legal framework on the utilitization of nuclear energy in Japan. The most important law in our system is the Atomic Energy Basic Act of Japan. This Act

clearly states that Japan uses nuclear energy for peaceful purposes only. Under this Act, we have many laws. One of them is called "The Regulation Law of Nuclear Reactor, Nuclear Fuel Material and Nuclear Source Material".

This law was originally concerned with the regulatory aspects of nuclear safety, radiation safety, and health physics. However, we saw that this law might be a good tool for accommodating the safeguards requirements, because this law compels the facility operators to meet many safety requirements, e.g. the need to make application for a facility design license, maintain records, report to the licensing authority, and have inspection by the National inspectors. We have had the possibility of incorporating the safeguards requirements into this law. We amended this law and added several new requirements. I would like to draw your attention to the following point. Generally speaking, the nuclear safety people are concerned with the facility design details. Also, if the design has not met the required level of the regulations, the right of the licensing authority to order the changes in design of the facility and equipment therein is clearly stated. We, therefore, decided to rely on the regulatory activities of nuclear safety to include the design of facility and of equipment contained therein. On the other hand, the safeguards authority requires a lot of information on the capability and performance of facility material accountancy. Normally it can not be done without adding new regulatory requirements. I would like to mention that in our country the safety and safeguards authorities

are in the same organization under the same Minister, namely the Minister for Science and Technology, who is a Cabinet Member (Figure 1).

In our country, each facility operator, if he is handling nuclear materials, has to submit to facility safety regulations and to get the approval from the licensing authority. The facility safety regulations, unfortunately, do not cover the material control system which is needed for safeguards implementation. Therefore, we put into "the Regulation Law" the new regulatory requirements of submitting to, and asking for approval from the safeguards licensing authority on the material accountancy regulations of the facility. By this addition, we can collect the information on the facility's material accountancy system.

As for other requirements of international safeguards (as well as domestic safeguards), namely, records, reports and inspections, we also incorporated those requirements into the "Regulation Law". Details on those items will be explained later.

C. Increase of Funding for Implementing Safeguards and Upgrading of Manpower and Quality of Safeguards Inspectors.

I would like to explain at a later stage the detailed safeguards requirement, records, reports etc. Now I want to explain other aspects of measures taken in connection with improvement of our national system. Apart from promulgation of legal requirement there were several measures which I had to take after having the responsibility for safeguards implementation,

and I think those will serve as a good examples for others who want to establish or to improve their national system.

Those are as follows: Expansion or upgrading of the safeguards organization, increased funding for safeguards implementation, increased manpower, upgrading of manpower capability, modernization of the system, and seeking understanding of high level people.

First, we enlarged and upgraded the Safeguards Office of the NSB. It was under the direction of the Director of Nuclear Material Regulation Division. The new organization is called the Safeguards Division and has a Director, who is the same rank as the Director of the Nuclear Material Regulation Division. By this upgrading we have a lot of flexibility to deal with the IAEA, and also with facility people (Figure 2).

Second, implementation needs much money. We aimed for a sophisticated national system which has a computer, its own analytical laboratory, good inspectors, and good instruments for inspections. Therefore, it became, very important to persuade the finance people. In Japan, as in most countries, it takes a lot of effort to persuade the people in the Ministry of Finance. When I came to this post in 1975, the total expenditures for safeguards implementation and R & D were 0.1 million dollars. (In our calculation we excluded the costs for personnel; therefore if salary, wages, travel fees are included, the figures would be more nearly 0.4 million dollars.)

The next fiscal year (1976), funding was increased to 0.35 million dollars, in 1977 to 1.25 million dollars and the last

year of my responsibility, 1978, to 2 million dollars. Within 4 years the safeguards budget was increased about 30-fold. It is very important to have sufficient money for successful implementation, so I recommend persons responsible for implementing safeguards to concentrate a large portion of their activities to obtaining funds.

Third, the increase of manpower. Even when we introduce a computer system for record keeping, and sophisticated instrumentation system, we still need a lot of manpower. This is especially so in our case, since we decided to implement the national safeguards inspections in addition to the international inspections. You may not know how difficult it is in my country to increase the number of employees for any government organization. Recently, the public opinion keeps saying that we have too many government employees, and should be reduced drastically. Having such an atmosphere in our country, we met with much difficulty in increasing manpower. But fortunately, we were able to increase the number of our inspectors.

It is very important again to have the understanding of people who control the number of personnel in the Government. Upgrading of inspectors and other personnel who work for safeguards implementation is also very important. In our country, those persons who work in the central safeguards organization of the State have the job of instructing the facility operators periodically. Therefore, maintaining their capability is a very important factor in the level of the whole national system. We

sent our inspectors to various seminars and training courses for education purposes.

Fourth, modernization of the system. By the word "modernization", I mean the introduction of a sophisticated computer system, various instruments for the inspectors, analytical laboratories, etc. Already you had many topics dealing with these matters in this training course, and I do not want to repeat the general discussions here. So far, I explained the effort of improving our national system, so that we could go along with the NPT-safeguards regime. In the remaining parts of my lecture, I would like to explain our experiences with (a) negotiation with IAEA on the Subsidiary Arrangements and Facility Attachments, (b) design examination, (c) records system, (d) reports system and "National Account", (e) inspection, and (f) verification activities.

IV. NEGOTIATION OF SUBSIDIARY ARRANGEMENTS AND FACILITY ATTACHMENTS

During my stay as Director of the Safeguards Division, my busiest and most important job was to negotiate and conclude the Subsidiary Arrangements and Facility Attachments with the IAEA. I should like to say several words for Mr. L. Thorne, the Chief of the Far East Section of the IAEA. He and I worked together and I was impressed with his ability, cooperative attitude, and also his very good judgement for the political moods. Without his cooperation, I could not have finished my job. This proves how the cooperation between the IAEA and State's people is

important, and the importance of the State's people in maintaining a good relationship with the IAEA. You should realize that the number of facilities in my country is more than 500, if we include "Nuclear Material Outside Facility". To save time, we at first took one from each type of facility as models of facilities to be discussed. The general part of the Subsidiary Arrangements is fairly normal and except for a few points we did not have real difficulty. The detailed process of negotiations will not be explained to you, very unfortunately, since those are confidential matters. However, I can make some general remarks. Since we decided that Japan's system should have independent verification capability, the coordination of activities between the IAEA and Japan is very important and needs further discussion.

As a reminder I would like to say the following. The fact that a State system has an independent verification capability does not mean that the Agency need not have independent capability.

On the contrary, Agency inspection capability must be secured firmly irrespective of national inspections. It took about two years to complete a set of the Subsidiary Arrangements and Facility Attachments. Apart from Subsidiary Arrangements, we concluded 72 separate Facility Attachments. The Agency has many types of Model Facility Attachments, and it has also formats for Design Information Questionnaires (DIQ) for each type of facilities. The first step for the National System people is to prepare the DIQ. We had the formats of the DIQ from the Agency

before official entry into force of the Safeguards Agreement. Since we are the licensing authority for safety as well as safeguards, we are supposed to have all of the design information and information on material accountancy for the facilities. Actually we did not. When we received them from the Agency, the Agency was in the process of elaborating the DIQs, so, half-way through completing the DIQs and FAs, we had to collect new types of information from the operator. And that meant we had to amend our domestic regulations. It cost additional effort to the State authority. We are careful about collecting commercially sensitive information without legal authorization. Finally, together with the operators we completed all of the DIQs and the FAs. In other words, we (the State authority) asked the help of the facility operator to complete the DIQs etc.

I do not know whether this is the right way or not, but with the completion of more than 70 DIQs within a limited time by less than 10 persons, we needed the assistance of the facilities. If I may say so, some DIQs, especially DIQs for the smaller facilities e.g. research center or NMOF, have too many columns to be filled with respect to the actual mode of utilization of nuclear material. We had lengthy discussions on the smaller facilities with the IAEA. In our country, any amount of nuclear material must be licensed. For example, if a research laboratory wants to use 1.8 mg of Pu for experimental use, or if an electron-microscope laboratory wants to use a bottle of uranyl acetate solution containing 25 g of natural uranium as a reagent, the laboratory must have a license from the authority. Our

national system of material control must take care of these very small user's. Even in a case of 25 g of natural uranium, the user is considered as an independent facility in the National System. However, from the safeguards point of view, it would be advisable to be consolidated into one large conceptual MBA or NMOF. We did such consolidation for the Agency's sake. Again, we encountered numerous tedious but must-be-solved accountancy problems like the exemption, or termination of safeguards of these minor quantities. If you would establish a national system, the problem of treating these minor user's is one of the problems to be solved for adjustments with the Agency's system.

After submission of DIQs to the Agency, Facility Attachment for each facility should be drawn up in cooperation with the State and IAEA. For the Japanese Authority, one of the focal points of the negotiations which started with the negotiation of the Safeguards Agreement between Japan and the IAEA was to complete the Facility Attachments. There are several reasons. First, these Attachments are of direct concern to the operator. It is understandable in our country that operators do not welcome any disturbing or hampering of their activities. Safeguards inspection, especially made by the foreigners who can not speak Japanese, is certainly a disturbance from the commercial point of view. Therefore, the first thing for us, the national system people, was to explain the importance of safeguards implementation and to pursuade the operators. Without cooperation of facility operators, the administration of national system would be a difficult job. Second, the implementation of safeguards was

of considerable interest to our politicians, and they watched the progress of our negotiations on Facility Attachments very carefully. They were most interested in the ARIE figures. Therefore, we had to prepare good explanations when we agreed on the specific figures of ARIEs and also these figures should be generally equivalent with ARIEs for the same type of facility in other countries. As I mentioned before, with the cooperation of IAEA people, we succeeded quite satisfactorily in the discussions on Facility Attachments, including the figures of ARIEs.

V. DESIGN INFORMATION

As I said before, we have two different legal ways to collect information from the facility. One way is by means of the facility application for a safety license. Necessary information on the design of the facility is attached to the documents submitted for the application. The other is submission of material management regulation of the facility to the authority.

After collecting design information, the problem faced was the following. What extent of detail is sufficient in the DIQ, for safeguards purposes, without disclosing unnecessary information. Operators of commercial facilities are always sensitive about disclosure of information associated with their commercial activities, for example capacity of some equipment, quality control measures, the capability of measurement, etc. I must say that sometimes the Agency wants more information than the minimum necessary for performing their safeguards responsibility. Japan wanted to have a computer system for storing and reviewing design

information submitted by the facility operators. We subsidized the R & D on this program, but up to this date I know nothing of the result of such R & D. Collecting, checking, and compiling more than 70 DIQs is quite a heavy task for us, not having English as a mother tongue. Even the translation became an enormous job. We had to postpone the prearranged submission date to sometime later due to the translation delay. Your country may not have such difficulty, but we need to pay due attention to such clerical matters. Of course, this type of caution must be paid not only DIQs but also FAs, and other documents. Otherwise an unexpected delay of submission to the Agency might occur.

VI. RECORDS SYSTEM

The kind of records that should be maintained in the specific facility are listed in the Subsidiary Arrangement and also in the Facility Attachment. I am not sure of the case in other State's, but in Japan's case the safeguards Agreement was concluded between the Japanese Government and the IAEA. Strictly speaking, the facility operator, therefore, is the third and independent party to this Agreement. Nothing concerning the Agency requirements to the Japanese Government, unless the Japanese Government requests otherwise, require the facility operators to obey obligations arising from international commitments made by the Government. Therefore, we needed to promulgate our Ministry's orders in which specific items of records must be recorded in a timely manner and be retained for a specified time. We did not specify the formats for the records.

As explained later, we needed to have a specific format for the reports, but we did not need the uniform format for the records. For example, in the case of a commercial fuel fabrication plant and of a small scale laboratory, the required type of records would be the same, but in the commercial plant they may keep their records in the memory cores in the computer, whereas in research laboratories a log book is enough. As you know already, records kept in the facility are the basis for checking the reports when our inspectors, as well as the Agency inspectors, visit the facility. Based on the articles in the Safeguards Agreement, reports to the Agency should be submitted in one of the Agency's official languages. However, there is no rigid limitation for the languages to be used for the records system in the Agreement. Normally, our facility operator keeps his records in Japanese.

Because material accountancy records have a lot of numerals in the log-books, an explanation is given in Japanese. Also, because the Agency inspectors who were once assigned to Japan had at least some sense of understanding the records, we have had no real difficulty. If the Agency inspector meets with difficulty, assistance will be supplied by the national system people.

VI. REPORTS SYSTEM

The Safeguards Agreement as well as subsidiary Arrangements specify the reports that should be submitted to the Agency. The most important ones are Inventory Change Report (ICR), Material Balance Report (MBR), and Physical Inventory Listing (PIL).

Concerning records and reports, we have to consider the relationship between the entire facility, the national system and the Agency. The flow of information between them is shown in Figure 3.

Conceptually, the facility reports are submitted to the national system (national safeguards authority), and after processing those reports by the authority, the national system will dispatch ICR, MBR etc. to the Agency. There are two types of reports in this flow. The first one is concerned with domestic reports, hence the national authority must check the validity and credibility of the reports. Once those reports are processed in the national system, then ICR etc. is sent from the national system to the Agency. These reports are the ones which are referred to in the international safeguards regime.

We have to pay special attention to the actual implementation of the reports system. First, contents and timing of reports to the national system should be specified by the promulgation of regulations concerned. Reports to the Government in our country (not only safeguards or nuclear activities but every type of the report) should be sent in Japanese, since Japanese is the only official language. Logically the national system should receive the reports in Japanese, translate them into English and send them to the Agency.

Recently we are also considering the use of magnetic tapes for the reports between the facilities and the national system. The reports from the Japanese Authority to the IAEA, i.e. ICR etc., are now being transmitted by magnetic tapes.

Unfortunately, our domestic legal system is still maintaining the old tradition, namely any reports submitted to our Government must be in the form of written documents appropriately signed and stamped. It should be in Japanese. Therefore, we invented a compromise way of merging the Japanese requirements and decreasing our effort of translation. By our instruction, the facility operators send us the material accountancy reports on forms specified by the national system. These forms include both the Japanese and English language. The forms are quite similar with the IAEA's formats, but a few columns are added. Added columns are used exclusively for our domestic usage (Figures 4-7).

Next, I would like to explain some special topics associated with the reports system. The first topic deals with computer processing at the national system. The second topic concerns the right of suppliers, and the third topic is on minor quantity.

A. Computer Processing

As I explained before we have many reports to be processed (Figures 8-13).

Normally a reactor facility consists of one MBA, but process facilities like fuel fabrication plants are divided into 2 or more MBAs. ICR should be reported whenever inventory changes occur or are consolidated in a one month period. Therefore, enormous amounts of data must be dealt with at the national system. Also, safeguards implementation requires reports to be dispatched in a timely manner. If you recall, the reports to the IAEA need two steps, from facility to the national authority, and then from the national authority to the IAEA. You might think

that this process must be done by using a computer in order to be timely. Actually, we permitted the facility operator two weeks for processing, checking, printing etc., after closing the account period for the reporting of ICR. In some facilities, like the Japan Atomic Energy Research Institute where many independent small quantities of nuclear material are used in various laboratories at the site, this time-limit is really a heavy burden. Anyway, we have only two remaining weeks for processing of the facility's reports into ICR, MBR, or PIL for the IAEA since we have to dispatch the reports before the end of the next month when inventory changes occur. Normally if an organization like the Government Safeguards Authority needs to use a computer, the Authority ought to purchase and own it. In our case, however, we used an outside agency. We already had a non-profitable independent organization, namely the "Nuclear Material Control Center (NMCC)". We decided to utilize this organization, and by designating this center as "Designated Information Processing Organization" which was impowered by the Law for Regulation of Reactor, Nuclear Fuel Material and Nuclear Source Material, we gave the NMCC special status for processing material accountancy data. Then, we asked NMCC to process the facility reports.

There are a few points which I should mention specifically. First, the data to be processed may contain commercially proprietary or confidential information. Therefore, we must secure confidentiality of this data during processing. We amended our Law, and included one Article, specifically asking

the persons who belong to the NMCC not to disclose any confidential information prior to release by the Government.

Second, the costs of processing data are borne exclusively by the Government. Therefore, prior to actual implementation, it was necessary to develop a computer program to deal with this task, and the costs of developing such program were also borne by the Government.

Third, I would like to explain why we asked an outside independent organization to process these data rather than equipping a computer by ourselves. This Center already had a computer and had experience associated with safeguards technology. That is one reason. Another reason is that maintenance of a computer requires not only money but also manpower. Our manpower in the Government is extremely tight and we do not want to use our precious manpower for it, if we can use outside manpower.

B. The Right of Supplier Country

This is a very important problem for the State Safeguards people, but it is not concerned with the IAEA. When we discussed and reached a conclusion on the model safeguards agreement under the NPT at the Safeguards Committee of IAEA back in 1970 - 71 in Vienna, we hoped the problem of duplicated control of supplier - or so to speak "Double Origin," could be solved. having the NPT Safeguards Agreement, as far as the Agency is concerned, solves this problem, because the model safeguards agreement called for a unified inventory irrespective of origin of nuclear material. Unfortunately, for the national authority people problems still have remained. If you had an electric utility in your country

that wanted to operate a light water nuclear power station, you would need fuel. You would have to purchase U_3O_8 (Yellow cake) somewhere and obtain enrichment work, unless you could find it domestically. In Japan's case, our electric companies purchase natural uranium in Canada and send it directly to the USA for enrichment. After enrichment, the uranium comes to Japan. Every transfer of uranium is governed by a bilateral Government-Government nuclear agreement. Thus for this transfer, Japan-US Agreement and Japan-Canada Agreement is necessary. This uranium I described just now is Canadian-origin, because it came from Canada, although not directly coming to Japan. At the same time, this nuclear material is US-origin, because it was enriched in the US and therefore it was "improved". The problem continues. The reason we are concerned about the origin, is that we have to keep the material account of the uranium on a supplier by supplier basis. Although the right of the supplier to safeguard nuclear material shipped to the recipient is transferred to the IAEA, the supplier country has another right which is not transferred to the Agency, and it may be executed sometime after the import of nuclear material. Whenever we wish to transfer the nuclear material to a third country, we must have a prior approval of the original supplier of this material (the famous MB#10 affairs). If the quantity of material is kept on a national record rather than a country-by-country basis, we do not know how much of X-country origin uranium is going to be shipped.

I would like to make one other point concerning supplier's rights. The Japan US Cooperation Agreement, states the

following: "In the event of termination of this Agreement, the supplier Government may require the receiving Government to effect the return of all special nuclear material supplied pursuant to this Agreement and still in the receiving country". Considering this right of requesting return, the National Authority must also know exactly the location of nuclear material imported from another country.

If I may continue the example which I just described to you, the enriched uranium is imported in Japan, fabricated at the fuel fabrication facility, transferred to the nuclear reactor, where it is burned and produces plutonium. The plutonium produced in this situation is of triple origin, Canadian-US-Japanese. I can continue further. From now on, my story is becoming just a possible case, but it may happen in the very near future. Plutonium produced in a Japanese Reactor is shipped in the form of spent fuel and reprocessed at a French Plant. Separated plutonium is converted and fabricated into a fuel assembly together with Australian origin uranium in a German Plant. The fabricated fuel assembly is sent back to Japan. Now how many countries can claim to be the origin country of this fuel assembly? Possibly, six countries. I can continue more but it seems a waste of time. In general, the bilateral government-government safeguards agreement contains not only the right of safeguarding but also other bilateral means of control. The right of safeguards was transferred to the IAEA, but other controls remain in the hand of the supplier which causes real difficulty. I hope that you now realize the necessity of the

country-by-country basis material accountancy by the illustration which I have given you.

Once we wanted to solve this problem at the occasion of the London Suppliers Group Meeting. However, the result of the consultation was not successful and we did not find a solution. The reason may be that this is much too detailed and technical for the people gathered at such a politico-technical meeting. Now, until we find a good solution, we must maintain our national account country-by-country basis and report to the IAEA in a unified manner. This is the reason why our format for the reports from facility to the state authority differs from ICR to the IAEA.

C. Minor Quantity

I would like to explain to you as a next topic the special problem of minor quantity. I stated that the Japanese nuclear material control is very strict and that even 1 mg of plutonium or 25 g of natural uranium must have a Government license, and an ICR from the facility to the National System. However, in the implementation of safeguards you do not worry about 25 g of natural uranium. So the reporting unit of ICRs, MBRs, etc. is one gram for special fissionable material and one kilogram for source material. You have to consider whether the objective of a State system really needs control of minor quantities of nuclear material less than the reporting unit. In my opinion the International safeguards system and material management system of the state are different. From the material management point of view, you need smaller quantities compared to the safeguards

purpose. In Japan, as you know people are quite sensitive to release of, or even existence of, any amount of nuclear material. So, for Japanese purposes, we need a limitation on smaller quantities, though it is sufficient to report to the IAEA based on the reporting unit.

VIII. INSPECTION AND VERIFICATION ACTIVITY

Japanese facilities have two different inspections related to the safeguards activity, at least conceptually, namely, the IAEA inspection and the national inspection. In order to keep the effectiveness of the system, however, we made an effort to minimize the burden to the facility operator. We coordinated our inspection activities by the following procedures. At first the Japanese Inspectorate decides the inspection schedule and informs the IAEA. Whenever possible the IAEA safeguards inspection coincide with the national inspection. When the Agency inspection takes place, a representative of the Government authority accompanies the Agency inspectors. So, if the representative of the Government has a capability to perform a national inspection, the inspections are simplified. Again I would like to repeat--national inspectors must not interfere with the activities of the Agency inspectors. So far, I can say we have maintained fairly good relationships between the two inspectorates.

I believe that the inspection itself is important but is only one part of the verification activities. The National inspectorate must consider not only an effective inspection but

more wider activities of verification. When we decided to enlarge our capability of safeguards verification we took the following measures; (a) increasing manpower for verification activities including inspectors: (b) developing and purchasing a portable and non-portable equipment for the verification activities: (c) constructing and operating a dedicated safeguard analytical laboratory: and (d) developing a MUF analysis theory.

I previously discussed the increased manpower. As for the development of instruments, we concentrated most of our efforts on the development of containment and surveillance devices, such as surveillance cameras and TV systems. We purchase portable non-destructive measurements equipment for our inspectors. Right now the Japanese inspectorate has several types of instruments, including SAM-II and so on. Also we have plastic and paper seals, but they are still conventional ones. We have had a fairly large amount of financing from our Finance Ministry for purchasing and developing the equipment, but I do not foresee this increase continuing in the budget. So, I wish one of the objectives of R & D for developing instruments to be to develop devices which are reliable, easy-to-handle and within reasonable price range, rather than to develop sophisticated and very expensive ones.

At that time, it was decided that a non-destructive instrument would be satisfactory, but that we should also have an analytical laboratory for safeguards purposes. Based on our idea, the Japanese Safeguards Analytical Laboratory (JSAL) would serve primarily for routine analysis of samples taken by national

inspectors, and would also be useful for providing assistance to the Agency, if necessary. At that time, we already had in Japan a lot of large analytical laboratories which had very good capability and performance. Tokai Laboratory of Japan Atomic Energy Research Institute, Tokai Works of Power Reactor and Nuclear Fuel Development Corporation are only two examples. But, from our viewpoint, the laboratories already established have their own nuclear material and therefore are subject to the safeguards. It is not preferable to ask to analyze the samples taken for safeguards purposes by a laboratory which is under the safeguards. Therefore, we decided to create a new analytical laboratory whose purpose would be solely safeguards. And we selected the site, got the money from the Finance Ministry, constructed the laboratory and it is now under operation.

One of the problems encountered during the operation of this analytical laboratory is transportation of the samples. From the safe transport point of view, the transport of plutonium is strictly regulated, and samples for safeguards purpose are not exempted from the regulation. I appreciated the efforts spent to solve this problem by the Japanese people, US-DOE and the IAEA. If I understand the situation correctly, however, it will take some time to solve this problem completely and it should be discussed further.

One additional comment. We asked the NMCC to operate JSAL. There was no strong reason why it was not a national laboratory. The only reason I recall is that the future increase of people

becomes very difficult if we established a national laboratory instead of an independent body, like NMCC.

Concerning MUF's analysis and associated evaluation of results of inspection, etc., I must confess that although this is the essence of our tasks, only a few experiences have been gained. Some of this is summarized in Figures 14 and 15. So far, many cases of inconsistencies in the material balance in my experiences are caused simply by miscalculation or inadequate data treatment. For detailed analysis of the results of the verification activities we need more experience and historical data. In this area the Japanese National Authority is now in the process of accumulating historical data. I am very interested to hear some thoughts on this matter from the Agency people.

IX. CONCLUSIONS

So far, I explained to you quite qualitatively my experiences when I served as a Director of Safeguards Division of Japanese Government. Detailed discussion on actual implementation of a national system especially from the facility operator's point of view, will be discussed by my respected co-lecturer Mr. Osabe of JNF. As for conclusions of my talks, I can say the following points should be very important to maintain and upgrade the level of any national system of material control.

(1) Take the necessary measures to improve the level of the national inspectors and other personnel working for this area and also maintain the highest possible level of standards of manpower.

(2) Maintain frequent contact with the IAEA and its officials and establish regular meetings on implementation and improvement of safeguards between the IAEA and State Systems. (Keeping a good relationship with the IAEA is essential, in my opinion.)

(3) Many inconsistencies of the material accountancy arise from misunderstandings of the facility operator. Since in a country it is expected that the operator's levels of understanding of safeguards implementation might differ greatly, the upgrading of the level of the operator is very important. We are organizing annually a training course for the facility operator which will be held jointly by the Safeguards Division of NSB and the NMCC. This is a very successful attempt so far and it will be continued in the future, if I understand correctly.

(4) This is the last point but not the least. I would like to stress the importance of public relations aspects. Especially troubles associated with implementation of the nuclear field stem from misunderstanding or less understanding of the people concerned. When we want to improve our own system, normally we need money, and/or manpower. To obtain more financing and manpower, we have to make an effort to seek better understanding of the finance people.

A Material control system is a part of the world wide regime of non-proliferation, and since it is a very important and topical issue, the press people are very

interested. Without a good understanding of the situation, a wrong article in the newspaper might cause serious trouble.

In many occasions, it is important to make an effort to give a better understanding to the people concerned. Without the cooperation of many people, a good system can not be established and maintained.

Fig. 1 Administrative Structure of Regulatory Activities on Nuclear Energy in Japan

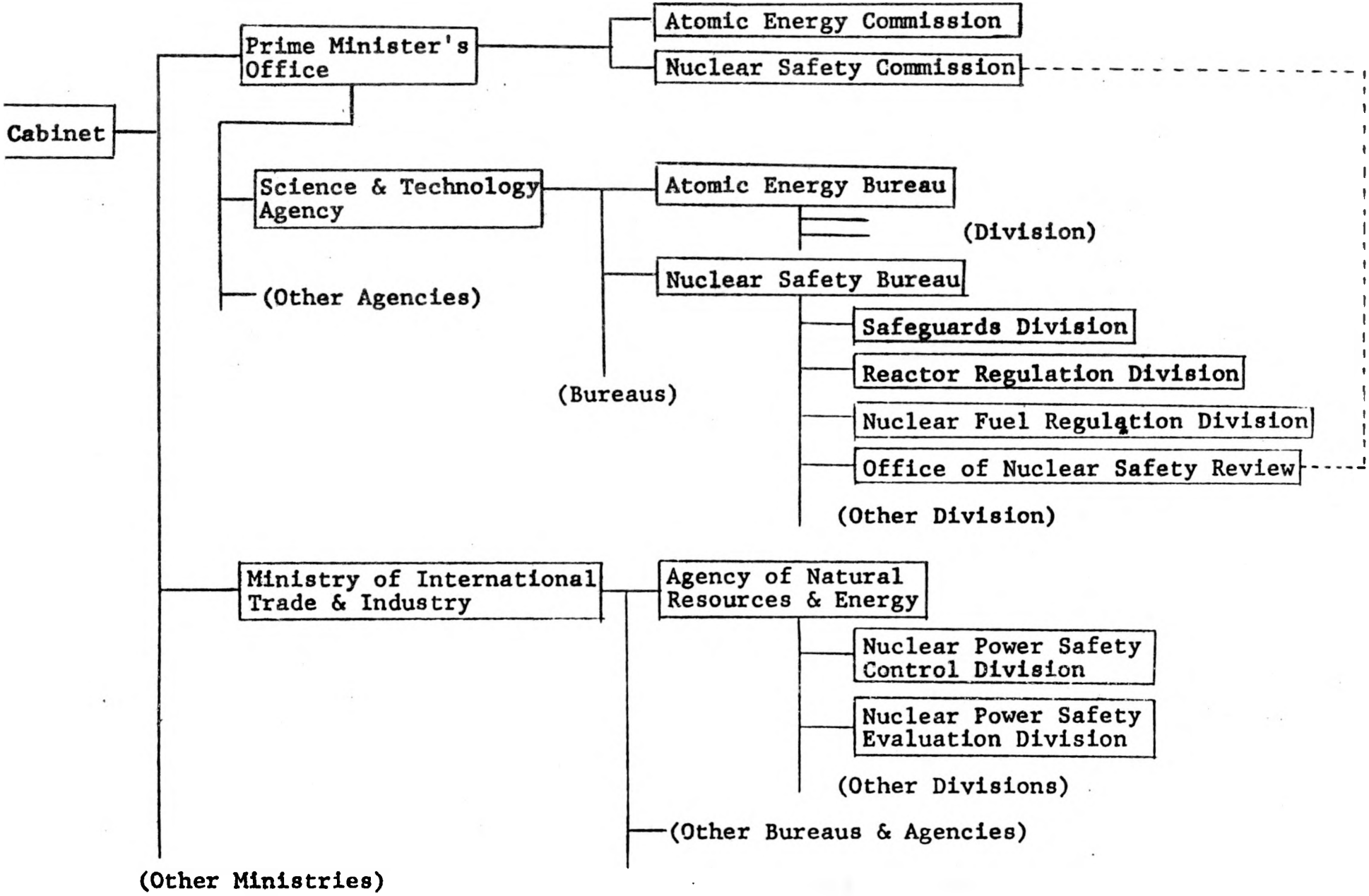


Fig. 2 Upgrading of Government Organization in Connection with Nuclear Safeguards

	Organization in the Government	International Activities and Domestic Regulation
.968		Japanese Government signed on the NPT (June) The NPT was opened for ratification (July)
.970	Safeguards Office was established in Internation Cooperation Division, AEB, STA (March)	IAEA Safeguards Committee model safeguards Agreement was concluded (1970.6 - 1971. 3)
.972		Informal negotiations of Japan-IAEA safeguards Agreement has started (June)
.975		It was concluded and initialed (February)
.976	Safeguards Office was transferred from International Cooperation Division of the AEB to Nuclear Fuel Regulation Division of the NSB (February)	Formal negotiation of the Agreement has started (June) Japan ratified the NPT (June)
.977	Safeguards Division was created in the NSB (October)	Japanese Government sent a bill for amendments of the Law for Regulation of Nuclear Reactors, Nuclear Fuel Materials and Nuclear Source Materials to the Japanese Diet (Parliament)(Feb.) It passed the Diet (October) The Safeguards Agreement was entry into the force (December)
1978		Subsidiary Arrangements and Facility Attachments were entry into the forc

Fig. 3 Flows of the Reports and Inspections

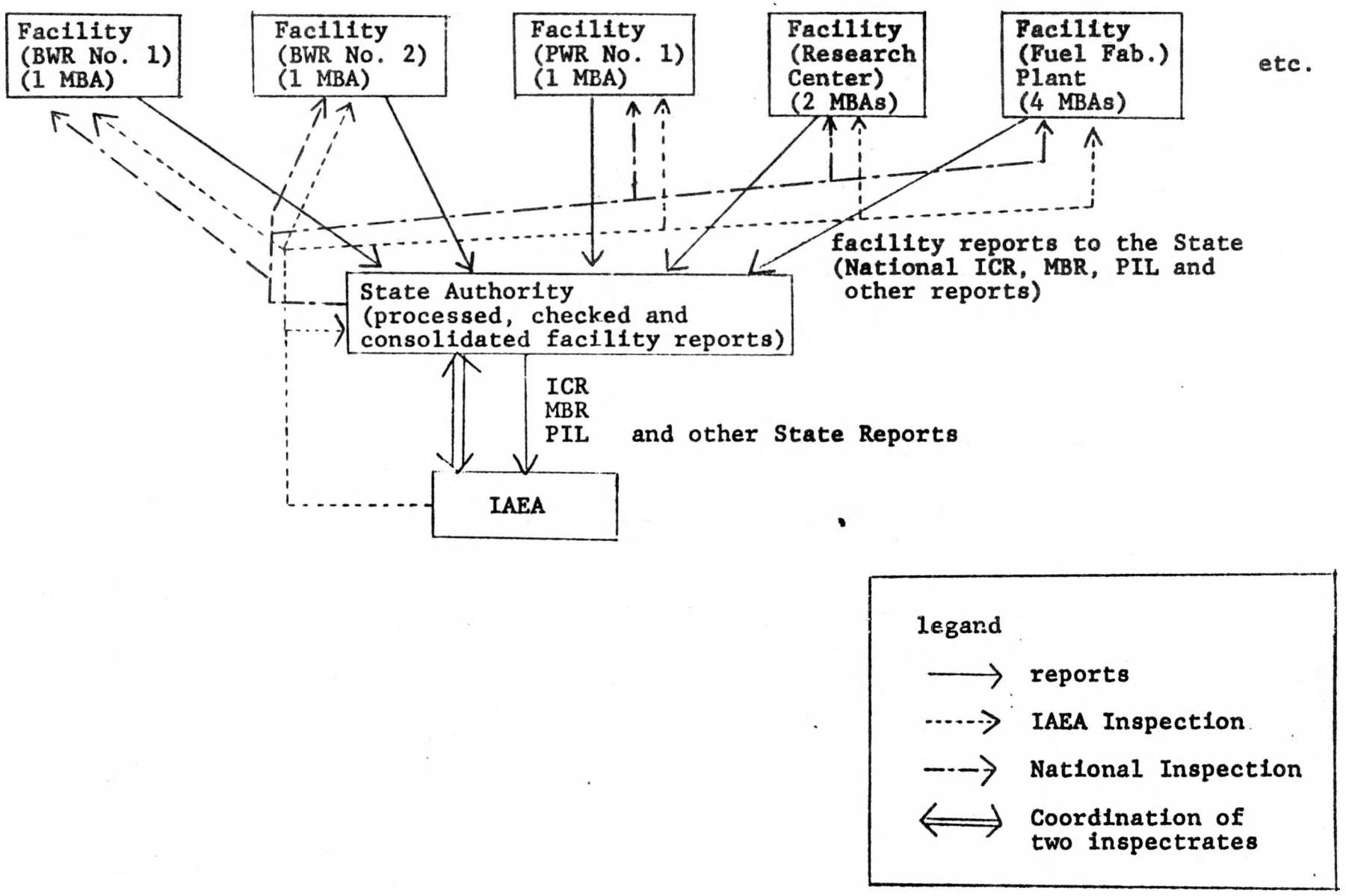


Fig. 4 Data Sheet of Inventory Change Report

在庫変動報告 (ICR)

INVENTORY CHANGE REPORT

様式 R01.1/J

原子力事業者又は使用者 名称 .... 所在地 .... 施設名 .... 物質収支区域名	報告期間 ...... 年 ...... 月 ...... 日から ...... 年 ...... 月 ...... 日まで 報告番号 .... 提出者名 ....

原子力事業者等コード ORGANIZATION 1–4	施設コード FACILITY 5–8	物質収支区域コード MBA 9–12	報告期間 年 月 日から 年 月 日まで PERIOD COVERED BY REPORT 13 FROM 18	19 TO 24	報告番号 REPORT No. 25–28	エントリー行数 No. OF LINE ENTRIES: 計量データ ACCOUNTANCY DATA 29 30	同位体データ ISOTOPE DATA 31 32	注釈データ CONCISE NOTE DATA 33 34		提出者氏名 SIGNATURE 63–77

物質収支区域コード MBA 1–4	報告番号 REPORT No. 5–8	エントリー番号 ENTRY No. 9 10	データ継続コード CONTINUATION 11	在庫変動年月日 DATE OF INVENTORY CHANGE 12–17	相手側物質収支区域又は国別コード MBA/COUNTRY FROM 18–21	MBA/COUNTRY TO 22–25	在庫変動種類コード TYPE OF INVENTORY CHANGE 26–29	在庫変動コード KMP 30	バッチ名又は番号 NAME OR NUMBER OF BATCH 31–38	バッチ内品目数 NUMBER OF ITEMS IN BATCH 39–42	物質記述コード MATERIAL DESCRIPTION 43–46	計量データ ACCOUNTANCY DATA: 供給国コード ORIGIN OF MATERIAL 47–50	元素コード ELEMENT 51	元素重量 WEIGHT OF ELEMENT 53–60	単位 unit 61	核分裂性物質重量 WEIGHT OF FISSILE ISOTOPES 63–70	同位体コード 71	測定ベースコード MEASUR. BASIS 72	注釈コード CONCISE NOTE 73	データ修正 報告番号 REPORT No. 74–77	エントリー番号 ENTRY No. 78 79
1 4	5 8	9 10	11	12 17	18 21	22 25	26 29	30	31 38	39 42	43 46	47 50	51	53 60	61	63 70	71	72	73	74 77	78 79

物質収支区域コード (   ) 枚中 枚

Fig. 5 Data sheet of Physical Inventory Listing

実在庫明細表 (PIL)

PHYSICAL INVENTORY LISTING

様式 RC2/J

事業者又は使用者
名称 ....
所在地 ....
施設名 ....
物質収支区域名 ....

棚卸し実施日 ....
報告番号 ....
提出者名 ....

事業者又は使用者コード ORGANIZATION 1–4	施設コード FACILITY 5–8	物質収支区域コード MBA 9–12	(X)	棚卸し実施日 DATE OF PIL 19–24	報告番号 REPORT No. 25–28	エントリー行数 No. OF LINE ENTRIES: 計量データ ACCOUNTANCY DATA 29–30	エントリー行数 No. OF LINE ENTRIES: 注釈データ CONCISE NOTE DATA 33–34	提出者氏名 SIGNATURE 63–77

物質収支区域コード MBA 1–4	報告番号 REPORT No. 5–8	エントリー番号 ENTRY No. 9–10	データ種類コード 11	主要測定点コード KMP 30	バッチ名又は番号 NAME OR NUMBER OF BATCH 31–38	バッチ単位数量 NUMBER OF ITEMS IN BATCH 39–42	物質記述コード MATERIAL DESCRIPTION 43–46	計量データ ACCOUNTANCY DATA: 供給国コード ORIGIN OF MATERIAL 47–50	元素コード ELEMENT 51	元素重量 WEIGHT OF ELEMENT 53–60	単位 UNIT 61	核分裂性核種重量 WEIGHT OF FISSILE ISOTOPES 63–70	同位体コード ISOTOPE CODE 71	測定ベースコード MEAS. BASIS 72	注釈コード CONCISE NOTE 73	データ修正 CORRECTION TO: 報告番号 REPORT No. 74–77	データ修正 CORRECTION TO: エントリー番号 ENTRY No. 78–79

物質収支区域コード (　　) 頁中 頁

Fig. 6 Data Sheet of Material Balance Report

物質収支報告 (MBR)

MATERIAL BALANCE REPORT

ORGANIZATION 1–4	FACILITY 5–8	MBA 9–12	PERIOD COVERED BY REPORT 13 FROM 18	19 TO 24	REPORT No. 25–28	29 30	33 34	SIGNATURE 63–77

MBA 1–4	REPORT No. 5–8	ENTRY No. 9 10	CONTINUATION 11	ENTRY NAME 26–29	ACCOUNTANCY DATA: BATCH OF MATERIAL 47–50	ELEMENT 51	WEIGHT OF ELEMENT 53–60	UNIT 61	WEIGHT OF FISSILE ISOTOPES 63–70	ISOTOPE CODE 71	CONCISE NOTE 73	CORRECTION TO: REPORT No. 74–77	ENTRY No. 78 79

1 4 5 8 9 10 11 26 29 47 50 51 53 60 61 63 70 71 73 74 77 78 79

Fig. 7 Data Sheet of Concise Note

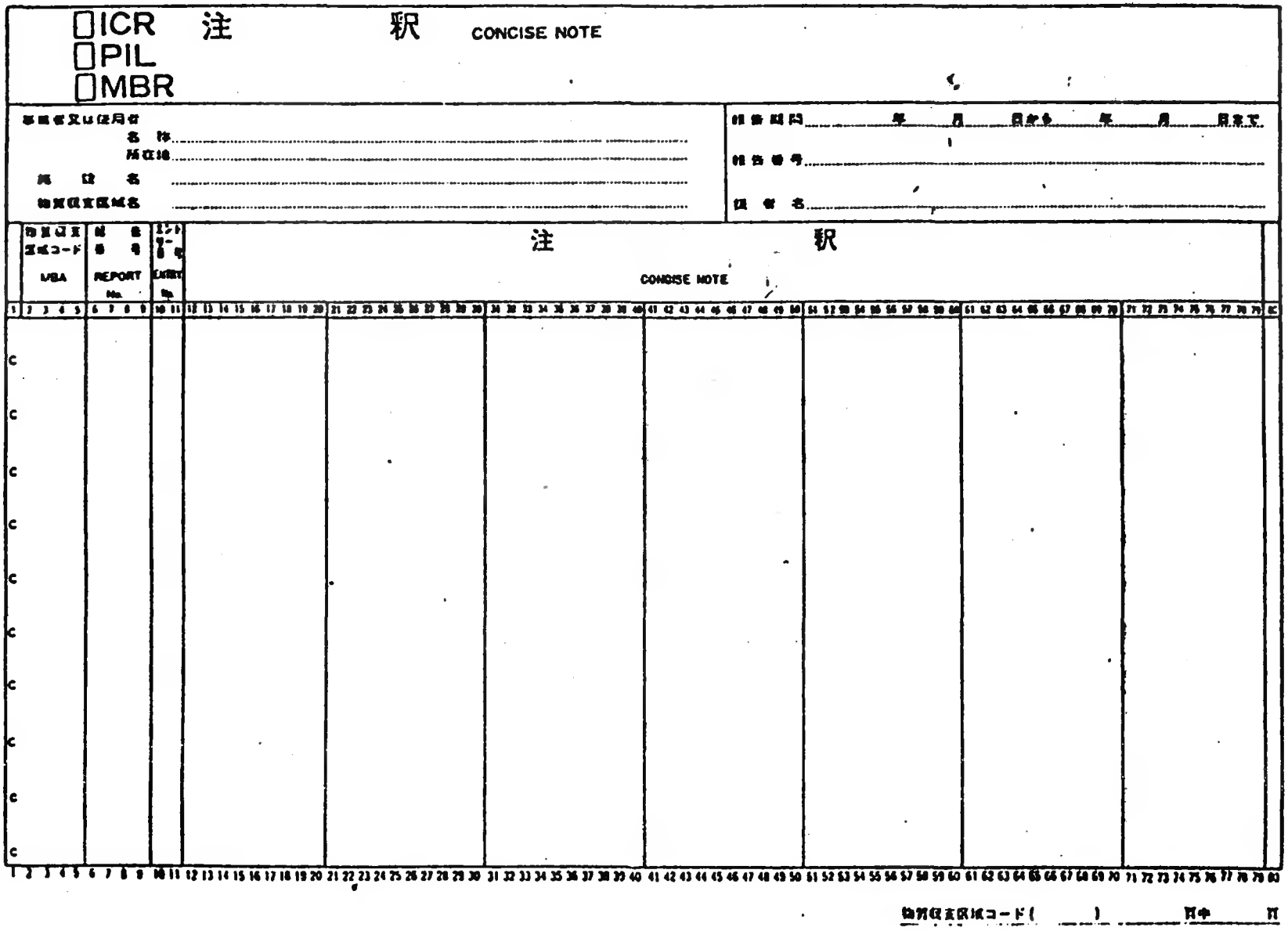
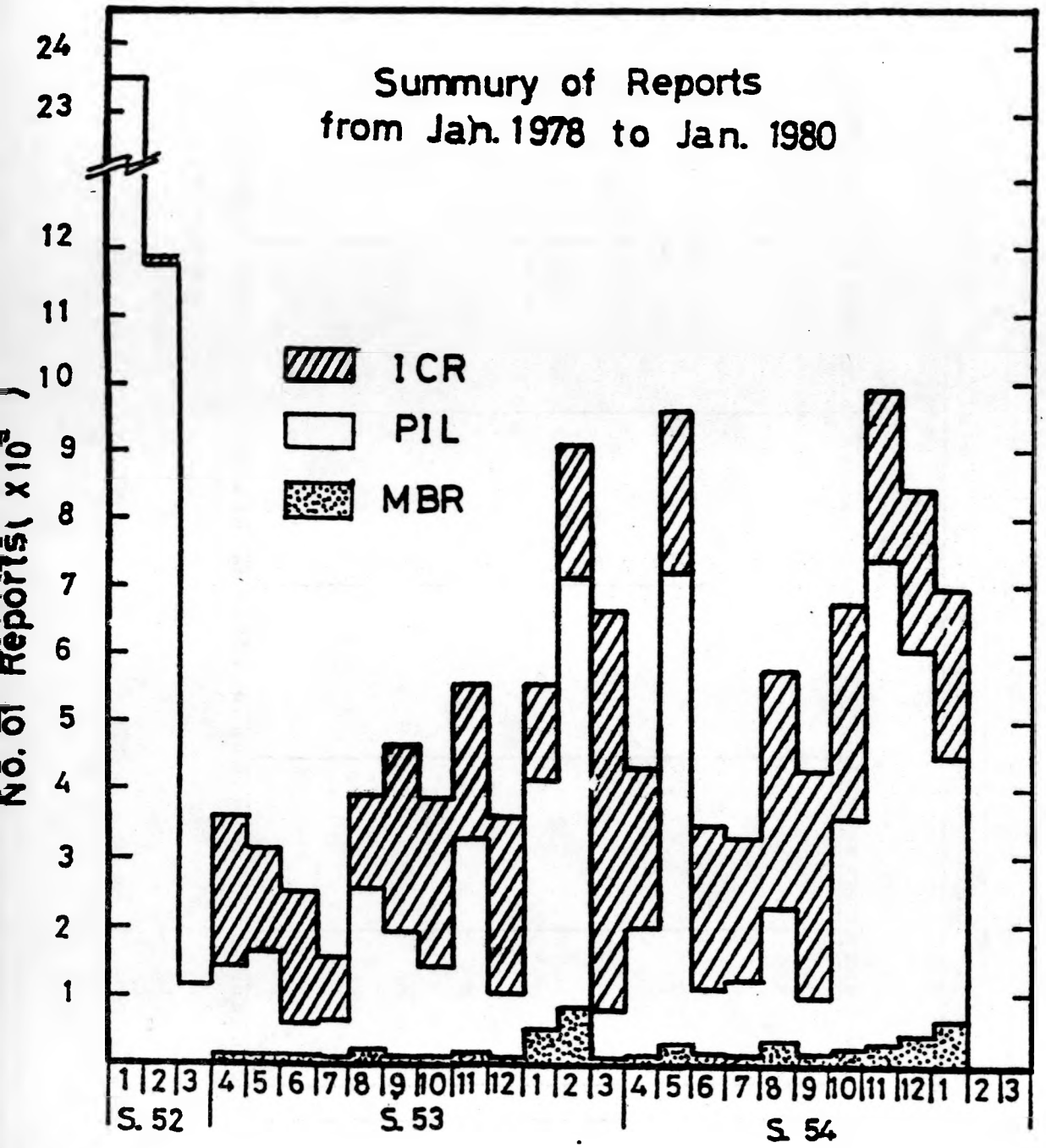


Fig. 8


Summury of Reports
from Jan. 1978 to Jan. 1980
ICR
PIL
MBR
No. of Reports (X10³)
24
23
12
11
10
9
8
7
6
5
4
3
2
1
1 2 3 4 5 6 7 8 9 10 11 12 1 2 3 4 5 6 7 8 9 10 11 12 1 2 3
S. 52
S. 53
S. 54

[illegible] のグラフ

(1980.1月現在のMBA数です。) No.8

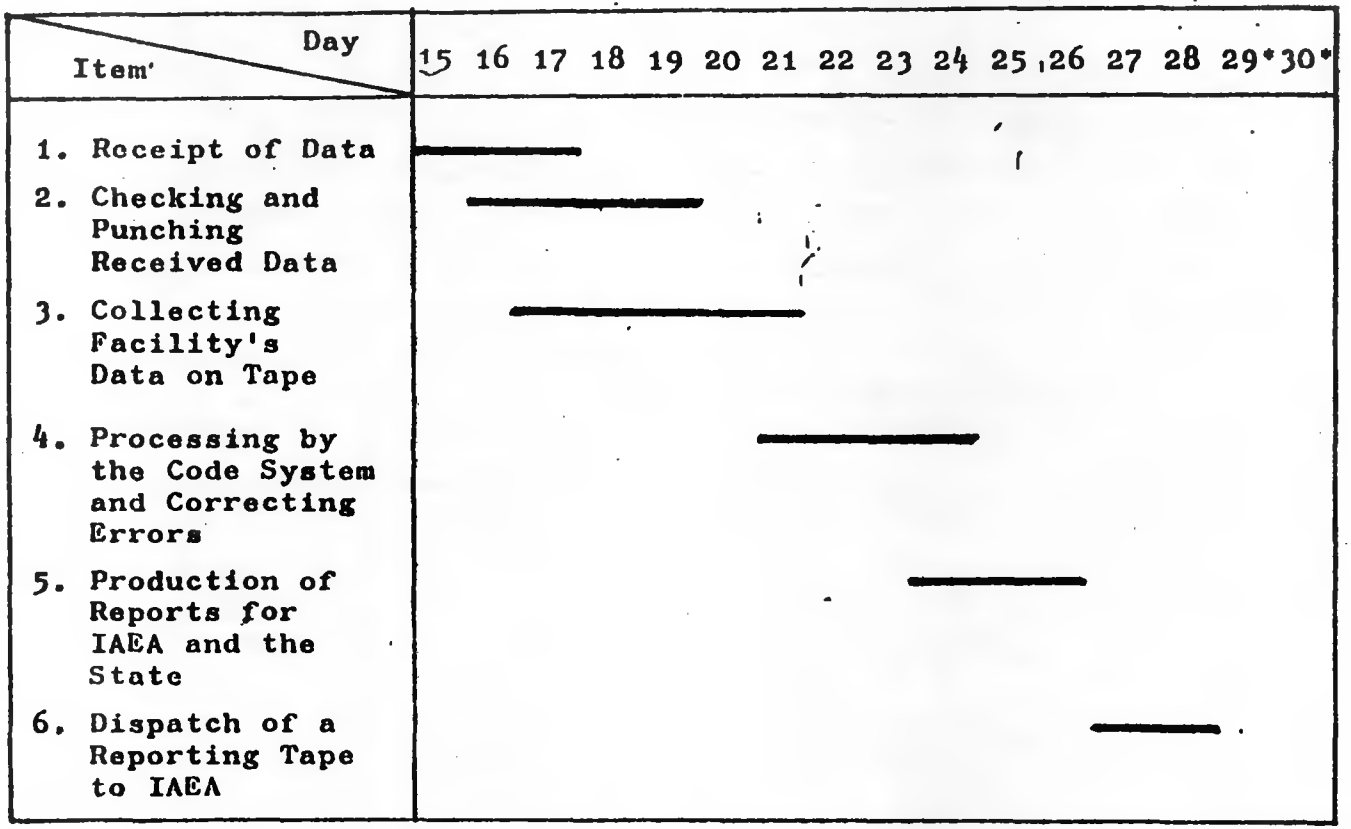
Table 1

Number of Nuclear Facilities and MBAs in Japan (as of January, 1980)

	Organization	Facility*	MBA
Electric Power	8	35	35
JAERI	1	13	13
PNC	1	5	22
Universities	15	31	32
Fabrication	4	10	23
Other Uses	89	113	113
Total	118	207	238

* The number of facilities is calculated after the provisions of the Control of Nuclear Reactors and Nuclear Materials Act. This defenition is defferent from that of the NPT Safeguards Agreements.

Fig. 9 Normal Monthly Schedule for Processing of the Accounting Data at the NMCC



* Reserved for Week End or Holiday

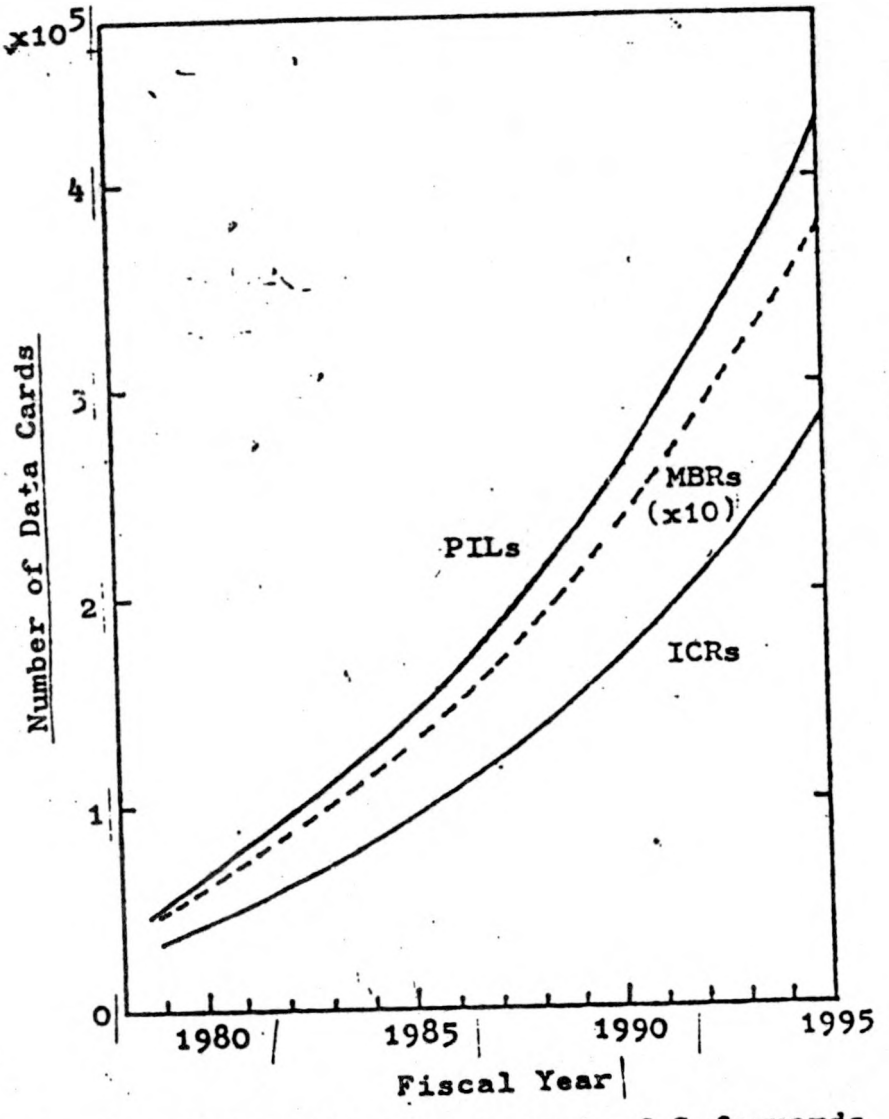


Fig. 10 Estimation for Amount of Safeguards Accounting Data (based on a Report Issued by the Bureau of Resources and Enegy)

Table 2. Amount of Nuclear Materials in Japan (June, 1978)

	Natural Uranium (Kg)	Depleted Uranium (Kg)	Enriched Uranium		Thorium (Kg)	Plutonium (g)	U-233 (g)
			U(g)	U-235(g)			
Refining	8,993	0	0	0	0	0	0
Reactor	278,641	130,254	2,076,673,055	51,058,440	1,567	2,989,417	0
Fabrication	16,097	19,851	464,481,311	12,063,744	0	1	0
Reprocessing	216	6,641	35,886,741	409,262	0	182,657	3
Other use of Nuclear Materials	90,271	20,577	5,836,143	233,485	6,392	101,982	22
Total	394,218	177,323	2,582,877,250	63,764,931*	7,959	3,274,057	25

	Natural Uranium (%)	Depleted Uranium (%)	Enriched Uranium		Thorium (%)	Plutonium (%)	U-233 (%)
			U(%)	U-235(%)			
Refining	2.28	0	0	0	0	0	0
Reactor	70.68	73.46	80.40	80.07	19.67	91.31	0
Fabrication	4.08	11.19	17.98	18.92	0	0	0
Reprocessing	0.05	3.75	1.39	0.64	0	5.58	12
Other use of Nuclear Materials	22.90	11.60	0.23	0.37	80.31	3.11	88
Total	100*	100	100	100	100*	100	100

* Error due to rounding off

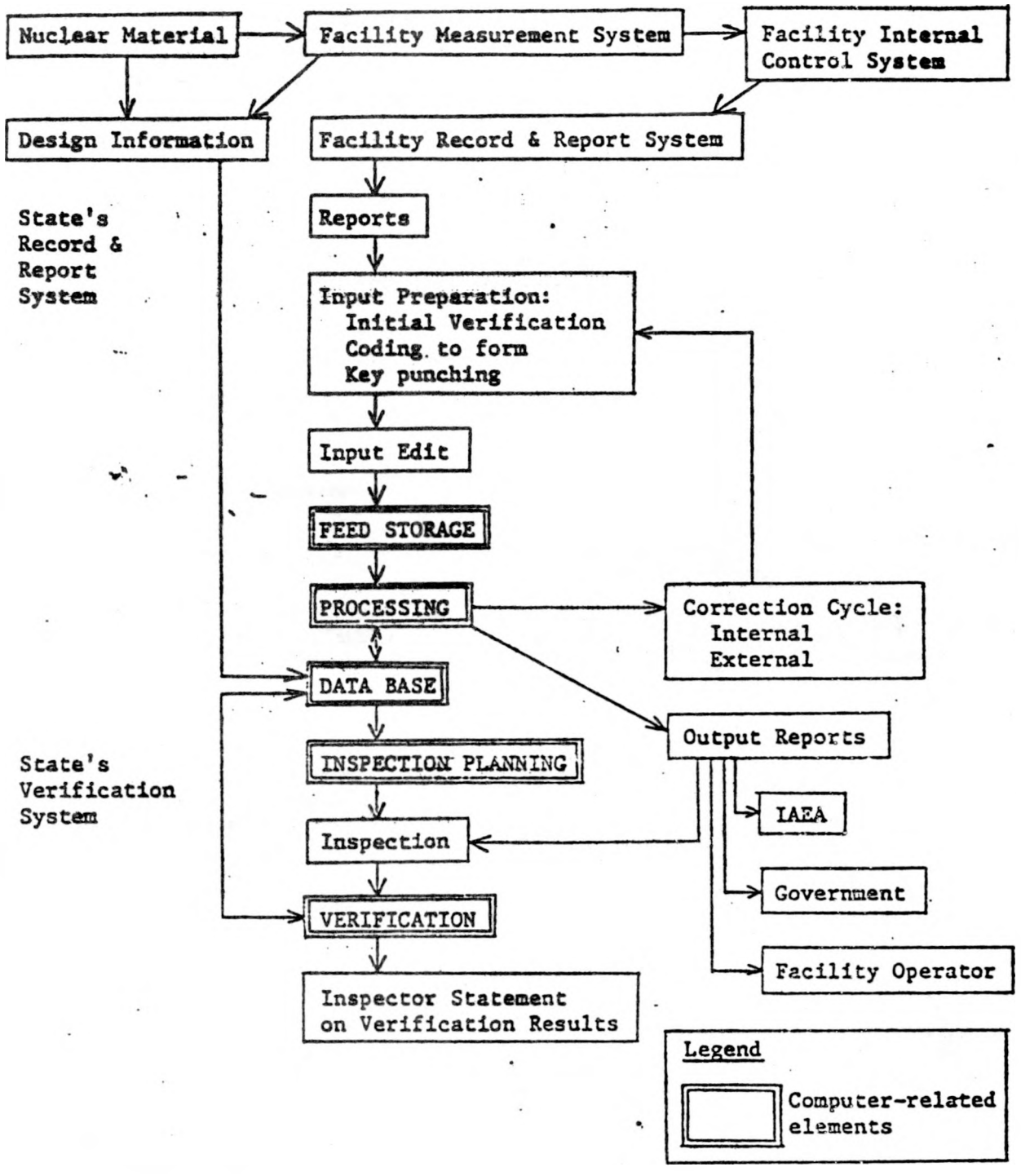


Fig. 11 Safeguards Information Flow in the State System

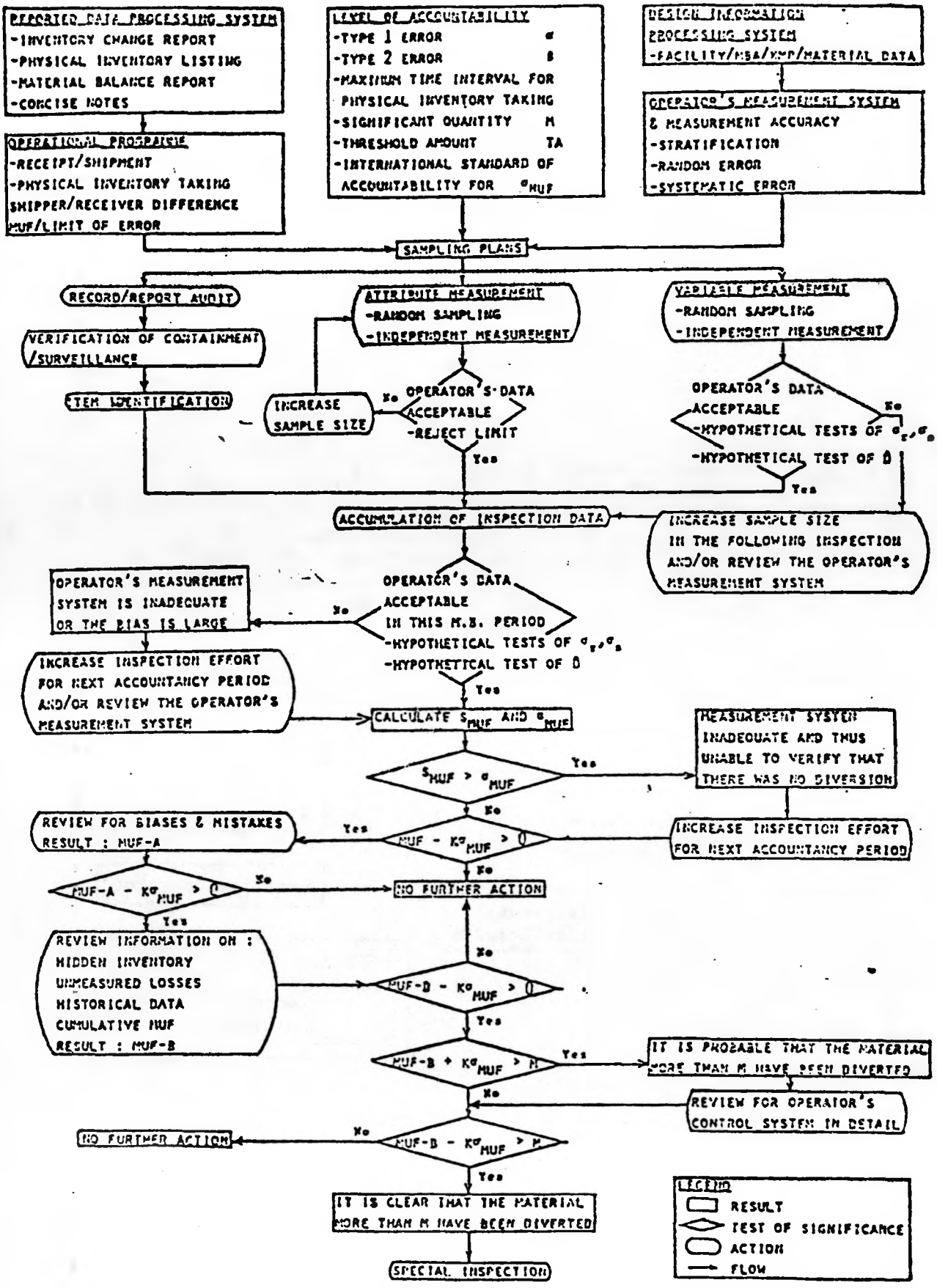


Fig. 12 Evaluation of MUF for Single MBA / PIT Interval

Table 3 Figures of Error Occurred in Processing of Accounting Report Presented from Facility Operaters

*1

E R R O R / W A R N I N G P R O F I L E (78/01)

ERROR TOTAL.....2627
WARNING TOTAL...6051

*1. ONLY POWER REACTOR

DISTRIBUTION OF ERRORS (EXCL. ZERO COUNTS)

ERROR NO.	COUNTS	% OF TOT
25	3	0.11
36	116	4.42
88	6	0.23
92	1645	62.62
108	311	11.84
111	543	20.67

DISTRIBUTION OF WARNINGS (EXCL. ZERO COUNTS)

WARNG NO.	COUNTS	% OF TOT
63	396	6.54
73	639	10.56
78	22	0.36
80	20	0.33
88	61	1.01
92	1872	30.94
94	153	2.53
113	2888	47.73

E R R O R / W A R N I N G P R O F I L E (78/02)

ERROR TOTAL.....1178

WARNING TOTAL...1121

DISTRIBUTION OF ERRORS (EXCL. ZERO COUNTS)

ERROR NO.	COUNTS	% OF TOT
5	2	0.17
20	15	1.27
30	1	0.08
55	29	2.46
56	2	0.17
80	49	4.16
88	1	0.08
92	343	29.12
95	1	0.08
108	79	6.71
109	133	11.29
111	459	38.96
177	17	1.44

DISTRIBUTION OF WARNING (EXCL. ZERO COUNTS)

WARNG NO.	COUNTS	% OF TOT
73	180	16.06
78	11	1.07
88	70	6.24
92	305	32.56
94	141	12.58
113	353	31.49

E R R O R / W A R N I N G P R O F I L E (78/05)

ERROR TOTAL.....2786
WARNING TOTAL...2275

DISTRIBUTION OF ERRORS (EXCL. ZERO COUNTS)

ERROR NO.	COUNTS	% OF TOT
5	3	0.11
30	24	0.86
51	2	0.07
55	50	1.79
80	14	0.50
81	2	0.07
88	12	0.43
92	764	27.42
95	4	0.14
97	143	5.13
108	176	6.32
109	280	10.05
111	770	27.64
135	27	0.97
140	1	0.04
161	1	0.04
168	1	0.04
175	62	2.23
176	10	0.36
177	16	0.57
178	43	1.53
179	70	2.51
180	24	0.86
182	286	10.27
221	1	0.04

DISTRIBUTION OF WARNING (EXCL. ZERO COUNTS)

WARNG NO.	COUNTS	% OF TOT
63	22	0.97
73	205	9.01
78	41	1.80
80	28	1.23
88	60	2.64
92	865	38.02
94	175	7.69
113	863	37.93
130	1	0.04
133	1	0.04
134	5	0.22
135	1	0.04
136	3	0.13
137	5	0.22

SESSION 29b: EXAMPLE OF OPERATING STATE SYSTEM (APPLICATIONS TO FUEL FAB PLANT)

T. Osabe
Japan Nuclear Fuel Co., Ltd.

I. INTRODUCTION

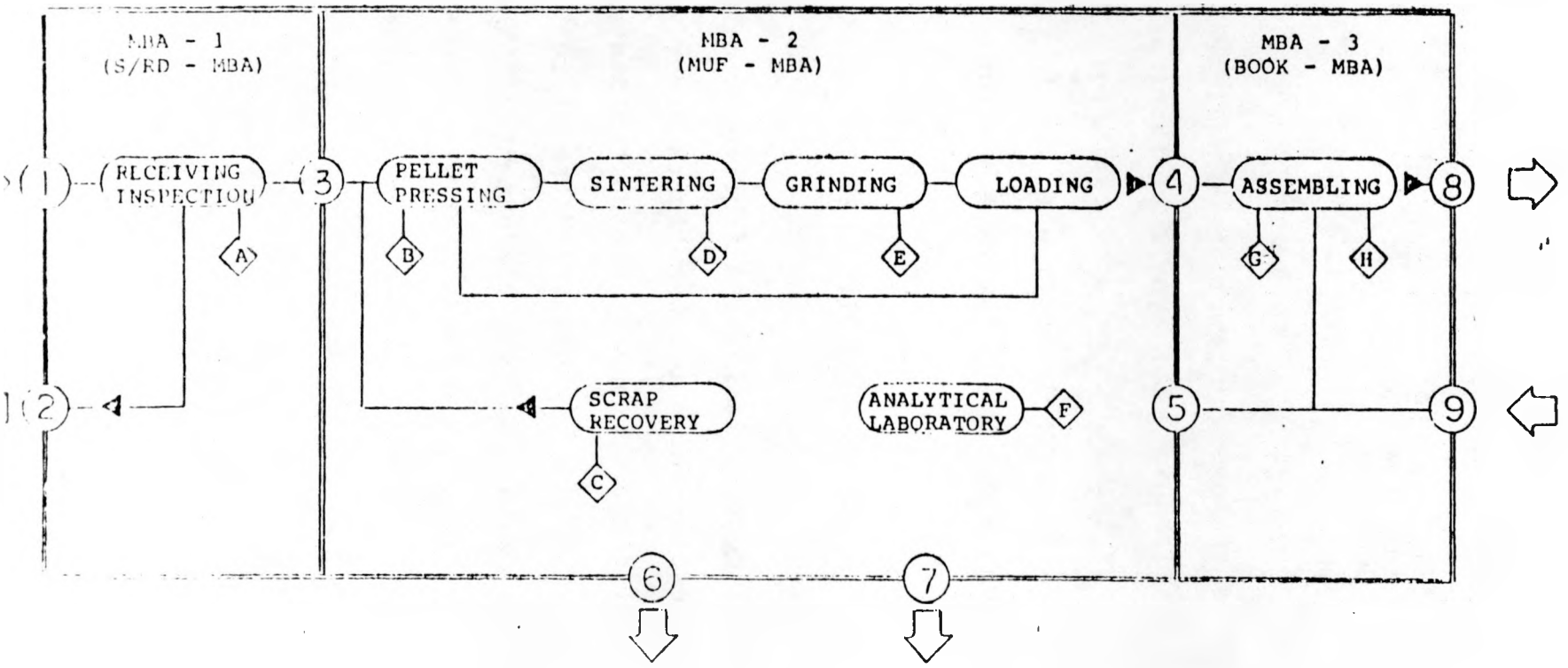
This lecture describes the current practice on nuclear material accountability utilized in a BWR type fuel fabrication facility in Japan and the current status of application of national and international safeguards inspection to the facility. Various problems being encountered by the facility and the inspection party are discussed.

II. FACILITY'S NUCLEAR MATERIALS ACCOUNTABILITY SYSTEM

A. Outline of the Facility

The Japan Nuclear Fuel Company (JNF) is a fabricator of BWR type nuclear fuel for commercial nuclear power plants under license from the General Electric Company of the United States. This facility has no UF_6/UO_2 conversion plant and therefore uranium dioxide powder (UO_2) with a maximum enrichment of 4% of U-235 is delivered to the facility as a feed material. The plant of this facility consists of six major manufacturing processes as shown in Figure 1: such as Pellet Pressing, Sintering, Grinding, Pellet Loading, Fuel Assembling, and Scrap Recovery.

MATERIAL FLOW AND MBA/KMP



FLOW KMP

(1) EXTERNAL RECEIPT
(2) EXTERNAL SHIPMENT
(3) DETERMINATION OF SRD
(4) ROD LOADING
(5) ROD UNLOADING
(6) EXTERNAL SHIPMENT
(7) DISCARD
(8) PRODUCTS SHIPMENT
(9) PRODUCT RETURNING

INVENTORY KMP

(A) FEED STORAGE (IN JM1A)
(B) FEED STORAGE (IN JM2A)
(C) SCRAP STORAGE
(D) GREEN PELLET STORAGE
(E) SINTERD PELLET STORAGE
(F) LABORATORY SAMPLE
(G) FUEL ROD STORAGE
(H) FUEL BUNDLE STORAGE

PROCESS

FLOW KMP

INVENTORY KMP

Figure - 1

B. Policy on Design of Accountability System

The design objective of the facility's accountability system may be broadly divided into the following two aspects. One is to have the accountability system contribute to the nuclear material inventory control, material balance determination, manufacturing process control, quality control, safety control, physical protection, and other managerial operations of the facility. The other is to have this system satisfy the legal requirements for national and international safeguards. The above two aspects regarding the function of the accountability system are interdependent and the system must be so designed that information necessary for the managerial operation and for the legal requirements is readily available. Figure 2 shows an example of the conceptual design of the accountability system.

C. Material Balance Area

In accordance with national and international safeguards requirements, the low enriched uranium fuel fabrication facility in Japan is required to establish at least three MBAs for accounting and controlling of the materials in the facility. As shown in Figure 1, JNF facility is divided into three MBAs: (1) Shipper-Receiver Difference MBAs, (2) a MUF MBA in which MUF will be generated and (3) Book MBA in which all materials will be accounted for by the measured value in the preceeding MBA.

From the safeguards point of view the MBA can be defined as a functional area and not as a specific area separated by any physical barrier or building structure.

CONCEPTUAL DESIGN OF FACILITY SAFEGUARDS

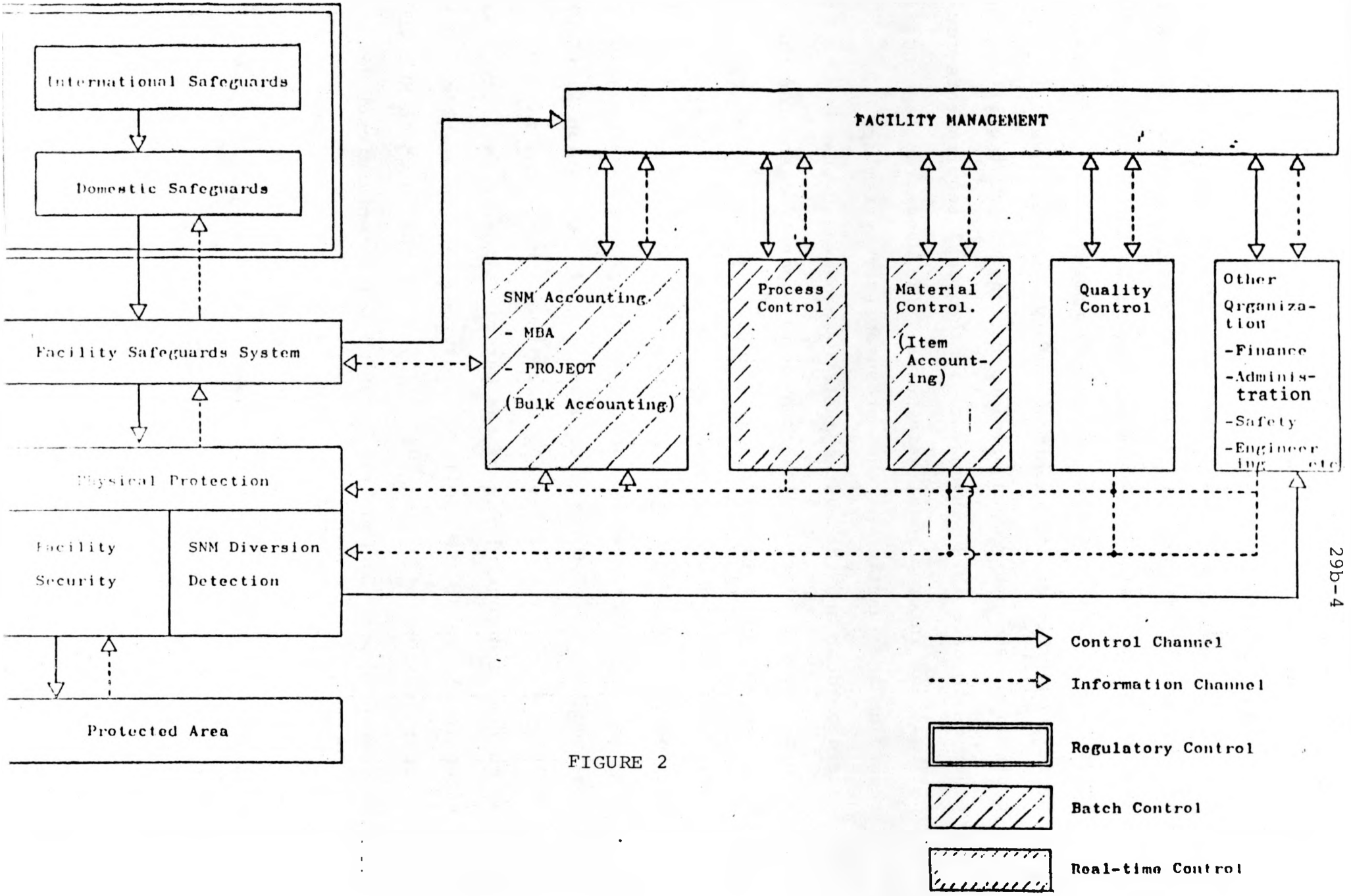


FIGURE 2

MBA-1: Shipper-Receiver Difference Area. This MBA includes all the nuclear material that is kept on shipper's data.

MBA-2: This MBA includes the fuel fabrication process up to pellet loading, the chemical laboratory and storage of intermediate materials.

MBA-3: This MBA includes the fuel bundle assembling process and storage of fuel rods and products kept on the basis of the facility's own measurements that were performed previously at MBA-2.

D. Key Measurement Point

Strategic points that serve as key measurement points are to be established for determination of material flow and inventory. At the JNF facility there are nine KMPs for determination of material flow at the boundary of MBAs which relate to inventory changes of the MBAs and eight KMPs to determine the inventory of each stratum which is classified by chemical and physical configuration of the material. (Refer to Figure 1.)

Flow KMPs (1-9) and INVENTORY KMPs (A-H)

KMP-1: Receipt of external nuclear material into MBA-1.

KMP-2: Shipment from MBA-1 to a destination outside the facility.

KMP-3: Shipment of nuclear material from MBA-1 to MBA-2 and determination of S/R Differences. This KMP also is used for receipt of nuclear material at MBA-2 from MBA-1.

KMP-4: Shipment of loaded fuel rods from MBA-2 to MBA-3.

KMP-5: Reshipment of loaded fuel rods from MBA-3 to MBA-2.

KMP-6: Shipment of various materials from MBA-2.

KMP-7: Measured discard and retained waste.

KMP-8: Shipment of final products from MBA-3 to outside of the facility.

KMP-9: Receipt of fuel assembly.

KMP-A: Storage of feed material kept according to shipper's data.

KMP-B: Storage of feed material kept on the basis of facility's measurement.

KMP-C: Storage of recoverable scrap.

KMP-D: Storage of green pellets.

KMP-E: Storage of sintered pellets.

KMP-F: Storage of various analytical samples.

KMP-G: Storage of fuel rods.

KMP-H: Storage of fuel assemblies.

E. Material Balance Accounting

The material balance accounting for each MBA shall be accomplished by determining changes in material inventory with such methods as item counting, weighing, volume measurement, sampling and analysis at the KMP's and by accounting through computerized data processing system. This system consists of four sub-systems as described below.

1. Feed Material and Scrap Control System (FASCS). This system is designed to maintain inventory control, calculation and statistical evaluation of shipper/receiver differences for both

feed material and recoverable scrap material. This system also provides an itemized listing for the purpose of taking the physical inventory.

2. Bundle Assembling Control System (BACS). This system is designed to control the accountability information regarding the fuel rod and fuel bundle. The calculation of uranium and isotopic weight for each fuel bundle and preparation of shipping document for the products are also made through this system. The system can provide an itemized list of fuel rods and fuel bundles for taking the physical inventory.

3. Safeguards Information System (SIS). This system is programmed to generate various regulatory reports such as ICR, PIL, and MBR as needed.

4. Project Accountability System (PAS). This system is to control and maintain material balance accounting for specific project material. The system is to provide project material accountability report for the project and maintain perpetual inventory for the project material.

The Data Transaction Diagram of this material balance accounting system is shown in Figure 3.

F. Measurement System

Various measurement methods for determination of special nuclear materials for each of the flow and inventory key measurement points are established in consideration of chemical and physical characteristics of the nuclear materials. The material descriptions and measurement methods are shown in Table I.

JNF'S SNM ACCOUNTABILITY SYSTEM DATA TRANSACTION DIAGRAM

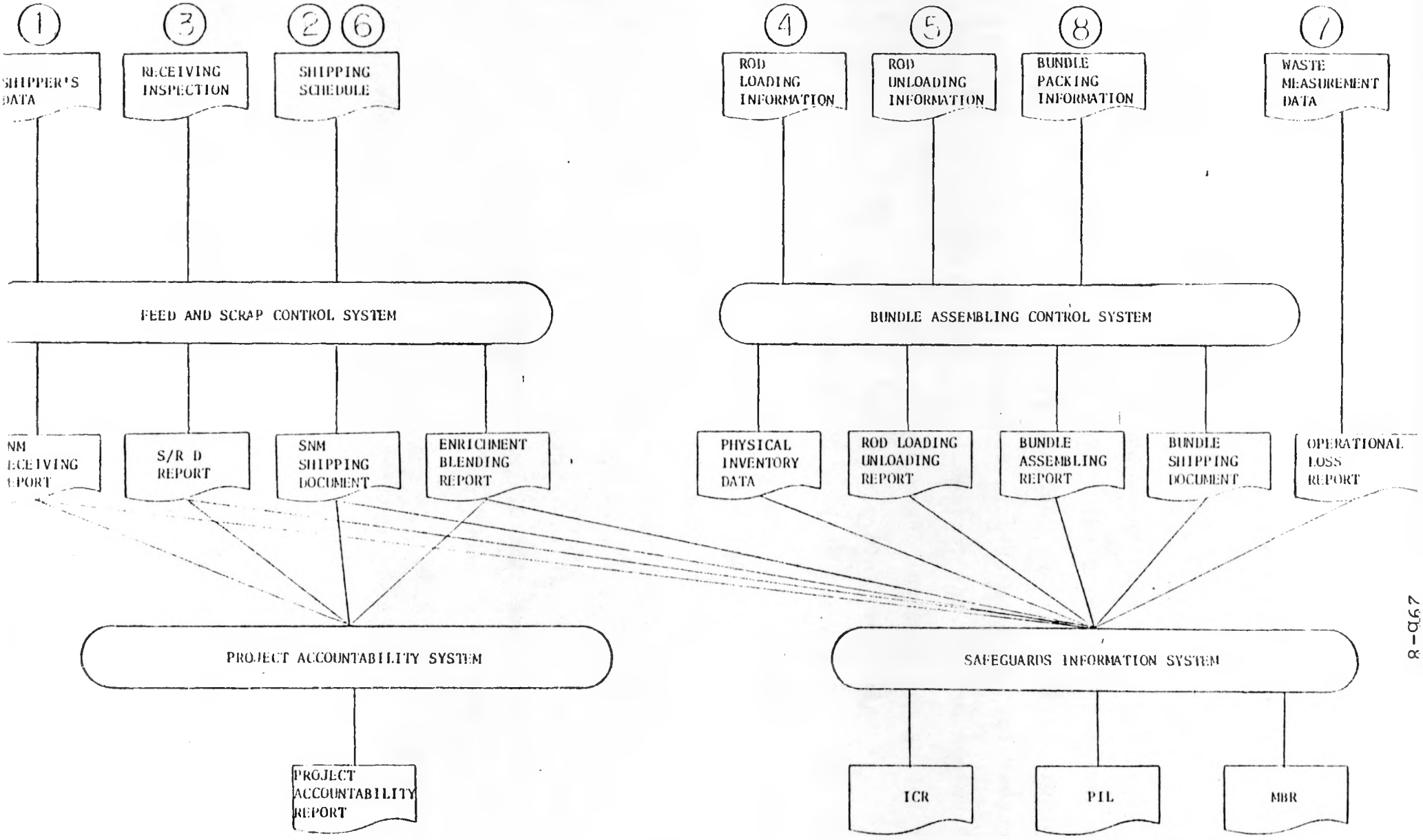


FIGURE 3

◯ : KEY MEASUREMENT POINT

8-967

KMP	Description of Nuclear Fuel Material				Type of Measurement	Method of Measurement/Analysis or Equipment
	Type	Chemical Form	Physical Form	Subject		
	Feed material	UO_2	Powder	BU-Type shipping container	Item count	
	Retained waste	Various	Various	Drum, Filter, etc.	Item count	
	Nuclear fuel material other than feed material	UO_2 U_3O_8	Powder Pellet Sludge Others	Shipping containe and others	Item count	
		Same as KMP-1 above				
	Feed material	UO_2	Powder	5 Gal. can	Weight U-w/o U235-w/o	Scale Oxidation method Enrichment analyzer-∫measurement
	Nuclear fuel material Other than feed material	UO_2 U_3O_8	Powder Pellet Sludge Others	5 Gal. can 2.5 Gal. can Others	Weight U-w/o U235-w/o	Scale (exclude less than 10g-U235) Oxidation or titration method Enrichment analyzer-∫measurement
	Fuel rod	UO_2	Pellet	Fuel rod	Weight U-w/o U-235-w/o	Scale Oxidation method Enrichment analyzer- measurement
	Fuel rod	Same as KMP-4 above				
	Nuclear fuel material other than fuel rod and fuel bundle	UO_2 U_3O_8 UO_4	Powder Pellet Sludge Others	5 Gal. can 2.5 Gal. can Others	Weight U-w/o U235-w/o	Scale (Except less than 10g-U235) Oxidation or titration method Enrichment analyzer-∫measurement
	Exhaust loss	-----	-----	-----	U-Concentration	∫Scintillation counter
	Sewage loss	-----	-----	-----	U-Concentration	∫Scintillation counter
	Retained waste	UO_2 U_3O_8 UO_4	Powder Others	50 ℓ or 200 ℓ drum Used filter	U235-quantity	measurement by SAM-II

Table - 1 (1/2)

Measurement/Analysis at Each KMP

KMP	Description of Nuclear Fuel Material				Type of Measurement	Method of Measurement/Analysis or Equipment
	Type	Chemical Form	Physical Form	Subject		
3	Fuel rod or fuel bundle	UO_2	Pellet	Fuel rod	Weight U-weight U235-weight	Measured value at KMP-4 above
	Fuel rod or fuel bundle	Same as KMP-8 above				
	Feed material (by shipper's data)	UO_2	Powder	DU-Type shipping container	Item count	
	Feed material (by JNF's data)	UO_2	Powder	5 Gal. can	Weight U-w/o U235-w/o	By transfer card Measured value at KMP-3 above
	Scrap	UO_2 U_3O_8 UO_4	Powder Pellet Sludge	5 Gal. can or 2.5 Gal. can	Weight U-w/o U235-w/o	By transfer card Oxidation or titration method Enrichment analyzer measurement
	Green pellet	UO_2	Pellet	Metal pellet boat	Same as KMP-C above	
	Sintered pellet	UO_2	Pellet	Metal boat or tray	Same as KMP-C above	
	Various lab. sample	UO_2 U_3O_8 UO_4	Various	Various	Weight U-w/o U-235-w/o	By record Average U-w/o Average enrichment or actual measured value
	Fuel rod	UO_2	Pellet	Fuel rod	Item count	
	Fuel bundle	UO_2	Pellet	Fuel bundle	Item count	

Table - 1 (2/2)

1. Measurement Methods.

a. Mass Measurements. Weight measurement at KMPs are performed with electronic scales with digital display of the weight value. The range of these scales is from 10 kg to 50 kg with divisions ranging from 1 gram to 20 grams. The scale is selected depending upon the weight of the items to be weighed.

b. Analytical Measurements.

Percent Uranium

- Dichromate Titration - This type of determination is based on the techniques devised by Davies and Gray which allow the determination of uranium in dilute nitrate solution and in the presence of a large quantity of impurities.
- Gravimetric Determination of Uranium - This technique is used for relatively pure uranium compounds and is based on oxidation of the sample to U_3O_8. The final value is then corrected for non-volatile impurities.

Enrichment

- Gamma Spectrometry. This technique is used for determination of the percent of Uranium-235. Samples are converted into relatively pure U_3O_8 to make their geometry constant.

Impurities

- Trace metallic impurities are determined using the standard emission spectrographic technique.

Nuclear Poison (Gd_2O_3)

- nuclear poison as an additive in fuel is determined using an energy-dispersive X-Ray fluorescence technique.

c. Non-destructive Measurements

- Alpha Counting is employed for measurement of uranium in atmospheric discharge and effluent discharged to the sewer.
- Passive Gamma Counting (SAM-II) is employed for counting containers of waste and used filters which are stored as retained waste.

2. Measurement Control Program.

a. Weight Measurement Control. All scales at KMPs will be checked daily for zero setting and calibrations with standard weights. In addition, the scales will be checked and calibrated once per every month with the first class standard weights by personnel who are qualified as measurers by the national government. The standard weights will be inspected by the Inspection Institute of Weights and Measures.

b. Analytical Measurement Control.

- Uranium content measurement.

  Analytical reagents for measurement are qualified with the national standard. Analytical balances will be calibrated once every six months.

- Enrichment measurement.

  Gamma spectrometry equipment is calibrated with the national standards. The equipment is calibrated at the beginning of each shift. The working standards are analyzed after every eleven samples and if the average of three readings is out of control limits, the equipment will be recalibrated.

c. Nondestructive Measurement Control. Calibration standards of this nature are not available either from national or international sources. The calibration is performed once every month with a known standard gamma source that is prepared by the facility.

3. Laboratory Correlation Program. As part of the measurement control program, this facility has participated in the Safeguards Analytical Laboratory Evaluation (SALE) Program.

Also various laboratory correlation programs are being conducted between related facilities.

G. Physical Inventory

The purpose of taking a physical inventory is to determine the quantities of nuclear materials on hand at a given time within a material balance area and to derive the differences between the book inventory and physical inventory that are called Book Physical Inventory Differences (BPID) or Materials Unaccounted For (MUF). The MUF is a very important figure both for plant management and safeguards because the MUF gives a useful indication of the effectiveness of the facility's nuclear materials accountability system. It is also useful to indicate no significant loss of nuclear material and no diversion of nuclear material. In order to meet the safeguards requirements, the physical inventory must be taken twice a year. The inventory frequency can be reduced when the annual throughput is less than 300 tons of uranium and when the safeguards authority has continued assurance that the plant material balance is closed with limits of error of MUF of not more than 0.3% relative. The

requirements further demand that the physical inventory must be conducted under the complete shutdown status of the process and all material movement which might change the inventory balance of each MBAs must be ceased after the book inventory cut-off for the inventory. In addition to this complete physical inventory, interim inventory will be taken upon completion of each fuel fabrication project to determine the material balance for the project accounting.

Inventory Procedure (For complete inventory). The physical inventory will consist of four major parts:

1. Equipment clean out and in-process inventory determination.

   All process equipment and systems containing nuclear material are thoroughly cleaned to minimize hidden inventory and equipment hold-up. However, in the case of the equipment or a system that cannot be disassembled for technical or economic reasons, the equipment hold-up will be estimated by means of appropriate NDA equipment or past experience.

2. Inventory item count

   This portion of the inventory will involve item identification and accounting for all nuclear materials. With respect to discrete items visually located, their item identification number, project number, enrichment, material type, and gross, tare and net weights will be recorded.

3. Weight verification

In order to test the gross weight assigned to inventory items, randomly selected containers are re-weighed, and if only systematic bias is detected throughout this examination, the tag weight will be corrected.

4. Analytical verification

A statistically based sampling plan is developed for various types of recoverable scrap to reconfirm the applicability of the standard uranium contents for each type of recoverable scrap.

The standard sequence of events for physical inventory is shown in Figure 4.

H. Records and Reports

Records and reports for accountability and safeguards purposes can be categorized as follows:

1. Accounting Records. Four major types of accounting records are maintained by the facility;

a. For Inventory Changes. Record all external shipments and receipts, material transferred between MBAs within the facility, measured discards, retained waste, accidental loss or gain, and all information concerning changes in the MBA inventory.

b. For physical inventory. Record all information used for determination of ending physical inventory, including sampling and analytical results, weight verification data, etc.

c. Adjustment and correction. Record any shipper/receiver differences and MUFs as adjustment and corrections due to

STANDARD SEQUENCE OF EVENTS FOR PHYSICAL INVENTORY

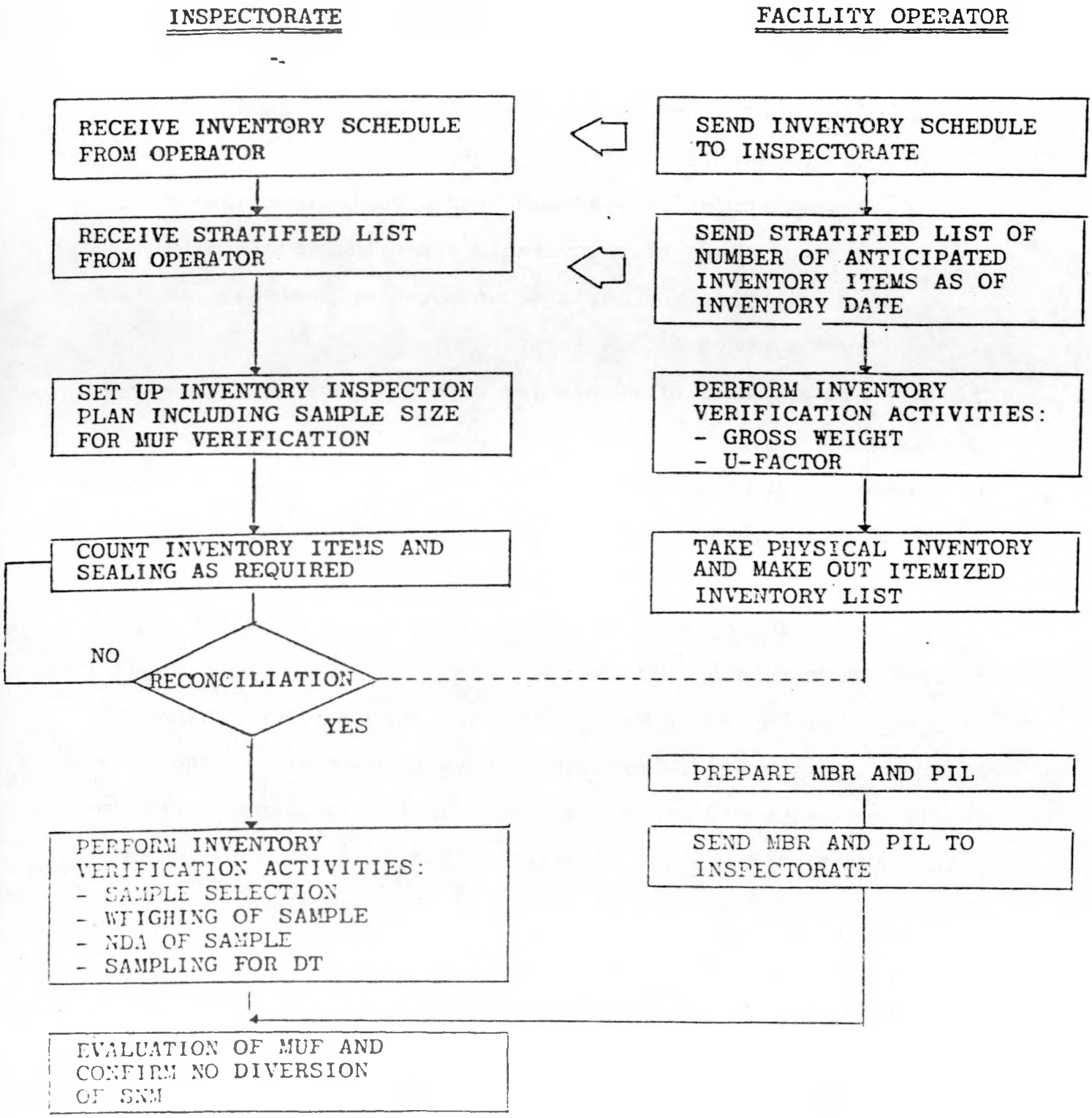


FIGURE 4

detection of errors in previous records or due to more precise later measurements, and corrections for measurement bias.

d. Changes in batch identities. Where a batch identification is changed, its previous batch identification and new batch identification must be recorded with traceability.

2. Operating Records. At least 6 types of operating record are to be maintained in accordance with regulations:

a. Rod loading operation. All accountancy data relevant to determination of the uranium and isotope weight for each fuel rod are recorded.

b. Bundle assembling operation. All the relevant data for the fuel rods assembled into each fuel bundle and the uranium and isotope weight for each fuel bundle are recorded.

c. Removal of seal or equipment. Whenever a facility operator removes a seal which has been installed by a safeguards official for any safeguards purpose, the date, seal identification number and the reason for removal are recorded.

d. Enrichment blending operation. Whenever enrichment blending is performed, accountancy data on the original materials used for blending including the name of the country of origin and on the material created by the enrichment blending are recorded.

e. Accident that results in loss or gain. For accidental losses or gains of nuclear material, information relevant to the accident including date, cause and features of the accident, and estimated or known amount of nuclear material which has been lost or gained is recorded.

f. Measurement control. With respect to measurement equipment and instruments, all relevant data for the facility measurement control program used for determination of random and systematic errors in each inventory change are recorded.

3. Regulatory Reports. Regulatory reports are required in connection with paragraphs 59 to 69 of the NPT safeguards Agreements. The specific requirements on reports are stipulated in Code 10 of the Subsidiary Arrangements and Facility Attachment. These reports are:

a. Inventory Change Report (ICR). This report is used to report all inventory changes of MBA including changes of batch identification and those due to blending, adjustment and corrections. The report must be submitted to the government office within 15 days after the end of the month in which the inventory changes occur.

b. Material Balance Report (MBR). This report is used to report the material balance of each MBA for the period between two physical inventories. The report must be prepared for each type of nuclear material for which the facility keeps a separate account and submitted to the government office within 15 days after the completion of the inventory.

c. Physical Inventory Listing (PIL). This report shall be attached to each MBR. All accountancy data for each batch of physical inventory must be entered.

d. Concise Note. This note shall be attached to ICR, MBR, and PIL to explain any unusual inventory changes or corrections to their previous reports respectively.

e. Special Report. This report must be prepared whenever any operational losses exceed the allowable limits or any other circumstance which might affect the safeguards measures occurs.

4. Project Accountability Report. This report is required to be in accordance with the fuel fabrication contract. The report is prepared upon completion of each fuel fabrication project to determine project material balance accounting and to assure operational losses have not exceeded the allowance which is stipulated in the contract.

III. CURRENT STATUS IN APPLICATION OF SAFEGUARDS INSPECTION TO THE FACILITY

Application of safeguards to the facility is performed in accordance with national and international safeguards requirements under NPT safeguards. The basic concept for implementation of the NPT safeguards in the facility is that the IAEA shall utilize the national safeguards system as much as possible in verifying that there has been no diversion of nuclear material from its peaceful use. However, IAEA can utilize direct inspection of the facility within the safeguards agreement in order to perform independent measurements and observations of the nuclear material in the facility.

A. Pre-Administrative Arrangements

Certain administrative arrangements are to be completed prior to the application of safeguards to the facility, in accordance with the national and international safeguards requirements.

First, the facility's Regulation for Nuclear Materials Accountability, defining the appropriate accountability system, must be submitted to the Prime Minister for his authorization in accordance with the "Law for Control of Nuclear Source Material, Nuclear Fuel Material and Nuclear Reactors." At the same time, the Design Information will be submitted to the IAEA as an informative document to establish the inspection strategy for detection of any diversion, including allowable limits of error of MUF for the facility. The Design Information is also utilized to complete the Facility Attachment which contains more detailed arrangements for the application of safeguards by the IAEA. Upon completion of the Facility Attachment the facility then submits the initial report to the IAEA. IAEA then performs an ad hoc inspection to confirm that the facility's accountability system is as stated in the design information.

B. Inspection

The inspection activities and the normal frequency of inspection are stipulated in the Facility Attachment. The IAEA inspection of the facility is usually performed together with Japanese government inspectors and the IAEA inspectors in accordance with the NPT safeguards agreement between the Japanese government and the IAEA.

The inspection mode can be categorized as flow verification and inventory (MUF) verification. Twelve flow verifications and one inventory verification were performed during 1979, with a total inspection effort of 78 man-days.

C. Flow Verification

The scope of flow verification activities are summarized below:

- Examination of records on verification for self consistency and consistency with the reports which were previously submitted to the safeguards authorities. This includes source data examination.
- Item identification, counting and measurement
- Calibration of measurement equipment used for accountability
- Verification of the quality of facility's measurements
- Taking representative analytical sample
- Flow verification of nuclear material at flow KMPs
- Application, examination, removal and renewal of seals
- Servicing and review of surveillance equipment

D. Inventory Verification

Normally, three days of plant shutdown are required for inventory verification activities carried out by the government inspectors and the Agency's inspectors. The inventory schedule is to be submitted to the regulatory officials a minimum of 30 days prior to the inventory date and the Stratified List approximately one week prior to the inventory. It is very important to discuss at this stage the details of the inventory practice of the operator's inventory and the inspector's verification plan in order to eliminate potential problems that might surface at the time of inventory. This discussion shall cover the availability of operator's man-power to assist

inspectors measurements, appropriate location for setting of inspectors measurement equipment, background of gamma rays in the measurement area, the facility's power supply voltage fluctuation which may affect the inspector's instruments, etc.
(Refer to Figure 4.)

The scope of inventory verification activities which are stipulated in the Facility Attachment is as follows:

- Verification of the operator's physical inventory taking for completeness and accuracy
- Weighing of containers with nuclear material on the basis of a random sampling plan
- Taking accountability samples
- Identification and counting of fuel assemblies and the use of NDA techniques
- Use of in-line NDA systems
- Application, examination, removal and renewal of seals
- Servicing and review of surveillance equipment

The inspector's sampling plan for inventory measurement will be established for two types of measurement methods. One is an instrumental method for quick detection of medium size to gross discrepancies of individual items with a high degree of certainty. The other is a more accurate measurement capable of detecting small discrepancies. These two methods are referred to as the attribute method and the variable method respectively. For reference purposes the actual number of samples taken at the 1979 physical inventory at JNF facility is shown in Table 2.

1979 PHYSICAL INVENTORY VERIFICATION

KMP	STRATA	ITEM COUNTING	NUMBER OF SAMPLE	MEASUREMENT (See note below)			ANALYTICAL SAMPLE
				①	②	③	
+ B	FEED MATERIAL	100 %	29	13	3	29	16
C	RECOVERABLE SCRAP	100 %	26	0	0	16	26
D	GREEN PELLET	100 %	4	0	0	4	4
E	SINTERED PELLET	100 %	22	0	0	22	16
F	FUEL ROD	100 %	21	21	0	21	-
G	FUEL BUNDLE	100 %	13	13	0	-	-

Note: Measurement ① = By NIS 315r - Spectrometer
Measurement ② = By Multi Channel Analyzer
Measurement ③ = By SAM - II

TABLE 2

IV. DISCUSSION

In conclusion, JNF's accountability system fully satisfies the current regulatory requirements in both national and international safeguards under NPT safeguards and plant management policy.

However, it is obvious that the accountability system needs to be modified or improved in the future as required by increased plant throughput, introduction of new measurement techniques, use of computerized material control systems, etc., but we must endeavor to minimize the economic impact and eliminate degradation of effectiveness of accountability through these modifications. When we establish the future plan for a facility's accountability system, it is necessary to adequately consider the trend of international policy against nuclear proliferation and the intensity of the safeguards requirements. Furthermore, the cost effectiveness of application of the accountability system must be considered in terms of software as well as hardware. However, it is recognized that the various current problems with both the inspecting party and the facility operator must be settled first for these purposes. The major problems identified through our past experiences are enumerated below:

(1) There are some ambiguities in the criteria for application of safeguards, and, therefore, it is unclear to what extent the accountability system ought to be improved.

(2) Lack of adequate inspection methods for verification of material flow at KMPs.

(3) There are differences in the technical level (e.g., in handling of NDA equipment) among individual inspectors.

(4) There are some ambiguities in the relationship of applicable legal regulations and the safeguards requirements.

(5) There is some room for reconsideration of the reporting format, contents of report, practice for correction of previous report, etc.

INTERNATIONAL TRAINING COURSE ON NUCLEAR MATERIALS ACCOUNTABILITY FOR SAFEGUARDS PURPOSES

SESSION #30: IMPLEMENTATION OF THE FACILITY SAFEGUARDS SYSTEM

SPEAKER: Dr. James A. Powers

Teknekron, Inc.
McLean, Virginia USA

Thursday, June 5, 1980
8:30 a.m.

BIOGRAPHY

Education: West Virginia University - B.S. Chemistry; Purdue University - Ph.D Physical/Nuclear Chemistry.

Present Position: Independent Consultant; senior scientist, Teknekron Research Institute.

Present Duties: Manage and participate in nuclear safety and safeguards study projects.

Past Positions: Supervisory positions with government; 10 years nuclear safeguards regulatory programs; 10 years nuclear fuels development. Safeguards regulatory program experience includes material control and accounting and physical protection regulation setting and licensing.

Session Objectives

SESSION #30: IMPLEMENTATION OF THE FACILITY SAFEGUARDS SYSTEM

This session will consist of a detailed survey of the manpower, equipment, and funding necessary to implement an effective system of materials accountability and control in a power reactor and/or spent-fuel storage facility. Special consideration will be given to the technical background and further training required for the safeguards personnel, the types of assay equipment necessary to provide useful accountability data, and the associated capital and operating costs. In addition, the impact of the required safeguards system on overall design and operation of the facility will be discussed.

After the session, participants will be able to

1. Identify the manpower resources and training needed to carry out an effective safeguards program in a representative reference reactor/spent-fuel storage facility.

2. Specify generally the type of equipment required for the reference facility safeguards system.

3. Have an appreciation of typical capital and operating costs associated with implementing the safeguards system envisioned.

4. Discuss the facility design features that must be considered in order to integrate the safeguards system into the reference facility.

SESSION 30: IMPLEMENTATION OF THE FACILITY SAFEGUARDS SYSTEM

Charles W. Emeigh,
U.S. Nuclear Regulatory Commission

and

James A. Powers,
Teknekron Research, Inc.

I. INTRODUCTION

This paper describes the manpower, equipment and funding necessary to implement the fundamental nuclear material controls essential to an effective material control and accounting (MCA) safeguards system at:

1) a low enriched uranium fuel fabrication facility, and

2) a power reactor.

For the United States domestic nuclear fuel processing facilities, the backbone of the MCA safeguards system is defined in the Code of Federal Regulations, Title 10, ENERGY, Part 70, paragraph 70.58, Fundamental Nuclear Material Controls (FNMCs). The MCA requirements for power reactors are given in paragraph 70.51(b), (c), and (d). The basic concepts for the controls documented in these paragraphs were developed and instituted in the late 1960s. The current controls merely represent a refinement and upgrade of those basic concepts.

Also of interest in the development of MCA safeguards is the manner in which the controls were and are now being applied.

Initially, they were applied on a case-by-case basis. Beginning in mid-1974, the controls were applied across the board, and each nuclear fuel processing facility, possessing more than one effective kilogram of special nuclear material (SNM), was required to submit a plan to demonstrate how the intent of regulations would be satisfied. This same procedure is still adhered to today.

II. FUNDAMENTAL NUCLEAR MATERIAL CONTROLS

Following is a synopsis of the nine basic controls.

A. Facility Organization

1. Corporate and Site. The responsibilities for MCA functions at the corporate and site locations must be delineated.

2. Single Individual Responsible for Overall Direction. Responsibility for the MCA functions must be assigned to a single designated individual.

3. Appropriate Separation of Functions. The facility organization must provide for appropriate separation of functions to assure independence of action and objectivity of decision.

4. Responsibilities and Authorities of Different Functions. The responsibilities and authorities of all functions whose activities impact on MCA must be delineated.

5. Training. The MCA must include provisions for training, qualifying the MCA personnel as well as their periodic retraining and requalification.

B. Material Control Areas

1. Plant. For MCA purposes, a set of processes or operations coordinated into a single manufacturing, R&D or testing effort is defined as a plant.

2. Material Balance Area (MBA)/Item Control Area (ICA). Subdivisions of a plant established to enhance material control and loss localization capabilities. Custody of material within an MBA/ICA is the responsibility of a single designated individual.

C. Measurements

1. SNM Measurements. Measurements are required for all SNM received, produced, transferred between MBAs, transferred from MBAs to ICAs, on inventory, or shipped, discarded, or otherwise removed from inventory.

2. Description of Measurement System. A description of all measurement systems employed for MCA purposes must be prepared.

3. Process Measurement Points. Specific points at which MCA measurements are performed must be indicated.

4. Process Materials. A listing of all material types and the associated measurement systems must be established.

D. Measurement Control

1. Quality Assurance (QA) Program. A quality assurance program covering all MCA measurement systems must be established and maintained. The Program shall assure that all measurements are controlled, measurements are traceable to the national system of measurements, and reliable data is available for estimating bias, random errors and limits of systematic error.

2. Measurement Errors. The QA program shall assure that the magnitude of errors is controlled such that the allowable limits of paragraph 70.51(e)(5) are met.

E. Physical Inventory

1. Inventory Procedures. Detailed written procedures covering all aspects of physical inventory preparation, performance, and reconciliation shall be established and maintained.

2. Inventory Measurements. All SNM on inventory must be listed on a measured basis. Prior measurements of tampersealed containers may be accepted provided the integrity of the tamperseal has not been violated.

F. Material Accounting System

1. System Description. A centralized accounting system employing double entry bookkeeping must be established.

2. Account Structure. A system of general and subsidiary ledgers and journals should be established to cover both plant and MBA/ICA transactions.

3. Records and Reports. The accounting system shall include a system of records and reports adequate to specify the location of all SNM at any time.

G. Internal Control

1. Internal Transfers, Item Control. The MCA program shall provide for accounting and control of all transfers between internal control areas. The system shall also be capable of providing current knowledge of the identity, quantity, and location of all SNM in discrete items and containers.

2. Scrap Control. Special handling procedures shall be established to assure that scrap materials with a measurement uncertainty of $>\pm$ 10% do not remain on inventory for longer than a pre-determined period.

3. Shipments and Receipts. The MCA program shall provide for accounting and control over all shipments and receipts of SNM and for the evaluation of statistically significant shipper-receiver differences.

4. Tamperseal Program. A tamperseal program, when employed, shall provide for adequate seal control, proper application of tamperseals and verification of seal integrity at appropriate times.

H. Management

1. Procedures. Written procedures shall be established, kept up-to-date and followed for all material control and accounting functions.

2. Reviews and Audits. A formal program of reviews and audits shall be maintained to assess the adequacy of and compliance with all MCA requirements.

3. Material Balance Discrepancies. The program shall include appropriate response actions for investigating excessive material balance discrepancies.

I. Limit of Error on Inventory Difference

The program shall provide for determination of the limit of error associated with an inventory difference through propagation of errors from all components of measurement.

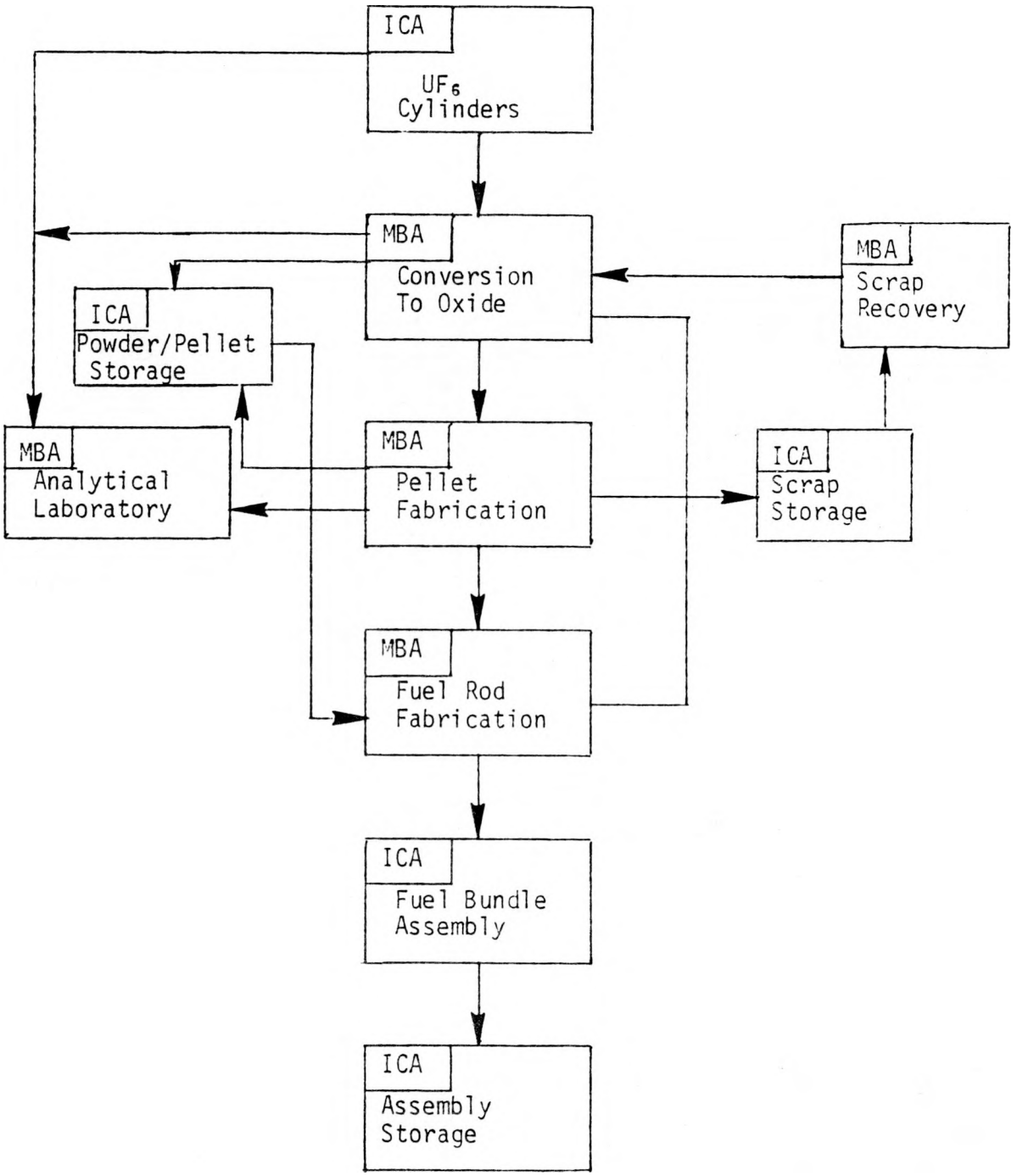
These FNMCs summarize the organizational and functional requirements of an MCA facility safeguards system. Independence of the MCA organization, responsibility, authority and staffing of qualified individuals are the essential organizational requirements. The functional requirements, in addition to identifying staffing requirements, can be used to identify equipment needs. While the MCA organization staffing requirements are specific to the facility safeguards system, the equipment can be used for other purposes such as production and quality control.

III. LOW ENRICHED URANIUM (LEU) FUEL FABRICATION

After the FNMCs, the controlling factor for the facility safeguards system is the facility itself. The operations performed in the facility determine the MBA/ICA structure, the measurement control program, quality control program and every other feature which directly or indirectly influences the safeguards program. The operations carried out at an LEU fuel fabrication plant are illustrated in Fig. 1, a simplified material flow sheet. Those areas or operations where the LEU is contained in item form are designated ICAs. MBAs are designated for those steps in the process where bulk LEU is processed. The plant organizations responsible for carrying out the safeguards duties or that frequently interact with the safeguards organization are shown in Fig. 2. The two figures will be referred to frequently to describe the implementation of safeguards at an LEU fuel fabrication plant.

FIGURE 1

SIMPLIFIED LEU FUEL FABRICATION FACILITY
FLOW DIAGRAM SHOWING MBA/ICA STRUCTURE



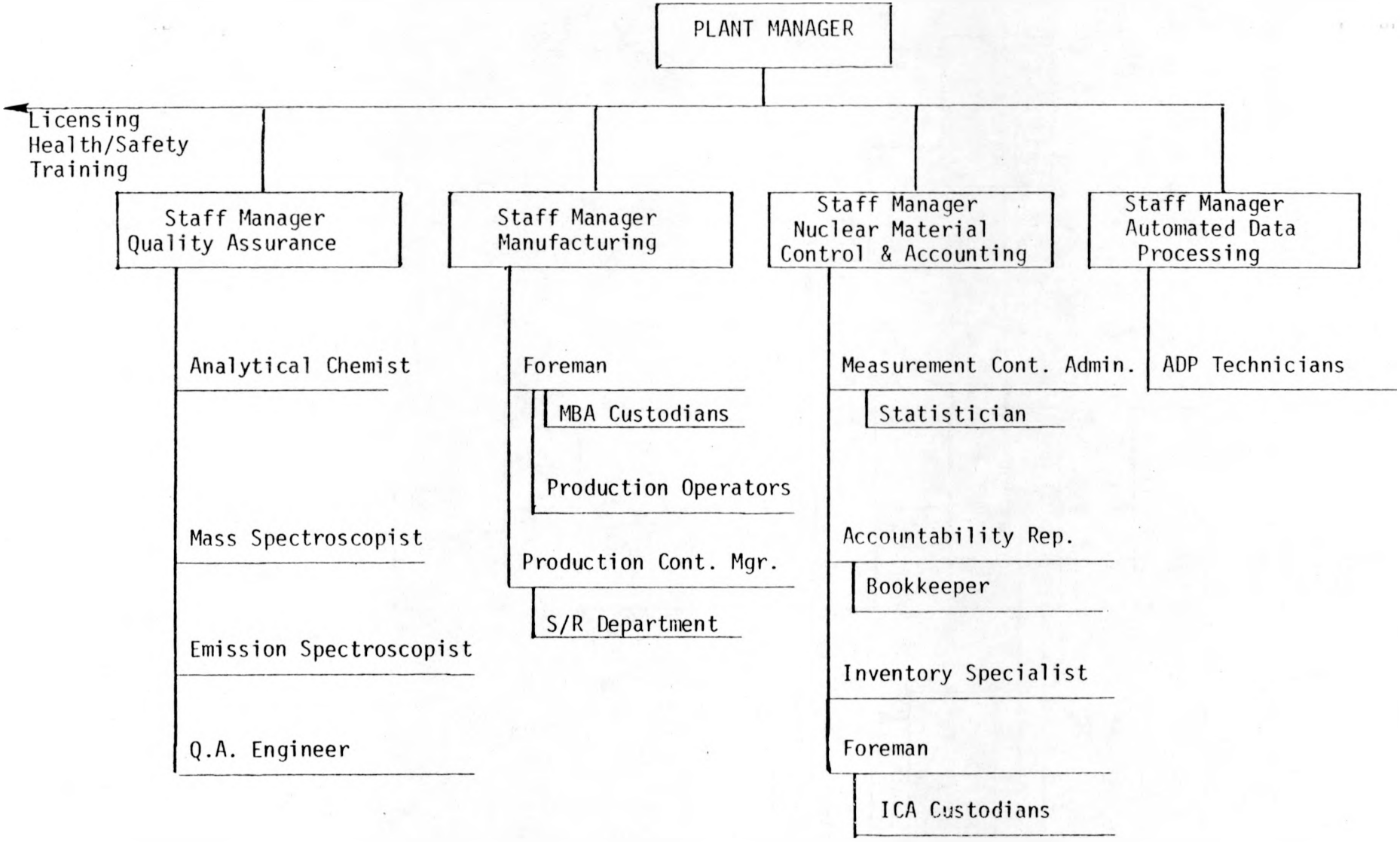


FIGURE 2

Plant Organization

The Nuclear Material Control and Accounting Department has the responsibility and authority for implementing the safeguards program. This department is supported by the Analytical Laboratory, Manufacturing, and Automated Data Processing (ADP) Departments. Table 1 lists the essential positions in the Nuclear Material Control and Accounting Department. Table 2 lists key support positions in some of the other departments. Not shown in Table 2 are the Manufacturing Department support personnel. Their position requirements for education, experience, and training are established by the Manufacturing Department.

A. Functions of the MCA Department Positions

1. Department Manager. The MCA manager has the responsibility for the overall planning, coordination, and administration of the program. In carrying out his responsibilities, he must interact with other department managers whose activities have an impact on safeguards. The interaction with the manufacturing manager is especially important since a safeguards program functions most effectively in a plant that is designed with safeguards considerations in mind. Considerations include such things as equipment layout, equipment design to minimize holdup and the MBA/ICA structure.

Additional important interfaces for the MCA manager involve the managers of quality assurance, the analytical laboratory, and data processing. The quality assurance manager usually provides review and audit services to monitor the adequacy of the MCA program. The laboratory manager provides analytical services for

TABLE 1

MATERIAL CONTROL and ACCOUNTING DEPARTMENT

POSITION	EDUCATION	EXPERIENCE	TRAINING
Manager	BS - Science, Chemistry, Engineering	10 Years Total - 5 Years in MCA, Q.A. or Manufacturing	Statistics, Advanced Concepts in MCA
Statistician	BS - Mathematics	3 Years - Industrial Firm	Statistical Methods in MCA
Bookkeeper	BA - Accounting	3 Years - Industrial Firm	Automated DATA Precessing: Accounting Concepts in MCA
Measurement Control Administrator	BS - Science	5 Years - Q.A., Analytical Laboratory, Manufacturing	Measurement Techniques, Statistics
Inventory Specialist	2 years College Chemistry/Math	2 years - Industry	Inventory Procedures, Measurement Techniques
Custodian (S) of MBA/ICA	High School Diploma	Nuclear Material Processing (2 yrs.)	MCA Practices, Procedures in Accounting, Records, Item Control, Measurements, Inventory, Tamper Safing

TABLE 2

SAFEGUARDS SUPPORT REQUIREMENTS

POSITION		EXPERIENCE	TRAINING
Quality Control Engineer	BS - Science	3 yrs. - Calibration and Control of Measurement Systems	General MCA Course
Analytical Laboratory Staff			
1. Analytical Chemist(s)	BS - Chemistry	--------	--------
2. Emission Spectroscopist	2 yrs. College Chemistry	--------	Instrument Training
3. Radiochemist	BS - Chemistry or Physics	--------	Use of Non-Destructive Assay Equipment
4. Mass Spectroscopist	BS - Chemistry, MS	--------	Instrument Training
Automated Data Processing Staff			Interpretation and Understanding Safeguards Data and Forms

the measurement of accountability samples. The data processing department provides for the automation of accountability records.

Within the MCA department, the manager is responsible for staff assignments and delegations of authority. Responsibility for coordinating the conduct of physical inventories, administering the measurement control program and for maintaining the bookkeeping system are normally delegated to the MCA department staff.

2. Statistician. The statistician participates in all phases of the safeguards program involving measurements and their uses in conducting inventories, establishing and maintaining the measurement control program and in resolving shipper-receiver differences. During the conduct of the physical inventory, the statistician is responsible for analyzing the error data and computing the limit of error for the inventory difference. The measurement control program includes practices and procedures for analyzing standards and process samples for the purpose of establishing the magnitude of biases and random and systematic errors. The statistician should assist in setting up this part of the measurement control program and in the routing monitoring of the data generated in the program. Shipper-receiver differences that are statistically significant and exceed 50 grams of contained U-235 must be investigated and resolved. The statistician reviews measurement results and supporting documentation and notifies the MCA department manager of unresolvable differences.

3. Bookkeeper. The bookkeeper has the responsibility for maintaining the accountability records. These records include all transactions for material movement, i.e., shipments, receipts, internal transfers, etc. These records constitute the "book inventory" against which the physical inventory is compared. They show the location and status of all material in the plant. The bookkeeper makes all adjustments to the accountability records as required, for instance, following a physical inventory or in the event of shipper-receiver differences.

4. Measurement Control Administrator. This individual, who may be the MCA department manager, is responsible for the program to assure accuracy and precision of nuclear material measurements. He is responsible for setting up procedures and practices for monitoring measurement system performance. He is responsible for setting up the program for measurements of standards and replicates measurements on process materials. He is also responsible for periodically auditing and continuously monitoring the SNM measurement program.

5. Inventory Specialist. An inventory specialist is needed to coordinate and oversee the conduct of physical inventories. This includes preparing for the inventory and preparing reports of transactions outside those related to the normal plant movement of SNM. The inventory specialist plans for the inventory and establishes procedures for carrying out the inventory, including assignments for those participating in the inventory. He schedules the inventory and, in coordination with

the manufacturing department, determines the cutoff for SNM to be included in the inventory. Additional duties for this individual may include the preparation of documentation for external transactions (Form 741 DOE/NRC) and SNM status reports (Form 742 DOE/NRC). He may also prepare discard/loss reports and maintain current records of customer-authorized possession limits.

6. Custodians. Custodians have the responsibility of custody for material within designated MBAs or ICAs. This responsibility entails authorizing the acceptance of material into or the removal of material out of the area, including the preparation of all associated documentation. Custodians normally have control over and oversee the application and removal of tamperseals.

7. Support Personnel. Support personnel have other primary responsibilities. Due to the nature of their job functions, they are essential to support of the MCA Department in tracking the movement of material in the plant and during the taking of inventories.

8. Quality Control Engineer. The quality control engineer periodically inspects and checks all equipment utilized for measuring material or items containing SNM. This responsibility is aimed at preventing the introduction of measurement biases or promptly correcting them should they occur.

9. Analytical Laboratory Staff. In coordination with the MCA staff, the analytical and instrument chemists establish and maintain a measurement program to assure reliable accountability measurements. The program includes: selection of an appropriate

assay technique, establishment of required sample size, sample preparation, analysis and calculation and reporting of results. In conjunction with the accountability measurements, this staff is responsible for performing standard measurements and replicate analyses of process materials in order to establish the magnitude of potential biases as well as the magnitude of the systematic and random errors.

10. Automated Data Processing (ADP) Department. Accountability records are automated and maintained by this department. The accountability data is entered into the ADP system and as necessary, accountability records and reports are generated, including inventory records.

11. Manufacturing Department Staff. The receiving-and-shipping group weigh incoming and outgoing shipments and provide that information to the MCA Department. The floor operators prepare the facility for inventory and, functioning in two-man teams, list and tag all material on inventory. This staff takes all bulk measurements and is responsible, along with QC, for taking samples of SNM to be analyzed for measurement control programs purposes.

B. Safeguards Program Costs

Costs fall into two categories, operating costs and equipment costs. Operating costs are given in terms of man-years (MY) of effort since salaries, fringe benefits and other costs elements can vary over a wide range. To obtain total costs for a full-time individual, a factor of 2.5 to 3.0 is common as a multiplier of the individual's salary. To totally support a

person on a salary of $20,000 per year, a company will pay from $50,000 to $60,000. Table 3 lists the positions described above and the man-years per year devoted to the facility safeguards program. These represent annual operating costs and should be considered as approximate figures. As a simplified example of the total safeguards operating costs, assume the department manager's salary is $35,000 per year and the remainder of the positions have a combined average salary of $20,000 per year (1980 U.S. dollars). Using a 2.75 multiplier, the operating cost for the manager is $96,250. The remainder of the positions total 10.2 MY (with 8 MBA/ICAs). The remainder of the operating costs are therefore, 10.2 x $20,000 x 2.75 = $561,000 for a total estimated operating cost of approximately $660,000 per year. Equipment costs are those costs associated with instruments and other equipment items that are needed not only for safeguards but for manufacturing purposes as well. It is, therefore, difficult to assign cost to the safeguards program unless those costs are for equipment specifically for that program. The approach taken here is to assign 50% of the analytical-laboratory instrument costs to the safeguards program on the basis that 50% of the laboratory staff is estimated to be needed for the safeguards program.

The equipment needs are listed in Table 4. This list includes all those items needed for material control and accounting whether they are necessary to other uses or not. From the list of equipment and instruments listed in Table 4, an estimate of safeguards costs has been made based on the following

TABLE 3. SAFEGUARDS OPERATING COSTS

Position	Annual Effort (MY)
Department Manager	1
Statistician	0.75
Bookkeeper	1
Measurement Control Administrator	0.50
Inventory Specialist	1
Custodian	0.4 per MBA/ICA
Quality Control Engineer	0.25
Analytical Laboratory Staff	0.50
ADP Department	1
Manufacturing Department	2

TABLE 4

SAFEGUARDS EQUIPMENT COSTS

Equipment Item	Purpose	Estimated Cost
A. Bulk Measurements		
Scale (7000 lb Capacity)	Weighing UF_6 Cylinders	$ 15,000
Scales-3 (100 lb Capacity)	Weighing UO_2 Powder/Pellets	15,000
Scales (2000 lb Capacity)	Weighing Drums of UO_2	3,000
Scale (8-10 kg Capacity)	Weighing Pellet Stacks for Rod Loading	3,000
Equipment for Measuring Liquids-Calibrated Collumns and Tanks; Manometer, Pressure Cells, Flowmeter	Measurement of In-Process Scrap Solutions	5,000
B. Wet Chemistry		
Gravimetric Analysis (Precision Balance, Crucibles, Furnace)	Uranium Compounds Analysis	10,000
Titrimetric Analysis (Precision Balance, Glassware, Potentiometer)	Uranium Compound Analysis	3,000
Spectrophotometer	Scrap and Waste Solutions	2,000
Fluorometer	Scrap and Waste Solutions	3,000
C. Instruments		
Mass Spectrometer }	Isotopic Assay	150,000
Gamma Spectrometer }		30,000
Emission Spectrograph	Impurity Assay	100,000
Fuel Rod Scanner (Active Non Destructive Assay)	Fuel Rod Assay	75,000
Passive Gamma Counter	Assay of Scrap and Waste Containers	15,000

assumptions: (1) bulk measurement equipment is utilized mainly for manufacturing and has no safeguards cost element; and (2) one-half the wet-chemistry equipment costs are assigned to safeguards. In Category C, the gamma spectrometer and passive gamma counter costs are assigned totally to the safeguards program, the remaining equipment totally to the manufacturing program. Using these assumptions, costs are obtained as follows:

Equipment Category	Estimated Cost
Wet Chemistry	$ 9,000
Instruments	45,000
TOTAL	$54,000

Equipment maintenance and replacement costs must be added to this cost. The estimated annual maintenance and replacement costs run from 10 to 20 per cent of the initial costs, for an added factor of $5400 to $10,800. Based on these figures, an estimate is made of $54,000 for the initial costs and $10,000 for the annual maintenance and replacement costs. The latter figure should cover the incremental safeguards costs for calibration and test standards for both destructive and nondestructive analyses, sampling and other equipment.

In summary, precise values for the cost of a safeguards program are not given in this report. These costs should be considered as approximate, rough estimate figures. While they are based on the operations conducted at a low enriched uranium fuel fabrication facility, they should apply to other types of fuel processing facilities as well.

IV. POWER REACTOR

Fundamental nuclear material controls do not apply, per se, to a power reactor. This is so for various reasons. There are no bulk materials, analytical measurements, or measurement control programs, and therefore no need for a separate MCA department. Those MCA functions necessary at a power reactor could be readily performed by individuals with other responsibilities for operating and maintaining the reactor. Only item control is necessary with three ICAs being distinguishable: fresh fuel storage, reactor core, and spent fuel storage. The movement of fuel assemblies into and out of these three ICAs must be recorded. Reactor core performance records are kept as part of the normal operating data. These records can be used to calculate U-235 depletion and plutonium buildup in a light-water reactor (LWR) or a heavy-water reactor (HWR). Similar calculations can be made for other reactor types. Custodial and bookkeeping functions, with management responsibilities assigned to an individual to see that these functions are performed, should satisfy the safeguards requirements for a power reactor. These requirements should be met with the conservative estimate of one man-year of effort. Using a composite salary figure of $25,000 per year and a factor of 3.0 for other related costs, an estimate of $75,000 per year is obtained.

SESSION #31: WORKSHOP ON FACILITY SAFEGUARDS SYSTEMS DESIGN

James Shipley, Coordinator; A. Hakkila, D. Cobb, J. Foley, C. Olson, D. Perricos, B. Pontes, D. Reilly, and L. Wirfs

Thursday, June 5, 1980
10:30 am

BIOGRAPHIES

James Shipley: See Session 17

A. Hakkila
Education: Central Connecticut State College, New Britain, B.S. Chemistry 1953; Ohio State University, Columbus, Analytical Chemistry Ph.D. 1957.

Present Position: Associate Group Leader, Q-4, Nuclear Safeguards Systems, Los Alamos Scientific Laboratory.

Present Duties: Participate in design of Advanced Safeguards Systems for Nuclear Facilities; US Technical Advisor to International Working Group on Reprocessing Plant Safeguards.

Past Positions: Seventeen years in development of analytical chemistry methods for nuclear fuel cycle materials; section leader for x-ray fluorescence and absorption edge analysis and electron microprobe analysis.

Other Information: Fellow, American Institute of Chemists; Member, American Chemical Society, Institute of Nuclear Materials Management, Microbeam Analysis Society. Editor of book, Nuclear Safeguards Analysis (over 60 publications).

D. Cobb: See Session 26
J. Foley: See Session 14
C. Olson: See Session 11
B. Pontes: See Session 3
D. Reilly: See Session 22

Session Objectives

SESSION #31: WORKSHOP ON FACILITY SAFEGUARDS SYSTEM DESIGN

The workshop will enable participants to apply the safeguards principles presented and discussed during the entire course to the design of a national system of accountability and control for a postulated reference-state nuclear power program. Participants will be divided into four working groups, each having appropriate guidance by Course Staff, as needed. Design considerations will include national safeguards system performance requirements, organization and training of personnel, equipment procurement and implementation, and evaluation of overall system effectiveness including diversion sensitivity and detection timeliness. Due attention will be given to the vital importance of compatibility with IAEA safeguards requirements. The workshop will be structured to emphasize key aspects of safeguards system design and implementation while eliminating irrelevant detail in the interest of maximizing clarity, understanding, and effective technology transfer.

SESSION 31: WORKSHOP ON FACILITY SAFEGUARDS SYSTEM DESIGN

E. A. Hakkila, D. D. Cobb, and J. P. Shipley
J. P. Shipley, Workshop Coordinator
Los Alamos Scientific Laboratory

Summary: Guidelines and Format for Workshop

Participants will be divided into two groups, each of which will consider the design and development of a State's System of Accountability and Control (SSAC) for a postulated reference nuclear power program. Each of these groups will be further subdivided into two subgroups, one of which will act as facility operators (the operator's subgroup) and the other as SSAC inspectors (the inspector's subgroup). IAEA representative(s) will be present for consultation and negotiation with both groups on the interface of their State's System design with IAEA safeguards requirements.

The reference plant will be a 1000 MW PWR reactor in a non-nuclear weapons state. The plant receives fabricated fuel elements from another independent State. Spent fuel will be stored on-site in a storage pool pending transfer to an away-from-reactor storage facility or to a fuel reprocessing facility, either of which is located in another state.

Both groups are to consider the problem of developing a safeguards approach and a SSAC for the facility using the facility design information that is provided. The safeguards approach will consider design verification, materials accounting and control, and inspection strategies for the fresh fuel storage area, the reactor, and the spent-fuel storage pool.

The objective for both groups is to have their respective operator and inspector subgroups develop an appropriate State's System of Accounting and Control for the reference facility and

then to negotiate a facility attachment with the IAEA representative(s). The operator subgroups will consider the problem of implementing the system to provide adequate materials accounting and control in a practical and cost-effective manner. The inspector subgroups (and the IAEA representative(s)) will consider the problem of implementing adequate inspection and verification both for domestic safeguards requirements and for international safeguards requirements, within realistic limitations imposed by the NPT and available IAEA resources.

I. FACILITY DESCRIPTION

The reference power reactor is a pressurized water reactor having a nominal electrical output of approximately 1,000 MW. The areas of safeguards interest in the facility include the reactor, a fresh-fuel handling and storage area, a spent-fuel handling and storage pool, and on-site capability for disassembly or repairs of fuel assemblies.

A. Fresh Fuel Handling and Storage

Fresh fuel is stored in the fuel handling building, which is located outside the primary reactor containment. The fuel is stored dry in the fresh-fuel vault, which consists of vertical racks arranged in parallel rows. An overhead fuel-handling crane in the building moves fresh fuel from the vault to the new fuel elevator for transfer to the spent-fuel pool. Fresh fuel is handled by the spent-fuel handling equipment once it is in the pool. A transfer carriage moves the fuel through a transfer tunnel from the pool to inside the containment. The refueling machine moves the fuel to the reactor core.

B. The Reactor

A normal reactor core loading comprises approximately 50,000 fuel rods, which are contained in 217 fuel assemblies. The fuel consists of sintered pellets of UO_2 having a nominal enrichment of 2-3% ^{235}U. Each fuel assembly can contain up to 236 fuel rods. However, up to 16 poison rods may be inserted into an assembly.

The reactor is refueled once per year with approximately one-third of the core loading replaced. The refueling operations last 3-5 weeks. Fuel burnup at discharge averages approximately 14,000 megawatts days per metric ton (MWD/Te) with a maximum burnup of approximately 21,000 MWD/Te.

TABLE I

PWR FUEL CHARACTERISTICS

	Fresh	Discharged
Dimensions, Rod	406.4 cm long 0.970 cm diam	406.4 cm long 0.970 cm diam
Rods/Assembly	∿220	∿220
Fuel	UO_2	$UO_2 + PuO_2$
Heavy Metal/Rod	1.805 kg	1.805 kg
Heavy Metal/Assembly	∿400 kg	∿400 kg
^{235}U Enrichment	2-3%	∿1%
Pu/Rod	-	∿18 g
Pu/Assembly	-	∿3.9 kg

Each fresh fuel assembly contains approximately 400 kg of uranium and between 8 and 12.5 kg of fissile ^{235}U. The total weight of each assembly is approximately 541 kg including cladding and end-fittings. Cladding is Zircaloy-4. Fresh-fuel characteristics are summarized in Table I.

C. Spent Fuel Storage Pool

The spent fuel storage pool has a capacity to store ∿600 assemblies using the normal storage configuration. However, the facility is considering alternative modes of spent fuel storage to increase the capacity of the spent-fuel pool.

II. STATE'S SYSTEM OF ACCOUNTING FOR AND CONTROL OF NUCLEAR MATERIAL

The requirements for a SSAC are established by the State Authority, for example, in the US by the Nuclear Regulatory

Commission. However, in establishing the State's system the State must bear in mind that the operator needs data on the flow of nuclear material through the facility for reasons such as health and safety, nuclear criticality, and materials value, as well as safeguards accounting. Also the IAEA will require safeguards information to verify that material has not been diverted from the facility. The State's system should be designed to take advantage of the operator's existing requirements and capabilities and to help meet the IAEA requirements.

A. SSAC Requirements

As examples, the following items are identified by the US NRC as necessary for the State's system of accounting for and control of nuclear material, and should be considered in the design of the SSAC.

1. Establish a safeguards system structure. This includes planning, coordination, and administration of nuclear materials accounting activities with responsibility for all functions delegated to one individual or organization.

2. Establish a materials balance structure. A materials balance area (MBA) is defined such that all movement of material into and out of the area is measured or counted and recorded. All such transfers are performed through key measurement points (KMPs).

3. Define a reporting system. Reports should cover all unauthorized as well as authorized movements of nuclear material through the facility and should include an annual report describing all material within the facility.

4. Establish a materials accounting system. This includes records for receipts, inventory, disposal, and transfer of nuclear material with sufficient information to form materials balances. Measurement of material may be defined where appropriate. Surveillance devices required for verifying the integrity of items in item control areas (ICAs) should be specified.

5. Establish a nuclear materials measurement program. Any sampling plan should be based on statistical grounds. Tamper-safing or seals can be used as a basis for accepting prior measurements.

6. Establish a measurement control program. This should determine and control random and systematic errors of all measurement processes used for nuclear materials accounting.

7. Define a statistical approach to determining nuclear materials losses. The approach should incorporate the considerations under items 5 and 6. The statistical methods should be designed to be compatible with desired performance criteria and to make most effective use of the available materials accounting data.

8. Establish a system of inspection and audits. This should include a definition of the activities, frequency of occurrence, and the kinds of results to be expected. An assessment of inspection manpower and capability should also be given.

B. Materials Balance Areas for Reactors

For purposes of nuclear materials accounting a reactor may be treated as a single materials balance area (MBA), as shown in Figure 1, with the following flow key measurement points (KMPs):

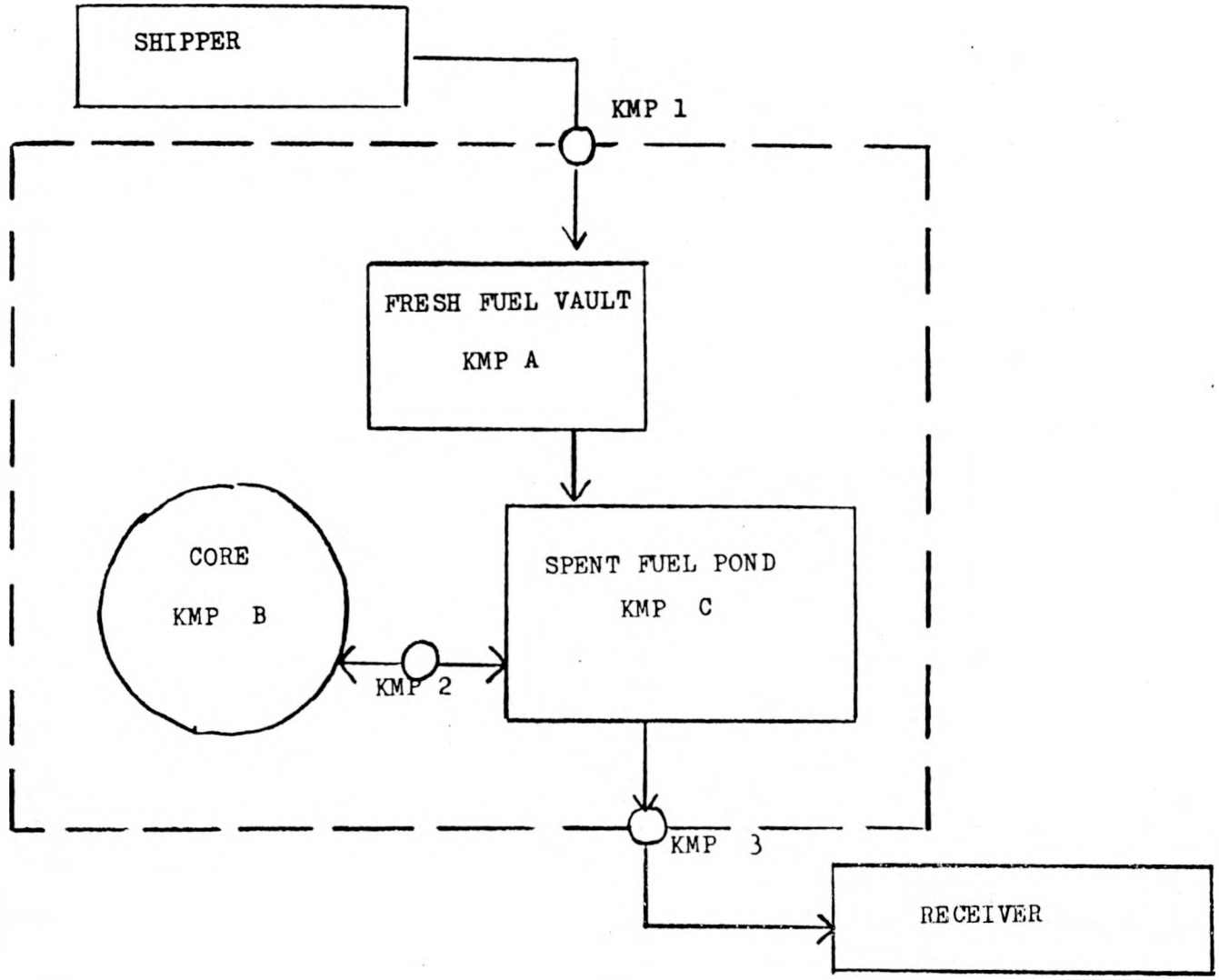


Figure 1.
Nuclear material flow sheet.

KMP#1 - receipt of fresh fuel to the reactor
KMP#2 - nuclear material consumed and produced in the reactor
KMP#3 - shipments of spent fuel from the reactor.

The following item accounting KMPs also may be defined.

KMP A - fresh fuel in the fresh fuel storage pool
KMP B - fuel in the reactor core
KMP C - spent fuel in the spent fuel storage pool
KMP D - other nuclear material in the facility (not shown in the figure).

III. IAEA VERIFICATION

The IAEA verification of the SSAC is designed for timely detection of the diversion of a significant quantity of nuclear material. For low-enriched-uranium power reactors, the detection times and goal quantities are summarized in Table II.

The IAEA verification procedures for materials accounting incorporate a combination of inspection and reports with a physical inventory taking (PIT). The frequency of PITs and inspections is dictated by the type of facility and the form of the nuclear material. As shown in Table II, the frequency of PITs for low-enriched-uranium power reactors is once per year, generally coinciding with the annual fuel reload operations. The inspection frequency is defined in Article 80 of INFCIRC/153 based on the amount and type of nuclear material in the facility. Power reactors may be inspected every 2-3 months. Inspection activities may include independent verification of measurement of nuclear material in the facility, such as fuel in the fresh-fuel storage area and the spent-fuel storage pool. Continuity of

TABLE II

IAEA GOAL QUANTITIES AND DETECTION TIMES FOR NUCLEAR REACTORS

Type of Material	Significant Quantity	Detection Goal	Detection Time	PIT Frequency	Inspections
LEU	75 kg ^{235}U	No. of fuel assemblies containing 75 kg ^{235}U	1 yr	1/yr	4-6/yr
Pu in fresh fuel	8 kg Pu	No. of fuel assemblies containing 8 kg Pu	1-3 wks		each 2-3 wks
Irradiated fuel	8 kg Pu	No. of fuel assemblies containing 8 kg Pu	1-3 months	1/yr	4-6/yr

knowledge of the fresh and spent fuel in the respective storage ponds is maintained through a complementary system of containment and surveillance, discussed in detail in Session 11.

Following each PIT verification, the IAEA inspection activities are evaluated relevant to meeting the Agency's safeguards objectives. This evaluation is performed for each materials balance area. A statement summarizing the findings is sent to the Member State.

IV. SAFEGUARDS SYSTEMS DESIGN APPROACH

In designing the State's system of accounting for and control of nuclear material it is assumed that the following records are available to the safeguards inspectorate.

1. Power history.
2. Fuel management records, necessary to calculate plutonium production and uranium depletion.

3. Internal transfer forms, describing movement of assemblies in the facility by assembly identification number.

4. Book inventory listing, including uranium and plutonium content.

5. State transfer documents.

6. Crane movement records.

7. Inventory change reports.

8. Physical inventory listings.

9. Materials balance reports.

10. Special reports.

The detection of the IAEA goal quantities of 75 kg of ^{235}U contained in low-enriched uranium in one year, or 8 kg of plutonium in spent fuel in a period of 2-3 months, should be considered as safeguards system objectives. This is equivalent to six fresh-fuel assemblies for ^{235}U or 2 spent-fuel assemblies for plutonium. The use of NDA measurements for fresh fuel received at the reactor and for spent fuel leaving the reactor may be incorporated for transfer measurements and/or verification.

The safeguards systems design should address the following points (at least):

- The structure of the SSAC, including physical protection and materials accounting; a specification of the points and natures of the interactions among the components of the SSAC, between the SSAC and the facility operator, and between the SSAC and the IAEA.

- The structure of the MBA(s), including measurement points.

- The specification of the materials accounting system, including types of measurements, measurement frequencies, sampling plans, and interfaces with the physical protection system to guard against measurement tampering.

- A measurement control program to assure the continued performance of the materials accounting system.

- A method of data analysis for diversion detection and an estimate of detection sensitivity and false-alarm rate.

- A system of inspections, audits, and reports, considering both the facility operator and the IAEA.

- An estimate of the required inspection manpower and capability.

A typical facility attachment for the IAEA is included as Appendix A.

LWR with spent fuel stored for a year or longer, with pin exchange

Safeguards Agreement under NPT between Atlantis and the IAEA

Subsidiary Arrangements

Facility Attachment No. 6: Atlantis Nuclear Power Station No. 4

Facility: AIE- MBA: AI-E

Total No. of pages 16 Page No. 1 Date: 3 March 1980

Code	General Part Reference (Codes)	Agreement Reference (Articles)	
1.		43(a)	Identification of the facility Facility identification code: AIE-
1.1			Name, owner and operator Atlantis Nuclear Power Station No. 4
1.2			Geographic location About 200 km east of the city of Seroza
1.3			Postal address Atlantis Electric Power Co. Cape Nese, Atlantis
1.4			Description The power station consists of one direct cycle light water reactor, 1000 MW(th).
1.5			Maps and plans Attached herewith.
2.	3.1.1	43,44, 46(a)	Information on the facility This facility attachment is based on the Design Information provided by the Government of Atlantis as of July 1979.
2.1		8(c)	Location of information Identical sets of the information provided on the facility are kept at the Agency, at the facility, and at the Atlantis Nuclear Regulatory Bureau.

Code	General Part Reference (Codes)	Agreement Reference (Articles)	
2.2	3.1.3	45	Changes in the information on the facility to be provided in advance (With reference to the relevant paragraphs of the design information questionnaire) 13 - Any change in the rated thermal output for continuous operation; 10,35 - Any change in the access routes to the reactor area; 19,20,22 - Any change in the design of the reactor fuel; 21 - Any change in the nominal enrichments of the fuel; 24 - Any change of the method of identifying individual fuel assemblies; 34,38 - Any change of the refuelling equipment or methods; 35 - Any change in connection with the reactor vessel or its cover influencing access to the core; 29 - Introduction of new irradiation positions inside the reactor vessel; - Introduction of new loop heat removal equipment; - Installation of any fuel assembly decladding or dissolution equipment; 10 - Any change in the routes of the shipping cask for irradiated fuel within the facility; 42 - Any change in the health and safety procedures affecting the conduct of inspection.
3.			Safeguards measures
3.1		29	Accountancy
3.1.1		46(b)	Material balance areas and their identification codes. The Atlantis Nuclear Power Station No. 4 consists of one material balance area.

Code	General Part Reference (Codes)	Agreement References (Articles)	
3.1.2		46(b) 98K 98S	Strategic points which are key measurement points (KMPs) (for their specifications see Code 4). (a) For determination of nuclear material flow: KMP 1 – Receipt and de-exemption of nuclear material. KMP 2 – Nuclear loss and production in fuel discharged from the reactor 1/, and rebatching KMP 3 – Shipment of nuclear material, exemption. (b) For determination of physical inventory: KMP A – Fresh fuel storage KMP B – Fuel in the reactor core KMP C – Spent fuel storage KMP D – Other locations of nuclear material at the facility.
3.1.3		46(c)	Physical inventory taking Nominal timing: once a year. As soon as possible after each refuelling and before the reactor is closed again. Procedures: Item counting and identification
3.2		29	<u>Containment and surveillance</u>
		46(f) 98S	Strategic points for application of containment and surveillance measures: – Reactor hall – Access routes to the reactor hall, including the spent fuel pond.
3.2.2		75(d)	Installed Agency instruments and devices:
		75(e)	(a) Seals to ensure the containment of the reactor vessel; (b) Camera s for surveillance of fuel movements into or out of the reactor containment, including the spent fuel ponds. (c) Seals on shipping casks with spent fuel.

1/ Fuel removed from the reactors shall be considered as discharged if it remair out of core for longer time than the duration of a routine refuelling shutdow

Code	General Part Reference (Codes)	Agreement Reference (Articles)	
			If there is a need to break a seal or interfere with the operation of safeguards instruments, the Agency shall be informed in advance and by the fastest means. This information shall include the (probable) date on which the operation will take place.
3.3		11 35(a)	Specific provisions and criteria for termination of safeguards on nuclear material None.
3.4		36	Specific provisions and criteria for exempting nuclear material from safeguards None.

Code 4

Agreement References (Articles): 57, 98C, 98R
Specifications for key measurement points.

Code 4.1 KMPs for the flow of nuclear materials

KMP	Inventory change	Description of a typical batch	Description of a typical item		Material description	Measurement basis
	Receipt	For fuel assemblies: One fuel assembly	One fuel assembly	For each fuel assembly: 1) Identification number 2) Weights of total and fissile uranium 2/ ~~(and of plutonium)~~ and the chemical composition based on shipper's data.	BQ2F BV2F	N
		For fuel pins: A number of separate fuel pins with the same nominal initial total and fissile uranium content, received in one consignment	One fuel pin	For each fuel pin: 1) Identification number 2) Weights of total and fissile uranium and the chemical composition based on shipper's data.	DQ2F DV2F	N
		For small quantities of nuclear material (each less than 0.01 effective kilogram): Any number of such quantities received in one calendar month from the same shipper or, if a physical inventory was taken during the month, separately before and after the time of physical inventory taking.	Not applicable	1) Weight of compound 2) Weights of total and fissile uranium and the chemical composition based on shipper's data.	QSØB	N

Code 4.1 Ps for the flow of nuclear material (continued)

KMP	Inventory change, Rebatching	Description of a typical batch	Description of a typical item	Source data	Material description	Measurement basis
2	Rebatching	For fuel: Each fuel assembly in which pins have been exchanged; all fresh fuel pins with the same nominal initial content; all irradiated fuel pins with the same initial nominal content.	One fuel assembly, one fuel pin	As at KMP 1 above	BQ1F BV1F DQ1F DV1F BQ1G BV1G DQ1G DV1G	N
	Nuclear production, Nuclear loss (burn-up)	For fuel: All fuel assemblies and separate fuel pins discharged together.	One fuel assembly	1)-2) As for fuel assemblies at KMP 1 above 3) Estimated burn-up of each fuel assembly (in MWD/tU) 4) Nuclear loss of total and fissile uranium and nuclear production of total plutonium as calculated for each fuel assembly discharged or recycled into the core	BQ1G BV1G	M
			One fuel pin	For each fuel pin: 1) Identification number of fuel assembly from which the fuel pin was removed. 2) As for fuel pin at KMP 1 above. 3) Nuclear production as calculated for each fuel pin discharged.	DQ1F DQ1G	M

1/ For assemblies recycled into the core, these data are to be subtracted from total nuclear loss and total production, respectively.

Code 4.1 KMPs for the flow of nuclear material (continued)

KMP	Inventory change	Description of a typical		Source data	Material description	Measurement basis
		batch	item			
2	Nuclear loss	For fuel: All fuel assemblies and separate fuel pins shipped together.	One fuel assembly One fuel pin	1)-2)-3) as above 1) Nuclear loss of total plutonium (Pu 241 decay) for each fuel assembly and separate fuel pin.	BQ1G BV1G DQ1G DV1G	M
3	Shipment	For fuel assemblies: One fuel assembly shipped.	One fuel assembly	Weight and, if available, isotopic composition of total and fissile uranium and weight of total plutonium in each fuel assembly as calculated to allow for nuclear loss and production.	BQ3G BV3G BQ2F BV2F	M N

Code 4.1 KMPs for the flow of nuclear material (continued)

KMP	Inventory change	Description of a typical batch	Description of a typical item	Source data	Material description	Measurement basis
3	Shipment	For separate fuel pins: A number of separate fuel pins with the same nominal initial total and fissile uranium content and the same material description shipped together.	One fuel pin	For each fuel pin: Weights and, if available, isotopic composition of total and fissile uranium and weight of total plutonium in each fuel pin shipped, as calculated to allow for nuclear loss and production.	DQ3G DV3G DQ2F DV2F	M N
	Shipment Nuclear production Nuclear loss	For small quantities of nuclear material (each less than 0.01 effective kilogram): Any number of such quantities shipped in one calendar month to the same recipient or, if a p ysical inventory was taken during that month, separately before and after the time of physical inventory taking.	Not applicable	1) Weight of compound 2) Weights of total and fissile uranium and chemical composition as given by the operator.	QSØB QSØJ	N M
	Exemption	The same as for shipments or receipts, respectively				
	Accidental loss					
1	De-exemption					

Code 4.2 KMPs for the physical inventory of nuclear material

KMP	Description of a typical batch	Description of a typical item	Source data	Material description	Measurement basis
A B	For fuel assemblies: One fuel assembly.	One fuel assembly	For each fuel assembly: 1) Identification number 2) Weights of total and fissile uranium and the chemical composition based on shipper's data.	BQ1F BV1F BQ4G BV4G	N N
A	For fuel pins: A number of separate fuel pins with the same nominal initial total and fissile uranium content.	One fuel pin	For each fuel rod: 1) Identification number 2) Weights of total and fissile uranium and the chemical composition based on shipper's data.	DQ1F DV1F	N
C	For fuel assemblies: One fuel assembly.	One fuel assembly	Weight and, if available, isotopic composition of total and fissile uranium and weight of total plutonium in each fuel assembly as calculated to allow for nuclear loss and production.	BQ1G BV1G BQ1F BV1F	M M N
	For separate fuel pins: A number of separate fuel pins with the same nominal initial total and fissile uranium content.	One fuel pin	For each fuel pin: 1) Identification number of fuel assembly from which the fuel pin was removed. 2) As for fuel pin at KMP 1 above. 3) Nuclear loss and production as calculated for each fuel pin. 4) Weights and, if available, isotopic composition of total and fissile uranium and weight of total plutonium in each fuel pin, as calculated to allow for nuclear loss and production.	DQ1G DV1G DQ1F DV1F	M N
,B, ,D	For small quantities of nuclear materials (each less than 0.01 effective kilogram): Any number of such quantities.	Not applicable	1) Weight of compound 2) Weights of total and fissile uranium based on shipper's data.	QSØB QSØJ	N N

Code	General Part Reference (Codes)	Agreement Reference (Articles)	
5.		46(d),51	Records system
5.1		56	Specific provisions for accounting records
5.1.1		56(a),98D	Inventory and batch changes (for the specifications of source data see Code 4.1 above), time of recording.
			- Receipts (KMP 1) Upon receipt, on shipper's data
			- Rebatching of pins and fuel assemblies when pins are exchanged in fuel assemblies: Upon pin replacement.
			- Nuclear production and nuclear loss (uranium burn-up)(KMP 2): For fuel assemblies and fuel pins, upon discharge from the reactor. 1/ For small quantities of nuclear material, upon shipment.
			- Nuclear loss (Pu 241 decay) (KMP 2): Upon shipment, if considered appropriate.
			- Shipment (KMP 3): Upon shipment.
	6.3,6.2		- Exemption/de-exemption (KMP 3/KMP 1) Upon transfer of nuclear material out of/into the facility respectively.
			- Accidental loss (KMP to be determined after the accident) Upon determining the amount of the loss.
5.1.2	4.2	56(b),98D	Measurement (item counting and identification) results used for determination of the physical inventory (for the specifications of source data see Code 4.2 above), time of recording.
			- All physical inventory KMPs Upon identification and counting of items during the physical inventory taking.
			- Itemized list of nuclear material quantities on inventory. Before inventory taking.
5.1.3		56(c)	Adjustments and corrections, time of recording.
			- Shipper/receiver difference (KMP 1): Not relevant.
			- MUF Normally identical to zero.
			- Corrections Whenever errors have been found.

1/ If fuel assemblies already recorded as discharged are returned to the core at any time later, negative values shall be recorded for nuclear loss and production so as to restore the shipper's values for the fuel contained in such fuel assemblies.

Code	General Part Reference (Codes)	Agreement Reference (Articles)	
5.2		58	Specific provisions for operating records
5.2.1		58(a)	Operating data used to establish changes in the quantities and composition of nuclear material. - Location of each fuel assembly or separate fuel pin at any time; - The relevant source data with respect to nuclear loss and production, including: (a) The integrated thermal power produced by the reactor - once per ~~[day][shift]~~; and (b) The estimated burn-up (in MWD/t) for each fuel assembly and separately discharged fuel pin after its discharge from the reactor. - Date and duration of any reactor shutdown; - Date and description of any dismantling operation of a fuel assembly (pin removal or exchange).
5.2.2		58(b)	Calibrations Not required.
5.2.3		58(c)	Sequence of the actions taken in preparing for and in taking the physical inventory. - All physical inventory KMPs: Dates and description of the actions taken and the results obtained.
5.2.4		58(d)	Actions taken in order to ascertain the cause and magnitude of any accidental or unmeasured loss: - Dates and description of the actions taken and the results obtained.
5.3		52	Location and language of records At the facility.
5.4		53	Retention period for records Five years.
6.		46(d),59	Reports system
6.1	3.4.1	63(a),64, 65	Specific provisions for inventory change reports (ICRs)
6.1.1	10	98D	Contents. The recorded entries to be reported are those type specified in Code 5.1.1 above. Nuclear production and nuclear loss (burn-up) will be reported upon discharge of irradiated fuel from the reactor. Nuclear loss (Pu-241 decay) may be reported upon shipment. Rebatching of pins and fuel

Code	General Part Reference (Codes)	Agreement Reference (Articles)	
			assemblies will be reported using Codes RP and RM. 1/ Forms R.01.1/c will be completed as specified in paragraphs 1-23 of Code 10, General Part. However, Forms R.01.2 will be completed as specified in paragraphs 1-28 of Code 10, General Part, to report shipments of irradiated fuel assemblies, in respect of which the isotopic composition is available.
6.1.2			Timing or frequency of dispatch. Within 30 days after the end of the month in which receipt, pin exchange in fuel assemblies, discharge, shipment, exemption, de-exemption or accidental loss occurred or was established.
6.2	3.4.1	64	Specific provisions for concise notes
6.2.1		64(a)	Concise notes explaining the inventory changes: - To be attached to each ICR containing the data on nuclear loss and production to state the burn-up in MWD/t of initial U for each fuel batch discharged; - To be attached to ICRs to explain unusual inventory changes (such as accidental loss) and corrections. They may also be used to explain any other part of information included in reports.
6.2.2		64(b)	Concise notes describing the anticipated operational programme; subject and time of dispatch: - Planned operations involving the removal of the reactor vessel seal, e.g. refuelling and fresh fuel receipts and spent fuel shipments; To be attached to each MBR (see Code 6.3.3 below) and to cover the period until the end of the next refuelling. - Precise forecast for the next refuelling physical inventory taking or spent fuel shipment, including information about the shipping casks to be used and the extent to which these are expected to be filled. To reach the Agency at least 30 days in advance, subsequent changes thereto as soon as they are known.

1/ If fuel assemblies already recorded as discharged are returned to the core at any time later, negative values shall be reported for nuclear loss and production so as to restore the shipper's values for the fuel contained in such fuel assemblies.

Code	General Part Reference (Codes)	Agreement Reference (Articles)	
			- Dates and identification numbers of fuel assemblies involved in any planned dismantling operation (pin removal or exchange): Normally to reach the Agency at least 30 days in advance.
6.3	3.4.2	63(b),67	Specific provisions for material balance reports
6.3.1	10	98D	Contents. The consolidated inventory changes to be reported are those types specified in Code 5.1.1 above. Forms R.03 will be completed, as specified in paragraphs 44-54 of Code 10, General Part.
6.3.2	10	98D	Physical inventory listings (PILs) to be attached to MBRs: The batch data included in PILs will be based on the shipper's data on the initial nuclear material content of the fuel for fuel in KMPs A and B and for small quantities and on operator's data for fuel in KMP C. Forms R.02/c will be completed, as specified in paragraphs 29-43 of Code 10, General Part.
6.3.3			Timing or frequency of dispatch: Within 30 days of the physical inventory taking under Code 3.1.3 above.
6.4	3.5.1	68	Special reports
6.4.1			Specification of circumstances requiring submission of special reports. (a) Loss limits: One fuel element (pin). (b) Changes in containment: - Physical integrity of a fuel assembly as an accounting unit is accidentally broken; - Any Agency containment and surveillance device, referred to in Code 3.2.2 such as a seal or camera, is interfered with or removed in the absence of Agency inspectors, unless the Agency has been informed in advance as provided. 1/
6.4.2			Contents, as appropriate. - Date when the incident or circumstance occurre

1/ In respect of seals on shipping casks, this requirement applies only while the cask remains in the facility.

Code	General Part Reference (Codes)	Agreement Reference (Articles)	
			- Description of the actions taken in order to ascertain the cause of the incident or circumstance and the magnitude of the loss; - Cause and features of the incident or circumstance; - Estimated amount of nuclear material which has been lost.
7.			Inspections
7.1		78,80	Mode of routine inspections
			Intermittent.
7.2	Annex	80	Applicable formula and procedure for determination of maximum routine inspection effort
			Article 80(a) of the Agreement.
7.3		78,81	Indication of the actual inspection effort under ordinary circumstances
			An estimate of the actual routine inspection effort, as far as can be foreseen and assuming:
	3.1		(a) Circumstances at the facility to be as described in the information provided in respect of the facility;
	2		(b) The continued validity of the information on the national system of accounting for and control of nuclear material, as set out in the General Part:
			(c) That there shall be one refuelling(s) a year and two irradiated fuel shipments, including no more than one involving casks that are not completely filled; and/or
			(d) That there are means of correlating the identity and number of fuel assemblies in the spent fuel pond at the reactor site and in the reception pond at the reprocessing plant.
			.20.... man-days per year
7.4		74,75	Indication of the scope of routine inspections under ordinary circumstances
7.4.1			General:
			- Examination of the records, verification for self-consistency and consistency with reports.

Code	General Part Reference (Codes)	Agreement Reference (Articles)	
7.4.2			At inventory KMPs: - Verification of the inventory, e.g. by item counting, identification and integrity checks, non-destructive measurements.
7.4.3			At flow KMPs: - Verification of inventory changes, e.g. by item counting, identification and integrity checks, non-destructive measurements of fresh and irradiated fuel, including the use of seals on containers of fresh and irradiated fuel.
7.4.4			At strategic points for containment and surveillance (reactor hall and access routes thereto): - Observation of refuelling and spent fuel removal operations; - Observation of dismantling of fuel assemblies (pin removal or exchange); - Application, examination and removal of Agency seals used in accordance with Code 3.2.2, as well as of other seals; - Servicing and maintenance of the surveillance equipment.
7.5		75(d)	Arrangements for the use by the Agency of equipment for independent measurement Specific arrangements for the use of equipment to be made as the need arises.
7.6	9.4	75(a), 75(c)(i)	Duplicates and additional samples Not relevant.
7.7		87,88	Persons to whom a request for any operation or for services at the facility should be addressed The representative of the ANRB.
7.8			Contacts at the facility The representative of the ANRB.
7.9		15,88	Services and charges
7.9.1			Services provided by the operator free of charge: - Health and safety services (protective clothin dosimeters); - Office space for the Agency's inspectors; - Power supply for the Agency instruments; - Personnel for handling the fuel assemblies during their measurements;

Code	General Part Reference (Codes)	Agreement Reference (Articles)	
			- Available equipment for handling the fuel assemblies during their measurements.
7.9.2			Services provided by the operator with the charges to the Agency, as quoted below:
			- Means of communication (telephone, telex, cable): according to existing rates.
			If any specific request by the Agency for services not covered above gives rise to expenses for which reimbursement is requested from the Agency, the Agency shall be notified of the expenses before the service is performed. The Agency will only reimburse such expenses if it has confirmed its initial request and agreed in writing to the amount involved.
7.9.3			Mode of reimbursement of the expenses charged to the Agency:
			- By cheque after receipt of the invoice by the Agency.
7.10		44,89	Specific facility health and safety rules and regulations to be observed by the Agency's inspectors
			As specified in paragraph 42 of the Design Information provided by the Government of Atlantis dated July 1979.
8.	4.1.3	30,90	Agency statements
8.1		90(a)	A summary statement will be made on the result of each inspection within 30 days of its completion.
8.2		90(b)	A statement on the conclusions the Agency has drawn from its verification activities in respect of the facility will be made within 60 days after the end of the month in which the Agency has verified the physical inventory. The statement will include, as appropriate, conclusions drawn from:
			(a) Records examination; (b) Reports to the Agency; (c) Verification of containment and surveillance measures; (d) Verification of inventory changes; (e) Verification of material accountancy; (f) Verification of the quality and functioning of operator's measurement system; (g) Activities in respect of MUF, shipper/receiver differences and/or losses.

SESSION #32: PLENARY SESSION AND WRAP UP OF SAFEGUARDS SYSTEMS DESIGN WORKSHOP

James Shipley, Coordinator; A. Hakkila, D. Cobb, J. Foley, C. Olson, D. Perricos, B. Pontes, D. Reilly, L. Wirfs

Friday, June 6, 1980
10:30 a.m.

Session Objectives

The national safeguards system designs generated by each of the working groups during Session 31 will be presented and compared. The relative capabilities and trade-offs of the designs will also be evaluated and constructively critiqued by the Course Staff. If desired, supplemental staff time can be made available to individual participants on Saturday morning, June 7, for assistance with special problems, technical consultation, etc., as needed.

INTERNATIONAL TRAINING COURSE ON NUCLEAR MATERIALS ACCOUNTABILITY FOR SAFEGUARDS PURPOSES

LECTURERS & STAFF

Dr. Carleton D. Bingham
US Department of Energy
New Brunswick Laboratory
9800 South Cass Avenue
Argonne, IL 60439

Dr. Carlos Buechler
International Atomic
Energy Agency
Box 100
A-1400, Vienna, Austria

Dr. Thomas R. Canada
Los Alamos
Scientific Laboratory
Group Q-3, MS 539
P. O. Box 1663
Los Alamos, NM 87545

Dr. Donald D. Cobb
Los Alamos
Scientific Laboratory
Group Q-4, MS 541
P. O. Box 1663
Los Alamos, NM 87545

Mr. John Dailey, DP-343
US Department of Energy
Office of Safeguards and Security
Mail Station A2-1016, GTN
Washington, DC 20545

Dr. John E. Foley
Los Alamos Scientific Laboratory
Group Q-5, MS 537
P. O. Box 1663
Los Alamos, NM 87545

Dr. Dipak Gupta
Gesellschaft Fur
Kernforschung MBH
D-75 Karlsruhe
Postfach 3640, F. R. Germany

Dr. E. Arnold Hakkila
Los Alamos
Scientific Laboratory
Group Q-4, MS 541
P. O. Box 1663
Los Alamos, NM 87545

Dr. Charles R. Hatcher
Los Alamos
Scientific Laboratory
Group Q-1, MS 540
P. O. Box 1663
Los Alamos, NM 87545

Mr. William A. Higinbotham
Brookhaven National Laboratory
Upton, LI, NY 11973

Mr. Gordon Hough
International Atomic
Energy Agency
Box 100
A-1400, Vienna, Austria

Karen Humphrey
Los Alamos
Scientific Laboratory
DAD/SS, MS 550
P. O. Box 1663
Los Alamos, NM 87545

Mr. John L. Jaech
Staff Consultant
Exxon Nuclear Company, Inc.
2955 George Washington Way
Richland, WA 99352

Dr. Brian Jenkins
RAND Corporation
1700 Main Street
Santa Monica, CA 90406

Mr. Ed R. Johnson
E. R. Johnson & Associates
11702 Bowman Green Drive
Reston, VA 22090

Dr. G. Robert Keepin
Los Alamos
Scientific Laboratory
DAD/SS, MS 550
P. O. Box 1663
Los Alamos, NM 87545

Dr. Donald M. Kerr
Director,
Los Alamos
Scientific Laboratory
P. O. Box 1663, MS 100
Los Alamos, NM 87545

Dr. Ronald A. Knief
GPU Service Corp.
P. O. Box 480
Middletown, PA 17057

Dr. Hiroyoshi Kurihara
Embassy of Japan
Suite 900, Watergate
600 New Hampshire Ave. NW
Washington, DC 20037

Dr. David M. Lee
Los Alamos
Scientific Laboratory
Group Q-5, MS 537
P. O. Box 1663
Los Alamos, NM 87545

Dr. Ralph F. Lumb, President
NUSAC, INC.
7926 Jones Branch Drive
McLean, VA 22102

Dr. John J. Malanify
Los Alamos
Scientific Laboratory
Group Q-3, MS 539
P. O. Box 1663
Los Alamos, NM 87545

Dr. Ugo Miranda, Chief
EURATOM Safeguards Inspection
18 Rue Doct. Peffer Howald
Kirchberg, LUXEMBOURG

Dr. Nicholas Nicholson
Los Alamos
Scientific Laboratory
Group Q-2, MS 562
P. O. Box 1663
Los Alamos, NM 87545

Mr. Christopher E. Olson
Div. 1758
Sandia Laboratories
P. O. Box 5800
Albuquerque, NM 87185

Mr. Takeshi Osabe
Japan Nuclear
Fuel Company, Limited
3-1, 2-chome,
Uchikawa, Yokosuka-shi
Kanagawa-Ken, Japan

Mr. James G. Partlow
Chief, Material Control
Lic. Br.
US Nuclear Regulatory Comm.
Washington, DC 20555

Mr. Dimitri Perricos
International Atomic
Energy Agency
Box 100
A-1400, Vienna, Austria

Mr. Bernardino Pontes
International Atomic
Energy Agency
P. O. Box 100
A-1400 Vienna, AUSTRIA

Mr. James A. Powers
Teknekron, Inc.
1483 Chain Bridge Road
McLean, VA 22101

Mr. Cordell Reed
Vice President for
Nuclear Activities
Commonwealth Edison Co.
Chicago, IL 60604

Dr. T. Douglas Reilly
Los Alamos
Scientific Laboratory
Group Q-5, MS 537
P. O. Box 1663
Los Alamos, NM 87545

Dr. W. Roehnsch, Vice President
National Board for
Nuclear Safety
DDR - 1157 Berlin-Karlshorst
Waldowallee 117

Mr. David B. Sinden
Atomic Energy Control Board
P. O. Box 1046
Ottawa, Canada K1P5S9

Dr. James P. Shipley
Los Alamos
Scientific Laboratory
Group Q-4, MS 541
P. O. Box 1663
Los Alamos, NM 87545

Dr. Hastings A. Smith
Los Alamos
Scientific Laboratory
Group Q-1, MS 540
P. O. Box 1663
Los Alamos, NM 87545

Dr. Gerald F. Tape, President
Associated Universities, Inc.
1717 Massachusetts Ave., NW
Washington, DC 20036

Dr. Fred H. Tingey, Director
Idaho Falls Center for
Higher Education
University of Idaho
Idaho Falls, ID 83401

Dr. Les Thorne, Head
Far East Section
Division of Operations A
Department of Safeguards
International Atomic
Energy Agency
Box 100
A-1400, Vienna, Austria

Dr. Adolph von Baeckmann
Director
Division of Development
and Technical Support
Department of Safeguards
International Atomic
Energy Agency
Box 100
A-1400, Vienna, Austria

Mr. George Weisz, DP-30
US Department of Energy
Mail Station A2-1016, GTN
Washington, DC 20545

Mr. Lawrence Wirfs
US Nuclear Regulatory Comm.
Division of Safeguards
Washington, DC 20555

Dr. H. Thomas Yolken
National Bureau of Standards
Bldg. 221, Rm. B318
Washington, DC 20760

INTERNATIONAL TRAINING COURSE ON NUCLEAR MATERIALS ACCOUNTABILITY FOR SAFEGUARDS PURPOSES

COURSE ATTENDEES

Mr. Shafi Ahmad
Karachi Nuclear Power Plant
P.O. Box 3183
Karachi 18
PAKISTAN

Miss Aysun Anacan
Division of Nuclear Installation
and Safety
Turkish Atomic Energy Commission
Alacam Sokak No. 11
Cankaya, Ankara, TURKEY

Mr. Gilberto Gomes de Andrade
Furnas Centrais Electricas S/A
Rua Real Grandeza 219
Reio de Janeiro - RJ - BRAZIL

Dr. Flavio Argentisi, Director
Biological Science and Statistics
Sr. Officer, Computer Center, ISPRA
JRC Euratom ISPRA, Varese,
ITALY 21010

Ms. Maria Luz Ascano
Philippine Atomic Energy Commission
Don Mariano Marcos Avenue
Diliman, Quezon City
PHILIPPINES

Mr. David Bacon
Commission of the
European Communities
12 Av Victor Hugo
Kirchberg, LUXEMBOURG

Dr. Tamas Biro
Institute of Isotopes
P. O. Box 77
H-1525 Budapest, HUNGARY

Dr. Jan J. W. Bogaards
Atomic Energy Control Board
Safeguards Division
P. O. Box 1046
Ottawa, CANADA K1P5S9

Mr. Paulo Ferruz Cruz
Comision Chilena
De Energia Nuclear
Los Jesuitas #645
Santiago, CHILE

Dr. Hafiz Ramadan Higgy
Atomic Energy Establishment
6 El Sheikh Ahmed
El Sakandary Heliopolis
Cairo, EGYPT

Mr. Jong-Lai Kim
Division of Safeguards
Atomic Energy Bureau
Ministry of Science
and Technology
Sejong-Ro, Jong-Ro Gu
Seoul 110
Republic of KOREA

Mr. Nand Kishore
Reactor Physicist
Rajasthan Atomic Power Station
Dept. of Atomic Energy
H3A/6 RAPP Township
Vikaram Nagar
Via Kota
Rajasthan, INDIA

Mr. Ji-Box Lee
Korean Atomic Energy
Research Institute
Cheong Ryang
P. O. Box 7
Seoul 110
Republic of KOREA

Mr. Martin Littlejohn
Commission of the
European Communities
12 Av Victor Hugo
Kirchberg, LUXEMBOURG

Mr. Chung-Lu Lo
Assist. Scientist for Safeguards
Institute of Nuclear
Energy Research
P.O. Box 3
Lung-tan, Taoyuan
TAIWAN 325

Mr. Shimon Malkiely
Israel Atomic Energy Commission
Nuclear Research Centre
P. O. Box 9001
Beer-Sheva, ISRAEL

Mr. Eduardo de Braga Melo
Comissao Nacional
de Energia Nuclear
Rua General Severiano 90
Botafogo
Rio de Janeiro - RJ -
BRAZIL 20000

Mr. Satoshi Nishizawa
A-10-5 Nagahori-Jutaku
Tokai-Mura
Naka-Gun, Ibaraki-Ken, JAPAN

Mr. Arumugan Palamalai
Reactor Research Center
Thungapuram
(SO) Trichy Dist.
Tamil Nadu, INDIA

Mr. Hudihastowo Peosposoetardjo
National Atomic Energy Agency
Jl Babarsari
P. O. Box 8
Yogyakarta, INDONESIA

Mr. Syed Yusuf Raza
Pakistan Atomic
Energy Commission
29-School Road
E-F/1
Istamabal, PAKISTAN

Mrs. Marita Stute
Gesellschaft Fur
Reaktorsicherheit (GRS)
Glockengasse 2 D-5000
Koeln 1, F.R. GERMANY

Mr. Adel A. M. Tolba
Atomic Energy Organization
of Egypt
5-A Foud Basha St.
Mansit El Bakery
Cairo, EGYPT

Mr. Nimrod M. Tole
Department of
Diagnostic Radiology
University of Nairobi
P. O. Box 30588
Nairobi, KENYA

Mrs. Isabel Torres
Depatment of Nuclear Energy
Avenida de Republica 45-5
1000 Lisboa, PORTUGAL

Mr. Yi-Ching Yang
Atomic Energy Council, Taiwan
Republic of China
67 Lane 144,
Keelung Road, Section 4
Taipai, TAIWAN 107

International Training Course on Nuclear Materials Accountability for Safeguards Purposes

May 27—June 6, 1980

Lecturers and Staff


Carleton D. Bingham
US Department of Energy


Carlos Buechler
International Atomic Energy Agency


Thomas R. Canada
Los Alamos Scientific Laboratory


Dimitri Perrices
IAEA


Donald D. Cobb
Los Alamos Scientific Laboratory


John E. Foley
Los Alamos Scientific Laboratory


Gesellschaft Fur Kernforschung MBH


Arnie Hakkila
Los Alamos Scientific Laboratory


Charlie Hatcher
Los Alamos Scientific Laboratory


William A. Higinbotham
Brookhaven National Laboratory


Gordon Hough
International Atomic Energy Agency


Karen Humphrey
Los Alamos Scientific Laboratory


Exxon Nuclear Company


Brian Jenkins
RAND Corporation


E. R. Johnson and Associates


G. Robert Keepin
Los Alamos Scientific Laboratory


Donald M. Kerr, Director
Los Alamos Scientific Laboratory


Ronald A. Knief
University of New Mexico


Hiroyoshi Kurihara
Embassy of Japan


David M. Lee
Los Alamos Scientific Laboratory


Ralph F. Lumb, President
NUSAC, Inc.


John J. Malanify
Los Alamos Scientific Laboratory


Ugo Miranda, Chief
EURATOM


Nicholas Nicholson
Los Alamos Scientific Laboratory


Christopher E. Olson
Sandia Laboratories


Takesha Osabe
Japan Nuclear Fuel Company


James C. Partlow
US Nuclear Regulatory Commission


Bernardino Pontes
International Atomic Energy Agency


James A. Powers
Teknekron, Inc.


Corbell Reed
Commonwealth Edison Co.


D. Douglas Reilly
Los Alamos Scientific Laboratory


Walter Rochbach
Nat'l. Board for Atomic Safety


James P. Shipley
Los Alamos Scientific Laboratory


David B. Smith
Atomic Energy Control Board


Hastings A. Smith
Los Alamos Scientific Laboratory


Gerald F. Tape, President
Associated Universities, Inc.


John Dusley
US Department of Energy


Fred H. Tingey
University of Idaho


Adolph von Baeckmann, Director
International Atomic Energy Agency


George Wise
US Department of Energy

US Nuclear Regulatory Commission

National Bureau of Standards

International Training Course on Nuclear Materials Accountability for Safeguards Purposes

May 27—June 6, 1980

Course Attendees


Shafi Ahmad
Pakistan


Aysun Anacan
Turkey


Gilberto Gomes De Andrade
Brazil


Flavio Argentin
Italy


Arumugan Palamalai
India


Maria Lua Asuano
Philippines


David Bacon
Commission of the European Communities
EURATOM


Tamás Bíró
Hungary


Jan Jacobus Bogaards
Canada


Chung-Lu Lo
Taiwan


Paulo Ferrus Cruz
Chile


Yi-Ching Yang
Taiwan


Jong-Lai Kim
Korea


Nand Kishore
India


Ji-Bok Lee
Korea


Martin Littlejohn
Commission of the European Communities
EURATOM


Shimon Malkiely
Israel


Hudihastowo Poespowetardjo
Indonesia


Eduardo De Braga Melo
Brazil


Satoshi Nishizawa
Japan


Hafiz Ramadan Higgy
Egypt


Syed Yusuf
Pakistan


Marita
F.R. Germany


Isabel Torres
Portugal


Abel Ahmed Telba
Egypt


Nuured Tole
Kenya